UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

June 5, 2000

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

SUBJECT: NRC INSPECTION REPORT NO. 50-298/2000-06

Dear Mr. Swailes:

This refers to the inspection conducted on April 2 through May 13, 2000, at the Cooper Nuclear Station facility. The enclosed report presents the results of this inspection. The results of this inspection were discussed during meetings on April 20 and May 10, 2000, with Mr. J. McDonald and Mr. B. Rash, respectively, and other members of your staff.

The inspectors examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspectors examined a selection of procedures and representative records, observed activities, and conducted interviews with personnel.

Based on the results of this inspection, the NRC has determined that four violations of NRC requirements occurred. These violations are being treated as noncited violations (NCV's), consistent with the NRC Enforcement Policy. These NCV's are described in the subject inspection report. If you contest any of the violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper facility.

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Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/ by Ken E. Brockman for

Charles S. Marschall, Chief
Project Branch C
Division of Reactor Projects

Docket No.: 50-298
License No.: DPR-46

Enclosure:
NRC Inspection Report No.
50-298/00-06

cc w/enclosure:
G. R. Horn, Senior Vice President
of Energy Supply
Nebraska Public Power District
1414 15th Street
Columbus, Nebraska 68601

John R. McPhail, General Counsel
Nebraska Public Power District
P.O. Box 499
Columbus, Nebraska 68602-0499

S. R. Mahler, Assistant Nuclear
Licensing and Safety Manager
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

Dr. William D. Leech
Manager - Nuclear
MidAmerican Energy
907 Walnut Street
P.O. Box 657
Des Moines, Iowa 50303-0657

Ron Stoddard
Lincoln Electric System
1040 O Street
P.O. Box 80869
Lincoln, Nebraska 68501-0869

Michael J. Linder, Director
Nebraska Department of Environmental
Quality
P.O. Box 98922
Lincoln, Nebraska 68509-8922

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, Nebraska 68305

Cheryl K. Rogers, Program Manager
Nebraska Health and Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P.O. Box 95007
Lincoln, Nebraska 68509-5007

Ronald A. Kucera, Director
of Intergovernmental Cooperation
Department of Natural Resources
P.O. Box 176
Jefferson City, Missouri 65102

Jerry Uhlmann, Director
State Emergency Management Agency
P.O. Box 116
Jefferson City, Missouri 65101

Vick L. Cooper, Chief
Radiation Control Program, RCP
Kansas Department of Health
and Environment
Bureau of Air and Radiation
Forbes Field Building 283
Topeka, Kansas 66620

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DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(JAC)**
Branch Chief, DRP/C **(CSM)**
Senior Project Engineer, DRP/C **(DPL)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**
Jim Isom, Pilot Plant Program **(JAI)**
Sampath Malur, Pilot Plant Program **(SKM)**

Only inspection reports to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
CNS Site Secretary **(SLN)**
Wayne Scott **(WES)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR 46
Report No.:	50-298/00-06
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	April 2 through May 13, 2000
Inspectors:	J. Clark, Senior Resident Inspector M. Hay, Resident Inspector J. Melfi, Project Engineer M. Shannon, Senior Health Physicist
Approved By:	Charles S. Marschall, Chief, Project Branch C Division of Reactor Projects
ATTACHMENTS:	1. Supplemental Information 2. NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report 50-298/00-06 (DRP)

This report covers a 6-week period of baseline resident inspection and an announced inspection by a regional radiation specialist.

The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609. The body of the report is organized under the broad categories of Reactor Safety and Other Activities as reflected in the summary below.

Cornerstone: Initiating Events

- Green. On two occasions, maintenance personnel failed to follow maintenance procedures when working on a control rod drive flow control valve. Maintenance workers failed to perform a specified step of a work order. As a result, the control rod subsequently operated at approximately 3 times normal rod speed. Planners also deleted a postmaintenance test that would have verified the rod's speed. The planners did not follow maintenance procedures that required work order revision approval for such changes. Both examples were in violation of Technical Specification 5.4.1(a) that requires written procedures to be established, implemented, and maintained. The licensee documented these issues in their corrective action process as Resolved Condition Report 2000-0046 and Resolved Condition Report 2000-0061, respectively.

  This noncited violation was characterized as a "green" finding using the significance determination process. The increased control rod speed had very low significance because reactor engineers demonstrated that excess margins were available to thermal limits during all times that the control rod was able to be moved (Section 1R19).

- Green. Licensed operators armed and withdrew Control Rod 42-19, after determining that the rod was inoperable, in violation of Technical Specification 3.1.3. The rod had exhibited excessive rod speed during a reactor startup. Technical Specification 3.1.3 requires that an inoperable control rod be fully inserted and disarmed. Operators inappropriately applied the permissive of Technical Specification 3.0.5 to manipulate the control rod for troubleshooting and rod speed adjustment. Technical Specification 3.0.5 permits testing of equipment solely to determine operability following corrective maintenance. The licensee documented these issues in their corrective action process as Resolved Condition Report 2000-0046 and Resolved Condition Report 2000-0061, respectively.

  This noncited violation was characterized as a "green" finding using the significance determination process. The increased control rod speed had very low safety significance because reactor engineers demonstrated that excess margins were available to thermal limits during all times that the control rod was able to be moved (Section 1R19).

Cornerstone: Mitigating Systems

- Green. The inspectors determined that a maintenance procedure was inadequate to address the seismic qualification of service water system piping when an idle section of piping was removed. Procedure 7.2.57.1, “”Pipe Support Removal and Re-installation,” provided guidance for the removal of snubbers, hangers, and other such equipment. However, the procedure did not address the impact from removal of the piping itself. As a result, operations personnel determined that residual heat removal service water booster pump system components were operable when seismic reviews to support operability had not been completed. This was in violation of Technical Specification 5.4.1(a) that requires written procedures to be established, implemented, and maintained. The licensee documented these issues in their corrective action process as Resolved Condition Report (RCR) 2000-0108.

  This noncited violation was characterized as a “green” finding using the significance determination process. This issue was determined to be of very low significance because, while the repairs affected the operability of one system loop, redundant safety capability was still available from the other loop. Also, operators and engineers determined that previous repairs were all conducted within the most restrictive Technical Specification allowed outage times (Section 1R15).

Cornerstone: Occupational Radiation Safety

- Green. A radiation worker and a radiation protection technician failed to follow the requirements of their radiation work permits. Specifically, the radiation worker failed to rinse equipment being removed from the reactor cavity pool to reduce the possible spread of contamination and radioactive particles, and the radiation protection technician failed to perform air sampling during the installation of the reactor cavity/fuel pool shield plugs to monitor the radiological airborne work conditions. The licensee documented the above occurrences in Problem Identification Reports 4-07254 and 4-08306, respectively.

  This noncited violation was characterized as a “green” finding using the occupational radiation safety significance determination process. This issue was determined to be of very low significance because these incidents did not result in an overexposure or have a significant potential to cause an overexposure (Section 2OS2).

Report Details

The plant was in a refueling outage at the beginning of the inspection period. On April 18, 2000, the licensee redesignated the ongoing outage as a forced outage for the resolution of an issue involving the environmental qualifications of electrical components. The plant remained in the forced outage for the remainder of the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignments

a. Inspection Scope

The inspectors performed a partial walkdown of Loop A of the residual heat removal system while this loop was providing shutdown cooling. The inspectors also reviewed logs and maintenance schedules to determine if outage activities affected the residual heat removal system alignment. In particular, the inspectors looked for items that could potentially cause a reactor vessel draining event through the residual heat removal system.

b. Observations and Findings

The inspectors did not identify any significant findings.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed routine fire protection tours to assess the material condition of plant fire protection equipment and proper control of transient combustibles. The specific risk-significant areas inspected included the emergency diesel generator Rooms and the residual heat removal heat exchanger rooms.

b. Observations and Findings

The inspectors did not identify any significant findings.

1R06 Flood Protection

.1 Seasonal

a. Inspection Scope

The inspectors reviewed external flood protection barriers for those susceptible areas containing risk significant structures, systems, and components, as analyzed by plant engineering personnel. Specifically, the inspectors reviewed the flood barriers for the diesel generator building, reactor building, and control building. The inspectors performed a review of the licensee's Emergency Procedure 5.1.3, "Flood," Revision 28. This procedure describes the operator actions required to mitigate the consequences of a flood and the materials needed to combat an external flooding event. The inspectors toured the applicable areas, to verify the appropriate inventory of materials.

b. Observations and Findings

The inspectors did not identify any significant findings.

.2 Periodic

a. Inspection Scope

The inspectors toured the diesel generator building and examined sumps, floor drains, and paths of external flooding for any potential impact to safety-related equipment. The inspectors selected the diesel generator building because it contained equipment important to plant safety and was specifically covered in the plant flooding analysis.

b. Observations and Findings

The inspectors did not identify any significant findings.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee's maintenance rule implementation for the following systems:

- High Pressure Coolant Injection,
- Essential 4160 volt, and
- Standby Gas Treatment.

The inspectors also reviewed selected problem identification reports associated with these systems for the previous 12 months, to determine if licensee staff properly captured potential maintenance rule issues.

b. Observations and Findings

The inspectors did not identify any significant findings.

1R13 Risk Assessments

a. Inspection Scope

The inspectors reviewed the risk assessment performed for various aspects of equipment alignments conducted during outage conditions. The inspectors verified that planning, maintenance, and operations personnel considered the affects to redundant equipment, potential losses of safety function, and overall plant safety for each activity being performed. Specifically, the inspectors reviewed:

- actions to commence and secure shutdown cooling,
- tagging operations to secure motor control centers for forced outage work, and
- electrical splicing activities for the forced outage.

b. Observations and Findings

The inspectors did not identify any significant findings.

1R14 Nonroutine Plant Evolutions

a. Inspection Scope

The inspectors reviewed licensee event reports (LERs) for potential human errors, to assess operator actions, and to evaluate the risk significance of events described. The inspectors evaluated operator actions to determine if actions were in accordance with procedures and training.

b. Observations and Findings

The inspectors did not identify any significant findings. The inspectors noted that the events described in each LER had been previously reviewed and documented in NRC Inspection Report 50-298/00-04. In each instance, the inspectors had identified and documented a noncited violation and made the finding that there was very low risk significance. As a result, the following reports are administratively closed because they are within the licensee's control and do not warrant further NRC attention:

(Closed) LER 50-298/00005-00: Scaffold Construction Places Plant In Condition
(Closed) LER 50-298/00006-00: Torus to Drywell Vacuum Breaker Misalignment
(Closed) LER 50-298/00007-00: Failed MOV Places Plant In Condition Prohibited

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the adequacy of Operability Evaluation PIR 4-00683 pertaining to increased drywell temperature profiles calculated for the small break loss of coolant accident. The drywell and containment spray valves were evaluated and are risk significant components. Therefore, the inspectors reviewed this evaluation to ensure that the operability of the drywell was properly assessed.

The inspectors also reviewed the operability assessments and Technical Specification actions taken by operations personnel for a series of maintenance activities on the service water booster pump (SWBP) system during January and February 2000. The

SWBP system is a risk significant system used in the mitigation of design basis accidents. The inspectors reviewed the maintenance activities and operator actions to ensure proper plant configuration control was maintained.

b. Observations and Findings

The inspectors did not identify any significant findings with the drywell evaluation.

During the period of January 4 through February 16, 2000, maintenance personnel performed a series of activities to replace discharge piping sections for each of the SWBPs. On February 14, 2000, the inspectors noted that, while SWBP A discharge piping was being replaced, operations personnel had declared all of Loop A inoperable. During previous piping replacements for Pumps B and C, operators had only declared the associated subloop inoperable. The SWBP system is divided into two loops, with two subloops each. Each subloop consists of an SWBP and its associated piping. A subloop is then connected to the other subloop, by common piping, to form one of the two SWBP operating loops. The operators stated that, during the replacement of SWBP A discharge piping, a piping support was also removed that potentially affected the seismic characteristics of the other subloop.

As a result, the inspectors questioned whether the loops were seismically analyzed for removal of sections of piping. Licensee engineers informed the inspectors that there was no current analysis to support this. The licensee entered the issue into their corrective action program and initiated analyses to determine the adequacy of maintenance procedures and technical specification conditions for SWBP work. The engineers completed their study on May 4, 2000. The analysis indicated that, for SWBPs A through C, there were no seismic concerns and, therefore, only the associated subloop would be inoperable throughout the repairs. However, the engineers determined that, when the SWBP D discharge piping was removed, Loop B was insufficiently supported. Therefore, the entire loop would be inoperable.

The inspectors reviewed Maintenance Procedure 7.2.57.1, “Pipe Support Removal and Re-Installation.” While specific guidance was given for the removal of seismic supports, no guidance was given to evaluate the seismic impact prior to the removal of a section of piping. The engineers stated that they would submit a change to the procedure to address piping replacements.

Technical Specification 5.4.1(a) requires, in part, that written procedures be established, implemented, and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures for performing maintenance are referenced in Appendix A.

The failure of Maintenance Procedure 7.2.57.1 to adequately address the seismic concerns when removing piping sections is a violation of Technical Specification 5.4.1(a). This violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy (50-298/0006-01). The licensee documented these issues in their corrective action process as Resolved Condition Report (RCR) 2000-0108.

The inspectors characterized this issue as having very low safety significance through the use of the significance determination process. With one loop of the SWBP system unavailable, the issue is "green" under all applicable accident scenarios. Also, the licensee performed a thorough review of previous piping repairs and determined that although the appropriate Technical Specification action statement was not always entered, all such activities were conducted within the applicable Technical Specification allowed outage times.

1R19 Postmaintenance Testing

a. Inspection Scope

The inspectors observed or evaluated postmaintenance testing performed on the following equipment to determine whether the tests adequately confirmed equipment operability:

- Tests performed on Control Rod Drive 42-19 following replacement of its exhaust water withdrawal/settle solenoid valve

- Test runs of the Division 1 emergency diesel generator after refueling outage maintenance

- Postmaintenance leakage testing of torus-to-drywell vacuum breakers

b. Observations and Findings

During a reactor startup on January 15, 2000, at approximately 30 percent rated thermal power, the control room operators noted excessive rod speed from Control Rod 42-19 while it was being withdrawn from the reactor core. Reactor engineers determined that the rod speed was approximately 9 inches per second as opposed to the nominal rod speed of 3 inches per second. At approximately 60 percent rated thermal power the licensee performed Nuclear Performance Procedure 10.4.1, “Control Rod Drive Speed Timing,” Revision 0, adjusted the speed of the rod, and declared Control Rod 42-19 operable. The inspectors reviewed this occurrence and identified the following three findings:

1. On January 14, 2000, maintenance personnel replaced Control Rod Drive 42-19 Solenoid Operated Valve SO120. This activity was controlled by Maintenance Work Request 00-0144 and required that the flow control setting for the replacement valve be adjusted to the as-found setting of the valve being replaced. During the licensee’s review, maintenance supervision identified that maintenance personnel failed to follow a step in the maintenance work request to set the flow control valve as required.

2. Following a review of this event, maintenance supervision determined that the control rod speed postmaintenance testing was inappropriately removed from the maintenance work request package without performing an approved revision.

Maintenance Work Request 00-0144 had originally required the performance of Procedure 10.4.1 as postmaintenance testing. However, during the planning process, work planning personnel removed this testing requirement because they failed to understand that the control rod drive speed adjusting valve was affected by the maintenance. Processing the package for revision could have resulted in identification of the error.

Technical Specification 5.4.1(a) requires, in part, that written procedures be established, implemented, and maintained covering applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedures for performing maintenance are recommended in Appendix A.

Administrative Procedure 0.40, "Work Control Program," Revision 18, Section 10.1, states, in part, "Changes that affect post maintenance testing require a maintenance work request revision." Also, Administrative Procedure 0.40, Section 5.6.2.3, states, in part, "Work shall be performed per work instructions." The failure to perform a maintenance work request revision, and the failure to follow the instructions in Maintenance Work Request 00-0144, are two examples of a violation of Technical Specification 5.4.1(a). This violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy (50-298/0006-02). The licensee documented these issues in their corrective action process as RCR 2000-0046 and RCR 2000-0061, respectively.

3. Technical Specification 3.1.3 requires that an inoperable control rod, for reasons other than sticking, be fully inserted and disarmed. While this was initially performed, during the aforementioned reactor startup, operations personnel armed the control rod and withdrew it for the purposes of adjusting the rod speed in accordance with Procedure 10.4.1. Operators administratively entered Technical Specification Limiting Condition for Operation 3.0.5 to permit arming and withdrawing the rod. Technical Specification LCO 3.0.5 states that equipment removed from service or declared inoperable may be returned to service under administrative control solely to perform testing required to demonstrate operability. The evaluation of an identified excessive rod speed, and the subsequent adjustments of Control Rod 42-19 speed, were not testing required to demonstrate operability.

   Therefore, the arming and withdrawal of Control Rod 42-19, outside the scope of Technical Specification LCO 3.0.5, was in violation of Technical Specification 3.1.3. This violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy (50-298/0006-03). The licensee documented these issues in their corrective action process through RCR 2000-0046 and RCR 2000-0061, respectively.

Reactor operations with Control Rod 42-19 having excessive rod speed was characterized as having very low significance. Control rod speed affects the amount of control rod travel when demanded. The bounding evaluation for rod speed is only applicable during operations requiring the operability of the rod block monitor. At all times when Control Rod 42-19 was operated, prior to adjusting the rod speed, the rod block monitor was not required to be operable because of the existing core thermal power and excess margin for the minimum critical power ratio. Therefore, no core operating thermal limits could have been exceeded.

The inspectors did not identify any significant findings with the diesel generator or vacuum breaker testing.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed the following tests for effective control of risk:

- Surveillance Procedure 6.1DG.302, "Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 1)," Revision 11

- Surveillance Procedure 6.2DG.302, "Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (DIV 2)," Revision 8

b. Observations and Findings

The inspectors did not identify any significant findings.

2. RADIATION SAFETY

OS2 ALARA (as low as reasonably achievable) Planning and Controls

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs throughout the radiologically controlled area during Refueling Outage 19. Independent radiation surveys of selected work areas within the radiologically controlled area were performed. The following items were reviewed:

- ALARA program procedures

- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information

- Scheduled refueling outage maintenance work and the associated exposure estimates and results

- ALARA job packages for reactor disassembly/reassembly, in-service inspection activities, under reactor vessel activities, local leak rate testing, and motor-operated valve maintenance and repair

- Hot spot tracking and reduction program
- Plant related source term data
- Temporary shielding program
- Declared pregnant worker dose monitoring controls
- Radiation protection department self-assessment
- Twelve operational radiation protection problem identification reports

b. Issues and Findings

ALARA Results

The following table shows the licensee's annual collective dose (in person-rem), 3-year average, and the national 3-year average results for boiling water reactors (BWR). This data is documented here to support future usage of the occupational radiation safety significance determination process.

	1997	1998	1999
Licensee's Yearly Totals	174	182	48
Licensee's 3-Year Average	150	135	135
BWR National 3-Year Average (Calculated from NUREG 0713)	277	234	***

**** Data not available**

Problem Identification Reports

Through a review of selected, risk significant problem identification reports, the inspector determined that the licensee typically identified appropriate and timely corrective actions to prevent recurrence of an issue. However, during the review, the inspector noted the following two examples of a violation of requirements directly related to the inspection area.

- On March 13, 2000, the licensee identified that a radiation worker failed to rinse in-service inspection equipment being removed from the reactor cavity pool as required by Radiation Work Permit 2000-1073, Revision 0. Rinsing items being removed from the reactor cavity pool was required to reduce the possible spread of contamination and radioactive particles. The licensee documented this issue in Problem Identification Report 4-07254.

- On April 10, 2000, the licensee identified that a radiation protection technician failed to provide air sample monitoring, as required by Radiation Work

Permit 2000-1048, Revision 0, while workers were installing the reactor cavity/fuel pool shield plugs. Air sampling was required to monitor radiological airborne work conditions. The licensee documented this issue in Problem Identification Report 4-08306.

Technical Specification 5.4.1.a. requires, in part, that written procedures be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section 7.e(1), includes procedures for the radiation work permit system. Section 7.3 of Procedure 9-ALARA-4, “Radiation Work Permits,” Revision 1, stated that it was each individual's responsibility to comply with radiation work permit requirements.

The failure to follow the requirements of Section 7.3 of Procedure 9-ALARA-4 on two occasions was identified as two examples of a Technical Specification 5.4.1.a violation. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the violation had a low safety significance. This violation is being treated as a noncited violation, consistent with the Enforcement Policy (NCV 298/0006-04). The licensee documented the above occurrences in Problem Identification Reports 4-07254 and 4-08306, respectively.

OTHER ACTIVITIES

4OA1 PI Verification

Inspection Scope

The inspectors reviewed logs and plant reports to verify the accuracy of reported data for the reactor coolant system specific activity and the reactor coolant system leak rate performance indicators. The inspectors also performed routine checks that locked high radiation areas were properly secured, while on tours throughout the plant.

b. Observations and Findings

The inspectors did not identify any significant findings.

4OA2 Meetings

.1 Exit Meeting Summary

On April 20, 2000, the inspectors conducted a meeting with Mr. J. McDonald, Plant Manager, and other members of plant management and presented the inspection results of the radiation protection inspection. The plant management acknowledged the findings presented. Plant management also informed the inspectors that no proprietary material was examined during the inspection.

On May 10, 2000, the inspectors conducted a meeting with Mr. B. Rash, Senior Engineering Manager, and other members of plant management and presented the inspection results of the resident inspection. The plant management acknowledged the findings presented. Plant management also informed the inspectors that no proprietary material was examined during the inspection.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Blan, Licensing Engineer, Licensing
T. Chard, Manager, Radiation Protection
L. Corey, ALARA Technician, Radiation Protection
J. Dixon, ALARA Supervisor, Radiation Protection
M. Boyce, Risk and Regulatory Affairs Manager
L. Dugger, Engineering Support Manager
C. Fifler, Assistant Maintenance Manager
M. Kaul, Operations Support Specialist
M. Gillan, Outage Manager
B. Houston, Quality Assurance Operations Manager
D. Kimball, Assistant Manager, Radiation Protection
W. Macecevic, Operations Manager
S. Mahler, Assistant Licensing Manager
E. McCutchen, Senior Licensing Engineer
J. McDonald, Plant Manager
B. Rash, Senior Engineering Manager
R. Sessoms, Quality Assurance Senior Manager
C. Sunderman, Supervisor, Radiological Operations
K. Tanner, Crew Leader, Radiological Operations
C. Weers, Radiological Support Supervisor

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

50-298/0006-01	NCV	Failure to Enter Appropriate LCO for SWBP Condition
50-298/0006-02	NCV	Failure to Follow Maintenance Procedures
50-298/0006-03	NCV	Inappropriate Use of Technical Specification LCO
50-298/0006-04	NCV	Failure to Follow ALARA Procedures

Previous Items Closed

50-298/00005-00	LER	Scaffold Construction Places Plant In Condition
50-298/00006-00	LER	Torus to Drywell Vacuum Breaker Misalignment
50-298/00007-00	LER	Failed MOV Places Plant In Condition Prohibited

PARTIAL LIST OF DOCUMENTS REVIEWED

Problem Identification Reports

4-02015
4-03553
4-04327
4-04769
4-04771
4-05355
4-05574
4-05606
4-06406
4-07188
4-07923
4-07926
4-07569
4-07942
4-07955
4- 07044
4-07130
4-07254
4-07452
4-07489
4-07562
4-07773
4-07993
4-08003
4-08014
4-08241
4-08306

Radiation Protection Assessments

ALARA Functional Area department self-assessment, conducted March 13-17, 2000

A summary of operational radiation protection Problem Identification Reports written since March 1, 2000

Radiation Protection Procedures

9.ALARA.1	"Personnel Dosimetry and Occupational Radiation Exposure Program," Revision 5
9.ALARA.3	"In-Vitro and In-Vivo Bioassays," Revision 1
9.ALARA.4	"Radiation Work Permits," Revision 1
9.ALARA.5	"ALARA Work Review," Revision 4

9.ALARA.6	"ALARA Reports," Revision 1
9.ALARA.7	"ALARA Job Files," Revision 1
9.ALARA.8	"ALARA Document Control," Revision 1
9.RADOP.3	"Area Posting and Access Control," Revision 5

ATTACHMENT 2

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the significance determination process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.