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

September 11, 2000

EA-00-207

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

SUBJECT: NRC INSPECTION REPORT NO. 50-298/00-11

Dear Mr. Swailes:

This refers to the inspection conducted on June 25 through August 12, 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 June 29 and August 17, 2000, with Mr. J. McDonald 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 two violations of NRC requirements occurred. These violations are being treated as noncited violations (NCV), consistent with Section VI.A of the NRC Enforcement Policy. The NCVs are described in the subject inspection report. If you contest these 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.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

K. M. Kennedy, Chief
Project Branch C
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report No.
50-298/00-11

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Only inspection reports to the following:
David Diec **(DTD)**
NRR Event Tracking System **(IPAS)**
CNS Site Secretary **(SLN)**
Dale Thatcher **(DFT)**

R:_CNS\CN2000-11RP-JAC.wpd EA-00-207; IE 14

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR-46
Report No.:	50-298/00-11
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	June 25 through August 12, 2000
Inspectors:	J. Clark, Senior Resident Inspector M. Hay, Resident Inspector W. Sifre, Project Engineer L. Ellershaw, Senior Reactor Inspector P. Goldberg, Reactor Inspector W. McNeill, Reactor Inspector L. Ricketson, P.E., Senior Health Physicist D. Carter, Health Physicist
Approved By:	K. Kennedy, Chief, Project Branch C Division of Reactor Projects

ATTACHMENTS:
1. Supplemental Information
2. NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

IR05000298-00-11; on 06/25-08/12/00; Nebraska Public Power District, Cooper Nuclear Station. Integrated Resident & Regional Report; Maintenance Risk Assessments and Emergent Work Evaluation, Surveillance Testing.

This inspection report covers a 7-week period of inspection by resident and region-based inspectors.

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, Safeguards, and Other Activities as reflected in the summary below.

Mitigating Systems

- No color. On July 2, 2000, engineering and operations personnel revised a surveillance procedure to raise the drywell temperature limit from 148° F to 150° F. The licensee's basis for raising the limit was that the instrument inaccuracy was already accounted for in the calculated net positive suction head margin for emergency core cooling systems. However, the inspectors determined that adequate margin did not exist in these calculations. As a result, during the licensee's review of the procedure change, the licensee failed to identify that the change involved an unreviewed safety question and therefore required Commission approval. The failure to obtain Commission approval prior to raising the drywell temperature limit was a violation of 10 CFR 50.59. This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy and is in the licensee's corrective action program as Problem Identification Report 4-10381.

  This noncited violation was determined to have very low safety significance because actual drywell temperature exceeded the original limit of 148° F for a short period of time, approximately 10 hours (Section 1R22).

Barrier Integrity

- Green. On June 28, 2000, operations personnel declared the drywell floor drain sump flow monitoring system inoperable to perform a surveillance test. Operators did not recognize that the drywell atmospheric monitoring system had previously been declared inoperable on June 23 due to a failed sample pump. As a result, all reactor coolant system leak detection instrumentation required by Technical Specification 3.4.5, "RCS Leakage Detection Instrumentation," was inoperable for 1 hour and 9 minutes. Because the operators did not recognize this condition, the requirements of Technical Specification 3.0.3 to initiate actions within 1 hour to shut down the plant were not satisfied. This was determined to be a violation and is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy. It is in the licensee's corrective action program as Significant Condition Report 2000-0701.

  This noncited violation was characterized as a green finding using the significance determination process. It was determined to have very low safety significance because the condition existed for a short period of time (Section 1R13).

<u>Report Details</u>

At the beginning of the inspection period, the plant was at approximately 55 percent power for fuel leak localization. Following efforts to suppress neutron flux around the fuel leak, reactor power was restored to 100 percent on June 28, 2000. The plant operated at 100 percent power throughout the remainder of the inspection period, with the exception of minor power reductions for rod line adjustments and control rod surveillance.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 <u>Equipment Alignments</u>

a. <u>Inspection Scope</u>

The inspectors performed a partial walkdown of the high pressure coolant injection system and the service water system. The inspectors also reviewed logs and maintenance schedules to determine if maintenance activities affected the system alignments.

b. <u>Findings</u>

There were no findings identified during this inspection.

1R05 <u>Fire Protection</u>

a. <u>Inspection Scope</u>

The inspectors performed routine plant tours to assess the material condition of fire protection equipment and proper control of transient combustibles. The specific risk-significant areas inspected included the reactor building southwest quadrant and the Division 1 critical switchgear room.

b. <u>Findings</u>

There were no findings identified during this inspection.

1R12 <u>Maintenance Rule Implementation</u>

a. <u>Inspection Scope</u>

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

- Reactor equipment cooling system
- Reactor core isolation cooling system
- Service water booster pump system

The inspectors reviewed the selected systems based on their risk importance to the operation of the plant. The inspectors verified that engineering personnel were adequately tracking and trending failures and performance data for these systems. The inspectors also reviewed problem identification and resolution forms to verify proper documentation of maintenance rule-related functions.

b. Findings

There were no findings identified during this inspection.

1R13 Risk Assessments

a. Inspection Scope

The inspectors reviewed the risk assessments performed for various maintenance activities. The inspectors reviewed selected maintenance activities to determine if operations personnel considered the impact of these activities on redundant equipment, potential losses of safety function, and overall plant safety. The inspectors verified that work control and operations personnel were aware of risk categories and applicable contingency actions. Specifically, the inspectors reviewed:

- Surveillance of the drywell floor drain sump flow monitoring system with the drywell atmospheric monitoring system inoperable

- Contingencies in effect concerning high drywell temperatures that were approaching administrative limits

b. Findings

On June 23, 2000, operators declared the drywell atmospheric monitor inoperable due to a sample pump failure and entered Technical Specification 3.4.5.B. At 11:52 a.m. on June 28, 2000, control room operators authorized the performance of Surveillance 6.DWLD.302, "Drywell Floor Drain Sump 1F Flow Loop Channel Calibration," Revision 2C1, and entered Technical Specification 3.4.5.A. Surveillance 6.DWLD.302 instructs performance of channel functional tests to ensure the required reactor coolant system leakage detection flow instruments are operable. These two conditions resulted in all required reactor coolant leakage detection systems being inoperable in the drywell. Technical Specification 3.4.5.D requires immediate entry into Limiting Condition for Operation (LCO) 3.0.3 when all required leakage detection systems are inoperable. The control room operators failed to recognize this condition resulting in their failure to enter LCO 3.0.3.

At 1:01 p.m., on June 28, 2000, Surveillance 6.DWLD.302 was completed and operators exited Technical Specification 3.4.5.A. At 3:52 p.m. during review of control room logs by the control room shift technical engineer, it was discovered that performing the surveillance on drywell floor drain Sump 1F, while the drywell atmospheric monitor was inoperable, required entry into Technical Specification LCO 3.0.3. LCO 3.0.3 requires that action be initiated within one hour to place the unit in Mode 2 within 7 hours. The failure to initiate actions within one hour as required by Technical Specification LCO 3.0.3 is considered a violation of NRC requirements. This violation is being treated as a noncited violation (50-298/0011-01) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Significant Condition Report 2000-0701.

This noncited violation was characterized as having very low safety significance. The licensee was actually in LCO 3.0.3 for a duration of 1 hour and 9 minutes. Therefore, the requirement to place the unit in Mode 2 within 7 hours was not exceeded.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the adequacy of the operability assessment for repeated reactor equipment cooling pump trips. The inspectors noted that multiple reactor equipment cooling pumps out of service would present an increased risk to plant operation.

The inspectors also reviewed the operability assessment for increased noise in the Turbine Generator 2 governor valve caused by sticking. The inspectors determined that thermal limits could potentially be exceeded if this valve were stuck in the closed position.

b. Findings

There were no findings identified during this inspection.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed procedures governing plant modifications to evaluate the effectiveness of the licensee's programs for implementing modifications to risk-significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility. The inspectors also reviewed 24 change evaluation documents (permanent plant modification packages) to verify that they were performed in accordance with plant procedures. Procedures and permanent plant modifications reviewed are listed in the attachment.

The inspectors conducted field walkdowns of five permanent plant modifications, identified in the attachment. The cognizant design and system engineers for the identified modifications were interviewed to determine their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed problem identification reports and the subsequent corrective actions pertaining to licensee-identified problems and errors in the performance of permanent plant modifications. Problem identification reports reviewed are listed in the attachment.

b. Findings

There were no findings identified during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed the following tests:

- Surveillance Procedure 6.SLC.101, “Standby Liquid Control Operability Test,” Revision 8.
- Surveillance Procedure 6.LOG.601, “Daily Surveillance Logs - Modes 1, 2, and 3,” Revisions 20 and 21.

b. Findings

There were no findings identified with Surveillance Procedure 6.SLC.101.

On July 12, 2000, the inspectors reviewed the data of Surveillance Procedure 6.LOG.601 for the current shift. The inspectors noted that the limit for drywell temperature had been raised on July 2, 2000, from 148°F, to 150°F in Revision 21 of the procedure. Drywell temperature is used for analysis in several design features of the plant, including environmental qualifications and net positive suction head for emergency core cooling systems. The inspectors also determined that drywell temperature had been approximately 148.5 °F for a period of approximately 10 hours on July 10-11, 2000.

The inspectors questioned operators and engineering personnel about the procedure revision. The inspectors were informed that the revision resulted from a change in the method for incorporating instrument uncertainty for drywell temperature. Previously, the 148°F value was an explicit limit used to account for this uncertainty. The limit of 148°F provided assurance that actual drywell temperature would not exceed 150°F. The inspectors noted that, for Revision 21, the instrument uncertainty had been implicitly accounted for in the margin for net positive suction head calculations. The inspectors determined that both the explicit and implicit methods for addressing instrument uncertainty were industry-accepted practices.

On July 14, 2000, the inspectors reviewed a docketed letter dated November 24, 1998, from the U.S. Nuclear Regulatory Commission to Cooper Nuclear Station which addressed correspondence regarding Generic Letter 97-04, “Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps.” The letter detailed NRC concerns associated with the review of information pertaining to Generic Letter 97-04. Specifically, the NRC identified an unresolved safety question due to the apparent operation of the Cooper facility outside of its design basis. The inspectors found that the calculations and methods used to prove net positive suction head, based partially on containment temperature, were not what was described in the original license basis of the plant. The inspectors also noted that the licensee had committed to produce a justification for continued operation and to submit a license amendment request. The inspectors verified that both actions had been performed. However, the amendment request was subsequently retracted due to modeling errors.

The inspectors determined that the engineering assessment for the July 2 change to Surveillance Procedure 6.LOG.601 was based upon the new calculations and methods that had not been approved by the NRC. The inspectors concluded that there was no margin associated with the net positive suction head for the emergency core cooling systems. Therefore, removing the explicit temperature limit, which ensured the actual drywell temperature did not exceed Technical Specification limits, constituted an unreviewed safety question. The inspectors were concerned that actual temperature of the drywell could exceed 150°F, with no approved margin.

A maximum drywell temperature of 150°F is specified in LCO 3.6.1.5 of the Technical Specifications. The drywell temperature parameter is also used in various design basis assumptions and is referenced in several sections of the Updated Safety Analysis Report (USAR). Section V.2.3.9.7 of the USAR specifies that drywell temperature is used to monitor essential drywell parameters that are used in the “Station Safety Analysis” of Section XIV. Section XIII.6.4 of the USAR lists instrumentation procedures, including drywell and suppression chamber instrumentation lists.

Section 59 of 10 CFR Part 50 states that a licensee may make changes to a procedure described in the safety analysis report without prior Commission approval, unless the proposed change involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question. The change in drywell temperature limits in Surveillance Procedure 6.LOG.601 represents an unreviewed safety question due to the lack of an approved analysis that permits drywell temperature to exceed a design basis limit addressed in the Technical Specifications and the Updated Safety Analysis Report. As a result, the change decreased the margin of safety as defined in the basis of the Technical Specification and increased the probability of a malfunction of equipment important to safety. The failure by operations and engineering personnel to identify that raising the drywell temperature limit was an unreviewed safety question, requiring prior Commission approval, is a violation of 10 CFR 50.59. This violation is being treated as a noncited violation (50-298/0011-02) in accordance with Section VI.A of the NRC Enforcement Policy.

Due to the nature of unreviewed safety questions, and the requirements of 10 CFR 50.59, the significance determination process of Inspection Manual Chapter 0609 does not apply to this issue. This issue is considered to have very low safety significance, due to the limited time drywell indicated temperature was above 148°F. Operations personnel implemented a standing order to reimpose the 148°F limit on July 14, 2000, after discussing the issue with the inspectors. Operations personnel also initiated a problem identification report (PIR 4-10381) to enter the issue into their corrective action program.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector interviewed cognizant licensee personnel and reviewed the following items:

- Calibration, operability, and alarm setpoint, when applicable, of portable radiation detection instrumentation, area radiation monitors, continuous air monitors, containment high range monitor, main steam line monitor, whole-body counting equipment, electronic alarming dosimeters, and personnel contamination monitors

- Calibration expiration and source response check currency on radiation detection instruments staged for use

- The status and surveillance records of self-contained breathing apparatuses staged and ready for use in the plant

- The licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions

- Control room operator and emergency response personnel training and qualifications for use of self-contained breathing apparatuses

- Licensee self-assessments and audits, focusing on radiological incidents that involved personnel internal exposures

- Selected exposure significant radiological incidents that involved radiation monitoring instrument deficiencies or self-contained breathing apparatuses since the last inspection in this area

b. Findings

There were no findings identified.

OTHER ACTIVITIES

4OA1 PI Verification

Inspection Scope

The inspectors completed the following inspection elements:

- Reviewed logs and plant reports to verify the accuracy of reported data for heat removal system unavailability and scrams with loss of normal heat removal

- During plant tours, verified that locked high radiation areas were properly secured

b. Findings

There were no findings identified during this inspection.

4OA6 Meetings

.1 Exit Meeting Summary

The results of the inspections conducted during this period were discussed with licensee management on June 29 and August 17, 2000. 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

A. Bacha, Civil Engineer, Design Engineering
M. Boyce, Risk and Regulatory Affairs Manager
P. Caudill, Senior Manager, Technical Services
T. Chapin, Auditor, Quality Assurance
J. Dixon, ALARA Supervisor, Radiation Protection
J. Falls, Technician, Radiation Protection
C. Fidler, Assistant Maintenance Manager
M. Gillan, Outage Manager
M. Hale, Senior Manager, Site Support
S. Jobe, Shift Technical Engineer
K. Jones, Manager, Design Engineering
D. Madsen, Senior Licensing Engineer
W. Macecevic, Operations Manager/Acting Plant Manager
S. Mahler, Assistant Licensing Manager
E. McCutchen, Senior Licensing Engineer
J. McDonald, Plant Manager
W. Previn, Electrical and I & C Supervisor
B. Rash, Senior Engineering Manager
B. Stander, Mechanical Engineer, Design Engineering
D. Van Der Kamp, Training/Licensing
A. Weise, Design Engineer
J. White, Technician, Radiation Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

50-298/0011-01	NCV	Failure to initiate actions within one hour as required by Technical Specification Limiting Condition for Operation 3.0.3
50-298/0011-02	NCV	Failure to attain prior Commission approval for a procedure revision involving an unreviewed safety question

DOCUMENTS REVIEWED

Administrative Procedures

0.5 Problem Identification and Resolution, Revision 19 C2

CAP Deskquide 1 PIR Processing, Revision 12
CAP Deskguide 2 Resolution of “OTHER” PIRs, Revision 0
CAP Deskguide 5 Problem Classification Guideline, Revision 1

Radiation Protection Procedures

6.PRM.323 High Range Containment Monitor Victoreen Model 875 Source Calibration Check, Revision 5

9.INST.20 Calibration of the Canberra Whole-Body Counter, Revision 0

9.INST.41 SAIC Model PD-1 Electronic Dosimeter System, Revision 2

9.INST.50 Portable Beta-Gamma GM Survey Meter Eberline E-140, Revision 0

9.INST.53 Extender Model 2000 W GM Survey Instrument, Revision 1

9.RESP.4 Bauer FS-9 Air Compressor Air Quality Monitoring, Revision 1

Surveillance Procedures

6.PRM.324 Main Steam Line Process Radiation Monitor Channel Calibration, Source Test, and Setpoint Determination, Revision 8

6.PRM.326 Drywell Air Sampling System Known Source Calibration, Revision 2 C1

6.PRM.327 Drywell Air Sampling System Electronic Channel Calibration, Revision 10 C1

15.ARM.302 Area Radiation Monitors Calibration and Functional Test, Revision 2

15.ARM.322 Containment High Range Monitor Channel Calibration and Setpoint Determination, Revision 7

Cooper Nuclear Station Operations Manual

Alarm Procedure 2.3.2.21 Panel 9-3 - Annunciator 9-3-1

Training

Job Performance Measure for Radiological Protection (Lesson Number SKL018-01-04)

Audits and Self-Assessments

1999 CNS Radiation Monitoring Instrumentation Self Assessment
CNS Quality Assurance Surveillance Report #S302-9801, “Self Contained Breathing Apparatus”

CHANGE EVALUATION DOCUMENTS

*CED 1999-0144, "Replacement of Valves SW-MO-89A and 89B," Revision 0 and Change Notice 4

*CED 1999-0083, “HPCI - M058 Insulation,” Revision 0

CED 1999-0070, “HPCI ASD Flow Controller Enhancement,” Revision 0

*CED 1998-0180, “HV-SOV-(SPV-259) and HV-SOV-(SPV-261) Replacement,” Revision 0

*CED 1998-0110, “PC PENT-X104A Gauge Replacement,” Revision 0

CED 1998-0205, “Replacement of Diesel Generator Lube Oil Filter Gages, CNSNO23092,” Revision 0

CED 1998-0253, "MSIV Seat Design Change," Revision 0

CED 1999-0043, "HPCI Suction Vent Valve," Revision 0

CED 1998-0128, "Evaluation of Changes Required for CEP 95-0236,” Revision 1

CED 1998-0268, "Addition of Throttling Capability for RHR-MOV-MO12A and RHR-MOV-MO12B," Revision 0 and OSC 1

CED 1998-0284, "REC Quad FCU Flow Control Valve Replacement," Revision 0 and OCD 4

CED 1998-0282, "MSIV Riblet Guide Pads," Revision 0

*CED 1998-0179, "Upgrade of Air Supply to Control Valves HV-SOV-(SPV-259) and HIV-SOV-(SPV-261)," Revision 0

CED 1998-0013, “CS-MOV-MO12A/B Motor Replacement,” Revision 0 with Change Notice 1

CED 1998-0014, “RHR-MO18 Motor Replacement,” Revision 0 with Change Notice 1

CED 1998-0029, “REC-MOV-700MV Motor Replacement,” Revision 0 with Change Notice 1

CED 1998-0030, “Documentation of Configuration at HPCI Stop Valve HPCI-HO-HOV10,” Revision 0

CED 1998-0201, “250 VDC Fuse Replacement with Higher Interrupting Current Rating Fuses,” Revision 1

CED 1999-0121, “125 VDC Battery Cell Replacement,” Revision 0 with Change Notices 1 through 4 and On-the-Spot-Changes 1 through 6

CED 1999-0122, “250VDC Battery Cell Replacement,” Revision 0 with Change Notices 1 through 3 and On-the-Spot-Changes 1 through 8

CED 1999-0170, “RW-LS-Z4 Hi-Hi Contact Bypass,” Revision 0 with Change Notice 1

CED 1997-036A, “Installation of Auxiliary Relays and Fuses to Alleviate Distribution System Undervoltage Concerns,” with On-the-Spot-Changes 1-2

CED 1997-070, “SRV Stellite 21 Pilot Disc Replacement,” with On-the Spot-Change 1

CED 1997-090, “Motor Operator Upgrades for CS-MO-MO5A and MO5B,” Revision 0

NOTE: Those packages identified (*) received field walkdowns.

PROBLEM IDENTIFICATION REPORTS

2-15575	3-53230	4-03242	4-03528	4-04582	4-10160
3-20404	3-53461	4-03365	4-03543	4-04675	4-67777
3-40308	3-53512	4-03476	4-04053	4-05809	
3-40354	4-03202	4-03479	4-04217	4-06875	

Lists of reports involving radiation instrumentation and self-contained breathing apparatuses from January 1, 1999, to June 23, 2000

ENGINEERING PROCEDURES

7.0.8.1, "Inservice Leak Test," Revision 11

6.SC.201, "Secondary Containment (Reactor Building H&V) Valve Operability Test," Revision 12C1

6.SC.301, "Secondary Containment Isolation AOV Accumulator Functional and Check Valve IST Exercise Test," Revision 4

6.MS.201, "Main Steam Isolation Valve Operability Test," Revision 2

6.1REC.102, "REC Critical Subsystem Emergency Mode Flow Test," Revision 2

6.MISC.401, "Position Indicator Inservice Testing (IST),” Revision 6

6.1RHR.201, "RHR Power Operated Valve Operability Test (IST),” Revision 8c3

6.CSCS.601, "Technical Specifications Verification of Flowpath Valve Lineup," Revision 2

2.4.2.4.1, "RHR Loss of Shutdown Cooling," Revision 20

2.69, "Residual Heat Removal System," Revision 59

6.RHR.201, "RHR Motor Operated Valve Operability Test from ASD-RHR Panel," Revision 5

2.2.69.2, "RHR System Shutdown Operations," Revision 34

6.SC.603, "Technical Specification Verification of Secondary Containment Manual Valves and Blind Flanges," Revision 3

2.2.33, "High Pressure Coolant Injection System," Revision 48

7.2.24, "Main Steam Isolation Valve Maintenance," Revision 26C1

6.PC.513, "Main Steam Local Leak Rate Tests," Revision 5

6.MS.201, "Main Steam Isolation Valve Operability Test (IST)," Revision 5C1

13.21, "RHR Service Water Booster Pump Isolation Valves SW-MOV-89A & B Seat Leakage Test," Revision 0

3.4, “Configuration Change Control,” Revision 28

EDP-06, “Design Inputs,” Revision 04

DESIGN BASIS DOCUMENTS

DBD-5, “DC Electrical Distribution System,” Revision 3
DBD-12, “Core Spray (CS) System,” Revision 2
DBD-13, “Residual Heat Removal(RHR) System,” Revision 2
DBD-16, “Reactor Equipment Cooling (REC) System,” Revision 1

MISCELLANEOUS DOCUMENTS

Inservice Testing Program Basis Document, Revision 1
Maintenance Work Request 99-3139 dated 11/2/99

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

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

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. 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.