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

December 6, 2000

Randal K. Edington, Vice President - Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

SUBJECT: RIVER BEND STATION--NRC INSPECTION REPORT NO. 50-458/00-14

Dear Mr. Edington:

On November 11, 2000, the NRC completed an inspection at your River Bend Station facility. The enclosed report documents the inspection findings which were discussed with you and members of your staff on November 17, 2000.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of the inspection, the inspectors identified two findings of very low safety significance (Green) and one other finding (No Color). Two of these findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these findings as noncited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these noncited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the River Bend Station facility.

In addition, adverse trends in two cross-cutting areas were identified. A declining human performance trend was identified with failure of personnel to adhere to plant procedural requirements or to maintain a questioning attitude as common elements. Additionally, a declining problem identification and resolution trend was identified with not implementing timely corrective actions as the common element.

Two unresolved items were identified concerning the storage of chemicals used for alternate standby liquid control system injection and the standby service water system station blackout valve. These matters are discussed in Sections 1R04.1 and 1R12.1 of the enclosed inspection report. The items are unresolved pending a review of the safety significance of the issues by a NRC senior reactor analyst.

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/

William D. Johnson, Chief
Project Branch B
Division of Reactor Projects

Docket No.: 50-458
License No.: NPF-47

Enclosure:
NRC Inspection Report No.
50-458/00-14

cc w/enclosure:
Executive Vice President and
Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Vice President
Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

General Manager
Plant Operations
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

Director - Nuclear Safety
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

Wise, Carter, Child & Caraway
P.O. Box 651
Jackson, Mississippi 39205

Mark J. Wetterhahn, Esq.
Winston & Strawn
1401 L Street, N.W.
Washington, DC 20005-3502

Manager - Licensing
River Bend Station
Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana 70775

The Honorable Richard P. Ieyoub
Attorney General
Department of Justice
State of Louisiana
P.O. Box 94005
Baton Rouge, Louisiana 70804-9005

H. Anne Plettinger
3456 Villa Rose Drive
Baton Rouge, Louisiana 70806

President
West Feliciana Parish Police Jury
P.O. Box 1921
St. Francisville, Louisiana 70775

Ronald Wascom, Administrator
and State Liaison Officer
Department of Environmental Quality
P.O. Box 82135
Baton Rouge, Louisiana 70884-2135

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(TWP)**
Branch Chief, DRP/B **(WDJ)**
Senior Project Engineer, DRP/B **(RAK1)**
Branch Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
Scott Morris **(SAM1)**
NRR Event Tracking System **(IPAS)**
RBS Site Secretary **(PJS)**
Dale Thatcher **(DFT)**

R:_RB\RB2000-14RP-TWP.wpd

RI:DRP/B	SRI:DRP/B	C:DRP/B		
SMSchneider	TWPruett	WDJohnson		
E - WDJohnson	E - WDJohnson	/RA/		
12/1/00	12/1/00	12/6/00		

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/00-14
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	September 24 through November 11, 2000
Inspectors:	T. W. Pruett, Senior Resident Inspector S. M. Schneider, Resident Inspector
Approved By:	William D. Johnson, Chief, Project Branch B Division of Reactor Projects

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

SUMMARY OF FINDINGS

River Bend Station
NRC Inspection Report 50-458/00-14

IR 05000458-00-14; on 9/24-11/11/2000; Entergy Operations, Inc; River Bend Station. Resident Report. Equip. Align., Fire Prot., Maint. Rule Impl., Op. Evals., Drill Eval., and cross-cutting issues.

The inspection was conducted by resident inspectors. The inspection identified two Green findings, one of which was a noncited violation, three findings of No Color, one of which was a noncited violation, and two unresolved items. The significance of most of the findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, "Significance Determination Process." Findings for which the significance determination process does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- To be determined. The licensee had not maintained the required inventory of sodium borate and boric acid to support operation of the alternate standby liquid control system. Additionally, the licensee had not completed annual walkdowns of emergency operating procedure enclosures. Due to the variance in the preliminary safety significance determinations, which ranged from very low to substantial, the issues involving the inventory of sodium borate and boric acid and emergency operating procedure enclosure walkdowns is an unresolved item pending a review of the safety significance by a NRC senior reactor analyst. This issue is in the licensee's corrective action program as Condition Reports 2000-1680 and 2000-1723 (Section 1R04.1).

- Green. The licensee did not adequately assess or conduct fire drills. During the October 11, 2000, fire brigade drill, the licensee failed to identify and assess several deficiencies. For example, brigade members incorrectly donned protective clothing, the brigade leader did not establish communications between the control room and scene, there was no simulated demonstration of the ability to pressurize a hose or use a hose nozzle, two brigade members did not actively participate in the simulated extinguishing of the fire, and objective criteria were not developed to evaluate the fire brigade's performance. Additionally, the licensee performed unannounced drills within 4 weeks of each other and did not use members of the management staff responsible for plant safety and fire protection to critique unannounced drills. The failure to adequately assess the effectiveness of the fire brigade and to adequately conduct fire brigade drills was a violation of Attachment 4 to Facility Operating License 50-458. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This finding was entered in the licensee's corrective action program as Condition Report 2000-1848.

  The inspectors determined that the safety significance of the fire brigade training issues and fire brigade performance was very low in that plant fire barriers and automatic suppression capability were maintained in accordance with the fire protection program (Section 1R05.1).

- To be determined. The licensee did not recognize that the failure of station blackout Valve SWP-AOV599 was a maintenance preventable functional failure. Consequently, additional corrective actions were not implemented even though the failure resulted in the standby service water system exceeding the maintenance rule performance monitoring criteria. Due to the variance in the preliminary safety significance determinations, which ranged from very low to moderate, the issues involving the failure of station blackout Valve SWP-AOV599 to automatically open is an unresolved item pending a review of the safety significance by an NRC senior reactor analyst. This issue is in the licensee's corrective action program as Condition Reports 2000-1627 and 2000-1645 (Section 1R12.1).

- Green. Engineering personnel did not properly assess the significance of system air leakage on the ability to maintain station blackout Valve SWP-AOV599 open for the 12-hour duration specified in the probabilistic safety assessment. Specifically, engineering personnel only considered the minimum air pressure necessary to open the valve and did not determine the minimum air pressure needed to maintain the valve in the open position.

  The poor engineering review of air leakage on station blackout Valve SWP-AOV599 was of very low safety significance in that subsequent air drop testing of the system and engineering analysis demonstrated that the valve would have remained open for the 12-hour duration specified in the probabilistic safety analysis (Section 1R15).

Other Activities: Cross-cutting Issues

- No Color. Operations personnel inappropriately accessed nonjob related information on the operations shift superintendent's computer. The participation in potentially distracting activities at the operations shift superintendent's watch station was a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This finding was entered in the licensee's corrective action program as Condition Report 2000-1709.

  The inspectors determined that the safety significance of the potentially distracting activity at the operations shift superintendent's watch station was very low in that no actual plant problems occurred during the time in question which would have required the operations shift superintendent's response. The inspectors also determined that the finding was representative of an isolated human performance cross-cutting issue involving the failure to follow plant procedures (Section 1R04.2).

- No Color. The inspectors identified a declining human performance trend with failure of personnel to adhere to plant procedural requirements or to maintain a questioning attitude as common elements. Approximately 27 findings, which were documented as violations of NRC requirements during the previous 12 months, had a direct or credible impact on safety. This adverse performance trend is considered a cross-cutting finding not captured in individual findings (Section 4OA4).

- No Color. The inspectors identified a declining problem identification and resolution trend with not implementing timely corrective actions as common elements. Approximately 9 findings, which were documented as violations of NRC requirements during the previous 12 months, had a direct or credible impact on safety. This adverse performance trend is considered a cross-cutting finding not captured in individual findings (Section 4OA2).

Report Details

Summary of Plant Status: The facility operated at essentially 100 percent power throughout the inspection period. The licensee received approval to increase the thermal power of the plant from 2894 to 3039 megawatts thermal on October 6, 2000. On October 14, 2000, the licensee commenced the increase in power and reached 3039 megawatts thermal on October 25, 2000.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R04 Equipment Alignment (7111104)

.1 Verification of the Standby Liquid Control System

a. Inspection Scope

The inspectors performed an equipment alignment check on the standby liquid control (SLC) system to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system and thereby potentially increase risk. The inspectors reviewed documents to determine the correct system lineup and performed a walkdown to identify any discrepancies between the existing system lineup and the correct lineup. The inspectors also reviewed outstanding maintenance work requests and deficiencies which would preclude the system from performing its function and reviewed outstanding design issues and items tracked by the licensee to ensure equipment alignment problems had been properly identified and resolved. The following procedures and documents were reviewed during the assessment:

- SOP-0028, “Standby Liquid Control”
- EOP-0005, Enclosure 15, “Alternate SLC Injection and SLC TK GAL to LB Conversion”
- OSP-0009, “Authors Guide/Control and Use of Emergency Operating and Severe Accident Procedures”
- Updated Safety Analysis Report (USAR)

b. Findings

The inspectors identified an unresolved item involving the storage of chemicals for alternate SLC injection and walkdowns of equipment needed to implement emergency operating procedure enclosures.

On March 3, 1999, a quality assurance chemistry audit identified chemical storage discrepancies (i.e., deteriorated packaging) with chemicals stored in the onsite warehouse. On July 14, 1999, warehouse personnel removed the identified chemicals from inventory as a corrective action for the deteriorated packaging problems identified during the quality assurance audit. On July 28, 1999, the chemicals that had been removed from inventory were transported to the environmental storage yard and subsequently shipped offsite.

On September 14, 2000, at 9:36 a.m., chemistry personnel found a 10-inch square piece of plastic in the SLC tank during the SLC system monthly sample. The SLC tank manway cover was removed and several additional pieces of plastic were observed floating on the surface. Approximately 20 pieces of plastic which ranged in size from 2 square feet to one-half square inch were identified and removed. Further inspection of the tank with an underwater camera identified additional pieces of plastic on the sparger supports and the mixing heater located at the bottom of the SLC tank.

At 2:22 p.m., due to the potential for the plastic to enter the SLC pump suction piping, both SLC subsystems were declared inoperable. The plastic material was removed and further inspections of the SLC tank were performed with an underwater camera. No additional foreign material was identified. Additionally, a boroscope inspection of the horizontal suction piping from the tank outlet to the downward elbow was performed. No material was identified during the inspection. Based on the inspection and the buoyancy of the plastic material, the licensee concluded that the remaining SLC pump suction piping would not contain any significant amount of plastic material.

The licensee evaluated the effect that 1 square inch pieces of plastic would have on system performance. The evaluation determined that pieces of plastic less than 1 square inch which might be introduced into the SLC pump suction would not preclude the system from injecting sodium pentaborate into the reactor. The licensee believed that any remaining plastic material would be less than 1 square inch.

At 8:59 p.m., both SLC subsystems were declared operable due to the inspections of the SLC system, the removal of the plastic material, and the engineering evaluation. This event was documented in Condition Report (CR) 2000-1618.

The licensee established a significant event review team to investigate the event and evaluate the safety significance of the plastic material in the SLC tank. The licensee determined that the incremental risk from this event, assuming that the SLC system was out of service for a one year time frame, was 3.0E-8. Since the upper limit for nonrisk significant changes in conditional core damage probability was 1.0E-6, the licensee concluded the event was nonrisk significant. Due to low flow velocities and the buoyancy of the plastic, the licensee believed that the SLC system would have been able to perform its function provided the SLC storage tank was in a steady state condition. The licensee specified that additional technical evaluations and computer modeling could prove the SLC system past operability during steady state conditions. However, given their low safety significance determination result, no actions were planned by the licensee to provide an additional evaluation of operability of the SLC system.

The only time period the licensee could not postulate the SLC system's behavior was during the 10-minute air sparge before the monthly chemistry sample of the SLC tank contents. During and immediately following an air sparge, the location of the plastic material in the SLC tank could not be postulated. Therefore, the licensee concluded that the SLC system may not have been able to provide its intended safety function during periods in which the SLC tank was being sparged with air.

Procedure EOP-0005, Enclosure 15, “Alternate SLC Injection and SLC TK GAL to LB Conversion,” required that operations personnel contact the main warehouse/storeroom to transport approximately 2500 pounds each of sodium borate and boric acid to the auxiliary building. In the event the SLC system did not function, these chemicals would be mixed together to form a sodium pentaborate mixture. The mixture would then be injected into the reactor pressure vessel via the high pressure core spray system to shut down the reactor.

On September 25, 2000, the inspectors conducted a walkdown with Procedure EOP-0005, Enclosure 15, and identified that the sodium borate and boric acid chemicals were not available in the warehouse or onsite for alternate SLC use. The inspectors notified the licensee and replacement chemicals were shipped from the Grand Gulf Nuclear Station and arrived onsite at approximately 4 a.m., on September 26, 2000. Further investigation by the licensee revealed that these chemicals had been removed from the site as a followup action to the removal of the chemicals from the warehouse inventory on July 28, 1999. CR 2000-1680 was generated to document this condition.

Sections 11.1.1 and 11.1.2 of Procedure OSP-0009, “Authors Guide/Control and Use of Emergency Operating and Severe Accident Procedures,” required that operations personnel perform yearly walkdowns of each emergency operating procedure enclosure. On October 2, 2000, in response to the inspectors’ observation of the missing chemicals, operations personnel completed a review of their documentation of emergency operating procedure enclosure audits and yearly walkdowns. The review determined that the yearly walkdowns of each emergency operating procedure enclosure had not been performed since November 26, 1996. This issue was documented in CR 2000-1723.

The inspectors completed a Phase 2 significance determination process (SDP) evaluation to assess the preliminary safety significance of the SLC and alternate SLC system unavailability for greater than 30 days for the anticipated transient without scram event. The Phase 2 SDP indicated that the loss of both the normal and alternate SLC functions, with no remaining mitigation capability, would have substantial safety significance. The licensee’s probabilistic risk analysis determined that the loss of the SLC functions would be of very low safety significance. Due to the variance in the safety significance determinations, the unavailability of the normal and alternate SLC systems is considered an unresolved item pending review of the safety significance by an NRC senior reactor analyst (URI 50-458/0014-01).

.2 Improper Use of Computers by Operations Personnel

a. Inspection Scope

The inspectors performed a deep backshift assessment of control room activities.

b. Findings

The inspectors identified operations personnel using a computer inappropriately at the operations shift superintendent watch station.

Section 5.7.1 of Procedure ADM-0022, “Conduct of Operations,” specified, in part, that potentially distracting activities in the control room and other watch stations are prohibited (including reading that is not job related).

On September 29, 2000, the inspectors observed operations personnel accessing an internet site involving a barbecue grill message board on the operations shift superintendent’s computer.

The inspectors determined that the safety significance of the potentially distracting activity was low since no plant problems actually occurred during the time the material was being viewed. Nevertheless, the inspectors determined that this potentially distracting activity was more than minor since the degraded oversight of main control room activities by the operations shift superintendent while viewing this material could have a credible impact on safety. The inspectors also determined that the finding was representative of an isolated human performance cross-cutting issue involving the failure to follow plant procedures.

Technical Specification 5.4.1.a requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Section 1.b of Appendix A of Regulatory Guide 1.33 requires the licensee to have administrative procedures for authorities and responsibilities for safe operation and shutdown. Section 5.7.1 of Procedure ADM-0022, specified, in part, that potentially distracting activities in the control room and other watch stations are prohibited (including reading that is not job related). The inspectors determined that accessing a nonjob related message board at the operations shift superintendent’s watch station was a violation of Technical Specification 5.4.1.a (NCV 50-458/0014-02). This violation is in the licensee’s corrective action program as CR 2000-1709.

1R05 Fire Protection (7111105)

.1 Observation of Fire Brigade Drill

a. Inspection Scope

The inspectors observed a fire brigade drill in order to evaluate the readiness of personnel to fight fires. Aspects assessed during the drill included the ability to don protective clothing, use of emergency breathing apparatus, use of fire hoses, ability to enter the fire area in a controlled manner, staging of fire equipment at the scene, command and control by the fire brigade leader, use of emergency communications, assessment of fire propagation, use of firefighting strategies, and conduct of the fire drill and critique. The following procedures and documents were reviewed during the assessment:

- ADM-0009, "Station Fire Protection Program"

- FPP-0010, "Fire Fighting Procedure"
- TPP-7-021, "Fire Protection Training and Qualifications"
- National Fire Protection Association Standard 27, "Recommendations for Organization, Training, and Equipment of Private Fire Brigades"
- Lesson Plan FD-92-07, "Fire Drill CCP Heat Exchanger Room Elevation 95'/Auxiliary Building"
- Fire Drill Training Data Base
- Fire Hazards Analysis
- Fire Strategies

b. Findings

The inspectors identified three examples of a noncited violation of Attachment 4 to Facility Operating License 50-458, which involved the implementation of the licensee's fire brigade training program.

On October 11, 2000, the inspectors observed the fire brigade respond to a simulated fire in Motor Control Center ENB-MCC1A. Motor Control Center ENB-MCC1A supplied safety-related power to various reactor core isolation cooling system components.

Following the drill, the controller critiqued the performance of the brigade members. Items identified during the critique included one brigade member not properly donning protective clothing, not taking an extra fire hose to the scene, not bringing radio communications to the scene, and poor utilization of fire strategies. Following the critique, the controller informed the inspectors that the performance of the brigade was satisfactory with the deficiencies.

The inspectors informed the controller of several additional deficiencies which were not discussed at the critique. Specifically: (1) three additional brigade members incorrectly donned protective clothing, (2) the initial two responders went directly to the fire, in protective clothing, without any extinguishing agent, (3) the brigade leader did not establish communications between the control room and the scene, (4) the brigade leader walked directly into the affected space without checking the door for heat, did not enter the affected area low to floor elevation, and had not received a report from the brigade members dispatched to the scene, (5) the first two brigade members approached the fire from the wrong direction, then, instead of backing out, they went past the fire (3 foot space between damaged electrical equipment and wall) to find an extinguishing agent, (6) the brigade members did not know that there were no carbon dioxide fire extinguishers on the auxiliary building 95 foot elevation, (7) one brigade member was dispatched to bring a carbon dioxide extinguisher to the scene; however, no effort was made to locate and bring additional carbon dioxide extinguishers to the scene, (8) the fire hose was simulated being placed into service by one individual stating that he was simulating getting the hose ready over a period of 3-5 seconds, and no demonstration of the ability to pressurize a hose or use a hose nozzle was performed in the simulation, (9) no attempt was made to secure area ventilation, (10) no attempt was made to contact radiation protection for a fire in a radiologically controlled area, and (11) two brigade members did not actively participate in the simulated extinguishing of the fire.

In addition to the deficiencies, the inspectors identified that one individual was responsible for simulating, controlling, and evaluating the fire drill. The inspectors determined that the use of one person to conduct these activities significantly impacted the ability to conduct a meaningful assessment of brigade member performance during the drill. Additionally, no objective criteria were established to evaluate the performance of the brigade members. Following discussions of the inspectors' observations, the controller determined that the overall performance of the brigade was unsatisfactory and that the performance of the brigade leader was unsatisfactory.

Attachment 4 to Facility Operating License 50-458 specified that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report. Section 9.5.1.5 of the USAR specified that the licensee conformed to Section III.I of Appendix R to 10 CFR Part 50. Section III.I.3.e of Appendix R to 10 CFR Part 50 required, in part, that fire brigade drills include the following:

- An assessment of the place and use of equipment and firefighting strategies. The inspectors determined that the licensee did not adequately assess the inability of brigade members to locate carbon dioxide fire extinguishers or to use the fire strategies in determining the correct approach to the fire or the location of firefighting equipment.

- An assessment of each brigade member's knowledge of their role in the firefighting strategy. Assessment of the brigade member's conformance to firefighting procedures, use of equipment, communication equipment, and ventilation equipment. The inspectors determined that the licensee did not assess the knowledge of the brigade members' role in the firefighting strategy or the brigade members' conformance to firefighting procedures. The licensee inadequately assessed the failure to use emergency radios to establish communications with the main control room and the failure to secure ventilation in the affected space.

- The simulated use of firefighting equipment. The licensee did not adequately simulate the use of fire hoses. Specifically, the fire hose was not laid out, no simulation of pressurizing a hose was conducted, no demonstration of using a hose nozzle on an electrical fire was performed, and two brigade members did not participate in the use of firefighting equipment.

- Assessment of the fire brigade leader's direction of the firefighting effort. The licensee inadequately assessed the ability of the brigade leader. Specifically, the brigade leader's performance was initially considered acceptable, even though the brigade leader did not adequately consult fire strategies, did not establish communications, improperly donned protective clothing, and did not safely enter the affected fire area.

The inspectors determined that the failure to conduct a fire drill, which adequately assessed the effectiveness of the fire brigade, was a violation of Attachment 4 to Facility Operating License 50-458 and is being treated as a noncited violation (NCV 50-458/0014-03). This violation is in the licensee's corrective action program as CR 2000-1848.

Section III.I.3.b of Appendix R to 10 CFR Part 50 required, in part, that unannounced drills not be scheduled closer than 4 weeks. The inspectors determined that the licensee scheduled and conducted three unannounced drills in February 2000. The inspectors determined that the failure to properly schedule fire drills was a second example of a violation of Attachment 4 to Facility Operating License 50-458. This example was entered in the licensee's corrective action program as CR 2000-1848.

Section III.I.3.c of Appendix R to 10 CFR Part 50 required, in part, that unannounced drills shall be planned and critiqued by members of the management staff responsible for plant safety and fire protection. The inspectors determined that the controller for unannounced fire drills was a senior training instructor and not considered a member of the management staff responsible for plant safety and fire protection. The failure to use management staff to plan and critique unannounced fire drills was a third example of a violation of Attachment 4 to Facility Operating License 50-458. This example was entered in the licensee's corrective action program as CR 2000-1848.

The fire brigade training issues were considered more than minor because the inability to properly respond to a plant fire could impact the capability of mitigating systems during an actual event. The inspectors determined that the safety significance of the fire brigade training issues and fire brigade performance was very low in that plant fire barriers and automatic suppression capability were maintained in accordance with the fire protection program.

.2 Tours of Plant Areas

a. Inspection Scope

The inspectors toured auxiliary building Elevation 95, the high pressure core spray pump room, and the low pressure core spray pump room to assess the control of transient combustible material and ignition sources, operational effectiveness of fire protection equipment, and the material condition of fire barriers. The following procedures were reviewed during the assessment:

- FPP-0030, "Storage of Combustibles"
- FPP-0050, “Handling of Flammable Liquids and Gases"
- FPP-0040, “Control of Transient Combustibles”
- Fire Strategies

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (7111106)

a. Inspection Scope

The inspectors verified that the licensee's flooding mitigation plans and equipment were consistent with the licensee's design requirements and the risk analysis assumptions. The area inspected was Pipe Tunnel D. The following documents were reviewed during the assessment:

- Calculation G13.2.2, "Pipe Tunnel Flooding due to Service Water Pipe Break"
- Calculation G13.2.3 PN 317, "Max Flood Elevations for Moderate Energy Line Cracks in Cat I Structures"
- Updated Safety Analysis Report

b. Findings

No findings of significance were identified.

1R07 Heat Exchangers (7111107)

a. Inspection Scope

No risk significant heat exchangers were tested during the inspection period. Consequently, this inspection was not completed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (7111111)

a. Inspection Scope

The inspectors observed the testing of operations personnel in the simulator on October 23, 2000. The observation was performed to determine if there were deficiencies or discrepancies with the training and if the licensee's evaluators conducted an adequate critique of the training. The following procedures were reviewed as part of the assessment:

- EOP-1, "Reactor Pressure Vessel Control," Revision 16
- EOP-1A, "Reactor Pressure Vessel Control - Anticipated Transient Without Scram," Revision 16

- EOP-2, "Primary Containment Control," Revision 12
- EOP-3, "Secondary Containment and Radioactive Release Control," Revision 11
- EOP-4, "Contingencies - Reactor Pressure Vessel Flooding," Revision 8
- EOP-4A, "Contingencies - Anticipated Transient Without Scram," Revision 8

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (7111112)

.1 Review of Maintenance Rule Determination for Station Blackout Valve

a. Inspection Scope

The inspectors selected a performance problem associated with the failure of station blackout Valve SWP-AOV599 and evaluated the licensee's maintenance rule determination and corrective actions.

b. Findings

The inspectors identified one unresolved item involving implementation of corrective actions when performance monitoring criteria for the standby service water system were exceeded.

Station blackout Valve SWP-AOV599 is a nonsafety-related valve which actuates automatically during a station blackout event. With Valve SWP-AOV599 open, a cooling water return flow path to the standby service water cooling towers would be enabled. Therefore, the Division III emergency diesel generator and high pressure core spray system would be available during a station blackout event.

Valve SWP-AOV599 has a Division I standby service water maintenance rule function to supply cooling water to the Division III emergency diesel generator during a station blackout event. In addition, Procedure AOP-0050, "Station Blackout," specified that a diesel generator is not allowed to be run for more than one minute without cooling water.

On March 4, 1999, engineering personnel initiated CR 1999-0263 to evaluate the need for testing of Valve SWP-AOV599. Investigation into this condition by the licensee revealed that on March 2, 1995, testing of Valve SWP-AOV599 was deleted from the inservice testing program because the air supply was not safety related.

On March 10, 2000, the licensee tested Valve SWP-AOV599 in accordance with maintenance action item (MAI) 330213. Valve SWP-AOV599 failed to open automatically due to blown fuses on the station blackout valve air supply line control solenoid Valve SWP-SOV602C. The licensee initiated CR 2000-0531 to document the failure. Followup investigation identified that Valve SWP-SOV602C was last tested on

September 25, 1997, under Engineering Request 97-410. ER 97-410 performed a postmodification test of Valve SWP-SOV602C following a modification involving rerouting a fire protection conduit.

On March 24, 2000, engineering personnel reviewed CR 2000-0531 and determined that a maintenance rule functional failure had not occurred in Division I standby service water due to the remaining capability for manual actuation of Valve SWP-AOV599. Engineering personnel reasoned that the valve could still be operated manually, therefore, the failure to open automatically was not considered a maintenance rule functional failure.

On September 4, 2000, the inspectors conducted a review of the maintenance rule determination for CR 2000-0531. The inspectors questioned engineering personnel to determine what procedure provided guidance for operations personnel to manually operate Valve SWP-AOV599 in the event the valve did not operate automatically. Engineering personnel stated that Procedure AOP-0050 required verification that Valve SWP-AOV599 had opened during a station blackout event. The inspectors reviewed Procedure AOP-0050 and noted that there was no specific direction in the procedure to manually open Valve SWP-AOV599 if it failed to open automatically.

The inspectors also questioned main control room personnel to determine what action would be taken if Valve SWP-AOV599 did not open automatically during a station blackout event. Operations personnel informed the inspectors that the Division III emergency diesel generator would be secured due to a caution statement in Procedure AOP-0050 which did not allow operation of the emergency diesel generator for more than one minute without cooling water. The inspectors determined that it would not be realistic to assume that, within one minute, operations personnel could diagnose the event, implement immediate operator actions specified in plant procedures, recognize that cooling water was not being supplied to the Division III emergency diesel generator, and take actions to open Valve SWP-AOV599.

On September 13, 2000, the inspectors again discussed the failure of Valve SWP-AOV599 with engineering personnel to determine if the unrealistic crediting of operator actions altered their view on the initial maintenance rule functional failure determination. Engineering personnel subsequently reviewed Nuclear Energy Institute Report 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” which provided guidance on maintenance rule functional failure determinations. Question 25 (d) of NEI 93-01 asked whether or not the failure of an automatic function would be considered a maintenance rule functional failure if there was a manual or a backup function available. The NEI 93-01 response specified that it would be considered a maintenance rule functional failure unless specific credit had been taken in the accident analysis for the manual backup. In this case, there was no specific analysis for the manual backup of Valve SWP-AOV599.

Engineering personnel subsequently determined that the occurrence should have been characterized as a maintenance preventable functional failure. Engineering personnel considered that the most likely cause of the blown fuses on station blackout valve air supply line control solenoid Valve SWP-SOV602C was the performance of signature testing on standby service water Pump 2C discharge motor-operated Valve SWP-MOV40C. Since Valve SWP-SOV602C operated in conjunction with

Valve SWP-MOV40C, engineering personnel believed that some environmental condition (e.g., moisture) existing in solenoid Valve SWP-SOV602C, at the same time signature testing of SWP-MOV40C was completed, could have caused the blown fuses in Valve SWP-SOV602C. Because the signature test was a maintenance activity, engineering personnel determined that the occurrence was a maintenance preventable functional failure.

On September 15, 2000, engineering personnel initiated CR 2000-1627 to document the revised maintenance rule functional failure determination for Valve SWP-AOV599. On September 18, 2000, system engineering initiated CR 2000-1645 to document that Division I of the standby service water system had exceeded the maintenance rule performance monitoring criteria.

The inspectors determined that, at the time of the initial functional failure determination for Valve SWP-AOV599 on March 24, 2000, Division I of the standby service water system was already in maintenance rule category a(1) due to exceeding the system availability percentage of less than or equal to 99.5 percent during a rolling 18-month period. Actual Division I standby service water system availability had decreased to 99.30 percent during March 1999, and remained below 99.5 percent through June 2000. On June 12, 2000, a maintenance preventable functional failure for Division I standby service water occurred as a result of the loss of indication and power to the Division I standby service water air release valves for the auxiliary building and containment. Had the Valve SWP-AOV599 issue been determined to be a maintenance preventable functional failure on March 24, 2000, then the June 12, 2000, maintenance preventable functional failure would have been the second maintenance preventable functional failure within 18 months. Two failures within an 18-month period would have required separate entry into maintenance rule category a(1) for Division I standby service water and additional corrective actions would have been necessary.

The inspectors performed a preliminary Phase 2 SDP evaluation of the failed station blackout valve and the resultant effect on the Division III emergency diesel generator during a loss of offsite power event. The Phase 2 evaluation indicated that the issue was of very low safety significance. The inspectors also reviewed the change in core damage frequency using the licensee’s no maintenance model probabilistic risk analysis. Based on the January 1993 probabilistic risk analysis, the installation of Valve SWP-AOV599 using Modification Requests 91-0126 and 92-0012 changed the core damage frequency by 7.12E-5/year. A review of the current maintenance model probabilistic risk analysis performed by the licensee identified a change in core damage frequency of approximately 9.8E-7/year.

Due to the variance in the safety significance determinations, the issues involving the failure of station blackout Valve SWP-AOV599 to automatically open is an unresolved item pending a review of the safety significance by a NRC senior reactor analyst (URI 50-458/0014-04). This issue is in the licensee’s corrective action program as CRs 2000-1627 and 2000-1645.

.2 Review of Maintenance Rule Determinations

a. Inspection Scope

The inspectors selected the following two performance problems associated with the SLC system and evaluated the effectiveness of the licensee's corrective actions and maintenance rule determinations:

- CR 2000-0287, "Failed Lift Setpoint for Standby Liquid Control System Relief Valve"
- CR 2000-1430, "Standby Liquid Control Pressure Gauge Came Apart During System Testing"

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (7111113)

a. Inspection Scope

The inspectors evaluated the effectiveness of risk assessments performed by the licensee for the work weeks beginning October 8, 15, and 22, 2000. The following procedures were reviewed during the assessment:

- Maintenance Planning Guideline
- On-line Maintenance Guidelines
- Weekly Maintenance Schedules

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (7111115)

a. Inspection Scope

The inspectors reviewed the following documents to ensure that operability was properly justified, the components remained available, and there was not a significant increase in risk:

- CR 2000-1758, "Throttling of Residual Heat Removal Heat Exchanger Service Water Valve E12-MOVF068 A and B"
- CR 2000-1793, "Containment Unit Cooler A and B Flow Rates Out of Specification"

- MAI 334993, “Leakage on Station Blackout Valve Air Supply Line Control Solenoid Valve SWP-SOV602C”

b. Findings

On May 22, 2000, operations personnel initiated MAI 334993 to rework standby cooling tower station blackout valve air supply line control solenoid Valve SWP-SOV602C. The MAI was initiated because the solenoid leaked by, causing the Nitrogen air bottle pressure to decrease. When the MAI was initially written, the Nitrogen bottle required replacement approximately every 2 weeks. By the time the work to replace the solenoid valve was completed (August 18, 2000), the replacement interval for the Nitrogen bottle was approximately once every 3 days.

On August 16, 2000, the inspectors questioned engineering personnel to determine if an evaluation had been performed when the MAI was written on May 22, 2000, to assess if the air leakage rate would impact the ability of station blackout Valve SWP-AOV599 to remain open following a station blackout event. Additionally, the inspectors questioned whether or not the Nitrogen bottle sizing calculation accounted for air leakage between the Nitrogen bottle and the valve’s air actuator.

Engineering and operations personnel stated that an operability evaluation of the degraded condition of the air supply to Valve SWP-AOV599 had not been performed. Engineering personnel subsequently initiated CR 2000-1486 to document that an evaluation had not been performed. Additionally, CR 2000-1486 specified that shiftly monitoring ensured the in-use Nitrogen bottle was greater than 300 psig and that per calculation G13.18.2.6-31, “Nitrogen Bottle Sizing for SWP-AOV599,” the minimum bottle pressure at which adequate capacity existed to operate the valve once was 60 psig. Therefore, there was no concern that Valve SWP-AOV599 could not be actuated upon demand.

The inspectors determined that Valve SWP-AOV599 required air to remain open and that CR 2000-1486 and Calculation G13.18.2.6-31 did not consider the normal actuator air leakage or the air leakage from Valve SWP-SOV602C in the ability to maintain Valve SWP-AOV599 open for the 12-hour duration specified in the probabilistic risk assessment. The inspectors determined that, even though the ability to provide cooling water to the Division III emergency diesel generator during a station blackout event was not a function described in the USAR, it was a maintenance rule function and did contribute to a significant reduction in the contribution to core damage frequency. Therefore, the inspectors determined that the licensee did not properly assess the significance of air leakage on the ability to maintain the functional capability of Valve SWP-AOV599.

On August 18, 2000, the licensee completed as-found and as-left air drop testing of the air supply to Valve SWP-AOV599. The licensee’s testing demonstrated that, at 300 psig in the Nitrogen bottle, a sufficient air supply existed to maintain Valve SWP-AOV599 open for the 12-hour duration described in the probabilistic risk analysis.

The poor engineering review of air leakage on station blackout Valve SWP-AOV599 was of very low safety significance in that subsequent air drop testing of the system and engineering analysis demonstrated that the valve would have remained open for the 12-hour duration specified in the probabilistic safety analysis.

1R19 Postmaintenance Testing (7111119)

a. Inspection Scope

The inspectors reviewed the postmaintenance testing requirements specified for the MAIs listed below to ensure that testing activities were adequate to verify system operability and functional capability:

- MAI 336891, "Replace Hydrogen Analyzer Control Panel A Components"
- MAI 337057, “Verify Torque Switch Setting for Residual Heat Removal System Valve E12-MOVF040"
- MAI 332569, “Add Time Delay Relay to Low Pressure Core Spray Pump Minimum Flow Bypass Valve E21-MOVF011"

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (7111122)

a. Inspection Scope

The inspectors reviewed the surveillance tests listed below to verify that systems were capable of performing their intended safety functions and to ensure that requirements for Technical Specifications, the USAR, and procedures were met:

- STP-043-7301, “Containment Purge System Isolation Valve Leak Rate Test”
- STP-053-3001, ”Jet Pump Operability Test”
- STP-CSP-0100, “Chemistry Required Surveillances and Actions,” Attachment 1, “Standby Liquid Control Tank Surveillances”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (7111123)

a. Inspection Scope

No risk significant temporary modifications were implemented by the facility since the last review of the area. Consequently, this inspection was not completed.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (7111406)

a. Inspection Scope

The inspectors observed the licensee's October 17, 2000, emergency preparedness drill in order to evaluate the adequacy of the drill and critique. The event declaration and notification elements of the drill contributed to the performance indicator statistics. The following procedures and documents were reviewed during the assessment:

- NRC Emergency Preparedness Position 2, "Emergency Preparedness Position on Timeliness of Classification of Emergency Conditions,"
- EIP-2-001, "Classifications of Emergencies"
- EIP-2-002, "Classification Actions"
- EIP-2-006, "Notifications"
- EP-04, "Scenario 04 Site Drill Manual"

b. Findings

NRC Emergency Preparedness Position 2, "Emergency Preparedness Position on Timeliness of Classification of Emergency Conditions," dated August 1, 1995, specified that a 15-minute goal is a reasonable period of time for assessing and classifying an emergency once indications are available to control room operators that an emergency action level has been exceeded. The licensee determined that operations personnel failed to declare the site area emergency within 15 minutes of the indications in the main control room being available. The licensee stated that the late declaration would be included during the next submittal of performance indicator data to the NRC.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors used NRC Inspection Manual Procedure 71151, Performance Indicator Verification, to verify the accuracy and completeness of data associated with the safety system functional failures, reactor coolant system activity, and reactor coolant system leak rate performance indicators. The following procedures and documents were reviewed during the verification:

- STP-000-0001, "Daily Operating Logs"

- Performance indicator technique sheets for reactor coolant system leakage and reactor coolant system activity for July, August, and September 2000

- Performance indicator data summary report for the third quarter of 2000

- Reactor coolant system activity chemistry data for July, August, and September 2000

- River Bend Technical Specifications

- Licensee Event Reports for 1999 and 2000

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

Adverse Trend in the Problem Identification and Resolution Cross-cutting Area

The inspectors identified a declining trend in problem identification and resolution with not implementing timely corrective actions as the common element. Within the past 12 months, a number of related findings were identified. Those identified prior to the implementation of the revised Reactor Oversight Program in April 2000 were Severity Level IV violations or noncited violations. Those identified under the revised Reactor Oversight Program had very low safety significance, while one item was still unresolved. Specifically:

- Nine months prior to this, inadequate corrective actions were implemented in response to personnel not completing adequate control panel walkdowns (VIO 50-458/9915-01).

- Six months prior to this, personnel did not initiate corrective action documents for two conditions adverse to quality (NCV 50-458/0002-01).

- Five months prior to this, personnel did not implement corrective actions to prevent the recurrence of an inadvertent opening of the residual heat removal minimum flow valves (NCV 50-458/0009-01).

- Three months prior to this, personnel did not implement corrective actions to restore compliance for a minor violation involving inspections of portable fire extinguishers and did not implement corrective actions to ensure manual valves in the main flow path of safety-related systems were locked (VIO 50-458/0011-03 and NCV 50-458/0011-01).

- Two months prior to this, personnel did not implement corrective actions to ensure scaffolding was properly installed and to ensure that technical deficiencies with procedures were corrected (NCV 50-458/0013-01 and 02).

- One month prior to this, personnel did not properly characterize the failure of hydrogen ignitors as a maintenance preventable functional failure. Therefore, the system was not classified as a maintenance rule (a)(1) system and additional corrective actions were not implemented (NCV 50-458/0017-01).

- During the current inspection period, personnel did not properly characterize the failure of station blackout Valve AOV-599 as a maintenance preventable functional failure. As a result, the service water system was not classified as a maintenance rule (a)(1) system and additional corrective actions were not implemented (URI 50-458/0014-04).

The inspectors noted that the causal relationship of these findings was failure of personnel to implement timely corrective actions. The findings individually had a direct or credible impact on safety. This adverse performance trend is considered a cross-cutting finding not captured in individual findings and is characterized as "No Color."

4OA4 Cross-cutting Issues

Adverse Trend in the Human Performance Cross-cutting Area

The inspectors identified a declining human performance trend with failure to follow procedures and poor questioning attitude as the common elements. Within the past 12 months, a number of related findings were identified. Those identified prior to the implementation of the revised Reactor Oversight Program in April 2000 were Severity Level IV violations or noncited violations. Those identified under the revised Reactor Oversight Program had very low safety significance, while one item was still unresolved. Specifically:

- Twelve months prior to this, personnel were not completing adequate control panel walkdowns, did not identify unauthorized operator aids, improperly used computers in the at-the-controls area of the main control room, completed inadequate operability evaluations of degraded equipment, and did not stop work to revise a procedure for turbine valve testing (NCVs 50-458/9913-01, 02, and 05).

- Ten months prior to this, personnel did not implement requirements for cold weather protection and did not complete adequate walkdowns of heat trace panels (NCV 50-458/9914-01).

- Nine months prior to this, personnel were not completing adequate control panel walkdowns and did not perform required radiological surveys (VIO 50-458/9915-01 and NCV 50-458/9915-02).

- Seven months prior to this, personnel installed an unauthorized operator aid, left a residual heat removal heat exchanger vent valve out of position, did not report an automatic engineered safety features actuation, did not ensure qualified personnel were conducting scram time testing, caused an inadvertent isolation of the reactor core isolation cooling system, failed to wear required anticontamination clothing, and made an unauthorized entry into a high radiation area (NCVs 50-458/0001-01, 02, 03, 04, and 05).

- Six months prior to this, personnel did not prepare operability evaluations for degraded conditions and failed to make a required notification of a condition prohibited by Technical Specifications (NCVs 50-458/0002-01 and 03).

- Five months prior to this personnel did not complete an adequate closeout of the drywell and did not obtain a control rod drop analysis before withdrawing control rods (NCVs 50-458/0009-01 and 02).

- Four months prior to this, personnel did not specify the appropriate postmaintenance testing requirements in four maintenance packages (NCV 50-458/0010-01).

- Three months prior to this, personnel did not ensure that penetrations for drywell purge isolation valves were isolated (NCV 50-458/0011-02).

- Two months prior to this, personnel did not complete adequate operability evaluations for three degraded conditions and radioactive waste was not properly classified in two waste shipments (NCV 50-458/0013-03 and 04).

- During the current inspection period, personnel did not verify that boron was available for alternate SLC injection, improperly used an operations watch station computer, and did not adequately assess a fire brigade drill (URI 50-458/0014-01and NCVs 50-458/0014-03 and 04).

The inspectors noted that the causal relationship of these findings was a failure of personnel to adhere to procedural requirements in plant procedures or to maintain a questioning attitude. The findings individually had a direct or credible impact on safety. This adverse performance trend is considered a cross-cutting finding not captured in individual findings and is characterized as “No Color.”

4OA6 Management Meetings

Exit Meeting

The inspectors presented the results of the inspection to Mr. Dwight Mims, General Manager Plant Operations, and other members of licensee management at the conclusion of the inspection on November 17, 2000.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Biggs, Coordinator, Licensing
W. Brian, Director, Engineering
E. Bush, Superintendent Operations
R. Edington, Vice President-Operations
J. Fowler, Manager, Quality Assurance
T. Hildebrandt, Manager, Maintenance
J. Holmes, Manager, Technical Support
R. King, Director, Nuclear Safety Assurance
J. Leavines, Manager, Licensing
J. McGhee, Manager, Operations
D. Mims, General Manager
D. Myers, Senior Specialist, Licensing
A. Shahkarami, Manager, System Engineering
M. Wyatt, Manager, Planning and Scheduling/Outage

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-458/0014-02	NCV	Unauthorized use of computer at an operations watch station (Section 1R04.2)
50-458/0014-03	NCV	Failure to adequately assess and conduct fire drills (Section 1R05.1)

Opened

50-458/0014-01	URI	Safety significance determination for unavailability of the normal and alternate standby liquid control systems (Section 1R04.1)
50-458/0014-04	URI	Safety significance determination for the failure of station blackout valve to open automatically (Section 1R12.1)

LIST OF ACRONYMS AND INITIALISMS USED

CFR	Code of Federal Regulations
CR	condition report
MAI	maintenance action item
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
SDP	significance determination process
SLC	standby liquid control
URI	unresolved item
USAR	Updated Safety Analysis Report

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.