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

August 23, 2000

EA-00-196

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. 50-458/00-11 AND NOTICE OF VIOLATION

Dear Mr. Edington:

On August 5, 2000, the NRC completed inspections at your River Bend Station facility. The enclosed report presents the results of these inspections. The results of the inspections were discussed with Dwight Mims, General Manager Plant Operations, and other members of your staff.

This inspection was an examination of 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of the inspection, the NRC has determined that a violation of NRC requirements occurred. The violation is cited in the enclosed notice of violation (notice) and the circumstances surrounding it are described in detail in the subject inspection report. The violation is being cited in the notice because plant staff did not inspect 17 portable fire extinguishers at 30 day intervals between March 2000 and July 5, 2000, even though your staff was informed of the noncompliance with fire protection program requirements on April 10, 2000, and at an exit meeting on May 9, 2000.

You are required to respond to this letter and should follow the instructions specified in the enclosed notice when preparing your response. The response should include those actions you will take to ensure that violations of NRC requirements are resolved within a reasonable time after they are identified. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Based on the results of this inspection, the NRC has also identified two issues that were evaluated under the risk significance determination process as having very low safety significance (green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. The noncited violations are described in the subject inspection report. If you contest the violation or significance of these noncited 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, 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 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

Enclosures:
1. Notice of Violation
2. NRC Inspection Report No.
 50-458/00-11

cc w/enclosures:
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 Drector **(ATH)**
Senior Resident Inspector **(TWP)**
Branch Chief, DRP/B **(WDJ)**
Senior Project Engineer, DRP/B **(RAK1)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
RBS Site Secretary **(PJS)**
Dale Thatcher **(DFT)**
G. F. Sanborn, D:ACES **(GFS)**
K. D. Smith, RC **(KDS1)**
R. W. Borchardt, OE **(RWB1)**
OE:EA File **(RidsOeMailCenter)**

DOCUMENT NAME:R:_RB\2000\RB2000-11RP-TWP.wpd

RIV:RI:DRP/B	SRI:DRP/B	RP:DRS/PSB	SHP:DRS/PSB	C:DRS/PSB
SMSchneider	TWPruett	DRCarter	MPShannon	GMGood
WDJ for	/RA/ - E to WDJ	/RA/	/RA/	/RA/
8/22/00	8/21/00	8/18/00	8/21/00	8/21/00
C:DRP/B	D:ACES	D:DRP		
WDJohnson	GFSanborn	KEBrockman		
/RA/	/RA/	EEC Acting for		
8/22/00	8/22/00	8/23/00		

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

ENCLOSURE 1

NOTICE OF VIOLATION

Entergy Operations, Inc.
River Bend Station

Docket No. 50-458
License No. NPF-47
EA-00-196

During an NRC inspection conducted on June 25 through August 5, 2000 a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> Facility Operating License No. NPF-47, Attachment 4, "Fire Protection Program Requirements," License Condition 1, specified, in part, that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Safety Analysis Report.
>
> Updated Safety Analysis Report Sections 9.5.1.2.11, “Portable Extinguishers,” and 9A.3.6.6, “Portable Extinguishers,” specified that portable fire extinguishers are provided utilizing the guidance from National Fire Protection Association Standard 10, "Portable Fire Extinguishers."
>
> National Fire Protection Association Standard 10, Section 4-3.1, “Frequency,” specified that fire extinguishers shall be inspected when initially placed in service and thereafter at approximately 30 day intervals.
>
> Contrary to the above, between June 1999 and March 2000, 17 portable fire extinguishers located in high radiation areas in the reactor core isolation cooling pump room, radioactive waste building, and turbine building, were not inspected at 30 day intervals. In addition, the licensee did not inspect the 17 portable fire extinguishers at 30-day intervals between March 2000 and July 5, 2000, even though they had been informed of the noncompliance with fire protection program requirements on April 10, 2000, and at an exit meeting on May 9, 2000.
>
> This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Entergy Operations, Inc. is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this notice, within 30 days of the date of the letter transmitting this notice of violation (notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this notice, an order or a Demand for

Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available to the public, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the basis for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790 (b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this day of August 2000.

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/00-11
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	June 25 through August 5, 2000
Inspectors:	T. W. Pruett, Senior Resident Inspector S. M. Schneider, Resident Inspector M. P. Shannon, Senior Health Physicist D. R. Carter, Health Physicist
Approved By:	William D. Johnson, Chief, Branch B Division of Reactor Projects

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

SUMMARY OF FINDINGS

River Bend Nuclear Station
NRC Inspection Report 50-458/00-11

IR 05000458-00-11; on 6/25-8/5/2000; Entergy Operations, Inc; River Bend Station. Integrated Resident/Regional Report. Equipment Alignment, Fire Protection, Performance Indicator Verification.

The report covers a 6-week period of resident inspection and an announced inspection by two regional health physicists. The significance of issues is indicated by their color (green, white, yellow, or red) and was determined by the significance determination process in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- Green. The inspectors determined that the licensee did not implement adequate corrective actions in response to noncited violation 50-458/9913-01, to ensure manual valves in the main flow path of safety related systems were locked. Consequently, the inspectors identified approximately 70 manual valves which were not locked as required by plant procedures and the Updated Safety Analysis Report. The failure to implement corrective actions was a violation of Criterion XVI of Appendix B to 10 CFR Part 50. This issue was entered into the licensee's corrective action system as Condition Reports 1999-1557 and 2000-1405.

  The safety significance of this issue was very low because the unlocked manual valves were in the correct position for plant operation. Therefore, the safety function of the associated systems was not affected (Section 1R04.1).

- Green. The licensee failed to adequately perform surveillance testing to verify that the drywell purge isolation valves were sealed closed. Consequently, two drywell penetrations were inoperable during MODE 1 operations. The failure to seal closed the affected drywell purge isolation valve penetrations by isolating their motive air was a violation of Technical Specification 3.6.5.3. The circumstances involving this issue were discussed in Licensee Event Report 50-458/0009. This issue was entered into the licensee's corrective action system as Condition Report 2000-1139.

  The safety significance of this issue was very low because the drywell purge isolation valves were administratively controlled by tags in the main control room and a caution note in plant procedures specified that drywell purge was not to be operated while in MODE 1, 2, or 3. Therefore, the inspectors determined that the drywell purge valves should have remained closed during accident conditions (Section 1R04.2).

- No color. The inspectors determined that fire protection personnel did not implement corrective actions to restore compliance in response to a minor violation identified on April 10, 2000, which involved the failure to complete inspections of portable fire extinguishers located in high radiation areas. During tours of the auxiliary building on June 25, 2000, the inspectors again determined that fire protection personnel were not completing inspections of portable fire extinguishers located in high radiation areas. The failure to perform inspections of fire extinguishers was a Severity Level IV violation of License Condition 1 of Attachment 4 to Facility Operating License No. NPF-47. This issue was entered into the licensee's corrective action system as Condition Report 2000-0969.

Fire protection personnel failed to implement corrective actions to restore compliance within a reasonable period of time. The safety significance of this issue was very low because redundant methods of automatic and manual fire suppression were available (Section 1R05).

Report Details

Summary of Plant Status: The plant operated essentially at 100 percent power throughout the inspection period.

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors toured the standby cooling tower area, the diesel generator building, and the protected area in order to assess the effectiveness of the licensee’s actions for mitigating adverse weather conditions involving tornados, hurricanes, and high winds. The following procedures were reviewed as part of the assessment:

- ADM-0018, "Plant Housekeeping"
- ADM-0081, "Cleanliness Control"
- AOP-0029, “Severe Weather Operation”
- EDS-ME-002, "Control of Loose Items"
- RBNP-0089, “Hurricane Readiness”
- River Bend Station Emergency Plan
- River Bend Station Individual Plant Examination of External Events
- Updated Safety Analysis Report (USAR)

b. Findings

There were no findings identified.

1R04 Equipment Alignment

.1 Verification of Residual Heat Removal and Standby Gas Treatment Systems

a. Inspection Scope

The inspectors performed a partial equipment alignment check on standby gas treatment Train B and residual heat removal (RHR) Trains B and C to verify that the systems were properly configured. The following procedures were reviewed during the assessment:

- EDP-AA-77, "Control of Locked Valves List"

- EOP-0005, Enclosure 32, "Defeating Shutdown Cooling Injection Valves Isolation Interlocks"

- EOP-0005, Enclosure 36, "Defeating Shutdown Cooling Isolation Interlocks"

- GMP-0101, "Scaffold Installation and Removal"
- SOP-0031, "Residual Heat Removal System"
- SOP-0043, "Standby Gas Treatment System"
- USAR

b. Findings

The inspectors found that the licensee failed to implement adequate corrective actions in response to NCV 50-458/9913-01, which involved the failure to lock safety related manual valves.

NRC Inspection Report 50-458/9913 described an NCV for the licensee's failure to lock approximately 20 manual valves in the main flow path of safety related systems involving emergency diesel generators (EDGs) and instrument air. The licensee initiated Condition Report (CR) 1999-1557 in response to the violation. On December 9, 1999, as part of the corrective actions for CR 1999-1557, the licensee completed a review of all safety related systems to determine which safety related manual valves required locking devices.

On July 17, 2000, during equipment alignment checks of the RHR system, the inspectors identified that manual valves in the main flow path of the service water supply to the RHR room unit coolers were not locked. The inspectors reviewed the evaluation completed for CR 1999-1557 and determined that the licensee had evaluated unit cooler valves as not requiring locking devices because mispositioning of the valves would be identified during startup and that mispositioning of the valves would be apparent since area temperatures would increase.

The inspectors questioned engineering personnel on the assumptions used to determine potentially mispositioned unit cooler valves. Specifically, how long a delay would be required before an area temperature alarm was actuated; would the valves be accessible following an accident, would an automatic isolation of the equipment in the room occur on high temperature, and would seasonal conditions (low service water temperature, low demand for cooling water, and low room temperature) potentially mask a valve which was out of position.

The inspectors also reviewed additional exceptions to locking valves described in CR 1999-1557 and determined that engineering personnel did not provide an adequate basis for cooling water to the EDGs and the main steam isolation valve leakage control system compressors. For the EDGs, engineering personnel specified that a flow element was available to monitor flow and a mispositioned valve would be apparent due to a low flow condition. Engineering did not specify if a low flow condition would result in an alarm or if the flow rate was verified during routine monitoring of the system. For the main steam isolation valve leakage control system compressors, engineering personnel stated that the compressors periodically cycle on and off and a loss of service water would result in a high discharge air temperature which would alarm in the main control room. Engineering personnel did not specify if the duration of the periodic operation of the compressors was sufficient to result in a high temperature condition.

Following the discussions with the inspectors, engineering personnel initiated CR 2000-1405 to document that the original disposition of CR 1999-1557 did not receive an adequate review of items that affected different disciplines. Engineering personnel initiated a corrective action item to reevaluate the exceptions to locking valves noted in CR 1999-1557. Additionally, operations personnel initiated corrective actions to lock approximately 70 safety-related manual valves in the main flow path of unit coolers.

The inspectors determined that not locking manual valves in the main flow path of safety related systems was of very low safety significance. The unlocked manual valves were in the correct position; therefore, the safety function of the associated systems was not affected.

Criterion XVI of Appendix B to 10 CFR Part 50 requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. The inspectors determined that the licensee did not implement corrective actions to ensure manual valves in the main flow path of safety related systems were locked as required by Attachment 1, “Design Criteria for Locking or Sealing Manual Valves,” of Procedure EDP-AA-77, “Control of Locked Valves List," and Section II.K.1.5 of Appendix 1A of the USAR. The failure to implement corrective actions is a violation of Criterion XVI of Appendix B to 10 CFR 50, which is being treated as a noncited violation (NCV 50-458/0011-01). The issue was entered into the licensee's corrective action system as CRs 1999-1557 and 2000-1405.

.2 <u>(Closed) Licensee Event Report (LER) 50-458/0009</u>: Noncompliance with Technical Specifications for drywell purge valve configuration during power operations. One noncited violation was identified for the failure to verify that drywell purge isolation valves were sealed closed as required by the Technical Specifications.

Technical Specification Bases B.3.6.5.3, "Drywell Isolation Valves," specified that the drywell purge system is used to remove trace radioactive airborne products prior to personnel entry. The drywell vent and purge system is not used in MODE 1, 2, or 3; therefore, the drywell purge isolation valves are sealed shut during power operation.

Technical Specification Surveillance Requirement 3.6.5.3.1 requires the licensee to verify each 24 inch drywell purge isolation valve is sealed closed every 31 days. The Technical Specification bases specified that each drywell purge isolation valve must have the motive power to the valve operator removed. The inspectors determined that the method for removal of motive power from the drywell purge isolation valves was by isolating the air supply to the valve operators.

On May 19, 2000, with the plant operating in MODE 1, training personnel identified that the air supply to the four air-operated drywell purge isolation valves had not been isolated. Consequently, the drywell purge valves were not sealed closed as required by Technical Specifications. The licensee subsequently determined that the drywell purge isolation valves had not been sealed closed between April 6 and May 19, 2000.

On April 5, 2000, while making preparations for reactor startup, operations personnel miscommunicated procedural requirements and failed to complete Step B-16 of Procedure GOP-0001, "Plant Startup." Specifically, Step B-16 of Procedure GOP-0001 required operations personnel to verify that the switches are tagged out for drywell purge backup isolation Valves HVR-AOV125 and HVR-AOV126 and drywell purge isolation Valves HVR-AOV147 and HVR-AOV148. Additionally, Step B-16 of Procedure GOP-0001 required operations personnel to verify that the air supply to the drywell purge isolation valves had been removed. Operations personnel tagged the control switches for the valves, but did not remove the air supply from the valve actuators.

On April 25, 2000, due to an inadequate procedure, operations personnel failed to recognize that the air supply valves to the drywell purge isolation valves were open. Specifically, Section 1-7 of Procedure STP-000-0201, "Monthly Operating Logs," included instructions to verify that drywell purge isolation valves were locked closed. However, Section 1-7 of Procedure STP-000-0201 did not include specific instructions to verify that the air supply valves were isolated.

The inspectors determined that the failure to ensure the drywell purge isolation valves were sealed closed was of very low safety significance. Between April 6 and May 19, 2000, the licensee had administratively controlled the drywell purge isolation valves by placing tags on the control switches in the main control room. Additionally, a caution note in Section 5.8 of Procedure SOP-0059, "Containment HVAC System," specified that operation of drywell purge while in MODE 1, 2, or 3 was in conflict with Technical Specification 3.6.5.3 and that drywell purge was not to be operated while in MODE 1, 2, or 3. Therefore, the inspectors determined that the drywell purge valves should have remained closed during accident conditions.

Technical Specification Surveillance Requirement 3.6.5.3.1 required that each 24-inch drywell purge isolation valve be verified sealed closed every 31 days. Technical Specification Surveillance Requirement 3.0.1 specified, in part, that a failure to meet a surveillance requirement is a failure to meet the limiting condition for operation. Technical Specification 3.6.5.3 required that each drywell isolation valve be operable. With one or more penetration flow paths with two drywell isolation valves inoperable, the affected penetration must be isolated within 4 hours, or be in MODE 3 in 12 hours and MODE 4 within 36 hours. The inspectors determined that the licensee failed to adequately perform surveillance testing to verify that the drywell purge isolation valves were sealed closed. Consequently, two drywell penetrations were inoperable during MODE 1 operations. The failure to isolate the affected drywell purge isolation valve penetrations (including deactivating the isolation valves by removing their motive power) within 4 hours is a violation of Technical Specification 3.6.5.3, which is being treated as a noncited violation (NCV 50-458/0011-02). The issue was entered into the licensee's corrective action system as CR 2000-1139.

.3 <u>(Closed) LER 50-458/0008</u>: Unplanned isolation of reactor water cleanup system. The inspectors reviewed this LER and CR 2000-1104. The inspectors determined that the issue is minor and warrants no additional inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured standby gas treatment Room B and RHR Pump Rooms B and C to assess the control of transient combustible material, operational effectiveness of fire protection equipment, and the material condition of fire barriers. Additionally, the inspectors reviewed the adequacy of corrective actions for a minor violation involving monthly inspections of portable fire extinguishers in high radiation areas. The following procedures were reviewed during the assessment:

- FPP-0030, "Storage of Combustibles"
- STP-251-3201, "Fire Hose Station Visual Inspection"
- STP-251-3603, "Fire Hose Station Hose Removal, Rerack, and Inspection"
- National Fire Protection Standard 10, "Portable Fire Extinguishers"
- USAR

b. Findings

There were no findings identified during tours of the standby gas treatment Room B and RHR Pump Rooms B and C.

One noncited violation was identified for the failure to complete inspections of 17 portable fire extinguishers in high radiation areas. USAR Sections 9.5.1, "Fire Protection System," and 9A.2, "Fire Hazards Analysis," specified that portable fire extinguishers follow the guidance provided in National Fire Protection Association (NFPA) Standard 10, "Portable Fire Extinguishers." No exception to the guidance in NFPA Standard 10 was documented in the USAR. Additionally, USAR Section 9A.2, specified that portable fire extinguishers are a part of the fire suppression system. The high radiation areas in which the portable fire extinguishers were not inspected (reactor core isolation cooling room (RCIC), radwaste building, and turbine building) each credited the use of portable fire extinguishers as a method of fire suppression.

On April 10, 2000, during a tour of the RCIC pump room, the inspectors identified that the dry chemical portable fire extinguisher had not been inspected between June 1999 and March 2000. During subsequent discussions with fire protection personnel, the inspectors determined that 17 portable fire extinguishers located in high radiation areas within the auxiliary building, radwaste building, and turbine building, were not inspected at the required monthly frequency.

Fire protection personnel stated that the inspections had not been completed because lower tier Procedure FPP-0095, “Fire Extinguisher Inspection and Maintenance,” did not require monthly inspections of portable fire extinguishers in high radiation areas. Specifically, Section 7.2 of Procedure FPP-0095 specified that if abnormal radiological conditions are present which rendered an area inaccessible, a normally high radiation area, then the data for the extinguisher should be marked N/A. Procedure FPP-0095 also specified that the inaccessible extinguisher should be tested as soon as practical after the abnormal radiological condition is restored to normal or during the next cold shutdown exceeding 7 days.

The licensee acknowledged that there were no exceptions for high radiation areas in the fire hazards analysis and that the monthly inspections should have been performed. The inspectors dispositioned the issue as a minor violation which was not documented in an NRC inspection report because the issue appeared to be isolated to portable fire extinguishers in high radiation areas, there were redundant methods of automatic and manual fire suppression available, and the deficiency was entered into the corrective action program as CR 2000-0969.

During a tour of the RCIC room on June 25, 2000, the inspectors determined that the licensee had not developed or initiated corrective actions to complete the monthly inspections of portable fire extinguishers in high radiation areas. Fire protection personnel were in the process of developing a proposed exception to completing inspections of portable fire extinguishers in high radiation areas; however, no attempt was made by fire protection personnel to restore compliance by completing the requisite inspections on an interim basis until the final corrective actions could be developed, reviewed, and approved. The inspectors determined that the failure to perform inspections of portable fire extinguishers was of more than a minor concern because fire protection personnel did not implement corrective actions to restore compliance within a reasonable period of time.

License Condition 1 of Attachment 4 to Facility Operating License No. NPF-47 specified, in part, that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the USAR. USAR Sections 9.5.1.2.11, “Portable Extinguishers,” and 9A.3.6.6, “Portable Extinguishers,” specified that portable fire extinguishers are provided utilizing the guidance from NFPA Standard 10. Additionally, USAR Section 9A.2, “Fire Hazards Analysis,” specified, in part, that for fire suppression, portable extinguishers are provided throughout the building. Section 4-3.1, “Frequency,” of NFPA Standard 10, specified that fire extinguishers shall be inspected when initially placed in service and thereafter at approximately 30 day intervals. Failure to perform monthly inspections of fire extinguishers as specified by NFPA Standard 10 is a violation of License Condition 1 of Attachment 4 to Facility Operating License No. NPF-47 (VIO 50-458/0011-03). The issue was entered into the licensee's corrective action system as CRs 2000-0969 and 2000-1459.

On July 6, 2000, fire protection and maintenance personnel completed the inspections of the 17 portable fire extinguishers in high radiation areas. No deficiencies were identified during the inspections. Additionally, the licensee initiated a review of all fire protection inspection requirements associated with equipment located in high radiation areas.

1R06 Flood Protection

a. Inspection Scope

The inspectors reviewed the licensee's adequacy of procedures for coping with external flooding. In addition, the inspectors completed walkdowns of selected external flood protection barriers to assess the licensee's susceptibility to external flooding. The following procedures were reviewed as part of the assessment:

- River Bend Station Emergency Plan
- River Bend Station Individual Plant Examination
- River Bend Station Individual Plant Examination of External Events
- USAR

b. Findings

There were no findings identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed the testing of operations personnel in the simulator on July 27, 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

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors selected the following four performance problems associated with the standby gas treatment system and RHR system and evaluated the effectiveness of the licensee's corrective actions and maintenance rule determinations.

- CR 1999-2025, "Closure of Standby Gas Treatment System Dampers"
- CR 2000-0931, "Downscale RHR Line Break Instrument"
- CR 1999-0863, "RHR A Check Valve Indicates Full Open When Closed"
- CR 1999-1662, "Inadvertent Closure of Primary Containment Isolation Valve E12-F105"

b. Findings

During a review of system event failure determinations associated with the RHR system, the inspectors identified CR 1999-1662, which described an inadvertent isolation of RHR C pump suction Valve E12-F105, while conducting maintenance on alternate decay heat removal system Valve E12-F067. While repacking Valve E12-F067, maintenance personnel removed the torque arm and caused the limit switch to indicate the valve was open. Valve E12-F067 was interlocked with Valve E12-MOV105, such that if the limit switch for Valve E12-F067 indicated open, Valve E12-F105 would automatically close.

During the maintenance activity, the licensee had already declared RHR Train C inoperable for an unrelated reason. However, RHR Train C was considered available because it was able to perform its intended safety function of supplying water to the reactor vessel following a loss of coolant accident.

Engineering personnel completed a maintenance rule functional failure review and determined that a functional failure had not occurred because the system function was not required while RHR Train C was inoperable. Engineering personnel also determined that the inadvertent isolation made RHR Train C switch from an available to unavailable status.

The inspectors determined that a maintenance preventable functional failure had occurred because the maintenance on Valve E12-F067 resulted in the unplanned loss of the maintenance rule function of RHR Train C. In response, engineering personnel issued CR 2000-1411 and submitted a frequently asked question to the NRC to obtain clarification on when a system function is required per the maintenance rule. The inspectors considered the review of functional failure determinations for systems which have been declared Technical Specification inoperable, but available for performing their intended safety functions, an unresolved item pending a review by NRC personnel of the licensee's frequently asked question (URI 50-458/0011-04).

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated the effectiveness of risk assessments performed by the licensee for the work weeks beginning July 2, 9, and 23, 2000. The following procedures were reviewed during the assessment:

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

b. Findings

There were no findings identified.

1R15 Operability Evaluations

a. Inspection Scope

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

- CR 2000-1169, "Voltage Calculation E-225 did not Reflect Normal Lineup per Station Operating Procedures"

- CR 2000-1255, "Air Blowing From Switchgear Ventilation Unit HVC-ACU2A"

b. Findings

There were no findings identified.

1R19 Postmaintenance Testing

a. Inspection Scope

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

- MAI 336303, "RCIC Pump Discharge Header Pressure Relief Valve"
- MAI 336294, "RCIC Lube Oil Cooler Pressure Control Valve"
- MAI 311105, "Refurbishment of Actuator for Valve 1E12-MOVF074B"
- MAI 324011, "Replace E22-S001 K21, 27, 42, and 56 HFA Relays"

b. Findings

There were no findings identified.

1R22 Surveillance Testing

.1 Review of Surveillance Testing

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-204-1301, “Low Pressure Coolant Injection Pump B Start Time Delay”
- STP-204-6302, “Division II Low Pressure Coolant Injection (RHR) Quarterly Pump and Valve Operability Test”
- STP-204-6304, “Division II RHR Quarterly Valve Operability Test”

b. Findings

There were no findings identified.

.2 (Closed) LER 50-458/9703: High pressure core spray (HPCS) breaker trip during testing. The inspectors reviewed this LER and determined that the issue is minor and warrants no additional inspection.

1R23 Temporary Plant Modifications

a. Inspection Scope

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

b. Findings

There were no findings identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (7112102)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel throughout the controlled access area, conducted independent radiation surveys of selected work areas, and reviewed the following items, to determine whether the licensee has an adequate program to maintain occupational exposure as low as is reasonably achievable.

- ALARA program procedures
- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- ALARA job packages for Refueling Outage 9's source range monitor work and inservice inspection work which resulted in some of the highest personnel collective exposures during the inspection period
- Hot spot tracking and reduction program
- Use of engineering controls to achieve dose reductions
- Individual exposures of mechanical maintenance and instrument and controls work groups
- Plant related source term data, including source term control strategy
- Radiological work planning
- Declared pregnant worker dose monitoring controls
- Eleven ALARA related condition reports
- Problem identification and resolution

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors used NRC Inspection Manual Temporary Instruction 2515/144, "Performance Indicator Data Collecting and Reporting Process Review," to verify that the licensee properly implemented NRC and industry guidance for performance indicator data collecting and reporting. In addition, the inspectors used NRC Inspection Manual Procedure 71151, "Performance Indicator Verification," to verify the accuracy and completeness of data associated with the safety system unavailability performance indicator (emergency ac power system and HPCS system), unplanned scrams per 7,000 critical hours, and scrams with a loss of normal heat removal for the period of January 1 through June 30, 2000.

b. Findings

The inspectors determined that the licensee understood the indicator definitions, data reporting elements, and calculational methods for performance indicators involving unplanned power changes per 7,000 critical hours, emergency response organization drill participation, occupational exposure control effectiveness, and protected area security equipment. Additionally, no findings were identified with the accuracy and completeness of performance indicator data associated with unplanned scrams per 7,000 critical hours and scrams with a loss of normal heat removal.

Safety System Unavailability Data Collection

Section 2.2 of Revision 0 to Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," specified that if the unavailability of a support system causes a train to be unavailable, then the hours the support system was unavailable are counted against the train as either planned or unplanned unavailability hours. Draft Revisions C and D of NEI 99-02 specified that safety system unavailability data reported as part of the World Association of Nuclear Operators (WANO) performance indicators may be used in the January 2000 report without modification beyond correction of known reporting errors. Engineering personnel stated that the licensee used the data collected for the WANO performance indicators in their January 2000 submittal to the NRC.

In April 2000, during a review of the safety system unavailability performance indicators, the inspectors identified that the data reported by the licensee did not include the unavailability of support systems for some monitored systems. For example, when a standby service water (SWP) pump was removed from service, the licensee did not include the unavailability time of the SWP system in the unavailability of the RHR, EDG, RCIC, and HPCS systems. Additionally, not all periods in which unit coolers for monitored systems were removed from service were counted as unavailable hours. The inspectors questioned the licensee on the validity of the performance indicator data and determined that engineering personnel were not aware that the WANO performance indicator data erroneously did not include support system unavailability.

The inspectors determined that quality assurance personnel had completed two surveillances of performance indicator data. Surveillance Reports 20001002 and 20004001 each documented that performance indicator data submitted to the NRC for mitigating systems was in conformance with NEI 99-02. The inspectors determined that quality assurance personnel missed two opportunities to identify the omission of support system unavailability in the performance indicator data submitted to the NRC.

Because support system unavailability was not properly accounted for in the performance indicator data, the inspectors determined that the licensee did not have a clear understanding of the data reporting elements and indicator definitions for safety system unavailability. The licensee initiated CR 2000-1213 to resolve the issue.

The inspectors determined that the licensee credited the use of the alternate decay heat removal system as a method of RHR when either or both of the Technical Specifications required RHR trains were removed from service. When the alternate decay heat removal system was used in place of the RHR system, the licensee recorded the entire duration as unavailable hours for both trains of the RHR system. The licensee did not collect or report data on the availability/unavailability of the alternate decay heat removal system. At the end of the inspection period, the licensee was evaluating whether or not the alternate decay heat removal system should be credited when the system is used in place of the RHR system. The determination on the tracking of unavailable hours for periods when the alternate decay heat removal system is used in place of the RHR system is an unresolved item pending further review by NRC personnel (URI 50-458/0011-05).

Effect of Support System Unavailability on Performance Indicators

Section 9.2.7 of the USAR specified that the SWP system consists of four 50 percent capacity pumps. Two pumps are provided in each redundant supply header. One SWP pump is capable of meeting the cooling requirements of all equipment with the exception of a RHR heat exchanger. The RHR heat exchangers are not required during the initial phase of an event and are manually aligned to the suppression pool cooling mode after approximately 30 minutes. Only one train of RHR is required to provide adequate cooling to the suppression pool. The two redundant divisions of SWP also merge to supply a single component in two locations, the HPCS diesel generator jacket water cooler and the HPCS pump room unit cooler.

In June 2000, the licensee stated that removal of one SWP pump from service resulted in the unavailability of divisional loads supplied by the affected train of SWP and the HPCS components. Specifically, with one of the two divisional SWP pumps removed from service and a single failure resulting in a loss of the redundant division, only one 50 percent SWP pump would be available to supply cooling water to essential equipment following a plant event. A single SWP pump was not capable of meeting all the safety functions of the SWP system.

The licensee reanalyzed the data for the fourth quarter of 1999 and the first quarter of 2000 and determined that the inclusion of support system unavailability increased the total number of unavailable hours for the monitored systems. The licensee also identified instances where unavailability hours should not have been reported. For example, the licensee had not excluded planned overhaul maintenance completed on-line from the unavailable hours reported to the NRC. Nevertheless, the licensee determined that the resultant increase in hours did not cross the threshold for changing any of the safety system unavailability performance indicators from GREEN to WHITE.

The inspectors reviewed the limiting condition for operation tracking log for the SWP system between November 1, 1997, and April 1, 2000. Using the licensee's revised methodology, the inspectors determined that had a full 12 quarter review been completed, the HPCS performance indicator would have changed from GREEN to WHITE. Therefore, the inspectors questioned engineering personnel to determine if a full 12 quarter review would be performed given the known reporting error in the WANO data.

In July 2000, the licensee reevaluated the effect of the removal of a SWP pump from service and determined the following:

- With one SWP pump removed from service, only the affected train of RHR would incur unavailability hours. Specifically, the licensee determined that the remaining SWP pump in the affected train would not be able to supply water to the RHR heat exchanger to support suppression pool cooling 30 minutes following an event. Additionally, the licensee stated that the redundant train of RHR, which would have two available SWP pumps, could be aligned to the suppression pool.

- HPCS components would not be unavailable with one SWP pump removed from service. Specifically, the redundant train of SWP would have two SWP pumps in service and could provide cooling water flow to HPCS components. Additionally, one SWP was capable of supplying cooling water to all components with the exception of the RHR heat exchanger which required a manual alignment by operations personnel.

- The licensee determined that the 12 quarters of WANO data submitted to the NRC would not be reevaluated. The licensee determined that performance indicator data was not readily accessible to complete a 12 quarter review and decided to revise the submittal to the NRC to include a revision of data beginning on January 1, 1999. Specifically, the data was not easily retrievable from the main control room logs and the licensee had not consistently documented the effect of removing a support system from service on the monitored system. Additionally, fault exposure hours were to be included starting the fourth quarter of 1999.

- The licensee submitted a frequently asked question, dated July 11, 2000, to have NEI evaluate the affect on monitored system unavailability when one of four 50 percent SWP pumps was unavailable.

The inspectors reviewed NEI 99-02 Revision 0 and had the following concerns regarding the licensee's methodology for reporting performance indicator data:

- Section 2.2 of NEI 99-02, Revision 0, Page 33, Line 30, specified limitations on the source of cooling water. Specifically, unavailable hours for emergency generators need not be reported when cooling water provided by a pump powered from another class 1E power source can be substituted, provided that a pump will maintain electrical redundancy requirements such that a single failure cannot cause a loss of both emergency generators.

  For River Bend Station, Division I SWP Pump A is supplied power from the Division I EDG, while Division I SWP Pump C is supplied power from the Division III (HPCS) EDG. Division II SWP Pumps B and D are supplied power from the Division II EDG. Assuming the removal of a single SWP pump for maintenance and a single failure which results in a loss of a separate Division of SWP (3 of 4 SWPs pumps not available), the remaining SWP would not be capable of supplying adequate cooling to support all of the monitored systems.

Due to the unique design considerations for the River Bend Station, the inspectors determined that the licensee may need to report the unavailability of one SWP pump as unavailable hours for monitored systems supplied by the SWP system.

- Section 2.2 of NEI 99-02, Revision 0, Page 30, Line 35 and Page 31, Line 9, indicated that in order to credit an installed spare and not incur unavailability hours, the system must be capable of meeting the design bases requirements with one train in maintenance and a single failure of another train. Once again, due to the unique design considerations at the River Bend Station, the SWP system could not withstand a design bases accident with one SWP pump in maintenance and a single failure affecting the opposite train. Therefore, the inspectors determined that the licensee may need to report the unavailability of one SWP pump as unavailable hours for monitored systems supplied by the SWP system.

- The inspectors determined that the licensee only revised 6 quarters of data as a result of not including support system unavailability in the WANO data. Because the original submittal was made using WANO data, the inspectors believed that the known reporting error should be corrected by completing a full 12 quarter review. The inspectors determined that the information was readily available in that the limiting condition for operation tracking log and tagging log were kept in a computer database. The databases reflected periods in which monitored and supporting systems were removed from service. Therefore access to the corresponding hand written control room log entries appeared manageable.

The inspectors considered the adequacy of the licensee’s data reporting methods for safety system unavailability an unresolved item pending a review by NRC personnel on the applicability of unavailable hours of monitored systems due to one of four 50 percent SWP pumps being removed from service (URI 50-458/0011-06).

The adequacy of the resubmitted data which only utilized revised information from January 1, 1999, in lieu of a full 12 quarters, was considered an unresolved item pending a review by NRC personnel (URI 50-458/0011-07).

4OA6 Exit Meeting Summary

The health physicist inspectors presented the ALARA inspection results to Mr. D. Mims, General Plant Manager, and other members of licensee management at an exit meeting on July 21, 2000. The licensee acknowledged the findings presented.

The inspectors presented the resident inspection results to D. Mims, General Plant Manager, and other members of licensee management at the conclusion of the inspection on August 3, 2000. The licensee acknowledged the findings presented.

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. Azzarello, Manager, Training and Emergency Planning
R. Biggs, Coordinator, Licensing
W. Brian, Director, Engineering
E. Bush, Superintendent Operations
R. Edington, Vice President-Operations
J. Fowler, Manager, Quality Assurance
D. Heath, Supervisor, Health Physics Shift
T. Hildebrandt, Manager, Maintenance
H. Holmes, ALARA Coordinator, Radiation Protection
J. Holmes, Manager, Technical Support
R. King, Director, Nuclear Safety Assurance
J. McGhee, Manager, Operations
D. Myers, Senior Licensing Specialist, Nuclear Safety Assurance
C. Miller, Superintendent, Composite Team
D. Mims, General Manager, Plant Operations
P. Page, Supervisor, Radiation Protection
A. Shahkarami, Manager System Engineering
D. Wells, Superintendent, Radiation Protection
M. Wyatt, Manager, Planning and Scheduling/Outage

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-458/0011-03	VIO	Failure to complete monthly inspections of portable fire extinguishers (Section 1R05)
50-458/0011-04	URI	Review of functional failure criteria for inoperable but available structures, systems, and components (Section 1R12)
50-458/0011-05	URI	Review of the inclusion of alternate decay heat removal system in performance indicator data (Section 4OA1)
50-458/0011-06	URI	Review of support system unavailability hours for one of four standby service water pumps out of service (Section 4OA1)
50-458/0011-07	URI	Review of process to revise performance indicator data based on error in historical data (Section 4OA1)

Opened and Closed

50-458/0011-01	NCV	Failure to implement corrective actions for locked valves (Section 1R04.1)
50-458/0011-02	NCV	Failure to isolate drywell purge isolation valve penetrations (Section 1R04.2)

Closed

50-458/0009	LER	Noncompliance with Technical Specifications for drywell purge isolation valve penetrations (Section 1R04.2)
50-458/0008	LER	Unplanned isolation of reactor water cleanup system (Section 1R04.3)
50-458/9703	LER	High pressure core spray breaker trip during testing (Section 1R22)

LIST OF ACRONYMS AND INITIALISMS USED

CFR	Code of Federal Regulations
CR	condition report
EDG	emergency diesel generator
HPCS	high pressure core spray
LER	licensee event report
MAI	maintenance action item
NCV	noncited violation
NEI	National Energy Institute
NFPA	National Fire Protection Association
RCIC	reactor core isolation cooling
RHR	residual heat removal
SWP	standby service water
URI	unresolved item
USAR	Updated Safety Analysis Report
WANO	World Association of Nuclear Operators

LIST OF DOCUMENTS REVIEWED

Assessment Reports

Quality Assurance Surveillance Report 200003006

Radiation Protection Department Self-Assessment, “ALARA Planning and Controls,” dated June 26-29, 2000

1999 Annual ALARA Report

Radiation protection sections of the refueling nine outage report

Calculations

G13.18.2.2*034 Reactor Core Isolation Cooling Required Speed with Pump Degraded 10 Percent

Condition Reports

Condition Report 2000-0919, "Loss of Air Sample Volume due to Inclement Weather"

Condition reports initiated between June 25, 2000 and August 5, 2000

Radiation protection related condition reports: 2000-0778, 2000-1035, 2000-1123, 2000-1165, 2000-1232, 2000-1334, 2000-1335, 2000-1337, 2000-1338, 20001347, and 2000-1374

Maintenance Rule

Maintenance rule database for standby gas treatment system

Maintenance rule database for residual heat removal system

Procedure PEP-0219, "Reliability Monitoring Program"

Procedures

ADM-0039, “ALARA Program,” Revision 07

ADM-0046, “Temporary Shielding Control,” Revision 04

RBNP-024, “Radiation Protection Plan,” Revision 09

RPP-0005, “Posting of Radiologically Controlled Areas,” Revision 22

RSP-0200, “Radiation Work Permits,” Revision 20

RSP-0222, “Hot Spot Tracking,” Revision 00

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, and 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.