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

March 8, 2000

EA 00-043

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

SUBJECT: NRC INSPECTION REPORT NO. 50-458/99-15 AND NOTICE OF VIOLATION

Dear Mr. Edington:

This refers to the inspection conducted on December 26, 1999, through February 12, 2000, at the River Bend Station facility. The enclosed report presents the results of this inspection.

Based on the results of this 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 of concern because the failure of operations personnel to observe suspect indications was documented in three consecutive NRC inspection reports between August 22 and December 25, 1999. This was compounded by the fact that your corrective actions were inadequate to prevent recurrence. Consequently, between December 29, 1999, and February 7, 2000, eight additional examples were identified where operations personnel did not recognize suspect indications.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. 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 determined that one additional Severity Level IV violation of NRC requirements occurred. The violation is being treated as a noncited violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or severity level of the NCV, 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, its enclosures, and your response, if requested, will be placed in the NRC Public Document Room (PDR).

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.
   50-458/99-15

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ENCLOSURE 1

NOTICE OF VIOLATION

Entergy Operations, Inc.	Docket No.: 50-458
River Bend Station	License No.: NPF-47
	EA 00-043

During an NRC inspection conducted on December 26, 1999, through February 12, 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:

A. Technical Specification 5.4.1.a required, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2. Section 2 of Appendix A of Regulatory Guide 1.33 required the licensee to have general plant operating procedures for power operation and process monitoring. Section 4.2 of Procedure ADM-0022, "Conduct of Operations," specified that the operations shift superintendent, the control room supervisor, and plant operators on shift must be aware of, and responsible for, the plant status at all times.

   Contrary to the above, between December 29, 1999, and February 7, 2000, the operations shift superintendent, control room supervisor, and plant operators were not aware of the plant status at all times. Specifically, operations personnel were unaware of eight suspect plant indications involving a rod position bypass status light, the liquid radioactive waste system, the spent fuel pool cooling system, and plant main stack exhaust flow.

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, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, 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 placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR 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 bases 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.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 8th day of March 2000

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-458
License No.:	NPF-47
Report No.:	50-458/99-15
Licensee:	Entergy Operations, Inc.
Facility:	River Bend Station
Location:	5485 U.S. Highway 61 St. Francisville, Louisiana
Dates:	December 26, 1999, through February 12, 2000
Inspectors:	T. W. Pruett, Senior Resident Inspector N. P. Garrett, Resident Inspector
Approved By:	William D. Johnson, Chief, Project Branch B Division of Reactor Projects

Attachment:	Supplemental Information

EXECUTIVE SUMMARY

River Bend Station
NRC Inspection Report No. 50-458/99-15

This routine announced inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7-week period of resident inspection.

Operations

- The inspectors identified eight examples where operations personnel were not aware of the plant status as required by Procedure ADM-0022, "Conduct of Operations." Specifically, operations personnel were not aware of suspect indications involving main plant stack exhaust flow, a rod position bypass status light, the spent fuel pool cooling system, and the liquid radioactive waste system. Similar occurrences of poor awareness of suspect indications by operations personnel were previously documented (including Noncited Violation 50-458/9912-01) in three NRC inspection reports between August 22 and December 25, 1999. Corrective actions were inadequate to prevent recurrence. The failure of operations personnel to be aware of suspect plant indications was a repeat Severity Level IV violation of Technical Specification 5.4.1.a (EA 00-043). This item was entered in the licensee's corrective action program as Condition Report 1999-1448 (Section O1.2).

- The inspectors identified three examples where operations personnel had not identified equipment deficiencies associated with a residual heat removal unit cooler, a main steam tunnel unit cooler, and a service water supply valve to the high pressure core spray unit cooler (Section O1.3).

- The inspectors determined that operations personnel responded appropriately to a loss of spent fuel pool level event on November 24, 1999, and a high conductivity in the reactor plant event on January 19, 2000 (Sections O1.4 and O1.5).

Engineering

- The inspectors identified one example of a failure to translate design requirements into drawings involving heat trace for the hydrogen analyzer sample line. Specifically, Modification Request MR 87-0262 removed the heat trace for portions of the Train B hydrogen analyzer sample line; however, piping and instrument diagram PID-33-2A, "Containment Atmosphere and Leakage Monitoring," was not revised to delete the affected heat trace circuit. The circumstances addressed in Licensee Event Report 50-458/9915 are addressed in the licensee's corrective action program as Condition Report 1999-1967. This issue was treated as an additional example of a violation of Criterion III of Appendix B to 10 CFR Part 50, which was described in NRC Inspection Report 50-458/9913. This item was entered into the licensee's corrective action program as Engineering Request 2000-0095 (Sections O1.3 and E8.1).

- The inspectors determined that the licensee resolved age degradation and over-pressurization protection issues for the containment upper fuel pool gate seals. However, the licensee did not implement over-pressure protection for the non-safety related spent fuel pool gate seals in the fuel building. Consequently, on November 24, 1999, the fuel building fuel transfer pool gate seal failed due to either age degradation or over-pressurization of the seal (Section O1.5).

Plant Support

- The inspectors determined that radiation protection personnel did not conduct surveys as required by 10 CFR 20.1501. Specifically, on December 15, 1999, the licensee did not complete an evaluation of the radiological conditions associated with infrequently accessed fire penetration seals located greater than 6 feet above floor level. This Severity Level IV violation of 10 CFR 20.1501 is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This item was entered in the licensee's corrective action program as Condition Report 2000-0016 (Section R4.1).

Report Details

Summary of Plant Status

At the beginning of the period the facility operated at 100 percent reactor power. On December 30, 1999, reactor power was reduced to 90 percent in preparation for the Year 2000 (Y2K) transition. The plant was returned to 100 percent power on January 1, 2000. On January 15, 2000, reactor power was reduced to approximately 65 percent to perform a rod sequence exchange. On January 16, 2000, reactor power was increased to 96 percent and held to resolve a problem with the heater drain system. On January 19, 2000, reactor power was reduced to 30 percent in response to a main condenser tube rupture. The plant was returned to 100 percent reactor power on January 22, 2000. On January 24, 2000, reactor power was reduced to 75 percent to perform a final rod pattern adjustment and then returned to 100 percent power where it remained for the rest of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Y2K Transition

a. Inspection Scope (71707)

The inspectors observed activities associated with the Y2K transition.

b. Observations and Findings

On December 31, 1999, through January 1, 2000, the inspectors observed activities for the Y2K transition. The licensee augmented the on-shift operating crew to support potential Y2K related problems. On-shift personnel closely monitored the plant for potential problems related to the Y2K transition prior to and during the central time zone transition. Following the Y2K transition, two minor computer issues which involved the 3-D Monicore system and the leak detection computer were noted. Both of the computer issues were determined to be insignificant.

c. Conclusions

The licensee's Y2K preparations were effective in preventing significant Y2K related problems.

O1.2 Operator Awareness of Plant Indications

a. Inspection Scope (71707)

The inspectors performed frequent walkdowns of main control room and auxiliary control room panels and questioned operations personnel on suspect instrument indications.

b. Observations and Findings

Technical Specification 5.4.1.a required, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2. Section 2 of Appendix A of Regulatory Guide 1.33 required the licensee to have general plant operating procedures for power operation and process monitoring. Section 4.2 of Procedure ADM-0022, "Conduct of Operations," specified that the operations shift superintendent, the control room supervisor, and plant operators on shift must be aware of, and responsible for, the plant status at all times. Nevertheless, between December 29, 1999, and February 7, 2000, the inspectors identified eight examples where plant operators were not aware of suspect indications.

On December 29, 1999, the inspectors identified that the rod position bypass light on main control room Panel P-680 was flashing intermittently. The reactor operator was not able to provide an explanation as to why the light was flashing. The control room supervisor stated that the condition may be due to a scan function from the rod control system. The control room supervisor subsequently contacted the system engineer who determined that the rod position bypass light should not be flashing intermittently. Following the discussion with the system engineer, the control room supervisor initiated Maintenance Action Item (MAI) 331255.

On January 27, 2000, the inspectors identified the following five meters in the auxiliary control room displaying suspect indications:

- Liquid radiological waste backwash pump Strainer 2A differential pressure Indicator LWS-PIZ620A was pegged low and the on-shift operators were not aware of the condition and did not recognize that a pegged low instrument was a symptom of a suspect indication.

- Liquid radiological waste backwash pump Strainer 2B differential pressure Indicator LWS-PIZ620B was pegged low and the on-shift operators were not aware of the condition and did not recognize that a pegged low instrument was a symptom of a suspect indication.

- Liquid radiological waste backwash pump analyzer Indicator LWS-AI324 was pegged low and the on-shift operators were not aware of the condition and did not recognize that a pegged low instrument was a symptom of a suspect indication.

- Spent fuel cooling demineralized water to containment pools flow Indicator SFC-FI124 was pegged low and the on-shift operators were not aware of the condition and did not recognize that a pegged low instrument was a symptom of a suspect indication.

- Spent fuel cooling demineralized water to spent fuel pools flow Indicator SFC-FI126 indicated approximately 95 gallons per minute with system flow secured. The on-shift operators were aware of the condition but did not question if the indication was a symptom of a suspect indication or an improper system lineup.

On January 31, 2000, the inspectors again identified the same five meters in the auxiliary control room to different on-shift operators. Following discussions with the inspectors, the on-shift operators initiated MAIs 322086, 322087, and 332084 for indications LWS-PIZ620A, LWS-PIZ620B, and LWS-AI324, respectively.

On February 3, 2000, the inspectors discussed the spent fuel cooling indication with two other on-shift operators and an operator who was present at the January 31, 2000, discussion. These operators then initiated MAIs 332186 and 332185 for indications SFC-FI124 and SFC-FI126, respectively.

On February 7, 2000, the inspectors identified the following:

- Main stack flow Indication RMS-FI126 on main control room Panel P-863 differed from main stack flow Indication RMS-FI125 by approximately 30,000 cfm. The main control room operators were unaware of the difference in the flow indicators. The unit control operator subsequently initiated MAI 332210.

- Liquid radioactive waste recovery sample tank pump discharge pressure trouble annunciator on auxiliary control room Panel PNL-187 was lit. The on-shift operators were unable to explain why the annunciator was lit since the pump discharge pressure indicated in the normal band. The auxiliary control room operator subsequently initiated MAI 332223.

NRC Inspection Report 50-458/99-12 documented the inspectors' identification of four examples of main control room operators not being aware of control room indications. Noncited Violation 50-458/9912-01 was issued and the licensee entered the inspectors' observations and findings in the corrective action program as Condition Report (CR) 1999-1448. The observations included:

- On August 25, 1999, operations personnel were not aware of temperature deviations between the eight computers points for suppression pool temperatures.

- On September 1, 1999, operations personnel were not aware of temperature deviations between the 10 computer points for drywell temperatures.

- On September 3, 1999, operations personnel did not recognize an invalid computer point displaying drywell temperature.

- On September 7, 1999, operations personnel were unaware of the pressure deviations in the differential pressure indications between drywell and containment pressure.

NRC Inspection Report 50-458/99-13 documented the inspectors' identification of two examples where operations personnel were not aware of suspect indications in the main control room. The observations included:

- On October 14, 1999, operations personnel were unaware that the amperage indication for the high pressure core spray pump was pegged high.

- On October 14, 1999, operations personnel were unaware that one computer point for suppression pool temperature was 20°F lower than the seven other computer points.

NRC Inspection Report 50-458/99-14 documented the inspectors identification of operations personnel not being able to determine the position of the heat trace panel HAND-OFF-AUTO switch even though operators were required to verify the correct position of the switch once per shift.

On November 9, 1999, the licensee completed the corrective action response for the observations made during NRC Inspection Report 50-458/99-12. CR 1999-1448 specified that the previous management expectation for operator knowledge of equipment status had not required the level of detail that is now desired. Although not listed as corrective actions in CR 1999-1448, the apparent cause disposition described the operations management desire to improve operator knowledge of plant indications by continuing the following four ongoing efforts:

- Discussing new expectations on operator knowledge of equipment status and differing control room indications given to the shift superintendents. The inspectors determined that the discussions had occurred; however, the discussions did not include efforts to improve nonlicensed operator performance.

- Operations upper management would periodically question operators on control room indications and equipment status. The inspectors determined that this activity was occurring and that operations management had identified additional examples of indications and equipment status issues in the main control room. The inspectors also determined that no efforts were implemented to question nonlicensed operators on plant conditions and equipment status.

- Challenging senior reactor operators to question personnel in depth on equipment and control room indication status. Operations senior management was not able to provide examples where the senior reactor operators were questioning operations personnel on equipment and control room indications.

- The corrective action group would make periodic observations in the main control room. The inspectors determined that the observations to be completed by the corrective action group were not formalized. Consequently, between November 9, 1999, and February 7, 2000, no observations were performed in the main control room by the corrective action group. The inspectors did identify that three management observations were completed by previously licensed managers from outside the operations department.

On December 15, 1999, operations personnel initiated a corrective action item to perform a self-assessment or effectiveness review of the ongoing efforts described in CR 1999-1448 before June 12, 2000.

The inspectors determined that the corrective actions had not been successful in improving operator awareness of plant indications. Additionally, no efforts or corrective actions were initiated to improve the ability of nonlicensed operators to recognize and question suspect plant indications. Consequently, the inspectors determined that the repetitive failure of shift supervisors, control room supervisors, and plant operators to be aware of indications affecting plant operation as required by Procedure ADM-0022 was a Severity Level IV violation of Technical Specification 5.4.1.a (VIO 50-458/9915-01). This item was entered in the licensee's corrective action program as CR 1999-1448.

On February 15, 2000, operations management informed the inspectors that the apparent cause for the failure of operations personnel to be aware of plant indications had not been identified and that additional corrective actions were being added to expand the scope of CR 1999-1448. The additional planned corrective actions included, in part, conducting a root cause analysis, briefing all operations personnel on plant indications and expectations, development of a nonlicensed operator subcommittee to address watch-standing issues, re-enforcing the expectation that main control room operators will perform panel walkdowns once every 3 hours, scheduling operations management to perform in-plant monitoring, and developing observation skills training.

c. Conclusions

The inspectors identified eight examples where operations personnel were not aware of the plant status as required by Procedure ADM-0022, “Conduct of Operations.” Specifically, operations personnel were not aware of suspect indications involving main plant stack exhaust flow, a rod position bypass status light, the spent fuel pool cooling system, and the liquid radioactive waste system. Similar occurrences of poor awareness of suspect indications by operations personnel were previously documented (including Noncited Violation 50-458/9912-01) in three NRC inspection reports between August 22 and December 25, 1999. Corrective actions were inadequate to prevent recurrence. The failure of operations personnel to be aware of suspect plant indications was a repeat Severity Level IV violation of Technical Specification 5.4.1.a (EA 00-043). This item was entered in the licensee’s corrective action program as Condition Report 1999-1448 (Section O1.2).

O1.3 Operator Awareness of Plant Equipment

a. Inspection Scope (71707)

The inspectors questioned operations personnel on equipment issues following a tour of the auxiliary building on February 6, 2000.

b. Observations and Findings

On February 6, 2000, the inspectors toured the auxiliary building and identified the following three equipment issues:

- Residual heat removal Unit Cooler 9 had three handles on a access panel which were not secured. Operations personnel subsequently determined that the handles would not tighten and initiated MAI 332205.
- Service water supply to Unit Cooler 5 Valve SWP MOV-74B had a small oil leak. Electrical maintenance personnel investigated and subsequently initiated MAI 332204.
- Main steam tunnel Unit Cooler 8 had approximately 3 feet of gasket pulled out of the access panel. Operations personnel subsequently initiated MAI 332206.

The inspectors determined that the three examples of plant equipment issues represented a lack of awareness on the part of operations personnel.

Criterion III of Appendix B to 10 CFR Part 50 required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures and instructions. During the February 6, 2000, tour of the auxiliary building, the inspectors also identified that the hydrogen analyzer cabinet supply line for Train A was completely insulated and heat traced; however, the supply line to hydrogen analyzer cabinet Train B was not. Operations personnel subsequently determined that Modification Request MR 87-0262 added an accumulator tank in the Train B hydrogen analyzer sample line and that the line should not be insulated and heat traced. However, piping and instrument diagram PID-33-2A, "Containment Atmosphere and Leakage Monitoring," specified that the entire line was to be heat traced. Engineering reviewed the issue and determined that PID-33-2A should have been revised following the completion of MR 87-0262 and initiated Engineering Request 2000-0095 to change the drawing. The inspectors determined that the licensee has not completed the root cause evaluation of similar design issues previously identified in NRC Inspection Report 50-458/9913. Therefore, the failure to translate design requirements into drawings for the installation of heat trace on the hydrogen analyzer sample lines is an additional example of a violation of Criterion III of Appendix B to 10 CFR Part 50, which was described in NRC Inspection Report 50-458/9913. This additional example was entered into the licensee's corrective action program as Engineering Request 2000-0095.

c. Conclusions

The inspectors identified one example of a failure to translate design requirements into drawings involving heat trace for the hydrogen analyzer sample line. Specifically, Modification Request MR 87-0262 removed the heat trace for portions of the Train B hydrogen analyzer sample line; however, piping and instrument diagram PID-33-2A, "Containment Atmosphere and Leakage Monitoring," was not revised to delete the affected heat trace circuit. The circumstances addressed in Licensee Event Report 50-458/9915 are addressed in the licensee's corrective action program as Condition Report 1999-1967. The circumstances addressed in Licensee Event Report 50-458/9915 are addressed in the licensee's corrective action program as Condition Report 1999-1967. This issue was treated as an additional example of a violation of Criterion III of Appendix B to 10 CFR Part 50, which was described in NRC Inspection Report 50-458/9913. This item was entered into the licensee's corrective action program as Engineering Request 2000-0095.

The inspectors identified three examples where operations personnel had not identified equipment deficiencies associated with a residual heat removal unit cooler, main steam tunnel unit cooler, and the service water supply valve to the high pressure core spray unit cooler.

O 1.4 Condenser Tube Rupture

a. Inspection Scope (71707 and 93702)

The inspectors responded to the January 19, 2000, high conductivity in the reactor plant, observed the response of the main control room personnel, and assessed the licensee's review of the event.

b. Observations and Findings

On January 16, 2000, the Train A third point heater experienced a high level alarm. Operations personnel investigated the problem and found Train A third point heater drain Valve HDL-LV4A, the normal heater level control, fully open and Train A third point heater dump Valve HDL-LV24A, the high level control valve which bypasses drain flow to the main condenser, modulating to control heater level. On January 18, 2000, operations personnel determined that a flow blockage had occurred at Valve HDL-LV4A and made the decision to secure the operating heater drain pump in the Train A heater string and allow the heater level to be controlled by diverting flow to the main condenser through Valve HDL-LV24A.

On January 19, 2000, a tube rupture occurred in the main condenser, resulting in a chemistry excursion. To isolate the leak and return chemistry to normal operational specifications, operations personnel decided to shut down the reactor. The reactor shutdown was terminated at 30 percent power when reactor plant conductivity started to return to specification. Power ascension was resumed following plugging of the ruptured main condenser tubes and returning chemistry to specification.

The follow events occurred on January 19, 2000:

2:53 p.m. Received condenser conductivity high alarm in main control room.

3:10 p.m. Chemistry personnel confirmed a condenser tube leak had occurred.

3:41 p.m. Operations personnel commenced a power reduction to 75 percent reactor power.

3:46 p.m. Chemistry personnel estimated condenser tube in-leakage to be 95 gallons per minute.

4:21 p.m. Operations personnel removed condenser Waterbox A from service.

9:12 p.m. Operations personnel commenced a power reduction to 60 percent reactor power.

10:20 p.m. The Technical Requirements Manual Limiting Condition for Operation 3.4.13, "Chemistry," was entered. Conductivity exceeded the specification of 1.0 μmho/cm. The Limiting Condition for Operation allowed 72 hours to reduce conductivity or be in Mode 2 within 6 hours.

11:15 p.m. Operations personnel commenced a reactor shutdown due to high conductivity.

On January 20, 2000, the following events occurred:

12:52 a.m. Reached maximum conductivity for the event (1.24 μmho/cm).

2:20 a.m. Operations personnel observed conductivity trending down and decided to maintain 28 percent reactor power.

2:39 a.m. Conductivity was less than 1.0 μmho/cm.

Two leaking tubes were located in the region where flow from Valve HDL LV24A entered the main condenser. The licensee plugged an additional 21 tubes in the estimated heater drain impingement path. When condenser Waterbox A was reflooded, additional leaks were indicated. The licensee plugged an additional 122 tubes in the main condenser for a total of 145 tubes plugged.

On January 20, 2000, the licensee disassembled and repaired Valve HDL-LV4A. The licensee determined that the flow blockage was due to a separation of the valve stem and valve disk. The licensee restored normal heater level control before raising reactor power on January 22, 2000.

c. <u>Conclusions</u>

The inspectors determined that operations personnel responded appropriately to the high conductivity in the reactor plant event.

O1.5 Loss of Level in the Fuel Building Spent Fuel Pool

a. Inspection Scope (71707 and 93702)

The inspectors responded to the November 24, 1999, loss of level in the fuel building spent fuel pool, observed the response of the main control room personnel, and assessed the licensee's review of the event.

b. Observation and Findings

Event Description

On November 18, 1999, the fuel building fuel transfer pool was isolated from the fuel building spent fuel pool by shutting the fuel building fuel transfer pool gate and inflating the gate seal. The level in the fuel building fuel transfer pool was reduced approximately 90 percent to allow maintenance on the inclined fuel transfer equipment.

On November 24, 1999, at approximately 3:20 p.m., the seal on the fuel building fuel transfer pool gate ruptured allowing the fuel building spent fuel pool and fuel building cask transfer pool water to refill the drained fuel building fuel transfer pool. The resultant level decrease in the fuel building spent fuel pool was approximately four feet. During the event, operations personnel secured the fuel building spent fuel pool cooling system which resulted in an increase in fuel building spent fuel pool temperature from 106 °F to 110 °F. Radiation levels at the boundaries of the fuel building spent fuel pool increased from 7 mR/hr to 40 mR/hr. Operations personnel recovered level in the fuel building spent fuel pool by transferring water from the condensate storage tank to the fuel building spent fuel pool.

Main Control Room Operator Response to Decreased Level in Fuel Building Spent Fuel Pool

The inspectors responded to the main control room and observed the response of operations personnel to the decreasing level in the fuel building spent fuel pool and the subsequent restoration of fuel building spent fuel pool level. Operations personnel appropriately diagnosed and recovered from the loss of level in the fuel building spent fuel pool.

Failure of the Fuel Building Fuel Transfer Pool Seal

On January 15, 1996, CR 1996-0143 was written to answer questions asked during the NRC Fuel Integrity and Reactor Subcriticality inspection. This inspection resulted in Notice of Violation 50-458/9601-05, for the failure to assure the design basis of the upper fuel pool gate seals in containment, the spent fuel pool gate seals in the fuel building, and the reactor cavity pneumatic seals in containment were translated into drawings, procedures, and instructions. In response to the notice of violation, the licensee committed to establish a seal service life, evaluate a modification to install a permanent backup nitrogen supply to the containment gate seals, and reclassify the gate seals as non-safety related components. The NRC reviewed the reclassification of the gate seals and documented the licensee's actions as acceptable in NRC Inspection Report 50-458/97-16.

On September 9, 1997, the licensee installed a modification that provided a permanent air manifold and backup nitrogen system with over-pressure protection for the upper pool gate seals in containment but did not install the system for the fuel transfer pool gate seal and cask transfer pool gate seal in the fuel building.

On August 3, 1999, the licensee issued Engineering Request 99-0588 which extended the service life of all gate seals to 17 years. The licensee also determined that replacement could be performed during Refueling Outage 10 which is scheduled to begin September 2001.

Following the November 1999 event, the licensee established a significant event response team to investigate the failure of the fuel building fuel transfer pool gate seal. Two potential failure modes were identified which involved, seal age and seal over-pressurization. The inspectors determined that the licensee resolved the containment upper fuel pool gate seals over-pressure protection issue described in Notice of Violation 50-458/9601-05. However, the licensee did not implement the same modification for the spent fuel gates in the fuel building.

Following the event, the licensee replaced the damaged fuel transfer pool gate seal and installed a temporary modification to provide over-pressure protection. The inspectors consider the licensee's corrective actions to be appropriate.

c. <u>Conclusions</u>

The inspectors determined that the licensee resolved age degradation and over-pressurization protection issues for the containment upper fuel pool gate seals. However, the licensee did not implement over-pressure protection for the non-safety related spent fuel pool gate seals in the fuel building. Consequently, on November 24, 1999, the fuel building fuel transfer pool gate seal failed due to either age degradation or over-pressurization of the seal.

The inspectors determined that operations personnel responded appropriately to the fuel building fuel transfer pool seal failure and subsequent recovery of fuel building spent fuel pool water level.

II. Maintenance

M1 Conduct of Maintenance

a. <u>Inspection Scope (61726 and 62707)</u>

The inspectors observed all or portions of the following maintenance and surveillance activities:

- MAI 329746 — E22 FTN056, Install damping card on E22 FTN056
- STP-203-4203 — Emergency Core Cooling System/Division 3-High Pressure Core Spray System Flow Rate-Low, Channel Calibration Test (E22-N656, E22-N056)
- MAI 330451 — EGF-LT15A, Calibrate Fuel Tank Level Gauge
- MAI 331248 — EGO-E1A, Repair oil leak on Division I Emergency Diesel Generator Lube Oil Cooler
- MAI 331233 — E22-S004-ACB3, Clean, Functional Test, Inspect, and Insulation Test Breaker
- STP-000-3602 — Fire Barrier Visual Inspection
- STP-000-3604 — Fire Barrier Sealed Penetration Inspection

b. Observations and Findings

The inspectors determined that the performance of maintenance and surveillance activities were generally thorough and professional.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Violation 50-458/EA 99158-01013: Technical Specification 3.8.1.b required three diesel generators be operable in Modes 1, 2, and 3. This issue involved the mechanical failure of the Division I emergency diesel generator fuel oil pump coupling during a surveillance test. As a result, the emergency diesel generator was inoperable for greater than the allowed Technical Specification Limiting Condition for Operation. The mechanical failure resulted from the improper installation of the fuel oil pump during engine maintenance. The issue was initially addressed in NRC Inspection Report 50-458/99-03 and NRC Inspection Report 50-458/99-07.

The licensee attributed the root cause of the Division I emergency diesel generator failure to be inadequate workmanship during the engine reassembly and inadequate work package preparation. The corrective actions implemented by the licensee included training in the installation of taper pins, an evaluation of past work practices for the Divisions I, II, and III emergency diesel generators to ensure compliance with the vendor recommended maintenance practices, and a review of a sampling of other systems to identify potential inadequacies in implementing vendor recommendations. The inspectors consider the licensee's corrective actions appropriate.

III. Engineering

E1 Conduct of Engineering

E1.1 Receipt of New Fuel

a. Inspection Scope (71707 and 37551)

The inspectors observed receipt inspection of new fuel assemblies.

b. Observations and Findings

The inspectors observed the receipt inspection of new fuel assemblies in preparation for Refueling Outage 9. The inspectors determined that all inspections were performed in accordance with Procedure REP-0005, "New Fuel Receipt." No deficiencies were identified.

E8 Miscellaneous Engineering Issues (92700)

E8.1 (Closed) Licensee Event Report 50-458/9915: Residual heat removal Train C and reactor core isolation cooling operablity affected by an unsealed wall penetration between the two rooms.

Criterion III of 10 CFR Part 50 requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. On December 9, 1999, the licensee determined that unsealed Penetration 73K between the residual heat removal room Train C and the reactor core isolation cooling room required sealing based on the effects of medium energy line crack and high energy line break flood considerations. Both systems were declared inoperable until the appropriate seal was installed.

The licensee determined that the drawing for Penetration 73K had been inappropriately changed during a 1985 drawing review. Specifically, the seal for Penetration 73K was initially determined to be a radiological seal. The drawing review determined that a radiological seal was not required and the seal classification was inappropriately determined to be none required. The licensee performed a review of penetration tabulation drawings from this period and found no similar problems.

The licensee identified a failure to translate design requirements into drawings and specifications involving a penetration seal between residual heat removal Room C and the reactor core isolation cooling room. The circumstances addressed in Licensee Event Report 50-458/9915 are addressed in the licensee's corrective action program as Condition Report 1999-1967. The inspectors determined that the licensee has not completed the root cause evaluation of similar design issues previously identified in NRC Inspection Report 50-458/9913. Therefore, this issue was treated as an additional example of a violation of Criterion III of Appendix B to 10 CFR Part 50, which was described in NRC Inspection Report 50-458/9913.

IV. Plant Support

R4 Staff Knowledge and Performance in Radiation Protection and Chemistry

R4.1 Failure to Perform Radiological Surveys

a. Inspection Scope (71750)

The inspectors observed a radiation protection (RP) technician conduct surveys during the inspection of fire penetrations which were located greater than 6 feet above floor level.

b. Observations and Findings

On December 15, 1999, during the performance of fire penetration inspections, the inspectors observed an RP technician perform very detailed and methodical radiation dose rate surveys. The RP technician completed the dose rate surveys before personnel ascended to elevations greater than 6 feet above floor level. In addition, the RP technician performed frequent dose rate surveys while the activity was in progress.

Following the activity the inspectors realized that the RP technician had not completed contamination surveys of work locations which were greater than 6 feet above floor level. The inspectors questioned the RP supervisor to determine the requirements for completing surveys for work that is to be performed greater than 6 feet above floor level. The inspectors were informed that an evaluation of the radiological conditions or actual contamination surveys were required to be performed by the RP technician. An adequate evaluation of the radiological condition would have included, at a minimum, a review of historical survey data of the work locations. The inspectors noted that the areas inspected by the quality control technicians were accessed approximately once every 10 years and that limited historical survey data would have been available.

10 CFR 20.1501 requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. In 10 CFR 20.1003, the definition of *survey* includes an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation.

The inspectors determined that the specific locations of the fire penetrations were not known by the RP technician before the commencement of the seal inspections and that an evaluation of historical survey data was not conducted. Additionally, the inspectors determined that no contamination surveys were completed by the RP technician for the areas located greater than 6 feet above floor level. Consequently, the inspectors determined that the licensee did not make surveys, as required by 10 CFR 20.1501, to assure compliance with 10 CFR 20.1201, which limits radiation exposure to less than 5 rem total effective dose equivalent. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (NCV 50-458/9915-02). This item was entered in the licensee's corrective action program as CR 2000-0016.

c. Conclusions

The inspectors determined that radiation protection personnel did not conduct surveys as required by 10 CFR 20.1501. Specifically, on December 15, 1999, the licensee did not complete an evaluation of the radiological conditions associated with infrequently accessed fire penetration seals located greater than 6 feet above floor level. This Severity Level IV violation of 10 CFR 20.1501 is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This item was entered in the licensee's corrective action program as CR 2000-0016.

V. Management Meetings

X1 Exit Meeting Summary

The exit meeting was conducted on February 15, 2000. The licensee did not express a position on any findings in the report. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Bakarich, Manager, Emergency Preparedness
R. Edington, Vice President-Operations
T. Hildebrandt, Manager, Maintenance
J. Holmes, Manager, Radiation Protection and Chemistry
R. King, Director, Nuclear Safety and Regulatory Affairs
D. Mims, General Manager, Plant Operations
J. McGhee, Manager, Operations
D. Pace, Director, Engineering

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902	Follow-up Maintenance
IP 93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-458/9915-01	VIO	Failure of operating personnel to be aware of plant indications (Section O1.2)

Opened and Closed

50-458/9915-02	NCV	Inadequate radiological survey (Section R4.1)

Closed

50-458/EA99158-01013	VIO	Technical Specification 3.8.1.b required three diesel generators be operable in Modes 1, 2, and 3 (Section M8.1)

50-458/9915	LER	Residual heat removal Train C and reactor core isolation cooling operability affected by an unsealed wall penetration between the two rooms (Section E8.1)

Discussed

50-458/9913-04	NCV	Two additional examples of the failure to translate design requirements into specifications and drawings involving hydrogen analyzer heat trace and a room penetration seal (Section O1.3 and Section E8.1)

LIST OF ACRONYMS USED

CR	Condition Report
MAI	Maintenance Acton Item
NCV	Noncited Violation
NRC	U.S. Nuclear Regulatory Commission
PDR	Public Document Room
RP	Radiation Protection
VIO	Violation
Y2K	Year 2000