ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-458/96-02

License: NPF-47

Licensee: Entergy Operations, Inc.
P.O. Box 220
St. Francisville, Louisiana

Facility Name: River Bend Station

Inspection At: St. Francisville, Louisiana

Inspection Conducted: January 14 through February 24, 1996

Inspectors: W. F. Smith, Senior Resident Inspector
D. L. Proulx, Resident Inspector

Approved: P. H. Harrell, Acting Chief, Project Branch D — Date 3-13-96

Inspection Summary

Areas Inspected: Routine, announced inspection of licensee actions in response to events, plant operations, maintenance and surveillance observations, onsite engineering, plant support activities, followup of previously identified items, and review of licensee event reports (LER).

Results:

Plant Operations

- During containment isolation system surveillance testing, the operator did not question why the instrument and control (I&C) technician directed, in error, 16 containment isolation system key-lock switches to be placed in the test position, when they were required to be verified in the normal position. An unexpected isolation occurred that might have been prevented had the operator been more familiar with the prerequisites and precautions of a procedure. The inspectors identified this as the first example of operator unfamiliarity with procedure requirements (Section 2.1).

- The operators responded appropriately to the unexpected containment isolations that occurred during Refueling Outage 6 (RF06). The correct

abnormal operating procedures were used to recover the isolated systems (Sections 2.1 and 2.2).

- Operators performed well in starting up the plant in accordance with procedures and exhibited good command and control practices (Section 3.1).

- The licensee exercised good management controls and teamwork in the execution of RF06. Essentially all planned modifications and work was accomplished in a 40-day period (Section 3.1).

- The inspectors identified three instances of operators failing to perform independent verification of safety system lineups following system manipulation. The inspectors also found three instances of senior reactor operators failing to independently verify proper entry and exit into limiting conditions for operation (LCO) action statements. These multiple failures indicated that operator attention to detail, supervisory oversight, and turnovers required strengthening. Failure to perform independent verification of system lineups as required by procedures was a violation of Technical Specification (TS) 5.4.1.a (Section 3.2).

- The licensee failed to shut and deactivate 17 containment isolation valves to meet the provisions of TS 3.6.2.10, "Primary Containment - Shutdown," as described in the bases. This indicated that the licensee did not thoroughly review and implement the bases section for this TS. Failure to close and deactivate automatic containment isolation valves was a violation of TS 3.6.2.10 (Section 3.3).

- The licensee failed to properly implement Operating License NPF-47, Condition 2.C(17) by not tagging hoses and cabling, which passed through an open containment air lock during core alterations, with instructions for expeditious removal. Although the licensee identified this violation, it is being cited because immediate corrective action did not include questioning and correcting other related discrepancies. The licensee did not verify that all provisions of the implementing procedure were in place. Consequently, the inspectors identified that signs were not posted at the air lock that described the requirements for tagging and removal (Section 3.4).

Maintenance

- An I&C technician demonstrated poor self-checking techniques and a lack of attention to detail by failing to follow prerequisite steps in a containment isolation system surveillance procedure. As a result, an unexpected balance-of-plant isolation occurred. A noncited violation was identified for failure to follow procedure (Section 2.1).

- The licensee performed well in the diagnostic testing and repairs of two motor-operated valves (MOV) (Section 4).

- The licensee did not perform a thorough evaluation of test data associated with a local leak rate test on an MOV. Personnel incorrectly determined that postmaintenance local leak rate test results were as-found testing data. The licensee retracted a 10 CFR 50.72 notification based on this incorrect information. Consequently, the licensee was re-evaluating the reportability of this issue at the end of the inspection period. This is an inspection followup item (Section 4).

- The inspectors noted that the control rod scram time test procedure required that control rods should not be moved until satisfactory scram time tests were completed. However, an operator cycled control rods from full in to full out for venting of the control rod hydraulic system. Once the inspectors identified this conflict, the licensee changed the procedure to clarify that the intent was to prevent pulling rods for criticality and power operations prior to scram timing. The inspectors identified this as another example of an operator not being thoroughly familiar with the precautions and prerequisites of a procedure being performed (Section 5.1).

- The licensee satisfactorily completed control rod drive mechanism (CRDM) housing support surveillance checks in accordance with TS requirements prior to startup (Section 5.2).

- During the reactor pressure vessel inservice leakage test, operators were not aware of the test procedure precaution that the pressure monitor/controller <u>should</u> have no other duties. The inspectors identified that the designated pressure monitor/controller performed several plant manipulations and relieved the at-the-controls operator. There was no loss of pressure control observed by the inspectors; however, licensee management expected personnel to comply with <u>should</u> statements in procedures unless they obtain concurrence from their supervisors to deviate. This was a third example personnel performing a procedure without being fully cognizant of the applicable prerequisites and precautions (Section 5.4).

<u>Engineering</u>

- The licensee identified that they did not adhere to the provisions of the Updated Safety Analysis Report (USAR) because toxic refrigerants were stored on site in unanalyzed compositions and quantities. This unanalyzed condition initially called into question the habitability of the control room. The licensee failed to perform a 10 CFR 50.59 evaluation for this apparent change to the facility. The licensee-identified and corrected failure to perform a 10 CFR 50.59 evaluation was a noncited violation (Section 6).

Plant Support

- Overall plant housekeeping during RF06 was good. Plant management provided strong oversight as evidenced by the inspectors finding the drywell cleaner than it was at the end of the previous refueling outage. Also, because of housekeeping practices during the outage, the post outage recovery of plant cleanliness was simplified (Section 3.1).

- Because of a lack of understanding of fire seals, operators did not adequately resolve the inspectors' concern that a main steam tunnel access plug was not properly sealed. Consequently, the licensee started up the plant and operated at power for four days with a degraded fire seal until maintenance personnel identified the impairment on a condition report (CR). However, an hourly fire watch was already patrolling the area for other identified fire barrier impairments (Section 7).

Summary of Inspection Findings:

Open Items

- A noncited violation was identified that involved a cognitive error resulting in a failure to follow a surveillance procedure (Section 2.1).

- Unresolved Item 458/9602-01: Followup on breaker retesting methods (Section 2.2).

- Violation 458/9602-02: Failure to follow an administrative procedure that required independent verification of system restoration (Section 3.2).

- Violation 458/9602-03: Failure to comply with TS 3.6.1.10 for containment integrity while shut down (Section 3.3.4).

- Violation 458/9602-04: Failure to meet Operating Licensee NPF-47, Condition 2.C(17) requirements for control of hoses and cables passing through containment air locks during core alterations (Section 3.4).

- Inspection Followup Item 458/9602-05: Followup on reassessment of possible loss of function of penetration valve leakage control system (Section 4).

- A noncited violation was identified involving a failure to comply with 10 CFR 50.59 related to refrigerant storage on site (Section 6).

Closed Items

- Inspection Followup Item 458/9505-01 (Section 8).

- LER 458/96-001 (Section 9.1).
- LER 458/96-004 (Section 9.2).
- LER 458/96-005 (Section 9.3).

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

At the beginning of this inspection period, the plant was in Mode 5, Refueling, for RF06. On January 29, 1996, operators placed the plant in Mode 4, Cold Shutdown, after refueling the reactor.

On February 11, operators commenced the reactor startup, and the plant entered Mode 2, Startup. The main generator was synchronized to the power grid on February 13, thus ending the 40-day refueling outage. Full power operation was achieved on February 18; however, power was reduced to 92 percent on February 21, because of abnormally high temperatures on the inboard pump bearing of Reactor Feedwater Pump C. On February 23, power was restored to 100 percent, and the plant continued to operate at 100 percent power through the end of this inspection period.

2 ONSITE RESPONSE TO EVENTS (93702)

2.1 Containment Isolation Actuation Through Human Error

On January 15, 1996, with the plant in Mode 5 and the refueling cavity filled to at least 23 feet above the reactor vessel flange, an unexpected balance-of-plant containment isolation occurred. During the actuation, I&C technicians were performing Procedure STP-058-4209, "Containment and Drywell Manual Isolation Actuation 18 Month LSFT and Channel Functional Test," Revision 2. The systems isolated in accordance with design, and shutdown cooling was not affected. The operators responded appropriately and restored the affected systems as required. CR 96-0136 was initiated to enter the event into the corrective action program.

The investigation into the event found that the I&C technician misread Procedure Prerequisite 6.5 and, as a result, directed the licensed operator to place 16 key-lock switches in the test position when the procedure required the switches to be verified in the norm (normal) position. The licensed operator proceeded to implement the instructions received from the I&C technician without questioning the action until the isolation occurred. When the I&C technician read the procedure again, he recognized his error.

As immediate corrective action, the licensee removed the I&C technician from surveillance testing activities for the remainder of RF06, so that the individual could consider the importance of following procedures and paying attention to detail. Other I&C technicians were informed of the event causes in a shop meeting on January 15. The licensee stated that they were reviewing the event with operations personnel to emphasize the need for a questioning attitude. As part of corrective actions from previous communication and procedure compliance errors identified in the past, the licensee was in the process of changing procedures to require the operators to sign off their own

actions. The Procedure Upgrade Program personnel had responsibility to add operator sign-offs, but Procedure STP-058-4209 had not yet been changed.

The licensee indicated that this action will help, but attention to detail was still essential to preventing errors. The corrective actions taken by the licensee were appropriate to the circumstances. The licensee issued LER 458/96-004 on February 12.

Failure to follow Step 6.5 of Procedure STP-058-4201 is a violation of TS 5.4.1.a. This self-disclosing and corrected violation is being treated as a noncited violation, consistent with Section VII of the NRC Enforcement policy. Specifically, the violation was identified by the licensee, was not willful, actions taken, as a result of a previous violation, should not have corrected this problem, and appropriate corrective actions were being or had been completed by the licensee.

2.2 Engineered Safety Features Actuation Because of Electrical Protection Assembly Breaker Trip

On January 13, 1996, with the plant in Mode 5, a loss of the Division II reactor protection system bus resulted in a half-scram and several engineered safety feature actuations. Upon loss of the reactor protection system bus, shutdown cooling was not protected for a loss of the suction path and, therefore, operators manually tripped the residual heat removal pump. Shutdown cooling was interrupted for 19 minutes, but reactor coolant temperature increased less than 1°F. The operators responded to the event appropriately and restored the isolated system to resume shutdown cooling. The licensee initiated CR 96-0115 to document the event and reported this event in LER 458/96-003. The licensee was unable to determine the root cause of this event, but the electrical protection assembly breakers have had a history of spurious trips. The licensee replaced the breakers with a new design during RF06 as part of a planned modification. The inspectors followed up on the this event and noted the complications discussed below.

2.2.1 MOV Failure

One of the engineered safety feature signals received during this event was a nuclear steam supply shutoff system isolation. However, Valve 1CCP*MOV144, component cooling water return drywell isolation, did not fully shut. The inspectors followed up on the failure of Valve 1CCP*MOV144 to operate properly. The licensee found the torque switch on the actuator tripped but could not determine the root cause of the failure. Because of poor communications, maintenance personnel had disassembled the MOV and removed the old actuator prior to engineering being notified of the failure. Therefore, the as-found condition of the MOV was not determined.

The licensee noted that the set-up and testing committed in their response to Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," had not yet been performed for Valve 1CCP*MOV144. The licensee

reviewed the maintenance history and found no time where the torque switch was worked. Consequently, the licensee concluded that the actuator may have been improperly set up by the vendor originally or the torque switch may have had a non-repeatable setting. Upon replacement of the actuator, the licensee performed the set-up and baseline testing required by Generic Letter 89-10 and declared the MOV operable.

The inspectors discussed this MOV failure with the plant manager. The inspectors expressed concern that, without the as-found condition of failed equipment preserved, the root cause of failure could not normally be determined, which decreased confidence that corrective actions to prevent recurrence would be effective. The licensee stated that they would continue to emphasize the importance of preserving as-found conditions to craft personnel as part of their continuing efforts to improve human performance. The inspectors considered the corrective actions acceptable because the inspectors have observed improvements in the area of root cause determinations, particularly with the Corrective Action Review Boards that have been recently convened by the licensee.

2.2.2 Failure of Motor Control Center Breaker (1EJS-ACB66) to Reset

The inspectors reviewed a failure of Breaker 1EJS-ACB66 to reset. The breaker supplied power to Motor Control Center MCC-102B that, in turn, supplied power to the drywell unit coolers and had tripped as designed. During recovery from the event, operators could not reset Breaker 1EJS-ACB66. The licensee initiated a maintenance action item to troubleshoot the failure of the breaker to reset. The licensee found that personnel had racked in the breaker too far causing the breaker switches to over toggle and prevented the breaker from resetting. The licensee attributed the root cause to operator error while racking in the breaker. The licensee correctly racked in the breaker and tested it satisfactorily. In addition, the licensee conducted training on this event and the proper techniques of racking in breakers with operations personnel.

The inspectors reviewed Procedure ADM-0022, "Conduct of Operations," Revision 18A, and noted that the method identified to retest a breaker after racking it in included only to close the breaker and start the associated equipment. This method did not verify the trip and reset functions of the breaker. The inspectors were concerned because, during a loss of offsite power and loss of coolant accident, the safety-related loads are stripped and then sequenced on to the buses, following emergency diesel generator start. This requires breakers to trip, reset, and then close again. If a breaker is improperly racked in, as was Breaker 1EJS-ACB66, such that the breaker will close but not reset after a trip, the breaker may not reclose when loads are sequenced onto the safety bus.

The licensee considered the improper racking of Breaker 1EJS-ACB66 as isolated and, therefore, planned no additional controls for breaker retesting. After the inspectors raised the question about retesting or otherwise having confidence in the reset feature, the licensee indicated that they would

evaluate their practices for verifying breakers are properly racked in. The inspectors will evaluate the adequacy of the method used to retest breakers during subsequent inspections. This is an unresolved item (458/9602-01) pending review by the inspectors of industry information related to common practices used after racking in a breaker.

3 PLANT OPERATIONS (71707)

3.1 Routine Observations

The inspectors observed the overall controls exercised by the licensee over RF06 activities, as the outage progressed to completion on February 13, 1996. RF06 was originally scheduled for approximately 33 days and was successfully completed in 40 days. The licensee completed all of the 50 planned modifications and completed over 2000 maintenance work orders, now referred to as maintenance action items. The inspectors noted excellent teamwork throughout the plant, and plant housekeeping practices were better than the previous outage.

The operators exhibited good overall control of plant conditions with few clearance problems, which were appropriately identified on condition reports and corrected. The Work Management Center, in concert with the outage management control center and highly visible management involvement, kept much of the day-to-day pressures from the control room operators, allowing them to concentrate better on plant conditions in support of the work.

Plant startup progressed smoothly. The inspectors toured the plant just prior to startup and noted the drywell was cleaner than at the end of the previous refueling outage. Cleanup of the rest of the plant was well advanced. The operators maintained good command and control as the plant was taken to full power operation. Very few steam leaks disrupted the startup, and the unidentified leak rate for the drywell was 0.1 gpm with a TS limit of 5 gpm. During the past fuel cycle, the unidentified leak rate typically exceeded 1 gpm.

Reactor Feedwater Pump C troubles delayed achieving full power operation because of abnormally high temperatures on the inboard pump bearing. The licensee resolved the temperature problems after adjusting the inboard pump bearing clearances on several occasions.

At the end of this inspection period, the inspectors noted no discernable seat leakage from any of the safety relief valves. The inspectors independently estimated that during the past fuel cycle the combined leakage from the safety relief valves exceeded 4 gpm. This leakage had caused suppression pool temperatures and level to be constantly rising and resulted in frequent, routine operation of suppression pool cooling and required pumping excess water to radwaste, respectively.

In NRC Inspection Report 50-458/95-26, the inspectors identified several discrepancies in control rod manipulation documentation. In support of the

startup from RFO6, the format of the control rod sequencing document was much improved, and the signoffs were complete and appropriate. The inspectors concluded the licensee implemented effective actions to prevent recurrence of a noncited violation.

3.2 Lack of Independent Verification

The inspectors monitored log-keeping practices of control room personnel to determine whether the operators complied with the USAR, TS, and licensee procedures and policies. The inspectors noted several discrepancies in log-keeping that indicated improvement was necessary in this area.

Section 6.3.7 of Procedure ADM-0022 states, in part, that control board lineups shall be performed after completion of any operation or component manipulations on safety-related TS systems. Control board lineups performed in this manner shall be verified by a different operator and documented in the control room log book. Procedure ADM-0022 also requires independent review of entries and exits into LCO action statements.

On January 29, 1996, during a review of the control room operator log, the inspectors noted that on January 28 the licensee vented the containment in accordance with Procedure SOP-0059 "Containment HVAC System," but did not independently verify system restoration. The inspectors noted that the licensee had performed three shift turnovers since this missed verification and did not recognize the error. The inspectors notified the shift superintendent who wrote CR 96-0366 to document this occurrence and propose corrective actions.

For immediate corrective actions, the licensee independently verified restoration of the containment ventilation with satisfactory results. In addition, the licensee reviewed the January 1996 operator logs and noted that on January 26 operators failed to independently verify restoration of the fuel pool cooling system. The licensee added this additional discrepancy to CR 96-0366. For long-term corrective actions, the licensee required the off-going shift superintendent to thoroughly review the shift operating log, prior to turnover, to ensure that all shift actions were properly accomplished. In the past, only the oncoming shift superintendent was required to review the log.

On February 5 the inspectors identified more instances of failure to perform independent verification. On February 2 the licensee did not independently review the entry into or exit from an LCO by two senior reactor operators as required by Procedure ADM-022. On February 3 operators failed to independently verify the restoration valve lineup for the residual heat removal system following reactor vessel water rejection to the radwaste system. Also, on February 3 the licensee failed to independently verify exit from an LCO. On February 18, on two occasions, the licensee failed to independently verify entries into and exits from LCOs.

The inspectors discussed these instances with the operations manager. The inspectors noted that, because six separate occasions of failure to independently verify items required by Procedure ADM-0022 occurred, these items were not isolated events. The inspectors identified that these errors existed for several shifts until identified by the inspectors, indicating that shift turnovers and supervisory oversight required strengthening. The operations manager acknowledged the inspectors' comments.

Individually, these findings were of low safety significance. Each system was found correctly aligned when operators performed the verifications. In addition, prior to startup, the licensee performed a complete lineup of all systems as required by licensee procedures. However, the repetitive nature of these errors, among the other weaknesses described previously, indicate that effective corrective action is necessary to ensure continued good operator performance. The failure to perform the independent verifications required by Procedure ADM-0022 is a violation of TS 5.4.1.a (458/9602-02).

3.3 Improper Containment Isolation Provisions

3.3.1 Background

On January 18, 1996, during a review of primary containment integrity, the inspectors noted that the licensee had not deactivated 17 automatic primary containment isolation valves that were required to be closed during accident conditions. The inspectors were concerned because the licensee had been handling irradiated fuel and conducting core alterations in the primary containment. TS 3.6.1.10, "Primary Containment-Shutdown" required primary containment to be operable during irradiated fuel handling, core alterations, and operations with a potential to drain the reactor vessel. Without automatic valves such as Containment Purge Supply and Outlet Valves HVR*AOV123 and -128 deactivated, the licensee did not appear to meet the TS bases definition of primary containment-shutdown. Therefore, primary containment could be considered inoperable during this time period. The inspectors informed the operations manager and the licensee suspended core alterations until the issue was resolved.

The operations staff initiated a CR to evaluate this concern. The licensee stated that they had previously interpreted an automatic containment isolation valve as a valve that receives an automatic open signal. However, the inspectors noted that the technical requirements manual and the USAR provided a listing of all valves considered to be automatic isolation valves. All of the required automatic valves were closed but not deactivated. Procedure STP-000-0702, "Primary Containment Shutdown Verification," Revision 8, did not require all of the required closed valves to be deactivated.

After discussion with the inspectors, the licensee revised Procedure STP-000-0702 to require closing and deactivating all of the valves listed as automatic containment isolation valves required to be closed during

an accident. On January 19, the licensee implemented the revised Procedure STP-000-0702 and then recommenced core alterations.

Subsequently, the licensee determined that they had fully met the intent of the TS. The licensee stated that only valves that received an open signal were required to be deactivated. The licensee indicated that the automatic containment isolation valves that received a closed signal could be taken out of the "auto" position to the "close" position, thereby making the valve manually operated. In addition, the licensee stated that, although the TS bases were written to preclude a "single active failure," the licensee believed that they were consistent with the TS bases.

Further, the licensee indicated that the intent of the bases was for a "credible single active failure" and interpreted a credible single active failure to mean an open valve fails to go shut when required or a valve that gets an automatic signal to open opens. The licensee stated that the list of automatic valves in the USAR applied to operational periods and not to shutdown periods. Therefore, they believed that no automatic valves at River Bend Station met the requirement to be closed and deactivated.

The inspectors disagreed with the reassessment. The inspectors noted that ANSI 58.9, "Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems" specifies, in part, spurious operation of a powered component because of a failure originating within its automatic actuation or control systems shall be regarded as an active failure unless specific features or operating restrictions (such as racking out an MOV breaker) are incorporated to prevent such spurious operation. An example of a spurious operation is the unintended energizing of a powered valve to open or close. Therefore, the inspectors concluded that the licensee's interpretation of a single active failure was incorrect. The licensee voluntary LER 458/96-006 to describe their position on this issue.

3.3.2 Regulatory and Safety Assessment

TS 3.6.1.10 states, in part, primary containment shall be operable during movement of irradiated fuel assemblies in the primary containment, during core alterations, and during operations with a potential to drain the reactor vessel. The action statements require, with primary containment inoperable: (A.1) suspend movement of irradiated fuel assemblies in the primary containment immediately, (A.2) suspend core alterations immediately, and (A.3) initiate action to suspend operations with a potential to drain the reactor vessel. Surveillance Requirement 3.6.1.10.1 requires the licensee to "Verify each penetration flow path, required to be closed during accident conditions, is closed" every 31 days.

The bases for TS 3.6.1.10 state, in part, that the isolation devices for the penetrations in the primary containment boundary are a part of the primary containment leak tight barrier. To maintain this leak-tight barrier for accidents during shutdown conditions, all penetrations required to be closed

during accident conditions are closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, or the equivalent.

The bases for Surveillance Requirement 3.6.1.10.1 states, in part, that this surveillance requirement verifies that each primary containment penetration that could communicate gaseous fission products to the environment during accident conditions is closed. The surveillance requirement helps to ensure that postaccident leakage of radioactive gases outside of the primary containment boundary is within design limits. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, a blind flange, or equivalent.

3.3.4 Safety Significance

This issue was potentially safety significant, because the licensee handled new and irradiated fuel within primary containment from January 13 until operators suspended fuel movement on January 18 without the provisions of TS 3.6.1.10 being met. The safety analysis portion of the TS bases states, in part, that primary containment operability is maintained by providing a contained volume to limit fission product escape following a fuel handling accident or other unanticipated reactivity or water level excursion. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Since no credit is assumed for automatic isolation valve closure and since any leakage that would occur prior to valve closure is similarly not accounted for, all penetrations that could communicate gaseous fission products must remain closed.

The inspectors noted that, although several automatic containment isolation valves were not deactivated closed, the applicable valves were shut so a release path did not exist. However, the inspectors considered the plant susceptible to a single active failure. The inspectors noted that the basis for the TS defined the method in which the licensee must comply with the TS. Compliance with "Primary Containment-Shutdown" includes properly isolating each penetration. Because several penetrations were not properly isolated, primary containment was not operable as was required during irradiated fuel handling and core alterations.

The failure to maintain Primary Containment-Shutdown during irradiated fuel handling operations and core alterations, as defined in the basis for the TS, is a violation of TS 3.6.1.10 (458/9602-03).

3.4 Improper Implementation of License Condition

On January 16, 1996, the licensee identified that they were not in compliance with License Condition 2.C(17) of the River Bend Station operating license.

License Condition 2.C(17) states, in part, that primary containment air lock doors may be open during core alterations, except when moving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 11 days), provided hoses and cables running through the air lock employ a means to allow safe, quick disconnect and are tagged at both ends with specific instructions to expedite removal. The licensee used Procedure OSP-0034, "Primary Containment/Fuel Building Operability during Movement of Irradiated Fuel Assemblies," Revision 1A, to implement these requirements.

During an inspection of the Fuel Building 113-foot elevation containment air lock, a quality assurance representative found a fire hose and cabling for ultrasonic test equipment running through the air lock. Neither of these items were tagged with specific removal instructions as required. The fire hose had tags hanging on it, but these tags had no removal instructions on them. The ultrasonic test cabling was not tagged at all and was not in the obstruction log. When these items were found, the licensee was performing core alterations. The control room was notified of these problems, and core alterations were suspended until the discrepancies were resolved. The licensee initiated CR 96-0167 to document this discrepancy and to propose corrective actions. After the items were corrected, the licensee recommenced core alterations. The licensee submitted LER 458/96-005, prior to the end of the inspection period.

As follow up, the inspectors reviewed Procedure OSP-0034. On January 19 the inspectors walked down the fuel and containment buildings, including the air locks to verify that all of the provisions of Procedure OSP-0034 were met. The inspectors noted that Section 6.1 of Procedure OSP-0034 stated, in part, when an airlock door is held open for an obstruction the following requirements in Section 6.1.1 shall be met: "A sign will be placed on the respective airlock cage or door. The sign will require the following: (1) obstruction must be labeled at both ends; (2) if an obstruction is unlabeled it will be removed by Operations; and (3) control room must be notified of placement of an instruction."

The inspectors found that this sign was not in place on or near the airlock cage or door. The inspectors notified the control room and CR 96-0226 was initiated to document this additional problem. Because the inspectors found this additional failure to follow Procedure OSP-0034 three days after the initial licensee identification of this issue, the inspectors concluded that the initial corrective actions were not adequate. The licensee had failed to fully implement all of the requirements of Procedure OSP-0034.

The failure to tag and provide instructions for removal of two containment airlock obstructions is a violation of River Bend Station Operating License Condition 2.C(17), and the failure to follow Procedure OSP-0034 is a violation of TS 5.4.1.a (458/9602-04).

4 MAINTENANCE OBSERVATION (62703, 37551)

On January 26, 1996, the inspectors witnessed portions of maintenance and MOV signature testing on Valves SAS*MOV102 and -103 in accordance with Maintenance Action Item 304226. Since these valves failed their as-found local leak rate tests, the licensee performed signature testing to determine the as-found condition.

Initially, the licensee could not determine a leak rate for either of the valves during the local leak rate tests. The licensee postulated that the excessive leakage would result in an isolation of the penetration valve leakage control systems on high flow and render both penetration valve leakage control system trains inoperable. Subsequently, the licensee determined an as-found leak rate value for Valve SAS*MOV103, which was less than the flow rate necessary to trip the outboard train of the penetration valve leakage control system compressor. Therefore, the licensee retracted the 10 CFR 50.72 report.

Following the end of the inspection period, quality assurance personnel identified that the as-found leak rate value used for Valve SAS*MOV103 was actually obtained following maintenance on Valve SAS*MOV103. Also, the licensee could not locate a data sheet that reflected actual as-found testing of Valve SAS*MOV103. The licensee continued to reassess operability of the penetration valve leakage control system and reportability of the event. The inspectors will followup on this reassessment (Inspection Followup Item 458/9602-05).

Signature testing of the valves found that the actuators for both valves had thrust values well within the previously established windows. The licensee disassembled the valve and performed relapping of the discs and seats. The licensee repeated the local leak rate tests for each of these valves and the results were satisfactory. The inspectors noted that licensee personnel were highly knowledgeable of the signature testing evolution, and performed all of the maintenance satisfactorily and in accordance with procedures.

5 SURVEILLANCE OBSERVATIONS (61726)

5.1 Control Rod Scram Time Testing

On February 1 and 2, 1996, the inspectors witnessed preparations for, and performance of, control rod scram time testing in accordance with Procedure STP-052-3701, "Control Rod Scram Testing," Revision 10B. The inspectors reviewed the surveillance procedure for technical adequacy and determined that it was satisfactory.

The inspectors witnessed the test preparations. The inspectors noted that Step 5.7 of Procedure STP-052-3701 required, in part, after control rod drive maintenance on a control rod or the control rod drive system that could affect the scram time, the control rod shall not be declared operable or withdrawn from position 00 for purposes other than scram time testing, until satisfying

scram time criteria in accordance with TS Surveillance Requirement 3.1.4.3. The inspectors noted on February 1 that operators cycled control rods from position "00" to "48" to adjust the operating speed of the control rods. The operator performed this evolution prior to scram time testing and had replaced and/or repaired several of the control rods being operated. The inspectors were concerned that the licensee was not adhering to Step 5.7 of Procedure STP-052-3701.

The inspectors notified the reactor engineering supervisor and the operations manager. The licensee stated that Step 5.7 of Procedure STP-052-3701 was intended to prevent personnel from moving a control rod while the reactor was being taken critical or during normal operation, and not to preclude all rod movement. The licensee revised Step 5.7 of Procedure STP-052-3701 to state, in part, "...the control rod shall not be... withdrawn for continued operation in Modes 1 or 2 until satisfying scram time criteria..." This revision satisfied the intent of reactor engineering and clarified the procedure for performance. The inspectors considered the corrective actions appropriate.

During the performance of scram time testing on February 2, the inspectors noted good coordination among operators and reactor engineering personnel in timing the control rods. Following testing the inspectors reviewed the test data. The inspectors verified that all of the control rod scram times met the TS limits. In addition, the inspectors noted that reactor engineering trended the scram times to help identify potentially degrading equipment.

5.2 Control Rod Drive Housing Support Verification

Prior to the startup from RFO6, the inspectors determined the CRDMs that had maintenance performed on them during the outage. The licensee replaced CRDMs 32-33, 36-33, and 44-33, which required disassembly of the CRDM housing supports. CRDM housing supports have a safety function to restrict the outward movement of a control rod less than 3 inches if a CRDM housing fails during plant operation. Section 3.1.9.1 of the Technical Requirement Manual specifies the requirement to have properly assembled CRDM housing supports, which was implemented by Procedure STP-052-7701, "Control Rod Drive Housing Support Check," Revision 7.

Since another facility had found similar CRDM housing supports partially disassembled after a period of plant operation, the inspectors requested copies of completed Procedure STP-052-7701 to confirm that the licensee had reassembled the affected supports prior to plant startup from RFO6. The inspectors reviewed the documentation and noted that for the three CRDMs listed above the supports were satisfactorily inspected and the proper clearances verified on January 30, 1996. Procedure STP-052-7701 was repeated for CRDM 32-33 on February 7 to stop leakage found during the reactor pressure vessel pressure test. The licensee replaced the o-rings associated with control Rod 32-33, and the leakage was corrected. The inspectors considered the CRDM housing support to have been properly inspected to ensure it was intact at startup.

5.3 Reactor Pressure Vessel Inservice Leakage Test

On February 1, 1996, the licensee commenced inservice leakage testing of the reactor pressure vessel in accordance with Procedure PEP-0042, "Reactor Pressure Vessel Inservice Leakage Test," Revision 7, to satisfy the ASME Section XI requirements for a leak check of the reactor pressure vessel. The inspectors witnessed this test to determine if the licensee performed the evolution in accordance with NRC and licensee requirements. The inspectors determined that the reactor pressure vessel inservice leakage test was performed in a generally conservative manner.

The inspectors attended the test briefing. Licensee management was present at the briefing to ensure that their expectations were met during the test. The inspectors determined that the licensee provided an excellent briefing to ensure all personnel were aware of their duties and the test would be performed in a conservative manner.

Upon reaching test pressure, the licensee performed their inspections of the pressure boundary after a 4-hour wait. The licensee found eight CRDMs leaking. In addition, the licensee identified and corrected a small flange leak at Safety-Relief Valve B21*RVF041B. Operators performed the control rod scram time testing concurrent with the leakage inspection. Operators used this opportunity to scram each of the leaking CRDMs, which stopped the leakage by reseating the seals. This method stopped the leakage for seven of the eight leaking CRDMs. However, CRDM 32-33 continued to leak and was repaired as described in Section 5.2.

During the test, the inspectors noted one concern. Paragraph 5.1.11 of Procedure PEP-0042, stated, in part, personnel responsible for monitoring and/or controlling reactor pressure should have no other duties once pressurization commences. This was to ensure that action could be taken immediately to trip the control rod drive pump used to raise pressure, open a reactor water cleanup valve to prevent overpressurization, or depressurize in an emergency. The inspectors observed that the operator assigned to monitor and control pressure also attended to several other control room tasks at this time. The operator responded to several annunciators, performed manipulations on the standby service water system, and took the routine control room operator logs.

In addition, this operator relieved the at-the-controls operator during his pressure watch with control rod scram timing about to commence. The inspectors informed the operator, the control room supervisor, and the shift superintendent that their practices did not meet the direction in Procedure PEP-0042. These licensee personnel stated that they were unaware of the direction of paragraph 5.1.11 for the pressure monitor/controller to have no concurrent duties. The control room supervisor had the operator relieved as at-the-controls operator, and briefed him on his responsibilities as pressure monitor/controller in accordance with paragraph 5.1.11 of Procedure PEP-0042.

The inspectors informed the operations manager of this observation. The operations manager stated that the failure to follow the "should" statement in Procedure PEP-0042 did not meet his expectations. The licensee informed all of the crews involved with the inservice leakage tests of the requirements in paragraph 5.1.11 of Procedure PEP-0042. The inspectors monitored the rest of the performance of Procedure PEP-0042 and had no other concerns. The inspectors were concerned regarding the conduct of the operators during the above activity, specifically, the crew's lack of familiarity with the procedure requirements. The inspectors considered this as weak operator performance.

5.4 Remote Shutdown Panel (RSP) Operability Test

On February 5, 1996, the inspectors witnessed portions of Procedure STP-200-0602, "Division II Remote Shutdown System Control Circuit Operability Test," Revision 8B. The inspectors considered that licensee personnel performed the surveillance well and in accordance with the procedure. The inspectors reviewed the test data and determined that it was satisfactory.

However, the inspectors were concerned that the procedure was written such that the procedure could be improperly performed. The surveillance procedure requires the user to shift control of components to the RSP, attempt to operate the components from the main control room, then operate the components from the RSP. Operators should be able to operate the components from the RSP but not be able to operate them from the main control room.

The inspectors expressed concern that this process for motor-operated throttle valves may not yield the required result. For throttle valves, the surveillance procedure required the operator to "throttle open" the valve, then "attempt to close" the valve from the main control room. No direction was given by the procedure as to how far to throttle open the valve, nor how long to hold the switch in the "close" position. If a valve is throttled partially open and indicates mid-position and if operators close the valve for a short period such that the valve only indicated mid-position, operators would be unable to confirm the actual valve travel. Therefore, following the procedure as written could result in a perceived satisfactory surveillance.

When performing Procedure STP-200-0602 on February 5, operators were able to avoid the above scenario by fully opening the throttle valves until a full open indication was observed from the RSP and then holding the main control room switch to the closed position for about five seconds. The inspectors determined that this was a good practice and in accordance with the procedure. The inspectors informed licensee management of this concern. They agreed that the steps in Procedure STP-200-0602 for testing throttle valves could be potentially misleading and would take action to clarify the procedures, as appropriate, to assure meaningful results to both the Division I and II RSP.

6 ONSITE ENGINEERING (37551)

In December 1995 the licensee identified that a significant amount of freon was stored onsite without prior evaluation. The licensee initiated CR 95-1199 to evaluate the significance of this discrepancy and to implement corrective actions. Upon notification of the potential significance of CR 95-1199, the inspectors performed a followup on the CR. The inspectors were concerned that the unanalyzed amount of freon stored onsite could challenge the evaluation for control room habitability.

During preparation of Engineering Study G13.18.20.95-011, Toxic Chemical Analyses for Control Room Habitability, the licensee discovered that Refrigerants R-11 and R-114 were stored onsite in significant quantities. The warehouse contained 11,700 pounds of R-11 and 29,850 pounds of R-114. The R-11 was stored in 117 separate 100-pounds containers, and the R-114 was stored in 199 separate 150-pounds containers.

The licensee performed an operability evaluation to assess habitability of the control room envelope. Standard Review Plan Section 6.4 endorses Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release." Regulatory Position C.5.a of Regulatory Guide 1.78 states, in part, that the quantity of hazardous chemicals assumed to be released is equal to the contents of the largest container stored onsite unless the containers were interconnected. Therefore, the licensee reasoned that they were required to evaluate for a 100-pound release of R-11 and a 150-pound release of R-114. The licensee noted that they were previously evaluated for a single release of 500 pounds of R-22, which has identical toxicity to R-11 and R-114. Therefore, the storage of the R-11 and R-114 was bounded by previous calculations, and the habitability of the control room was no longer in question.

The inspectors reviewed the evaluation documented in CR-1199 and the USAR requirements. The inspectors found the evaluation satisfactory. The inspectors noted that the USAR states, in part, that the basis for the control room habitability evaluation was for the items listed in Table 2.2-5. Table 2.2-5 states, in part, that there were 2,390 pounds of R-22, 160 pounds of R-12, and 5,600 pounds of R-114. The USAR does not list R-11 as being onsite at all.

The licensee had brought the unanalyzed gases on site in December 1994. The inspectors determined that by bringing R-114 and R-11 onsite, in quantities not previously evaluated, the licensee had made a change to the facility as described in the USAR. The licensee did not perform a 10 CFR 50.59 safety evaluation to ensure that an unreviewed safety question did not exist when the R-11 and R-114 was initially brought onsite. In addition, the licensee did not update the USAR, as required. The failure to perform a safety evaluation for a change to the facility as described in the USAR is a violation of 10 CFR 50.59. As immediate corrective action, the licensee issued a memorandum to procurement personnel to ensure that the USAR is consulted for

bringing new material onsite and began evaluating procedure changes for this process. In addition, a 10 CFR 50.59 safety evaluation was completed. The inspectors review confirmed that an unreviewed safety question did not exist.

This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII of the NRC Enforcement Policy. Specifically, the violation was identified by the licensee, was not willful. Actions taken as a result of a previous violation should not have corrected this problem, and appropriate corrective actions were being or had been completed by the licensee.

7 PLANT SUPPORT ACTIVITIES (71750)

On February 10, 1996, during a tour of the auxiliary building, the inspectors found that one of the main steam tunnel plugs did not have the rubber sealant applied around it. The inspectors expressed concern that the plant was about to start up with an unrecognized degraded condition that could have a detrimental affect on secondary containment, environmental qualifications, or fire protection. The inspectors notified the shift superintendent and the control room supervisor. The shift superintendent and the control room supervisor were in agreement that the missing sealant was not required and did not pursue the issue further. Based on this information, the inspectors did no additional followup at the time.

Subsequently, on February 14 a mechanical maintenance supervisor also noted that the sealant around the main steam tunnel plug was not installed. The licensee initiated CR 96-0500 to document this issue and to propose corrective actions. This CR stated that the problem resulted from miscommunication among maintenance personnel and indicated that the sealant provided a fire protection barrier. Therefore, an impaired fire barrier existed without establishing a fire watch within 1 hour, as required by the fire protection program. Upon notification of CR 96-0500, the shift superintendent added the impaired fire barrier to the hourly fire patrol. Mechanical maintenance then corrected the missing fire sealant and operations removed this item from the hourly fire patrol watch.

The licensee evaluated this condition and noted that it had no safety significance. The licensee noted that they had performed previous evaluations that had removed the need for similar fire sealant around shield blocks in other areas. In addition, almost all of the auxiliary building was already on the fire tour, so the fire patrol watch would be traversing through the area in any case.

The inspectors agreed with the assessment and the conclusions. However, the inspectors noted that they had informed two senior reactor operators in the control room of the missing sealant four days earlier, prior to the plant startup. These senior reactor operators made an incorrect determination of the deficient condition, without sufficient knowledge of the function of the sealant and without obtaining the proper technical assistance. Therefore, the licensee did not initiate a CR, or add the item to the hourly fire patrol

watch. The inspectors discussed these concerns with the operations manager. The operations manager stated that the senior reactor operators involved did not meet his expectations, and the operations manager discussed with operations personnel the need for complete evaluations considering all relevant technical input prior to increasing power. The inspectors considered the corrective actions appropriate.

8 FOLLOWUP OF PREVIOUSLY IDENTIFIED ITEMS (92903)

8.1 (Closed) Inspection Followup Item 458/9505-01: Review of New Procedure Covering Guidance on Operability Evaluations

On June 28, 1995, system engineering submitted an operability evaluation declaring the high pressure core spray pump operable even though the pump failed to start upon demand during a surveillance. The operability evaluation had a weak basis in that it did not identify the cause of the failure. Also, a question remained concerning the relationship between actual starting current and overcurrent relay settings.

The technical concerns were resolved, and the licensee had already implemented a Quality Action Team to assess operability evaluations and recommend improvements. A guidance procedure was being developed, and the inspectors opened this inspection followup item to review the procedure when it was issued.

On December 19 the licensee published Procedure RBNP-078, "Operability Determinations," Revision 0. The inspectors reviewed the new procedure and found that it contained sufficient processes and guidelines necessary for licensee personnel to make operability decisions for systems or components with degraded and/or nonconforming conditions. The procedure contained attachments covering the operability policy and guidelines for engineering evaluations. The guidelines reflected the operability concepts delineated in NRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability."

The inspectors determined that the new procedure provided the type of guidance needed for consistent and technically adequate engineering operability evaluations. The procedure also provided instructions relative to replacing manual actions for automatic actions, which were consistent with NRC generic guidance.

9 ONSITE REVIEW OF LERs (92700)

9.1 (Closed) LER 458/96-001: Manual Reactor Scram Because of High Turbine Vibration

This event was discussed in NRC Inspection Report 50-458/95-26. At the time of the inspection, the root causes of the necessity to scram the reactor and manually trip the main turbine because of high vibration were not known. The

LER identified the causes as: (1) tight clearances in the new low pressure turbine rotors installed during Refueling Outage 5 and (2) moisture separator reheater control anomalies, which have caused uneven heat distribution in the low pressure turbines in the past.

Corrective actions included maintenance on, and modification of, reheater controls during RF06. The inspectors noted that there were no turbine vibration problems during the February 1996 startup from RF06. The final proof test will be the next time the plant is shut down without a scram. The inspectors determined that the licensee took appropriate corrective actions to address the apparent root cause of the turbine vibrations that required operators to manually scram the reactor and that no violations of regulatory requirements occurred.

9.2 (Closed) LER 458/96-004: Containment Isolation of Various Balance of Plant Valves Because of Inattention to Detail

This event was addressed in Section 2.1 of this inspection report. A noncited violation was identified for failure to comply with the surveillance procedure.

9.3 (Closed) LER 458/96-005: Noncompliance with License Condition by Inadequate Tagging of Hose and Cabling Because of Change Management

This issue was addressed in Section 3.4 of this inspection report. A violation was cited because immediate corrective actions, to ensure compliance with all provisions of Procedure OSP-0034, were not completed.

10 REVIEW OF USAR COMMITMENTS (37551)

A recent discovery of a licensee operating their facility in a manner contrary to the USAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the USAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the USAR that related to the storage of chemicals onsite and control room habitability. The following inconsistency was noted between the wording of the USAR and the plant practices, procedures and/or parameters observed by the inspectors.

As described in Section 6, control room habitability related to toxic chemicals is based on the amounts of chemicals stored onsite and listed in USAR Table 2.2-5. The licensee stored more freon onsite than the values listed in the table. Also, the licensee stored R-11 onsite but the chemical is not listed in the table. The licensee had not performed a 10 CFR 50.59 evaluation, as required, and the inspectors identified this as a noncited violation. The licensee is evaluating procedure changes to assure that materials brought onsite would be compared to the USAR requirements.

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

R. J. Alexander, Manager, Project Management
J. P. Dimmette, General Manager, Plant Operations
D. T. Dormandy, Manager, System Engineering
J. O. Fowler, Supervisor, Quality Assurance
K. J. Giadrosich, Acting Manager, Maintenance
T. O. Hildebrandt, Manager, Outage Management
G. C. Hockman, Quality Assurance Specialist
J. Holmes, Superintendent, Chemistry
M. A. Krupa, Manager, Operations
T. R. Leonard, Director, Engineering
L. G. Lewis, Manager, Training
D. N. Lorfing, Supervisor, Licensing
J. R. McGaha, Vice President-Operations
W. H. Odell, Superintendent, Radiation Control
B. R. Smith, Superintendent, Mechanical and Electrical
J. E. Venable, Manager, Licensing
L. W. Woods, Superintendent, Operations
G. A. Zinke, Manager, Quality Assurance

1.2 Other NRC Personnel

D. A. Powers, Chief, Maintenance Branch, Division of Reactor Safety, RIV

The above personnel attended the exit meeting. In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on March 5, 1996. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee acknowledged the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.