ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-458/96-01

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 8 through March 5, 1996

Inspectors: Lawrence E. Ellershaw, Reactor Inspector, Maintenance Branch
Division of Reactor Safety

Paul C. Gage, Reactor Inspector, Engineering Branch
Division of Reactor Safety

Charles J. Paulk, Reactor Inspector, Maintenance Branch
Division of Reactor Safety

Gregory E. Werner, Reactor Inspector, Maintenance Branch
Division of Reactor Safety

Approved: Dale A. Powers 04/11/96
Dr. Dale A. Powers, Chief, Maintenance Branch — Date
Division of Reactor Safety

Inspection Summary

Areas Inspected: Routine, announced inspection of the in-core fuel loading and fuel storage configurations, core component performance, potential for fuel-related problems identified at other facilities, and fuel handling procedures and practices. In addition, followup of maintenance activities, and inspection of the licensee's inservice inspection program and related nondestructive examination activities were performed.

Results:

Plant Operations

- The policies and procedures in place for adequate shutdown margin during refueling activities satisfied the minimum requirements; however, precautions for inadvertent criticality and misloaded or misoriented fuel assemblies were lacking. The licensee's reliance on a bounding shutdown margin analysis for refueling operations without an increased use of and focus on plant nuclear instrumentation was questionable, especially given the frequency of fuel placement errors at the River Bend Station (Section 2.2).

- Management oversight of new fuel handling activities needed strengthening. A noncited violation was identified regarding a receipt inspector who failed to perform a procedurally required dimensional inspection on new fuel assemblies. Confusion resulted in a channel fastener on a new fuel assembly being damaged when it was misloaded and misoriented in the spent fuel pool. Inadequate training resulted in poor new fuel handling techniques which led to damaged channel fastener springs and an improperly seated fuel assembly in the spent fuel pool (Section 2.4).

- The licensee's training and qualification program for fuel handling personnel included both classroom and practical training on the use of fuel handling equipment prior to actual fuel movement. However, on-the-job training did not provide practical training for responding to fuel handling abnormal events or emergencies, and the abnormal operating procedure contained vague operator guidance (Section 2.5).

- The licensee maintained very good refueling water clarity, camera resolution, and pool lighting (Section 2.8).

- The inspectors observed numerous inconsistencies between the various refueling crews and shifts, including differences in command and control practices and communications techniques. The inspectors expressed concerns to licensee personnel regarding potential performance problems (Section 2.11).

- The inspectors observed a refueling crew mishandle a double blade guide and attempt to circumvent procedural controls. This event showed a lack of concern regarding core component protection, and a reluctance to involve management in problem resolution (Section 2.11).

- A violation of the fuel movement plan and the fuel handling procedure was identified on January 21, 1996, when a refueling crew incorrectly placed a fuel bundle in the wrong core location. The refueling crew had failed to properly verify the correct core location prior to depositing the fuel assembly and relied upon a nonproceduralized visual aid to position the fuel bundle (Section 2.11).

- Operations personnel's knowledge of the air supply systems for the lower and upper pool pneumatic gate seals was poor (Section 2.13).

Maintenance

- Certain operators performing fuel handling area ventilation system surveillances were incorrectly recording pressures; however, Technical Specification surveillance requirements were satisfied (Section 2.6).

- The inspectors identified a violation of Technical Specification 5.4.1 when a technician failed to verify refueling platform zone computer interlocks in accordance with the specified requirements of the procedure (Section 2.7).

- A maintenance technician's lack of understanding of site lubrication requirements and conflicts between maintenance procedures and the lubrication manual provided a potential for use of an incorrect lubricant on fuel handling equipment (Section 2.7).

- The inspectors identified a violation of Technical Specification 5.4.1 regarding procedures not being adequate for testing the main hoist interlocks on the fuel handling platform bridge and the refueling platform. Further, the main hoist interlock of the fuel handling platform bridge was not tested within 7 days of its use for handling fuel assemblies or control rods (Section 2.7).

- The inspectors identified a violation of Technical Specification 5.4.1 regarding the failure to establish accountability of foreign material in the spent fuel pool area and the failure to properly control foreign material in the upper pool areas (Section 2.10).

- The inspectors identified a violation of Technical Specification 5.4.1 regarding foreign material control in the suppression pool. The ability to positively demonstrate system operability under design basis accident conditions was questionable, in that the administrative controls which required the initiation of condition reports by personnel dropping items into the suppression pool or losing track of items that could have fallen into the suppression pool, were not routinely being complied with by site personnel (Section 2.10).

- The physical layout and protection of the emergency nitrogen backup air supply for the pneumatic gate seals was poor, in that the plastic tubing had no permanent support, was taped to electrical conduits, hand railings, and the floor and was routed in high traffic areas, thus, creating a high susceptibility to damage (Section 2.13).

- The observed inservice inspections were performed in accordance with requirements (Section 3).

- A failed weld on a residual heat removal system minimum flow/test return line was due to lack of fusion and fatigue of the fused areas. The lack of fusion was the result of improper weld technique, which was used by the fabricator in 1982. Additionally at that time, the indication identified during the associated radiographic inspection was misinterpreted to be misalignment rather than lack of fusion (Section 4.1).

- Upon identifying the failed dissimilar metal weld in a residual heat removal system minimum flow/test return line, the licensee acted appropriately in identifying root causes, establishing corrective actions, and evaluating possible generic implications (Section 4.1).

- During review of Licensee Event Report 50-458/94-17, the inspectors concluded that the licensee-identified and corrected violation regarding the failure to test check valves in accordance with the ASME Code, was a noncited violation (Section 4.3).

Engineering

- The licensee's review of NRC Information Notice 88-92 and Supplement 1 was limited and lacked a questioning attitude. It failed to review plant drawings, procedures, Updated Safety Analysis Report, and design documents to determine the actual design, as-built conditions, and use of the seal and support systems. For other fuel-related information notices, the licensee consistently employed conservatism in evaluations (Section 2.1 and 2.13).

- The licensee had taken appropriate measures to ensure reliable fuel performance. Where a small number of fuel failures did occur, post-irradiation examinations were conducted in an effort to determine the cause and, subsequently, improve performance (Section 2.9).

- Over the years, engineering documents and condition reports were initiated concerning the pneumatic gate seals. Several of the engineering documents and condition reports addressed a number of potential problems, including seal life, maintenance, inspection, and partial draindown of the dryer pool. These documents provided recommended actions; however, licensee personnel did not take effective corrective actions (Section 2.13).

- The inspectors identified a violation in the area of design control. Licensee personnel had classified the pool gates and pneumatic seals as safety-related. Yet, appropriate measures (i.e., procedures, instructions, or drawings) commensurate with the safety-related classification designated by the licensee were not established to assure that the pneumatic gate seals would be maintained as safety-related components (Section 2.13).

Plant Support

- Cognizant licensee personnel did not adequately prepare for the new fuel receipt inspection process, particularly with respect to inspector training and sensitivity to proper and safe fuel handling (Section 2.4).

Management Overview

- With the exception of reactor engineering management oversight during portions of the first day of irradiated fuel movement, the inspectors did not observe any management oversight of fuel handling activities on the refueling and fuel handling bridges (Section 2.11).

Summary of Inspection Findings:

- A Noncited Violation was identified (Section 2.4).
- Violation 50-458/9601-01 was opened (Section 2.7, 2.10).
- Violation 50-458/9601-02 was opened (Section 2.7).
- Violation 50-458/9601-03 was opened (Section 2.11).
- Inspection Followup Item 50-458/9601-04 was opened (Section 2.13).
- Violation 50-458/9601-05 was opened (Section 2.13).
- Violation 50-458/9419-01 was closed (Section 4.2).
- Licensee Event Report 50-458/94-17 was closed (Section 4.3).
- A Noncited Violation was identified (Section 4.3).

Attachments:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Documents Reviewed

DETAILS

1 PLANT STATUS

During this inspection period, the plant transitioned from Mode 5 (Refueling Outage RF-6) to Mode 1 (full power operations).

2 FUEL INTEGRITY AND REACTOR SUBCRITICALITY (60705/60710/86700)

The objectives of fuel integrity and reactor subcriticality inspection were to review, inspect, and determine the adequacy of the licensee's activities related to the protection of reactor fuel. Attachment 2 to this inspection report is a tabulation of documents that were reviewed by the inspectors during the inspection and provided some of the basis for the findings documented in this report. In general, the reviews of procedures and records were not detailed in nature, but were broad overviews to determine that essential issues were addressed in reasonable fashion.

NRC Inspection Manual Procedures 60705, "Preparation for Refueling"; 60710, "Refueling Activities"; 86700, "Spent Fuel Pool Activities"; and, 92902, "Followup - Maintenance," provided partial guidance for this inspection effort.

2.1 Fuel-Related Incidents at Other Facilities

The inspectors discussed with licensee personnel several fuel-related events that have occurred at other commercial nuclear power plants. The specific incidents discussed are described in NRC information notices and bulletins that were issued during the past decade (See Attachment 2).

Portions of Information Notice 88-92 and Supplement 1, which dealt with spent fuel pool issues, were not adequately addressed by the licensee. This is discussed in detail in Section 2.13. With the exception of this notice, the inspectors considered the actions taken by the licensee, with respect to the generic communication documents reviewed, to be appropriate. The inspectors, after review and verification of licensee evaluations associated with other fuel-related information notices, concluded that the licensee was responsive and consistently employed conservatism in these evaluations.

2.2 Shutdown Margin and Premature Criticality

The inspectors noted that the license requirements for refueling (Mode 5) shutdown margin specified by Technical Specification 3.1.1 required a minimum margin of core reactivity for any vessel core configuration that resulted from fuel handling activities. The inspectors reviewed the assumptions and results of these determinations for the current refueling outage (RF-6).

Licensee operators administratively controlled shutdown margin by implementing a fuel movement plan which was independently reviewed to assure the minimum shutdown margin was satisfied. Reactor engineering personnel utilized a bounding analysis for determining the shutdown margin required by Technical Specification 3.1.1.1. This was permitted by Step 7.3.1 of Procedure STP-050-3601, "Shutdown Margin Demonstration," Revision 9A, and was documented in the bases section of the Technical Specification. The inspectors were informed that the bounding analysis was based on another boiling water reactor operated by Entergy Operations, Inc.

The inspectors conducted a review of the bounding analysis for adequate shutdown margin. It was noted that the bounding analysis designated certain core locations be defueled prior to fuel being relocated within the vessel, and that fuel could not be reinserted in those restricted locations until all other fuel was in its final, Beginning-of-Cycle 7, configuration. Based on the bounding analysis, reactor engineering personnel developed the fuel movement plan, which the inspectors verified for adherence to these restricted fuel locations.

The inspectors noted that the bounding analysis also assumed a maximum of four fuel assemblies placed in an incorrect location in the core (i.e., misloaded), and a maximum of ten fuel assemblies misoriented in a correct core location. During review of Procedure REP-0029, "Fuel Movement," Revision 2, the inspectors identified that steps were provided to check for misloaded fuel and, subsequently verify the shutdown margin limitations. Further review indicated that for misoriented fuel, there was no reference to the specific limitation imposed by the bounding analysis for adequate shutdown margin. The inspectors noted that the fuel movement procedure inappropriately allowed an undetermined number of fuel assemblies to be misoriented without verifying the adequacy of the shutdown margin.

The inspectors observed that minimal guidance existed relative to an inadvertent criticality. Procedure AOP-0027, "Fuel Handling Mishaps," Revision 9, listed one symptom (steady increasing count rate with a measurable period), and directed operators to manually insert a reactor scram and monitor neutron instrumentation to verify decreasing count rate. The inspectors noted that this procedure had significant steps for addressing fuel handling equipment problems, but was limited in direction and guidance provided for an inadvertent criticality while shutdown during refueling operations.

The policies and procedures in place satisfied the minimum requirements for shutdown margin during refueling; however, precautions for inadvertent criticality and misloaded or misoriented fuel assemblies were lacking. The licensee had no specific written instructions or guidance for the control room reactor operators concerning monitoring for inadvertent criticality. The inspectors were informed, during telephone conversations with a program representative from the Office of Nuclear Reactor Regulation, that it was not the intent that the licensee employ a bounding shutdown margin calculation without specific monitoring of instrumentation for startup.

2.3 Licensee Commitments on Fuel-Handling Activities

The licensee committed to Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)," Revision 2, Appendix A, February 1978, in Technical Specification 5.4.1.a. The guide endorsed Standard ANS 3.2-1972, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." This standard required written procedures for core alteration, accountability of fuel, and partial or complete refueling operations. Specific procedures (listed in Appendix A of Regulatory Guide 1.33) were also required for each refueling outage and for receipt and shipment of fuel.

2.4 New Fuel Receipt and Receipt Inspection

The inspectors reviewed the receipt inspection activities associated with the new fuel scheduled to be used during Refueling Outage RF-6. The inspectors became aware of three condition reports initiated by licensee personnel during the receipt inspection effort.

- Condition Report 95-1148 (Improper Inspections)

On December 5, 1995, licensee personnel initiated Condition Report 95-1148 because personnel had not properly performed inspections for several new fuel assemblies. Step 9.8.9 of Procedure REP-0005, "New Fuel Receipt," Revision 5, required, in part, that the maximum dimension between the outer surface of the channel fastener guard and the channel be verified as less than 0.208 inches. This check ensured that there was adequate clearance for placing the bundle in the core. Licensee personnel identified one fuel inspector who was accepting this dimension through visual observation, rather than taking a direct measurement. By the time licensee personnel recognized the fuel inspector's incorrect activities, several other improperly inspected assemblies had been placed in the spent fuel pool. The licensee temporarily suspended new fuel receipt to resolve this issue.

The failure to follow the requirements specified in Procedure REP-0005 is a violation of Technical Specification 5.4.1.a. This licensee-identified violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

The corrective actions implemented by the licensee included the following:

(1) A vendor representative discussed proper use of the measuring gage to check the gap between the channel fastener and the channel with new fuel inspectors.

(2) Quality control inspectors were assigned to observe fuel inspections.

(3) The licensee reviewed the training and certification of fuel receipt inspectors to ensure that the inspectors had the required training, and

(4) The licensee reinspected the suspect assemblies in the spent fuel pool.

The corrective actions were satisfactory and no further instances of improper receipt inspection were identified.

- Condition Report 95-1149 (Fuel Channel Fastener)

On December 5, 1995, licensee personnel initiated Condition Report 95-1149 which stated that the fuel channel fastener for Fuel Assembly GGE218 was bent. This condition was caused when personnel attempted to place the fuel assembly in an incorrect location and misoriented configuration in the spent fuel pool. The fuel movement plan designated spent fuel pool Location QQ-15 (with a northeast orientation) as the storage location for Fuel Assembly GGE218. The fuel mover manipulated the fuel assembly into a northeast orientation in preparation for placement in Location QQ-15. The reactor engineer and the fuel mover somehow confused two separate sheets in the fuel movement plan and attempted to place the fuel assembly in spent fuel pool Location NN-14 (which had a northwest orientation). The error was identified when an attempt was made to locate Fuel Bundle GGE216 into its designated, but already occupied, location (NN-14). This error was attributed to personnel error (i.e., lack of attention to detail). This resulted in the bent fuel channel fastener. This current example of confusion in properly following the procedural guidance sequence of fuel movements is similar, but not identical, to another example that occurred in the reactor vessel during the previous refueling outage.

New fuel receipt activities were temporarily suspended to resolve this problem. Licensee personnel revised the fuel movement plan to correct the misload. The personnel involved were counseled to provide more strict attention to detail. No similar incidents occurred and licensee representatives indicated that they would replace the bent channel fastener prior to placing the fuel bundle in the core.

- Condition Report 96-0025 (Damaged Channel Fastener Springs)

On January 3, 1996, during an underwater video inspection of the new fuel assemblies in the spent fuel pool, licensee personnel found five channel fastener springs damaged and hung up on the spent fuel racks. Condition Report 96-0025 was initiated to address this issue. Investigation revealed that deficiencies existed in the training program for movement of new fuel. Licensee representatives identified that the individuals moving fuel were not fully aware of the techniques used at the River Bend Station to ensure that the fuel bundles were properly seated in the spent fuel pool. For corrective action, the training and qualifications of the new fuel handlers were revised and proficiency examination was required for each fuel handler.

The inspectors reviewed the corrective actions and questioned how a channel fastener spring could hang up on the fuel racks if properly inspected and verified flush with the channel fastener bracket. The licensee's review determined that the condition of the channel fastener springs was not inspected by the fuel vendor or by their own new fuel receipt inspection plan.

Vendor drawings indicated that the channel fastener springs should be flush with the channel fastener bracket. With this information, the licensee revised the condition report response to reflect the new information, and requested that the fuel vendor revise their channel fastener inspection requirements. In addition, the licensee stated that they would revise Procedure REP-0005 to include requirements for inspection of the channel fastener springs. While the inspectors considered the final disposition of the condition report to be appropriate, initial licensee efforts to establish the root-cause and corrective actions were not comprehensive, and may not have been sufficient to prevent future channel fastener spring damage.

Cognizant licensee personnel did not adequately prepare for the new fuel receipt inspection process, particularly with respect to inspector training and sensitivity to proper and safe fuel handling. Also, as evidenced by the number of incidents and actual damage to fuel core components, management oversight of fuel handling activities needed strengthening.

2.5 Fuel Handling Personnel Qualifications and Training Program

For the first time at the River Bend Station, all fuel handling activities during the refueling outage were to be performed by licensee personnel (Entergy Operations, Inc.) from River Bend and other licensee sites. In addition, a limited number of General Electric Company personnel were assigned to assist during the preparation for, and actual fuel handling. The refueling activities were overseen by senior reactor operators, serving as refueling coordinators. The licensee personnel from the other sites had varying technical backgrounds; however, those interviewed by the inspectors were found to have been involved in previous fuel handling activities at their respective sites. For the personnel from Arkansas Nuclear One and Waterford Steam Electric Station, however, this was the first time that they had handled fuel at a boiling water reactor. The Entergy personnel had attended one week of training at the General Electric Company's training facility, and had also received classroom training at River Bend Station on the refueling equipment. Site management stated that in the future, all Entergy sites would use, essentially, these same individuals for refueling activities.

The inspectors reviewed the training received by members of the licensee's fuel handling team identified by Letter DED 96-02, Revision 1, dated January 9, 1996. The inspectors noted practical training, as addressed in Procedure REP-0029, was documented for fuel handlers, fuel spotters, and fuel movement supervisors. The inspectors also noted that the licensee formalized the qualification process on January 4, 1996, by issuing on-the job training qualification cards to all fuel handling operators, and that although this training included practical experience in using or simulation of actual fuel handling equipment, it did not include any examination of the operators ability to address normal or abnormal circumstances. The inspectors' review of training records indicated that all fuel handling team personnel had completed the required classroom and practical training prior to any fuel movement.

Although a variety of fuel handling mishaps were indicated in Procedure AOP-0027, the inspectors noted during review of the qualification cards for fuel handling personnel that only one abnormal condition performance item was required to be performed or simulated, that being, lineup of cooling water to the transfer tube.

The inspectors observed that Lesson Plan HLO-535-0 included a review of Procedure AOP-0027 as part of the classroom training. The inspectors noted that Procedure AOP-0027 contained vague operator guidance. Specifically, if any difficulty arose while handling fuel, then the operator was to cease fuel handling after the assembly was placed in a safe condition. The licensee's procedures did not define or specify what a safe condition for a fuel assembly was. The management-approved fuel handler qualification process for on-the-job training did not provide practical training for responding to fuel handling abnormal events or emergencies.

2.6 Fuel Handling Area Ventilation System

The inspectors reviewed Technical Specification 3.6.4.5, "Fuel Building," and compared the surveillance requirements to those contained in Surveillance Test Procedure STP-000-0103, "Irradiated Fuel Handling in Fuel Building," Revision 3, which was signed off as complete on January 12, 1996.

The inspectors noted that this procedure did not reference the Improved Technical Specification 3.6.4.5, but still referred to previous Technical Specifications 4.6.5.2.a, b, c, and d, which were required to be completed within 24 hours prior to fuel movement and at least once every 7 days during irradiated fuel handling in the fuel building. The frequency for the previous Technical Specifications were more restrictive than the Improved Technical Specifications with one exception. Surveillance Requirement 3.6.4.5.1 required fuel building vacuum to be ≥ 0.25 inches of vacuum water gauge (same as previous Technical Specification), but required the pressure to be checked every 24 hours as opposed to every 7 days. The inspectors determined that Procedure STP-000-0103 did not satisfy Surveillance Requirement 3.6.4.5.1. However, all other surveillance requirements were appropriately tested in Procedure STP-000-0103.

Additional followup by licensee personnel indicated that Procedure STP-000-0005, "Daily Refueling Logs," Revision 9, Step 39, did satisfy Technical Specification Surveillance Requirement 3.6.4.5.1. However, the inspectors noted inconsistencies in logging the value of fuel building pressure. Some operators logged fuel building pressure as negative while others logged the number as positive. The inspectors observed the fuel building pressure gauge (Gauge 1HVF-PRI103) and found that the gauge had a positive scale from 0 to 1 inches of vacuum water gauge.

Licensing personnel were informed of the above observations and appropriate actions were taken to revise Procedure STP-000-0103. Operations personnel were instructed on how to properly interpret Gauge 1HVF-PRI103.

The inspectors concluded that Technical Specification surveillance requirements for fuel building vacuum had been satisfied.

2.7 Fuel Handling Equipment Maintenance and Surveillance

The inspectors reviewed fuel handling equipment maintenance and surveillance activities, and identified the following issues:

- Pre-operational Check of the Refueling Platform Zone Computer Interlock

Procedure ME01597, "Refueling Platform," Revision 8, delineated inspection criteria for various components of the refueling platform, and was used to prepare the platform for the refueling outage. Maintenance Work Order P581791 was used to implement the requirements contained in Procedure ME01597.

Step 9.4.9.14 was signed off as complete on January 2, 1996. Procedure ME01597 stated, "Zone Computer Interlock: Verify zone computer permissive zones are in accordance (with) Reference 3.19." Reference 3.19 was identified as Technical Manual 3224.110-000-016A, "Refueling Platform, GEK-83294." Step 3.19.16 of the technical manual gave numerous checks to be completed for the testing of zone computer interlocks. During conversations with the system engineer and the technician who performed the procedure, the inspectors learned that Step 3.19.16 was not performed. The technician stated that he used his experience and a copy of a zone map contained within the technical manual (Figure 3.10) to check the zone computer interlocks.

The inspectors and the reactor engineer compared the zone map to Step 3.19.16 of the technical manual and noted that there was not a one-to-one match up of coordinates on the zone map to all the coordinates in Step 3.19.16. Neither the technician nor the system engineer could confirm whether all the checks prescribed by Step 3.19.16 were in fact tested by the use of Figure 3.10. The failure to verify the zone computer interlocks in accordance with the specified requirements of the Procedure ME01597, Step 9.4.9.14, constituted a violation of Technical Specification 5.4.1 (458/9601-01).

The purpose of the zone computer was to prevent the refueling platform and main mast from moving past certain three-dimensional coordinates (i.e., to preclude possible fuel damage by preventing collisions with fixed objects in the pool). While there were no instances identified where the zone computer interlocks were challenged during this outage, the inspectors were concerned that a technician signed off a step in a work package as being complete without fully understanding whether all the requirements of that step had been performed. The inspectors were further concerned because the system engineer approved of this action, apparently without verifying that the actions taken by the technician were in accordance with the procedure and encompassed all of the checks specified in the technical manual.

- Preventive Maintenance - Lubrication of Fuel Handling Platform

The inspectors compared the technical manual for the fuel handling platform to Preventive Maintenance Procedure ME01600, "Maintenance," and several differences in lubrication requirements were identified. Step 9.5.6 of Procedure ME01600 stated that a lithium-base grease was to be used to lubricate all fuel handling components. The technical manual recommended using either a NLG #2 lithium-base or a silicon-base greases.

Procedure GMP-0015, "Lubrication Procedure," stated that the General Maintenance Lubrication Manual was the only document to be referenced concerning lubrication requirements. Further, licensee personnel indicated that the General Maintenance Lubrication Manual contained all equipment lubrication points. A licensee representative stated that all technicians were aware of the requirement to use only the General Maintenance Lubrication Manual for lubrication requirements.

The inspectors asked an experienced mechanic to cross reference the 14 grease lubrication points identified in Procedure ME01600 with the Lubrication Manual. The mechanic could not match five points and was unsure of three others (due to differing or missing component nomenclature or identification). When questioned what he would do in this case, he indicated that he would check the technical manual for the proper type of lubrication. This was not in accordance with the requirement in Procedure GMP-0015, and it did not meet management's expectations.

The inspectors determined that the missing and/or differing nomenclature of component lubrication points was confusing, and could lead to putting an incorrect type of grease in a component. No instances of using the incorrect type of grease were identified during this inspection.

The cognizant engineering supervisor initiated Condition Report 96-0417 on February 2, 1996, to document and review the problems identified by the inspectors.

- Surveillance Requirements - Fuel Handling Platform

The inspectors reviewed Procedure STP-055-0705, Revision 9, "Fuel Handling Platform Operability Test," and noted that it stated all surveillance requirements of Technical Requirement 3.9.13 were satisfied. However, during review of the procedure, the inspectors identified that Surveillance Requirement TSR 3.9.13.7 may not have been adequately tested. Surveillance Requirement TSR 3.9.13.7 required that the main hoist loaded hoist interlock be tested and demonstrated operable within 7 days prior to the start of handling fuel assemblies or control rods. Step 1.1.7 of Procedure STP-055-0705 stated that the purpose of the surveillance was to demonstrate operation of the main hoist loaded interlock for Surveillance Requirement TSR 3.9.13.7. Procedure STP-055-0705, Step 7.1.10, was shown to satisfy this technical manual surveillance requirement. Section 8.0, "Acceptance Criteria," stated, "The Fuel Handling Platform Main Hoist is operable when each of the following is satisfied: The loaded interlock functions on the Main Hoist before the total cable load exceeds 300 to 400 pounds as demonstrated by Step 7.1.10." However, Procedure STP-050-0705, Step 7.1.10 appeared to only check a hoist loaded indicating light, not the interlock.

The inspectors questioned the system engineer concerning the function of the main hoist loaded interlock and if Procedure STP-055-0705, Step 7.1.10 adequately tested the main hoist interlock. The system engineer did not know the function of the interlock and was unsure if Step 7.1.10 satisfied the surveillance requirement for testing the main hoist loaded interlock. Later, the system engineer informed the inspectors that the function of the main hoist loaded interlock was to block main hoist in the hoist direction (up direction) when two conditions existed: hoist load of 350 +/- 50 lbs and grapple not engaged. The inspectors identified that the hoist loaded interlock contained Contacts MHL1 and GC (Drawing C-23200-03-B). Contact MHL1 was to open when the load exceeded 350 +/- 50 lbs and Contact GC was to open when the grapple was not engaged. Procedure STP-055-0705 only checked the operation of Contact MLH1.

The system engineer also stated that Step 7.1.10 in Procedure STP-055-0705 did not satisfy the requirement to test the interlock. The system engineer stated that Procedure STP-055-0705, Step 6.6, required that applicable sections of Procedure ME01600 be completed prior to performing Procedure STP-055-0705. The system engineer further stated that Section 6.6 was signed off as complete and this, therefore, satisfied Surveillance Requirement TSR 3.9.13.7. The inspectors reviewed both procedures and no references were given as to which steps were applicable to Procedure STP-055-0705. In addition, the inspectors discussed the testing requirements with the senior reactor operator who signed off the test as being completed. The senior reactor operator described the acceptance criteria for Step 7.1.10 of Procedure STP-055-0705 as a light test only, and that no other criteria or attribute was tested. In addition, the senior reactor operator stated that no other procedure was required to be completed to satisfy testing of the main hoist loaded interlock.

The inspectors reviewed Maintenance Work Order P581784 which was used to implement Revision 10 of Procedure ME01600. It showed that the main hoist loaded interlock was tested on November 21, 1995. Licensee personnel stated that the platform was idle for approximately 3 weeks, thus, this test occurred more than 7 days prior to use of the fuel handling platform.

Procedure STP-055-0705, as written, did not adequately test the main hoist loaded interlock and did not contain appropriate guidance to ensure applicable steps of Procedure ME01600 were completed prior to releasing the fuel handling platform bridge for operational use. The failure to test the main hoist loaded interlock as required in Procedure STP-055-0705 constituted a violation of Technical Specification 5.4.1 to properly establish and implement procedures (458/9601-02).

2.8 Primary Water Clarity

During the inspection, the clarity of water in the cavity was very good with degradations noted only during residual heat removal system lineup changes. The inspectors observed that while thermal variations did present occasional visual difficulties, all irradiated fuel bundle serial numbers were visible and able to be verified using the mast camera.

While there are not any specific regulatory requirements pertaining to water clarity, safe fuel handling necessitates visual confirmation of fuel components for their movement. Procedures FHP-002, "Refuel Platform Operation," and FHP-003, "Fuel Handling Platform Operation," included a precautions and limitations section, which required the cessation of any fuel or equipment movement upon loss of visual indication and identification of the fuel or equipment. The refueling senior reactor operator had the responsibility to determine when water clarity was appropriate for fuel movement. During this outage, fuel movement was suspended several times to allow temporary filters to be used to clean up the cavity water after residual heat removal lineup changes and vessel nozzle cleaning.

The inspectors noted that the lighting in all reactor cavity and vessel locations was very good. In addition, the resolution of the mast-mounted camera was very good.

2.9 Fuel Assembly Post-Irradiation Examination and Reconstitution

Reconstitution of fuel assemblies with failed rods had been attempted but not actually performed at River Bend Station. Therefore, the inspectors did not review those procedures, nor prior activities.

The licensee had established procedures which designated responsibilities and addressed fuel reliability, including required activities associated with fuel failures. These included Procedures NF-101, "Nuclear Fuel Program and Divisions of Responsibility," Revision 3, and NF-102, "Corporate Fuel Reliability," Revision 0.

The inspectors were informed that a total of five fuel failures had previously occurred during Cycles 3 (two failures), 5 (two failures), and 6 (one failure), and that post-irradiation examinations had been performed.

Manufacturing tubing reduction flaws were attributed to both of the Cycle 3 failures and one of the Cycle 5 failures, while an endcap weld defect caused the other failure during Cycle 5. A joint effort between the fuel fabrication vendor (GE Nuclear Energy) and the licensee resulted in an excellent evaluation of the Cycle 6 failure. This included visual examination of the failed rod and adjacent rods, encircling-coil eddy-current examination of the adjacent rods, and a review of operating history, fission product activity trends, and manufacturing records. Determination of the likely cause of failure consisted of comparing the results of the examinations and reviews of related information to known failure mechanisms. While the inspectors considered the joint evaluation to be an excellent effort in terms of depth and quality, licensee representatives stated that different conclusions were arrived at by the two organizations. The licensee concluded the primary cause of failure was unknown, while GE Nuclear Energy concluded the primary cause of failure to be debris fretting.

Based on the small number of fuel failures to date, it appeared that the licensee had taken appropriate measures to ensure reliable fuel performance. Where fuel failures did occur, post-irradiation examinations were conducted in an effort to determine the cause and subsequently improve performance.

2.10 Loose Parts and Foreign Material Exclusion

The inspectors examined the licensee's controls for general cleanliness, loose parts, and housekeeping activities associated with the reactor building pools (upper fuel transfer pool, dryer storage pool, separator storage pool, and refueling cavity pool) and the fuel building pools (lower fuel transfer pool, shipping cask pool, and spent fuel pool).

Attachment 2 in Procedure ADM-0081, "Cleanliness Control," Revision 3, listed the spent fuel pool and upper cavity pools as areas requiring controls for foreign material exclusion. The procedure required accountability by logging personnel, tools, and other items brought into the foreign material exclusion areas. Additionally, it required items to be "fail-safe" (through the use of rope, tape, or lanyard devices) to prevent them from falling into a pool area.

During the course of the refueling observations, the inspectors observed numerous items being brought into the fuel building and reactor building foreign material exclusion areas. A number of items brought into the spent fuel pool area were not properly recorded or logged in (binoculars), quantified (pens and pencils), or tracked and logged out (4 of 11 items removed). However, the inspectors did not observe any foreign material in the spent fuel pool.

Condition Report 96-0047 was initiated by the foreign material coordinator to address these observations. Subsequent observations by the inspectors found foreign material control to be improved. The foreign material coordinator accounted for all items logged into the spent fuel pool area either by physical inventory or through discussions with the personnel responsible for bringing items into the area. The coordinator removed those items found in the area, but not logged in.

The foreign material exclusion zone in the reactor building pools was generally better controlled after the issuance of the condition report. Items were being tracked correctly; however, licensee personnel found several foreign material items in the dryer pool and the reactor cavity. Licensee personnel removed a 2-inch piece of yellow tape, a radiation material tag, and several other small items.

Later observations by the inspectors continued to identify poor work practices associated with foreign material exclusion control. The inspectors observed several instances of personnel carelessness (e.g., a rubber glove was discarded on the floor of the refueling platform bridge; a cotton glove liner was positioned so that it nearly fell into the dryer pool; a spool of cord on the refueling bridge was not secured; radiation protection personnel did not fail-safe radiacs; and maintenance personnel left unsecured rags and a bag on the fuel handling platform). Had it not been for the inspectors calling these instances to management attention, it was unlikely that the conditions would have been corrected. It was not until after the inspectors notified management personnel that actions were taken to either remove or secure the items. Sufficient time had existed for either peer workers or management personnel to have taken the initiative to stop or correct these poor work practices.

The failure to establish proper accountability of foreign material in the fuel building pool areas, and failure to properly control and account for or fail-safe, foreign material in the reactor building pool areas, is the second example of a violation of Technical Specification 5.4.1 (458/9601-01).

Licensee personnel undertook several corrective actions in an attempt to improve foreign material exclusion: signs posting housekeeping zones were moved to more visible locations and the signs were upgraded; personnel identified as having improperly logged items into or out of foreign material exclusion zones were counseled; and, new accountability sheets were instituted. As a result of continuing efforts and increased management oversight, foreign material exclusion practices did improve during the remaining portion of the outage.

At the start of the outage, the inspectors were informed that a foreign material exclusion assessment was performed by a team of 10 people consisting of both in-house personnel and industry personnel to review foreign material exclusion practices. Strengths and weaknesses were identified during the assessment, with the most significant weaknesses related to numerous problems of foreign material exclusion in and around the suppression pool. The suppression pool had been drained and cleaned during RF-4. A final inspection of the pool was performed by divers in September 1994. Subsequently, during RF-5, limited inspection of the suppression pool did not identify any foreign material problems.

The inspectors were informed that Procedure ADM-0081, Section 4.0, "Responsibilities," stated that the maintenance manager was responsible for establishing foreign material exclusion controls for the suppression pool areas and the drywell areas during shutdown periods consistent with the work activities planned. Attachment 8 to the procedure identified the suppression pool as a Housekeeping Zone III area. Section 3.1.3 described the cleanliness requirements applicable to a Zone III area, and also described when a written accountability record of personnel and material was not required. Section 8.2.9 provided steps to be used to minimize the potential for foreign material entering the suppression pool, such as using lanyards and tool bags. This section also required the initiation of a condition report whenever an item could not be accounted for. Additionally, Section 8.4.3 required the initiation of a condition report whenever foreign material was introduced into an open system or component and not immediately retrieved and that the foreign material exclusion monitor and discipline supervisor be notified.

The inspectors learned that the licensee had initiated numerous condition reports for items dropped into the suppression pool. Prior to work beginning for the outage, the licensee installed a vertical wire mesh along the walkways above the suppression pool and also placed herculite sheeting on metal grating on two levels of walkways above the suppression pool in an attempt to limit the introduction of foreign objects into the pool.

The inspectors questioned the responsible system engineer as to how he tracked items that fell into the suppression pool, and how he used this information to assure continued operability of the emergency core cooling systems that rely on the pool. The system engineer stated that when copies of condition reports were received, the item(s), including an estimated surface area, were entered into a data base (Suppression Pool Lost Item Log). A cumulative object surface area total was tracked to ensure total object surface area in the pool remained less than 1500 square inches, thus, maintaining system operability. The system engineer stated that 100 percent system flow could be maintained with suction strainer blockage of 50 percent, and that the worst case conditions assumed all items in the suppression pool were collected on one strainer. The system engineer stated that the items remained on the data base until removed from the pool, and that the pool was going to be cleaned during the current refueling outage.

The inspectors requested a copy of the suppression pool lost item log and observed that the log showed a total of 37 items, identified on 32 condition reports, as being apparently lost in the suppression pool. Based on the assigned surface area of the items, a total of 386 square inches was calculated (well below the designated 1500 square inches).

However, upon completion of the scheduled suppression pool cleaning on January 30, 1996, over 800 items were retrieved. These items included small fasteners (i.e., nuts, bolts, washers), tie wraps, weld rod, pens, plastic bags, rolls of tape, dosimeters, tools, a 12 X 16 X 0.250 inch rubber mat, a 1 X 4 foot piece of plywood, and an 8-foot section of scaffolding. This information caused the initiation of Condition Report 96-0428, dated January 30, 1996.

Since over 800 items had been retrieved from the pool and only 37 items were identified in the suppression pool log, the inspectors questioned the effectiveness of the administrative controls and the licensee's ability to positively demonstrate system operability under design basis accident conditions. Further, licensee personnel had attempted to match the retrieved items with associated condition reports. This resulted in about 33 items being matched with 29 condition reports, some of which had not been either provided to, or logged in, by the system engineer. This information clearly established that the vast majority of workers did not initiate condition reports when they either dropped items into the suppression pool, or lost accountability of items which had the potential for entering the suppression pool.

The failure of site personnel to initiate condition reports in accordance with Procedure ADM 0081 constituted the third example of a violation of Technical Specification 5.4.1 (458/9601-01).

During the Operational Safety Team Inspection (NRC Inspection Report 50-458/93-25), conducted October 25-29 and November 8-12, 1993, the inspectors identified excessive amounts of loose and unattended material inside containment. The licensee removed approximately 10 gallons of material from the suppression pool swell area. The inspectors were concerned that the loose material could potentially enter the suppression pool during an accident and clog the emergency core cooling system strainers. The licensee had initiated Condition Report 93-0753 to review operability.

Condition Report 93-0753 determined that operability had not been affected. However, the licensee did identify that housekeeping problems in containment was programmatic in nature. Numerous efforts were undertaken by the licensee to ensure all plant personnel, including contractors, were aware of requirements and restrictions of each housekeeping zone. In addition, emphasis was placed on personnel responsibilities for maintaining, correcting, and identifying housekeeping deficiencies. The condition report also identified the need for high level management attention to ensure requirements and responsibilities for housekeeping were properly implemented and enforced.

As identified during this outage, the number of unexpected items found in the suppression pool, as well as, the inadequate control of items brought into foreign material exclusion areas, demonstrated that previous corrective actions were ineffective and oversight was not sufficient to prevent recurrence of the violation.

Subsequent to the onsite inspection effort, numerous telephone conversations took place between the Office of Nuclear Reactor Regulation, NRC Region IV personnel, and the licensee representatives regarding operability of the suppression pool. In view of the fact that the licensee drained down and cleaned the suppression pool during this refueling outage, it was concluded that there was not an operability issue for the current cycle of operation.

2.11 Observation of Refueling Activities

The inspectors observed numerous fuel bundle moves during this outage, including the removal of the first bundle of irradiated fuel which occurred on January 13, 1996. New fuel movement was being conducted in accordance with Fuel Movement Plan FMP-STO-07-02 and irradiated fuel movement was conducted using Fuel Movement Plan FMP-STO-07-03. Procedure FHP-0001, "Control of Fuel Handling and Refueling Operations," Revision 14, was the governing procedure that provided instructions for the movement of fuel. During the inspectors' observations, all fuel movements were conducted in accordance with the procedures. However, the inspectors noticed numerous inconsistencies between the various crews/shifts, including differences in command and control, and communications between the senior reactor operators and fuel handling personnel. The inspectors pointed out these differences to licensee personnel.

During fuel handling observations on the refueling and fuel handling bridges, the inspectors did not observe any management or quality assurance oversight of fuel handling activities with the exception of reactor engineering management oversight during portions of the first day of irradiated fuel movement. Followup conversations with licensee personnel (quality assurance supervisor and outage manager) indicated that no independent management or quality assurance oversight on the refueling and fuel handling bridges was planned or performed, with the exception noted above. However, during a review of quality assurance surveillance reports, the inspectors noted that one quality assurance specialist did observe one fuel bundle movement (January 16) on the refueling bridge for approximately 30 minutes. The quality assurance supervisor stated that no quality assurance oversight was planned on the refueling bridge for this outage since they had not had any repetitive problems during the last outage. Subsequent to the inspectors' conversations with the quality assurance supervisor, additional quality assurance surveillances were conducted on January 18, 20, and 21, 1996, with no problems being identified.

Licensee personnel informed the inspectors that they considered the licensed senior reactor operator, mandated by Procedures FHP-0001 and REP-0029 to be assigned to the refueling floor prior to commencing core alterations and directly responsible for all core alterations, as fulfilling the management oversight function. It was the inspectors' contention that the senior reactor operator was an integral part of each refueling crew and was assigned specific supervisory responsibilities for the actions of the crew. Therefore, that person could not provide an independent and objective management assessment or oversight of his or her performance, or the crew's performance. This was borne out by the above discussion regarding numerous inconsistencies between the various crews/shifts, including differences in command and control and communications between the senior reactor operators and fuel handling personnel.

On January 15, the inspectors observed the refueling crew attempt to move a control rod double blade guide from its storage location in the separator pool to its designated core location. During the initial attempt to traverse through the "cattle chute" the blade guide impacted the base of the chute. The bridge operator overrode the upper limit switch in order to raise the blade guide higher. A second attempt was made, with similar results. The blade guide had been raised to a point where it was being dragged through the chute (i.e., sufficient contact to preclude maintaining a 90-degree perpendicularity). Rather than continue to drag the blade guide through the chute, the senior reactor operator halted the operation and revised the Fuel Movement Plan to return the blade guide to its starting storage position. The inspectors questioned reactor engineering personnel about this event since, during each outage, blade guides, which are longer than fuel bundles, are routinely moved into and out of the core. Reactor engineering personnel informed the inspectors that there were no available records that documented past similar conditions; however, the reactor engineering supervisor acknowledged the inspectors' concerns and agreed to take actions to administratively control blade guide movements to preclude possible damage. In a letter (RXE 96-008) to the Manager of Operations, the reactor engineering supervisor stated that Procedure FHP-0003, "Refuel Platform Operation," should be modified to add specific instructions for moving a full-blade guide through the cattle chute since it is approximately 4 inches longer than a fuel bundle. The letter continued with additional instructions to be followed by the bridge operator regarding the hoist and bridge overrides. The inspectors were not provided a date as to when the procedure would be revised.

On January 21, 1996, a refueling crew placed a fuel bundle in the wrong core location and Condition Report 96-0247 was initiated. Fuel Bundle YJ2151 was placed in core Cell Location 55-24 rather than Cell Location 55-26, as required by Step 1148 in Fuel Movement Plan FMP-COR-07-03. Section 2.15 in Procedure FHP-0001 required the reactor engineer and spotter to verify the correct bundle location prior to lowering the fuel bundle. Verification was to be performed by checking the X-Y coordinate display, visually confirming from the bridge through a comparison with the core map, and using the grapple camera to check the assembly number. The failure to properly verify the correct core location, which led to the incorrect placement of a fuel bundle, constituted a violation of Technical Specification 5.4.1 (458/9601-03).

The inspectors noted several factors involved with the placement error. Step 1148 required the loading of Fuel Bundle YJ2151 into Core Cell 55-26, which was an empty peripheral cell located adjacent to empty peripheral Core Cell 55-24. The bridge core map, which had not been maintained up-to-date, showed two peripheral, adjacent empty cell locations, 55-28 and 55-26. The refueling crew, rather than using core reference points (i.e., blade guides or counting cells from the periphery) or verifying the zone computer X-Y coordinates, referred to the core map. Since the fuel movement plan required insertion of the bundle into Core Cell 55-26, the crew located Cell 55-26 on the core map. They then placed the seating verification camera into the adjacent cell designated on the map as 55-28. However, since the map was not up-to-date, the crew was unaware that the cells designated on the map as 55-26 and 55-28 were really core cells 55-24 and 55-26, respectively. The camera then became the reference point based on the incorrect core map and the fuel bundle was placed into Map Cell 55-26, rather than Core Cell 55-26.

After detecting the error, the licensee immediately suspended fuel movement operations. Cognizant licensee personnel conducted a root-cause analysis, which was documented in a report dated January 25, 1996. The licensee determined that there was no safety significance associated with this incident since the single fuel misload error was well within the bounding shutdown margin analysis (up to four concurrent misloads). The root-cause analysis report determined that there were five causes for the error (personnel, procedure, communications, human engineering, and lack of administrative controls). With respect to the lack of administrative controls, Section 6.0 in Procedure REP-0029, "Fuel Movement," Revision 2B, required that, prior to the start of fuel movement in the reactor core, an enlarged core map be placed on the refuel bridge for use by the fuel handler/spotter. As noted above, Procedure FHP-0001 required visual confirmation through a comparison with the core map. However, neither of these procedures provided guidance regarding limits or expectations as to how the core map was to be used or maintained.

The inspectors reviewed previous fuel movement mishaps identified at River Bend Station.

- During Refueling Outage RF-3, a violation was identified with respect to the identification of five misoriented fuel bundles. The misorientation was attributed to refueling personnel error, even though performance and verification signatures were required to document each step of the fuel movement plan.

- During Refueling Outage RF-5, a fuel bundle was removed from its core location and moved to its new location, which was already occupied by another fuel bundle. This error was attributed to an accidental page turning of the fuel movement plan, compounded by the operators' failure to verify the next correct sequential step.

- On December 5, 1995, a fuel bundle was incorrectly loaded in the spent fuel pool in December 5, 1995 (See Section 2.4).

Personnel error appears to be the common link between these incidents, and increased management involvement is warranted.

2.12 Heavy Loads

The inspectors reviewed NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants," and compared the guidance contained within the NUREG to the specification contained in the Updated Safety Analysis Report (USAR), River Bend Safety Evaluation Report, and plant procedures. It was determined that the licensee was maintaining and operating the containment polar crane in accordance with all requirements. While observing refueling operations, the inspectors noted that the polar crane load block passed over the open vessel. Further reviews identified that this was evaluated as being acceptable in the NRC technical evaluation report for the River Bend polar crane due to certain design features such as dual load brakes and redundant and independent up-hoist interlocks. However, during conversations with a mechanical engineering manager, he indicated that this was not a good work practice and he would submit a procedure change to give guidance on avoiding moving the load block over the open vessel.

2.13 Spent Fuel Pool and Reactor Cavity Pneumatic Gate Seals

USAR Chapter 9, "Auxiliary Systems," described the design of equipment for the storage of new and irradiated fuel, the containment building fuel pool, fuel building fuel pool, watertight gates, and other systems/subsystems used to support storage of fuel. Section 9.1.2.2 stated that the gates were watertight, and classified them as Seismic Category 1 structures designed to withstand all loads as defined in Section 3.8 of the USAR (includes seismic loading).

USAR Section 3.2.3, "Quality Assurance," discussed structures, systems, and components whose safety functions require conformance to 10 CFR Part 50, Appendix B, and were summarized in USAR Tables 3.2-2 and 3.2-3. Table 3.2-3, "Summary of Safety Class Design Requirements," lists 10 CFR Part 50, Appendix B as being required for Seismic Category 1 equipment.

While the USAR described the gates as being watertight, no separate discussion on pneumatic gate seals was found; consequently, the inspectors reviewed Procurement Specification 92D72742, Specifications 219.721 and 219.722, applicable piping and instrumentation diagrams, and drawings. Drawing 12210-EV-6E-6, "Gate Details," gave detailed design information on seal and gate construction and was identified as "Nuclear Safety Related QA CAT I." Specification 219.721, "Shop Fabrication of Liners for the Containment Refueling Pools and the Fuel Building Pools," described the gates as "watertight gates." The specification also listed the inflatable seal in the material specifications. In addition, the specification stated that the equipment provided in the specification were basic components. Basic components were defined in 10 CFR Part 50 as safety-related. Specification 219.722, "Field Erection of Refueling Cavity and Fuel Pool Liners, Embedments, and Ancillary Equipment," contained pneumatic gate seal testing requirements, which were to demonstrate leak tightness of gates and seals and also contained provisions to correct the cause of any leakage.

Based upon the information contained in the USAR, and the construction/fabrication drawings and specifications, the inspectors determined that the licensee had classified the gates and seals as safety-related; thus, requirements of 10 CFR Part 50, Appendix B were applicable to these components.

During this refueling outage, the cavity pneumatic gate seals were maintained inflated by the nonsafety-related instrument air system with nitrogen backup supplied by nitrogen bottles connected through temporary 0.250 inch polyvinyl chloride tubing. Procedure GMP-0102, "Reactor Vessel Disassembly," Revision 3, Step 8.1.14 required that backup nitrogen bottles be connected to the pneumatic gate seals for emergency use. Because the pneumatic gate seals were classified as safety-related and the pneumatic gate seal instrument air supply system was nonsafety-related, the emergency nitrogen backup system constituted the safety-related pneumatic gate seal air supply system. The nitrogen backup supply system to these components had no procedural guidance or equipment requirements other than recommended regulator pressure settings

(Procedure OSP-0029, "Auxiliary, Reactor, and Fuel Building Rounds," Revision 3A). Licensee personnel indicated that no procedures or work instructions were used to install the backup system, and that they had relied on skill-of-the-craft.

Licensee personnel did not know how the nitrogen pressurization was tied into the pneumatic gate seals; therefore, the inspectors accompanied design engineers on a field walkdown in order to establish the actual installation of the nitrogen pressurization supply. The nitrogen supply consisted of two bottles hooked up in parallel with one supplying nitrogen through a regulator set at approximately 20-25 psig. This backup nitrogen supply not only supplied backup air to the pneumatic gate seals but also supplied backup air to the siphon plug for the separator pool and four main steam line plugs. The lines from the nitrogen bottles were teed into a common line which later split into two lines. One line supplied nitrogen to two main steam line plugs and the other line tied into a temporary manifold that supplied the other components described above. The tubing had no permanent support but was taped to electrical conduits, hand railings, and the floor. It was routed in high traffic areas and was susceptible to damage. The nitrogen temporary manifold had one nitrogen supply isolation valve and three supply valves for the dryer to cavity pneumatic gate seal, siphon plug, and two main steam line plugs. The valves were labeled with duct tape and grease pencil. Several of the valves were taped open and had "open" written on the tape.

The inspectors questioned two senior reactor operators (a control room supervisor and a shift superintendent) and two senior nuclear equipment operators concerning the source of normal air supply to the pneumatic gate seals. Only one individual, a senior nuclear equipment operator, was able to identify that the pneumatic gate seals were supplied by instrument air. The other operators stated that the pneumatic gate seals were supplied by service air.

The licensee's review of NRC Information Notice 88-92 and Supplement 1, along with their review of the 1993 drain down event at Comanche Peak Unit 1 (SEN 115 Event 5) provided the licensee an opportunity to identify similar problems at their facility; however, the reviews were limited, lacked a questioning attitude, and failed to review plant drawings, procedures, USAR, and design documents to determine the actual design, as-built conditions, and use of the seal and support systems.

Information Notice 88-92 and Supplement 1 described a loss of spent fuel pool level caused by leaks on the air supply for the pneumatic gate seals which occurred at isolation valve packings, check valves, and at quick-connect air fittings that were not in accordance with design drawings. The licensee's evaluation of these notices focused on differences in design features and administrative controls which would mitigate the consequences of pneumatic gate seal leakage; regardless, licensee personnel failed to consider those items that were similar and made several initial poor assumptions concerning the performance of their seals on loss of air. The licensee also used quick connect fittings that were not in accordance with Drawing PID-12-1C, "Engineering P&I Diagram System 122 Air-Instrument," and a nitrogen backup supply that had no design drawings.

Since 1984, licensee personnel have generated numerous condition reports and engineering documents concerning the pneumatic gate seals. Several of these documents recognized potential problems with service life of the seals, inspections, preventive maintenance, and several other problems including an event involving partial draindown of the dryer pool. However, none of the documents described below caused the licensee to take any corrective actions or implement inspection/maintenance activities (an exception was that limited corrective maintenance was performed in response to Condition Report 90-1128).

- Nonconformance and Disposition Report 5185, initiated February 16, 1984, identified that the reactor cavity pneumatic gate (1FNR*Gate 1) and upper fuel transfer pool pneumatic gate (1FNR*Gate 2) leaked a small amount of water. This condition was accepted in spite of the fact that the USAR described the gates as being watertight. Licensing personnel stated that through interviews with operators and maintenance personnel, it was determined that no pneumatic gate seal leakage existed for this outage or the previous outage.

- Condition Report 87-1277, initiated on October 12, 1987, was written to address GE SIL 86. The condition report recommended that the pool pneumatic gate seals should be inspected prior to each use and after the seals are replaced; a minimum service life for each seal should be specified; preventive maintenance procedures should be generated for the seals; and, a minimum and maximum air pressure for the seals should be specified. None of these actions were implemented at the time; however, they were deferred for further evaluation under Engineering Evaluation and Assistance Request 88-E0029.

- Engineering Evaluation and Assistance Request 88-E0029 evaluated the above actions and recommended criteria to satisfy each of the recommendations contained in Condition Report 87-1277. No actions were taken to implement any of these recommendations.

- Engineering Calculation G13.18.014.003*02, completed July 18, 1990, was initiated to determine the pneumatic gate seal life based on radiation exposure. The description of the problem recognized that the seals had a limited radiation exposure lifetime. The calculation established a radiation exposure to the seals during fuel movement for one refueling outage; however, the calculation failed to take into account radiation exposure to the seals from fuel stored in the pools and it did not determine a lifetime exposure.

- Condition Report 90-1128, initiated November 16, 1990, identified a slow partial draindown of the dryer pool to the reactor cavity. The dryer pool lost approximately 2 feet of water when the pneumatic gate seal inflate/deflate valve drifted to the deflate position. Seal pressure decreased slightly thereby allowing water to leak past the seal. This valve, as well as, the three other inflate/deflate valves for the other pneumatic gate seals were identified to have worn-out detents (these positively hold the valve open). The valves were replaced and the condition report recommended preventive maintenance for the valves; nonetheless, no preventive maintenance was done or developed for any of the valves.

- Engineering Evaluation and Assistance Request 91-R0129, initiated December 31, 1991, identified conditions where water level above fuel placed in the containment fuel storage racks could drop below the Technical Specification required level of 23 feet above irradiated fuel. Water level could decrease on failure of the various containment pneumatic gate seals and water level above the fuel could be between 9 and 1.5 feet. The licensee determined that conditions required for this to occur would be unlikely and no further evaluation or followup was completed.

Based on documents reviewed, the inspectors determined that the licensee had numerous opportunities to identify the same concerns expressed by the inspectors; however, licensee personnel failed to understand the licensing basis previously established for the pneumatic gate seals, and the important role they played in protecting nuclear fuel.

After the inspectors communicated their concerns regarding the pneumatic gate seals, engineering personnel initiated Condition Report 96-0143 to address a number of issues, including immediate operability, qualified service life, maintenance requirements, and safety classification. The analysis performed by the licensee found the seals to be operable. A service life of 10 years was established based upon conservative radiation estimates. The licensee determined that only Part (iii) (offsite exposure in relation to 10 CFR Part 100) of the definition of a basic component in 10 CFR Part 50 potentially applied to the pneumatic gate seals. The licensee concluded that the seals were not required to meet the requirements for offsite dose limits contained in 10 CFR Part 100, and thus the pneumatic gate seals could be classified as nonsafety-related.

In order to determine the appropriateness of the pneumatic gate seal reclassification, NRC will conduct a review of Condition Report 96-0143 and its associated engineering analyses. Performance of this review is considered an inspection followup item (458/9601-04).

Since at least the beginning of commercial operations, the licensee had classified the pool pneumatic gates and seals as safety-related. Due to the concerns raised by the inspectors, licensee personnel determined in January 1996 that reclassification of the pneumatic gate seals to a nonsafety-related status was appropriate. However, the inspectors considered this long-term oversight to be a breakdown regarding design control. Even though considerable industry information was available, and a number of internal reviews had occurred in which recommendations were made, appropriate measures (i.e., procedures, instructions, or drawings), commensurate with the safety-related classification designated by the licensee, were not established to assure that the pneumatic gate seals would be treated as safety-related components. This failure to establish appropriate design control measures constituted a violation of Criterion III of Appendix B to 10 CFR Part 50 (458/9601-05).

3 INSERVICE INSPECTION (73753)

The purpose of this portion of the inspection was to determine whether the inservice inspection, repair, and replacement of Class 1, 2, and 3 pressure retaining components were performed in accordance with Technical Specifications, the applicable ASME Code, correspondence between the Office of Nuclear Reactor Regulation and the licensee concerning relief requests, and requirements imposed by NRC/industry initiatives.

The inspectors observed limited nondestructive testing. No discrepancies were identified during the test equipment calibration or during the performance of the examinations. The inspectors also reviewed the qualifications and certifications of the inspection personnel performing the examinations and did not identify any discrepancies.

The inspectors noted that site examination personnel were augmented with personnel from other sites operated by the licensee. This permitted better control over the contracted examination personnel who performed the examinations, by providing essentially full-time coverage of the work tasks.

The inspectors concluded that the observed inservice inspections were performed in accordance with the requirements.

4 FOLLOWUP - MAINTENANCE (92902)

4.1 Failure of a Dissimilar Metal Weld Joint

At approximately 6:30 a.m. on January 5, 1996, with the plant in Mode 3 in preparation for Refueling Outage RF-6, a building operator noticed that the minimum flow/test return piping of Residual Heat Removal "B" system loop was missing from a level approximately 1 foot above the suppression pool. This was observed while the upper pool gravity drain for flushing Residual Heat Removal "B" system loop was in progress. The main control room personnel were notified and the upper pool flush was secured. In accordance with Technical Specifications, the containment penetration was isolated and the associated engineered safety feature systems were declared inoperable. Since the allowed out-of-service times were less limiting than the scheduled entry into Mode 4 (within 12 hours), no change to the scheduled shutdown was necessary. In Mode 4, containment isolation was not required and the remaining engineered safety features met Technical Specification requirements. The licensee initiated Condition Report 96-015 to document the condition and request engineering resolution concerning system and containment operability.

Divers were able to view the broken pipe, submerged pipe support, and surrounding area and concluded that no damage to the pool or pipe support resulted from the broken pipe. The divers recovered the broken pipe on January 6, 1996, and a detailed inspection was performed by design engineering. It was determined that the failure occurred at a dissimilar metal weld joint (stainless steel and carbon steel). Samples from both sides of the fracture were subjected to a thorough metallurgical examination by an independent testing laboratory. The results of the metallurgical examination showed the existence of lack of fusion in excess of 80 percent on the carbon steel side of the weld joint. The remaining 20 percent of the carbon steel side of the weld joint (consisting of two locations) showed fatigue fracture in both locations. The laboratory concluded that corrosion was not a factor and that failure was the result of a lack of fusion and fatigue of those areas that were fused.

The exact time of failure could not be established; however, the pipe was intact on December 19, 1995. This was documented by the system engineer in a memorandum which was used to record residual heat removal system noise levels while Residual Heat Removal "B" system was running in suppression pool cooling mode on that date.

The inspectors were informed that the failed dissimilar metal weld was a shop weld that had been fabricated in 1982 by B. F. Shaw Co. The inspectors reviewed the applicable manufacturing records pertaining to this weld (SW-002 in Fab Mark 1-RHS-9-2-029), including the shop traveler, welding procedure specification, procedure qualification record, welding material certified material test report, and the radiographs of the weld joint. This review concluded that the proper welding process (gas tungsten arc welding) and welding filler material (ER309) had been used. The inspectors further concluded that the most likely cause of the lack of fusion was improper weld technique (i.e., inadequate amperage or excessive travel speed), which led to a lower than required heat input. This was subsequently supported by the results of the metallurgical examination performed by the independent testing laboratory.

The B. F. Shaw Co. manufacturing records showed that an acceptable fitup inspection was performed on March 25, 1982, the weld was completed on April 7, 1982, and subsequent radiography was performed on April 20, 1982. The radiographic inspection report stated that the weld was acceptable, despite misalignment being evident in all three radiographs of the weld. The radiographic film was also reviewed and accepted by the authorized nuclear inspector and the customer (Stone & Webster Engineering Corp.) representative. With respect to the noted misalignment, the inspectors observed a note on the applicable B. F. Shaw drawing which stated that the pipe ends to be welded (SW-002) were to be counter bored and the finished weld was to be ground.

This was in consideration of future inservice inspection requirements. This, taken in conjunction with the acceptable fitup inspection performed prior to welding, should have raised questions regarding the validity of the misalignment interpretation that was made.

The inspectors reviewed the B. F. Shaw radiographs of SW-002, and observed a consistent, unsharp, linear type indication that extended 100 percent of the weld length. The indication was one that should have warranted a visual examination of the welded joint in order to support the misalignment interpretation.

The licensee's evaluation of the pipe break contained in Root Cause Analysis Report SERT 96-01 dated January 16, 1996, concluded that safety significance was not a concern, and that the offsite dose consequences of the failed pipe weld were bounded by the analysis for engineered safety feature systems leakage. Therefore, there would not be an increase in offsite dose due to the failed pipe weld.

The Residual Heat Removal "B" minimum flow/test return line was restored to meet the original design requirements. All repair welding was completed on January 7, 1996. Subsequent nondestructive examinations (i.e., visual, liquid penetrant, and radiography) were completed on January 8, 1996. The inspectors reviewed completed Manufacturing Work Order 303278, which specified the welding and nondestructive examination activities to be used during repair of the welded joint, and verified that the welding material specification and welding filler material were appropriate and that the welders were qualified. The inspectors also reviewed Radiographic Inspection Report 96 [illegible] 20400 and viewed the radiographs of the completed weld joint. The radiographic inspection report showed that the weld was acceptable, and was consistent with the inspectors' interpretation of the radiographs.

Since Residual Heat Removal "A" and "B" system loops had been identified as having excessive noise and vibration levels, the licensee performed an ultrasonic examination on the corresponding dissimilar metal weld joint in the Residual Heat Removal "A" system loop. No reportable indications were observed.

With respect to possible generic implications regarding the failure of dissimilar metal welds, the licensee established a sampling population of 14 welds based on the following criteria: dissimilar metal welds made by B. F. Shaw Co.; welds not subjected to high system pressures or in open ended pipes; and, safety-related welds that are not in the inservice inspection program (exempted by an NRC-approved request for relief). The intent was to review the original radiographs to assure that any existing similar flaws or indications were properly identified and reported. A qualified nondestructive examination Level III examiner evaluated the original radiographs and did not identify any rejectable indications. The inspectors also reviewed the radiographs of the 14 sample population and concluded that rejectable indications were not evident.

The inspectors concluded that the failed weld was due to lack of fusion and fatigue of the fused areas. The lack of fusion was the result of improper weld technique. The radiographic inspection report stated that misalignment was evident in all three radiographs of the weld. Since weld joint fitup of the counter bored pipe ends was inspected and accepted, questions should have been raised regarding the validity of the misalignment interpretation that was made, thus, prompting a closer visual examination of the welded joint.

The inspectors considered the licensee's inspection plan and criteria for establishing sample selection appropriate.

4.2 Previous Inspection Findings

(Closed) Violation 458/9419-01: Failure to Follow a Maintenance Work Order

The inspectors verified the corrective actions described in the licensee's response letter, dated December 9, 1994, to be reasonable and complete. No similar problems were identified.

4.3 Licensee Event Report Followup

(Closed) Licensee Event Report 50-458/94-17: Seven Testable Check Valves not Properly Tested in Accordance With ASME Section XI Requirements

On June 17, 1994, seven testable check valves were discovered by licensee personnel to have not been tested in accordance with ASME Section XI requirements. The applicable surveillance test procedures did not contain requirements to establish, determine, or note the force or torque required to reposition the testable check valves. On November 20, 1994, Revision 1 to this event report was sent to the NRC. In Revision 1, the categorization of the event report submittal was changed from 10 CFR 50.73(a)(2)(ii) to voluntary.

The root cause was stated in the event report to have been an inadequate procedure which resulted from the failure of licensee personnel to recognize the discrepancy between the test method described in ASME Section XI and the surveillance test procedures. The inspectors reviewed the immediate corrective actions taken and found them to be acceptable. Additional corrective actions to upgrade the inservice test program were detailed in the event report; however, the inspectors did not review those actions.

This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

ATTACHMENT 1

PERSONS CONTACTED AND EXIT MEETING

1 PERSONS CONTACTED

1.1 Licensee Personnel

*K. Aitken, Security Coordinator
*R. Alexander, Manager, Project Management
*D. Dormady, Manager, System Engineering
*J. Fowler, Supervisor, Quality Assurance
*K. Giadrosich, Superintendent, Maintenance
*T. Hildebrandt, Outage Manager
*J. Holmes, Superintendent, Chemistry
*M. Krupa, Manager, Operations
*M. Laris, Senior Reactor Engineer
*T. Leonard, Director, Engineering
*L. Lewis, Manager, Training
*D. Lorfing, Supervisor, Licensing
*R. McAdams, Senior Licensing Engineer
*P. Schlesinger, Technical Support Coordinator
*G. Scronce, Fuel Fabrication Coordinator
*J. Venable, Manager, Licensing
*L. Woods, Superintendent, Operations
*G. Zinke, Manager, Quality Assurance

1.2 NRC Personnel

*D. A. Powers, Chief, Maintenance Branch
*W. F. Smith, Senior Resident Inspector

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

* Denotes personnel that attended the exit meeting held on March 5, 1996.

2 REVIEW OF UFSAR COMMITMENTS

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the UFSAR description. While performing the inspections discussed in this report during the inspection period February 1-2, 1996, the inspectors reviewed the applicable sections of the UFSAR that related to the areas inspected. The following inconsistency was noted between the wording of the UFSAR and the plant practices, procedures and/or parameters observed by the inspectors: Spent fuel pool and reactor cavity pneumatic gate seals were not controlled as safety-related equipment (see Section 2.13).

3 EXIT MEETING

An exit meeting was conducted on March 5, 1996. During this meeting, the inspector reviewed the scope and findings of the report. The licensee's representatives, with one exception, did not express a position on the inspection findings documented in this report. The licensee's representatives presented their position with respect to the inspectors' finding regarding a lack of management oversight of refueling activities. As discussed in Section 2.11 above, the licensee's representatives considered the use of a senior reactor operator as part of the refueling crew to fulfill the management oversight function. The inspector acknowledged the licensee's position, but noted that the independent and objective aims of management oversight were not served by the senior reactor operator being part of the refueling crew. The licensee's representatives did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 2

DOCUMENTS REVIEWED

Information Notices and Bulletins

IN 81-23 - Fuel Assembly Damaged Due to Improper Positioning Of Handling Equipment

IN 84-93 - Potential For Loss Of Water From The Refueling Cavity

IN 86-58 - Dropped Fuel Assembly

IN 88-21 - Inadvertent Criticality Events at Oskarsham and at US Nuclear Power Plants

IN 88-65 - Inadvertent Drainages of Spent Fuel Pools

IN 88-92 and Supplement 1 - Potential for Spent Fuel Pool Draindown

IN 89-51 and Supplement 1. Potential Loss of Required Shutdown Margin During Refueling Operations

IN 90-02 - Potential Degradation of Secondary Containment

IN 92-25 - Potential Weakness In Licensee Procedures For a Loss of the Refueling Cavity Water

IN 92-39 - Unplanned Return to Criticality During Reactor Shutdown

IN 93-70 - Degradation of Boraflex Absorber Coupons

IN 93-82 - Recent Fuel and Core Performance Problems In Operating Reactors

IN 94-13 and Supplements 1 and 2 - Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment

IN 94-64 - Reactivity Insertion Transient and Accident Limits For High Burnup Fuel

IN 95-03 - Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition

Bulletin 89-03 - Potential Loss Of Required Shutdown Margin During Refueling Operations (Affected PWRs Only. While this addressed PWRs, some information was applicable to BWRs. For example, the intermediate storage of fuel bundles during fuel movement and the increased fuel enrichment.)

Procedures

REP-0006, "Reactor Engineering Qualification Matrix," Revision 2

MSP-0024, "Qualifications and Training for Load Handling," Revision 3

MSP-0009, "Qualification of Maintenance Personnel," Revision 13

AOP-0027, "Fuel Handling Mishaps," Revision 9

STP-050-3601, "Shutdown Margin Demonstration," Revision 9A

REP-0029, "Fuel Movement," Revision 2

FHP-0001, "Control of Fuel Handling and Refueling Operations," Revision 12

REP-0012, "Criticality Rules," Revision 1A

REP-0010, "SNM Accounting and Control," Revision 10

FMP-COR-07-03, "Fuel Movement Plan"

Calculation Report

NEAD-SR-95/082.RO, "River Bend Station RF06 Refueling SDM Calculation"

Condition Reports (CR)

91-0041
91-0041A
93-0213
92-0231

Letters, Records, and Memoranda

Entergy Letter RXE 96-003, "RF-6 Fuel Handling Training," January 7, 1996

Entergy Letter DED 96-03, "Fuel Handling Training Summary," January 7, 1996

Entergy Letter DED 96-02, "Fuel Handling Training Matrix," January 7, 1996, and Revision 1 dated January 9, 1996

OJT Card SP-02J-1 TR Number 0227, "Inclined Fuel Transfer Tube on the Job Training and Evaluation," December 11, 1995

OJT Card SP-01-01J Rev.1 TR Number 0227 SP001, "Fuel Handling Platforms on the Job Training and Evaluation," November 27, 1995

OJT Card HLO-06J-0 TR Number T0339J, "Instant Senior Reactor Operator on the Job Training and Evaluation," February 13, 1995

Memorandum CEO-95/00306, "RBS RF06 Shutdown Margin Analysis," October 12, 1995