ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-458/96-04

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: April 7 through May 18, 1996

Inspectors: W. F. Smith, Senior Resident Inspector
D. L. Proulx, Resident Inspector
M. E. Murphy, Reactor Engineer (Examiner), Division of Reactor Safety

Approved: Gregory A. Pick
P. H. Harrell, Chief, Project Branch D — Date 6/6/96

Inspection Summary

Areas Inspected: Routine, announced inspection of plant operations, maintenance and surveillance observations, onsite engineering, plant support activities, followup-maintenance, onsite review of licensee event reports (LERs), and review of Updated Safety Analysis Report (USAR) commitments.

Results:

Plant Operations

- Control room operators demonstrated good performance in the routine operation of the plant. Licensee management was frequently observed monitoring control room activities (Section 2.1).

- Past corrective actions taken by the licensee were not sufficiently effective to prevent recurring deficiencies in the maintenance of the equipment qualification configuration of Rosemount transmitters. A violation of 10 CFR 50.49(a) was identified because two safety-related transmitters did not have the required stainless steel plugs installed in the spare conduit openings (Section 2.2).

Maintenance

- On-line maintenance performed on the Division I control room air conditioning (CRAC) system was not well planned, resulting in delays that caused the safety-related system to be out of service longer than planned. Precautions to mitigate the possibility of losing the redundant division were acceptable (Section 4).

- The procedure used for the performance of inservice testing of the residual heat removal line fill pump was inadequate. This caused extra work and repeated testing in a high radiation area, resulting in unnecessary personnel radiation exposure. A violation of Technical Specification (TS) 5.4.1 was identified (Section 5).

- On two occasions, the licensee performed incomplete investigations into the test failures of safety-related motor-operated valves. The licensee initiated an inaccurate condition report (CR) and then later concluded, in error, that as-left test data was as-found test data. Although these errors ultimately did not significantly affect safety, they indicated that deficiencies existed in the investigation process when senior management was not involved. Quality Assurance (QA) personnel performed an excellent followup of this issue by identifying the improper evaluations (Section 8).

Engineering

- Inservice test data was not documented in accordance with Procedure STP-052-6301, "Control Rod Drive Valve Operability Test," Revision 6. Data was recorded by the operators as "acceptable" when, in fact, the data was in the "Alert" range, requiring more frequent testing. This was an example of a violation of TS 5.5.6 (Section 6).

- Inservice test engineers failed to review test data within the 30 days required by procedure. As a result, the reference values for the stroke times of two safety-related valves were not rebaselined nor was the surveillance frequency increased, as required by the Inservice Testing (IST) program. This was an example of a violation of TS 5.5.6 (Section 6).

- Engineering personnel did not maintain the IST program consistent with industry standards. The program failed to include justifications to defer testing to cold shutdown conditions and had out of date relief requests. Licensee management had previously signed off the Long-Term Performance Improvement Plan (LTPIP) action item for improvement of the IST program in September 1995; however, an effectiveness review revealed numerous deficiencies. This issue is an inspection followup item (Section 6).

Plant Support

- Housekeeping was observed to be excellent throughout the plant. A notable improvement was the high pressure core spray pump room, which was cleaned and painted during this inspection period (Section 2.1).

- A radiation worker entered the radiologically controlled area without a direct reading dosimeter. The individual did not receive any radiation dose prior to the Radiation Protection Department discovering the problem. Three examples of a violation concerning entrance into the radiologically controlled area without a direct reading dosimeter were cited in NRC Inspection Report 50-458/96-03, but the corrective actions for the previous examples were not yet completed. The inspectors identified this as another example of a previous violation (458/9603-02). The licensee committed to include this example in their response to the previous violation (Section 7).

Summary of Inspection Findings:

New Items

- Violation 458/9604-01: Environmental qualification of Rosemount transmitters not maintained (Section 2.2).

- Violation 458/9604-02: Failure to maintain an adequate IST procedure (Section 4.2).

- Violation 458/9604-03: Two examples of failure to follow IST procedures (Section 6).

- Inspection Followup Item 458/9604-04: Followup inspection of the IST program (Section 6).

Closed Items

- Inspection Followup Item 458/9602-05 (Section 8).

- LER 458/94-028 (Section 9.1).

- LER 458/94-030 (Section 9.2).

- LER 458/94-032 (Section 9.3).

- LER 458/95-001 (Section 9.4).

- LER 458/95-012 (Section 9.5).

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

At the beginning of this inspection period, the plant was at 100 percent power. On April 20, 1996, power was reduced to 70 percent to perform a reactor control rod configuration change. By April 21, power was restored to 100 percent, where it remained through the end of this inspection period.

2 PLANT OPERATIONS (71707)

2.1 Routine Observations

During visits to the control room, the inspectors noted that operators were attentive to their duties and operated the plant in accordance with the appropriate procedures. Annunciator response procedures were properly referenced and three-way communications were appropriately employed upon receipt of alarms. Operators carefully logged plant manipulations and system lineups, as required. Licensee management was frequently observed monitoring control room activities.

The inspectors toured accessible areas of the plant and concluded, based on inspections of safety-related valves, breakers, and equipment material conditions, that the plant was being maintained within its design basis and in accordance with requirements. Housekeeping was excellent throughout the plant. One notable improvement was the high pressure core spray pump room, which was cleaned and painted during this inspection period.

2.2 Environmental Qualification of Rosemount Transmitters

During a tour of primary containment on April 5, 1996, the inspectors identified that inservice Rosemount Transmitter 1MSS-PDT-107 had the factory-installed plastic plug in a spare conduit port instead of the stainless steel plug recommended by the vendor. The inspectors performed the evaluation because of sensitivity to recent, similar concerns identified at another nuclear facility. The stainless steel plug is required to maintain equipment qualification. The licensee initiated CR 96-0756 to enter this issue into the corrective action program.

Engineering determined that this transmitter was nonsafety-related equipment that provided input to the control room annunciator for low flow of air to the main steam isolation valve pilot solenoids. However, the licensee considered this transmitter important to plant reliability. Because the transmitter was installed in the reactor containment, which is a potentially harsh environment, the stainless steel plug should be installed. The licensee determined that this problem had existed since initial plant startup.

The inspectors inspected 50 additional Rosemount transmitters to ensure that configuration control (i.e, environmental qualification) was maintained for safety-related applications. On April 9, the inspectors identified that Rosemount Transmitter 1DFR*LT-136, reactor core insolation cooling (RCIC) pump room level, did not have a stainless steel plug installed in the spare conduit port. The inspectors informed the shift superintendent, who initiated CR 96-0770. After licensee management became aware that the inspectors found the Rosemount transmitter improperly installed, 150 of the 265 safety-related Rosemount transmitters were inspected for proper installation. The remaining 115 transmitters had been inspected in 1994, as discussed below. The inspection also identified that Rosemount Transmitter 1DFR*LT-137, Residual Heat Removed (RHR) Pump C room level, did not have a stainless steel plug installed in the spare conduit port. The licensee initiated CR 96-0771 to address this issue.

Rosemount Transmitters 1DFR*LT-136 and -137 are safety-related equipment and are required to be environmentally qualified. 10 CFR 50.49(a) requires, in part, licensees to establish programs to environmentally qualify equipment important to safety. Because the stainless steel conduit plugs were not installed in these transmitters, the licensee did not maintain the environmental qualification of Transmitters 1DFR*LT-136 and -137. This is a violation of 10 CFR 50.49(a) (458/9604-01).

The licensee corrected the specific deficiencies found and initiated a Maintenance Action Item (MAI) to inspect all Rosemount transmitters for all of the correct installation and configuration requirements to ensure no further deficiencies existed. The licensee indicated that these inspections will take several weeks to complete.

Section 6.3.1.1.3 of the USAR states, in part, that each emergency core cooling system (ECCS) equipment room is provided with a safety-related level transmitter, wall mounted near the floor, to detect a rising water level condition that annunciates an alarm and provides level indication in the control room. These transmitters are referenced in Emergency Operating Procedure EOP-3, "Secondary Containment Control," Revision 10, which requires, in part, entry into this emergency procedure when levels in the ECCS and RCIC rooms exceed 40 percent of scale. Operators must then take action to mitigate a flooding event or break in an ECCS or RCIC pipe. Operators have other means of detecting flooding or a break in the ECCS or RCIC rooms (such as area temperature monitors and/or the initiation of the sump pumps), which mitigates the potential safety consequence of this issue.

The inspectors noted that the licensee experienced previous problems with the environmental qualification of Rosemount transmitters. In December 1994, a maintenance technician identified that a safety-related Rosemount transmitter did not have a stainless steel plug installed in the spare conduit port. The licensee installed the plug, inspected a sample of 115 out of the 265 safety-related Rosemount transmitters for proper installation and found no other problems. The licensee stated that this sampling size was adequate to ensure no other problems existed. In addition, the corrective actions were

focused on training only instrument and control technicians to be sensitive to the proper configuration of Rosemount transmitters. In March 1995, the licensee identified another nonsafety-related Rosemount transmitter that was missing the stainless steel plug, but performed no further inspections.

The inspectors concluded that the previous corrective actions were too narrowly focused to prevent a recurrence. Specifically, the licensee did not perform a 100 percent sample and failed to train/sensitize other work groups to the potential for inoperable transmitters because of environmental concerns related to their installed configuration (i.e., plastic versus stainless steel plugs).

3 MAINTENANCE OBSERVATIONS (62703)

From April 22-26, 1996, the inspectors witnessed portions of work associated with the Division I control room air conditioning (CRAC) outage. The work involved several maintenance items and modifications that converted certain power-operated valves to manual valves. A major item included replacement of the fan hub for Air Handling Unit 1HVC-AHU-1A.

Division I of the CRAC system consists of Chillers 1HVC-CHL-1A and -1C, which serve Air Handling Unit 1HVC-AHU-1A. CRAC provides cooling to the control room and Chillers A and C also provide cooling to the switchgear for all of the Division I equipment. TS 3.7.3 allows one division of CRAC to be inoperable for 30 days, otherwise the plant must be shut down.

The inspectors questioned the licensee as to why this maintenance was being performed on-line versus the past or upcoming refueling outage, given the risk significance of this maintenance. Loss of the Division I Chillers could lead to multiple failures of Division I safety-related systems.

To partially compensate for the risk increase, the licensee prohibited work on Division II equipment and treated the Division II CRAC system and Chillers B and D as protected systems. The licensee placed signs on the entrance to the Division II chiller rooms that indicated entry was prohibited. The licensee stated that they had received the new fan hubs for the air handling units in December 1995, but the documentation certifying the quality of these hubs was not provided until late in Refueling Outage 6. Therefore, the licensee decided not to perform this maintenance during the refueling outage.

The inspectors questioned whether the documentation was expedited. The licensee stated that they were aggressive in their attempts to obtain the documentation, but the vendor was difficult to deal with on this issue.

The licensee considered it important to replace these hubs at the next opportunity, (i.e., on-line train outage) and not wait until the subsequent refueling outage. The fan hubs in the CRAC system were of the same design of the fan hub for the high pressure core spray room cooler that had catastrophically failed in August 1995.

Although the craft performed the work in accordance with the work instructions, a number of problems extended the system outage approximately 24 hours longer than originally planned. The following were examples of delays experienced because of poor planning:

- The work instructions required the motor of the fan to be run uncoupled. Because of the motor interlocks with the ventilation dampers, the dampers needed to be removed from service to defeat the interlocks. The MAI did not include direction on how to do this. After 8 hours of delay, the damper electrical leads were lifted and the work proceeded.

- The MAI required the fan motor to be lifted out for bearing replacement. The work instructions contained rigging instructions that had not been approved by engineering. Obtaining engineering's approval resulted in several hours delay.

- During the clearance process, the tagging computer was removed from service by computer maintenance personnel. This resulted in several hours delay in releasing a clearance, because the information needed was in the database. This delay could have been avoided by better scheduling of the computer maintenance.

The inspectors concluded that, although the work on the Division I CRAC system was performed in accordance with the appropriate requirements, the licensee did not adequately manage the train outage from a risk perspective. Work documents were not adequately reviewed and validated prior to the work. The licensee did not appear to consider the impact of other tasks (i.e., the tagging computer maintenance) in order to minimize delays.

The inspectors discussed these conclusions with the Manager, Plant Operations, who had already decided to perform a critique of the CRAC outage. The licensee provided the inspectors with a draft copy of the critique results. The inspectors noted that the critique provided good recommendations for improvement. The inspectors considered the management response to poor performance during this system outage to be adequate.

4 SURVEILLANCE OBSERVATIONS (61726)

4.1 Division I Diesel Generator Operability Test

On April 22, 1996, the inspectors witnessed performance of Surveillance Test Procedure STP-309-0201, "Division I Diesel Generator Operability Test," Revision 16. The inspectors noted that operators employed good self-checking techniques and performed the test in accordance with the procedure. The inspectors reviewed the completed test data and found it to be satisfactory.

4.2 IST of RHR System Pumps and Valves

On May 7-9, 1996, the inspectors observed ASME Code, Section XI testing of selected valves, RHR Pump C, and the line fill pump for RHR Pumps B and C. The test was conducted in accordance with Procedure STP-204-6302, "Division II LPCI (RHR) Quarterly Pump and Valve Operability Test," Revision 10.

The operators experienced difficulties in attempts to test the RHR line fill pump. Initially, the pump suction test gauge did not respond because the instrument root valves were still closed. When the operator opened the root valves, the suction pressure test gauge indicated 7.5 psig, when the expected value was 2.27 psig. The operator replaced the gauge, but the new gauge indicated 7.3 psig which exceeded the calibrated range of the test pressure instrument of 5 psig. Nevertheless, the test was continued; however, pump differential pressure failed to meet the acceptance criteria. With a discharge pressure of 34.5 psig, the pump differential pressure was 27.2 psig, which was in the required action range. The ASME Code requires that the pump be declared inoperable until the cause of the deviation is determined and corrected. The operators initiated a CR to enter the problem into the corrective action program. Although the line fill pump was technically inoperable, it was adequately performing its intended function of keeping the Division II RHR piping full and pressurized above the minimum pressure, as noted by a gauge in the control room. The operators concluded, with engineering concurrence, that the RHR system was operable. The inspectors considered this to be a correct disposition.

On May 9, an operator, with additional IST experience, was called upon to assist and recognized that the pressure instruments were not aligned to obtain the correct pump discharge and suction pressures. The equalizing valve for the permanently installed, but normally isolated and equalized, pump differential pressure gauge was open when it should have been closed for the test. As a result, the crossconnected test gauges at the suction and discharge of the pump caused the erroneous data. After closing the equalizer valve and repeating the test, satisfactory test results were obtained.

The inspectors were concerned that unnecessary radiation exposures occurred, because the pump was in a high radiation area. Further, the inspectors had concerns that on May 7 the procedures did not provide adequate instructions for an operator with the expected "skill of craft" to properly align the instrument valves and obtain valid pressure data. As a result of reviewing Procedure STP-204-6302, the inspectors found the following deficiencies:

- Section 7.3.7 directed the operator to obtain pressure readings from the test gauges at the differential pressure gauge test connections, without any direction for valve position verification or manipulations. The operators referred to a generic procedure, Procedure OSP-6001, "Gauge Installation," Revision 1. This procedure simply stated, in part, that when data were required to be taken, then open the root/instrument valve(s) as applicable. Although the procedure contained, and the

operators used, a record of valve manipulations to maintain configuration control, the procedure relied on the knowledge of the operators to assure the correct lineup.

- Section 7.3.7 also instructed the engineer to obtain pump vibration readings at locations indicated on Attachment 8. The readings were not taken at the locations specified by the procedure, but rather, at the dimples installed on the pump structure. The dimples were in a good location to obtain consistent data; however, the conflict with the procedure attachment could cause confusion, delays, and unnecessary radiation exposure. The engineer initiated a CR to resolve the problem. Similar problems were identified with the spent fuel pool cooling pump IST in March 1996, which was reported in NRC Inspection Report 50-458/96-03 as a minor concern.

- Other editorial errors were identified. The operators identified an incorrect valve number in Section 7.2.8. In Section 7.2.16, the inspectors noted that the procedure directed the operators to secure RHR Loop C from Suppression Pool Cooling. Suppression Pool Cooling does not exist on RHR Loop C.

The inspectors were concerned that Revision 10 of Procedure STP-204-6302 may not been adequately reviewed. When the inspectors questioned the IST implementation procedure validation process, the licensee explained that Revision 10 was a last minute change that was not validated as a result of schedular pressures. In addition, there was insufficient detail in the procedure for the operators to satisfactorily accomplish the testing without a special level of qualification training. Failure to maintain an adequate procedure for implementation of the IST program is a violation of TS 5.4.1 (458/9604-02).

During the test of RHR Pump C, the pump discharge pressure was 187 psig, when the expected value was 163 psig. This placed the pump differential pressure above the required action range, and as such, the operators initiated a CR and declared the pump inoperable. The proper TS limiting condition for operation was already in effect.

The permanent local discharge pressure gauge was used for testing RHR Pump C, and this was the first time the licensee used permanent equipment instead of measuring and test equipment. A new permanent gauge had been installed, meeting the accuracy and range requirements of the ASME Code, so that there would be no need to temporarily connect test gauges. The permanent gauge was checked for accuracy and was found to be inaccurate by as much as 12 psig. The gauge was recalibrated, and the test was repeated with satisfactory results. The inspectors questioned the suitability of that particular gauge for reliable test results. The licensee stated that there was an MAI initiated for performing a post-test calibration check of this gauge to ensure

it remained in calibration for the test. This gauge had no safety function with respect to system operability.

Despite problems with the procedures and test instrumentation, the operators followed the procedure in a step-by-step manner and demonstrated good communications between the control room and the pump room. All personnel involved with the testing made a conscious effort to minimize radiation exposures by standing in low dose areas or, whenever possible, leaving the room, which was designated a high radiation area. The testing personnel were all very familiar with area radiological surveys.

The inspectors reviewed the completed test documentation and found that the data entries and signoff sheets were properly and legibly entered. All acceptance criteria were met.

6 ONSITE ENGINEERING (37551)

The inspectors reviewed the IST program to ensure that the licensee was meeting the ASME Code, Section XI requirements for IST. During this review, the inspectors noted that the licensee still had Relief Request RV-2 to the IST program in effect. This NRC-approved relief request allowed the licensee to test several safety-related valves during the drywell bypass leak rate test each outage. The inspectors were concerned that this relief request was still in effect, because the licensee had changed the frequency of the drywell bypass leak rate test to every 10 years. The licensee did not receive NRC approval to change the testing interval of the valves in question. The inspectors contacted the IST engineer, who stated that the licensee changed the method of testing the valves associated with Relief Request RV-2 to performing radiography of these valves every 18 months to assure no blockage in the line, which is an acceptable alternative to full-flow testing, as allowed by the ASME Code. The licensee had not yet applied to the NRC to have Relief Request RV-2 deleted from the program. The licensee stated that they would send in a retraction of Relief Request RV-2.

The inspectors noted that the IST program manual had not been kept up to date at one of the controlled locations (the technical library). The IST engineer stated that they were not keeping any of the IST program controlled copies up to date and should be considered "for information only." The licensee stated that they were keeping the IST program on a computer data base, ensuring that all procedures related to IST were current, and that the IST program manuals would be updated. The licensee stated that they considered the implementing procedures to constitute the "IST Program" and thus they had fully met the intent of the Improved TS.

The inspectors noted that NUREG-1482, "Requirements for Developing Inservice Testing Programs," April 1995, states, in part, that the IST program manual should be treated as a quality-related document and be controlled in accordance with Appendix B of 10 CFR 50. In addition, NUREG-1482 states, in part, that changes to the IST program should be controlled in accordance with 10 CFR 50.59, for changes that reflect changes to the facility as described in

the USAR. Furthermore, NUREG-1482 provides direction for the contents of each section of the program manual (e.g., NRC-approved relief requests), cold shutdown frequency justifications, justifications for using the preapproved exceptions contained in Generic Letter 89-04, and process diagrams that describe each testing circuit. Maintaining a listing of each test performed on a database program does not meet the recommendations of NUREG-1482. The licensee also did not have documented justifications for a number of items that they had deferred to cold shutdowns rather than testing quarterly. In addition, a number of justifications in place were out of date because they referenced the old TS instead of the Improved TS, which were implemented at River Bend in October 1995.

In parallel with the inspectors' concerns, the licensee had performed a peer review of the IST program using personnel from Grand Gulf Nuclear Station and IST engineers from Arkansas Nuclear One. This peer review also noted that the control of the IST program did not meet established industry standards. The licensee initiated CR 96-0651 to address these issues. Licensee management committed in the CR response to resolve the issues with the IST program plan, to meet industry standards, and to update the IST manual by June 12, 1996.

The inspectors noted that the IST program was included in the LTPIP. The licensee stated that their intent was to ensure that the implementing procedures were being upgraded. The procedure upgrades were completed in September 1995 and the LTPIP item for IST was considered closed. The licensee assigned IST Engineering to maintain the program and transferred responsibility of the implementing procedures and performance of IST to the Operations Department.

During this period, the licensee initiated some CRs that indicated that problems may have also existed with procedures and implementation. On April 15, 1996, the licensee identified that during the performance of Procedure STP-052-6301, "Control Rod Drive Valve Operability Test," Revision 6, on February 4, two valves had measured stroke times in the "Alert" range, but the frequency of testing was not increased. Section XI of the ASME Code, IWV-3417 requires, in part, that the testing frequency be increased from quarterly to monthly. Since the two valves had not been tested for over two months, this was considered a missed surveillance. The IST engineers wrote CR 96-0808 to enter this deficiency into the corrective action system.

The licensee's investigation found that Procedure STP-052-6301 contained direction for the user to evaluate the data, which was not followed. The data tables contained an acceptable range, an alert range, and a required action range. The table further defined "Acceptable, Conditionally Acceptable, and Unacceptable" data as acceptable, alert range, and required action range, respectively. All valves were circled as "Acceptable" in the data table, even though Valves 1C11*AOVF180 and -181, outboard scram discharge volume vent and drain valves, respectively, had documented stroke times in the "Alert" range. The cover sheet for this completed procedure, to be reviewed by the IST engineers, contained a note stating, "Due to maintenance and valve adjustment, valves will be rebaselined with new stroke time acceptance

criteria. Valve stroke times are acceptable." The senior reactor operator that wrote this statement was aware that Section XI of the ASME allows for rebaselining reference values following maintenance. However, the rebaselining tests had not yet been performed and the data should not have been considered acceptable. Failure to follow Procedure STP-052-6301 is the first example of a violation of TS 5.5.6 (458/9604-03).

In addition, IST engineers were required to review test data within 30 days of completion, as required by Section 7.5.2 of Procedure PEP-0009, "ASME Section XI IST Program Documentation," Revision 7A. The IST engineer received the completed copy of Procedure STP-052-6301 on February 5, but deferred review of the data. The engineer had not reviewed the data as of April 15, approximately 70 days after test completion. The failure to follow Procedure PEP-0009 is the second example of a violation of TS 5.5.6 (458/9604-03).

The inspectors concluded that the licensee had not fully corrected the problems noted in 1994 that generated the LTPIP item. In 1994, the IST program was considered out of date, lacking in technical justifications, and weak in personnel knowledge and training. Increased management oversight of the IST improvement process was not effective in preventing the IST program from degrading since the LTPIP item was closed. Further review of the IST program documentation and implementation will be required in order to adequately assess licensee performance in the IST program and will be tracked as an inspection followup item (458/9604-04).

7 PLANT SUPPORT ACTIVITIES (71750)

On May 6, 1996, a radiation protection technician initiated a CR to identify an occurrence on May 6 where a radiation worker entered the radiologically controlled area without wearing a direct reading dosimeter. The individual withdrew a dosimeter from the rack, logged on the radiation work permit using the computer, activated his dosimeter but returned the dosimeter to the rack instead of attaching it to his person, and entered the radiologically controlled area. This was in violation of Section 4.6.1 of Procedure RSP-0203, "Personnel Monitoring," Revision 11, which requires, in part, personnel to wear direct reading dosimeters for all entries into the radiologically controlled area. Fortuitously, another worker withdrew the same dosimeter from the rack and preceded to process in, whereupon the computer commenced exit processing with the previous worker's name. The worker questioned this and the technician paged the previous worker back to the access control point. The individual was in the radiological controlled area for approximately 6 minutes and did not receive any measurable radiation dose, based on a coworker's dosimeter reading.

Similar problems related to use of dosimetry were documented in NRC Inspection Report 50-458/96-03, and a violation was identified with three examples of failure to wear proper dosimetry. As of May 6, all corrective actions to prevent a recurrence were not completed. The training sessions to be held by supervisors of radiation workers were not completed nor were the modifications

completed that caused the entry processing computer to alarm if a dosimeter is left in the receptacle too long.

In response to the May 6 occurrence, the licensee shut down all access points except for the main access point. Also, management required all personnel to be briefed and granted specific Radiation Protection Department permission to enter the radiologically controlled area. This action was designed to get the attention of all site personnel, and the inspectors observed that it was doing so. In addition, notices were published to focus employee attention on management concerns about radiation worker practices. The specific individual was held accountable for his error. The licensee committed to address this fourth example in their response to the previous violation described earlier.

8 FOLLOWUP - MAINTENANCE (92902)

(Closed) Inspection Followup Item 458/9602-05: Questionable Operability of the Penetration Valve Leakage Control System (PVLCS)

On January 21, 1996, the licensee initiated CRs 96-0248 and 96-0249 indicating that Valves SAS*MOV-102 and -103 were leaking excessively by their seats, such that the test volume could not be pressurized during the local leak rate tests (LLRT). The licensee performed an evaluation and noted that the excessive leakage could potentially result in excessive flow in the PVLCS, causing the PVLCS to trip off. The licensee determined that the apparent degraded condition of Valves SAS*MOV-102 and -103 resulted in a loss of function of the PVLCS compressor. Maintenance personnel repaired the valves and satisfactorily retested them.

After reporting this issue, on February 16, the licensee located the as-found leakage data for Valve SAS*MOV-103. Although this leakage (17,000 sccm) exceeded the allowable leak rate for the valve, this leakage would not have tripped the PVLCS. Therefore, on February 16, the licensee retracted the 10 CFR 50.72 report.

Subsequent review by QA personnel noted discrepancies in the investigation. In particular, QA personnel noted that the 17,000 sccm leak rate, was measured on January 31, 10 days after the original CRs for the failures of the system to pressurize during the LLRT for the two valves. Additional investigation revealed that the 17,000 sccm leak rate was the as-left data following the first seat repair of Valve SAS*MOV-103. QA initiated CR 96-0564 to address this additional concern. Mechanics repaired Valve 1SAS*MOV-103 a second time and the valve passed the LLRT.

The licensee determined that, without the as-found data, they had no evidence that the PVLCS system had been inoperable. In addition, the LLRT personnel actually aborted the LLRTs of Valves SAS*MOV-102 and -103 because the boundary valves of the test were leaking and Valves SAS*MOV-102 and -103 may not have been leaking. With no further evidence of excessive leakage past Valves SAS*MOV-102 and -103, the inspectors agreed with the licensee's conclusions.

The inspectors reviewed the evaluation and noted that the licensee performed incomplete evaluations of this event on two occasions. The initial CRs, written by an individual who did not witness the actual conditions, stated that Valves 1SAS*MOV-102 and -103 leaked excessively past their seats and attributed the probable root causes to improper torque switch settings. Actually, the LLRT craft personnel aborted the test because of excessive boundary valve leakage. Investigators did not interview the craft that performed the LLRTs when evaluating these conditions for root cause. This information would have provided the licensee sufficient knowledge to properly determine the root cause and to evaluate the condition for safety significance. The second evaluation was also incomplete because personnel reviewed records discussing the issue with the personnel who developed the records. This resulted in an engineer inappropriately designating postmaintenance testing to be as-found testing.

The inspectors discussed this issue with plant manager. The plant manager agreed that the initial evaluations were not adequate and that corrective action would be taken with the individuals that initially evaluated this issue. The inspectors noted that previous licensee investigations, particularly when management was involved, were thorough. This appeared to be an isolated issue. The inspectors considered the management actions, once the sequence of events was correctly established, to be appropriate; therefore, this item is closed.

9 ONSITE REVIEW OF LERs (92700)

9.1 (Closed) LER 458/94-028: Manual Reactor Scram Due to High Low Pressure Turbine Vibration Caused by Temperature Sensitive Rotor

This issue was addressed in Section 2.2 of NRC Inspection Report 50-458/94-21. No violations of NRC regulations were identified.

9.2 (Closed) LER 458/94-030: Reactor Scram Resulting from Inadvertent Main Steam Isolation Valve Isolation Due to Failure to Follow Test Procedure

This issue was addressed in Section 2.1 of NRC Inspection Report 50-458/94-22. A violation was identified for failure to follow the surveillance test procedure.

9.3 (Closed) LER 458/94-032: Failure of the Division III Diesel Generator Due to Improper Synchronization Attempt by an Operator

This issue was addressed in Section 2.2 of NRC Inspection Report 50-458/94-22. A violation was identified for failure to follow the appropriate procedures.

9.4 (Closed) LER 458/95-001: Unintentional Division II Reactor Core Isolation Cooling Isolation During Surveillance Testing

This issue was addressed in Section 2.2 of NRC Inspection Report 50-458/95-01. A violation was identified for a repeated failure on the part of maintenance

planners to establish sufficient guidance to prevent inadvertent actuations of safety-related components while implementing work activities.

9.5 (Closed) LER 458/95-012: Manual Scram Due to Reactor Recirculation Pump Transient

This issue was addressed in Section 2.1 of NRC Inspection Report 50-458/95-26. No violations of NRC regulations were identified.

10 REVIEW OF USAR COMMITMENTS (71707)

A recent discovery of a licensee operating a 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 descriptions.

While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the USAR that related to the areas inspected. No inconsistencies were noted between the wording of the USAR and the plant practices, procedures, and/or parameters observed by the inspectors.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

W. R. Brian, Manager, Strategic Planning
M. A. Dietrich, Director, Quality Programs
J. P. Dimmette, General Manager, Plant Operations
D. T. Dormandy, Manager, System Engineering
J. R. Douet, Director, Plant Projects and Support
E. C. Ewing, Manager, Maintenance
G. C. Hockman, Quality Specialist IV
J. Holmes, Superintendent, Chemistry
H. B. Hutchens, Superintendent, Plant Security
R. J. King, Director, Nuclear Safety and Regulatory Affairs
M. A. Krupa, Manager, Operations
T. R. Leonard, Director, Engineering
L. G. Lewis, Manager, Training
D. N. Lorfing, Supervisor, Licensing
T. P. Lucy, Outage Coordinator
J. R. McGaha, Vice President-Operations
W. H. Odell, Superintendent, Radiation Control
A. Shahkarami, Manager, Engineering
G. A. Zinke, Manager, Quality Assurance

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 May 23, 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. On June 6, 1996, J. Dimmette, General Manager Plant Operations, committed to include, as part of their corrective actions for Violation 482/9603-02, the fourth example of failure to have a direct reading dosimeter while in the radiologically controlled area.