APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-458/92-05 Operating License: NPF-47

Docket: 50-458

Licensee: Gulf States Utilities
P.O. Box 220
St. Francisville, Louisiana 70775

Facility Name: River Bend Station

Inspection At: St. Francisville, Louisiana

Inspection Conducted: January 19 through February 29, 1992

Inspectors: E. J. Ford, Senior Resident Inspector
D. P. Loveless, Resident Inspector

Approved: P. H. Harrell, Chief, Project Section C — 3-20-92 Date

Inspection Summary

Inspection Conducted January 19 through February 29, 1992 (Report 50-458/92-05)

Areas Inspected: Routine, unannounced inspection of followup of a previously identified item, followup of licensee event reports (LERs), followup on TMI Action Plan (NUREG-0737) items, onsite followup of events, operational safety verification, maintenance and surveillance observations, engineered safety features system walkdown, and results of the midcycle performance review.

Results:

- The safety parameter display system was installed in accordance with licensee commitments and the program was fully implemented in accordance with NUREG-0737, Supplement 1 (paragraph 5).
- The failure of the licensee to perform a formal safety review prior to opening a sealed penetration in the auxiliary building was adequate to satisfy minimal Technical Specification requirements; however, it was not representative of conservative management practices (paragraph 6.a).
- Expanded use of self-checking communications by the control room operators was noted as a strength (paragraph 7.a).

- Standardized notices, warnings, and instructions throughout the plant has contributed to better worker understanding and a more professional appearance of the plant (paragraph 7.a).

- The inspector concluded that tracking limiting conditions for operations were adequately compensated for and did not pose a significant safety problem either individually or as a group (paragraph 7.b).

- The inspector observed good radiation work practices by plant workers and radiation protection personnel (paragraph 7.c).

- Clearance orders reviewed were properly initiated and released for work (paragraph 7.d).

- Maintenance of the penetration valve leakage control system was found to be adequate and the system remained operable (paragraph 7.e).

- The suppression pool pumpback system was found to be in proper flowpath alignment and operable in accordance with Technical Specifications (paragraph 7.f).

- Maintenance practices observed throughout this period were good (paragraph 8).

- Surveillance procedures reviewed met the Technical Specification surveillance requirements. Workers performed the tests in accordance with the procedures in a commendable fashion (paragraph 9).

- Residual Heat Removal (RHR) System C was found to be in proper flow path and electrical alignment. The system, procedures, drawings, and system maintenance were considered to be in superior condition (paragraph 10).

DETAILS

1. Persons Contacted

#J. E. Booker, Manager, Nuclear/Industry Relations
* J. W. Cook, Technical Assistant
*#T. C. Crouse, Manager, Administration
* W. L. Curran, Cajun Site Representative
*#L. A. England, Director, Nuclear Licensing
*#P. D. Graham, Plant Manager
* G. R. Kimmell, Director, Quality Assurance
* D. N. Lorfing, Supervisor, Nuclear Licensing
* I. M. Malik, Supervisor, Quality Operations
*#W. H. Odell, Manager, Oversight
#J. J. Pruitt, Manager, Business Systems
* S. Radabaugh, Supervisor, Maintenance Services
#M. F. Sankovich, Manager, Engineering
* J. P. Schippert, Assistant Plant Manager - Operations, Radwaste and Chemistry
#K. E. Suhrke, General Manager, Engineering and Administration
* C. W. Walling, Supervisor, System Engineer Process System

NRC Personnel

#D. H. Harrell, Chief, Project Section C
#E. J. Ford, Senior Resident Inspector
#E. E. Collins, Project Engineer

* Denotes personnel who attended the exit interview conducted on March 2, 1992. In addition to the above personnel, the inspectors contacted other personnel during this inspection period.

# Denotes personnel who attended the midcycle performance evaluation meeting conducted on March 3, 1992.

2. Plant Status

At the beginning of this inspection period, the reactor was in startup, with operators pulling control rods to achieve criticality.

On January 19, 1992, the reactor was taken critical. After generator testing, the output breaker was closed on January 20 and immediately tripped on a C phase fault. The breaker was reclosed on January 21, following repairs to the exciter cabling, and power escalation began.

On January 22, the power escalation was halted, at approximately 60 percent power, to investigate apparent condenser tube leaks. The reactor achieved 100 percent power on January 23.

On February 15, the reactor scrammed from 100 percent power when a faulty reactor protection system power supply failed. This event was reviewed and is documented in paragraph 6.b. Following repairs, the plant was returned to power on February 18.

At the end of this inspection period, the plant was operating at 100 percent power, steady-state operations.

3. Followup of a Previously Identified Item (92701)

(Closed) Inspector Followup Item 458/9026-02: Recirculation Pump B Failed to Transfer to Low Frequency Motor Generator Following High Speed Trip

On September 29, 1990, at approximately 40 percent reactor power, Reactor Recirculation Pump 1B was intentionally tripped, followed by a manual reactor scram in accordance with the planned activities of Refueling Outage 3. Following the trip, the pump should have downshifted to slow speed, but failed to do so. Approximately 5 minutes later, the pump again failed to transfer to slow speed after an attempt to restart the pump was made.

The licensee completed review of Condition Report (CR) 90-0820 and undertook corrective actions, as dcoumented in NRC Inspection Report 50-458/90-26. As part of the actions, system engineering reviewed logic circuits, schematics, and equipment to determine the reason for the failure of Breaker 1NNS*SWG5B-ACB042 to close. The licensee's documentation reflected that, of four affected relays, three relays (K134B, K135B, and K141B) and the low frequency motor generator control logic and components were checked for proper operation and calibration with no discrepancies found. The fourth relay (K136B), which appeared visually to be stuck closed, did not prevent a correct motor start and transfer to slow speed during repeated testing. The relay was subsequently removed under Maintenance Work Order Request 056593 to check physical parameters (clearances, bearings, contact resistance, and calibration) with no problems detected. However, due to contact discoloration and a disk sticking, which could not be duplicated, the relay was replaced.

A background search by the licensee disclosed previous instances of the failure of the pump to transfer or start due to various components. A review of the instances did not indicate common-mode or repetitive problems. The licensee has concluded that the root cause was indeterminate.

4. LER Followup (92700, 90712)

a. (Closed) LER 91-002: Isolation of Reactor Water Sample Containment Isolation Valve

While performing Surveillance Test Procedure STP-051-4510, "Reactor Protection/Residual Heat Removal (RPS/RHR) Reactor Vessel Steam Dome

Pressure High," Fuse 1B21H-F76B failed, causing the automatic isolation of Reactor Water Sample Valve 1B33*AOVF019. This isolation constituted an engineered safety feature actuation.

As documented in the LER, the licensee could not determine the cause of the fuse failure. The licensee identified three previous failures of Fuse 1B21H-F76B, which were also indeterminate. The licensee reviewed the sizing of the fuse and other potential problems and did not identify a root cause for the failures. Therefore, the licensee revised all associated surveillance test procedures that test this circuit to require Valve 1B33*AOVF019 to be closed prior to the performance of surveillance testing.

b. (Closed) LER 91-020: Hydrogen Ignitors Declared Inoperable - Surveillance Requirements Not Met

This LER documents that the licensee inappropriately classified certain hydrogen ignitor assemblies as inaccessible. These assemblies were, therefore, not tested in accordance with the Technical Specifications.

This event was previously reviewed, as documented in NRC Inspection Report 50-458/91-27. As a result of the review, a Notice of Violation (458/9127-01) was issued. The licensee's response to this violation, dated January 22, 1992, repeated the corrective actions stated in the LER. The corrective actions will be tracked and closed by the NRC under the violation; therefore, this LER is considered closed.

c. (Closed) LER 91-023: Reactor Water Cleanup Isolation During Standby Liquid Control Testing

This event was previously reviewed in detail, as documented in NRC Inspection Report 50-458/91-33, prior to the issuance of this LER. The previous inspector reviewed the root cause of the event and the licensee's initial corrective actions.

As permanent corrective action, the licensee revised Surveillance Test Procedure STP-201-3302, "Standby Liquid Control Valve Operability," to prevent the reactor water cleanup system from isolating during the testing sequence. The new procedure also restored the automatic isolation upon completion of testing. The licensee stated that a review of all standby liquid control surveillance procedures, for similar procedural deficiencies, had been performed. No errors were found.

The inspector reviewed Temporary Change Notice 92-0270 to Procedure STP-201-3302, Revision 6, and determined that the procedural instructions would no longer cause an inadvertent reactor water cleanup isolation.

5. Followup on TMI Action Plan (NUREG-0737) Items (TI 2515/065)

a. (Closed) Item I.D.2.2: Safety Parameter Display System (SPDS) - Installed

Item I.D.2.2 required applicants for an operating license to provide an SPDS in the control room. The Commission-approved requirements for the SPDS are defined in Supplement 1 to NUREG-0737. The River Bend Station SPDS function is provided by two computer systems: the Emergency Response and Information System (ERIS) and a computer console that provides a summary of the data from the Digital Radiation Monitoring System (DRMS).

The NRC staff reviewed the licensee's safety analysis report on the SPDS and several supplemental submittals, and conducted an onsite audit of the licensee's SPDS program.

On June 27, 1988, the NRC published a draft Safety Evaluation Report of the SPDS. The following conclusions were made:

- The NRC found the design and implementation program for the SPDS to be generally acceptable based on the fact that: (1) the variables selected for display were adequate to assess critical safety function, (2) the SPDS was suitably isolated from plant safety systems through the use of fiber optics cables, (3) the design provided a means to ensure that displayed data were validated, and (4) the design and implementation of the SPDS design included appropriate commitments to a human factors engineering program that should allow reasonable assurance that the information provided will be readily perceived and comprehended by its users.

- However, some concerns were identified. The NRC requested responses to these concerns for confirmatory review. These concerns are: (1) the licensee had not included several verification items on the SPDS in their human factors engineering evaluation, (2) the SPDS was determined to be cumbersome and awkward in use of keys, selection of displays, location of terminals, poor color coding, dummy screens, and indication of failure, (3) incoming radiation level alarms could be masked by other alarms because of a poor priority scheme in the DRMS, and (4) control functions could be performed by terminals in locations outside of the control room.

By letter dated December 17, 1986, the licensee responded to this NRC request for information. The licensee stated that they had completed the human factors engineering evaluation of the SPDS and provided a description of the evaluation methodology in Enclosure 1 to the letter.

The items involving cumbersome use of the SPDS were either justified or included as human engineering deficiencies to address the concerns. Modification Request 86-0362 was initiated to determine corrective action for masking of incoming radiation level alarms. Additionally, the licensee modified the SPDS to allow control functions to be performed only from the main control room.

By letter dated July 24, 1987, the licensee provided an updated status of the SPDS human engineering discrepancies identified in the draft Safety Evaluation Report. Those items that were not already closed were classified with a criticality rating of 3 or 4 (relatively low importance).

In a letter dated April 5, 1988, the NRC informed the licensee that, after review of the licensee's submittal, the NRC concluded that the actions proposed were acceptable. The NRC then closed its review of the SPDS.

Because the original audit was performed on site utilizing the licensee's installed SPDS, inspection activities were limited to those issues addressed in the Safety Evaluation Report. The following items were performed:

- NRC reviewed and accepted the licensee's human factors engineering evaluation methodology.

- In an NRC meeting summary, dated October 17, 1984, the NRC stated that with confirmation by the licensee that its DRMS verification and validation program was identical to the program applied on the Seabrook DRMS, the program would be acceptable to the NRC. The licensee's submittal, dated December 17, 1986, confirmed that the programs were identical. Therefore, the licensee's verification and validation software methodology was determined to be acceptable.

- The inspector verified that the dummy displays causing operator confusion had been removed from the SPDS. These displays had initially been installed to put signals in for testing of system algorithms.

- The inspector verified that Modification Request 86-0362 had been converted to Software Database Change Request SBD 87-004 and that SBD 87-004 had been implemented to correct the masking of radiation level alarms. The licensee implemented a sixth grid, which showed only Technical Specification required radiation monitors. This allowed alarms on this grid to be received regardless of the status of other radiation monitors.

- The inspector verified that control functions could not be performed from the terminal located in the emergency offsite facility station.

In addition to the above, the inspector verified that the licensee had taken the stated actions on the following human engineering deficiencies (HED), as documented in the licensee's submittal:

- HED 882: ERIS keyboards contain many keys not used by the operators.

  The licensee determined that no action was required on this item. The inspector verified by interview and observation that the arrangement of the keyboard did not affect operator interactions.

- HED 889: Operators do not understand all of the meanings associated with the SPDS color codes.

  The licensee added a screen to both ERIS and DRMS that explains the color codes. The inspector verified the presence of these screens and that operators could locate and utilize them.

- HED 894: Displays cannot be selected by keyboard entry of the display number.

  This format still exists for all screens other than the menu. The licensee redesignated the function keys to provide operators with single key display of the 24 displays utilized during use of the emergency operating procedures. The inspector verified that the operators could select screens from function keys or through use of the menu.

- HED 902: ERIS system failure is not sufficiently conspicuous to alert the operators.

  The licensee added improved software, which greatly reduced the problem causing the screen to stop updating. In addition, an ancillary program was added, which causes the screen to turn magenta within 1 minute if the processes are not updating.

- HED 904: The XMIT key used to enter format numbers was in an awkward location and its label did not properly reflect the key's function.

  The licensee determined that a change was not necessary. The inspector determined, through interview and observation, that the key location was not inhibiting.

- HED 919: Cyan is the color code used by ERIS to indicate valid information. The same color is used by DRMS to indicate questionable data.

The licensee determined that changing this color code would require extensive rewriting of system codes. The inspector noted that the primary color coding for normal, alert, and alarm was consistent between the systems. The inspector determined, between interview and observation, that this discrepancy did not affect the operator's use of the SPDS.

The NRC published Generic Letter 89-06, "Task Action Plan Item I.D.2 - System Parameter Display System," on April 12, 1989. This generic letter required licensee's to provide information, including photographs, on the parameters, screens, hardware, and location of the SPDS at their facility.

This letter documented that the NRC review was completed and verified that the licensee had a fully satisfactory SPDS. The licensee was, therefore, exempted from the reporting requirements of this generic letter.

Based on the inspection efforts and NRC reviews, as discussed above, and the exemption from the reporting requirements of Generic Letter 89-06, Item I.D.2.2 is closed.

b. (Closed) Item I.D.2.3: SPDS - Fully Implemented

Item I.D.2.3 required applicants for an operating license to provide procedures that describe the timely and correct safety status assessment when the SPDS is and is not available, to train the operators to respond to accident conditions with and without the SPDS available, and to provide an SPDS implementation that can be reliably assessed by control room personnel.

The inspector reviewed the following system operating procedures (SOP) and emergency operating procedures (EOP):

- EOP-1, "RPV Control," Revision 10
- EOP-1A, "RPV Control - ATWS," Revision 10
- EOP-2, "Containment Control," Revision 8
- EOP-3, "Secondary Containment and Radioactive Release Control," Revision 8
- SOP-0086, "Digital Radiation Monitoring System," Revision 3A

The inspector noted that the EOPs referenced the SPDS screens by function key number. Most references included both the top-level display, which provided many current parameter values, and the lower-level display that gave the specific parameter trend. In addition, ERIS two-dimensional plots required for evaluation of

safety status were referenced in the EOPs. The inspector noted that the EOPs provided sufficient information for a licensed operator to respond to accident conditions should the SPDS fail to operate. In NRC Inspection Report 50-458/90-07, the inspectors evaluated the EOPs and determined that they could be used by trained operators to mitigate the consequences of an accident.

The inspector reviewed EOP-3 and determined that DRMS radiation monitor grid numbers were referenced to provide the operators with the location of current radiation level indication. References of alternate control room indication were also provided. SOP-0086 provided the operators with the steps to retrieve data from the safety-related channels of the DRMS should the SPDS fail to function.

The SPDS does not have a written operating procedure. However, the licensee considers the function to be skill-of-the-craft for licensed operators. Through interviews, control room observations, and simulator observations, the inspector determined that licensed operators had a good working knowledge of the SPDS. The inspector has also noted good use of the SPDS by operators during licensing and license requalification examinations.

The inspector reviewed the following operator training modules and qualification cards:

- Hot License Operator Lesson Plan HLO-078-3, "Emergency Response Information System (ERIS)"
- Hot License Operator Lesson Plan HLO-069-2, "Digital Radiation Monitoring System"
- License Requalification Program Lesson Plan REQ-320-1, "Emergency Response Information System (ERIS)"
- License Requalification Program Lesson Plan REQ-322-0, "Digital Radiation Monitoring System (DRMS)"
- Licensed Operator Training Manual LOTM-68-2, Chapter 68, "Emergency Response and Information System (ERIS)"
- Licensed Operator Training Manual LOTM-65-2, Chapter 65, "Digital Radiation Monitoring System (DRMS)"
- Reactor Operator Qualification Card HLO-01J, "OJT Card," Items 272, "Area Radiation Monitoring," 273, "Process Radiation Monitoring," and 283, "Plant Process Computers/ERIS."

The inspector determined that the training provided to licensed operators for use of the SPDS was good. The inspector performed interviews and walkthroughs with senior reactor operators and observed their use of the system during simulator scenarios. The

observations indicated good operator knowledge and use of the SPDS. The licensed operator trainers informed the inspector that the ERIS portion of the SPDS was routinely turned off during accident training and that one scenario actually required this training. During interviews, the operators discussed knowledgeably the indications and techniques utilized should the SPDS fail during an accident.

The inspector noted that the licensee had developed training scenarios that failed the DRMS RM-11 computer console. The simulator did not contain simulation of the safety-related RM-23 modules. However, the operator was required to request information that would normally be monitored on the back panels. The inspector also verified that classroom training modules include extensive discussions on use of these safety-related modules as a backup to the SPDS.

The licensee utilizes the shift technical advisor to retrieve information from the SPDS during emergency situations to assist the control operating foreman. During simulator observations, the inspector observed that the knowledge level of these individuals was poor. Difficulties were noted in retrieving information and in verifying alarm status. The licensee acknowledged the inadequate training of these individuals. The licensee is currently changing the staffing, training, and use of control room shift technical advisors. Operations management stated that the training of these new individuals would include additional SPDS training.

The inspector performed inspections of the SPDS system and licensee documentation to determine system reliability. The inspector reviewed the licensee's work document data base for the last 2 years. Of the 26 maintenance work orders written on the SPDS during this time, 25 were written for routine testing or for single computer points being out of calibration. The other maintenance work order (RI32085), was written to repair a faulty system input module, which was replaced in a timely manner. Additionally, the inspector noted that the licensee had developed and was performing an extensive preventive maintenance program for the SPDS. These reviews did not indicate problems with overall system operability or reliability.

The inspector reviewed the licensee's CR data base for the life of the plant and found the following two reports were the only reports written against the SPDS:

- CR 90-0444: Time indication is wrong on the SPDS. The licensee added a National Bureau of Standards satellite-driven clock signal to the SPDS to synchronize the time signals with other plant computers.

- CR 90-0744: This CR was written to document additional problems with timing systems used on control room computers. The licensee documented that the systems would remain on Central Standard

Time because of data time stamp skews inherent in a periodic change of time.

Neither of the above CRs indicated a problem with the operability or reliability of the SPDS. The inspector concluded, based on inspections, interviews with licensed operators, the licensee's preventive maintenance program, and a review of maintenance and corrective action data bases, that the SPDS provides reliable information to the control room personnel; therefore, Item I.D.2.3 is closed.

c. (Closed) Item I.D.2: SPDS Console

Item I.D.2.1 was written to document the receipt and initial review of the licensee's proposal for the SPDS. This item was accomplished by the licensee on April 24, 1984, by forwarding the River Bend Station Safety Parameter Display System Safety Analysis Report. The NRC review of this safety analysis report was covered under Item I.D.2.2 reviews, thus Item I.D.2.1 was closed previously.

Items I.D.2.2 and I.D.2.3 are closed as documented in Paragraphs 5.a and 5.b; therefore, Item I.D.2 is closed in total.

Conclusion

The SPDS was installed in accordance with licensee commitments and the program was fully implemented in accordance with NUREG-0737, Supplement 1.

6. Onsite Followup of Events (93702)

a. Secondary Containment Integrity

On February 3, 1992, the licensee opened a sealed auxiliary building penetration to pass a 3/4-inch hose into the 98-foot elevation of the building. The inspector questioned the operability of the secondary containment enclosure with this penetration open. Technical Specification 3.6.5.1 requires that secondary containment integrity be maintained while the reactor is in Operational Conditions 1, 2, or 3.

Surveillance Requirement 4.6.5.1.b.3 requires that the licensee verify that all secondary containment penetrations, not capable of being closed by automatic isolation dampers, are closed. The licensee stated that this requirement was only for large penetrations, as defined by the Updated Safety Analysis Report and licensee procedures. The inspector reviewed the licensee's analysis of the 4-inch opening. The licensee concluded that running a 3/4-inch hose through this 4-inch penetration did not effect the operability of secondary containment because it would not impair the operation of the standby gas treatment system and the auxiliary building would still be maintained at a negative pressure.

The inspector discussed the licensee's interpretation of the requirements of Technical Specifications with the Office of Nuclear Reactor Regulation staff. The staff concluded that the licensee may not have been using the most conservative judgement; however, their interpretation was adequate. The inspector presented this interpretation to the licensee, but stated that a more appropriate safety review should have taken place prior to opening the penetration. This would have provided for more conservative management oversight.

b. Reactor Scram

On February 15, 19[illegible], the licensee determined, through surveillance testing, that the scram discharge volume high level circuit was not working properly. In accordance with Technical Specification 3.3.1.a, the licensee placed the channel in the tripped condition, causing a one-half scram signal on Reactor Protection System B.

Later that evening, while repairs to the circuit were under way, Power Supply PS-23, to Average Power Range Monitors C and G, failed. This failure caused a trip of RPS C. The trip of both RPS divisions caused a scram of the reactor. All control rods inserted and initially all equipment operated, as designed. Approximately 7 seconds into the event, reactor water level decreased below the Level 3 setpoint, causing the engineered safety feature isolation of reactor water sample lines. The level recovered within 10 seconds.

Approximately 4 minutes into the event, the main generator tripped on reverse power, as designed. At this point, Switchgear 1NPS-SWG1A failed to fast transfer to offsite power; however, the bus did slow transfer to offsite power. This caused a momentary loss of power to certain large nonsafety-related loads (Reactor Recirculation Pump A, Circulating Water Pumps A and C, Service Water Pumps A and C, Condensate Pumps A and C, and Instrument Air Compressor A). The loss of the service water pumps caused an automatic start of all divisions of standby service water. No additional safety functions occurred.

The licensee determined that the failure of the power supply was an internal fault and the power supply was replaced. The scram discharge level circuit had failed because of a loose relay. Other RPS relays were inspected for mounting problems. No additional problems were identified.

The licensee evaluated the failure of Switchgear 1NPS-SWG1A to fast transfer. The offsite supply breaker to the bus, Breaker 11, a General-Electric (GE) breaker, was found to have hardened grease in the area of the latch roller bearing. Upon disassembly, the licensee determined that the bearing had frozen to the shaft causing the shaft to rotate, as opposed to the bearing. This caused the breaker function to slow enough that a six-cycle fast closure was not completed. The licensee replaced the breaker.

The licensee stated that the Switchgear A breakers had not received preventive maintenance performed since Refueling Outage 2; however, the licensee has never had a failure of a GE breaker to close. Preventive maintenance is scheduled to be performed on these breakers during Refueling Outage 4. All other large GE breakers in the plant had been surveilled within the last 18 months. Additionally, the licensee stated that the safety-related breakers are exercised regularly during Technical Specification required surveillance.

On February 18, 1992, following the above described repairs, the reactor was returned to power upon completion of a posttrip review and approval by the Facility Review Committee.

Conclusions

The failure of the licensee to perform a formal safety review prior to opening a sealed penetration in the auxiliary building was adequate to satisfy minimal Technical Specification requirements; however, it was not representative of conservative management practices.

7. Operational Safety Verification (71707)

a. Plant Tours

On February 28, 1992, the inspector noted a large air leak in the main steam safety relief air system dryer skid. This system provides air to the automatic depressurization system air accumulators. The operators isolated the leak and determined that the system header pressure had not dropped below required values while the leak was present. The air leak was repaired by the licensee.

The inspectors routinely observed evolutions in the control room for procedural compliance and appropriate communications. The licensed operators conducted themselves in a professional manner and were aware of plant status when questioned. During one testing sequence, the inspector noted that the control operating foreman was monitoring the licensed operators for proper communication practices. The inspector routinely observed self-checking practices taken by licensed operators as a result of a new licensee initiative.

On February 19, 1992, the inspector toured all elevations of the control building, normal switchgear building, and diesel generator spaces and associated tunnels. The inspector did not note any housekeeping problems and observed the use of improved firedoor placards throughout those areas. There has been a general improvement in the use of standardized notices, warnings, and instructions throughout the plant, which has contributed to a more professional appearance of the plant.

b. Review of Tracking Limiting Conditions for Operation (LCOs)

During this inspection period, the inspector noted that the numbers of tracking LCO in the licensee's system was increasing. The licensee uses this system to keep track of equipment out of service, which is referenced by Technical Specifications, but is not required in the current operational condition or plant configuration. In addition, the licensee may use this system to track administrative requirements of the Technical Specifications like consequential testing requirements or alternative equipment alignments. The inspector reviewed the specific circumstances to determine if the out-of-service backup equipment was at a level to cause concern.

The total number of open tracking LCOs was determined by review of licensee records to be increasing over the last 9 months. However, the records indicated that the total number closed during any given period remained fairly constant. This suggested that the licensee was actively working to correct equipment problems.

The inspector reviewed the specifics of each tracking LCO, which were open on February 24, 1992. Of the 32 open, the inspector concluded that 8 affected equipment that could only be worked during outage times. Thirteen were determined to be administrative in nature and did not affect major equipment. The inspector evaluated the remaining 11 tracking LCOs and determined that each was actively being worked by the licensee and that valid reasons existed for the equipment to remain out of service.

c. Radiological Protection

The inspector reviewed Radiation Work Permit 92-0014, "Special Maintenance - Inspections, Surveillances, MWORs." The inspector noted that special surveys were being performed, as necessary, prior to authorizing work under the radiation work permit. The inspector observed good work practices by plant workers and radiation protection personnel.

The inspector observed the monitoring of closed Cooling Water Pump 1CCP-PB, prior to its release from the radiological control area. The technician was well acquainted with the monitoring equipment and procedures. The pump was properly frisked, bagged, and allowed to be removed to the maintenance shop for rebuild work.

d. Clearance Orders

The inspector reviewed Clearance Order RB-1-92-2017 on the closed cooling water system. The inspector reviewed Drawing PID-9-1A, "Closed Cooling Water - Reactor Plant," and verified that the clearance boundary was appropriately selected. Additionally, the inspector walked down the clearance order boundary and verified that the clearance order had been properly hung prior to release for work.

e. Penetration Valve Leakage Control System (PVLCS)

The inspector evaluated the materiel condition of PVLCS Compressors 1LSV*C3A and 1LSV*C3B. In 1985, the licensee removed the intervals of the vendor-supplied air intake and discharge valves because the valve function and logic did not properly work for the application at the River Bend Station. This has caused continuous seal water leakage and water rejection during startup of the compressor.

Six years of service water leaking onto the compressor skid, and the puddling of this water around structural components, has caused coating deterioration and corrosion of the base metal. The inspector notified the licensee that, in this condition, the corrosion rate and effect on the structural steel was indeterminate.

The licensee initiated CR 91-0524 to request the maintenance organization to evaluate the condition of the PVLCS compressor skid. The licensee determined that all dimensions were above the military standard minimum for the structural members. However, as corrective action to prevent further degradation of the skid, the licensee installed custom-fitted drip pans on each compressor. The licensee intends to clean the surfaces of the channel irons to bare metal and reapply coatings.

f. Suppression Pool Pumpback System (SPPS)

The inspector reviewed the design basis and operability of the SPPS. The system consists of two subsystems, as required by Technical Specification 3.5.3.c. Each subsystem contains two pumps and an isolation valve to the radioactive waste system. Additionally, a common motor-operated valve (1DFR*MOV146) is provided to open a pathway to the suppression pool. One check valve in each subsystem provides the second containment isolation valve upstream of Valve 1DFR*MOV146.

The inspector walked down the system and determined that both subsystems were aligned for proper flowpath. The inspector questioned the design of the system discharge through Valve 1DFR*MOV146. A failure of this valve to open would cause both subsystems to fail.

The Updated Safety Analysis Report discusses additional design requirements for piping between the containment wall and the outermost containment isolation valve, which supports the assumption of no postulated failures. As a result, the licensee assumed that the greatest leak in the sump area for this system would be a 50-gpm leak through an emergency core cooling system isolation valve. Therefore, the design requirement for SPPS is to operate in the event of a failure of a watertight cubicle. In this respect, only one subsystem would be necessary to support plant operation, because failure of the SPPS alone does not result in a failure of the suppression pool or the emergency core cooling system. In conclusion, the SPPS was found to be in compliance with the Technical Specifications.

Conclusions

Expanded use of self-checking of communications by the control room operators was noted as a strength.

Standardized notices, warnings, and instructions throughout the plant have contributed to better worker understanding and a more professional appearance of the plant.

The inspector observed good work practices by plant workers and radiation protection personnel.

Clearance orders reviewed were properly initiated and released for work.

Maintenance of the PVLCS was found to be adequate and the system was operable.

The SPPS was found to be in proper flowpath alignment and operable in accordance with the Technical Specifications.

8. Maintenance Observations (62703)

a. On February 4, 1992, the inspector observed portions of the work being performed under Prompt Maintenance Work Order 059008, "1CCP-P1B - Troubleshooting Indicates Thrust Bearing Failure." The inspector observed the alignment check and removal of the Component Cooling Water Pump B. The mechanics followed the work plan and utilized Corrective Maintenance Procedure CMP-9002, "Gould Pumps, Models 3196 XLT, MT and ST Disassembly, Inspection, Rework and Reassembly."

The inspector verified that the clearance order was properly hung and that the receipt was in the work package. Additionally, the mechanics were using equipment removal tags and disassembled component identification and control tags, as required by Maintenance Section Procedure MSP-0021, "Equipment Removal/Disassembly Identification Tag." The inspector noted that open equipment was properly covered, good housekeeping practices were used, and rigging was performed in accordance with General Maintenance Procedure GMP-0017, "General Rigging Practice."

The mechanics were knowledgeable of the equipment and the specific job. Additionally, they utilized practices to preserve the pump as-found condition for later review by maintenance management. Proper health physics practices were observed throughout the work and during the removal of the pump from the radiologically controlled area.

b. During the performance of Procedure STP-251-3300, "Quarterly Diesel Fire Pump Battery Test," the licensee determined that the specific gravity of Cell A in the battery for Diesel Fire Pump B had fallen below 1.200. The licensee replaced a portion of the electrolyte and

protection system was scheduled for review under their system reliability program. This will provide a living system for trending of this type of failure in the future.

Conclusion

Maintenance practices observed throughout the period were good.

9. Surveillance Observation (61726)

On February 3, 1992, the inspector observed the performance of Procedure STP-601-0201, "RWCU SLC Channel A, B Isolation Actuation." The inspector verified that this procedure had been performed within the last 31 days, as required by Technical Specifications. The inspector independently verified that the revision in use (Revision 5) was the most current, an official work copy was being utilized, and the operators were qualified to perform this procedure in accordance with Administrative Procedure ADM-0007, "Selection, Training, Qualification and Evaluation of Plant Staff Personnel."

The inspector reviewed Procedure STP-601-0201 and determined that the procedural steps provided a good methodical approach to the test performance. The inspector determined that the procedure, if properly implemented, met the surveillance requirement of Technical Specification 4.3.2.1, Item 4.3.2.1-1.4.h. Additionally, the procedure required that, if the testing evolution was delayed, the operator must restore the containment isolation valves to operable status within 4 hours so that Technical Specification 3.6.4 would be met.

The inspector reviewed Electrical Drawing GE 828E445AA, Sheet 15, "Nuclear Steam Supply Shutoff System," and verified that the test, as performed, tested the entire circuit. This included a review of the ERIS data points used for indication during the test. The operators performing the test were knowledgeable of the procedure, followed the procedural steps, and were aware of potential plant effects of the testing.

Conclusions

Surveillance procedures reviewed met the Technical Specification surveillance requirements. Workers performed the tests in accordance with the procedures in a commendable fashion.

10. Engineered Safety Feature System Walkdown (71710)

The inspector independently verified the status of RHR System C. This system provides low pressure coolant injection in response to an accident signal.

The inspector verified that the System Operating Procedure SOP-0031, "Residual Heat Removal," valve lineup checklist matched the system

Engineering Piping and Instrumentation Diagram PID-27-7C, "Residual Heat Removal - LPCI." A detailed walkdown of all system components outside the drywell was performed. No discrepancies were noted.

The inspector observed the general system status and determined that hangers and supports were properly aligned, housekeeping was good, labeling of components was excellent, and system components appeared to be in good condition. One minor packing leak was noted on Drain Isolation Valve E12*VF056C; however, the inspector verified that the leak was documented on a maintenance work order request.

The inspector verified that the station electrical alignment was in accordance with the electrical alignment required by Procedure SOP-0031. In addition, the inspector observed the inside of an associated motor control center. The physical condition of the breakers and bolting and terminations was good and no loose material, jumpers, or evidence of rodents was observed.

The inspector verified the instrument alignments of a sampling of system process instrumentation, and determined that the calibration was current. Control room alignments were verified and pump room cooling was available.

Conclusion

RHR System C was found to be in proper flow path and electrical alignment. The system, procedures, drawings, and system maintenance were considered to be in superior condition.

11. Midcycle Performance Review

On March 3, 1992, a meeting was held on site between licensee and NRC personnel to discuss the NRC's evaluation of the licensee's performance at the approximate midpoint of the current SALP cycle. The licensee's SALP cycle extends from April 1, 1991, to July 4, 1992.

The NRC conducted the performance review to provide feedback to the licensee on the current status of their performance. The attendees at the meeting are listed in paragraph 1. The outline of the information presented at the meeting is included as an attachment to this inspection report.

12. Summary of Open Items

The following is a synopsis of the status of all open items generated and closed in this inspection report. The following items were closed:

- Inspector Followup Item 458/9026-02
- LERs 91-002, 91-020, and 91-023
- Three-Mile Island Items I.D.2, I.D.2.2, and I.D.2.3

set the cell on a trickle charge. This action brought the battery cell's specific gravity back into specification tolerances and the fire pump was returned to an operable status.

The inspector reviewed the following maintenance work orders to determine if a generic condition existed:

- Fire Pump A

  R137925, "Battery A Has Low Voltage Ready"

  R169698, "Severely Burned Positive Terminal on Battery"

  R169781, "Battery B Pulled Excessive Amps on Starting Engine"

  R122174, "Corrected Specific Gravity for Cell #A-7 of 1FPW P1A Battery"

  R056523, "Diesel Fire Pump 1A Battery Terminals are Burned and Partially Melted"

  R132662, "Battery Bank 2 There's a Crack on Top of Battery Between Cells 2-2 and 2-3"

  R169781, "Elect. Reports One Battery Reading 9 VDC"

- Fire Pump B

  R149078, "Cell Number 8 did not Pass STP-251-3300, Replace Battery"

  R150009, "Battery "A" Failed Acceptance Criteria of Specific Gravity"

  R169698, "Fire Pump Battery "A" Voltimeter Reads 24 VDC, it Should Read Greater Than 25 VDC"

On January 29, 1992, the inspector discussed this issue with the Assistant Plant Manager for Operations, Chemistry and Radwaste. The licensee then initiated CR 92-0059 to document the repeat problem and provide for corrective actions to prevent more battery failures.

The inspector reviewed Maintenance Policy Guideline MPG-002-005, "Planning Guideline." Section 5.3.2 requires that planners record any five related failures of a given component on a CR. Although the above documented work orders may be related upon review, the inspector determined that the review required by MPG-002-005 would not necessarily have identified a problem with the batteries. Additionally, the licensee informed the inspector that the fire

13. Exit Interview

An exit interview was conducted with licensee representatives identified in paragraph 1 on March 2, 1992. During this interview, the inspectors reviewed the scope and findings of the report. The licensee did not identify as proprietary, any information provided to, or reviewed by, the inspectors.

ATTACHMENT

MIDSALP CYCLE PERFORMANCE REVIEW

OPERATIONS

* Operations staff were implementing radiological control in an effective manner.

* Management oversight was apparent and conservative.

* Response to equipment-related events was good.

* The fire brigade was knowledgeable and well trained.

* Housekeeping was very good during plant operation, but needs additional attention during an outage.

* Repeat-back communications in the control room were a strong point.

* Operator performance during initial examinations is good overall.

* Training and operations staff were very professional during initial examinations.

* Lack of attention to details resulted in containment isolation not being verified

* Concern identified during license requalification examinations with the guidance provided in procedures, contributed to the failure of two operators.

* Overall evaluation

  + Management has successfully addressed concerns identified with operator compliance with radiological protection requirements.

  + The Operations staff performing well.

  + Management oversight and conservatism were apparent.

  + Housekeeping is generally good, but declines were noted during the outage.

  + Repeat-back communications in the control room are a strength.

  + There were procedure concerns with respect to operators being able to complete required actions

RADIOLOGICAL CONTROLS

* Implementation of the radiological protection program was very good.

* Very good audits were being performed in this functional area.

* Improvement in postings for radiological conditions has been noted

* Two incidents, early in the assessment period, were identified with the failure to control high radiation area barriers

* Overall evaluation

    + Improvements were noted in the implementation of the radiological protection program.

    + Quality assurance audits appeared to be comprehensive and performance-based.

MAINTENANCE/SURVEILLANCE

* Supervisor observation of maintenance activities was a strength.

* Surveillance test procedures were being revised, when required.

* Maintenance and surveillance activities were being performed well.

* Crafts personnel displayed good performance and working knowledge.

* Scheduling of surveillance tests is a strength.

* The quality of surveillance records is high.

* Interpretation of Technical Specifications for air lock barrel testing was nonconservative.

* Surveillance testing for hydrogen igniters was not performed in accordance with the Technical Specification requirements.

* The reactor coolant system sump drain flow surveillance test was not done within the time limit specified in the Technical Specifications.

* Overall performance

    + Management oversight of this area was apparent.

    + Maintenance and surveillance activities were performed well.

+ Nonconservative interpretation of Technical Specifications were identified as a concern.

+ Crafts personnel displayed good performance.

EMERGENC PREPAREDNESS

* The licensee has made a commitment to improve the quality of emergency directors and recovery managers.

* The licensee has made commitment to seek methods to develop and test command and control skills

* Poor internal communications contributed to failure to declare an unusual event when a contaminated individual was transported offsite.

* Overall

  + Actions were being implemented by management to improve performance in emergency response.

  + Internal communications need to be reviewed to ensure the appropriate information is provided to the appropriate personnel.

SECURITY

* Security personnel were knowledgeable.

* Day-to-day activities of the security force were excellent.

* A violation for the control of visitors was identified.

* Daily radio checks were not performed with the local law enforcement agencies.

* Good audits of the security program were being performed.

* Overall performance

  + Apparent improvement in the performance of the security force was noted.

ENGINEERING/TECHNICAL SUPPORT

* Performance of a safety-system functional inspection on Division III was proactive.

* A good relationship was shown between training, operations, and maintenance.

* Hydrogen mixing systems were inoperable since plant startup.

* A temperature switch was removed from the fuel building ventilation system without completion of a 10 CFR Part 50.59 evaluation.

* There was no apparent linkage of simulator scenarios to licensed operator task lists.

* Tasks for instrumentation and control technician tasks evaluated outside the laboratory were not supported by on-the-job training cards.

* Overall performance

    + Concerns continue to be identified with the licensed operator and nonlicensed training program.

    + Requirements for the performance of Part 50.59 evaluations need to be reviewed.

SAFETY ASSURANCE/QUALITY VERIFICATION

* The licensee had a good 10 CFR Part 21 evaluation program.

* Actions to resolve the cracking concerns with the hydraulic control units was proactive.

* Response to the hydrogen mixing system inoperability problems, once the problems were identified, was very good.

* Procurement activities were effectively implemented.

* Self-assessment and corrective action programs were effective.

* Root cause evaluations were generally adequate.

* The threshold for requiring routine review of condition reports by the Facility Review Committee and the Nuclear Review Board was too high.

* Overall performance

    + Management displayed a proactive approach to resolving issues

    + Programs for identifying potential safety deficiencies were effective.

    + Review of the threshold for review of nonconformance reports by the independent oversight committees should be performed.

bcc distrib. by RIV:

R. D. Martin	Resident Inspector
DRP	Section Chief (DRP/C)
Lisa Shea, RM/ALF	MIS System
DRSS-RPEPS	RSTS Operator
Project Engineer (DRP/C)	RIV File
DRS	Senior Resident Inspector, Cooper
Senior Resident Inspector, Fort Calhoun	