APPENDIX B

U.S. NUCLEA. 'ULATORY COMMISSION
. ON IV

NRC Inspection Report No. 50-458/92-09

Operating License No. NPF-47

Licensee: Gulf States Utilities Company (GSU)

Facility Name: River Bend Station (RBS)

Inspection At: RBS, St. Francisville, Louisiana

Inspection Conducted: March 23-27, 1992

Inspectors: L. Ellershaw, Reactor Inspector, Materials and Quality Programs Section, Division of Reactor Safety

L. Gilbert, Reactor Inspector, Materials and Quality Programs Section, Division of Reactor Safety

Approved: I. Barnes — 4-17-92
I. Barnes, Chief, Materials and Quality Programs Section, Division of Reactor Safety — Date

Inspection Summary

Inspection Conducted March 23-27, 1992 (Report 50-458/92-09)

Areas Inspected: Routine, unannounced inspection of the licensee's programs for handling and feedback of operational experience information, inservice inspection, and activities associated with the feedwater nozzle safe end replacement effort.

Results: Programmatically, the handling and feedback of operating experience information appeared to be well defined. The inspectors did not identify any instances where information, considered to be important for the safe operation of RBS, was not provided in a timely fashion to the operations and/or maintenance staff. One violation was identified (paragraph 2.2), pertaining to the failure by responsible departments to respond to Nuclear Licensing's requests for evaluations of identified potentially reportable conditions (PRCs) within the specified time limits. It was additionally noted that Nuclear Licensing was not timely in generating required statements of action in order for commitments to be identified and tracked. The inspectors also questioned the licensee's inability to demonstrate, for those cases where responses to requests had not been made, that an initial documented operability decision had been made. The licensee informed the inspectors that this problem had been recognized and that a program change was in-process, whereby PRCs would be handled within the condition report program.

In general, the inspectors found the ISI examination procedures to be well written and consistent with the requirements of the ASME Code. However, the inspectors noted that the ISI instructions and approvals could be enhanced by specifying the procedure which would be used to perform a specific ISI examination in the ISI Plan. One anomaly was identified (paragraph 3) concerning the acceptance standard specified in a fluorescent magnetic particle examination procedure which was resolved during the inspection.

The inspectors determined that the overall approach to the feedwater nozzle repair activity was well thought out and conceptually sound. The building of a full size mockup was considered commendable and should contribute to achieving ALARA goals.

The inspectors noted an inconsistency with respect to the temperature tolerances specified in the preliminary specification for postweld heat treatment of the feedwater nozzle repair activity. The inspectors also questioned the wording in the postweld heat treatment specification that permitted the nozzle to exceed 1200°F for up to 1/2 hour during postweld heat treatment. Licensee personnel indicated that these anomalies had been recognized and would be corrected prior to final approval and release for use of the specification.

DETAILS

1. PERSONS CONTACTED

RBS

*J. Blakely, Quality Assurance (QA) Supervisor, Inservice Inspection (ISI)
*J. Booker, Manager, Nuclear Industry Relations
*J. Deddens, Senior Vice President
*L. Dietrich, Supervisor, Nuclear Licensing
*G. Dolney, System Engineer
*L. England, Director, Nuclear Licensing
*P. Graham, Plant Manager
*J. Hamilton, Director, Design Engineering
*R. Jackson, Technical Specialist, ISI
*B. Kienlen, Nondestructive Examination (NDE) Level III Examiner
*G. Kimmell, Director, QA
*D. Lorfing, Supervisor, Nuclear Licensing
*G. Mahan, Senior Welding Engineer
*J. McQuirter, Licensing Engineer
*W. Odell, Manager, Oversight
*M. Sankovich, Manager, Engineering
*K. Suhrke, General Manager, Engineering and Administration
*C. Walker, Supervisor, Operations Quality Control
D. Wells, Operating Experience Program Manager

EBASCO SERVICES, INC.

*C. Latiolais, Project Manager and NDE Level III Examiner

NRC

*D. Loveless, Resident Inspector

The inspectors also interviewed other licensee employees during the inspection.

*Denotes those attending the exit meeting on March 27, 1992.

2. FEEDBACK OF OPERATIONAL EXPERIENCE INFORMATION (90700)

The purpose of this inspection was to determine the effectiveness of the licensee's program to assess and disseminate operational experience information pertinent to plant safety, which originated outside the organization.

2.1 Program Verification

The program and associated responsibilities are controlled and described in the following procedures. Procedure RBNP-043, "River Bend Station Nuclear Network Coordination Activities," Revision 1, provided guidance to Nuclear Licensing personnel who access the Institute of Nuclear Power Operations (INPO) nuclear network telecommunications system, and applied to all items entered into or received from the nuclear network system. Procedure RBNP-026, "Processing 10 CFR 21 Reports," Revision 1, established a method for fulfilling the reporting requirements of 10 CFR Part 21 and provided the guidance for handling suspected hardware deficiencies on site and vendor correspondence that identified potential hardware deficiencies. Procedure NLP-10-007, "Processing Evaluations of Reportability Under 10CFR21," Revision 1, established the methodology and documentation required to evaluate conditions which are potentially reportable pursuant to 10 CFR Part 21. Procedure NLP-10-006, "Processing and Tracking of Regulatory and Industry Correspondence," Revision 2 through Interim Procedure Change IPC-10-006-2-1, provided the method for processing and tracking licensing correspondence from or to the NRC, INPO, or other industry sources.

Known hardware deficiencies typically require a Condition Report to be initiated. Condition Reports are routed to the control room so that an operability decision can be made. It was not apparent to the inspectors that PRCs would receive the same control room operability review. Procedure RBNP-026 provided for an evaluation by the designated/applicable department heads in order to establish whether a deficient condition exists, and subsequent reportability. However, as discussed in paragraph 2.2, untimely evaluations which subsequently identify deficient conditions preclude the immediate control room review for operability. The inspectors were informed in response to questions on this subject that a program change was in-process, which would provide for initiating a Condition Report for all PRCs and thereby resulting in immediate control room review. This should also ensure that corrective actions are properly identified and implemented.

2.2 Program Assessment

The inspectors requested printouts that would provide the status of three types of externally initiated correspondence (i.e., INPO's Significant Operating Experience Reports [SOERs], NRC Information Notices [INs], and vendor notification of PRCs).

It was noted with respect to responsibilities, that Procedure NLP-10-006 required responsible department heads to provide information requested by Nuclear Licensing within the designated due date so that Statements of Action could be prepared. Statements of Action are written summaries comprised of actions taken or planned as a result of a concern or recommendation, and are issued by Nuclear Licensing. They are the principal means by which commitments are identified and entered into the commitment tracking system. The interim procedure change dated June 21, 1991, provided more specificity regarding response times. It stated that Information Data Sheets (IDSs) used in conjunction with INs, contain a comment to the responsible department heads that a response is requested within 60 days, and if that time is insufficient, the responsible person shall provide a new date for the response and a justification as to why it is acceptable to extend the time for review. The IDS, which serves as a cover sheet for information sent to responsible department heads, is used by Nuclear Licensing as an aid in tracking licensing correspondence. With respect to PRCs, Procedure RBNP-026 required responsible department heads to provide responses to Nuclear Licensing's request for information regarding reportability within 30 days, or provide a response schedule. The language of Procedure NLP-10-007 was less concise, in that it stated that department heads were responsible for timely responses to Nuclear Licensing's requests for information to establish reportability, and that efforts should be made to complete the evaluation within 30 days.

The inspectors' review of seven SOERs revealed that the licensee performed timely evaluations and, where applicable, identified and established appropriate corrective actions. The actions had been identified as commitments and entered into the commitment tracking system. The inspectors did not verify the implementation of the corrective actions; however, the tracking system showed that the commitments had been performed and the SOERs closed out.

The inspectors selected three NRC INs that were shown by the licensee's printout "NRC Information Notices," to be open.

IN 87-34, dated July 24, 1987, was received by Nuclear Licensing and distributed to Design Engineering with a request for review under IDS RBC-36017 in conjunction with Engineering Evaluation and Assistance Request (EEAR) 87-R0370 dated August 10, 1987. The EEAR is a document used to identify, document, and request engineering assistance in resolving an operating, maintenance, licensing, or design problem. The subject of the IN dealt with single failures in auxiliary feedwater systems. Engineering's evaluation concluded that the IN was not applicable to boiling water reactors and the EEAR was completed on August 10, 1987. Subsequent review by the Independent Safety Engineering Group (ISEG) and Nuclear Licensing resulted in the issuance of a memorandum to engineering on November 2, 1989, in which a request for reconsideration of the original response was made. The memorandum stated that the real concern of the IN dealt with a common mode failure that might disable multiple injection pumps and that this should be considered for applicability to RBS. EEAR 91-R0089 was initiated on September 20, 1991, requesting an additional response from Design Engineering. The requested due date for the response was established as October 21, 1991. The records indicated that no further response was received by Nuclear Licensing. There were no ready explanations as to why it took over two years to determine that an additional engineering response was necessary or why engineering had not responded.

IN 90-33 dated May 9, 1990, was received by Nuclear Licensing and distributed to the Plant Staff Department (Plant Manager) with a request for review under IDS RBC-39706 dated May 31, 1990. The IN dealt with unexpected occupational radiation exposure at spent fuel storage pools. There were no additional records in Nuclear Licensing's files to show that any response had been received from the responsible department head. Additionally, the inspectors noted that while the licensee generated "NRC Information Notices" printout showed this IN to be open, Nuclear Licensing's computer tracking system for all licensing documents (TASK) showed the IN to be closed. This situation was compounded in that Nuclear Licensing, as a result of Quality Assurance Surveillance ES-90-10-06 performed during September 1990, agreed to a quality assurance recommendation regarding the issuance of a report that would identify overdue documents requiring evaluation and response. The inspectors' review of the latest bimonthly report dated February 10, 1992. revealed that IN 90-33 was not listed as being open.

IN 90-41, dated June 12, 1990, was received by Nuclear Licensing and distributed to the responsible department (engineering) on July 16, 1990, on IDS RBC-39814. EEAR 90-R0104 was initiated on September 6, 1990, and completed on November 20, 1991. Nuclear Licensing's Statement of Action had not been completed and was in draft form. Engineering's response, while not timely, did show that corrective actions had been identified and were being implemented.

The inspectors further reviewed the NRC Information Notice printout and the latest bimonthly report and noted that approximately 47 INs were shown to be open and the response due dates had been exceeded by as few as 30 days and by as much as 2 years.

The inspectors reviewed seven PRCs (88-07, 89-07, 90-02, 90-12, 90-17, 91-03, 91-09) and one Service Bulletin (SB 9012 from Target Rock Corporation) in order to assess the evaluations and timeliness of responses. The Service Bulletin and all of the PRCs, with the exception of 91-09, appeared to have been handled expeditiously. The inspectors considered the evaluations to be logical and thorough. While not all of the responses were generated within 30 days, the subject matter was not related to operability and evaluations could not be performed until the items could be disassembled. PRC 91-09 was a notification from Cooper Industries dated January 31, 1991, regarding information they had received from NEI Peebles-Electric Products, Inc., that identified a potential defect. That information was contained in a NEI Peebles-Electric Products, Inc. 10 CFR Part 21 notification to the NRC dated January 4, 1991, which addressed failed welds in emergency diesel generator air baffles resulting in loss of design required rigidity. RBS was identified as a recipient of equipment that might be affected. Nuclear Licensing, upon receipt of the information, initiated an IDS on April 9, 1991, which was distributed to engineering for evaluation. As of this inspection, the required response had not been received by Nuclear Licensing and this item was still open.

The inspectors, in order to get a better understanding of PRC status, requested the licensee to generate a PRC status report. This report, dated March 27, 1992, provided the following information. To date, there have been a total of 215 PRCs received and distributed by Nuclear Licensing, of which 197 have been responded to by the designated responsible departments. Of the 18 PRCs for which no response has been received by Nuclear Licensing, 17 are in excess of 30 days, 5 exceed 1 year and 1 exceeds 2 years. Further, the status report showed that of the 197 responses received from the responsible departments, Nuclear Licensing had completed just 82 Statements of Action. The inspectors did not determine the delinquency status of the 115 Statements of Action which had not been initiated.

The above examples, showing that the designated responsible departments failed to provide the requested responses within the specified time, or to establish a response schedule, comprise an apparent violation of Criterion V of Appendix B to 10 CFR Part 50 (458/9209-01).

3. INSERVICE INSPECTION - PROGRAM AND REVIEW OF PROCEDURES (73051 and 73052)

The purpose of the inspection was to ascertain whether the licensee's procedures pertaining to the preservice inspection (PSI) and inservice inspection (ISI) adequately cover all required aspects of the approved ISI program.

The inspectors were informed that the status of the NRC approval of the River Bend Station ISI Plan was unchanged from the previous inspection documented in NRC Inspection Report 50-458/91-28.

The inspectors were provided a list of those GSU and contractor procedures which had been approved by GSU for performing PSI and ISI nondestructive examinations scheduled for the current refueling outage. The inspectors were informed that there were other contractor procedures which had not completed the review and approval cycle, such as, the automated ultrasonic examination procedures. From the list of GSU and contractor procedures designated as approved by GSU, the inspectors selected the following for review: Procedure QCI-3.12, "Magnetic Particle Examination (MT) Dry Method," Revision 5; Procedure QCI-3.13, "Liquid Penetrant Examination (PT)," Revision 6; Procedure QCI-3.35, "Magnetic Particle Examination (MT) Fluorescent Method," Revision 3 and including Change Notices through CN-3.35-3-2; Procedure GS-MT-W81-1, "Magnetic Particle Examination of Welds & Bolting," Revision 4 and Addenda No. 1; Procedure GS-PT-W81-1, "Liquid Penetrant Examination," Revision 5; Procedure GS-UT-W81-6, "Ultrasonic Manual Examination of Class 1 & 2 Bolts and Studs," Revision 3; and Procedure GS-UT-W81-12, "Ultrasonic Manual Examination for Detection of Cracking in Alloy 182 Nozzle Weldments," Revision 1. The procedures were reviewed for consistency with the requirements of the 1980 Edition of the ASME Boiler and Pressure Vessel Code Section XI with addenda through Winter 1981. One anomaly was identified in regard to the procedure for performing fluorescent MT, QCI-3.35. This procedure was noted to contain acceptance criteria for examinations performed on pressure retaining welds in piping (Category B-J) which differed from the acceptance standard specified in Subsection IWB-3514 of Section XI. For example, the procedure allowed indications of 1/4 inch in length for PSI examinations of piping with nominal wall thickness of 1 inch, where as the Code only permitted indications of 3/16 inch in length or less. The inspectors were informed that the fluorescent MT procedure was only used for examining reactor vessel bolting materials and, although the procedure was not used for examining piping welds, the acceptance criteria specified for piping would be corrected so that it was consistent with the Code acceptance standard for performing preservice and ISI examinations. The inspectors reviewed the other four surface examination procedures specified above and found the acceptance standards to be consistent with the Section XI acceptance standard for piping welds. Therefore, the inspectors considered the issue satisfactorily resolved. In discussing Procedure QCI-3.35 with GSU's Level III examiner, the inspectors were informed that for those examinations where Section XI of the Code does not provide an acceptance standard, such as the reactor vessel nuts, the acceptance standard in Section III of the Code would be specified by the Level III using the provisions specified in the examination procedure. The inspectors considered this to be a satisfactory approach.

With the exception noted above, the inspectors found the ISI examination procedures to be well written and consistent with the requirements of the Code. However, the inspectors noted that the ISI instructions and approvals could be enhanced by specifying the procedure which would be used to perform a specific ISI examination in the ISI Plan.

4. NUCLEAR WELDING GENERAL INSPECTION PROCEDURE (55050)

The purpose of this inspection was to determine whether the licensee's welding specification and procedures for replacement of the N4A feedwater nozzle safe end and thermal sleeve meet applicable ASME Code, regulatory, and contract requirements.

The inspectors observed the full size mockup which had been constructed to simulate the N4A feedwater nozzle safe end and thermal sleeve conditions. The inspectors were informed that the mockup would be used for training of personnel required to remove the existing N4A feedwater nozzle safe end and thermal sleeve and installation of a new feedwater nozzle safe end and thermal sleeve of a different design. In order to reduce the radiation exposure during repair of the feedwater nozzle and installation of the new nozzle safe end and thermal sleeve, the welding will be performed using automated welding equipment with remote controls. The mockup training will also be used for refinement of the replacement techniques, craft training, and equipment checkouts. The inspectors were informed that the welding procedure specification and postweld heat treatment procedures had not been completed but the welding procedure qualification and training of welders were now in process.

The inspectors reviewed the licensee's specification for the replacement of the feedwater nozzle safe end, GSU Specification No. 221.101, "Design Specification for Feedwater System's N4A Nozzle Safe End and Thermal Sleeve Replacement," Revision 1; and the following documents in preliminary status that had not been released for utilization in work activities. Documents in the review and approval process included: GE Nuclear Energy Certified Stress Report DC25A5110, "Replacement Feedwater Safe End and Thermal Sleeve Analyses," Revision 1; GE Nuclear Energy Specification No. 25A5080, "Preheat and Postweld Heat Treatment of Feedwater Nozzles," Revision 1; GE Nuclear Drawing No. 107E6099, "Feedwater Safe End Replacement," Sheets 1 through 4, Revision 1; GE Nuclear Drawing No. 112D4999, "Safe End," Revision 0; GE

Nuclear Drawing No. 112D6000, "Thermal Sleeve Transition," Revision 0; GE Nuclear Drawing No. 112D6001, "Thermal Sleeve Assembly," Revision 1; and the preliminary instructions drafted by Welding Services, Inc., detailing the steps and sequences which will be performed to accomplish the removal and replacement of the feedwater nozzle safe end and thermal sleeve. In addition, the inspectors reviewed the following documents: GSU's request to NRC dated January 31, 1992, for the use of ASME Section III Code Case N-483, "Alternate Rules to the Provisions of NCA-3800, Requirements for the Purchase of Material, Section III, Division 1," pursuant to 10CFR50.55a; and the preliminary results dated February 11, 1992, of an independent technical review of the stress analysis and specification for the postweld heat treatment of the N4A feedwater nozzle performed by MPR, Inc., in accordance with Purchase Order No. 91-L-74996.

The inspectors determined that the overall approach to the feedwater nozzle repair activity was well thought out and conceptually sound. The building of a full size mockup was considered commendable and should contribute to achieving ALARA goals.

The inspectors noted an inconsistency with respect to the temperature tolerances specified in the preliminary specification for postweld heat treatment of the feedwater nozzle repair activity. The inspectors also questioned the wording in the postweld heat treatment specification that permitted the nozzle exceed 1200°F for up to 1/2 hour during postweld heat treatment. Licensee personnel indicated that these anomalies had been recognized and would be corrected prior to final approval and release for use of the specification.

5. EXIT INTERVIEW

An exit interview was conducted on March 27, 1992, with the personnel denoted in paragraph 1 in which the inspection findings were summarized. The licensee did not identify as proprietary any of the materials provided to, or reviewed by, the inspectors during this inspection.