APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-382/89-11 Operating License: NPF-38

Docket: 50-382

Licensee: Louisiana Power & Light Company (LP&L)
317 Baronne Street
New Orleans, Louisiana 70160

Facility Name: Waterford Steam Electric Station, Unit 3 (W-3)

Inspection At: Taft, Louisiana

Inspection Conducted: March 27-31, 1989

Inspectors: P. Wagner, Reactor Inspector
D. Hunter, Senior Reactor Inspector
M. Murphy, Reactor Inspector

Accompanying
Personnel: S. Klein, Consultant, ERCI
S. Kobylarz, Consultant, ERCI

Approved: I. Barnes 5-8-89
I. Barnes, Chief, Materials and Quality Date
Programs Section, Division of Reactor Safety

Inspection Summary

Inspection Conducted March 27-31, 1989 (Report 50-382/89-11)

Areas Inspected: Routine, announced inspection consisting of evaluating the engineering and technical support activities and the QA audits of those activities. The engineering organization was evaluated for size, workload, qualification, and training. The quality of the engineering performed was evaluated by reviewing completed station modification and design change work packages. The QA audits of the engineering and technical support organization were also reviewed, and the corrective actions taken with respect to the audit findings were evaluated.

Results: Within the engineering and technical support activities, two examples of violations related to inadequate design review were identified (paragraphs 2.c.(1) and 2.c.(2)). Both of the examples were identified when the NRC inspectors reviewed completed design changes and discovered a discrepancy in the original design. Although the implementing procedures were good, the licensee's engineering evaluation summaries did not, in general, contain sufficient detail to ascertain the significance of the project being

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evaluated. In addition, the NRC inspectors identified two instances in which normally accepted levels of conservatism did not appear to have been utilized.

Within the area of QA audits, one unresolved item was identified (paragraph 3.c) pertaining to adequacy of corrective actions taken in response to identified discrepancies. The NRC inspectors found that the QA audits and surveillances of the design function had been of sufficient scope and depth to detect the engineering weaknesses in most all cases. The areas of audit checklists, documentation of findings (observations and closed findings), tracking and trending, and notification of management could be strengthened.

DETAILS

1. Persons Contacted

LP&L

D. Baker, Nuclear Operations Support Manager
T. Brennan, Design Engineering Manager
A. Briody, Nuclear Operations Engineering and Construction (NOEC) Deputy Assistant
N. Carns, Plant Manager
M. Ferri, Supervisor, Modification Engineering
S. Fisher, Principal Oversight Engineer
C. Gaines, Events Analysis Supervisor
T. Gray, Operations Quality Assurance (QA) Supervisor
J. Holman, Safety and Engineering Analysis Supervisor
J. Howard, Procurement Engineering Manager
D. Klinksiek, Supervisor, Mechanical/Civil Design Engineering
L. Laughlin, Site Licensing Supervisor
G. Koehler, QA Audit Supervisor
J. McGaha, NOEC Manager
B. Thigpen, Nuclear Operations Construction Manager

NRC

T. Staker, Resident Inspector

The above personnel attended the exit interview conducted on March 31, 1989. The NRC inspectors also contacted and interviewed other licensee personnel during the course of the inspection.

2. Engineering and Technical Support Activities

The NRC inspectors evaluated the effectiveness of the LP&L nuclear engineering program in the areas of adequacy of staffing levels and experience, training, design changes, and QA audits. The evaluation consisted of documentation reviews and personnel interviews to verify that the license requirements included in the Technical Specifications (TS) and codes and standards were being implemented and that the commitments contained in the Updated Safety Analysis Report (USAR) and other correspondence were being followed.

a. Organization and Staffing (37702 and 40703)

The NRC inspectors reviewed the Nuclear Operations Engineering Procedure (NOEP) manual, the Nuclear Operations Engineering Instruction (NOEI) manual, and selected nuclear operations administrative procedures, a partial listing of which is included in an Attachment to this report. This review verified that administrative controls had been established which described the responsibilities, authority, and lines of communication for personnel performing support functions. These functions consisted of design, technical support, quality assurance, construction, and procurement. The procedures reviewed were in conformance with 10 CFR Part 50, Appendix B, and the licensee's approved QA program.

The licensee's organization was structured so that the Nuclear QA Manager and Vice President-Nuclear reported to the Senior Vice President Nuclear Operations. Reporting to the Vice President-Nuclear were the Nuclear Plant Operations Manager and the Nuclear Operations Engineering and Construction (NOEC) Manager. Reporting to the Nuclear Plant Operations Manager was the Assistant Plant Manager Technical Services (APMTS). The major engineering and technical support functions reported to the NOEC manager and the APMTS. These functional areas were all located at the W-3 site. The organizational alignment at the time of the inspection, reflected the management reorganization that became effective August 8, 1988.

Plant engineering reported to the APMTS and was composed of two groups; these were systems engineering and reactor engineering and performance. Systems engineers performed all aspects of interim modifications, became involved in and tracked regular modifications, and acted as test engineers for postmodification test performance. Reactor engineers were involved in fuel handling, poststartup physics tests, and surveillance tests on core performance.

Procurement/programs engineering, construction, modification control, safety and engineering analysis, and design engineering reported to the NOEC manager. Design engineering was composed of mechanical/civil, electrical, and instrumentation and control engineering.

The total engineering complement, at the time of this report, was 144 individuals with specialized experience. Members of the engineering staff were required to have, as a minimum, a Bachelors degree in engineering or in an appropriate field. The licensee stated that the average experience of the engineering staff was 12.4 years with 4.39 years average experience at W-3. The engineering turnover rate in 1987 was 2 percent and in 1988 was approximately 8 percent.

The NRC inspectors determined that the engineering workload was level and manageable in all areas except design engineering. The heaviest backlogs in this area were old modification package closeout and resolving plant engineering information requests. Major modifications for the next outage were scheduled to be handled by consultants. The licensee's goal was to become an independent design engineering group. To this end, the engineering organization had moved from project to discipline type engineers with a significant increase in staff size over the past 2 years. The existing consultant work was being integrated into the LP&L organizations with direct supervision by licensee personnel. Consultant group efforts that function independently were assigned to a senior engineer as coordinator/interface.

No violations or deviations were identified in the area of organization and staffing.

b. Training (40703)

The NRC inspectors reviewed the licensee's training program for the W-3 technical staff. This program had been INPO certified. All new technical staff members were required to complete "Introductory Training" within the first 6 months. Upon completion of the required training, the training department offered courses in applied fundamentals and plant systems. Assignments for further training were developed by the staff supervisors dependent on assignment, experience, and need.

No violations or deviations were identified in the area of training.

c. Design Changes and Modifications (37700 and 37701)

At W-3, conditions requiring engineering evaluation included station modifications (SMs), (recently instituted) design change packages (DCPs), nonconformances (NCRs), and spare parts equivalency evaluation reports (SPEERs). The NRC inspectors selected a sampling of modifications to the plant design for review, including SMs and DCPs. In general, the modifications which had been made did not involve major changes to the plant design and were limited in scope. The following modifications were reviewed in detail:

- DC 3056, Revision 2, dated January 17, 1989, "Installation of Pressure Bleed-Off Valve for Air Accumulator Check Valve Test"
- SM 896, Revision 5, dated January 22, 1988, "CCW Surge Tank Vent Isolation"
- DC 3005, Revision 0, dated February 26, 1988, "Provide Throttle Capability for SI-135A and SI-135B"
- SM 1432, dated June 13, 1986, "Add Throttle Settings for SI Throttle Valves"
- SM 83-0035, "Replace bypass Transformers for ELGAR Inverters, SUPS 3A-S and 3B-S"
- SM 84-0511, "Frequency and Alarm Trip Setpoints for SUPS 3MA-S, 3MB-S, 3MC-S, and 3MD-S"

The NRC inspectors found that the modification descriptions, Safety Reviews, and Safety Evaluations (performed to meet 10 CFR 50.59), were not detailed and did not always clearly describe the changes being made. Safety evaluations did not always include sufficient rationale to substantiate the conclusions established. Without improvement in descriptive details and substantiation of the safety evaluations, future design changes could be made without consideration of important safety elements affected by the modification being evaluated. In addition to these general observations, the NRC inspectors identified concerns related to the modifications discussed below:

(1) DC 3056

Valves SI-602A&B are air actuated, safety-related, containment sump recirculation isolation valves. These valves are required to be closed during the safety injection phase and open on a recirculation actuation signal (RAS) during a design basis accident. The valves are positioned by the instrument air system, which was not safety-related or seismically qualified. In the event of a loss of instrument air, an air accumulator is supposed to supply air to operate its associated valve. A check valve was located immediately upstream of each accumulator to maintain accumulator pressure.

A modification was performed to install a bleed-off valve upstream of the check valves to permit depressurization of the air line when isolated from the instrument air system to facilitate periodic leak testing of the check valves. The NRC inspectors identified weaknesses in the postmodification testing, the analysis performed to establish leakage acceptance criteria, and a concern related to the design basis for the size of the accumulator tank.

LP&L Calculation EC-M89-014 (Revision 0, dated March 6, 1989, Allowable Air Leakage Rate - Valves SI 602A&B) was performed to determine the maximum allowable leak rate from the air accumulator system for operating Valves SI 602A&B. The input criteria for the calculation indicated that the accumulators were sized to cycle the valves once (one stroke closed followed by one stroke open) in 1 hour based on a referenced Ebasco Specification LOU 1564.109A. While LP&L was unable to provide the basis for this requirement, the NRC inspectors concluded that the 1 hour was based on the time required to generate the RAS signal during a design basis accident, which is nominally on the order of 20 minutes. (This is the time necessary to pump down the refueling water storage pool in a postulated large break loss of coolant accident (LOCA)). In that case, the air accumulators were sized to sustain leakage for up to 1 hour while maintaining sufficient capacity to operate the valves upon demand.

However, during a small break LOCA, the time required to drain the refueling water storage pool could be substantially greater than 1 hour. The NRC inspectors were concerned that the accumulators were not adequately sized for these accidents.

In their February 21, 1989, response to NRC Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," the licensee indicated that the possibility that "these valves may need to operate beyond the present 60 minute limit" was being further reviewed.

In addition to the above concerns, the NRC inspectors determined that the analysis to establish allowable leakage rates for check valve leak testing was not conservative. The calculation (EC-M89-014 above) used an average air system pressure to determine allowable leakage, rather than the lowest anticipated operating pressure (95 psig), and no margin was applied to the calculated allowable leakage (pressure decay rate) for use in the test. Consequently, the full calculated decay rate of 9 psi/hour was utilized as the acceptance criteria for leak testing (OP-903-032, Revision 6). Using the methodology in the calculation, the NRC inspectors independently determined that the allowable pressure decay rate would be less than 8 psi/hour based on the lowest system air pressure of 95 psig. Testing performed on March 27, 1989, for these check valves, showed pressure decay rates of 6 and 7 psi/hour; following repairs, testing subsequent to the inspection showed decay rates of 0.5 psi/hour on both valves.

The DCP included a brief Acceptance Test performed for the modification to "verify capacity of accumulators to cycle SI-602A(B) after IA isolated for 1 hour." The test consisted of depressurizing the air system between the isolation valve and the check valve by opening the new bleed valve and stroking Valves SI-602A&B after 1 hour. The NRC inspectors found that:

- There were no initial or minimum instrument air pressures specified for the test initial conditions,

- There were no provisions to record pressures observed or time required to stroke the valve, and

- The only documentation required to confirm satisfactory valve performance was a check mark on the data sheet that the results were "SAT" or "UNSAT."

The NRC inspectors reviewed the data from the DCP required test and noted that some data not specifically required by the test procedure had been recorded. The data indicated that the valves could fully stroke at 95 psig. However, the initial pressures recorded were higher than the lowest anticipated air system operating pressures. In addition, a handwritten note on the margin of the data sheet stated that the pressures were approximate since the "needle in pressure gage tends to stick occasionally."

Surveillance Procedure OP-903-032, Revision 6, was performed on Valves SI-602A&B to confirm that the valves' stroke times were within acceptable limits. While the data indicated that stroke times for both valves were within limits, there was no basis for the 15 second stroke time acceptance criteria used in the test. LP&L personnel informed the NRC inspectors that vendor recommended stroke times were used, if no other requirement was identified. The NRC inspectors found that the vendor (Fisher Controls data sheet for the valves) indicated a valve stroke time of 5 seconds, maximum. Using this criterion, Valve SI-602B, which tested at an actual stroke time of 6 seconds, would have failed the test.

As a result of these observations, the NRC inspectors were concerned that:

° The accumulators may not have adequate capacity to operate these critical safety-related valves on demand.

° Weaknesses may exist in the definition of postmodification testing requirements and acceptance criteria at W-3.

° This modification to the plant design appeared to have been made without an adequate review of the original design basis for the system to confirm that a sound basis existed for the changes being made.

In response to the NRC inspectors' concerns, LP&L personnel indicated that corrective actions had been initiated relative to the valves and accumulators. These actions included:

(a) A Non Conformance Report Condition Identification (NCRCI) was written to determine if the valves would operate after more than 1 hour; and

(b) Administrative procedures were initiated to maintain instrument air pressures above 107 psig header pressure. The 107 psig header pressure was based on an LP&L evaluation, performed after the inspection, of small break LOCAs, which could require more than 1 hour to drain the refueling water storage pool. In addition, these administrative procedures would include a set of curves depicting allowable system pressure decay rate versus required system pressure to assure two full strokes of the recirculation valves for at least 4 hours subsequent to a loss of instrument air. The 4 hour requirement was based on the LP&L evaluation which determined that the refueling water storage pool would be drained in less than 4 hours for break sizes exceeding .01 square foot. Two full strokes of the valves were required to account for the possibility that the valves could be open prior to the accident. The curves would also form the basis for establishing monthly surveillance testing allowable leak rate acceptance criteria.

The licensee discussed the above commitments with NRC Region IV and NRR personnel subsequent to the inspection. In addition to refining those commitments, LP&L personnel stated that an administrative requirement had been established for the instrument air system; i.e., if system pressure were to reduce to less than 90 psig and could not be restored to greater than 90 psig within 1 hour, the plant would be shutdown.

Violation: The failure to control the design of the accumulators adequately in order to provide assurance of containment sump recirculation valves (SIs 602A and 602B) operation is an apparent violation (382/8911-01).

(2) SM 83-0035

SM 83-0035 upgraded the bypass transformers for the Class 1E static uninterruptible power supplies (SUPS) 3A-S and 3B-S. Although no problems were identified with this modification, the NRC inspectors identified a potential problem with the availability of power from the inverter during conditions when a fault occurred on a branch feeder circuit fed from a downstream distribution panel.

The NRC inspectors reviewed the instruction manual for SUPS 3A-S and 3B-S, which were manufactured by ELGAR, and determined that the model UPS 103-1-151 inverter would shut itself down if the inverter output was subjected to a fast overload condition. A "fast overload condition" shutdown could occur at an inverter load current greater than 165 percent of rated, as noted in paragraph 4-4.8.1 of the manual, which could result from a fault condition on the output of the inverter. An inverter shutdown on fast overload would be automatic and would last for approximately one-half second before the inverter would restart. Upon restart after a fast overload, the output voltage of the inverter would be slowly "ramped-up," and this could take on the order of seconds to complete.

Since the design of the SUPS were such that the inverter bypass power source was utilized only during maintenance of the inverter and since the bypass source could only be connected to the distribution bus by manual operator action, the Class 1E distribution bus powered by the inverter would be temporarily de-energized during the fast overload or fault condition described above.

FSAR Section 8.3.1.1.2.11, "Electric Circuit Protection Systems," stated that the electrical protection had been designed for selective tripping, so that only the affected circuit, close to the point of fault, would be isolated. During a fast overload, the inverter shutdown mode of operation would effectively be racing the operation of the downstream circuit protective devices (both molded case circuit breakers and fuses) to interrupt the fault. The licensee's engineers contacted the inverter manufacturer, during the inspection, to confirm whether the inverter would or would not go into a shutdown mode for a fast overload or fault condition. The inverter manufacturer could not confirm that the inverter would not shutdown during fast overloads such as fault conditions. Since the electrical overload protective devices for the inverter and the distribution system branch circuits were apparently not coordinated for all fault conditions, the design did not appear to satisfy the FSAR commitment for selective electrical protection.

The NRC inspectors were concerned that the vital distribution system powered from SUPS 3A-S and 3B-S could be subject to a common loss of power event, as a result of fault conditions, if the Class 1E power distribution panels powered Non-Class 1E circuits. Non-Class 1E circuits may fail during design basis accidents, such as a seismic event, causing a loss of power in both of the redundant safety/shutdown control trains if each redundant train powered Non-Class 1E control circuits. The licensee's design engineers reviewed the loads on the distribution panels powered by the SUPS 3A-S and 3B-S and found that Power Distribution Panel (PDP) No. 390-SA, supplied by SUPS 3A-S, powered one Non-Class 1E control circuit. Drawing No. LOU1564, B-289, Sheet 147A, Revision 8, for PDP 390-SA showed a Non-Class 1E Telephone Cabinet (PEC) was powered from circuit No. 65. This was the only Non-Class 1E circuit found on the SUPS 3A-S and 3B-S distribution systems.

The NRC inspectors also reviewed the Associated Circuit Analysis Calculation EE5-32-02, Revision 0, which was prepared as part of the licensee's fire protection program. Part 4B of the analysis demonstrated the selectivity and the coordination of the electrical protective devices on the 120V AC SUPS panels. Part 4B.3.1 stated that the SUPS 3A-S and 3B-S operated in a similar manner as SUPS 3MA-S, 3MB-S, 3MC-S, and 3MD-S, which were manufactured by Solidstate Controls, Inc. The NRC inspectors found this analysis to be incorrect in that the fast overload shutdown mode of the ELGAR SUPS (3A-S and 3B-S) resulted in a loss of power; licensee personnel stated that the inverters supplied by Solidstate Controls, Inc. did not exhibit the automatic fast overload shutdown feature of the ELGAR inverters.

During the course of the inspection, the licensee initiated NCRCI 262261 to address the potential loss of power event for SUPS 3A-S during design basis accident conditions. At the exit meeting, the licensee indicated that the immediate corrective action would be to relocate the feeder for the PEC from SUPS 3A-S to the Non-Class 1E computer SUPS.

Violation: The failure to adequately control the design of the SUPS 3A-S loading is an additional example of the apparent violation discussed in paragraph 2.c.(1) above (382/8911-01).

(3) SM 896

This modification installed a check valve in the component cooling water (CCW) surge tank vent line to prevent waste gas activity from entering the tank. A vacuum breaker was also added to the line to preclude excessive vacuum pressures which might result from reductions in tank level.

The NRC inspectors found that the modification package contained no documented analysis to substantiate the size of the vacuum breaker. In addition, there was no analysis to demonstrate that the size of the vent line was adequate. The licensee initiated calculations to determine the maximum vacuum pressure the tank could safely accommodate and stated, during the exit meeting, that the results indicated the tank could sustain external pressures near 125 psig. The NRC inspectors had no further concerns with the capability of the tank to sustain vacuum pressure levels. However, this was an example which contributed to the NRC inspectors' concern that modifications to the plant design could be made without an adequate review of the design basis for the systems being changed.

3. Audits of the Support Functions (37702)

The NRC inspector reviewed applicable QA program procedures, audits, surveillances, and corrective actions associated with plant modification activities.

a. QA Audits-Modifications

The licensee scheduled annual audits of selected design control activities. Two audits were scheduled and performed; one in January - February 1987, and the other during September - November 1988. The NRC inspector reviewed these two audits for scope, content, and auditor complement. The QA audits appeared to be acceptable and were performed by lead auditors with additional personnel assigned to the audit team. The certifications of the auditors were reviewed by the NRC inspector and found to be acceptable. The audit schedule was provided to the Safety Review Committee (SRC) subcommittee for concurrence in order to ensure SRC cognizance of the QA audit program. The completed audits were distributed to the audited organization and other licensee management personnel.

The review of the detailed audit checklists by the NRC inspector revealed that the design activities were, in most instances, checked only against the procedure requirements and not against both the program and procedure requirements. Interviews revealed that the procedures were considered to be acceptable, based on previous QA group independent reviews of the procedures and procedure changes. The audit checklists could have been more comprehensive (e.g., included a comparison of selected modification procedures/procedure changes with the QA program and regulatory requirements). This matter was discussed with the licensee representative for consideration. Interviews revealed that the licensee was in the process of including members of the SRC on the audit team to strengthen the overall audit program and to increase SRC involvement on the performance of the QA audits.

The NRC inspector reviewed the audits to verify that the audit results and effectiveness of the QA program associated with the area audited were addressed. The specific documentation of the summary of the audit results was not apparent; however, interviews with licensee representatives revealed that the results of the audit were considered to be indicated by the audit findings (or lack of negative findings) and represented the effectiveness of the QA program elements which were audited. This observation was discussed with the licensee representatives, in that an annual or biennial summary of audit results provided to management based on the QA audits, QA surveillances, and the associated findings would likely provide a significant enhancement to the effective implementation of the overall QA program.

Discussions with licensee personnel and records review revealed that the QA group performed additional, special audits as requested. Also unscheduled audits of an onsite contractor's (Paul-Monroe) activities were routinely audited. Unscheduled and special audits were addressed in the Nuclear Operations Management Manual (Section VII, Chapter 9, and Section V, Chapter 18); however, the quality program lacked specific definition of unscheduled and special audits. This item was discussed with the licensee for consideration. The NRC inspector had no further questions regarding this matter.

b. QA Surveillances - Modifications

Document review and personnel interviews conducted by the NRC inspector revealed that QA surveillances of specific activities were performed to supplement the QA audit program. The NRC inspector selected eight surveillances concerning modification activities for review. The surveillances were performed by QA group personnel when deemed appropriate or as a result of special requests. The NRC inspectors had no questions regarding this matter.

c. QA Audit and Surveillance Findings

The NRC inspector reviewed selected QA audit and surveillance findings to assess the reports and to determine if the dispositions of the findings were thorough and the corrective actions implemented in a timely fashion. In most instances, the findings were acceptable. The following comments were noted.

The review of QA scheduled Audit SA-88-005.1 (Organization and Quality Assurance Program) revealed that a finding regarding the apparent "misclassification of findings" had been identified by the licensee. The finding was documented on a quality notice (QN) (QA-88-036) and corrective action was nearing completion. The finding identified the practice of the QA group of identifying a "condition adverse to quality" and closing the finding without issuing a QN. Interviews and document reviews revealed that the licensee had identified this item previously and planned to change the procedures to require that a QN be issued to identify all "conditions adverse to quality" to ensure tracking and trending. The NRC inspector had no further questions regarding this item.

The review of QA scheduled Audit SA-87-006.1 (Station Modification/Design, January 28 - February 11, 1987) revealed that a finding regarding the established controls for safety-related structural welding inspection activities was identified. Similar deficiencies regarding structural and ASME welding activities had also been identified previously and a QN (QA-86-133, January 7, 1987, welding - testing and inspection) had been issued. Since the condition had been previously identified and the QN was still "open," an additional QN based on the specific audit findings was not initiated. Document review and interviews revealed that the earlier QN (QA-86-133), which addressed deficiencies concerning AWS-D1.1 and ASME requirements, was not revised, or modified to address the specific audit (SA-87-006.1) findings. In addition to the specific welding deficiencies, it was noted in QN QA-86-133 by QA that the findings had potential generic implications to other safety-related welding activities. The QA audit (SA-87-006.1) also identified a deficiency regarding the established controls for safety-related grouting activities and a finding was issued (QN QA-87-065 grouting-procedural controls).

The NRC inspector reviewed the documentation associated with the condition adverse to quality documented in QN QA-86-133. The NRC review revealed a protracted closure of the issues - March 1987 through final closeout of all identified matters in January 1989. The NRC inspector's review of the corrective actions associated with

QN QA-86-133 revealed that the finding placed certain station modification packages (SMPs), performed after 1985 through identification of the deficiency in January 1987, in question. The documentation review revealed that the engineering department performed a selected sample of SMPs (12) which had been completed during the identified time period; however, it was not apparent to the NRC inspector that the sample was representative of all completed work activities or the specific findings (observations) identified in QA audit SA-87-006.1. Further, the closeout of QN QA-86-133 by the QA group did not address the specific QA audit findings (SMP-84, SMP-1297, and SMP-195) nor, the generic implications (review of all applicable work activities completed during the period of concern to ensure that all the requirements were addressed) associated with the SA 87-006.1 audit findings. This apparent deficiency was discussed with the licensee for consideration.

Subsequent to the completion of the required corrective actions concerning the program and procedures in early 1988, a QA surveillance (QS-88-062) was conducted in May and June 1988, to check a specific modification (SMP-138, Reactor Vessel Water Level Indication System). This surveillance identified significant deficiencies concerning safety-related grouting activities (QN QA-88-084, grouting - attention to details and QN QA-88-085, grouting - procedural requirements and clarification) and safety-related welding activities (QN QA-88-082, welding - testing and inspections). The specific welding activity findings (i.e., failure to perform required nondestructive examination of welds) resulted in a formal report to the NRC (LER 88-022, Revision 1). The licensee performed the required nondestructive examinations during a forced outage in September 1988. Training of selected personnel was also provided.

The noted recurring conditions raise a concern regarding the adequacy of actions taken to preclude recurrence. This subject is considered an unresolved item pending further NRC review during a future inspection. (382/8911-02)

The review of QA scheduled Audit SA-88-005.1 (Organization and Quality Assurance Program, November 21, 1988 - February 10, 1989) revealed that a number of findings were identified, including the failure to identify recurring findings (QN QA-89-031) and the misclassification of findings (QN QA-89-036). Both issues were being actively pursued by the licensee to develop a mechanism to identify recurring QNs and to ensure that all conditions adverse to quality were identified and entered in a tracking and trending system. The practice of utilizing closed findings and observations in the tracking program was also noted by the NRC inspector during the review of selected audit and surveillance report.

The review of QA unscheduled Audit QA-88-003 (Modification and Test of Hydraulic Snubbers - April 8 - May 6, 1988) revealed that the audit results (findings and observations) were transmitted to the contractor (Paul-Monroe), licensee contract management, and licensee management. However, the licensee audit results were not sent directly to the vendor's audit group to provide information to be considered in the next vendor audit.

The review of QA scheduled Audit SA-88-006 (Design/Station Modifications - September 12 - November 29, 1988), revealed that the audit of the design activities identified a number of programmatic and procedural deficiencies as a result of the design program revisions which occurred in 1988. The deficiencies had been addressed by the licensee, including the finding regarding "interim modifications."

The review of QA surveillance QS-88-072 (Post Audit Sampling, June 25, 1988) revealed that an observation was documented regarding gas sample disagreement. The samples were required to agree within a factor of two. The samples were performed again on September 1, 1988, and the three samples were acceptable. The QA group maintained the QS open until the observation (deficient samples) was corrected and verified. This was an example where the condition adverse to quality could have been documented on a QN, to ensure specific and generic corrective action and tracking and trending of the finding. The licensee had identified the misclassification of findings on QN QA-89-036 and was pursuing the corrective actions.

The NRC inspector had no further questions regarding the above items.

d. Temporary Modifications

The NRC inspector reviewed selected controls established regarding temporary modifications to ensure that the licensee requirements were properly implemented. The review of Procedure UNT-5-004, Temporary Alteration Control, Revision 6, revealed the licensee provided controls of temporary jumpers, lifted leads, flanges, hoses, relays, and setpoints. The procedure addressed the review of proposed quality-related temporary alterations (modification/change) by the plant operations review committee (PORC) and approval by the plant manager (PM). For nonsafety-related alterations, immediate implementation could be accomplished, provided the PORC review, and PM approval was performed within 14 days after implementation. The procedure, Attachment 6.3, Section II, Item B.11, addressed the update of necessary drawings; however, the procedure did not address the update of necessary procedures. This item was discussed with licensee personnel for consideration and the NRC inspector had no further questions regarding this matter.

4. Unresolved Item

An unresolved item is one about which more information is requested in order to determine whether or not it is a violation, a deviation, or acceptable. One unresolved item concerning corrective action adequacy is delineated in paragraph 3.c of this report.

4. Exit Interview (32703)

The NRC inspectors met with the personnel identified in paragraph 1 on March 31, 1989, to discuss the findings and conclusion reached during the inspection. The licensee personnel acknowledged the findings. No information was presented to the NRC inspectors that was identified by the licensee as proprietary.

ATTACHMENT

LIST OF DOCUMENTS REVIEWED

Drawings No.	Title
LOU1564, G-285	Main One Line Diagram
LOU1564, G-286	Key Auxiliary One Line Diagram
LOU1564, G-287	125V DC and 120V AC One Line Diagram
LOU1564, B-289, Sheets 147 & 147A	Power Distribution and Motor Data 120V Distribution Panel No. 390-SA
LOU1564, B-289, Sheets 148 & 148A	Power Distribution and Motor Data 120V Distribution Panel No. 391-SB

Specifications No.	Title
LOU1564.282A, Revision 9	Static Uninterruptible A-C Power Supply for Class 1E Control Systems
LOU1564.109A, Revision 4	Butterfly Valve Data Sheet, Revision 7, Sheet 16

Instruction Manual	Title
457000387, Volume 1	ELGAR Model UPS 103-1-151, Uninterruptible Power Supply
457000387, Volume 2	Instruction Manual for ELGAR AC Power Line Conditioner, Model PLC 253-1-04

Vendor Drawings	Title
1564-1897, Revision 5	Schematic, 20 kVA Inverter Solidstate Controls, Inc. (Drawing No. 014D10915, Sheet 1 of 2)
1564-1898, Revision 7	Schematic, 20 kVA Inverter Solidstate Controls, Inc. (Drawing No. 014D10915, Sheet 2 of 2)

Calculations No.	Title
EE5-32-02, Revision 0	Associated Circuit Analysis
EC-M89-014, Revision 0	Allowable Air Leakage Rate - Valves SI602A&B

Procedures

NOP-14, Revision 1, Design Changes

NOP-15, Revision 1, Justification for Continued Operation

QAP-000, Revision 4, Quality Assurance Charter

QAP-301, Revision 2, Quality Assurance Review of Station Modification Packages

QAP-302, Revision 6, Conduct of Operations Quality Assurance Audits

QAP-304, Revision 1, Quality Assurance Group Revision of Programs, Procedures, and Instructions

QAP-305, Revision 1, Planning and Scheduling of Operations Quality Assurance Audits

QAP-306, Revision 1, Conduct of Operations, Quality Assurance Surveillance

QAP-350, Revision 1, Review of Work Authorization Packages

QAP-366, Revision 0, Operations Inspections - General

PE-TEM-012, Revision 1, Plant Engineering Station Modification

UNT-5-004, Revision 6, Temporary Alteration Control

UNT-5-015, Revision 8, Work Authorization

QA Audits - Scheduled (SA) and Unscheduled (UA)

SA-87.006.1, Station Modification/Design, January 28 - February 11, 1987

SA-88-005.1, Organization and Quality Assurance Program, November 21, 1988 - February 10, 1989

SA-88-006, Design/Station Modifications, September 12 - November 19, 1988

UA-88-033, Modification and Testing of Snubbers, April 8 - May 6, 1988

SA-88-016.1, Fire Protection and Loss Prevention Program, December 2-9, 1988

QA Surveillances

QS-87-076, Yellow/Orange Jumper Wires, May 20-27, 1987

QS-88-050, Main Steam Isolation Valves, April 18 - May 22, 1988

QS-88-053, Station Modification (Paul Monroe), April 27 - May 6, 1988

QS-88-062, Station Modification Package 138, May 31 - June 20, 1988

QS-88-072, Postaccident Sampling System, June 25, 1987

QS-88-087, Acceptance of Station Modification 818, August 8-9, 1988

QS-88-089, Evaluations on SMP-1332, May 5-8, 1988

QS-89-010, Status of Safety-Related SMs and DCs, March 2-3, 1989

Other Documents

Condition Identification #262265, 3/30/89, Accumulators for SI Sump Outlet Isolation Valves Are Sized For One Hour Supply Of Air to Valve Operators (NOP-19)

OP-903-032, Revision 6, Surveillance Procedure, Quarterly ISI Valve Tests, Section 8.24 Instrument Air Check Valves (performed 3/27/89), Section 8.3, Safety Injection (1/24/89)

LP&L Letter to U.S. Nuclear Regulatory Commission, W3P89-0028, dated 2/21/89, Generic Letter 88-14