APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-382/87-29 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 (WSES)

Inspection At: Taft, Louisiana

Inspection Conducted: November 30 through December 4, 1987

Inspector: J. I. Tapia, Project Engineer, Project Section A, Division of Reactor Projects — 1/5/88 Date

Approved: J. P. Jaudon, Chief, Project Section A Division of Reactor Projects — 1/8/88 Date

Inspection Summary

Inspection Conducted November 30 through December 4, 1987 (Report 50-382/87-29)

Areas Inspected: Routine, announced inspection of previously identified items (one violation and two unresolved items), and implementation of licensee actions taken in response to NRC Compliance Bulletin No. 87-02.

Results: Within the two areas inspected, no violations or deviations were identified.

DETAILS

1. Persons Contacted

WSES

*G. Wuller, Operational Licensing Supervisor
T. Garrets, Nuclear Services Manager
*R. Bennet, Supplier Audits Quality Assurance (QA) Supervisor
B. Toups, QA Representative

Partek Laboratories

T. Blanchard, Marketing Manager
R. Sutton, Chief Engineer

*Denotes those present at the exit interview.

In addition to the above personnel, the NRC inspector held discussions with various operations, engineering, technical support, and administrative members of the licensee's staff.

2. Followup of Previously Identified Items

(Closed) Unresolved Item (382/8602-01): Procedure for Documenting Evaluations of Events - During an inspection in January 1986, the NRC inspectors found entries in the shift supervisor and reactor operator logs for the 7 a.m. to 3 p.m. shift on January 8, 1986, indicating that control element assembly (CEA) No. 1 had dropped with the reactor at approximately 40 percent power. The logbooks indicated that the electrical breaker was found open, was reclosed, the dropped CEA recovered, and power ascension continued. The NRC inspectors questioned licensee management at the time of the event on the extent of the evaluation of the incident since no evidence of a formal evaluation could be identified. It was explained that although an evaluation had been performed by the assistant plant manager for operations and maintenance, the operations superintendent, and the maintenance superintendent, it had not been documented. The plant manager committed to conduct a review to determine if a weakness existed in the licensee's event reporting and evaluation program. Since the dropped CEA and ensuing actions did not violate Technical Specification requirements, the dropping of the CEA did not constitute a reportable event. Nevertheless, the pending results of the licensee's review were considered an unresolved item.

This item was subsequently addressed in June 1986 (NRC Inspection Report 50-382/86-13). During that inspection, the NRC inspector reviewed the results of the licensee's evaluation of the event reporting and evaluation program. The inspection disclosed a memo from the plant manager to various members of plant management on the subject of determining accountability when investigating and reporting events and also disclosed various event reports documented by memos to file, including one for the dropped CEA incident. The NRC inspector noted that it appeared that an independent review of event reports would be appropriate in some instances and that the threshold for initiating a report was not clear in all instances. Also, the method of documentation required was not always clear (e.g., potential reportable event, condition identification work authorization, quality notice, memo to file, etc.). Based on these observations, the unresolved item remained open.

In response to these additional observations, the licensee established an Event Analysis and Reporting Group consisting of a Senior Engineer and two Associate Engineers and having primary responsibility for the investigation, documentation, and closure of Potentially Reportable Event (PRE) reports. Administrative Procedure No. UNT-6-010, Revision 4, "Event Notification and Reporting," provides the instructions to all plant personnel for reporting conditions potentially adverse to quality. The procedure also references Administrative Procedure Nos. UNT-5-002, Revision 7, "Condition Identification," and QP-015-001, Revision 3, "Nonconformance and Corrective Actions." These three documents provide the threshold and define the appropriate document to obtain resolution of reported conditions. In addition, Procedure No. UNT-6-010 now requires approval of PRE reports by the Plant Operations Review Committee (PORC). Discussions with the head of the Event Analysis and Reporting Group disclosed pending changes, as yet unfinalized, in the overall method for reporting problems. This licensee representative indicated that a generic problem report modeled after the Institute for Nuclear Power Operations (INPO) Human Performance Evaluation System is forthcoming. Based on the actions taken to date, the unresolved item concerning the procedure for documenting evaluations of potentially reportable events is considered closed.

(Closed) Unresolved Item (382/8606-01): Component Cooling Water Valve Checklist and Drawing Comments - During March 18-21, 1986, NRC inspectors reviewed the component cooling water system standby valve lineup and performed a walkdown of accessible portions of the system to verify operability. As a result of that inspection, discrepancies were noted between the valve lineup procedure and the affiliated drawings. Although all main flow path valves were correctly identified in the valve lineup procedure and were found correctly positioned during the walkdown, the noted discrepancies were considered an unresolved item. Previously, on January 9, 1986, the NRC inspectors performed a walkdown of the Essential Services Chilled Water System and generated comments similar to the discrepancies which are the subject of this unresolved item (see NRC Inspection Report 50-382/86-02, paragraph 8). At the time of the January inspection, the Assistant Plant Manager for Operations and Maintenance and the Operations Superintendent responded to the NRC inspectors' comments by describing an upgrading program for retagging all plant valves and reviewing and updating each safety-related valve line-up checklist. During following subsequent inspections, the NRC inspectors augmented the li discrepancies noted during ESF system walkdowns:

NRC Inspection Report No.	ESF System Walked Down
50-382/86-11	Auxiliary Component Cooling Water
50-382/86-15	High Pressure Safety Injection
50-382/86-17	Low Pressure Safety Injection
50-382/86-29	Emergency Feed Water
50-382/87-01	Emergency Diesel Generators

The upgrading program completion and correction of the deficiencies identified in the listed NRC inspection reports were reviewed during this inspection. Although the upgrading program was at one point expanded to include the labeling of all electrical breakers, only the quality of valve tagging and associated valve line-up procedures were addressed. The results of the upgrading program indicate that all noted discrepancies were corrected and that the program encompassed all safety-related systems. In order to ensure that the quality of valve tagging remains current, Procedure Nos. PMI-315, "Instructions for Dispositioning Detailed Construction Packages," and PE2-006, "Administrative Procedure - Plant Engineering Station Modification," require that all components be properly tagged before any modification is tested and placed in service. Based on the implementation of these procedural requirements and on the results of the valve line-up checklist walkdowns, this matter is considered closed.

(Closed) Violation (382/8634-01): Failure to Quantify Containment Isolation Valve Leakage - During an inspection conducted December 8-11, 1986, it was determined that the local leak rates for six valves required to be tested had recorded as-found leak results of "off-scale" with no recorded value, so that the continuous running sum of local leak rates was not known and the required comparison with the Appendix J criterion could not be made for the as-found condition. This was identified as a violation of the requirements of Appendix J, Section III.C.3, which states, in part, "The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.6 La." At the time that the local leak rate testing was being performed, the plant was in Mode 5, cold shutdown. The six valves in question had as-found leak rates which exceeded the capability of the test equipment. Prior to performing maintenance on the six valves, the local leak rate testing equipment was modified to provide a greater flow measuring capability. Corrective action was initiated by maintenance on all six valves and the appropriate retests were conducted. Of the six valves identified as having leak rates of "off scale," numerical values were identified for five of these valves. The sixth valve, a 24-inch containment vacuum check valve (CVR-102) could not be pressurized with the modified test unit, thus no leak rate could be established. In accordance with the requirements of Appendix J, it was assumed that the leakage for Valve CVR-102 would result in exceeding the 0.60 La limit. Accordingly, per the requirements of Appendix J, Section V.B.3, Valve CVR-102 is the subject of a separate accompanying summary report to the "Reactor Containment Building Integrated Leak Rate Test" report.

In order to prevent recurrence, Surveillance Procedure No. PE-5-002, "Local Leak Rate Test (LLRT)" was modified to require that, "After each individual test a cumulative leakage total shall be updated to reflect the new as-found leakage." In addition, the procedure now requires that if the leakage is higher than the current measuring capabilities, then it will be assumed that the 0.6 La limit has been exceeded and the applicable portions of Technical Specification 3.6.1.2 will be invoked. Based on the review of the actions taken in response to the Notice of Violation, this matter is considered closed.

3. Response to NRC Compliance Bulletin No. 87-02: Fastener Testing to Determine Conformance With Applicable Material Specifications

The licensee's ongoing actions taken in response to NRC Compliance Bulletin No. 87-02 were addressed during this inspection. The bulletin was issued to request that licensees review their receipt inspection requirements and internal controls for fasteners and determine, through independent testing, whether fasteners in storage meet the required mechanical and chemical specification requirements. The bulletin also required the participation of an NRC inspector in the selection of the sample for testing.

The NRC inspector's examination of ongoing actions included reviewing the licensee's receipt inspection program and procedures for safety-related and nonsafety-related fasteners to determine what characteristics are inspected and in reviewing the maintenance and warehousing procedures for the issuance and control of fasteners. The following site quality procedures, including historical revisions, were reviewed:

QI-010-006, "Materials Receipt Inspection"
QI-010-002, "Material Storage Inspection"
QI-004-001, "Site Review of Procurement Documents"
QI-004-002, "Records Review Checklist"
UNT-8-044, "Requisition and Return of Items to Stores"

The following QA procedures, including historical revisions were also reviewed:

QAP-208, "Procurement Administration"
QAP-250, "Materials Receipt Inspection"

The NRC inspector participated in a meeting held onsite between licensee representatives and personnel from Partek Laboratories. This meeting was held to discuss bulletin requirements and the testing schedule. The NRC inspector verified that the testing laboratory was on the licensee's qualified suppliers list and reviewed the results of the licensee's audit which was performed to provide the basis for the testing laboratory's qualification. It was agreed that LP&L would conduct a surveillance of all ongoing testing. The following safety-related sample site was decided upon after discussion of the Bulletin requirements and plant fastener usage:

Type	Quantity to be Tested	Approximate Percent of Use
A193, Grade B7	6	10
A193, Grade B8	6	10
A325, Type 1	24	45
A490	12	25
A307	6	10

The total sample of 54 safety-related fasteners was selected in increments of three to provide one chemical, one hardness, and one tensile test. For nonsafety-related fasteners, six lots of four fasteners each were selected for a sample site of 24. No tensile testing is required for nonsafety-related fasteners.

With the NRC inspector participating, the samples selected for testing were pulled from warehouse stock. The samples were bagged by lot and identified by type and quantity. The NRC inspector confirmed that the sample taken was properly tagged. The results of all tests, together with supporting information, will be reviewed during a subsequent inspection.

4. Exit Interview

The inspection scope and findings were summarized with those persons indicated in paragraph 1 at the end of this inspection.