U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-382/88-25 Operating License: NPF-38

Docket: 50-382

Licensee: Louisiana Power & Light Company (LP&L)
142 Delaronde Street
New Orleans, Louisiana 70174

Facility Name: Waterford Steam Electric Station, Unit 3

Inspection At: Taft, Louisiana

Inspection Conducted: September 17 through October 31, 1988

Inspectors: _____ 11/5/88
W. F. Smith, Senior Resident Inspector Date

_____ 11-1-88
T. R. Staker, Resident Inspector Date

Approved: Dwight D. Chamberlain 11-17-88
D. D. Chamberlain, Chief, Project Section A Date
Division of Reactor Projects

Inspection Summary

Inspection Conducted September 17 through October 31, 1988 (Report 50-382/88-25)

Areas Inspected: Routine, unannounced inspection of: (1) plant status, (2) onsite followup of events, (3) operational safety verification, (4) monthly maintenance observation, (5) monthly surveillance observation, (6) followup of previously identified items, (7) licensee event report followup, and (8) plant status.

Results: In the previous inspection period, as discussed in NRC Inspection Report 50-382/88-21, the NRC inspectors identified a number of examples which reflected weaknesses in the licensee's corrective action programs. This report discusses additional problems encountered as a result of inadequate and untimely corrective action. While it is understood by the NRC staff that this is a complex problem that will have a complex and time-consuming solution, the licensee should recognize that management emphasis must be placed on a priority basis to get this program under control as soon as possible.

There are also examples where licensee actions to achieve compliance to, and adequacy of, procedures apparently have not reached all disciplines. There was one violation identified in this area. The violation involved failure to follow procedures. A Safety Injection Containment isolation valve bypass, required by procedure to be locked closed, was found unlocked.

One new unresolved item was identified in paragraph 3.a, requiring further review to determine whether there was a violation of NRC regulations requiring a Notice of Violation to be issued pending review of the Architect-Engineer's analysis on whip restraints which did not meet design requirements.

DETAILS

1. Persons Contacted

LP&L

*R. P. Barkhurst, Vice President, Nuclear Operations
*N. S. Carns, Plant Manager, Nuclear
S. A. Alleman, Nuclear Quality Assurance (QA) Manager
P. V. Prasankumar, Assistant Plant Manager, Technical Support
D. F. Packer, Assistant Plant Manager, Operations and Maintenance
J. J. Zabritski, QA Manager
D. E. Baker, Manager, Nuclear Operations Support and Assessments
J. R. McGaha, Manager, Nuclear Operations Engineering
W. T. Labonte, Radiation Protection Superintendent
*G. M. Davis, Manager, Events Analysis Reporting & Responses
*L. W. Laughlin, Onsite Licensing Coordinator
D. W. Vinci, Maintenance Superintendent
A. F. Burski, Manager, Nuclear Safety and Regulatory Affairs
R. S. Starkey, Operations Superintendent
*R. L. Azzarello, Modification Control Manager

*Present at exit interview.

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

2. Plant Status (71707)

At the start of the inspection period, on September 17, 1988, the plant was operating at 90 percent power. This power level was being maintained in order to minimize possible damage from noise believed to be originating from the No. 2 Steam Generator.

On September 23, the plant was shut down to hot standby (Mode 3) to install instrumentation for the No. 2 Steam Generator noise investigation. The plant was back on the grid by September 25 and at 90 percent power by September 26. On September 29, 1988, power was increased to 100 percent to take noise data and then returned to about 90 percent power.

The No. 2 Turbine Governor Valve failed shut with No. 1 already shut on October 20, 1988. Consequently, power was reduced and maintained at 75 percent.

The plant was shut down and cooled down on October 21, 1988, in order to investigate the No. 2 Steam Generator noise. The plant entered Mode 5 on

October 22, 1988, and remained in Mode 5 while finding and correcting the source of the noise in No. 2 Steam Generator. See paragraph 3.b of this report.

Preparations for startup were underway at the end of the inspection period; however, preparations were halted when Reactor Coolant Pump 2B shaft seal exhibited staging problems and had to be replaced. This will be discussed in the next resident inspectors' report.

No violations or deviations were identified.

3. Onsite Followup of Events (93702)

a. Missing Seismic Restraints in CPC Panels

During an exit interview conducted by the resident inspectors on September 23, 1988, the NRC inspectors reiterated earlier concerns over the licensee's failure to implement timely corrective action after a nonconforming condition was identified on safety-related equipment. The details of these concerns were documented in NRC Inspection Report 50-382/88-21. In response, the licensee conducted reviews of all open nonconformance condition identification (NCI) reports to ensure that there were no other problems that should have been corrected earlier.

On September 29, 1988, the licensee found NCI-256700, dated June 15, 1988, which identified missing seismic restraints on the fixed incore amplifier (FICA) drawers located in Bays A, B, C, and D of Core Protection Calculator (CPC) Panel CP-22 in the control room. Upon reviewing available documentation related to the restraints and discussing the problem with the NSSS contractor, Combustion Engineering Corporation, the licensee concluded that the restraints must be installed, and without them, the CPCs might not function as designed in the event of a seismic event.

At 6:20 p.m., the shift supervisor was informed of the problem. Technical Specification (TS) 3.3.1 requires at least two CPC channels to be operable when the plant is in Modes 1 or 2 (reactor critical and/or at power). The other two channels must be in bypass and tripped respectively. This action statement could not be met. At 6:20 p.m., the licensee entered TS 3.0.3 which required shutdown of the reactor to be commenced within 1 hour.

The NRC senior resident inspector was onsite and was kept informed by the licensee as the problem developed. At about 7 p.m., the licensee held a conference telephone call with NRC Region IV management and requested enforcement discretion to allow continued operation of the plant while installing the required seismic restraints. Briefly, the basis of the request was that: (1) it would take about the same amount of time to restore the CPC cabinets to a qualified state as it

would to shut down the plant; (2) the licensee would take compensatory measures to manually trip the reactor should a seismic event occur; and (3) the probability of an earthquake was about 2 x 10E-6 in a 24-hour period based on the analysis in Section 2.5.2.7 of the FSAR.

The request was granted at about 7:15 p.m. for a period not to exceed 24 hours. By 2:40 a.m. on September 30, 1988, the missing seismic restraints were installed in two CPC cabinets, thus enabling the licensee to have the minimum two operable CPC channels required by TS 3.3.1. By 10:30 a.m. the same day, the seismic restraints were installed in all four CPC cabinets. The licensee was then able to exit the TS 3.3.1 Action Statements and resume normal operation.

The licensee determined that on October 21, 1982, (about 2 years prior to initial fuel load) the seismic restraints were installed, inspected, and accepted in accordance with Field Change Request E-2759, Revision 1. However, the licensee noted that the Plant Monitoring Computer/Auxiliary Protective Cabinet Multiplexer (MUX) drawers were installed just below the FICA drawers on November 1, 1984. It was apparent that in order to facilitate installation of the MUX drawers, the seismic restraints on at least two of the FICA drawers would have been removed. There was no record of removal or reinstallation. On June 7, 1988, the licensee noted the missing restraints and by June 14 had determined that they were required. On June 15, NCI-256700 was initiated. The control room supervisor, who reviewed the NCI at the time, apparently did not recognize the impact of the nonconformance on the operability of the CPC cabinets.

On October 13, 1988, the licensee completed an engineering evaluation which documented the chronology of events and included a new analysis and calculation performed by Combustion Engineering. The evaluation showed that without the seismic restraints installed on the back and with the drawer front retaining hardware installed, the FICA drawers would not have become a missile hazard in the CPC cabinets during a seismic event. Thus, it was determined that the CPC channels were never rendered inoperable by the missing FICA drawer seismic restraints.

Actions taken by the licensee were prompt and appropriate once this NCI was resurfaced, and the licensee has been in the process of making programmatic changes to provide responsive and timely corrective action when appropriate. Included in the licensee's short-term corrective action was a comprehensive review of all open NCIs. As a result of this review, three other examples of nonconforming conditions not corrected in a timely manner were identified as discussed below:

NCI-255471, dated May 2, 1988, (during the second refueling outage) identified a bent strut on rigid 12-inch safety injection system pipe restraint RCRR-133. Apparently this NCI was given a 1-week priority for correction, which would have been adequate because plant startup was nearly a month away. However, the work did not get done nor was it flagged as a startup constraint. The licensee conducted an engineering evaluation on or about October 8, 1988, and determined, based on the engineer's recollection of the condition and a photograph taken during the outage, that the pipe could not have been overstressed with the bent strut connected. The evaluation also showed that the remaining support would be adequate to carry all design operation and seismic loads if the load carrying capability of the nonconforming restraint was not there. The restraint was replaced during the October 21, 1988, outage.

NCI-258220, dated September 16, 1988, identified several pipe whip restraints on the Reactor Coolant, Feedwater, and Main Steam Systems in the Reactor Containment Building which were too close to the piping to allow the design thickness of insulation to be installed. Apparently the plant had been operating in this condition since construction. The licensee made an operability call on October 1, 1988, stating that continued operation of the plant was acceptable. This was based on the logical assumption that if the restraints were closer to the pipe, the pipe could not reach the whip velocities anticipated by design. This was balanced against the minor degradation in physical properties of the restraints due to the higher temperatures they would have at closer proximity to the pipe (which was not insulated). The Architect-Engineer has been contracted to provide a detailed followup analysis to support the licensee's position. As such, this issue shall be an unresolved item (382/8825-01) pending review of the followup analysis and its conclusions.

On October 28, 1988, while reviewing an NCI-252577 related to level instrument tubing supports on the boric acid makeup tanks, the licensee noted that the drawings called for ASME Class 7 piping on the level indicators when it appeared that Class 3 should have been installed. This was previously identified on March 25, 1985, and accepted as is by engineering on the basis that if the piping broke off the tanks when boration was called for, there was sufficient volume below the tank connections to achieve the required reactor shut down conditions. One of the assumptions included the boric acid concentration being in accordance with the TS. On October 28, 1988, the licensee noted that Amendment 10 of the TS, dated January 8, 1987, reduced the concentration, rendering the basis for acceptance invalid. The licensee was in the process of promptly replacing all of the piping at the end of this inspection period. In addition, records are being received to determine if the actual boric acid concentration in the tanks were low enough since January 8, 1987, to

have caused an unreviewed safety question. The licensee has committed to provide the results to the resident inspectors during the next inspection period.

Since the issue of inadequate corrective action programs implemented by the licensee are the subject of discussion in NRC Inspection Reports 50-382/88-16 and 50-382/88-21, no purpose will be served in additional enforcement actions or tracking mechanisms. The NRC inspectors will continue to monitor licensee actions in this area.

b. Determination of Source of Metallic Noise in Steam Generator No. 2

In NRC Inspection Report 50-382/88-21, the NRC inspectors documented licensee efforts to determine the source of a noise that appeared to be coming from Steam Generator No. 2. The noise was first identified on August 30, 1988, by an operator while conducting his rounds near the Main Feedwater Isolation Valve on the +46 elevation. At the time, the plant was at full power, and the noise resembled a loose part hitting against the feedwater piping or something in the steam generator. By September 16, 1988, the licensee had concluded that the noise was coming from the steam generator based upon sound testing of the accessible feedwater piping. Assistance was obtained from Technology for Energy Corporation (TEC), Combustion Engineering (CE), and Anchor-Darling Valve Company. Later, assistance was obtained from Babcock and Wilcox (B&W) who had previous experience with a similar problem at another plant.

The licensee found that reducing power to about 90 percent significantly reduced the noise. Except when taking sound data, the plant was run at 90 percent to minimize possible damage by the unknown source.

On September 23, 1988, the plant was shut down to Hot Standby (Mode 3) so that the steam generator could be instrumented in an effort to locate the source of the noise. On September 25, 1988, the plant was returned to power and data was taken. By September 30, 1988, the licensee informed the NRC that preliminary results indicated that the source of the noise was in the area of the feedwater inlet into the steam generator.

On October 5, 1988, a CE preliminary technical evaluation suggested four candidates for the noise source: (1) a loose part in the feedwater distribution box; (2) a loose thermal sleeve at the feedwater inlet to the steam generator; (3) a loose seal ring on the feedwater distribution box; and (4) a loose feed ring discharge elbow in the immediate vicinity of the feedwater distribution box. In their final evaluation dated October 13, 1988, CE confirmed the above possibilities to the extent practicable and recommended a visual inspection of the steam generator internals including the thermal sleeve to be conducted at the first convenient opportunity but not

beyond January 1989. The report also recommended operation at a reduced power level, i.e., about 96 percent maximum, where the 100 percent noise amplitude would be reduced by at least one half. On October 14, 1988, the licensee increased power from about 90 percent to 96 percent.

During a conference call between Region IV, NRR, and the licensee on October 14, 1988, the staff expressed concern about the licensee's decision to operate above 90 percent and emphasized particular concern that the licensee was considering operation as late as mid-November 1988 without knowing what kind of damage might be occurring in the steam generator. Following the conference call, the licensee reduced power back to 90 percent and committed to shut down and inspect the steam generator no later than October 29, 1988.

On October 21, 1988, the plant was shut down, and by the evening of October 24, 1988, the steam generator internal inspection was underway. The feed ring, distribution box, thermal sleeve, and moisture separators showed no sign of damage or loose parts. Inspection of the distribution box revealed an unexpected bracket welded in place which was apparently used as a fabrication aid. The bracket did not appear on the drawing. CE will provide the licensee with the appropriate justification for the extra part being in the steam generator. The resident NRC inspectors will follow up to verify this does not create a safety issue.

On October 26, 1988, after finding no condition in the steam generator that would explain the noise, the licensee proceeded to remove the cap from a check valve that is located about 13 feet of piping upstream of the steam generator feedwater inlet. This was to obtain a better view of the feedwater piping immediately upstream of the steam generator. The 20-inch swing check valve was supposed to have had the internals permanently removed in 1982 prior to initial startup due to a design change. The reason for the design change was documented on December 21, 1982, under Significant Construction Deficiency No. 43, pursuant to 10 CFR 50.55(e). Up to this point, therefore, the empty check valve was not considered to be a possible source of the noise. When the cap was removed, the licensee found that only the valve disc was removed. The swing arm and pin were still installed. There was significant wear on the end of the arm indicating it had been impacting the top inside surface of the valve body for some time. It is believed that this was the source of the noise. The licensee went on to drain the steam generator and inspect the secondary side of the tube sheet for foreign or loose parts that could cause the noise and found none.

The swing arm and pin were removed from Check Valve V826B on October 26, 1988. On October 27, 1988, the licensee removed the cap on the similar check valve (V825A) from Steam Generator No. 1 and removed the swing arm and pin. The swing arm in V826B exhibited

much more wear than the arm in V825A, which might explain why the noise could only be heard on the Steam Generator No. 2 side.

The licensee's representatives commented that they heard faint and infrequent, but similar sounds on the Steam Generator No. 1 side while at full power, but they had to listen intently for at least 30 minutes. At the time, it was explained away by the noise experts as noise from No. 2 being transmitted through piping and structure. Once it was determined that the check valve swing arms were still installed on both sides, it became evident that the lesser sounds on the No. 1 side were probably coming from V825A.

The NRC inspectors questioned why the internals were only partially removed. Upon review of the construction documents and quality inspection reports, the NRC inspectors noted in several places there was direction and confirmation stating "check valve internals removed" or "remove internals from check valves," but no where did the documentation specify that all internal parts were to be removed. The objective was to prevent the check valves from trapping pressure in the steam generator during a feedwater line break accident. The objective was accomplished by removing the valve disc only, thus there was no safety significance to leaving the remaining parts in the valve.

The licensee plans to continue noise monitoring of the steam generator and feedwater check valves during unit startup to assure that the problem is corrected. The resident inspectors will continue to monitor licensee actions.

c. <u>QA Audit of Jamesbury Corporation</u>

During the period of August 9-11, 1988, the licensee conducted a vendor plant audit of Jamesbury Corporation, a division of Combustion Engineering. The findings were satisfactory in all areas, except the auditors noted that two purchase orders (L-106184-W and L-96954-P) were filled through the vendor's commercial quality program when the purchase orders specified nuclear quality by imposing such documents as the LP&L approved Jamesbury QA Manual, 10 CFR 21, Appendix B to 10 CFR 50, ANSI N45.2-1971 and ANSI N45.2.9. The purchase orders were delivered to the site with Certificates of Compliance stating conformance with all of the above nuclear requirements. The licensee did not request corrective action from Jamesbury until September 2, 1988, when a corrective action report was issued to the vendor. At the time, LP&L indicated that this was not potentially reportable pursuant to 10 CFR 21. Jamesbury responded on September 21, 1988, stating that the particular parts furnished have always been processed in accordance with their commercial quality program and acknowledged that they should have noted an exception on the certificates of compliance. Jamesbury committed to ensure that if

commercial quality parts were quoted but the purchase orders call for nuclear quality, the disparity would be resolved in writing before the order was entered.

The licensee reviewed all previous Jamesbury orders and showed the NRC inspectors that earlier purchase orders called for the same quality requirements as was imposed on the original components. In the case of noncode, nonpressure boundary parts, Jamesbury always applied the commercial quality program. This appeared appropriate, and the Certificates of Compliance correctly reflected what was specified. The disparity emerged when LP&L changed the specifications in recent purchase orders to require nuclear quality for all parts, which was not necessary for noncode, nonpressure boundary parts. Conducting business as usual, Jamesbury supplied the same parts as before, not realizing the purchase orders no longer specified original component quality requirements.

The only apparent safety issue here is the lack of timeliness on the part of the licensee to recognize the implications of the vendor supplying commercial grade material when nuclear grade was specified. As it turned out, the hardware was correct, thus there was no impact on plant safety; however, this is another indication of weaknesses in the licensee's corrective action program which have been addressed in previous inspection reports. Since the licensee is already taking actions to improve the timeliness and effectiveness of its corrective action programs, there will be no purpose served in issuing an additional Notice of Violation at this time. Region IV staff and the resident inspectors are monitoring licensee progress in this area.

4. Operational Safety Verification (71707)

The objectives of this inspection were: (1) to ensure that this facility was being operated safely and in conformance with regulatory requirements; (2) to ensure that the licensee's management controls were effectively discharging the licensee's responsibilities for continued safe operation; (3) to assure that selected activities of the licensee's radiological protection programs were implemented in conformance with plant policies and procedures, and in compliance with regulatory requirements; and (4) to inspect the licensee's compliance with the approved physical security plan.

During the outage of October 21, 1988, the NRC inspectors toured the containment building and observed maintenance activities, implementation of radiological controls, housekeeping, and equipment condition. The NRC inspectors observed the following conditions and noted that no Condition Identification (CI) tags were in place, and thus they were identified to the licensee for correction:

a. The cover on safety-related Conduit Box B3070SA was ajar and hanging from one bolt.

b. The face of the thermometer installed on cabinet C-6B was shattered.

c. The pipe whip restraint at Valve SI-332A was not installed properly. A retaining nut was loosened and approximately 1 inch from the stop bracket.

d. Boric acid deposits at a vent on Reactor Coolant System Pressure Transmitter RC-IDPT-0124 indicated leakage during plant operation.

e. A rope was hanging from the snubber attached to the safety injection system piping near Valve SI-302. Also, in the vicinity of SI-302, a cable was hanging freely from a pipe whip restraint.

f. Caps were not installed on safety injection system test Valves SI-2361 and SI-2391.

g. Valve SI-2421 (Safety Injection to Loop 1B Containment Isolation Check Valve SI-242 Bypass) at containment penetration No. 56 was closed, but no locking device was installed. Procedure OP-100-009, Revision 8, "Control of Valves and Breakers," required this containment isolation valve to be locked in the closed position. The NRC inspectors noted that the last time these valves were checked in accordance with OP-100-009 was in April 1988 during the early stages of the second refueling outage. It appeared that later during the outage the lock was removed. Section 5.1.2.1 of OP-100-009 requires documentation when the status of a locked valve is changed. None could be produced by the licensee. The NRC inspectors concluded, based on the unexplained missing lock, that the controls required by OP-100-009 were not implemented. Failure to comply with OP-100-009 is an apparent violation of NRC regulations (382/8825-02).

The licensee promptly corrected the above items.

The NRC inspectors observed boric acid deposits from leaking safety injection system Check Valve SI-512B on conduit, cable trays, firewrap, ducting pipe supports, snubbers and component cooling water system piping, and valves. The leaking valve was repaired and the boric acid deposits were cleaned up. The licensee analyzed the effects of the boric acid on the above equipment and found no problems.

The NRC inspectors visited the control room at least once each day when on site, and verified that proper control room staffing was maintained, operator behavior was professional and commensurate with plant conditions, and that approved procedures were utilized and complied with. The panels were inspected for anomolies and when found were satisfactorily explained by the operators.

In addition to the reactor containment tours above, the NRC inspectors toured accessible interior and exterior areas of the plant and noted no significant deficiencies in housekeeping, radiological work practices and

equipment condition. The NRC inspectors noted a continuing effort to paint components and structures, thus protecting surfaces and facilitating easy cleanup. The overall appearance of the plant is excellent.

The NRC inspectors attended daily plan-of-the-day meetings to keep abreast of problems and to observe the licensee's identification and managing of corrective maintenance issues.

The implementation of the licensee's security program was specifically observed at least once each week during this inspection period, with emphasis on personnel plant identification badges, vital area portals being kept locked, and processing of personnel through the primary access point. No deficiencies were noted.

5. Monthly Maintenance Observation (62703)

The below listed station maintenance activities affecting safety-related systems and components were observed and documentation reviewed to ascertain that the activities were conducted in accordance with approved procedures, technical specifications, and appropriate industry codes or standards.

a. Work Authorization 01025389. The NRC inspectors observed the inspection of the high pressure safety injection (HPSI) pump A/B motor heater and power supply cable terminations. The inspection was performed to verify that the splices were in conformance with Drawing LOU-1564-B-288, "Cable & Conduit List and Installation Details." Verification of the HPSI pump termination splices was performed after charging pump motor heater splices were found in nonconformance with the above drawing and subsequently replaced. The HPSI pump motor heater cables were landed on terminal blocks so no splices were required. The HPSI pump motor power supply cables were not found in the expected configuration per Drawing LOU-1564-B-288, but the power supply cable splices were of the same configuration as the emergency feedwater pump cables depicted in the same drawing and thus were considered acceptable by the licensee. Because of the observed inconsistencies in pump motor cable termination splices, the licensee performed additional verifications.

During the subsequent verifications, the licensee determined that several V-type Okonite splices used on safety equipment inside containment and shown on Drawing LOU-1564-B-288 were not consistent with the documented environmentally qualified (EQ) configurations. Drawing LOU-1564-B-288 did not require filling of the area between the cable legs on all V-type splices with insulation tape. The qualified configuration required filling of this area with insulation tape.

The licensee obtained a letter from Wyle Laboratories stating that these splices can be qualified for a minimum of 5.9 years, which

means some splices may have to be replaced as early as 1990. The licensee is currently determining the lifetime based on plant specific parameters which may turn out to be longer than 5.9 years. Once the plant-specific qualification lifetime is determined, the licensee has committed to implement a program for splice replacement (if required). The licensee indicated that this will take place in the next few weeks. Followup on the licensee's actions for program implementation shall be tracked as an open item (382/8825-03).

b. Work Authorization 01018979. The NRC inspectors witnessed the hydrostatic test of the welded pipe joints associated with the replacement of Main Steam Line No. 2 Drain Isolation Valve MS-1051B. The hydrostatic test was conducted in accordance with Maintenance Departmental Procedure MM-7-007, Revision 3, "Hydrostatic Pump Unit Setup and Operation for Testing." The test was completed satisfactorily in the presence of the authorized nuclear inservice (ANI) inspector and a plant quality inspector. However, there was one problem identified by the plant quality inspector prior to starting the test. Procedure MM-7-007 had a prerequisite (Step 3.2) requiring the test gauges to be calibrated within 2 weeks prior to the test. Only one of the two test gauges met this requirement. This delayed the test for about ½ hour until a properly calibrated test gauge was connected. This was indicative of inattention to procedure requirements by the maintenance personnel conducting the test. The NRC inspectors expressed concern to the licensee that plant management efforts to ensure procedure compliance are not yet fully effective. The maintenance supervisor initiated a change to the procedure to require a specific signoff of the test gauge calibration requirement.

c. Work Authorization 01025932. The NRC inspectors observed the removal of the swing arm and hinge pin from main feed system Check Valve 2FW-V826B. The reason for this work is discussed in paragraph 3.b of this report. The documentation appeared to be in order, and no problems were encountered during the work.

d. Work Authorization 0109107. The NRC inspectors observed portions of the installation of a new motor in containment fan cooler "C" and noted the following:

(1) The old fan motor was removed at the end of the refueling outage in May 1988. A note was written on the work site copy of the work authorization stating that electrical tools and parts were stored in Cabinet C-8. The NRC inspectors observed tools and copies of pages from the work package in Cabinet C-8. The tools were apparently stored in Cabinet C-8, which contains engineered safety features system transmitters (safety injection tank level and pressure) since May 1988. This was identified to the licensee and the tools were removed. All other instrument cabinets in the reactor containment building were checked prior

to containment closure, and no other tools or parts were found. This appears to be an isolated case and not normal practice by maintenance personnel.

(2) While the technicians were rigging the fan in place, the NRC inspectors observed spare bolts and washers in the fan assembly resting adjacent to the blades. Because of the potential for jamming the fan blades, this was identified to technicians performing the work, and the spare bolts and washers were removed. Plant management was informed of this poor work practice.

(3) Splicing of the fan motor power supply leads was performed per Procedure ME-4-809, Revision 4, "Low Voltage (600 Volts and Less) Power and Control Cable/Conductor Terminations and Splices," in conjunction with Drawing LOU-1564-B-288, "Cable and Conduit List Installation Details." The technicians performing the work had a copy of the proper detail from Drawing LOU-1564-B-288 but did not have the sheet containing the applicable notes at the work site. As discussed in paragraph 5.a above, the drawing detail did not provide for a splice configuration that was supported by EQ documentation. The electricians made the splices in accordance with ME-4-809 and added sufficient insulating tape between the legs of the conductors such that the work was supported by EQ documentation. The NRC inspectors expressed concern that the licensee's failure to correct the disparities between ME-4-809 and the drawings was placing the electricians in a difficult position. Actions taken to solve this problem, as identified previously in NRC Inspection Reports 50-382/87-31 and 50-382/88-21 have been unacceptable to date. The licensee's representatives met with the NRC inspectors on October 27, 1988, and presented what appeared to be an acceptable revision to both the drawings and the procedure. This action continues to be tracked under Violation 382/8731-03 and is another indicator of poor and untimely corrective action.

No violations or deviations were identified.

6. Monthly Surveillance Observation (61726)

The NRC inspectors observed the below listed surveillance testing of safety-related systems and components to verify that the activities were being performed in accordance with the technical specifications. The applicable procedures were reviewed for adequacy, test instrumentation was verified to be in calibration, and test data was reviewed for accuracy and completeness. The inspectors ascertained that any deficiencies identified were properly reviewed and resolved.

a. Procedure OP-903-024, Revision 6, "Reactor Coolant System Water Inventory Balance." On October 31, 1988, the NRC inspectors observed the performance of the reactor coolant system (RCS) inventory balance prior to entering Mode 4. The results indicated that RCS unidentified leak rate was 1.59 gallons per minute (gpm). Because the plant was in Mode 5, no action was required. Since no other indications of leakage were present and a large RCS temperature change occurred during the test (1.2°F with RCS at 190°F), the inventory balance was reperformed with more attention on maintaining RCS temperature. Unidentified leakage of 0.96 gpm was measured, satisfying the TS requirement (less than 1 gpm).

No violations or deviations were identified.

7. Followup of Previously Identified Items (92701)

(Closed) Issue on Main Feedwater Isolation Valves (MFIVs). Paragraph 2 of NRC Inspection Report 50-382/88-15 identified a need to perform an additional inspection on the review of the licensee's Potentially Reportable Event (PRE) Report 88-053. On May 13, 1988, the PRE reported, in short, that MFIVs FW-184A & B may not have met TS Table 3.3-5 minimum response times under full flow conditions. FSAR Section 6.2.1.4 assumes 100 percent feedwater flow for 5 seconds for the limiting accident for peak containment pressure (main steam line break) with a concurrent failure of one containment cooling train. The valves were tested in the past to prove that they could close in not more than 5 seconds, but the test was done at zero flow conditions. The licensee determined that 5 seconds under zero flow conditions corresponded to 7 to 9 seconds under full flow conditions. Initially, this appeared to be a potential unreviewed safety question until the licensee noted that the FSAR took credit for the feedwater regulating valves as a backup that would close on the same signal (Main Steam Isolation Signal) in not greater than 5 seconds, under full flow conditions. Thus, the plant was not in an unanalyzed condition due to the slower settings on the MFIVs. The licensee reset the MFIVs to close in 3 seconds under zero flow conditions, which corresponded to five seconds full flow. This was completed on May 13, 1988, before the plant was restarted from the second refueling outage. The licensee determined that this issue was not reportable under 10 CFR 50.73, which appeared to be a correct decision. This issue is closed.

No violations or deviations were identified.

8. Licensee Event Report (LER) Followup (92712)

The following LER was reviewed and closed. The NRC inspectors verified that reporting requirements had been met, causes had been identified, corrective actions appeared appropriate, generic applicability had been considered, and that the LER form was complete. The NRC inspectors

confirmed that unreviewed safety questions and violations of TS, license conditions, or other regulatory requirements had been adequately described.

(Closed) LER 88-022 (Revision 1), Missed Penetrant Test on Two Welds Due to Personnel Error. When Revision 0 of LER 88-022 was published, the NRC inspectors noted that the licensee had listed the event date as August 2, 1988, when in fact, as described in the text, the problem was found on June 20, 1988. The NRC inspectors expressed concern to the licensee that it was inappropriate to wait until September 1, 1988, to issue the LER when the problem was identified over 2 months earlier. The text explained the licensee's actions in detail which eventually led to the LER; however, it illustrated the lack of promptness built into the licensee's corrective action program. This issue has been addressed in previous NRC inspection reports and the licensee has already been tasked to improve its corrective action program. The LER indicated that it would take a plant cooldown to facilitate a penetrant test. On September 24, 1988, the licensee performed the test while shutdown in hot standby for other reasons. The NRC inspectors reviewed the test documentation and found the piping temperature was verified at 111°F. Thus, it was not necessary to cool down to correct the deficiency. All seven of the joints in the new piping were tested or retested at that time. The LER was revised on October 19, 1988, reflecting the actual event date and completion of the penetrant test. This LER is closed.

No violations or deviations were identified.

9. Exit Interview (30703)

The inspection scope and findings were summarized on November 4, 1988, with those persons indicated in paragraph 1 above. The licensee acknowledged the NRC inspectors' findings. The licensee did not identify as proprietary any of the material provided to or reviewed by the NRC inspectors during this inspection.