APPENDIX A

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-382/85-05

CP: CPPR-103
License: NPF-26

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: February 1 through March 31, 1985

Inspectors:

G. L. Constable, Senior Resident Inspector — Date: 5/16/85

T. A. Flippo
T. A. Flippo, Resident Inspector — Date: 5-16-85

for W. B. Jones, Reactor Inspector — Date: 5/16/85

for W. R. Johnson, Reactor Inspector — Date: 5/23/85

8506030518 850528
PDR ADOCK 05000382
Q PDR

Assisting Personnel:

T. L. Bell, Nuclear Training Instructor
(paragraph 6)

J. J. Harrison, Chief, Engineering Branch, Region III
(paragraph 11)

D. Tomlinson, Reactor Inspector
(paragraph 10)

Approved: W A Crossman 5/23/85
W. A. Crossman, Chief, Project Section B, Date
Reactor Project Branch 1

Inspection Summary

Inspection Conducted February 1 through March 31, 1985 (Report 50-382/85-05)

Areas Inspected: Routine, announced inspection of: (1) startup test procedure review; (2) review of licensee significant construction deficiencies; (3) Three Mile Island (TMI) open items; (4) shift turnover review; (5) followup on previous NRC Inspection findings; (6) test results evaluation; (7) Phase III test procedure witnessing; (8) followup on allegations; and (9) quality assurance (QA) personnel qualifications. The inspection involved 617 inspector-hours onsite by four NRC inspectors.

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

DETAILS

1. Persons Contacted

Principal Licensee Employees

*R. S. Leddick, Senior Vice President, Nuclear Operations
*R. P. Parkhurst, Plant Manager, Nuclear
*T. F. Gerrets, Corporate QA Manager
*P. N. Backes, Operations QA Manager
*L. F. Storz, Assistant Plant Manager, Operations and Maintenance
S. A. Alleman, Assistant Plant Manager, Plant Technical Staff
O. D. Hayes, Operations Superintendent
*J. N. Woods, Plant Quality Manager
*R. G. Pittman, Operations QA Audit Supervisor
*G. E. Wuller, Onsite Licensing Coordinator
*K. L. Brewster, Onsite Licensing Engineer
J. R. McGaha, Maintenance Superintendent
L. M. Meyers, Assistant Operations Superintendent

*Present at exit interviews.

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

2. Plant Status

The U.S. Nuclear Regulatory Commission issued Facility Operating License NPF-38 together with Technical Specifications and Environmental Protection Plan to LP&L for the Waterford Steam Electric Station, Unit 3 on March 16, 1985. License NPF-38 authorizes operation of the Waterford Steam Electric Station, Unit 3, at core power levels not to exceed 3390 megawatts thermal (100 percent power), and supersedes License NPF-26, issued on December 18, 1984.

At the end of the inspection period the licensee was in the process of conducting startup testing at the 20 percent power level.

3. Startup Test Procedure Review

The NRC inspectors reviewed the startup test procedures for performing power ascension testing of the plant. The procedures were reviewed for technical content, compliance with the Final Safety Analysis Report (FSAR), and compliance with licensee's administrative procedures. The startup test procedures reviewed are listed below:

SIT-TP-716 Core Performance Record

SIT-TP-717 CPC/Core Operating Limit Supervising System (COLSS) Verification

No violations or deviations were noted.

4. Review of Licensee Significant Construction Deficiencies (SCDs)

The NRC inspector reviewed the following SCDS:

a. (Closed) SCD-37, Temperature Detectors (RTDs) Failure

On July 27, 1981, Combustion Engineering, Inc. (C-E) notified EBASCO Services, Inc. (EBASCO) that errors in the RCS T-cold signal could result in a nonconservative thermal margin/low pressure (TM/LP) trip setpoint and could permit possible operation in excess of departure from nucleate boiling (DNB) limits. The cause of the erroneous RTD signals was due to resistance changes in the RTD circuitry as a result of corrosion degradation of the RTD leads at the thermal block connection in the head of the RTD assembly. The corrosion was believed to be caused by a galvanic reaction between the dissimilar metals in the RTD leads and the thermal blocks.

A total of 33 Rosemount RTDs were utilized at Waterford 3. Of these 33, 16 were safety-related RTDs installed in steam generators hot and cold legs which provide input to the plant protection system. Eight of these 16 Rosemount RTDs were replaced with Weed RTDs which left 8 dual element Rosemount RTDs which were required to be environmentally sealed. The NRC inspector reviewed the licensee's corrective action and supporting documentation for resolution of this deficiency. This issue is considered appropriately resolved.

No violations or deviations were identified.

b. (Closed) SCE-80, Unsatisfactory Stroking of EFW Pump Turbine Steam Supply Shut Off Valves

During hot functional testing (HFT), automatic operation of valves 2MS-V611A and 2MS-V612B were found to be unsatisfactory. Stroking of the valves was not smooth, and excessive force was needed to open the valves.

The licensee's corrective action was to replace the pneumatic operators for the valves 2MS-V611A and 2MS-V612B with D-C motors. The NRC inspector witnessed a retest of these valves during post-core HFT. The valves functioned satisfactorily per design in order to support the minimum emergency feedwater (EFW) control "lo-lo" setpoint analyses. This item is closed.

No violations or deviations were identified.

c. (Closed) SCD-93, Charging and Letdown Containment Ioslation Valve Deficiency

Following the cool down after HFT, it was discovered that the letdown containment isolation valve (CVC-103) was stuck in the open position. It appears that the malfunction occurred due to upstream float of the seat which reduced the clearance between the seat and the gate, thereby causing the gate to stick open.

The licensee's corrective action was to repair the valve with new seats which are sized for maximum interference fit in the seat pocket in order to eliminate the previously experienced seat float. The closure of the valve has since been satisfactorily tested in the cold fluid condition.

The NRC inspector witnessed the retest on the valve during post core hot functional and the valve operated satisfactorily and closed in approximately 1.5 seconds. FSAR commitment for closure of valve (CVC-103) is 10 seconds.

On February 21, 1985, it was discovered that valve (CVC-103) failed to close again after a plant cooldown. The valve has since been freed, stroked, and determined to be operable during cold shutdown. On March 3, 1985, at 0105 hours, when the plant was at normal operating temperature and pressure, the valve was stroked once. The closure time was 1.5 seconds which is within the allowable limits.

The NRC inspectors held discussions with the licensee concerning the operation of this valve and the licensee committed to the following actions in a letter dated March 8, 1985, to Mr. Robert D. Martin.

"The valve will be stroked in 100°F decrements during the next two cooldowns starting at about 100°F below normal operating temperature and pressure. The surveillance conducted during cooldown will be in accordance with procedure OP-10-001. If the valve fails to close during testing, Technical Specification 3/4.6.3 will be evoked and the plant personnel will respond accordingly.

A long term solution will be sought to relieve the plant from this valve cycling exercise; should cycling prove to be the necessary solution to current reliability concerns." This item is closed.

No violations or deviations were identified.

5. Three Mile Island (TMI) Open Items

a. (Closed), NUREG-0737 (TMI Item II.B.1) Reactor Coolant Vent System

The licensee has installed the reactor coolant gas venting system (RCGVS). This system allows for the remote venting of noncondensable gases, through either the reactor vessel head vent or pressurizer steam space vent, which may collect in the RCS following certain post accident conditions. The RCGVS is described in detail in FSAR, Section 5.4.15, "Reactor Coolant Gas Vent System." The NRC inspector reviewed the design basis for the RCGVS and determined that the design meets the requirements of NUREG-0737.

In addition, the NRC inspector reviewed Emergency Operating Procedure OP-902-002, Revision 0, "Loss of Coolant Accident Recovery Procedure," and OP-902-003, Revision 0, "Loss of Forced Flow Recovery Procedure." The NRC inspector determined that the procedures provide adequate information and guidance to the operator for initiating and terminating RCGVS usage. This item is considered closed.

b. (Closed), NUREG 0737 (TMI Item I.G.1) Training During Low Power Testing

The licensee has revised FSAR Sections 14.2.12.3.25 and 14.2.12.3.34 to reflect that natural circulation testing and training will be conducted in Mode 3 under conditions of actual decay heat removal following a reactor trip from 80% of rated thermal power level. This test is to be conducted in conjunction with the loss of flow test. The NRC staff evaluated the above proposal in NUREG-0737 Supplement 5, "SER Related to the Operation of Waterford SES, Unit No. 3," dated June 1982. The NRC staff concluded that the testing and training objectives stated in the NRC staff's position can be readily accomplished during post-80 percent power trip conditions. The NRC inspectors will verify that the licensee meets the NRC staff's objectives for training during low-power testing during Mode 3 natural circulation operation. This item is considered closed.

c. (Closed), NUREG-0737 (TMI Item II.E.3.1) Emergency Power Supply for Pressurizer Heaters

The licensee has completed work on the pressurizer heater electrical supply which provides the capability to supply from either the offsite power source or the emergency power source (when offsite power is not available). A redundant group of pressurizer proportional heaters, each with a capacity of 150 kW, may receive power from emergency diesel generators following a loss of offsite power.

The proportional heaters are powered from the 480 V nonsafety switchgear buses 3A32 and 3B32. The safety-related Class 1E breakers provide power to these buses from the 4.16 kv ESF buses 3A3-S and 3B3-S. These safety-related buses are designed to trip when they receive a loss of offsite power (LOOP) or safety injection actuation signal (SIAS) signal. The nonsafety 480 V switchgear buses will then trip due to bus undervoltage. This scheme ensures the pressurizer heaters are protected by safety Class 1E circuit breakers. The nonsafety 480 V switchgear breakers can be reclosed manually from CP-1 once the emergency diesels are on line, as long as a SIAS signal is not present.

LP&L's Emergency Operating Procedure OP-902-005, Revision 0, "Degraded Electrical Distribution Recovery Procedure," Section E, provides the necessary guidance to direct the operator on when and how the pressurizer proportional heaters are to be connected to the emergency buses. In addition, LP&L's Surveillance Procedure OP-903-28, Revision 0, "Pressurizer Heater Emergency Power Supply Functional Test," provides instructions to demonstrate the operability of the pressurizer heater emergency power supply, as required by LP&L Technical Specifications. This item is considered closed.

d. <u>(Closed), NUREG-0737 (TMI Item II.E.1.1) Auxiliary Feedwater System Evaluation</u>

The licensee reevaluated the EFW system using event tree and fault tree logic techniques for potential EFW failure under various loss of main feedwater transients. The results of the evaluation, stated in FSAR Appendix 10.4.9.B, were reviewed by the NRC inspectors and determined to adequately address failures which could result from human error, common causes, single point vulnerability, testing, and maintenance outages.

LP&L's staff also compared the EFW design with the requirements of Standard Review Plan (SRP) 10.49 and Branch Technical Position (BTP) ASB 10.1. The NRC inspector verified that the licensee's evaluation, stated in FSAR Appendix 10.3.9A as Table 10.3.9A-1, is in compliance with the above established acceptance criteria.

An analysis of the Waterford 3 EFW against the NRC's requirements for EFW flow to the steam generators to ensure adequate removal of core decay heat is documented in FSAR Appendix 10.4.9A as Table 10.4.9A-3. The NRC inspectors' review of the EFW flow design basis revealed that the licensee has adequately considered plant transient and accident conditions which could affect removal of reactor decay heat.

The NRC inspector determined that the licensee has met the requirements of NUREG-0737 for the evaluation of the EFW system. In addition, the licensee has met the NRC's short and long term

recommendations for EFW operability. Compliance with these recommendations is reflected in the Waterford 3 Technical Specifications and LP&L's Operating Procedure OP-9-003, "Emergency Feedwater," Revision 3. This item is considered closed.

e. (Closed), NUREG-0737 (TMI Item I.D.2) Plant Safety Parameter Display Console

The licensee has installed the safety parameter display system (SPDS) consoles in the control room, technical support center (TSC), and emergency operations facility (EOF) as required by NUREG-0737. However, the computer at Waterford 3, specifically CPU-3, cannot execute both the emergency response (including SPDS) and NSSS (including COLSS) software concurrently. LP&L committed, in their letter W3P85-0432, to install an additional redundant CPU by June 1985 to allow for simultaneous operation of NSSS and emergency response software. Until the additional CPU is available to allow for continuous operation of the SPDS software, the nuclear plant operators (NPO) will be able to initiate the SPDS program from the control room during Modes 6 through 3 and have it available within 15 minutes.

The NRC inspectors observed that the NPO was able to load the SPDS program from the control room. Although the SPDS program was available within 15 minutes on several occasions, portions of the SPDS program had to be reinitialized from the computer room. The licensee has agreed to maintain a computer operator on shift until an NPO from each shift has been trained to reinitialize the SPDS program from the computer room if necessary.

While reviewing the operability of the SPDS program, the NRC inspectors noted that several displayed parameters were incorrect, including steam generators and pressurizer level. Discussions with the licensee revealed that work is presently in progress to verify the accuracy of the displayed parameters. This is considered an open item (50-382/8505-01).

The NRC inspectors determined that the licensee has satisfactorily implemented the SPDS software as required by NUREG-0737, Supplement 1 using the proposed interim solution. The NRC inspectors will monitor the testing of CPU 4 and will verify that the installation of the unit will provide the operators with continuous display of required safety parameters. This item is considered closed.

f. (Closed), NUREG-0737 (TMI Item II.E.42) Containment Isolation Dependability

The licensee's evaluation of the containment isolation system design with the requirements of SRP 6.2.4 is documented in FSAR Section 6.2 as Table 6.3-32. This section of the SRP requires that there be diversity in the parameters used for initiation of containment

isolation. The NRC inspector reviewed the licensee's evaluation and determined that the containment isolation system design complies with the requirements of SRP 6.2.4.

In addition, the licensee reviewed each of the systems that impact containment isolation to determine if they are essential or nonessential based on whether the system is necessary to: (1) maintain the integrity of the reactor coolant boundary; (2) shut the reactor down and maintain in a safe condition; and (3) prevent or mitigate the consequences of accidents which could result in the potential for offsite exposure. The NRC inspector verified that each nonessential system, as described in FSAR Table 6.2-32, is maintained in a closed position or will close on receiving an isolation signal. These valves, once closed by an isolation signal, can only be reopened by deliberate operator actions following resetting of the containment isolation signal.

The containment isolation high pressure trip set point for initializing containment isolation of nonessential penetrations has been established below that permitted by the Technical Specifications limits. The licensee used explicit set point methodology to account for individual instrument uncertainty such as instrument loop error, set point variance, and instrument drift. This method ensures that the trip setpoint is sufficiently below the allowable value to prevent high containment pressure without initiating a containment isolation signal.

In addition to the containment high pressure isolation, the containment purge system is designed to automatically isolate on a containment purge isolation signal (high radiation). Operability of the containment purge isolation valves has been analyzed to ensure closing against the most severe design basis accident. The analysis showed that the containment purge isolation valves are capable of closing under any accident conditions when limited to an opening of 52°. Modifications to limit valve openings have been completed.

The NRC inspector determined that the licensee has fulfilled the requirements of NUREG-0737 for containment isolation dependability. LP&L has also implemented Operating Procedure OP-903-075, "Containment Purge Valve Isolation System Operability Check," Revision 3, and Off Normal Operating Procedure OP-901-020, "High Airborne Activity in Containment," Revision 2, to ensure the containment isolation system operates as designed. This item is considered closed.

No violations or deviations were identified.

6. Shift Turnover Review

The NRC inspectors reviewed the plant procedure for shift turnover (OP-100-07). The NRC inspectors reviewed the procedure to assure that the checklist provided for the oncoming control room operators included the following items.

a. Assurance that critical plant parameters are within allowable limits.

b. Assurance of availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

c. Identification of system and components that are in a degraded mode of operation permitted by the Technical Specifications.

It is the NRC inspector's observation that a new checklist or an upgrade of the old checklist is warranted to assure that the oncoming shift supervisor has adequate information regarding the availability and proper alignment of all systems essential to the prevention and mitigation of an accident. This item was discussed with the licensee and they have agreed to revise the procedure to address this area of concern. This is considered an open item (50-382/8505-02).

No violations or deviations were identified.

7. Followup on Previous NRC Inspection Findings

(Closed), Open Item 50-382/84-07 Construction Appraisal Team (CAT) Finding 6.2 and 6.3 on Masonry Walls

The NRC inspector reviewed the documentation associated with the rework on masonry wall S-24 and verified the wall was constructed as per design. This item is considered closed.

No violations or deviations were identified.

8. Test Results Evaluation

The NRC inspectors reviewed initial fuel load and Phase III test results to verify that: (1) all changes, including deletions to the test program, had been reviewed for conformance to the requirements established in the FSAR and Regulatory Guide 1.68; (2) deficiencies had been adequately addressed and corrective action completed; (3) the licensee correctly analyzed the test data and verified it met the established acceptance criteria; and (4) the startup organization as well as the Plant Operating Review Committee (PORC) had reviewed and accepted the test results. The following test packages were reviewed:

SIT-TP-400	Initial Fuel Load
SIT-TP-502	Post Core Reactor Coolant System Flow and Coastdown Measurements
SIT-TP-503	Rod Drop Measurements
SIT-TP-505	Pressurizer Effectiveness
SIT-TP-506	Leak Rate Test
SIT-TP-508	Reactor Coolant System Heat Loss

The NRC inspectors determined that each of the above test packages was properly reviewed by the licensee and met the applicable acceptance criteria. The following observations were made while reviewing the test results.

a. SIT-TP-502 - The reactor coolant flow coastdown measurements, following a planned simultaneous trip of all four reactor coolant pumps (RCP), was less conservative than that assumed in FSAR Section 15.3.2.1. The RCP flow coastdown curve is used by C-E to establish the maximum thermal margin for the COLSS and setpoints. To ensure that the COLSS is conservative and the plant operated within the analyzed operating parameters, the COLSS penalty factor EPOL1 has been changed from -4.7175 to -7.000. This new COLSS penalty factor will remain in effect until RCP flow coastdown can be reanalyzed during SIT-TP-727, "80% Loss of Flow - Natural Circulation."

b. SIT-TP-505 - The flow settings on the pressurizer continuous spray valves (RC-302A and RC-302B) could not be adjusted to maintain the pressurizer spray line temperature 25°F to 30°F colder (actual 50°F) than the average RCS cold leg temperature at steady state conditions as required by the acceptance criteria. C-E, in letter C-CE-9390, dated February 8, 1985, stated that the maximum allowable temperature differential for the spray nozzle could be increased to a value of 85°F.

    The HFT data for the pressurizer spray line temperature indicated readings of approximately 522°F with four RCPs operating and both continuous spray bypass valves fully open. The pressurizer spray line low temperature alarm is presently set at 525°F which is resulting in an almost constant alarm condition. C-E, in their letter C-CE-9390, recommended resetting the alarm to 520°F to void the constant alarm during steady state operating conditions. In addition, Technical Specification 5.7, "Component Cyclic or Transient Limits," Table 5.7.1, for the pressurizer spray nozzle should be revised to account for the reduced setpoint. The present Technical Specification limit calls for a differential temperature of 130°F between the pressurizer water (approximately 653°F at 2250 psia) and the spray water temperature, with less than four RCPs running. The licensee is presently reviewing a change to allow the limit to be revised to 140°F, to avoid the present required calculation of usage factors for virtually every operation of the pressurizer spray

system. The NRC inspectors will review any design setpoint or cyclic limit changes that are made to the pressurizer spray system.

No violations or deviations were identified.

9. Phase III Test Procedure Witnessing

The NRC inspectors witnessed the performance of portions of the following Phase III test procedures:

SIT-TP-500	Determination of Auxiliary Spray Flow Split
SIT-TP-650	Low Pressure Physics Tests
SIT-TP-704	Reactor Coolant System Delta T Power Determination
SIT-TP-705	Nuclear and Thermal Power Calibration
SIT-TP-708	Initial Turbine Startup

During the performance of the tests, the NRC inspectors verified the following:

a. The personnel conducting the test were cognizant of the test acceptance criteria, precautions, and prerequisites prior to beginning the test.

b. The test was conducted in accordance with an approved procedure and the test procedure was used and signed off by personnel conducting the test.

c. Data was collected and recorded as required by the test procedure instructions.

No violations or deviations were identified.

10. Followup on Allegations

Three allegers, all requesting anonymity, were interviewed by two NRC representatives on December 18, 1984. They expressed concern with specific welding practices employed at Waterford 3 during construction. These concerns and the NRC findings are stated:

a. Reactor surge line restraints were originally welded and then inspected using radiography. After "numerous" unsuccessful weld repair attempts the inspection requirements were changed to allow the acceptance of these welds based upon magnetic particle inspection. The allegers stated that the reason given for this change was that the welds were inaccessible for radiography, even though they had been radiographed previously.

The NRC inspector reviewed the documentation for these restraints and conducted interviews with cognizant engineering and NDE personnel. This documentation review and subsequent interviews revealed that, in some cases, the inspection requirements were changed in the original design of the surge line restraints, all of the welds were designated as full-penetration welds which require a full volumetric inspection. Normally this would be accomplished using radiography. In cases where component configuration, access limitations or weld geometry prevent 100 percent radiographic examination, all or part of the weld may be examined using an approved ultrasonic technique. Because of the complexity of these restraints and severe access limitations, an engineering decision was made to radiograph all accessible areas of these full-penetration welds and perform an ultrasonic examination of the remaining areas. Because of the difficulties encountered in using two methods of volumetric inspection on parts of the same weld joint, a subsequent engineering analysis was performed and it was decided that certain welds in these supports had been over designed and could be changed to partial-penetration welds without compromising the intended function of the structures. The acceptance criteria for partial-penetration welds is based upon the performance of a surface examination by either the liquid penetrant or magnetic particle method. No "blanket" change was made for the weld design change of these restraints. A provision was included in the engineering analysis for each of the welds in this category that each of the welds in this category must be documented in a Field Change Request (FCR) and evaluated individually by engineering prior to this "down grading." The NRC inspector reviewed four FCRs relating to these restraints and ascertained that each had been initiated, evaluated, and dispositioned in accordance with the appropriate procedures. In some cases the requirements for welds were changed and in other cases the changes were denied but the disposition for each weld under consideration was clearly stated on each FCR and each was signed by the design engineers performing the evaluations.

The NRC inspector determined that some weld and acceptance requirements were changed; however, each change was properly done in accordance with code and no improprieties were noted. The NRC inspector had no further questions concerning this item.

b. The allegers stated that a piping restraint in Cell 2B, located inside the containment building, had cracked. The restraint located at approximately the 15-foot level, was repaired unsuccessfully several times before the design was changed. Their concerns are with the cause of the original cracking and the adequacy of the design changes. They expressed concern with the possible damage done to the concrete behind the embedded plate to which the restraint is mounted. Their concern is that the preheating of this plate prior to welding caused spalling of the concrete wall behind the embed.

The NRC inspector reviewed several FCRs issued for support and restraint rework performed in the area cited by the allegers in an effort to identify the specific structure referred to. By comparing the above statement to the actual work performed, it was determined that the subject structure was a "C" stop as shown on drawing G-696 S06 R-3, sheet 6. CCR-AS-3625 states that repeated cracking of one weld occurred, necessitating a design review to establish the cause of the cracking and a means to prevent recurrence. The engineering analysis determined that locked-in weld stresses and rigidity caused by the restraint geometry were responsible for the repeated cracking in the final weld. A design change for this restraint was proposed and is shown on the FCR. The change not only altered the restraint configuration but changed some of the welds from full-penetration to partial-penetration welds. This changing of the type of weld being used automatically changed the inspection requirements from a volumetric examination to a surface examination. The signatures on the subject FCR indicate that all procedural requirements were met and that the FCR was properly dispositioned. The restraint was subsequently welded, inspected, and accepted without further cracking. Concerning the heat-induced spalling of the concrete behind the plate, the NRC inspector interviewed several civil and structural engineers who agreed that the weld preheat applied to an embed plate of this size and thickness, could not transmit sufficient heat to the concrete to cause even minor spalling behind the plate.

The NRC inspector had no further questions concerning this item.

c. The allegers expressed concern with the way welders are presently being tested and certified by the licensee. The allegers feel that there is insufficient monitoring by the licensee in all areas of the welder certification program.

The licensee has opted to qualify, certify, and maintain welders at the Waterford 3 plant rather than contract these services through another organization.

The licensee has developed approved welding procedures and established a welder training and certification center onsite. All welder testing is performed under the direction of the LP&L assistant mechanical superintendent with the accept/reject responsibility assigned to the LP&L plant quality group. The NRC inspector reviewed 12 randomly selected welder qualification records and conducted interviews with three representatives of the plant quality group. From this review and the interviews, it was apparent to the NRC inspector that the qualification and certification of welders was being conducted in accordance with Procedure MM-1-052, Revision 2, and the ASME Code. The records generated for each welder test indicated that, as a minimum, the material identification, welder identification, test coupon fit-up, and test coupon final weld

surfaces were verified or observed by a member of the plant quality group. Interviews with cognizant personnel also revealed that, in addition to the documented observations made by the licensee, unscheduled visits were made several times daily by the plant quality group. Acceptance or rejection of a test coupon is based solely on the results of radiography or physical bend test as observed by the licensee.

The subject of the allegation is whether the licensee is providing adequate monitoring of the welder certification program. Section IX, Article III, of the ASME Code states that the certification of the welders is the responsibility of the organization performing the welding (the licensee) and cannot be delegated. The Code further states that the welders or welding operators shall be tested under the full supervision and control of the licensee. The NRC inspector determined that the monitoring activities and the documentation maintained by the plant quality group in the area of welder certification is adequate and meets the intent of the Code. The NRC inspector had no further questions concerning this item.

d. The allegers stated that multiple chain-falls and "come-alongs" were utilized in the fit-up of an 8" diameter stainless steel piping spool in startup system 52-A2. The allegers stated that a QC inspector refused to accept the fit-up inspection with these devices in place, so all but one of the "come-alongs" were released. The allegers stated that the tack welds broke, indicating that the pipe had been "cold-sprung" into position. The pipe was allegedly manipulated back into place with the "come-alongs" and retacked. The QC inspector allegedly accepted the second fit-up inspection in the "cold-sprung" condition. The allegers stated that this acceptance was made "very reluctantly."

The NRC inspector interviewed cognizant licensee and EBASCO engineers in an attempt to ascertain the exact piping spool and weld in question as the allegers could not provide these details. Through these interviews and a review of the record package for startup system 52-A2, it was determined that the pipe spool and weld in question was in line 6RC8-21. This is the pressure safety valve outlet to the quench tank as shown on drawing 150-8469-724 R2/IC-899-E, Sheet 2, Revision 13. Although this line was not considered to be safety-related and required no inspection, EBASCO QC did perform a 100 percent visual inspection of all final weld surfaces.

The NRC inspector determined that during the performance of Design Change Notice (DCN) MP-688 R/1 which replaced an existing 2" x 8" safety relief valve on the pressurizer with a 6" x 8" valve, it was necessary to cut line 6RC8-21. This work was accomplished under Condition Identification Work Authorization (CIWA) 82A236

issued on August 31, 1982. When the line was cut to perform this work, the pipe moved, indicating that it had been "cold-sprung" into place during the fit-up operation. This was observed by EBASCO engineering personnel who were present during the repair.

The record package indicates that several welds were reworked, as a result of this discovery, to realign the pipe without forcing it into position. Records indicate that several QC inspectors were involved during the original welding and the rework of this pipe but none of them was the inspector mentioned by the allegers.

The NRC inspector determined that an 8" line in the startup system 52-A2 apparently had been "cold-sprung" prior to welding. This condition, however, had been discovered by EBASCO and corrected prior to the turnover of the system to the startup and test group. The NRC inspector had no further questions concerning this item.

e. The allegers stated that, in an effort to accept substandard welds and multiple weld repairs, the acceptance criteria for the reactor coolant system piping "C" stops were changed from radiography to magnetic particle inspection. The allegers feel that this downgrading of requirements could affect the functioning of the "C" stops.

The NRC inspector reviewed the fabrications and installation records of the RCS "C" stops and conducted interviews with the same personnel mentioned in the evaluation of Allegation "a" above. The reasons for changing the acceptance criteria and the justification for the changes were listed on individual FCRs for each "C" stop. Each was evaluated and dispositioned on a case-by-case basis by EBASCO engineering. It was noted by the NRC inspector that the reasons for changing the "C" stop design and examination was due to alignment and fit-up difficulties and not the weld cracking problem noted in Allegation "a" above. Following his interviews and record review, the NRC inspector had no further questions concerning this item.

No violations or deviations were identified.

11. <u>QA Personnel Qualifications</u>

During the week of February 25, 1985, the Waterford Task Force QA Team completed the review of qualifications of QA personnel at the Waterford site. The NRC staff committed to complete this review by March 1, 1985, in SSER 9.

QA personnel included managers, supervisors, auditors, record reviewers, clerks, and secretaries. These individuals were not assigned responsibilities or performed any functions that required them to be qualified under the requirements of ANSI N45.2.6; i.e., their

certification was not required. The requirement for these individuals (other than auditors) was various amounts of formal, and/or on-the-job training and some relative experience commensurate with their job description and responsibilities. The requirements for auditor qualifications are delineated in ANSI N45.2.12, Draft 3, Revision 4 - 1974 (LP&L's commitment). This standard requires training, relative experience, and independence from the area being audited. The NRC staff reviewed the site standard practices (procedures) pertaining to all QA personnel and found them to be acceptable and to meet LP&L's commitments.

The LP&L evaluation included background verification, qualification, and determinations. The NRC staff also reviewed the corrective action accomplished to resolve the problems associated with personnel who were determined to be not qualified by the LP&L evaluation. LP&L Procedure QASP 19.12 contains the guidelines used in this evaluation of QA personnel qualifications. The guidelines were based on training and indoctrination as well as ANSI N45.1.12-1974 requirements for auditors. Qualification requirements for other QA personnel are not specifically defined by ANSI/ASME standards or NRC regulations.

The NRC staff reviewed the LP&L evaluation of personnel qualifications using a statistically based sampling plan. The following is a summary of the NRC staff review.

a. Auditors

LP&L used the criteria of the Green Book, WASH 1283, Revision 1 (5/24/74)/ANSI N45.2.12 (2/22/74), Draft 3, Revision 4 to evaluate auditor qualifications. At least one or more of the following had to be apparent:

(1) Orientation to ANSI N45.2 and ANSI N45.2.12

(2) Training program on audit performance

(3) On-the-job audit training

The NRC staff reviewed a sample of auditor qualification records, including evidence of the above requirements, background verification, and disposition of auditors determined not qualified.

The NRC staff concurs with LP&L's evaluation and disposition of auditor qualifications.

b. Document Reviewers

The job responsibilities of document reviewers consisted of verifying that quality documents contained the information which had been previously identified as being required for inclusion. Essentially,

the fulfillment of these responsibilities did not require the need for a document reviewer to have a significant technical knowledge, but did require that he knew what QA information had to be contained in specific documents. Consequently, the LP&L evaluation of document reviewer's qualification consisted of determining if those personnel employed as document reviewers received appropriate training to acquire this knowledge. The results of this evaluation concluded that all personnel employed as document reviewers did receive the training needed to qualify them to perform QA reviews of one or more types of quality documents and concluded, therefore, that these personnel were adequately qualified to perform these tasks. The NRC staff review of this LP&L evaluation concurs with this conclusion. This conclusion was in concert with resolution of Allegations A-06, A-09, A-289b, A-196, and A-306t as stated in SSER 7.

c. Other QA Personnel

Other personnel performing QA tasks included QA engineers, specialists, supervisors and managers. Since QASP 19.12 does not contain guidelines for evaluation of the qualifications of these personnel, the evaluation of these qualifications were in accordance with the licensee, contractors and subcontractors, QA programs. The NRC staff reviewed the qualifications of selected personnel listed in this category and agreed with the LP&L assessment that they had sufficient qualifications to perform QA tasks normally associated with these job classifications. It was, therefore, concluded that these personnel possessed the necessary qualifications to perform the assigned QA task.

d. Personnel Identified as Not Qualified

Of the 657 personnel evaluated by LP&L, 115 were determined not qualified (questionable or indeterminate). Corrective action requests were prepared and issued to document the unqualified personnel, and the following was determined:

(1) Forty individuals were determined qualified based on additional information obtained; i.e., background checks and training records.

(2) Forty individuals had been previously addressed as a result of audits performed by EBASCO at Waterford 3. These audits were reviewed by LP&L and the staff and were found acceptable.

(3) Fourteen individuals were identified as performing document reviews characterized as a clerical function and did not perform the detailed review for final acceptance.

(4) The 21 remaining individuals did not meet the requirements of QASP 19.12 and were dispositioned as follows:

(a) Nineteen individuals were auditors performing audits. Seventeen individuals performed the assigned audits under the supervision and with the audit results approved by a qualified lead auditor. The audit conducted by one individual was reaudited by a qualified Level III/Lead Auditor utilizing the same audit checklist and produced the same results. This reaudit was conducted immediately following the questionable audit due to questionable qualifications. The remaining individual performed audits in accordance with plans and audit checklists approved by a qualified site QA manager. The individual was properly trained in auditing techniques based on the approved audit procedure. The audits were conducted under the supervision and the results of the audit were approved by a qualified QA manager.

(b) Two individuals performed the duties of QA engineers in accordance with their job descriptions. The initial problem was that the training records for these individuals could not be located. These individuals were onsite a limited amount of time (3 months and 7 months). These individuals acted as EBASCO liaison agents with two site subcontractors. Their duties included issuance and closure of NCRs, performing surveillances, and monitoring subcontractor performance. These individuals worked with qualified QA engineers and under the supervision of a qualified site QA supervisor. Their work was periodically reviewed by the QA supervisor and was found to be acceptable. These two QA engineers were also judged to have sufficient education/experience to perform their job assignments.

The NRC staff reviewed these corrective actions, responses, and dispositions, and concurs with LP&L conclusions. The proper resolution of the corrective action requests determined that there was no impact of the unqualified QA personnel on plant hardware or the QA program. The staff, therefore, has no further concerns with QA personnel at the Waterford 3 site.

No violations or deviations were identified.

12. Open Items

Two new open items were identified in this report in paragraphs 5e and 6.

8505-01 Verification of SPDS parameters

8505-02 Upgrade checklist for oncoming control room operators

13. Site Tour

At various times during the course of this inspection period the NRC inspectors conducted general tours of the reactor auxiliary building, turbine building, and reactor building to observe ongoing maintenance and testing.

No violations or deviations were identified.

14. Exit Interviews

The NRC inspectors met with the licensee representatives at various times during the course of the inspection. The scope and findings of the inspection were discussed.