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Vol. 21, No. 3

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
July - September 1997

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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A year's subscription of this report consists of four quarterly issues.

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
July - September 1997

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Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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ABSTRACT

This periodical covers the results of inspections performed between July 1997 and September 1997 by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations.

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This quarterly report contains copies of all vendor inspection reports issued during the calendar quarter for which it is published. Each vendor inspection report lists the nuclear facilities inspected. This information will also alert affected regional offices to any significant problem areas that may require special attention. This report lists selected bulletins, generic letters, and information notices, and include copies of other pertinent correspondence involving vendor issues.

INSPECTION REPORTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

July 23, 1997

Mr. John J. Connolly, Vice President
Business Development
ABB Service Incorporated
9050-A Red Branch Road
Columbia, Maryland 21045

SUBJECT: NRC INSPECTION REPORT NO. 99901281/97-01

Dear Mr. Connolly:

On May 14, 1997, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection at one of the ABB Service, Incorporated (ABB Service), repair and refurbishment facilities. The enclosed report presents the results of the inspection that was conducted at your Cleveland, Ohio facility, the discussion conducted with you during the exit meeting at the ABB Service Corporate office in Columbia, Maryland, on June 5, 1997, and subsequent discussions between July 14 through July 22, 1997, regarding the enclosed Notice of Violation (NOV).

During this inspection, the NRC inspectors found certain activities to be in violation of NRC requirements. Specifically, ABB Service failed to adequately evaluate several examples of potential defects regarding miswiring errors concerning K-Line circuit breaker solid state trip devices that had been shipped to the Perry Nuclear Power Plant (PNPP). The team found that ABB Service did not evaluate the errors or transmit the information to other ABB Service customers even though the concern was potentially generic.

This violation is cited in the enclosed NOV, and the circumstances surrounding the violation are described in detail in the enclosed report. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed NOV when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

The NRC considers this violation significant because, as early as January 1993, ABB Service was aware of potential premature tripping on new and refurbished K-Line breakers as a result of miswiring errors and reversal of current sensor polarity into the solid state trip device of K-Line low voltage circuit breakers. However, ABB Service did not assure that, as a minimum, all of its circuit breaker refurbishment facilities were aware of the potential deficiency. ABB Service was also aware in January 1993, that single phase testing methods of licensees and ABB Service would not identify a reversed polarity problem which could affect both safety and non-safety-related K-Line breakers. Similarly, ABB Service did not apprise their refurbishment facilities or licensees of the weakness in testing. If ABB Service had apprised licensees of the testing weakness, a report to the NRC would likely have been generated.

Although ABB Service's failure to inform its customers of a potential defect in January 1993 was not a violation of NRC requirements, the staff believes that ABB Service did not act appropriately given the potential problems that could have ensued at operating nuclear power plants. The staff is also concerned that the January 1993 ABB Service evaluation report that delineated this problem may not be the only ABB Service example where appropriate action was not taken.

In addition, the NRC inspectors found that the implementation of your quality assurance program failed to meet certain NRC requirements imposed on you by your customers. Specifically, ABB Service inspections performed on K-Line breakers failed to detect that the current transformers on two circuit breakers were assembled incorrectly. This nonconformance is cited in the enclosed Notice of Nonconformance (NON), and the circumstances surrounding it are described in detail in the enclosed report. You are requested to respond to the nonconformances and should follow the instructions specified in the enclosed NON when preparing your response.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC's Public Document Room (PDR).

Sincerely,

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket 99901281/97-01

Enclosures: 1. Notice of Violation
 2. Notice of Nonconformance
 3. Inspection Report 99901281/97-01

cc: Mr. Joseph M. Tate
 General Manager
 ABB Service Inc.
 Regional Service Center
 5311 Commerce Street
 Cleveland, OH 44130

NOTICE OF VIOLATION

ABB Service
Cleveland, Ohio

Docket No.: 99901281

During an NRC inspection conducted on May 14, 1997, and discussions conducted between July 14-22, 1997, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

10 CFR Part 21.21, "Notification of failure to comply or existence of a defect and its evaluation," requires in part that, (a) Each individual, corporation, partnership, dedicating entity, or other entity subject to the regulations in this part shall adopt appropriate procedures to: (1) Evaluate deviations and failures to comply to identify defects and failures to comply associated with substantial safety hazards as soon as practicable, and, except as provided in paragraph (a)(2) of § 21.21, in all cases within 60 days of discovery.

ABB Service Quality Assurance Procedure (QAP) 15.1, "Control of Nonconforming Items," Revision 4, dated December 15, 1996, implements, in part, requirements contained in 10 CFR Part 21.21. Section 3.5 of QAP No. 15.1 requires that any significant nonconformance in nuclear-safety-related equipment shall be evaluated to determine if a 10 CFR Part 21 report needs to be filed.

ABB Service QAP 15.2, "Reporting of Defects and Noncompliance in Accordance with 10 CFR 21," Revision 2, approved March 24, 1995, states in part, the evaluation of potential defects or potential failures to comply must be completed as soon as practicable, and in all cases, within 60 days of discovery.

Contrary to the above, even though ABB Service was aware of examples of potential defects regarding miswiring errors concerning K-Line circuit breaker solid state trip devices that they had shipped to Perry Nuclear Power Plant (PNPP), ABB Service did not adequately evaluate the errors or transmit information regarding the deviations to other ABB Service customers even though the concern was potentially generic. PNPP identified three refurbished K-Line breakers that had incorrectly wired or installed sensors (Serial Numbers 51817A-107073, 51817C-264135 and 51817D-211135). All three had been incorrectly assembled by ABB Service during refurbishment, tested, and sent back to PNPP by ABB Service. Additionally, PNPP found the "C" phase Power Sensor current transformer inverted on a K-600S low voltage breaker (Serial Number 51817D-211135), and identified the problem to ABB Service on April 28, 1997. (99901281/97-01-01)

This is a Severity Level IV violation (Supplement VII).

Notice of Violation

- 2 -

Pursuant to the provisions of 10 CFR 2.201, ABB Services Inc., is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 23rd day of July 1997

NOTICE OF NONCONFORMANCE

ABB Service
Cleveland, Ohio

Docket No.: 99901281

Based on the results of an inspection conducted on May 14, 1997, it appears that certain of your activities were not conducted in accordance with NRC requirements.

Criterion X of Appendix B to 10 CFR Part 50, "Inspection," requires, in part, that a program for inspection of activities affecting quality be established and executed by or for the organization performing the activity to verify conformance with documented instructions, procedures, and drawings for accomplishing the activity.

Contrary to the above, two safety-related K-Line breakers that had been returned by ABB Service, Cleveland, to a licensee after being refurbished were not properly inspected to verify their conformance to documented drawings, specifically: (99901281/97-01-02)

- On November 8, 1996, maintenance personnel at the licensee discovered that the control wires on terminals 12 and 13 of the Power Shield trip unit were reversed on breaker S/N 51817A-107073.
- On April 28, 1997, maintenance personnel at the licensee discovered that the power sensor on phase "C" was inverted (i.e., it was installed upside down) on breaker S/N 51817D-211135.

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed.

Dated at Rockville, Maryland
this 23rd day of July 1997

INSPECTION REPORT

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
DIVISION OF INSPECTION AND SUPPORT PROGRAMS

ORGANIZATION: ABB Service Incorporated
Cleveland Service Center
5311 Commerce Parkway West
Cleveland, Ohio 44130

REPORT NO.: 99901281/97-01

ORGANIZATIONAL CONTACT: Mr. D.E. Leckey
Quality Assurance (QA) Manager
(216) 267 2882

NUCLEAR INDUSTRY ACTIVITY: Can provide new ABB switchgear, procurement of replacement parts, maintenance, refurbishment and on-site switchgear services.

INSPECTION CONDUCTED: May 14, 1997

SUBSEQUENT DISCUSSIONS: June 5 and July 14-22, 1997

INSPECTORS: Kamalakar R. Naidu, NRR
Virgil L. Beaston, NRR
Joseph J. Petrosino, NRR

APPROVED BY: Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

1 INSPECTION SUMMARY

ABB Service Incorporated (ABB Service), is headquartered at its Columbia, Maryland, Service Center. Prior to 1996, ABB Service had approximately 17 service center facilities throughout the United States that could have potentially performed engineering, consulting services, or other work on, or related to, safety-related circuit breakers for NRC licensees. However, due to quality concerns, ABB Service modified that policy and currently allows only four of its service centers to process safety-related circuit breaker work. They are: Cleveland, Ohio Service Center; Columbia, Maryland Service Center; Houston, Texas Service Center; and Charlotte, North Carolina Service Center.

The ABB Service Center located in Cleveland, Ohio, has been performing refurbishment services on metal-clad, low and medium voltage, K-Line circuit breakers installed at Centerior Energy's Perry Nuclear Power Plant (PNPP) for several years. The K-Line breakers were originally designed and manufactured by I.T.E. Imperial Company (ITE), which changed ownership and became known as I.T.E.- Gould, Gould-Brown Boveri, Brown Boveri Electric and finally ASEA Brown Boveri. ABB Power Transmission & Distribution Company, Incorporated (ABB Power) manufactures metal-clad low-voltage and medium-voltage K-Line breakers at Florence, South Carolina. ABB Power at Sanford, Florida assembles complete switchgear installations. The Cleveland Service Center currently services safety-related K-Line breakers only for PNPP.

On May 14, 1997, the inspectors reviewed records documenting the refurbishment work performed on PNPP breakers, reviewed the actions taken to correct nonconformances identified in report 99901281/94-01, and performed subsequent review of records and documents associated with the reversed polarity issue in K-Line circuit breakers. Additionally, the inspectors reviewed an ABB Service Report performed by ABB Service for Public Service Electric and Gas Company's (PSE&G) Salem nuclear generating station (Salem) as discussed in Section 3.2.b.3. The inspection bases were:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50)
- 10 CFR Part 21, "Reporting of Defects and Noncompliance"

During this inspection, the inspectors identified one nonconformance (Section 3.1.b.3 and 3.1.b.5) and one violation (Section 3.2.)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

Unresolved Item 94-01-01 (Closed). The previous inspection identified that a current organization chart was not available. The team determined that ABB Service has a current organization chart which depicts the organization.

Unresolved Item 94-01-02 (Closed). The previous inspection identified that job descriptions and qualifications of persons performing safety-related activities were not available. The team identified that ABB Service has identified and delineated job descriptions and qualifications of persons performing safety-related activities.

Nonconformance 94-01-03A (Closed). The previous inspection identified that potential deviations were not appropriately dispositioned. Although the team determined that adequate corrective action was not performed for this issue, similar corrective action will be required as a result of Violation 99901281/97-01-01.

Nonconformance 94-01-03B (Closed). The previous inspection determined that not all contact resistances were delineated for purchaser use. The team determined that maximum acceptable contact resistance values are now furnished in the ABB Service test instructions.

3 INSPECTION DETAILS

3.1 K-Line Breaker Refurbishment

a. Scope

The inspectors reviewed circumstances that led to incorrectly assembled ABB Service K-Line breakers being delivered to PNPP. During and subsequent to the inspection at ABB Service's Cleveland facility, the inspectors:

- visited PNPP to examine the suspect K-Line circuit breaker,
- reviewed the purchase order (P.O.) PNPP issued to ABB Power-Florence, for the supply of new ABB K-Line low-voltage metal-clad K-Line breakers,
- examined the documentation related to the work performed by ABB Service on PNPP breakers to determine if it met the P.O.,
- observed ABB Service technicians perform work on selected K-Line breakers at the Cleveland Service Center,
- discussed K-Line breaker wiring problems identified by PNPP and other licensees with representatives of ABB Service, and
- reviewed associated documentation and conducted discussions regarding identified K-Line polarity problems including PNPP and Salem.

b. Observations and Findings

b.1 K-Line Breaker Failure at PNPP

On October 4, 1996, while PNPP was operating at 100 percent power, a new ABB K600S K-Line Breaker (Serial Number 934277-031295) supplying 480-volt power to a motor control center prematurely tripped. PNPP personnel investigated the cause of this incident and determined that ABB Power-Florence, had shipped the K-Line Breaker with two lead wires from one of the breaker's three phase sensors (current transformers) reversed. PNPP shipped the breaker to ABB Power for analysis. On March 26, 1997, ABB Power-Sanford, informed the Nuclear Regulatory Commission in accordance with 10 CFR Part 21 requirements that the two wires on the phase-C phase sensor were crossed on the breaker that was shipped to PNPP. Crossing the phase sensor leads changed the polarity of the phase sensor and caused the K-Line Breaker's Power Shield trip unit to trip at 350 amperes of primary current instead of the Power Shield trip setpoint setting of 660 amperes.

Six doughnut-shaped current transformers (CTs) are mounted on the lower molding current transformer assembly of the K600S K-Line breakers equipped with trip units. Three of the six CTs are referred to as phase sensors, and they are used to detect fault currents. The other three are referred to as power sensors, and they are used to develop a reference signal within the Power Shield trip unit. One phase sensor and one power sensor are installed on each phase of the K-Line Breaker. According to Revision 16 of ABB Power Distribution Inc. Drawing 709551, "Aux. Physical Wiring Drawing," the phase sensors are mounted on top of the power sensors and the leads of both the sensors are terminated on a terminal block attached to the lower molding. Two leads emerge from the top of each sensor, and a red dot (a polarity mark) on the top of each sensor distinguishes the polarity of the sensor. The other ends of the lower terminal blocks are connected to the appropriate terminals of the Power Shield trip unit using a multiconductor cable.

In the new breaker that tripped prematurely at PNPP, the blue wire from the phase-C phase sensor, which should have been landed on terminal 2 of the lower terminal block, was found fastened to terminal 1; the yellow wire, which should have been fastened on terminal 1, was founded to be fastened to terminal 2. This error caused an errant, phase-shifted signal to be sent from the phase-C phase sensor to the K-Line Breaker's Power Shield solid state trip unit. The errant signal caused the trip unit to sense an abnormally high current value, tripping the breaker.

After learning that an incorrectly wired phase sensor reduces the amount of primary current needed to actuate ABB's Power Shield trip unit, PNPP began conducting secondary wiring checks of ABB K-Line circuit breakers. The wiring checks performed by PNPP identified three refurbished K-Line breakers that had incorrectly wired or installed sensors (Serial Numbers 51817A-107073, 51817C-264135 and 51817D-211135). All three of these incorrectly assembled K-Line breakers had been refurbished and tested by ABB Service.

The inspectors determined that current and phase sensors (sensors) wiring and assembly errors may occur in any of the following manners:

- Crossing wires from the sensors to the terminal block attached to the lower molding current transformer assembly;
- Crossing wires from the terminal block on the lower molding to the terminal block on Power Shield Static Trip Unit.
- Incorrectly re-connecting selected wires on the Power Shield Static Trip Unit after the wires have been lifted to conduct K-Line Breaker tests.
- Installing a sensor upside down or incorrectly wiring a sensor when the lower molding current transformer assembly of the K-Line Breaker is disassembled (to replace a sensor).

Additionally, as discussed in Section 3.2.b.2 below, the reversed polarity problem in K-Line circuit breakers appears to have been initially identified by ABB Service in the 1992-1993 time period. ABB Service was aware that the polarity problem could have the potential to exist in new or refurbished K-Line breakers and that single phase testing methods typically used by the licensees and ABB Service organization would not identify the problem. The problem could affect safety and non-safety-related K-Line breakers. However, ABB did not take action to assure the circumstances of the problem were affectively disseminated.

b.2 Procurement Documents

Purchase Order (P.O.) S137920, Rev. 3, dated June 14, 1996, controlled the work performed on safety related K-Line breakers including S/N 51817D-211135. Attachment 1, Rev. 002, Section C, "Quality Assurance Program Requirements," to P.O. S137920 required certain QA program requirements to apply to all rework, replacement part procurement, inspection, testing, handling, storage and shipping of K-Line breakers returned for refurbishment, including 10 CFR Part 50, Appendix B, and 10 CFR Part 21.

Attachment 1, Section E, "Quality Assurance Records," Item 2 to P.O. S137920 required, in part, the following documentation to be submitted, as applicable:

- ABB Service Certificate of Compliance (CoC).
- Procedural Checklist for Safety-Related Nuclear Switchgear (ABB Service Form QA2).
- Final Inspection/Acceptance Criteria Checklist (ABB Service Form QA6).

Attachment 1A, Rev. 000 (01/16/95), "Technical Requirements Safety Related K-Line Breaker 10 Year Refurbishment," to P.O. S137920, Rev. 3, requires in part that:

A complete visual inspection of all areas of the K-Line Breaker shall be performed to verify configuration complies with factory approved physical assembly configuration... Any visual inconsistency... shall be further investigated. These activities shall be performed utilizing factory approved design documents... Any observed discrepancies shall be noted and the K-Line Breaker reworked to comply with the specified configuration

b.3 K-Line Breaker S/N 51817A-107073

K-Line Breaker S/N 51817A-107073 is a safety related type ABB K3000S breaker. PNPP sent this breaker to ABB Service for an inspection, lubrication evaluation, 10-year refurbishment, and repair in accordance with P.O. No. S137920. ABB Service refurbished and tested the breaker and returned it to PNPP with a CoC, dated August 23, 1995. The CoC was signed by the Cleveland Service Center QA Manager. On September 1, 1995, PNPP put the breaker in service.

On November 8, 1996, PNPP maintenance personnel discovered that control wires connecting the power sensors on the breaker lower molding to the Power Shield trip unit were incorrectly wired. Based on Perry's information, ABB Service generated Nonconformance Report (NCR) ABB Service Job # 43-02714-26 on November 8, 1996.

The ABB Service NCR stated that during primary current injection testing, Power Shield control wires 11 through 14 are lifted and shorted together. After testing, the control wires are re-connected to the Power Shield terminal block. The NCR stated that the wiring error discovered by PNPP may have occurred during the re-connection of the control wires following final testing of the breaker. The ABB Service NCR stated that the reversal of the control wires on terminals 12 and 13 of the Power Shield would not have affected proper operation of the breaker, and therefore there was no impact to plant operability. The line item "Potential Part 21 Evaluation Required?" on the NCR was checked "No" and this response was signed by both the Service Center Manager and the Service Center QA Manager.

The actions taken by ABB Service to correct this nonconformance and prevent recurrence were to add line items to checklists ABB Service Form QA2, "Procedural Checklist for Safety Related Switchgear," and ABB Service Form QA6, "ABB Service QA Final Inspection/ Acceptance Criteria Checklist," to require point-to-point wiring checks of all controls including Power Shield, power and phase sensors, and on both ends of the Power Shield wiring harness to verify wiring configuration. The inspectors verified that these corrective actions were implemented by ABB Service with the issuance of Rev. 8 of ABB Service Form QA2 and Rev. 3 of ABB Service Form QA6. Both revisions were dated November 14, 1996.

The inspectors informed ABB Service personnel that failure to perform an adequate inspection of K-Line Breaker S/N 51817A-107073 is a nonconformance contrary to Perry's P.O., and Criterion X of Appendix B to 10 CFR Part 50. (This is one example for Nonconformance 99901281/97-01-02).

b.4 K-Line Breaker S/N 51817C-264135

K-Line Breaker S/N 51817C-264135 is a non-safety-related type ABB K600S K-Line Breaker. In November 1995, PNPP sent this breaker to ABB Service where it was refurbished and returned to PNPP on September 4, 1996. Upon receipt, the breaker was placed in a PNPP warehouse.

On March 20, 1997, while performing pre-installation checks on the K-Line Breaker, PNPP personnel discovered that the phase-B phase sensor was incorrectly wired. Based on Perry's information, ABB Service generated NCR ABB Job # 43-02902-3. In this NCR also, the line item "Potential Part 21 Evaluation Required?" was checked "No" and this response was signed by the Service Center Manager. The NCR stated that PNPP was assured that the revised ABB Service Forms QA2 and QA6 were being utilized to prevent recurrence of future wiring errors.

From the available documentation, the inspectors could not determine who caused the wiring error. The leads were not reversed at the Power Shield, as in the example above, but at the current sensor terminal block on the lower molding. ABB Service personnel informed the inspectors that these leads are not disconnected during routine refurbishing work unless the current sensors are replaced. The documentation reviewed by the inspector indicated that the current transformers were not replaced for this breaker.

ABB Service personnel informed the inspectors that this error was not identified because the breaker was refurbished and shipped before the issuance of Rev. 8 of ABB Service Form QA2 and Rev. 3 of ABB Service Form QA6. Because this is a non-safety-related breaker, the inspectors did not identify this condition as a nonconformance. However, the team considered this as an example that should have been evaluated as affecting safety-related breakers since K-Line safety and non-safety-related breakers are identical in this area.

b.5 K-Line Breaker S/N 51817D-211135

K-Line Breaker S/N 51817D-211135 is an ABB K600S type safety-related breaker. PNPP sent this breaker to ABB Service for an inspection, lubrication evaluation, 10-year refurbishment, and repair in accordance with Purchase Order Number S137920, Rev. 3. On April 16, 1997, ABB Service tested the breaker and returned it to PNPP with a CoC, dated April 16, 1997, signed by the Cleveland Service Center QA Manager certifying that the breaker met the P.O. requirements. On April 28, 1997, PNPP personnel discovered that one of the breaker's power sensors was inverted (i.e., it was installed upside down).

PNPP reported this nonconforming condition to Corporate ABB Service and ABB Service documented this condition in NCR ABB Service Job # 43-02939, dated April 29, 1997. In the NCR, the line item "Potential Part 21 Evaluation Required?" was checked "No" and this response was signed by the Service Center QA Manager.

The inspectors determined from the available documentation that this nonconforming condition was caused by ABB Service during replacement of the lower molding current transformer assembly. The K-Line Breaker repair worksheet (ABB Service Form CBRW, Rev. 4., Dated March 10, 1995) for breaker S/N 51817D-211135 indicated that on April 2, 1997, ABB Service replaced the phase-C lower assembly which contained the upside-down power sensor. The ABB Service Form CBRW for breaker S/N 51817D-211135 was signed and approved by the Service Center QA Manager on April 15, 1997.

On April 16, 1997, ABB Service issued a CoC for Breaker S/N 51817D-211135, signed by the QA Manager, which certified that the repairs and final tests on the subject apparatus, as described in Service and Test Reports, complied with the requirements of the subject purchase order, applicable industry standards, and the specifications of the original contract.

Based on the above, the inspectors informed ABB Service personnel that failure to perform an adequate inspection of K-Line Breaker S/N 51817D-211135 is another example of a nonconformance contrary to Criterion X, "Inspections," of 10 CFR Part 50, Appendix B, PNPP's purchase order, and Step 9 of ABB Service Form QA2, Rev. 8, which required that a point to point wiring check be performed on all components, including the power and phase sensors (this is a second example for Nonconformance 99901281/97-01-02).

c. Conclusions

Based on the above, the inspectors determined that ABB Service had not properly developed and implemented an inspection program to assure that refurbished breakers conformed to their original design.

3.2 10 CFR Part 21 Procedure & Evaluations

a. Scope

The team reviewed the adequacy of the procedure that ABB Service adopted to implement the provisions of 10 CFR Part 21, Quality Assurance Procedure (QAP) 15.2, "Reporting of Defects and Noncompliance with 10 CFR 21," Revision 2, approved March 24, 1995, and reviewed selected deviations to determine whether they were dispositioned in accordance with the provisions of 10 CFR Part 21.

b. Observations and Findings

b.1 Procedure

The team reviewed QAP 15.2, Revision 2, to assess whether it effectively implemented the provisions of 10 CFR Part 21 and ensured that identified deviations or failures to comply were appropriately evaluated or transmitted to the purchasers. The team noted that revision 2 of QAP 15.2 is eight pages long with an additional seven pages of "supplemental guidance," and has a two page form, "Potential Noncompliance or Defect Report," to be completed by the ABB Service originator.

The inspectors reviewed the procedure as if a deviation had been identified, and determined that the procedure was comprehensive and contained useful notes and guidance to assist the evaluator. The team noted that several of the Part 21 definitions were not in accordance with the latest revision of Part 21 and some definitions also contained ABB Service clarifications or interpretations that were integrated into the definition. Although the clarifications appeared to be helpful, the integration of the ABB Service narrative could be misleading. Therefore, the inspectors recommended that ABB Service assure that the clarifications are discernable from the Part 21 definitions (such as, by the use of brackets). Additionally, Section 21.3, "Interpretations," of Part 21 does not allow interpretations of the meaning of Part 21 verbiage except for written interpretation by the NRC General Counsel.

The definitions that were contained in QAP 15.2 appeared to be from the 1995 revision. 10 CFR Part 21 was modified in 1996, but QAP 15.2 did not reflect the modifications. Consequently, some of the QAP 15.2 definitions are not in accordance with the current revision of 10 CFR Part 21.

The team also reviewed the ABB Service "Potential Noncompliance or Defect Report" form that was an attachment to QAP 15.2 and noted that it contained instructions/guidance that are not appropriate for the audience that would be expected to be use the form. The requirements on the form state:

- A. Identification of the basic component or activity which contains a "defect" or "failure to comply."
- B. Identification of the Company supplying the basic component or activity which contains the "defect" or "failure to comply."
- C. Nature of the "defect" or "failure to comply" and the safety hazard which is or could be created.
- D. Number and location of all affected components: (include identification of all purchasers to whom component has been supplied)
- E. Corrective action, identification of party responsible or action, and schedule for action.

The team determined that these requirements may not be fully understood by the average employee/technician. Additionally, the requirements could tend to have somewhat of a chilling effect. The team believes that an employee would not typically have the expertise to identify whether a "defect" or "failure to comply" exists. Therefore, the team felt it possible that an employee would not complete the form since the employee would be unable to provide the required information. The employee's responsibility should be limited to identifying problems or deviations to their supervision to ensure that all potential defects are identified and dispositioned in accordance with 10 CFR Part 21. This form, if not understood completely by the ABB Service personnel, may have prevented deviations from being reported in the past.

After discussing the 10 CFR Part 21 regulation, intent and QAP 15.2, with ABB Service staff, the team informed ABB Service that it had determined that it had not established an adequate procedure to appropriately implement 10 CFR Part 21. This failure is characterized as a violation of minor significance and will be treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. The ABB Service Director of Quality committed to resolving the NRC concerns.

b.2 PNPP Evaluations

The team reviewed licensee and ABB information regarding identified problems with the K-Line low voltage circuit breakers. On three separate occasions, as discussed above, PNPP reported that ABB Service refurbished and tested K-line low voltage breakers were found by PNPP to contain wiring or assembly errors. Although some of the errors could cause a breaker to trip prematurely, in each of the NCRs, the line item "Potential Part 21 Evaluation Required?" was checked "No" and signed by an ABB Service Manager.

Premature tripping, or false trips are a concern especially if false trips occur during transient events when the breaker is required to carry its full current rating. Further, because single phase calibration testing does not adequately verify the overcurrent trip set points of some circuit breakers equipped with three phase solid state trip units, the potential exists for incorrectly wired or assembled circuit breakers to pass ABB Service and licensee calibration testing, but prematurely trip during a design basis accident, resulting in a loss of safety function.

The team discussed the disposition of these NCRs with the ABB Service Quality Director. The team asked the Quality Director why there were not any 10 CFR Part 21 evaluations performed. The Quality Director stated that in retrospect, they should have evaluated the issue, regarding breaker S/N 51817D-211135, in accordance with 10 CFR Part 21. The ABB Service personnel also informed the inspectors that they did not evaluate the potential that they may have previously shipped safety related K-Line breakers with errors similar to those identified in the NCRs.

The inspectors informed ABB Service personnel that failure to identify potential defects and either evaluate or inform the purchasers was a violation of 10 CFR Part 21. Additionally, recurring current sensor errors in safety-related K-Line breakers in previously shipped safety-related K-Line breakers was a potentially generic problem that ABB service did not recognize, evaluate or inform ABB Service customers so its customers could determine if the condition existed at their facilities. Violation 99901281/97-01-01 was identified in this area.

b.3 ABB Service Evaluation

The team also reviewed an ABB Service report performed for Public Service Electric and Gas Company's Salem nuclear generating station (Salem) by the ABB Service Company's Mount Laurel, New Jersey facility. The report, "Harmonic Measurements and Circuit Breaker Tripping Evaluation," dated December 1992-January 1993, was requested by Salem due to unexplained K-Line circuit breaker

tripping during starting and normal operating conditions. The report stated the existing circuit breaker had been tested by single-phase fault simulation injection, and all tests indicated proper tripping functions; therefore, the cause of the premature tripping was unidentifiable by the licensee. Consequently, the licensee contracted ABB Service for further investigation. The ABB Service report stated in part:

- The computer and laboratory simulation identify the effects of improper current sensor connections to the solid state trip unit. The results clearly indicate the potential for premature tripping, if improper current sensor polarity wiring exists.
- It appears that the error in polarity may have existed prior to shipment from the ABB manufacturing factory....
- NOTE: Single-phase overload testing is necessary, but will NOT identify improper polarity of the... current sensor connections.
- The [ABB Service] laboratory simulation confirmed that the mathematical results which clearly indicate that the reversal of one phase will result in an artificially high input into the logic box sensing circuitry. Two times normal, to be specific. This would cause premature tripping....
- The comparison of the field measurements and the circuit breaker limitations, with one phase reversed, clearly indicate that having one phase reversed was a contributing factor to premature tripping during normal loading and inrush operation.
- In addition, if ABB type K-Line circuit breakers have been serviced, by in-house [licensee] personnel... these circuit breakers should be identified and have the polarities of the current sensors checked during the next scheduled outage... If any of the identified circuit breakers supply loads whereby premature tripping is of an operating or safety concern, then provisions should be made to allow these circuit breakers to have the polarities checked as soon as possible.

The team determined that although ABB Service personnel were aware of this problem as early as January 1993, ABB Service failed to inform its customers of the potentially generic latent defect that could have existed on any new or refurbished ABB K-Line breakers.

c. Conclusions

c.1 Procedure

The team concluded from its review of QAP 15.2, Revision 2, and its attachments that the ABB Service procedure was cumbersome, contained outdated 10 CFR Part 21 definitions, and could have resulted in misleading ABB Service employees into believing that they must perform a review of the circumstances

surrounding a deviation to determine the safety-significance and the safety hazard which could be created. Therefore, the team concluded that the current revision of QAP 15.2 would not effectively implement all of the provisions of 10 CFR Part 21.

c.2 PNPP Evaluations

The team concluded that ABB Service failed to either perform an evaluation of the potential reversed polarity and miswiring or to inform all of its customers so they could determine if a problem existed.

c.3 ABB Service Evaluation

The team concluded that although ABB Service was aware of the potential for reversed polarity on new or refurbished K-Line low voltage circuit breakers as early as January 1993, ABB Service failed to act appropriately given the potential problems that could have ensued at operating nuclear power plants as a result of the potentially generic K-Line series matter.

4. **PERSONS CONTACTED**

ABB Service Company

J.J. Connolly, Vice President, Business Development
D.E. Leckey, Quality Assurance Manager, Cleveland Service
E. Link, Manager, Cleveland Service
J.M. Tate, General Manager, North Central Region
J.O. Webb, Director of Quality

Centerior Energy Company

M.R. Fournier, Quality Engineer, PNPP Nuclear Assurance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

99901281/97-01-01	NOV	inadequate evaluation/failure to inform
99901281/97-01-02	NON	inadequate inspections

Closed

99901281/94-01-01	URI	unavailability of organization chart
99901281/94-01-02	URI	unavailability of job descriptions
99901281/94-01-03A	NON	inadequate qualitative criteria
99901281/94-01-03B	NON	acceptable contact resistance not available

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 29, 1997

Mr. Wilfred C. LaRochelle, Manager,
Quality Assurance
Hartford Steam Boiler Inspection and Insurance Company
One State Street
P.O. Box 5024
Hartford CT 06103-3102

SUBJECT: NRC INSPECTION REPORT 99900601/97-01 AND NOTICE OF
NONCONFORMANCE

Dear Mr. LaRochelle:

On July 18, 1997, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection of Hartford Steam Boiler Inspection and Insurance Company offices in Hartford, CT and Atlanta, GA. The enclosed report presents the results of that inspection.

During this inspection, the NRC inspectors identified several instances where the implementation of your quality assurance program failed to fully comply with NRC requirements imposed on you by your customers and with American Society of Mechanical Engineers (ASME) requirements that are applicable to your activities under the scope of your ASME Certificate of Accreditation. Specifically, the NRC inspectors determined that internal audits of the Home Office and regional office Engineering Services activities had not been conducted at the required intervals, and that nonconformity reports were not issued to document, correct, and disposition the findings of those audits that had been performed. A contributing factor to the identified conditions appeared to be a lack of controlled procedures or instructions for the performance and documentation of audits or for the disposition of audit findings. Additionally, the inspectors identified that the qualification files for two lead auditor candidates did not contain adequate documentation to support the point scores assigned on the basis of their nuclear industry experience.

These nonconformances are cited in the enclosed Notice of Nonconformance (NON), and the circumstances surrounding them are described in detail in the enclosed report. You are requested to respond to the nonconformances and should follow the instructions specified in the enclosed NON when preparing your response.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC's Public Document Room.

Sincerely,

ORIGINAL SIGNED BY GREGORY CWALINA FOR:

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No.: 99900601

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99900601/97-01

NOTICE OF NONCONFORMANCE

Hartford Steam Boiler Inspection
and Insurance Company
Hartford, CT

Docket No.: 99900601

Based on the results of an inspection conducted on July 14 through 18, 1997, it appears that certain of your activities were not conducted in accordance with NRC requirements imposed on you by your customers, or with the requirements of The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) that are applicable to your activities under the scope of your Certificate of Accreditation.

- A. Criterion XVIII, "Audits," of Appendix B to 10 CFR Part 50 requires, in part, that a comprehensive system of planned and periodic audits shall be carried out to verify all aspects of your quality assurance program and to determine the effectiveness of the program.

Engineering Services Manual (ESM) Chapter 4400, "Audits," states, in part, in Section 4421, that the Quality Assurance Manager is responsible for annual audits of the Regional Manager, Engineering Services (RMES) activities and that the Internal Audit Department shall audit the activities of the Home Office (HO) Engineering Services (ES) department ASME activities.

Contrary to the above,

1. Hartford Steam Boiler and Insurance Company (HSB) could only provide documented evidence that one audit (November 1995) of HO ES activities had been performed during the last five years. The HSB Quality Assurance Manager (QAM) stated that the Internal Audit Department no longer conducts audits of HO ES ASME activities and no other HSB organization or department has been assigned that responsibility.
 2. HSB did not perform the annual audits of RMES activities for San Francisco (1996), Atlanta (1995), and Northeast/Philadelphia (1994 and 1995).
(Nonconformance 99900601/97-01-01)
- B. Criterion XVI, "Corrective Action," of Appendix B to 10 CFR Part 50 requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected.

ESM Chapter 4300, "Control of Nonconformities," states, in Section 4310, that this Chapter outlines the requirements for the identification, documentation, and disposition of nonconformances to the ESM, supporting procedures or instructions.

Enclosure 1

Section 4340, states, in part, that the RMES/QAM shall have a nonconformity report prepared whenever a nonconformance is identified.

Contrary to the above,

1. Nonconformity reports were not issued to document, correct, and disposition the six audit findings that were identified and documented as part of the November 1995 Internal Audit Department audit of the HO ES ASME activities.
2. Nonconformity reports were not issued to document, correct, and disposition all of the audit findings identified during the HO ES QAM audits of RMES. The inspectors determined there was only one instance (1996 Atlanta regional office audit) where the RMES used a nonconformity report to document and disposition the audit findings that were identified during the QAM's annual audit of RMES activities. (Nonconformance 99900601/97-01-02)

- C. Sections 1-1, "The Authorized Inspection Agency," Subsection 1-1.2, "Duties," Paragraph 1-1.2.4 of ASME QAI-1-1995, requires, in part, that the agency shall establish and implement an internal program which shall provide assurance that those of its employees holding the positions of supervisor or authorized nuclear inservice inspector (ANII) perform work in accordance with the requirements of Part 1 of this Standard. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout the life of any agreement covering ASME Code Section XI work, in accordance with these policies, procedures, or instructions.

Contrary to the above,

1. The ESM, which documents the requirements necessary to perform ASME Code and engineering service activities, did not include or reference the "Engineering Service Audit Checklist" used by the HO ES QAM to perform annual audits of the RMES activities.
2. The "Engineering Services Audit Checklist," which is used by the HO ES QAM to perform the annual audits of RMES activities, did not include any provisions for reviewing the disposition and corrective actions implemented for findings identified during past audits of RMES activities.

3. No implementing procedure existed to control the internal audit process, and the ESM did not include guidance for conducting quality activities such as documenting internal audits, audit findings, and their closure.
 4. National Board forms NB-71 and NB-178, "Audit Verification Record," are referenced in Sections 4471 and 4475 of the ESM and used by HSB as the method to notify the National Board of completion of required audits, but are not included in Section H, "Forms," of the ESM. (Nonconformance 99900601/97-01-03)
- D. Section 1-2, "The Authorized Nuclear Inspection Supervisor," Subsection 1-2.2, "Duties," Paragraphs 1-2.2.6 & 1-2.2.7 of ASME QAI-1-1995, require, in part, that the ANIIS shall audit the performance of each ANII under his supervision on a planned and periodic basis. Each ANII actively engaged in Section XI Code inspection shall be audited at least twice a year at the site to which he is assigned. The audit shall be recorded in writing and shall contain a written comment regarding the status of each item audited.

Contrary to the above,

1. The HSB ESM did not include provisions that require the RMES to document and adhere to a schedule of two annual audits of each ANII.
 2. The audits conducted by the Atlanta region RMES of the assigned ANII performance did not contain written comments regarding the status of each item audited. Documented objective evidence consisted of a check for either satisfactory, unacceptable, not observed, or not applicable. (Nonconformance 99900601/97-01-04)
- E. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, requires, in part, that activities affecting quality be prescribed by documented instructions or procedures and be accomplished in accordance with these instructions.

Section C-5, "Experience," of Appendix C, "Qualification of Lead Auditors," to the ESM, permits 9 points maximum "experience" to be credited towards lead auditor qualification and states that time spent in various activities will be awarded points on a reasonable basis in line with ANSI N45.2.23 and NQA-1, Appendix 2A-3. This section of the ESM also contains a provision to score one (1) point maximum for each full year's experience classified as "Industry" with other companies if it meets the requirements of Paragraph 2.3.1.2 of ANSI N45.2.23 and Paragraph 2.2 of Appendix 2A-3 of NQA-1.

Section 2.3.1.2 of ANSI N45.2.23 states, "Experience (9 points maximum). Technical experience in engineering, manufacturing, construction, operation, or maintenance, score one (1) credit for each full year with a maximum of five (5) credits for this aspect of experience." Section 2.3.1.2 continues by providing guidance on scoring additional points for specific nuclear, quality assurance, and auditing experience. Similar provisions are contained in NQA-1.

Contrary to the above, two lead auditor candidates were credited the maximum of points (5) for 5 years of work experience towards lead auditor qualification without any objective evidence that the experience provisions contained in Section 2.3.1.2 of ANSI N45.2.23 or Appendix 2A-3 of NQA-1 had been met. (Nonconformance No. 99900601/97-01-05)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk Washington D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventative measures were or will be completed.

Dated at Rockville, Maryland
this 29th day of Aug 1997

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Report No: 99900601/97-01

Organization: Hartford Steam Boiler Inspection and Insurance Company
One State Street
Hartford, CT 06103-3102

Contact: Wilfred C. LaRochelle, Manager
Corporate Quality Assurance

Nuclear Industry
Activity: Authorized Nuclear Inspection Agency

Dates: July 14 - 18, 1997

Inspectors: Uldis Potapovs, Senior Reactor Engineer
Richard P. McIntyre, Senior Reactor Engineer
Larry L. Campbell, Reactor Engineer

Approved by: Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs

Enclosure 2

1 INSPECTION SUMMARY

During this inspection, the NRC inspectors reviewed the implementation of selected portions of Hartford Steam Boiler Inspection and Insurance Company's (HSB) quality assurance (QA) program for providing third party inspection services to NRC licensees. The first part of the inspection was conducted at HSB home offices (HO) in Hartford, CT and included a review of the corporate organization structure, 10 CFR Part 21 implementation program, and HO responsibilities for audits, training and qualification, and nonconformity control. The second part of the inspection was conducted at the HSB regional office in Atlanta, GA and focused on the control and oversight of plant site activities related to the implementation of American Society of Mechanical Engineers (ASME) inspection responsibilities.

The inspection bases were:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50)
- 10 CFR Part 21, "Reporting of Defects and Noncompliance"
- Section III, "Rules for Construction of Nuclear Power Plant Components" of the ASME Boiler and Pressure Vessel Code (Code)
- ASME QAI-1-1995, "Qualifications for Authorized Inspection"

During this inspection, two minor violations of NRC requirements were identified and are discussed in Section 3.2 of this report. The inspection also identified 5 instances where HSB failed to conform to NRC requirements imposed upon them by NRC licensees. These nonconformances are discussed in Sections 3.3.2 and 3.3.3 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of HSB activities performed under an ASME Certificate of Authorization.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Description of Facilities and Activities

HSB is an Authorized Inspection Agency (AIA) for performance of AIA activities controlled from One State Street, Hartford, CT for the following ASME Codes:

ASME Section III, Divisions 1 and 2, ASME Section XI, and ASME Sections I, IV, VIII, Divisions 1 and 2, and X. All inspection services provided to NRC licensees, including contract administration, are conducted through five regional offices. At the present time, HSB has services contracts with 27 nuclear utilities, at 44 plant sites.

3.2 10 CFR Part 21 Program and its Implementation

HSB's procedures for implementing 10 CFR Part 21 regulations are described in Appendix D, "NRC 10 CFR 50 Part 21," of the Engineering Services Manual (ESM). The current revision dates of the ESM and of Appendix D were December 20, 1995, and October 27, 1995, respectively.

Appendix D provided definitions of basic components, defects, and deviations, and required any person involved in nuclear inspection or supervision of such work, who notes a potentially reportable condition, to bring this condition to the attention of a responsible authority at the shop or plant site. The individual observing the potentially reportable condition was also required to document it in his bound diary and provide details to the Regional Manager of Engineering Services (RMES) who, in turn, was required to forward the information to the 2nd Vice President, Engineering Services (2nd VPES). From that point on, HSB's reporting obligations were to be handled by the 2nd VPES. The employee originally reporting the condition was also required to send a written report, describing the corrective action to his supervisor, who was required to forward this information to the 2nd VPES.

Appendix D noted that the provisions of the 10 CFR Part 21 also apply to the company's authorized nuclear inspectors (ANIs), and any defects or deviations in their work which may lead to a substantial safety hazard should be considered reportable. Appendix D also stated that, except in those cases which are clear and evident, it is not the Company's intent to require the inspectors to evaluate when a deviation may result in a substantial safety hazard, implying that all potentially reportable conditions are evaluated by the 2nd VPES. The following concerns were identified as a result the inspector's review of Appendix D.

- (a) Appendix D was based on, and referenced a superseded revision of 10 CFR Part 21 (October 21, 1991). Consequently, certain definitions quoted in the procedure were not consistent with the current revision of the regulation.
- (b) Although Appendix D defined a deviation as a departure from the technical requirements, and required such conditions to be reported to the 2nd VPES, it did not require that nonconformity reports be considered for

potential reportability. Since ESM Chapter 4300, "Control of Nonconformities," defined nonconformance as a deficiency in documentation, procedure, or instruction that renders an activity unacceptable or indeterminate, to comply with the requirements of 10 CFR 21.21, such conditions would need to be evaluated for reportability.

- (c) Although the procedure stated that the 2nd VPES will handle the Company's reporting obligations, it did not specify time limits associated with these obligations (evaluation, initial notification, interim and final reports, etc.)

Based on a record review and discussion with HSB management, the inspectors determined that no potentially reportable conditions had ever been identified and forwarded to the 2nd VPES, and that nonconformity reports, generated as a result of regional office or home office operations, had not been considered as potentially reportable conditions (ESM Chapter 4300 does not require that nonconformances be evaluated for potential 10 CFR Part 21 reportability).

The inspectors also determined that neither the HSB Home Office nor the Atlanta Regional Office had complied with the posting requirements specified in 10 CFR 21.6, "Posting Requirements," which state that each organization subject to the regulations in this Part shall post current copies of the regulation, Section 206 of the Energy Reorganization Act of 1974, and procedures adopted pursuant to the regulations in this Part in a conspicuous position on any premises within the United States where the activities subject to this part are conducted. None of the documents cited above were posted in HSB's offices.

The inspectors advised HSB management that failure to comply with the posting requirements as discussed above and failure to require that nonconformances are evaluated for potential reportability would be identified as violations of 10 CFR Part 21. However, these failures constitute violations of minor significance and are treated as Non-Cited-Violations, consistent with the NRC Enforcement Policy. During the inspection, HSB management revised Appendix D in response to the concerns identified in 3.2(a), (b), and (c), above, and committed to provide the required postings at their Home Office and all regional offices.

3.3 Quality Assurance Program and its Implementation

The program used by HSB to control inspection services provided to NRC licensees is described in their ESM. The ESM commits to providing inspection services consistent with the requirements of the ASME Code and ASME QAI-1 series standards. Although the requirements of 10 CFR Part 50, Appendix B, are invoked through purchase orders (PO) of several licensees, the ESM does

not specifically commit to compliance with this regulation. While a detailed review of programmatic compliance with 10 CFR Part 50, Appendix B, was not made during this inspection, HSB's control of the implementation of selected safety related services was reviewed against the requirements of the ESM as well as the criteria of 10 CFR 50, Appendix B.

3.3.1 ASME Inspection Responsibilities

a. Inspection Scope

The NRC inspectors reviewed ESM Chapter 4500, "ASME Inspection Responsibilities" and the authorized nuclear inservice inspector (ANII) bound diaries, inspection logs, and records of QA monitoring activities for selected nuclear plants to assess the program controls and their implementation.

b. Observations and Findings

Chapter 4500 of the ESM describes HSB policies and responsibilities for regional managers, supervisors, and inspectors involved in ASME inspections. Each RMES is assigned the responsibility for assuring that the inspection services meet all specified standards of performance and quality. The RMES is also responsible for the administration of service contracts with the licensees within his jurisdiction.

The direct supervision of ANIIs and audits of their performance is the responsibility of the authorized nuclear inservice inspector supervisors (ANIIS). Chapter 4500 also describes the duties and responsibilities of various categories of authorized inspectors consistent with the provisions of ASME -QAI-1.

The implementation of ASME inspection activities was evaluated by reviewing relevant documentation (ANII bound diaries, monitoring schedules and reports, qualification records, etc.) for selected nuclear plants and, in some cases, discussing specific issues with the assigned ANII by telephone. The NRC inspectors reviewed the ANII bound diaries for Comanche Peak, covering the period from April 10, 1995, to March 21, 1997; Browns Ferry, Unit 3 (November 22, 1995, to April 21, 1997); and Hatch (August 16, 1995 to October 21, 1995). Also reviewed were selected monitoring schedules and reports, records of inspection verifications, and identification and disposition of nonconforming conditions.

The ANII bound diaries, in general, were found to contain appropriate entries, consistent with the applicable ASME QAI-1 requirements. Records of monitoring activities indicated that these activities were performed in accordance with the schedules established by the ANIIs, using supplementary checklists (developed by the ANII), to identify specific program areas to be monitored and to provide for recording of objective evidence to support their findings for each area monitored. Program deficiencies identified as a result of the monitoring activity were recorded in the ANII's bound diary and documented on HSB Form 939, "ES Record for Monitoring QA/QC Programs." Form 939 is used to identify specific program areas reviewed, and, in cases of identified deficiencies, to request a response from the NRC licensee. Review of several monitoring records indicated that these activities were properly documented, and that identified concerns and their resolution were documented in the ANII's diary.

A general observation was that activities described above were performed and documented using different methods at the selected plant sites reviewed, apparently because there were no standard implementing procedures available to perform several of these activities. Similarly, it was also noted that tracking and resolution of identified concerns was being addressed by different methods. In some cases, HSB Form 939, "ES Record for Monitoring QA/QC Programs" was used for all identified concerns, while, at another plant site, the ANII had apparently developed and was using an "ASME XI Discrepancy Notice" for tracking and dispositioning of isolated (non-programmatic) deficiencies. These forms were being issued to the licensee and dispositioned after achieving resolution of the issue.

c. Conclusions

Review of the records of ASME activities performed at selected plant sites indicated that these activities were conducted and documented in accordance with the applicable ASME QAI-1 requirements and that the ANII records of these activities were generally well documented and complete. One observation in this area was discussed with HSB management as a program weakness. This related to the lack of controlled procedures for performing and documenting monitoring activities and dispositioning of isolated (non-programmatic) deficiencies.

3.3.2 Audits

a. Scope

The inspectors reviewed Chapter 4400, "Audits," of the ESM, which described the audit and survey requirements for internal audits, external audits, pre-review/

survey audits, and ASME Code required audits, and also described the accompanying requirements for audit report documentation. The following Sections of Chapter 4400 of the ESM were reviewed:

- Section 4420, "Internal Audits"
- Section 4430, "External Audits"
- Section 4470, "Nuclear Audits"

b. Observations and Findings

b.1 Internal Audits - Home Office Engineering Services

The Internal Audit Department (IAD) is responsible for conducting annual audits of HO Engineering Services (ES) department's ASME activities. The Quality Assurance Manager (QAM) is responsible for conducting annual audits of the various RMES activities to verify compliance with the ESM, supporting procedures, instructions, and the ASME QAI-1 standards, as applicable.

The inspectors reviewed implementation of the above internal audit processes for compliance to the ESM. While attempting to review IAD's audits of HO ES activities, the inspectors determined that only one audit (November 1995) had been performed and documented by IAD and was available for review. This IAD audit identified six findings and recommendations for compliance to the ESM. The inspectors were told by the QAM that IAD no longer conducts the audits of HO ES activities even though ESM Section 4421 still requires this activity. The failure to audit HO ES activities as required by the ESM was identified as an example of Nonconformance 99900601/97-01-01.

The inspectors also identified that nonconformance reports had not been written by the QAM to document, disposition, and correct the findings that were identified during the IAD audit conducted in November 1995. Section 4320 of Chapter 4300 of the ESM, "Control of Nonconformities," defines a nonconformance as a deficiency in documentation, procedure or instruction that renders an activity unacceptable or indeterminate. Section 4340 states that the RMES/QAM shall prepare a nonconformity whenever a nonconformance is identified. The failure to issue Nonconformance Reports for documented failures to implement the requirements of the ESM was identified as Nonconformance 99900601/97-01-02.

b.2 Internal Audits - Regional Manager, Engineering Services

All internal audits of the RMES activities are conducted by the QAM. Currently there are four regional offices that conduct ASME Section III and Section XI

nuclear inspection activities. The inspectors reviewed the schedule for audits of RMES activities for the last four years to select a sample of audit reports to review. When reviewing the regional office audit schedules, the inspectors noted that many of the scheduled audits had slipped beyond their scheduled date, including the 1996 audit of the San Francisco RMES and the 1995 audit of the Atlanta RMES, which were never performed. While attempting to review a sample of audit reports from each region, the inspectors were told that several of the documented audit reports and the RMES responses to audit report findings requested could not be located in the HSB HO or Regional files for Atlanta, Northeast/Philadelphia, and San Francisco in the 1993 to 1995 time frame.

During the inspection the QAM committed to issue a Nonconformance Report to address this issue. The failure to audit all RMES activities on an annual basis as required by the ESM was identified as another example of Nonconformance 99900601/97-01-01.

The audit reports reviewed by the inspectors documented what appeared to be a thorough review of the RMES activities and included pertinent findings against program implementation, when applicable. However, as was the case with the audit findings identified during the IAD audit of the Home Office Engineering Services ASME activities, no nonconformance reports or appropriate tracking and completion of corrective actions were documented by either the QAM or by the RMES. The inspectors determined there was only one instance (1996 Atlanta regional office audit) where the RMES documented and dispositioned the audit findings on nonconformance reports. This issue was identified as another example of Nonconformance 99900601/97-01-02.

When reviewing the audit reports that were available the inspectors determined that the Engineering Services Audit Checklist used by the QAM for RMES audits was not an approved and controlled quality document and was not referenced in ESM Section 4420 or in Appendix H, "Forms." During the inspection HSB committed to approve and control the checklist under the ESM program requirements. This issue was identified as an example of Nonconformance 99900601/97-01-03.

The inspectors also noted that Section 4420 of the ESM does not include in-depth guidance and detail for conducting and documenting internal audits and accompanying audit findings and that there was limited documented QA program requirements for accomplishing this quality activity. No implementing procedure existed that addressed the internal audit process. This issue was identified as another example of Nonconformance 99900601/97-01-03.

The inspectors also determined that none of the audit reports reviewed contained documentation of any follow-up review for corrective actions and disposition to previous audit findings, and that neither the ESM nor the checklist used for the QAM audits of the RMES included provisions that required evaluation and review of the closure of audit findings identified during previous audits. During the inspection and in follow-up documentation submitted to the NRC on July 25, 1997, HSB committed to implement Regional Office Audit Procedure, ES QP 03, Revision 0, to address these concerns.

b.3 ASME Nuclear Audits

Section 4470 of the ESM documents HSB requirements for the conduct of audits to comply with ASME QAI-1, 1995, ASME Section III, Division 1, and ASME Section XI. This included ANI and ANII audits of nuclear Section III and Section XI work activities, and ANI and ANII performance audits conducted by the authorized nuclear inspector and inservice inspector supervisors (ANIS and ANIIS). Chapter 4500 of the ESM, "ASME Inspection Responsibilities," documents the inspection responsibilities (including audits) for the RMES, ANIS, and the ANIIS.

The inspectors reviewed the current ANI "Nuclear Shop Assignments" and "National ANII Assignments" listing for ASME Section III shops in the Atlanta region and Section XI plant sites (including Atlanta region). The ASME Section III shop and Section XI site performance audits were reviewed for ASME Code compliance and for the implementation of the various sections of the shop's and site's applicable QA manuals. The nuclear shop ANI and ANII performance audits are required to be performed twice a year and are conducted using the ES Inservice Report checklist (Form 2163). National Board forms NB-71 and NB-178, "Audit Verification Record," forms are referenced in Sections 4471 and 4475 of the ESM and are used by HSB as the method to notify the National Board of completion of these required audits of ANI and the ANII by the ANIS and the ANIIS. These forms are not included in Section H, "Forms," of the ESM. This issue was identified as another example of Nonconformance 99900601/7-01-03.

The inspectors reviewed a sample of semiannual inspector's performance audits conducted by the ANIS at Section III nuclear shops and by the ANIIS at Section XI sites. The audit reports included the appropriate review and documentation as required by the HSB ESM with the exception of an issue pertaining to audit requirements included in Sections 0-2.2.7 and 1-2.2.7 of QAI-1, 1995. These paragraphs require the audits to be recorded in writing and to contain a written comment regarding the status of each item audited. The audits reviewed by the inspectors were accomplished using the Form 2163 checklist described above,

but did not contain a written comment regarding the status of each item audited or any real documented objective evidence of what was reviewed. This issue was identified as Nonconformance 99900601/97-01-04.

During the above review, the inspectors also identified that the ESM does not include any requirement for the ANIS and the ANIIS to document and implement an audit schedule for their applicable sites on an annual basis. This issue is identified as another example of Nonconformance 99900601/97-01-04.

c. Conclusions

Based on the above, the inspectors concluded that Hartford had failed to implement an adequate audit program.

3.3.3 Qualification of Lead Auditors

a. Scope

The NRC inspectors reviewed five HSB lead auditor qualification files to determine if the provisions contained in ESM Appendix C, "Qualification of Lead Auditors" were being met.

b. Observations and Findings

HSB uses lead auditors, qualified in accordance with ESM Appendix C, to perform internal audits of the implementation of its quality assurance program. HSB also uses these lead auditors to perform safety-related, non-ASME Code activities, such as independent reviews, assessments, and audits.

The lead auditor qualification provisions contained in Appendix C of the ESM are based primarily on the adoption of ANSI N45.2.23-1978, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants," and NQA-1-1989, "Quality Assurance Program Requirements for Nuclear Power Plants."

The inspector's review revealed that Appendix C contained a provision that was not fully consistent with Section 3.2, "Maintenance of Proficiency," of ANSI N45.2.23. ANSI N45.2.23 permits lead auditors to maintain their proficiency by several methods including the review and study of codes, standards, procedures, instructions, and other documents related to quality assurance programs and auditing. However, Section C-13, "Annual Recertification," of Appendix C to the ESM states, in part, that in lieu of audit participation, the lead auditor may demonstrate that his proficiency has been maintained by having actively participated as a member in an ASME Code Committee that reports to

the Board on Nuclear Codes and Standards. The NRC inspectors and HSB discussed the fact that even though a lead auditor participates on a committee that reports to the Board on Nuclear Codes and Standards, this does not ensure that the lead auditor's participation in the committee process requires a review and study of quality assurance programs and the auditing process. It was also discussed that depending on the committee or subcommittee, the lead auditor may not be involved in committee activities associated with the auditing process. The NRC inspectors considered the alternative to audit participation (having participated on a committee reporting to the Board on Nuclear Codes and Standards) to be a potential area of deviation from ANSI N45.2.23 and a weakness in the HSB lead auditor qualification program.

The NRC inspectors' review of the five HSB lead auditor qualification files identified that 3 of the qualification packages did not appear to meet the provisions of Appendix C. Specifically, Section C-3, "Qualification of Lead Auditors," requires that the lead auditor shall have verifiable evidence that a minimum of 10 points under the score system provided in Appendix C have been accumulated. The maximum number of points permitted for various categories are: a) Education, 4 points, b) Experience, 9 points, c) Professional Accomplishments, 2 points, and d) Management, 2 points.

Section C-5, "Experience," of Appendix C to the ESM, permits 9 points maximum for related experience and states that time spent in various activities will be awarded points on a reasonable basis in line with ANSI N45.2.23 and NQA-1, Appendix 2A-3. Further, Section C-5 contains a provision to score one (1) point maximum for each full year's experience with other companies in other capacities classified as "Industry" if it meets the requirements of Paragraph 2.3.1.2 of ANSI N45.2.23 and Paragraph 2.2 of Appendix 2A-3 of NQA-1. Section 2.3.1.2 of ANSI N45.2.23 states:

Experience (9 points Maximum). Technical experience in engineering, manufacturing, construction, operation, or maintenance, score one (1) credit for each full year with a maximum of five (5) credits for this aspect of experience.

Section 2.3.1.2 continues by providing guidance on scoring additional points for specific nuclear, quality assurance, and auditing experience.

The review indicated that HSB had credited each of three lead auditors with 5 points for work experience, based on two of the lead auditors having 5 years experience in the US Navy and one having 5 years experience with another company. The NRC inspectors questioned whether any of the 5 years experience was technical, or was in the nuclear or in the quality assurance or

auditing disciplines. HSB presented a resume of one of the three lead auditors, HSB Identification Number 02562, which indicated for the period 1972-1977, the individual had 4 years experience as a radiographer and welder and 1 year experience as a quality control manager. No records or resumes were presented for the other lead auditors (Identification Numbers 02026 and 02664).

c. Conclusions

The NRC inspectors determined that the lead auditor files for HSB employees Identification Numbers 02026 and 02664, initially qualified as lead auditors on June 28, 1991, and January 31, 1996, respectively, were inadequately processed by crediting the maximum of points for work experience without any objective evidence that the provisions contained in Section 2.3.1.2 of ANSI N45.2.23 or Appendix 2A-3 of NQA-1 had been met. This issue was identified as Nonconformance 99900601/97-01-05.

3.4 Service Contract Provisions and Applicability of 10 CFR Part 21 to HSB Activities

a. Scope

The NRC inspectors reviewed several licensee purchase orders (POs) issued to HSB for Authorized Inspection Agency inspection services to assess licensee control of subcontracted inspection activities and the extent to which applicable NRC requirements are passed down to the providers of these activities.

b. Observations and Findings

The review identified that certain NRC requirements were not consistently invoked by licensees using HSB to perform third party inspections required by the ASME Section XI Code. Specifically:

- (a) Virginia Power PO BKI 483582, dated November 11, 1995, stated that the HSB inspection services were nuclear safety related and required HSB to implement quality control and quality assurance programs that comply with the requirements of Appendix B to 10 CFR Part 50 and ANSI N45.2. The Virginia Power PO also invoked the requirements of 10 CFR Part 21 for this contract.
- (b) Section 26, "Quality Assurance Requirements" of Carolina Power & Light Company (CP&L) Document UFI No. PTC00004, "Contract No. XT00000026 between Carolina Power & Light Company and the Hartford Steam Boiler Inspection and Insurance Co.," hand dated April 5, 1990,

stated that the work to be performed by HSB had been determined to be non-nuclear safety related. In Section 27, "10CFR21," the CP&L contract stated that the provisions of Part 21 shall apply to any work within the definition of basic component in 10 CFR 21.3 and that CP&L shall be promptly notified of any reports made to the NRC pursuant to 10 CFR 21.21. However, Section 27 of the CP&L contract continued by stating that CP&L recognizes that the Contractor will not be required to perform nuclear safety related work.

- (c) Georgia Power Co. PO 60120550000 (latest revision dated May 21, 1997) did not invoke the requirements of either 10 CFR Part 50 or 10 CFR Part 21, but required the labor and supervisory personnel to meet the requirements of ANSI N18.1, and the AIA to meet the requirements of ASME 626.1-1982.
- (d) Duke Power Co. PO MN 12553, dated March 12, 1996 (for McGuire Nuclear Station) stated that the services to be supplied are safety related and that they shall be supplied in accordance with the suppliers quality assurance program, approved by Duke Power Co. It also stated that, if lower tier procurement is required, the applicable QA requirements must also be invoked on the lower tier subcontractors/suppliers.

The NRC inspectors and HSB discussed the applicability of both Appendix B to 10 CFR Part 50 and 10 CFR Part 21 to the inspection services provided by HSB at nuclear power plants. During these discussions the NRC inspectors identified that Section 21.3(1)(ii)(3) of 10 CFR Part 21 states: "In all cases, basic component includes safety related design, analysis, inspection, testing, fabrication, replacement of parts, or consulting services that are associated with the component hardware whether these services are performed by the component supplier or by others." It was also discussed that Section 50.55a, "Codes and Standards," of 10 CFR Part 50 identifies the applicable codes for the design, fabrication, erection, construction, testing, and inspection of systems, structures, and components and mandates the use of certain editions and addenda of the ASME Section XI Code.

It was further discussed that because Section IWA 2110, "Duties of the Inspector," of Article IWA 2000, "Examination and Testing," of the ASME Section XI Code identifies the duties of the Authorized Inspection Agency's inspector (the Inspector) assigned to perform IWA 2110 activities and these activities are mandatory by the ASME Section XI Code in order to assure compliance to the code, they are considered to be safety-related activities. It was further

discussed that because the Inspector's performance of certain verifications and reviews is a safety-related activity, the services of the Inspector are considered a basic component.

The inspectors also determined, from discussions with HSB management, that no licensee audits of HSB ASME authorized inspection activity implementation had ever been performed at any of the HSB offices.

c. Conclusions

The inspectors determined that major inconsistencies existed in licensee safety classification of the services provided by HSB and in the imposition of applicable NRC requirements (10 CFR Part 21 and Appendix B to 10 CFR Part 50) for the performance of these services. The inspectors also determined that licensees were not performing QA program implementation audits on HSB as a provider of safety-related services.

3.5 Entrance and exit meetings

In the Entrance Meeting on July 14, 1997, the NRC inspectors discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with HSB management. In the exit meeting on July 18, 1997, the inspectors discussed their findings and concerns.

PARTIAL LIST OF PERSONS CONTACTED

Wilfred LaRochelle, Corporate Quality Assurance Manager
Barry Bobo, National Manager, Engineering Services
Sidney Montgomery, Regional Supervisor, Engineering Services
Harold Robinson, National Safety and Training Manager
Lashanta Lewis, Engineering Support Assistant

ITEMS OPENED

Opened

99900601/97-01-01	NON	Failure to perform required audits
99900601/97-01-02	NON	Failure to issue nonconformity reports
99900601/97-01-03	NON	Inadequate procedures
99900601/97-01-04	NON	Inadequate control of audits
99900601/97-01-05	NON	Inadequate documentation of auditor qualification

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

September 26, 1997

Dama Wirries, President
NUS Instruments, Inc.
440 W. Broadway
Idaho Falls, ID 83402

SUBJECT: NRC INSPECTION REPORT 99901320/97-01 AND NOTICE OF
NONCONFORMANCE

Dear Ms. Wirries:

This letter addresses the inspection of your facility at Idaho Falls, Idaho, conducted by Bill Rogers and Robert Pettis, of this office on August 19-21, 1997, and the discussions of their findings with John McGimpsey, Cheryl Allen, and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector.

During this inspection it was found that the implementation of your Quality Assurance (QA) program failed to meet certain NRC requirements. It was determined that NUS Instruments, Inc. (NUS) did not adequately document the corrective actions taken to prevent recurrence of the inadequate soldering identified in modules shipped to Public Service Gas & Electric in the July 1995 to July 1996 time period. In addition, NUS did not take corrective actions for an extended period of time following the identification of less than acceptable work quality in the area of Quality Control inspection. The specific findings and references to the pertinent requirements are identified in the enclosures of this letter.

Please provide us within 30 days from the date of this letter a written statement in accordance with the instructions specified in the enclosed Notice of Nonconformance. We will consider extending the response time if you can show good cause for us to do so.

The responses requested by this letter and the enclosed notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Ms. Wirries

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Sincerely,

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No. 99901320

Enclosures: 1. Notice of Nonconformance
 2. Inspection Report 99901320/97-01

NOTICE OF NONCONFORMANCE

NUS Instruments, Inc.
Idaho Falls, Idaho

Docket No. 99901320

Based on the results of an inspection conducted on August 19 through 21, 1997, it appears that certain of your activities were not conducted in accordance with NRC requirements.

- A. Criterion XVI, "Corrective Action," of Appendix B to 10 CFR Part 50, requires, in part, that for significant conditions adverse to quality, measures will be established to assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

Section 16 of the NUS Instruments, Inc., (NUS) Quality Assurance manual, "Corrective Action," Fifth Issue, Revision 0, dated September, 1994, requires the prompt identification, documentation, and correction of conditions adverse to quality and, in the case of significant conditions adverse to quality, documentation of corrective actions to preclude recurrence.

Contrary to the above, (1) NUS did not adequately document the corrective actions taken in response to the identified occurrences of inadequate soldering on modules manufactured and provided to Public Service Gas & Electric in the July 1995 to July 1996 time period, and (2) NUS did not take prompt corrective action following the identification of less than acceptable work quality in the area of Quality Control inspection in July of 1996, for significant conditions adverse to quality.
(Nonconformance 99901320/97-01-01)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Quality Assurance, Vendor Inspection, and Maintenance Branch, Division of Reactor Controls and Human Factors, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each Nonconformance: (1) the reason for the nonconformance, or if contested, the basis for disputing the nonconformance, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further noncompliances, and (4) the date when your corrective action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 26th day of September 1997

Enclosure 1

U.S. NUCLEAR REGULATORY COMMISSION

Docket No: 99901320

Report No: 97-01

Facility: NUS Instruments, Inc.

Location: 440 W. Broadway
Idaho Falls, Idaho 83402

Dates: August 19 - 21, 1997

Inspectors: B. Rogers, Team Leader
R. Pettis, Senior Reactor Engineer

Approved by: G. Cwalina, Section Chief
Vendor Inspection Section
Special Inspection Branch

Enclosure 2

1 Inspection Summary

1.1 Background and Basis

NUS Instruments, Inc., (NUS) provides engineering and manufacturing capabilities for the replication, refurbishment, and redesign of electronic circuit card assemblies and modules which are provided to NRC licensees as safety-related equipment and services. NUS is owned by Sciencetech, Inc. (Sciencetech) which acquired NUS in October of 1996.

The basis for the NRC inspection of NUS included:

- 10 CFR Part 50, Appendix B
- 10 CFR Part 21
- American National Standards Institute/American Society of Mechanical Engineers Standard N45.2.6-1978, "Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants" as committed to in the NUS Quality Assurance Manual

1.2 Nonconformances

Nonconformance 99901320/97-01-01 was identified and is discussed in Sections 3.1 and 3.3 of this report

2 Status of Previous Inspection Findings

No previous inspections have been conducted at this facility.

3 Inspection Findings and Other Comments

3.1 10 CFR Part 21 Program and Corrective Actions

a. Scope

The inspectors reviewed the NUS 10 CFR Part 21 implementing procedure, 10 CFR Part 21 Posting, and the NUS records related to 10 CFR Part 21 evaluations performed by NUS or performed by customers and subsequently provided to NUS. The purpose of the review was to verify that NUS was meeting the requirements of 10 CFR Part 21 in the applicable activities.

b. Observations and Findings

On August 8, 1996, Public Service Electric & Gas Co. (PSE&G) notified the NRC that numerous NUS Model OCA801 Signal Isolator Modules and Model MTH801 Signal Summator Modules, used in the Salem Nuclear Plant, Reactor Control and Protection System (RPS), were found to have unsoldered or insufficiently soldered internal electrical connections.

The documentation provided by PSE&G indicated that the isolator module function is to provide electrical separation between the Reactor Protection System (Class 1E) and the Process Control System (non-Class 1E) portions of the RPS instrument loops, and the summator module function is to algebraically combine analog signals. The reactor protection system is designed to assure that the system can perform its required functions in the event of a design basis earthquake. The defect could affect the modules' ability to function during a design basis earthquake and could result in a loss of redundancy sufficient to constitute a major degradation of essential safety-related equipment.

NUS documentation indicated that two reviews of the PSE&G modules had been performed: (1) "Corrective Action/Root Cause Evaluation," which was completed in September of 1996 by the NUS Quality Assurance Supervisor and (2) "Quality Assurance Review of NUS Instruments," which was completed in October 1996 by a three person review team contracted by Sciencetech (NUS's parent company).

NUS Review

The NUS review listed the root cause of the inadequate soldering to be an organizational breakdown caused by inadequate prioritization of work and inadequate job skills, work practice, and decision making. Contributing causes included inadequate interface between organizations, inadequate supervisory program monitoring, inadequate self-verification practices, and differences in the soldering workmanship criteria of NUS and PSE&G. Corrective actions were specified to include (1) removal of the dual assignment of the manufacturing lead person, (2) increasing manufacturing supervision, (3) hiring an additional QC Inspector, and (4) providing supervisory skills training to supervisors. The performance of the NUS review had occurred just prior to the Sciencetech purchase of NUS (October 1996). Sciencetech and NUS management indicated in a telephone conversation with the NRC inspection team on August 26, 1997, subsequent to the completion of the inspection, that they considered the NUS review's root cause investigation to be generally accurate but that the conclusions and corrective actions had not been specific enough to be useful to Sciencetech management to prevent recurrence of the situation.

In addition, the NUS review indicated that manufacturing operating sheets (which specified the manufacturing and inspection steps) were not being followed and that

partial inspections were being performed in accordance with an agreement, between the manufacturing and inspection staff, to modify the operation process sheet flow. The Manufacturing Supervisor stated that there had been some indication during the period of the PSE&G module production (approximately July 1995 to July 1996) that there had been some modification of activities occurring at the staff level but that reorganization of the work activities and reassignment of staff had addressed any potential concerns in this area. Other NUS personnel, assigned to NUS management positions during this time period, did not have indication that modification of activities had occurred at the staff level. Sciencetech management stated that they had reviewed documentation and observed activities and did not agree with this conclusion of the NUS review. The inspectors noted that modification of the operation process sheet flow, to alter the specified inspections, would be, as defined by the NUS Quality Assurance Manual (QAM), a Level II inspection function and that the QC Inspector performing the work at the time was certified to Level I.

The inspectors reviewed applicable work records and interviewed numerous personnel and determined that there was documentation that the work had been performed in accordance with the appropriate procedures. Further, NUS had taken several corrective actions which would affect this area to prevent any potential for reoccurrence on modification of activities that might have occurred at the staff level.

Sciencetech Review

The Sciencetech review had some overlap with the conclusions and recommendation of the NUS review but was more detailed and specific. The contributing factors were determined to be related to organizational structure, training, attitude and awareness, fabrication overview, personnel qualification, and work station adequacy. The recommendations made in the Sciencetech review included evaluation of client action requests, evaluation of work station ergonomics, increasing the frequency of trend analysis, improving the coordination of training activities, refining the wave soldering process, and improving supervisory skills. Discussion with Sciencetech and NUS management indicated that there was agreement with the contents of the Sciencetech review and that the Sciencetech review (as opposed to the earlier NUS review) was the basis for the majority of the corrective actions taken by NUS to address the PSE&G soldering issue.

Immediate Corrective Actions

The inspectors reviewed the corrective actions taken by NUS, following the PSE&G notification, to address the inadequate soldering of the PSE&G Modules. The NUS Manufacturing Supervisor indicated that when PSE&G notified NUS in July of 1996 of the inadequate soldering, NUS had reviewed the manufacturing and inspection process and had taken numerous, immediate corrective actions.

The Manufacturing Supervisor had reviewed the soldering process and had determined that the majority of the manufacturing line (referred to as "assemblers" by NUS) were soldering at the "preferred" level while two assemblers were soldering at the "acceptable" level (a lesser quality level than "preferred"). While NUS had determined that the "acceptable" level was adequate for functioning electrical connections it did not meet the customer's (PSE&G) expectation. All of the assemblers were retrained in the soldering technique by being provided physical examples of "preferred" soldering, studying the applicable procedures, and undergoing a supervisory assessment of each person's soldering ability. In addition, the Manufacturing Supervisor assumed the manufacturing lead person's responsibilities (that person was reassigned), which allowed the supervisor to directly observe the manufacturing work being performed. Of the two assemblers who were soldering at the "acceptable" level, one person was adequately retrained and the other was removed from the manufacturing activities and within one month of the PSE&G notification the entire manufacturing line was soldering at the "preferred" level. The inspectors noted that the requirement to solder at the "preferred" level of quality had not been proceduralized by NUS.

During the period of July 1995 to July 1996, all of the NUS assemblers had performed work on the PSE&G modules. The two assemblers previously identified as soldering at the "acceptable" level had not performed any work on projects other than the PSE&G modules manufactured during that period. Work for customers other than PSE&G had been performed by two other assemblers, both of whom the Manufacturing Supervisor had considered excellent performers. NUS reviewed the work performed in the July 1995 to July 1996 time period, and had concluded that there was not a concern with work performed for customers other than PSE&G.

The inspectors noted that NUS had not documented the inadequate soldering discovered in the PSE&G modules in accordance with the requirements of the NUS QA program and had not documented the immediate corrective actions previously discussed.

Additional Corrective Actions

In addition to the immediate corrective actions taken by NUS to raise the level of soldering quality, NUS had taken additional corrective actions, based on the Sciencetech review, which included reorganizing the manufacturing line, installing a mechanical wave soldering machine, training the manufacturing supervisory and lead personnel, and upgrading the QC Inspector position to Level II. NUS also indicated that ongoing training of assembly and inspection personnel would occur, peer reviews had been instituted for manufacturing personnel, and deficiency logs had been established to track soldering rework, although these corrective actions had not yet been proceduralized at the time of the inspection.

c. Conclusions

The inspectors noted the indication that modification of the operation process sheets, to alter specified inspections, had occurred in the July 1995 to July 1996 period, that such a modification would have been a Level II inspection function, and that the QC Inspector performing the work at the time was certified to Level I. Indication of this modification highlighted a potential weakness in the NUS Quality Assurance program.

The inspectors concluded that NUS had taken reasonable corrective actions in response to the identification of inadequate soldering in the Salem modules. However, NUS had not adequately documented the corrective actions which NUS indicated were taken upon identification of the inadequate soldering. This was identified as an example of Nonconformance 99901320/97-01-01.

3.2 Manufacturing Procedure and Implementation of Soldering Technique

a. Scope

The inspectors reviewed the applicable procedures, discussed activities with manufacturing and inspection personnel, and observed ongoing inspection and manufacturing activities, to assess whether soldering activities were being adequately controlled.

b. Observations and Findings

The inspectors reviewed the training records for several assemblers to determine their qualifications to perform soldering. Training was documented on a "Certification of Training" which indicated that assembler was qualified to solder in accordance with the "Beckwith Training Course E- 5" and in accordance with the NUS Operating Procedures Manual, Appendix E, "Soldering," Revision 0, dated March 12, 1992. The certificate documented that the assembler had received the required instruction, had passed a written examination, and completed a practical demonstration of soldering ability.

The inspectors observed assemblers installing components, wrapping wire, and performing hand soldering. The inspectors also observed a demonstration by the QC Inspector on performance of an inspection of solder connections and discussed the levels of classification of soldering quality. The level of solder quality required for NUS product is the "preferred" level which is the premium level of solder quality. NUS indicated that the "preferred" level of solder quality was mandated and strictly adhered to as a corrective action to the PSE&G Part 21 report (See section 3.1.2).

c. Conclusions

The inspectors concluded that NUS had in place a program to train manufacturing personnel to solder at the required level of soldering quality and to verify the manufacturing personnel's ability and that this program was being adequately implemented. In addition, the QC Inspector possessed the ability to verify the adequacy of the soldered connections.

3.3 Review of Qualifications for QC Inspectors

a. Scope

The inspectors reviewed QC Inspector qualification records and selected documentation to determine compliance with American National Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) N45.2.6-1978, "Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants" as committed to in the NUS QAM. The requirements included in the ANSI Standard are for the qualification of personnel who perform inspection, examination, and testing of nuclear power plant items used in safety-related applications.

b. Observations and Findings

NUS required that the capabilities of a candidate for certification as a Level I, II, or III QC Inspector be initially determined by a suitable evaluation of the candidate's education, experience, training, test results, or capability demonstration. Once certified, the QC Inspector's job performance was reevaluated at periodic intervals not to exceed three years.

The NRC inspectors reviewed the training files of all QC Inspectors to verify that the above requirements were met. A total of seven QC Inspector files were reviewed and all appeared to be in compliance with the requirements for certification. One QC Inspector was certified to Level I, three QC Inspectors were certified to Level II, and three were certified to Level III. However, as of the inspection, only one Level II QC Inspector remained active since the others had either been reassigned to other positions or were no longer employed at NUS. The present NUS QA Manager, certified to Level III, supervised the work of the Level II QC Inspector. A review of the files indicated that the QC Inspectors were certified primarily on the basis of education and related experience and that the qualification records were in compliance with ANSI Standard N45.2-6.

The inspectors reviewed the work being performed by the QC Inspector to determine whether the work met the definition of Level I work as defined by ANSI standard N45.2-6 which states, in part, that Level I persons shall be capable of performing inspections, examinations, and tests in accordance with documented procedures. Level II persons

shall have, in addition to the Level I capabilities, demonstrated capabilities in planning, setting up, and supervising inspections and tests. The inspectors reviewed documentation for several POs which indicated that the QC Inspector had performed activities consistent with the Level I definition such as verifying component placement, verification of soldering, and final visual inspections. However, the inspectors cautioned NUS management that allowing manufacturing and inspection staff to modify the operation process sheet flow without management approval, as discussed in detail in Section 3.1.2, would be a Level II inspection function and inappropriate if the QC Inspector performing the work at the time was certified to Level I or if such operation process modifications were disallowed by the NUS QA program.

During review of the training files to verify QC Inspector qualifications, two documents were identified which indicated a quality control inspection activity weakness. Two memoranda from the QA Supervisor identified weaknesses in the QC Inspector's performance and one recommended management action. These memoranda, initiated in January and July of 1996, and were currently in the QC Inspector's training file. Discussion with the current NUS and Scientech management indicated agreement that the QC Inspector's performance was not adequate and that the QC Inspector had been subsequently reassigned shortly after the current management was made aware of the QC Inspector's performance in December of 1996. However, during the period of time from July 1996 until December 1996, NUS had been made aware of the inadequate performance of the QC Inspector and had not taken any corrective action. Subsequent to the new management being placed, NUS had reassigned the QC Inspector, and had employed two additional persons in that position. The inspectors reviewed the training files of the two most recent QC Inspectors, discussed their performance with the QA Supervisor and the Manufacturing Supervisor, reviewed work documentation, and observed work performance of the current QC Inspector. The inspectors did not observe any indication of inadequate performance in the documentation or work observations of the current QC Inspector.

c. Conclusion

Although NUS had been made aware of the potentially inadequate work performance of a QC Inspector by the inspector's direct supervisor in January and July of 1996, and that this information was available in the QC Inspector's training file, NUS had not taken any corrective action for this situation adverse to quality until December of 1996. The failure to take corrective action to correct a significant condition adverse to quality was identified as an example of nonconformance 99901329/97-01-01.

4 Personnel Contacted

Dama Wirries, President, NUS
John McGimpsey, General Manager, NUS
Cheryl Allen, QA Supervisor, NUS
Shauna Boyack, Manufacturing Supervisor, NUS
Heath Buckland, Testing Supervisor, NUS
Ron Todd, Quality Control Lead, NUS
Paul Sturm, Principal Engineer, Sciencetech, Inc.

**UNITED STATES
NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

September 9, 1997

Mr. Gregory M. Rueger
Pacific Gas & Electric Company
NPG-Mail Code A10D
PO Box 770000
San Francisco, CA 94177

Dear Mr. Rueger:

SUBJECT: NRC INSPECTION OF THE DIABLO CANYON POWER PLANT (REPORT
NOS. 50-275/97201 AND 50-323/97201)

During the period June 9-12, 1997, the Special Inspection Branch of the U.S. Nuclear Regulatory Commission's (NRC's) Office of Nuclear Reactor Regulation (NRR) performed an inspection of Pacific Gas & Electric Company's (PG&E's) activities related to the procurement, modification, testing and installation of replacement 4-kV circuit breakers at the Diablo Canyon Power Plant (DCPP), Units 1 and 2.

The primary purpose of the inspection was to determine if design verification testing of the modified breaker was accomplished in accordance with applicable requirements of American National Standards Institute (ANSI)/Institute of Electrical and Electronic Engineers (IEEE) Standards. A secondary purpose of the inspection was to examine several related issues involving production testing, modifications, and post installation testing in-service failures. The results of this inspection are contained in the enclosed inspection report.

With respect to design verification, the inspectors determined that PG&E's approach was generally consistent with the applicable NRC regulations in that PG&E undertook to verify, by engineering analysis and a testing program, that DCPP's safety-related 4-kV electrical distribution system as converted and modified would perform its safety functions under all design basis conditions. By NRC letter dated July 24, 1997 (Attachment 1 to the enclosed inspection report), the question of consistency of the approach with applicable industry standards, in particular, taking credit for certain design verification tests done by the manufacturer of the circuit breakers used in the conversion, was referred to the Standards Board of the IEEE for consideration by the appropriate subcommittee. The details of the IEEE response, contained in an August 21, 1997, letter from the IEEE Power Engineering Society Switchgear Committee (Attachment 2 to the enclosed report), are discussed in the report, but in summary, the PG&E approach is considered consistent with the intent of the standards. Although there is no regulatory guide endorsing the standards in question, PG&E has committed to them in the Final Safety Analysis Report (FSAR) for DCPP, and they are therefore relevant to the design and licensing basis of the plant. With regard to the engineering evaluations necessary to support the PG&E approach, the inspectors could not, on the basis of documentation available for review at DCPP, conclusively determine the adequacy of all the justifications for not reperforming certain design tests.

Mr. Gregory M. Rueger

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Therefore, the inspectors identified the need for further review at National Technical Systems (NTS), Inc., Acton, Massachusetts; and possibly also Power Distribution Services (PDS), Inc., Cincinnati, Ohio.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and inspection report will be placed in the NRC Public Document Room. Should you have any questions concerning the attached inspection report, please contact the inspection team leader, Mr. Stephen Alexander, at (301) 415-2995.

Sincerely,

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

cc: See next page
Docket Nos. 50-275 and 50-323
License Nos.: DPR-80 and DPR-82
Enclosure: Inspection Report No. 50-275,323/97201

cc:

NRC Resident Inspector
Diablo Canyon Nuclear Power Plant
c/o U.S. Nuclear Regulatory Commission
P.O. Box 369
Avila Beach, California 93424

Dr. Richard Ferguson, Energy Chair
Sierra Club California
1100 11th Street, Suite 311
Sacramento, California 95814

Ms. Nancy Culver
San Luis Obispo
Mothers for Peace
P.O. Box 164
Pismo Beach, California 93448

Chairman
San Luis Obispo County Board of
Supervisors
Room 370 P.O. Box 112
County Government Center
San Luis Obispo, California 93408

Mr. Truman Burns
Mr. Robert Kinosian
California Public Utilities Commission
505 Van Ness, Room 4102
San Francisco, California 94102

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Services
Post Office Box 942732
Sacramento, California 94232

Diablo Canyon Independent Safety
Committee
ATTN: Robert R. Wellington, Esq.
Legal Counsel
857 Cass Street, Suite D
Monterey, California 93940

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Harris Tower & Pavilion
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Christopher J. Warner, Esq.
Pacific Gas & Electric Company
Post Office Box 7442
San Francisco, California 94120

Mr. Robert P. Powers
Vice President and Plant Manager
Diablo Canyon Nuclear Power Plant
P.O. Box 56
Avila Beach, California 93424

Telegram-Tribune
ATTN: Managing Editor
1321 Johnson Avenue
San Luis Obispo, CA 93406

Mr. Gregory M. Rueger

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cc w/encl:

NRC Resident Inspector
Diablo Canyon Nuclear Power Plant
c/o U.S. Nuclear Regulatory Commission
P.O. Box 369
Avila Beach, California 93424

Dr. Richard Ferguson, Energy Chair
Sierra Club California
1100 11th Street, Suite 311
Sacramento, California 95814

Ms. Nancy Culver
San Luis Obispo
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P.O. Box 164
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Mr. Robert Kinosian
California Public Utilities Commission
505 Van Ness, Room 4102
San Francisco, California 94102

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Services
Post Office Box 942732
Sacramento, California 94232

Diablo Canyon Independent Safety
Committee
ATTN: Robert R. Wellington, Esq.
Legal Counsel
857 Cass Street, Suite D
Monterey, California 93940

Redwood Alliance
P.O. Box 293
Arcata, California 95521

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Harris Tower & Pavilion
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Christopher J. Warner, Esq.
Pacific Gas & Electric Company
Post Office Box 7442
San Francisco, California 94120

Mr. Robert P. Powers
Vice President and Plant Manager
Diablo Canyon Nuclear Power Plant
P.O. Box 56
Avila Beach, California 93424

Chief, Fuel Cycle and Decommissioning
Branch, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Mr. Peter H. Kaufman
Deputy Attorney General
State of California
110 West A Street, Suite 700
San Diego, California 92101

Mr. Thomas A. Moulia
Humboldt Bay Plant Manager
Humboldt Bay Nuclear Power Plant
1000 King Salmon Avenue
Eureka, California 95503

Chairman, Humboldt County Board
of Supervisors
County Courthouse
825 Fifth Street
Eureka, California 95501

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Docket Nos.: 50-275, 50-323

License Nos.: DPR-80, DPR-82

Report No.: 50-275/97201, 50-323/97201

Licensee: Pacific Gas & Electric Company

Facility: Diablo Canyon Power Plant, Units 1 and 2

Location: Avila Beach, California

Dates: June 9-12, 1997

Inspectors: Stephen D. Alexander, Team Leader, NRR
Billy H. Rogers, Reactor Engineer, NRR

Approved by: Gregory C. Cwalina, Section Chief
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Enclosure 1

EXECUTIVE SUMMARY

During the period of June 9-12, 1997, representatives of the U.S. Nuclear Regulatory Commission's (NRC's) Special Inspection Branch conducted an inspection of Pacific Gas & Electric Company's (PG&E's) activities related to the procurement, modification, testing and installation of replacement 4-kV circuit breakers at the Diablo Canyon Power Plant (DCPP), Units 1 and 2.

The inspectors reviewed engineering and quality assurance documentation, interviewed cognizant staff and examined equipment in order to evaluate PG&E's dedication and modification of 4-Kv Yaskawa circuit breakers for use in safety related applications at DCPP. The primary purpose of the inspection was to determine if prototype design verification testing of the modified breaker was accomplished in accordance with applicable NRC regulations and industry standards. Although there is no regulatory guide endorsing the standards in question, PG&E had committed to them in the Final Safety Analysis Report (FSAR) for DCPP, and they are therefore relevant to the design and licensing basis of the plant.

A secondary purpose of the inspection was to examine several related issues involving production testing, modifications, post installation testing, and in-service failures. The review included all documentation pertinent to the design, design verification (including prototype testing), conversion/modification, fabrication of adapting hardware, production testing, procurement and dedication done by PG&E, its principal contractor, National Technical Systems (NTS), Inc., of Acton, Massachusetts, and the NTS subcontractor who actually performed the conversions, Power Distribution Services (PDS), Inc., of West Chester, Ohio.

The inspectors determined that PG&E's approach to design verification was generally consistent with the applicable NRC regulations, primarily Criterion III, "Design Control," of 10 CFR Par 50, Appendix B, in that PG&E undertook to verify, by engineering analysis and a testing program, that DCPP's safety-related 4-kV electrical distribution system as converted and modified would perform its safety functions under all design basis conditions.

The question of consistency of PG&E's approach to the conversion and its design verification process with applicable industry standards, in particular, taking credit for certain design verification tests done by the manufacturer of the circuit breakers used in the conversion, was referred to the Standards Board of the IEEE for consideration by the appropriate subcommittee. In a letter to the NRC, dated August 21, 1997, the Chairman of the High-Voltage Circuit Breaker Subcommittee of the Switchgear Committee of the IEEE Power Engineering Society (sponsor of the principal applicable standard, C37.59) confirmed the licensee's understanding of the intent of the standard regarding required design verification testing for conversions using modular assemblies.

The letter further stated that if the modular assembly is in no way altered with respect to the coupling of the interrupting chambers and the operating mechanism, then some of the original design tests performed by the manufacturer, such as short circuit current interruption, load switching and capacitance switching tests, need not be repeated. However, the letter

stated, to apply this waiver, it must be shown that the mechanical operating characteristics of the interrupting chamber, such as contact parting times and contact travel, are still within the range specified for the original module prior to the conversion.

With regard to the engineering evaluations necessary to support the PG&E approach, the inspectors could not, on the basis of documentation for review at DCPD, conclusively determine the adequacy of all the justifications for not reperforming certain design tests. Therefore, this issue remains unresolved and the inspectors identified the need for further review at NTS, and possibly also at PDS. Accordingly, this issue is designated Unresolved Item 50-275,323/97210-01.

Report Details

III. Engineering

E1 Conduct of Engineering

E.1 4-kV Switchgear Conversion Design Verification

a. Inspection Scope

In order to verify that PG&E and its contractors, NTS and PDS, had properly verified the interrupting capacity of the 350-MVA Yaskawa SF6 interrupter breakers, adapted to fit into 250-MVA GE Magne-Blast switchgear, the inspectors reviewed engineering and quality assurance documentation, interviewed cognizant staff, examined equipment, and evaluated PG&E's dedication and modification of 4-Kv Yaskawa circuit breakers for use in safety related applications at DCPD.

The primary issues examined were:

- Whether the design verification testing conducted on the complete conversion (consisting of the Yaskawa "modular assembly" plus the hardware to adapt it to the Magne-Blast cubicle) for PG&E (conducted for NTS at PSM, Inc., of Pittsburgh, Pennsylvania) was consistent with ANSI/IEEE Std C37.59-1991, Paragraph 5.1.4.2(2), for conversions using adapted modular assemblies.
- Whether taking credit for the Yaskawa ANSI testing of the modular assembly supplemented by technical evaluations was consistent with Section 5 of C37.59.
- Whether the Yaskawa technical evaluations of its modifications made to the modular assembly after the original design verification tests were adequate to demonstrate that the original test results were still valid after those factory modifications.
- Whether the technical evaluations by NTS to demonstrate that its additional testing of the complete conversion as required by C37.59 was not invalidated by modifications made in response to installation, setup and operational problems made subsequent to the tests, and
- Whether PG&E adequately resolved the findings identified in its audits of NTS, PDS, and Yaskawa.

Also examined were:

- The material of the breaker secondary disconnect pins and
- The potential overtravel situation involving the stationary auxiliary switch (SAS) in the cubicle possibly preventing full closure of the breaker, and several other interface and operation issues brought to the VIS inspectors' attention by personnel from DCPD Operations.

The inspectors reviewed the PG&E documentation related to the design tests of the 5GYB-1-1200 AND 5GYB-1-2000 circuit breakers performed by Yaskawa, PDS, and PSM; the evaluations performed by NTS which demonstrated the equivalency of the circuit breakers supplied to PG&E to the prototype circuit breakers tested by Yaskawa and the two circuit breakers tested by PDS and PSM; and the review of the production modifications made by Yaskawa and PDS following the ANSI/IEEE testing. The inspector's review was performed to verify that the circuit breakers supplied to PG&E were equivalent to the circuit breakers tested by Yaskawa, PDS, and PSM and that any modifications made to the circuit breakers by Yaskawa or PDS did not invalidate the original design testing.

b. Observations

b.1 Validity of Design Verification Approach

During the initial stages of the original circuit breaker production, Yaskawa had performed the ANSI/IEEE design testing on prototype breakers as representatives of the circuit breaker which Yaskawa would subsequently sell as commercial grade products (non safety-related). This circuit breaker was the Yaskawa SF₆ gas "Fluopac" Series, medium-voltage (4.76-kV-rated), rotary-arc circuit breaker of 350-MVA interrupting capacity. PG&E contracted with NTS to provide the Yaskawa circuit breakers as Class 1E (safety-related) equipment. NTS purchased breakers from Yaskawa and subsequently subcontracted their modification and additional testing to PDS and PSM, Inc. (a high-energy test facility in Pittsburgh), to be performed under the QA coverage of NTS.

The particular models of these breakers that underwent design verification testing were Yaskawa Models 5GYB-1-1200 and 5GYB-1-2000NTS. The conversions performed by PDS used the interrupting chambers and their attached operating mechanisms and chassis (frame) from the circuit breakers. According to the applicable industry standard, ANSI/IEEE Standard C37.59-1991, these components constitute a "modular assembly." PDS adapted the modular assemblies for retrofit into the existing GE Magne-Blast cubicles at DCPD by mounting each assembly in a custom-fabricated enclosure and truck unit containing the necessary hardware with which to make the primary and secondary electrical connections in the cubicles to the 4-kV busses and 125-Vdc control power respectively, as well as the mechanical interfaces with the cubicle vertical lift apparatus, stationary auxiliary switch, truck-operated cubicle switch, and cubicle interlock devices.

The applicable NRC regulations were Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix A); 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants;" 10 CFR 50.59, "Changes, Tests, and Experiments," 10 CFR Part 21, "Reporting of Defects and Noncompliance." The 10 CFR Part 50, Appendix B, criteria that are especially applicable to this project are Criterion III, "Design Control," and Criterion VII, "Control of Purchased Material, Equipment and Services."

The principal industry standards applicable to switchgear conversions are ANSI/IEEE Standards C37.59-1991, "IEEE Standard Requirements for Conversion of Power Switchgear Equipment;" C37.09-1979, "IEEE Standard Test Procedure for AC

High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis," and C37.04-1979, "IEEE Standard Rating Structure for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis." Although there is no regulatory guide endorsing the standards in question, the inspectors determined that PG&E had committed to them in the Final Safety Analysis Report (FSAR) for DCP, and they are therefore relevant to the design and licensing basis of the plant.

Paragraph 5.1.4.2(2) of C37.59, which deals with conversions using modular assemblies, requires that the modular assembly undergo the entire series of design tests in accordance with C37.09. PG&E and NTS interpreted this to mean that it must be verified that the complete series of design tests per C37.09 has been performed on a prototype(s) of the modular assembly, e.g., by the manufacturer as part of original testing, with satisfactory results. However, on the basis of the general guidance in the beginning of Section 5 of C37.59, PG&E further interpreted the standard to provide that if engineering evaluations of all subsequent modifications can demonstrate that functions or characteristics (e.g., interrupting capacity) of the converted breaker are not adversely affected by the modifications such that the original tests would be invalidated, then the C37.09 design tests of those functions or characteristics need not be repeated on the complete conversion.

According to the documentation provided to PG&E by NTS (and reviewed by the inspectors), the original circuit breaker design tests had been performed by Yaskawa in accordance with ANSI/IEEE C37.09-1979, C37.20.2-1987, C37.59-1-1991 and PG&E Specification 1001-E-NPG Section 12.1. The purpose of the tests was to determine the adequacy of the design of this particular type and model and its components parts to meet its assigned ratings and operate satisfactorily under normal service conditions/special conditions defined by the PG&E specification. NTS Report No.60431-95N-C, "Equivalency Evaluation of ANSI Type Tests and ANSI C37 Test Reports," Revision 7, dated June 4, 1997, documented the required ANSI/IEEE tests performed, specifying a description of each test, which company had performed the test, and including applicable test report.

Paragraph 5.1.4.2(2) of C37.59 then requires specific additional testing of the complete conversion, which included dielectric, momentary (C37.20.3-1987), continuous current, interlock and auxiliary functions, and mechanical endurance testing (C37.06-1987). The inspectors confirmed by review of the test documentation that these tests were performed on two representative complete conversions with satisfactory results.

In reviewing the design verification of the breaker conversion, the inspectors learned that PG&E had consulted with several industry experts, including, most notably, Dr. Ward Laubach of the Low-Voltage Switchgear Device Subcommittee of the IEEE Switchgear Committee of the IEEE Power Engineering Society, sponsor of ANSI/IEEE C37.59-1991, who asserted that PG&E's approach was consistent with the provisions of the standard. However, the inspectors noted that Dr. Laubach was also employed as a consultant to NTS, PG&E's primary contractor on the project. In addition, the inspectors noted that two other parties consulted by PG&E were (1) one of the other bidders on the project, an employee of Pacific Breaker Systems, Inc. (who had used the same approach in a

project involving French Merlin-Gerin breakers for the Quad Cities and Dresden Nuclear Stations), and (2) an employee of Square D Company, the U.S. representative for Merlin-Gerin (both of which companies are owned by the Schneider Electric conglomerate). The inspectors determined that the industry experts consulted by PG&E were not totally disinterested parties because they were directly or indirectly involved in this or other similar projects.

Therefore, by NRC letter dated July 24, 1997 (Attachment 1 to this report), the question of consistency of the approach with applicable industry standards, in particular, taking credit for certain design verification tests done by the manufacturer of the circuit breakers used in the conversion, was referred to the Standards Board of the IEEE for consideration by the appropriate subcommittee. In a letter to the NRC, dated August 21, 1997 (Attachment 2 to this report), the Chairman of the High-Voltage Circuit Breaker Subcommittee of the Switchgear Committee of the IEEE Power Engineering Society (sponsor of the principal applicable standard, C37.59) confirmed the licensee's and the inspectors' understanding of the intent of the standard regarding required design verification testing for conversions using modular assemblies.

The letter stated that while it was intended that all design tests be performed on the converted equipment, in the case of a conversion using a modular assembly, in addition to the specific tests explicitly required to be performed on the complete conversion by Paragraph 5.1.4.2(2) of C37.59-1991, only those design tests that cover an area of performance affected by the modifications associated with the conversion must be repeated on the complete conversion. The letter further stated that if the modular assembly is in no way altered with respect to the coupling of the interrupting chambers and the operating mechanism, then some of the original design tests performed by the manufacturer, such as short circuit current interruption, load switching and capacitance switching tests, need not be repeated. However, the letter stated, to apply this waiver, it must be shown that the mechanical operating characteristics of the interrupting chamber, such as contact parting times and contact travel, are still within the range specified for the original module prior to the conversion. Although there is no regulatory guide endorsing the standards in question, PG&E had committed to them in the FSAR for DCP, and they are therefore relevant to the design and licensing basis of the plant.

b.2 Equivalency Evaluation

The inspectors found that the original Yaskawa breaker required extensive and substantial modification in order to successfully adapt it for use in the Magne-Blast switchgear; modification for which neither Yaskawa, nor PG&E, nor its subcontractors had originally or promptly provided comprehensive engineering evaluation(s) (at least in english) to establish that the modifications would not adversely impact the tested interrupting capacity, and would not invalidate the original design tests performed by Yaskawa.

According to the certifications provided to PG&E, NTS had established that several areas were critical to determining that the circuit breakers supplied to PG&E were equivalent to the circuit breakers tested by Yaskawa, PDS and PSM. The documentation further certified that NTS had maintained the design

control of materials, dimensions, and processes, verified that all ANSI/IEEE testing was performed by approved vendors (or surveilled) and performed in a calibrated test facility, and materials on the supplied circuit breakers were appropriately dedicated. In addition, NTS reviewed all design changes made by Yaskawa and PDS and certified that the modification did not invalidate the ANSI/IEEE testing performed by Yaskawa and PDS (and for NTS/PDS at PSM).

Yaskawa had provided NTS information on all design, parts, or material changes made to the circuit breaker since May of 1993, when the type tests had been performed, to the time of the NTS purchase. The Yaskawa changes were contained in Yaskawa document no. GA9400864 Statement of Design Change, Revision 2, dated July 31, 1995, in section 5.0 of NTS Report No. 60431-95N-C. Yaskawa had made numerous changes to the operating mechanism, the interrupter and the general assembly, including items such as material changes, dimensional changes, and drawing changes. Yaskawa indicated in the document that none of the indicated changes would impact the results of the ANSI/IEEE type tests which had been performed by Yaskawa in 1993, and provided certification to that effect with the circuit breakers shipped to NTS. In addition, NTS had reviewed the changes and performed an evaluation of those which NTS considered had the potential to affect critical characteristics of the as-tested design and had concluded that the Yaskawa changes had not affected the results of the type tests originally performed by Yaskawa.

In addition to Yaskawa's modifications to the modular assemblies supplied to NTS, PDS, under NTS controls, had also made modifications to the circuit breakers to facilitate the modular conversion and allow them to operate in the installed DCPM Magne-Blast switchgear. The PDS engineering change notice (ECN) table for Job #1466, included in NTS Report No. 60431-95N-C, contained all of the changes that NTS considered relevant to the conversion design. The PDS modifications were primarily mechanical changes to the circuit breaker frame, wheels, and hardware. Each ECN was accompanied by an evaluation for impact on the circuit breaker ANSI/IEEE testing which had been performed by Yaskawa, PDS or PSM. NTS Report No. 60431-95N-C concluded that none of the PDS modifications invalidated the circuit breaker testing performed by Yaskawa, PDS or PSM.

b.3 Low-Voltage Fault Current Test

The inspectors found that although PG&E did not conduct a rated-voltage, rated-fault current interrupting capacity test on the complete conversion, it ordered a special fault current test at PSM (in addition to the additional testing on the complete conversion required by C37.59), which was conducted at 480 volts instead of 4760 volts. PG&E explained that there might be adverse effects of fault-current magnetic fields on the components added to the complete conversion to adapt it to the auxiliary switches and mechanical interlocks of the Magne-Blast cubicle. These effects might cause the added hardware to impede breaker tripping on a fault; effects that may not have been covered by the testing required by the standards. Therefore, PG&E decided to conduct an interrupting capacity test of a prototype of the complete conversion unit that was already undergoing the testing required by Paragraph 5.1.4.2(2) of ANSI/IEEE Std C37.59 in July 1996 at PSM. However, due to some problem with or unavailability of PSM's main high-current test facility

generator, the test was being done at 480 Vac (although presumably at the required current level) instead of the 4760 Vac reportedly required for 4-kV breakers by C37.09 (Referenced in C37.59). PG&E argued, that even though the test was at low voltage, the fault current would produce magnetic fields to adequately simulate the fault interrupting conditions that might conceivably affect the operation components in question irrespective of the voltage at which the 41,000-amp test was conducted, as only the current, not the voltage gives rise to the magnetic fields.

c. Conclusion

The inspectors concluded that PG&E's approach to design verification was generally consistent with the applicable NRC regulations, primarily Criterion III, "Design Control," of 10 CFR Par 50, Appendix B, in that PG&E undertook to verify by engineering analysis and a testing program that DCP's safety-related 4-kV electrical distribution system as converted and modified would perform its safety functions under all design basis conditions. On the basis of the IEEE interpretation of the intent of C37.59, the inspectors further concluded that the PG&E approach was consistent with applicable industry standards, provided the required supporting engineering evaluations were adequate. On the basis of the review of the supporting documentation supplied by Yaskawa, NTS, and PG&E, the inspectors further concluded that the applicable ANSI/IEEE tests had been performed with satisfactory results.

However, although PG&E had certifications from NTS that equivalency had been established between the tested circuit breakers and those supplied to PG&E, the inspectors could not conclude on the basis of documentation available for review at DCP that the technical evaluations performed primarily by NTS demonstrated that the modification made by Yaskawa and PDS had not impacted the validity of the ANSI/IEEE tests performed on the circuit breakers; therefore, the inspectors could not conclusively determine the adequacy of all the justifications for not reperforming certain design tests. Accordingly, the inspectors identified the need for further review at NTS, and possibly also at PDS and designated this issue as Unresolved Item 50-275.323/ 97201-01.

E.2 4-kV Production Breaker Installation and Performance Concerns

a. Inspection Scope

In order to address three areas of concern that had been identified by the inspectors in preparation for this inspection: (1) 4-kV breaker secondary disconnect contact pin material, (2) breaker operation interference due to stationary auxiliary switch (SAS) overtravel, and (3) SAS adjustment/performance, the inspectors reviewed the associated DCP Action Requests (ARs) and their dispositions, interviewed cognizant engineering and operations personnel, and examined affected components.

b. Observations

b.1 Secondary Disconnect Pin Material

In the case of the secondary contact pin material, the pins on the breakers for DCP Unit 1, which had not yet been shipped were replaced with pins of a stiffer, more tempered material. The pins on the breakers in Unit 2 were not replaced en masse, but rather inspected and replaced if permanently deformed or otherwise damaged or degraded. In addition, PG&E had discovered that the reason the pins (of more malleable material than those of the original GE secondary contact blocks) were becoming deformed, in some cases enough to degrade electrical contact, was the manner in which maintenance electricians had become used to removing the secondary contact test position adapter cable and plug assembly, i.e., by yanking it off partially sideways. Accordingly, PG&E changed procedures and conducted training to ensure that the test cable plugs would be pulled off carefully and only with vertical force to prevent any future pin deformation. In addition, procedures were changed to require the use of a GE secondary contact pin spreading or gapping tool after each test cable removal to ensure that the four segments of each pin were properly spread for adequate electrical contact when the breaker was fully racked up into its operate position. The representative from DCP operations who had also related this concern to the team was satisfied that the corrective action was adequate. PG&E stated that there were sufficient replacement pins of the improved, stiffer/more tempered material on site to replace any pins that should become irreparably deformed despite improved handling procedures.

b.2 SAS Overtravel

With respect to the concern about the potential for an overtravel condition in the SASs potentially preventing the converted Yaskawa breakers from closing fully (which never occurred in service, only during experimentation), the inspectors determined that PG&E's minimum required gap (0.040") between the stationary auxiliary switch operating rod and the breaker's mechanism-operated cubicle plunger (set by adjusting shims in the plunger and thereafter by manually adjusting open breaker elevation in its cubicle), in conjunction with the procedures that required checking and establishing this gap each time a breaker was racked in, would prevent the overtravel condition from occurring.

b.3 SAS Adjustment and Performance

In addition, the DCP operations representative had related concerns to the inspectors regarding the several other interface problems that had been encountered relating to the SASs. Having suffered several equipment failures since the installation of the converted Yaskawa breakers attributable to problems with the SASs (all sets of contacts not always changing state with breaker operation), PG&E had determined (through testing) the worst case stroke requirement (they are somewhat variable) among all the SASs (i.e., stroke of the operating rod required to ensure that all contacts in the SAS, a GE SB-12 switch, will fully change state). Some older switches that had actually caused failures or were found through testing to be unreliable or out of tolerance were replaced. PG&E then determined the maximum allowable breaker- open, plunger-SAS operating rod gap that would ensure that all SAS contacts would change state when the breaker closed given the worst case (largest) required stroke of the all the SB-12 switches.

During this period, another related problem presented itself. Upon investigating the failure of a pump to start, PG&E found that another instance of 4-kV breaker-SAS adjustment to be the cause. Inspection revealed that even though the gap had been set by procedure when the affected breaker was last racked in, the as-found gap was too wide. Thus when this breaker was closed, not all of the contacts in its SAS had changed state. Through further testing and investigation, PG&E discovered that when the adjustment screw at the top of the breaker's SAS plunger is retorqued after replacing adjustment shims, the linkages that operate the SAS plunger become slightly cocked as joints in the linkages expand to their maximum end float. This condition raises the SAS plunger up as much as 0.050 or 0.060 inch above its normal, breaker-open, rest position. During the first subsequent closing operation after the gap has been set with the plunger in the slightly raised position, the end floats all take up which effectively shortens the plunger stroke and thereafter not all of the SAS contacts may change state. To prevent this and ensure the normally consistent SAS plunger stroke, PG&E changed procedures and conducted training to ensure that the plunger is tapped down into its fully withdrawn, breaker-open rest position before setting the plunger-to-SAS operating rod gap by manually adjusting the breaker elevation in the cubicle.

c. Conclusion

With respect to the two 4-kV breaker installation, interface, and performance concerns, the inspectors determined that PG&E's corrective action was appropriate and adequate.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Shawn LaForce	RS	Engineer
Brad Olson	USNRC	Project Engineer
Don Allen	USNRC	Resident Inspector
Stan Ketelsen	NSAL-RS	Supv.
Bill Bayne	Proc. Svs	Supv.
Bob Whitgell	NQS	Supv.
Charlie Nichols	Materials	Director
Michael Jacobson	NQS	Sr. Engineer
Chuck Lewis	NQS	Engineer
Bill Colry	Reg. Svcs.	Engineer
Tom Bennett	OS	Director
Dave Taggart	NQS-Engineering	Director
	Dir. & Procurement	
Thomas W. Packy	NQS-PA	Lead Auditor
Klemme Herman	NTS/DES/EE	Elect. Sup.
Ed Kahler	NTS/Tech	PM
Tom Fetterman	NTS/ES	Director
Pat Colbert	NTS/ES	Elect. Sup.
Eric Nelson	Mechanical Maint.	General Foreman
David Oatley	Maint. Serv.	Manager
Jim Molden	Operations	Manager
Terry Grebel	Reg. Services	Director

Open Items

This report categorizes the inspection findings as unresolved items and inspection follow-up items in accordance with the NRC Inspection Manual, Manual Chapter 0610. An unresolved item (URI) is a matter about which more information is required to determine whether the issue in question is an acceptable item, a deviation, a nonconformance, or a violation. The NRC Office of Nuclear Reactor Regulation will issue any enforcement action resulting from their review of the identified unresolved items. An inspection follow-up item (IFI) is a matter that requires further inspection because of a potential problem, because specific licensee or NRC action is pending, or because additional information is needed that was not available at the time of the inspection.

Item Number	Finding	Title
50-275;50-323/97201-01	URI	Meeting ANSI/IEEE C37.59-1991 (Section 5)

**UNITED STATES
NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

July 3, 1997

Mr. D. M. Smith, President
PECO Nuclear
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Smith:

**SUBJECT: LIMERICK GENERATING STATION ASSURANCE OF VENDOR QUALITY
INSPECTION (REPORT NOS. 50-352/96-201 AND 50-353/96-201)**

During the periods August 5 through 9, 1996, and March 10 through 14, 1997, the U.S. Nuclear Regulatory Commission's (NRCs) Office of Nuclear Reactor Regulation (NRR) performed a pilot Assurance of Vendor Quality Inspection at PECO Nuclear Offices in Wayne, Pennsylvania. The inspection was related to activities at the Limerick Generating Station, Units 1 and 2. This was the first of a series of pilot inspections being conducted to evaluate the implementation of licensee safety-related procurement programs.

The results of this inspection are contained in the attached inspection report. Overall, the team found that the procurement of safety-related items and services was adequately performed and the procurement process was being implemented per program requirements except for the concerns identified in the report.

An unresolved item was identified concerning the receipt inspection and acceptance of ASME Code, Section III items from material suppliers without the complete documentation required by Paragraphs NCA-3861(b) and NCA-3862.1(b) of Subsection NCA, Section III of the ASME Code.

In addition, the staff identified a weakness regarding PECO Nuclear Quality Assurance (NQA) group review and acceptance of an audit report performed by another utility under the auspices of the Nuclear Utilities Procurement Issues Committee (NUPIC) and use of this review of the audit report as the basis for maintaining ACCUTECH as an approved vendor on the PECO Evaluated Vendors List (EVL). The NQA review of the NUPIC audit report did not adequately address the commercial grade dedication sampling issues described in the NUPIC audit report for applicability to PECO procurement requirements and did not question the basis for verification of lot homogeneity for finished fasteners purchased from non-approved suppliers as described in the audit report.

You are requested to respond to the weakness identified in the inspection report. In your response please address, 1) your process for placing vendors on the EVL based upon third party review, 2) your process for identifying, reviewing and addressing audit findings and followup correspondence that identify issues applicable to your procurements, including findings from NRC inspections and 3) your assurance that past procurements from vendors, including ACCUTECH, similarly placed on the EVL are adequate based upon the weakness identified. Please send your response to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of receipt of this letter.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and inspection report will be placed in the NRC Public Document Room. Any enforcement action resulting from this inspection will be issued by the NRC Region I office via a separate correspondence. Should you have any questions concerning the attached inspection report, please contact the inspection team leader Mr. Richard P. McIntyre at (301) 415-3215.

Sincerely,

A handwritten signature in black ink, appearing to read "Robert M. Gallo". The signature is fluid and cursive, with a large, stylized "R" and "G".

Robert M. Gallo, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353
License Nos.: NPF-39 and NPF-85

Enclosure: Inspection Report No. 50-352/96-201
and 50-353/96-201

cc: See next page

PECO Energy Company

**Limerick Generating Station,
Units 1 & 2**

cc:

**J. W. Durham, Sr., Esquire
Sr. V.P. & General Counsel
PECO Energy Company
2301 Market Street
Philadelphia, PA 19101**

**Chief-Division of Nuclear Safety
PA Dept. of Environmental Resources
P.O. Box 8469
Harrisburg, PA 17105-8469**

**Manager-Limerick Licensing, 62A-1
PECO Energy Company
965 Chesterbrook Boulevard
Wayne, PA 19087-5691**

**Director-Site Engineering
Limerick Generating Station
P.O. Box A
Sanatoga, PA 19464**

**Mr. Walter G. MacFarland, Vice President
Limerick Generating Station
Post Office Box A
Sanatoga, PA 19464**

**Manager-Experience Assessment
Limerick Generating Station
P.O. Box A
Sanatoga, PA 19464**

**Plant Manager
Limerick Generating Station
P.O. Box A
Sanatoga, PA 19464**

**Library
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406**

**Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406**

**Senior Manager-Operations
Limerick Generating Station
P.O. Box A
Sanatoga, PA 19464**

**Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Limerick Generating Station
P.O. Box 596
Pottstown, PA 19464**

**Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803**

**Director-Site Support Services
Limerick Generating Station
P.O. Box A
Sanatoga, PA 19464**

**Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195**

**Chairman
Board of Supervisors
of Limerick Township
646 West Ridge Pike
Linfield, PA 19468**

**U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION**

Docket Nos.: 50-352 and 50-353
License Nos.: NPF-39 and NPF-85

Report Nos.: 50-352/96-201 and 50-353/96-201

Licensee: PECO Energy Company

Facility Name: Limerick Generating Station, Units 1 and 2

Location: Correspondence Control Desk
P.O. Box 195
Wayne, Pa 19087-0195

Dates: August 5-9, 1996
March 10-14, 1997

Inspectors: Richard P. McIntyre, Team Leader, NRR
Uldis Potapovs, Senior Reactor Engineer, NRR
Bill Rogers, Reactor Engineer, NRR

Approved by: Gregory C. Cwalina, Section Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

EXECUTIVE SUMMARY

From August 5 through 9, 1996, and March 10 through 14, 1997, representatives of the U.S. Nuclear Regulatory Commission's (NRC's) Special Inspection Branch conducted an inspection of PECO Energy Company (PECO) activities related to the procurement of products and services used in safety-related applications at the Limerick Generating Station, Units 1 and 2 (LGS).

The inspection team reviewed PECO's procurement program to assess its compliance with the quality assurance (QA) requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). The inspection scope included Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," and Regulatory Guide 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants." Applicable industry standards include the American National Standards Institute (ANSI) N45.2.13, "Quality Assurance Requirements for the Control of Procurement of Items and Services for Nuclear Power Plants," and ANSI N45.2.12, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants."

The NRC conducted this inspection, the first pilot inspection in this area, using draft Inspection Procedure, "Assurance of Vendor Quality," dated July 16, 1996, with the intent to use this inspection as input to finalize the inspection procedure for future inspections of this type. The objective of the inspection was to ascertain whether PECO is effectively monitoring the control of quality of safety-related products and services by contractors and subcontractors (hereafter referred to as "vendors"). This was done by assessing attributes of the licensee's vendor oversight program and verifying its implementation with regard to selected vendors based upon licensee and vendor documentation, NRC regulations, regulatory guides, and applicable industry standards.

The team reviewed the PECO program and its implementation for the procurement of items and services used in safety-related applications at LGS. The team also reviewed the PECO program and its implementation for determination or verification of suitability of those items for their intended or approved safety-related applications. The inspection included a review of procedures and representative records, including approximately 40 procurement packages for mechanical, material and electrical items; interviews with PECO staff, including senior management and LGS site personnel; and observations by the inspection team members. The inspection team findings were discussed with PECO's representatives and senior management at the interim exit meeting held August 9, 1996, and the final exit meeting held on March 14, 1997.

Overall, the team found that the procurement of safety-related items and services was adequately performed and the procurement process was being implemented per program requirements. However, the team identified a program weakness that concerned the PECO Nuclear Quality Assurance (NQA) group review and acceptance of an audit report performed by another utility in May 1996 under the auspices of the Nuclear Utilities Procurement Issues Committee (NUPIC) and use of this review of the audit report as the basis for maintaining ACCUTECH as an approved vendor on the PECO Evaluated Vendors

List (EVL). The NQA review of the NUPIC audit report did not adequately address the commercial grade dedication sampling issues described in the NUPIC audit report for applicability to PECO procurement requirements and did not question the basis for verification of lot homogeneity for finished fasteners purchased from non-approved suppliers as described in the audit report. PECO did not identify that ACCUTECH's sampling process needed improvement for them to be maintained on the EVL as an approved supplier. The NUPIC audit report described NRC inspection findings and follow-up correspondence issues from the December 1994 NRC Vendor Inspection Branch inspection at Cardinal Industrial Products (previous name of ACCUTECH). The sampling process reviewed at ACCUTECH in May 1996 was essentially the same process reviewed by the NRC in December 1994. The PECO NQA review used the NUPIC audit report for maintaining ACCUTECH on the EVL without adequately addressing the issues described in the NRC Notice of Nonconformance identified during the December 1994 inspection .

Another issue identified concerned the receipt inspection and acceptance of ASME Code, Section III items from material suppliers without the complete documentation required by Paragraphs NCA-3861(b) and NCA-3862.1(b) of Section III of the ASME Code. Paragraph NCA-3862.1(b) states that, when the required chemical analyses, tests, examinations, heat treatment, etc., are subcontracted, the approved suppliers certification for the operations performed shall be furnished as an identified attachment to the certified material test report (CMTR). In several instances the approved supplier certifications were not furnished with the CMTR and were not included in the document package.

Report Details

III. Engineering

E7 Quality Assurance in Engineering Activities

E7.1 Evaluated Vendors List

a. Inspection Scope

The inspectors reviewed PECO procedure P-C-9, "Evaluated Vendors List," Revision 1, dated December 15, 1993, and representative documentation to verify implementation. P-C-9 established requirements, assigned responsibilities and provided guidance for the preparation and maintenance of the PECO Evaluated Vendors List (EVL), the EVL Conditional Clauses, and the Alert List. The inspectors reviewed PECO procedure NQA-19, "NQA Evaluation of Vendors," Revision 4, dated February 20, 1996, and Revision 5, dated August 5, 1996, and representative documentation to verify implementation. NQA-19 established requirements, assigned responsibilities and provided guidance for the PECO Nuclear Quality Assurance (NQA) evaluation of vendors for acceptance on the PECO EVL.

b. Observations and Findings

The EVL was a listing of vendors which PECO had evaluated to determine their capabilities of supplying equipment, components, and services, in accordance with the conditions of a purchase order or contract. NQA-19 indicated that PECO evaluated vendors by several methods including assessments, commercial grade surveys, QA Manual reviews, and annual evaluations. Evaluated vendors were listed on the PECO EVL as Approved (A), Conditionally Approved (C), Commercial Grade (G), or Bidder (B). Approved vendors had a quality program complying with 10 CFR Part 50, Appendix B or another applicable standard and PECO had verified implementation of the quality program by an initial qualification audit or by a triennial follow-up audit. Conditionally Approved vendors were required to comply with a set of quality assurance conditional clauses imposed by PECO as a result of vendor evaluation activities. The applicable "C" vendor quality assurance conditional clauses were documented on the PECO EVL and imposed on PECO POs to the vendor. In addition, PECO had verified implementation of the quality program by an initial qualification audit or by a triennial follow-up audit for the "C" vendor. For Commercial Grade vendors, PECO had identified the vendor-controlled critical characteristics for the products to be purchased from the vendor, had obtained satisfactory results from a PECO evaluation of applicable vendor quality program documents, and had verified implementation of the quality program by an initial qualification audit or by a triennial follow-up audit for the "G" vendor. For Bidders (B), NQA reviewed the QA program documents to determine that the vendor had the capability to meet 10 CFR Part 50, Appendix B quality requirements. The bidder review process was documented on the PECO Vendor QA Manual Checklist.

For vendors listed on the EVL as Conditional, the additional requirements, listed in the Conditional Clauses section of the EVL, were imposed on the procurement documents or purchase orders. P-C-9 indicated that this status was applicable to vendors who operated with a limited scope 10 CFR Part 50 Appendix B program or who required additional requirements on POs based on audit, surveillance, or survey results, but were capable of supplying safety-related items or services. The Vendor Alert List identified areas to be reviewed concerning equipment and services and associated vendors based on information obtained from NRC Bulletins and Information Notices and PECO 10 CFR Part 21 reportable items.

The inspectors reviewed the PECO EVL dated August 5, 1996, to verify implementation of P-C-9. The EVL listed approximately 450 vendors including name, address, product or service available for PECO use, EVL status, and the ASME status. The vendors whose audits or purchase documents reviewed by the inspectors were determined to be appropriately listed on the EVL. In addition, the inspectors noted that all vendors reviewed, which were listed as Conditionally Approved, had corresponding entries in the Conditional Clauses section which identified conditions to be met.

c. Conclusions

The inspectors concluded that PECO had imposed adequate procedural controls on the EVL and that it was well organized and had been maintained to an up to date condition.

E7.2 Review of PECO Vendor Assessment and Surveillance Process

a. Inspection Scope

The inspectors reviewed PECO procedure NQA-20, "NQA Vendor Assessments and Surveillance," Revision 7, dated March 8, 1996, and representative documentation to verify implementation. NQA-20 established requirements, assigned responsibilities and provided guidance for the coordination, preparation, performance, and reporting of PECO Nuclear Quality Assurance (NQA) vendor assessments and surveillance.

b. Observations and Findings

NQA-20 defined Pre-award assessments as activities performed to evaluate a vendors's QA program prior to placement on the PECO EVL as an approved supplier and issuance of a PO to that vendor, and the triennial assessments as the activity performed to initially place and continue maintaining the vendor on the PECO EVL as a supplier. The inspectors noted that PECO used the term assessment to define the process of performing the on-site review of the applicable portions of the vendor's QA program and its implementation, to determine that it meets the applicable requirements of 10 CFR Part 50, Appendix B. NQA-20 defined "assessment" as a process equal to or exceeding the requirements of "audit" as defined in ANSI N45.2.

The PECO Assessment Team Leader (ATL) prepared an Assessment Plan and checklist listing based on a variety of sources which would indicate areas needing to be reviewed during the assessment which included the Vendor QA documents, PECO procurement documents, the PECO Vendor Alert List, the INPO Nuclear Network, NRC Vendor Inspection Reports, previous assessment results, EVL corrective actions requirements, and nonconformance data. The Assessment Plan was to be performance based using methods such as comparing performance objectives and acceptance criteria to the final result and observation of ongoing activities. In addition, NQA-20 indicated that the NUPIC Performance Based Supplier Audit Checklist should be used in developing the Assessment Plan when appropriate. The objective in development of the Assessment Plan was to determine what vendor program and processes should be evaluated during the assessment to verify product acceptability for its intended application based on the products essential function, product configuration, essential parts and components, and critical characteristics. The inspectors reviewed several Assessment Plans and concluded that they effectively implemented program requirements.

The assessment consisted of a pre-assessment conference, assessment performance, and a post-assessment conference. Nonconformances identified during the assessment were documented on Vendor Corrective Action Requests (VCR) and if the nonconformance could potentially affect hardware the ATL would request that the vendor document the identified nonconformance on a vendor Nonconformance Report. If the vendor dispositioned the nonconformance as "acceptable as is" or "repair" in accordance with the vendor's QA program the Assessor requested that the vendor initiate a Vendor Deviation Request (VDR) as required by the PECO PO. PECO issued the Assessment Report, within 30 days of the completion of the assessment, which included an Assessment Summary (scope, assessment results, strengths and weaknesses, recommendations, and evaluation of activities), Investigation Results (activity investigated, summary of acceptable and unacceptable results, reference to VCRs and recommendations (REC)), and Administrative Details (names of PECO personnel involved, vendor personnel contacted during the assessment, and VCRs and RECs). PECO required the vendor to respond to the issued VCRs, reviewed the received responses, and informed the vendor of the results. The inspectors reviewed several Assessment reports (discussed in Section E.7.3 of this report) and concluded that they effectively implemented the program requirements.

Surveillance were performed to meet a PECO EVL Conditional Approval requirement which indicated that the vendor contact PECO prior to performing a certain portion of the work so that PECO could perform a surveillance of the activities. Surveillance were also performed to meet PECO safety-related PO requirements which specified vendor surveillance requirements. Advance preparation for the Surveillance include development of a Surveillance Checklist to be approved by the Working Lead and contacting the vendor to initiate and coordinate activities. The Surveillance performance included observation and evaluation of activities and objective evidence for conformance with the PO requirements and completion of the Surveillance Checklist. Nonconformances identified during the surveillance were documented on a VCR and if the nonconformance could potentially affect hardware the Assessor would request that the vendor document the identified nonconformance on a vendor Nonconformance Report. If the vendor dispositions the as "acceptable as is" or "repair" in accordance with the vendor's QA program

the Assessor requested that the vendor initiate a VDR as indicated in the PECO PO. After completion of the Surveillance the Assessor completed a Certificate of Surveillance including the PO item number, quantity, description, and PECO nuclear code number for the item being released. A copy of the Certificate of Surveillance was provided to the vendor for inclusion into the documentation package shipped with the product. Any VCRs generated by the surveillance were provided to the vendor for response accompanying a PECO Nuclear Transmittal Letter issued within 30 days of the completion of the surveillance. The inspectors reviewed documentation associated with several surveillances and determined that the program requirements were being effectively implemented.

c. Conclusions

The inspectors concluded that PECO had developed adequate procedural requirements to establish a program to effectively perform assessments and surveillances to support verification of quality activities and that PECO had generally implemented these procedural controls on the performance of assessments and surveillance.

E7.3 Review of Selected PECO Assessments and Surveillance

a. Inspection Scope

The inspectors review of a listing of PECO performed assessments indicated that PECO typically performed twelve assessments yearly. A large portion of vendor placement on PECO's EVL was based on PECO's formal review and acceptance of NUPIC Joint Utility Audits or NUPIC Member Audits. In addition, PECO had led a number of NUPIC Joint Utility Audits recently including GE-Fenton, Bechtel, Crane Chem Pump, Amerace, ARCOS, Leeds & Northrup, Overly, NUS, and Ingersoll-Rand. To verify implementation of PECO vendor assessments, the inspectors reviewed the following NUPIC member audits and the accompanying PECO NQA reviews of these audits.

b. Observations and Findings

b.1 ACCUTECH

The inspectors reviewed the Comanche Peak Steam Electric Station TU Electric (TUE) QA Audit Report QAA-96-010 of ACCUTECH dated May 23, 1996. The audit, conducted April 29 through May 2, 1996, was led by TUE and performed in accordance with the requirements of the TUE QA program, under the auspices of NUPIC. The audit also included representatives of Iowa Electric Services (IES), Houston Light and Power, and Northern States Power. The audit was performed to assess ACCUTECH's QA program and its implementation in supplying ASME code and non-code materials to the applicable requirements of 10 CFR Part 50, Appendix B, ASME NQA-1, and ASME Section III, Subsection NCA 3800. The audit included the areas of Order Entry; Commercial Grade Dedication; Procurement; Material Control/ Handling, Storage, and Shipping; Fabrication, Assembly, and Special Processes; Tests and Inspection; Calibration; Document Control/Procedure Adequacy; Organization/Program Compliance; Nonconforming Conditions; Corrective Actions; Internal and External audits; and Training/Certification.

PECO performed a formal review of the TUE audit of ACCUTECH as documented on PECO Audit Report Review Form (ARRF) dated June 27, 1996. The ARRF indicated that TUE had performed an audit of the areas applicable to PECO's planned purchases using the appropriate criteria (10 CFR Part 50, Appendix B, NQA-45.2 and NQA-1). The ARRF required a review of numerous areas including Lead Auditor certification, audit scope, applicability of QA program, sufficient objective evidence in audit package, audit findings issued and applicability of items on order, in stock, or installed in the plant, and whether corrective actions for findings were adequate for the PECO application. PECO had concluded that the TUE audit was acceptable for PECO's purchases and supported ACCUTECH's status on the PECO EVL, however, the review did require that specific information be added to the EVL for PECO information only. This information dealt with, among other things, a description of the sample plans implemented by ACCUTECH.

The inspectors reviewed the applicable sections in the TUE audit report that addressed the nonconformances and issues identified in the December 1994 NRC Inspection (Report No. 99901076/94-01) at Cardinal Industrial Products (former name of ACCUTECH), concerning commercial grade dedication, and sampling relating to commercial grade item (CGI) dedication. The audit reviewed the applicable portions of the ACCUTECH program as it relates to sampling as part of commercial grade dedication; however, it did not appear that the NUPIC audit verified or attempted to verify the ACCUTECH basis for their sampling plans utilized for destructive and nondestructive testing. This issue had been identified as a nonconformance in the December 1994 NRC inspection at Cardinal. Also, when reviewing the documentation on the NUPIC audit checklist, the inspectors did not identify any evidence that the auditors reviewed ACCUTECH's rationale for verifying lot homogeneity for finished fasteners that are purchased from non-approved suppliers. This method, in turn forms the basis for ACCUTECH's selection of the CGI sampling plans.

PECO's review and acceptance of the NUPIC audit report of ACCUTECH did not question the sampling plans implemented by ACCUTECH or address the fact that the audit report identified that the NRC had issued a nonconformance to ACCUTECH concerning CGI dedication sampling deficiencies in December 1994 and they were still implementing basically the same dedication sampling process in May 1996. PECO stated that part of the reason the TUE audit was accepted was that the audit report concluded that the sampling plan(s) invoked by the ACCUTECH program for the products reviewed were deemed to be adequate. The inspectors stated that the TUE audit report described the NRC findings and follow-up correspondence from the December 1994 inspection at Cardinal and that the sampling process reviewed at ACCUTECH in May 1996 was essentially the same process reviewed in December 1994. In their formal AARF review conducted June 27, 1996, PECO did not identify that ACCUTECH's sampling process needed improvement for them to be maintained on the EVL as an approved supplier.

Review of the August 5, 1996, PECO EVL showed ACCUTECH listed as a Conditionally Approved supplier of nuclear fasteners and materials with two vendor conditional approval (CA) clauses describing the requirement of implementation of the Quality Systems Manual to be included on purchase orders and also included, among other items, the various sampling plans used by ACCUTECH under "For PECO Information." Since the initial inspection, PECO had made several changes to

the CA requirements to include specific information on commercial grade dedication and sampling. PECO stated that these revisions had been initiated prior to the NRC inspection in August 1996.

PECO stated they received NRC Information Notice (IN) 96-40 on July 31, 1996, and they initiated research on actions to be taken to address commercial grade dedication program weaknesses described in IN 96-40. As a result of this evaluation and further review of vendor issues contained in IN 96-40 and in Vendor Inspection Reports, PECO revised their EVL CA clauses to define PECO Nuclear expectations for acceptable commercial grade dedication, particularly in the area of sampling. These CA requirements were invoked on ACCUTECH and Allied Nut and Bolt for all future purchase orders in an August 20, 1996, letter. Several other vendors were also notified of the revised EVL CA requirements in the September - October 1996 time frame. In December 1996, letters were sent to each of the applicable vendors requesting a copy of their procedures used to implement the CA requirements. The inspectors verified that the vendors who did not submit their procedures to PECO Nuclear NQA for review were either downgraded or removed from the EVL.

During the March 1997 inspection at PECO, the inspectors discussed concerns with the use of the term "heat lot" traceability in the CA requirements and the resulting potential misinterpretation of what would be required for heat lot traceability regarding sampling to meet these CA requirements. PECO agreed that the CA requirements could be clarified and did so prior to the completion of the inspection.

b.1.1 Vendor Inspection at ACCUTECH

After the August 1996 inspection at PECO, follow-up vendor inspections were performed at ACCUTECH in November 1996 and January 1997 and documented in Inspection Report 99901307/96-01, dated March 4, 1997. During the November 1996 inspection at ACCUTECH the inspectors reviewed the sample plan methodology as part of the commercial grade dedication process currently in place. The inspectors determined that it was basically the same process that was reviewed during the NRC's December 1994 inspection at Cardinal, in that it places heavy reliance on visual and dimensional inspection to support the verification of lot homogeneity. No revisions had been implemented in the ACCUTECH QA program that supported the sampling process rationale described in various 1995 correspondence to the NRC, especially the August 30, 1995, letter. This letter was the last to formally respond to the sampling nonconformance identified in NRC Inspection Report No. 99901076/94-01.

During the November 1996 NRC inspection ACCUTECH could not provide a documented basis to support the sampling information that was described to the NRC in the August 30, 1995 letter. ACCUTECH then stated that they had recently written a "white paper" that described proposed changes to the ACCUTECH sample plan methodology for testing and examination. This document was dated November 7, 1996, but had not as yet been implemented as part of the ACCUTECH QA program.

In conclusion, the inspectors determined that ACCUTECH placed heavy reliance on visual and dimensional inspection to support the verification of lot homogeneity. Based upon this method for verification of lot homogeneity, ACCUTECH then utilized the ASTM A-325 shipping lot sampling plan for destructive testing (material chemistry and mechanical properties) and the EPRI guidelines for nondestructive testing (dimensional). The inspectors determined that visual inspection for shipping damage and manufacturing defects can not assure that all items in the same product lot were manufactured from the same heat of material or were heat treated under the same conditions and that, as discussed in NRC Inspection Report 99901076/94-01 and related correspondence, the use of ASTM A-325 shipping lot sampling plan is inappropriate for this application.

b.2 Dragon Valves

The inspectors reviewed the Illinois Power Company (IPC) Audit Report of Dragon Valves, Inc. (Dragon), dated October 27, 1995, which documented the September 25-28, 1995, nuclear assessment of Dragon. PECO had performed a review and accepted the Audit Report on December 13, 1995, as the basis for placing Dragon on the PECO EVL as an approved supplier.

IPC had led and performed the audit as a NUPIC Joint Utility Audit with Commonwealth Edison, Entergy Operations, and Illinois Power providing audit team members. The audit was performed to assess Dragon's QA program and its implementation in supplying ASME code and non-code valves, parts, and components to the applicable requirements of 10 CFR Part 50, Appendix B, and ASME Section III, NCA 4000. The audit included the areas of Order Entry; Procurement; Material Control/Handling, Storage, and Shipping; Fabrication, Assembly, and Special Processes; Tests and Inspection; Calibration; Document Control/Procedure Adequacy; and Program Compliance.

The audit report concluded that Dragon was implementing an effective program for the product to be supplied with the exception of two findings that were not considered significant enough to have any major impact on work previously performed by Dragon. When reviewing the NUPIC audit package the inspectors noted that the audit identified that Dragon does not audit vendors who hold current ASME Quality System Certificates (QSCs) and also that Dragon does not perform any commercial grade dedication. It further stated that CMTRs for material received from unapproved suppliers and QSC holders is validated through receipt inspection and independent chemical analysis performed by an approved supplier.

PECO had performed a formal review of the IPC audit of Dragon as documented on PECO ARRF dated November 13, 1995. The ARRF indicated that IPC had performed an audit of the areas applicable to PECO's planned purchases using the appropriate criteria (10 CFR Part 50, Appendix B, and ANSI N45.2). The ARRF required a review of numerous areas including Lead Auditor certification, audit scope, applicability of QA program, sufficient objective evidence in audit package, audit findings issued and applicability of items on order, in stock, or installed in the plant, and whether corrective actions for findings were adequate for the PECO application. PECO had concluded that the IPC audit had been applicable to PECO's purchases, that the findings and PECO's concerns had been adequately addressed, and PECO had sufficient basis for placement of Dragon on the PECO EVL with

conditional PO requirements. Review of the August 5, 1996, PECO EVL showed Dragon as a conditionally approved supplier of nuclear valves and replacement parts, with certain conditional clauses listed.

During the NRC inspection at Dragon in September 1996, the NRC reviewed specific program implementation areas and identified several instances where activities were not conducted in accordance with NRC and ASME code requirements. These included: (1) failure to provide adequate control of material procurement procedures and to prepare the Unqualified Source Material list; (2) failure to adequately verify supplier qualifications for certain ASME quality system certificate holders; (3) failure to establish and document the basis for chemical sampling plans used for dedicating unqualified material and; (4) failure to perform and document the required actions for a significant condition adverse to quality. However, these specific findings were not against PECO purchases.

b.3 ASTA Engineering, Inc.

The inspectors reviewed the PECO Assessment Report of ASTA Engineering, Inc. (ASTA), dated September 22, 1995, which documented the August 22-23, 1995, assessment of ASTA. PECO had performed the assessment using a performance based approach using the NUPIC Joint Audit Checklist, Revision 6, dated March 26, 1995. The Audit Team Leader and the Technical Specialist were both Gilbert/Commonwealth, Inc., personnel contracted by PECO to perform the assessment and prepare the Assessment Report.

The assessment was performed to address ASTA's QA program and its implementation as applicable to engineering services supplied to PECO. The portion of the assessment covered by the section of the Assessment Report titled "Investigative Summary" included the areas of Order Entry; Design; Software Quality Assurance; Procurement; Material Control, Handling, Storage, and Shipping; Fabrication, Assembly, and Special Processes; Tests and Inspection; Calibration; Document Control; and Program Control. The portion of the assessment covered by the section of the Assessment Report titled "Technical Specialist Summary" discussed the Technical Specialist's review of three ASTA packages for the dedication of molded case circuit breakers, dedication of Potter & Brumfield MDR relays, and the dedication of control transformers.

The PECO assessment concluded that ASTA was implementing an effective program for the services to be supplied to PECO with the exception of two areas identified in Vendor Corrective Action Requests (VCR). The VCRs documented that ASTA had not passed down a requirement (10 CFR Part 21), imposed by customer PO, to a subtier supplier as required by the ASTA QA program and ASTA had not taken actions to approve a calibration facility used in quality activities. PECO provided documentation of the correspondence between PECO and ASTA documenting acceptable closure of both findings. Review of the August 5, 1996, PECO EVL showed ASTA as an approved supplier of engineering consulting services and commercial grade dedication of equipment. There were no conditional clauses listed for ASTA.

b.4 Namco Controls Corporation

The inspectors reviewed the IES Utilities (IES) Audit Report of Namco Controls Corporation (Namco), dated January 25, 1996, which documented the January 8-12, 1996, assessment of Namco. PECO had performed a review and accepted this Audit Report as a basis for placing Namco on the PECO EVL as an approved supplier.

IES had led and performed the audit as a NUPIC Joint Utility Audit with Illinois Power, Tennessee Valley Authority, and IES providing audit team members. The audit was performed to assess Namco's QA program and its implementation in supplying electromechanical limit switches, limit switch receptacle assemblies, plug-in connector assemblies, and cable assemblies. The audit included the areas of Order Entry; Design; Procurement; Material Control, Handling, Storage, and Shipping; Fabrication, Assembly, and Special Processes; Tests and Inspection; Calibration; Document Control; and Program Control.

The audit report concluded that Namco was implementing an effective program for the product to be supplied with the exception of four areas identified in one finding and three observations. The finding indicated that Namco did not have procedural controls which adequately established storage temperature requirements. The observations indicated that Namco did not have documentation to support a referenced activation energy, did not specify certain management personnel in the organization chart, and the Namco QA program did not specify the frequency of internal audits and how resultant findings were to be addressed. PECO provided documentation of the correspondence between IES and Namco documenting acceptable corrective action for the finding and three observations.

PECO had performed a formal review of the IES audit of Namco as documented on PECO Audit Report Review Form (ARRF) dated February 20, 1996. The ARRF indicated that IES had performed an audit of the areas applicable to PECO's planned purchases using the appropriate criteria (10 CFR Part 50, Appendix B, ANSI N45.2 and NQA-1). The ARRF required a review of numerous areas including Lead Auditor certification, audit scope, applicability of QA program, sufficient objective evidence in audit package, audit findings issued and applicability of items on order, in stock, or installed in the plant, and whether corrective actions for findings were adequate for the PECO application. PECO had concluded that the IES audit had been applicable to PECO's purchases, that the findings and observations had been adequately addressed, and PECO had sufficient basis for placement of Namco on the PECO EVL. Review of the August 5, 1996, PECO EVL showed Namco as an approved supplier of nuclear limit switches and spare and replacement parts, with no conditional clauses listed.

b.5 Sorrento Electronics

The inspectors reviewed the Wolf Creek Nuclear Operating Corporation (Wolf Creek) Audit Report of Sorrento Electronics, Inc. (Sorrento), dated July 7, 1994, which documented the June 20-24, 1994, audit of Sorrento. PECO had performed a review and accepted this Audit Report as a basis for placing Sorrento on the PECO EVL as an approved supplier.

Wolf Creek had led and performed the audit as a NUPIC Joint Utility Audit with APS, TUE, WCN, and PSE providing audit team members. The audit was performed to assess Sorrento's QA program and its implementation in supplying radiation monitoring devices. The audit included the areas of Order Entry; Design; Software Quality Assurance; Procurement; Material Control, Handling, Storage, and Shipping; Fabrication, Assembly, and Special Processes; Tests and Inspection; Calibration; Document Control; and Program Compliance.

The audit concluded that Sorrento was implementing an effective program for the product to be supplied with the exception of the four findings identified. The findings included not including functional testing as a critical characteristic of an active electrical component; not assuring that items and services conformed to the procurement documents; inconsistencies in the software quality assurance program; and a measurement gage being calibrated at longer intervals than specified in the applicable QA procedure. PECO provided documentation of the correspondence between Wolf Creek and Sorrento documenting acceptable corrective action for the findings.

PECO had performed a formal review of the Wolf Creek audit of Sorrento as documented on PECO Audit Report Review Form (ARRF) dated October 21, 1994. The ARRF indicated that Wolf Creek had performed an audit of the areas applicable to PECO's planned purchases using appropriate criteria (10 CFR Part 50, Appendix B, and 10 CFR Part 21). The ARRF required a review of numerous areas including Lead Auditor certification, audit scope, applicability of QA program, sufficient objective evidence in audit package, audit findings issued and applicability of items on order, in stock, or installed in the plant, and whether corrective actions for findings are adequate for the PECO application. PECO had concluded that the Wolf Creek audit was had been applicable to PECO's purchases, that the findings had been adequately addressed, and PECO had sufficient basis for placement of Sorrento on the PECO EVL.

Review of the August 5, 1996, PECO EVL showed Sorrento as a conditionally approved supplier of nuclear radiation monitoring systems and spare and replacement parts. A conditional clause in the EVL specified four requirements to included on PECO POs to Sorrento: (1) Specific revision of Sorrento QA manual to be applied, (2) Commercial grade items were to be dedicated by a specific Sorrento procedure, (3) All material and documentation was to be shipped from San Diego Manufacturing facility, and (4) a PECO surveillance was required for Full Assembly Radiation Monitor POs. The inspectors reviewed a recent purchase order to Sorrento, PO No. LS 607244, dated March 13, 1996, for a General Atomic Processor component and determined that it contained the four conditional clauses as required by the PECO EVL.

c. Conclusions

c.1 ACCUTECH

The inspectors concluded that the NQA third party review of the NUPIC audit report did not adequately address the commercial grade dedication sampling issues described in the NUPIC audit report for applicability to PECO procurement requirements and did not question the basis for verification of lot homogeneity for

finished fasteners purchased from non-approved suppliers as described in the audit report. PECO did not identify, in their formal AARF review conducted June 27, 1996, that ACCUTECH's sampling process needed improvement for them to be maintained on the PECO EVL as an approved safety-related supplier. The inspectors also concluded that PECO reviewed and accepted the NUPIC audit report without adequately addressing the NRC Notice of Nonconformance and follow-up issues identified during a December 1994 Vendor Inspection Branch inspection at Cardinal Industrial Products (previous name of ACCUTECH). This issue was identified as a program weakness.

c.2 Dragon Valves

The inspectors concluded, based on the documentation reviewed, that PECO had taken adequate actions to place Dragon on the PECO EVL as a conditionally approved supplier of ASME code and safety-related non-code valves, parts, and components and had adequately documented these actions. Also, the inspectors verified during review of PECO POs to Dragon, that PECO had placed the appropriate conditional clause requirements in the POs.

c.3 ASTA Engineering, Inc.

The inspectors concluded, based on the documentation reviewed, that PECO had taken adequate actions to place ASTA on the PECO EVL as an approved supplier of engineering consulting and commercial grade dedication services and had adequately documented these actions.

c.4 Namco Controls Corporation

The inspectors concluded, based on the documentation reviewed, that PECO had taken adequate actions to place Namco on the PECO EVL as an approved supplier of nuclear limit switches and spare and replacement parts and had adequately documented these actions.

c.5 Sorrento Electronics

The inspectors concluded, based on the documentation reviewed, that PECO had taken adequate actions to place Sorrento on the PECO EVL as an approved supplier of nuclear radiation monitoring systems and spare and replacement parts and had adequately documented these actions.

E7.4 Requisition of Items From Other Utilities

a. Inspection Scope

The inspectors reviewed PECO procedure P-C-1, "Material Requisition Process," Revision 3, dated August 27, 1995, and representative documentation to determine the process used to requisition safety-related material from other utilities.

p. Observations and Findings

P-C-1 established purchase classification requirements and stated in Section 4.16 that a Purchase Class 1 item is a safety-related item that shall be procured from a vendor listed as approved or conditionally approved on the EVL or from a licensed commercial nuclear facility. Section 7.3, "Requisition of Items from Other Utilities," describes the documentation and quality requirements necessary for purchasing safety-related items from other Utilities. During the first phase of the inspection PECO stated that they do not audit other nuclear utilities for placement on the EVL, based upon the fact that they consider their QA programs as approved by the NRC by acceptance of Chapter 17 of their FSAR/USAR and done in combination with the quality and documentation requirements invoked BY PECO on the selling utility.

The inspectors determined that: PECO compares the PO used by the original purchasing utility to PECO's requirements; copies of the vendor PO documentation is obtained; the supplying utility is required to certify the item and its documentation was procured, received, and stored in accordance with the quality program and was not modified and; PECO ensures that the PO includes a statement requiring that PECO be notified of any deviations from the purchase requirements prior to shipment.

c. Conclusions

The inspectors concluded that this method was consistent with 10 CFR part 50, Appendix B requirements.

E7.5 Control of Purchased Material

a. Inspection Scope

The inspection scope included a review of PECO's control of purchased material with emphasis on the effectiveness of measures for assuring conformance with procurement document requirements. This phase of the inspection focused on receiving inspection activities for verifying the adequacy and completeness of vendor-supplied documentation. The control of these activities is described in Section 17.2.7, "Control of Purchased Material, Equipment, and Services," of the LGS UFSAR.

As a part of this review, the inspectors selected a sampling of purchase orders (PO) for safety related material that had been processed within the last year. The purchase orders and vendor-supplied documentation available at the plant site were reviewed to assess the effectiveness of this phase of the procurement process.

b. Observations and findings

A representative list of the document packages reviewed and inspector's observations are included in the following paragraphs.

- b.1 PECO PO LS 605919, dated January 1, 1996, to ACCUTECH Division of B&G Manufacturing Co. Inc.**

Item 1 of this PO was for 44, all thread 1-1/4 inch by 10 inches long, ASME SA-193 Grade B7 studs. The PO specified that the material must be manufactured and controlled through manufacturing and supplying (warehouse/delivering) under a quality program which satisfies the requirements of Section NCA-3700/NCA-3800 of Section III of the ASME Code.

The documentation package for this material included ACCUTECH's certification that the material had been manufactured in accordance with their ASME Quality Systems Certificate (QSC). The certification included ladle analysis (apparently transcribed from another document), tensile properties, impact properties, and a description of the heat treatment. The documentation package also included ACCUTECH's certificate of compliance with the applicable ASME Code and 10 CFR Part 50, Appendix B requirements and statements that impact specimens were prepared in accordance with the applicable Code requirements, a satisfactory macroetch test had been performed on this material and that ACCUTECH had conducted a satisfactory visual inspection (report attached). The documentation package also included a copy of PECO's receiving inspection report indicating that complete documentation was provided.

The inspector noted that, since this material was procured in accordance with the requirements of NCA 3800 of the ASME Code, and so certified under ACCUTECH's QSC, additional documentation should have been provided by the Material Organization (ACCUTECH) and maintained at the plant site. Specifically, Paragraph NCA-3862.1(b) states that, when the required chemical analysis (including the melting mill heat analysis), heat treatment, tests, examinations, or repairs are subcontracted, the approved supplier's certification for the operations performed shall be furnished as an identified attachment to the certified material test report (CMTR). From the documentation supplied by ACCUTECH, it could not be readily determined which of the operations described on their CMTR were subcontracted, but it appeared that the melting mill, who is responsible for providing the heat analysis for this material, would fall into that category. ACCUTECH's CMTR did not include the melting mill certification, including heat analysis, as an identified attachment.

Failure to verify conformance with procurement document requirements was identified as an example of Unresolved Item 50-352;50-353/96-201-01.

- b.2 PECO PO LS 607801, dated March 28, 1996, to ACCUTECH.**

Items 1 and 2 of this PO were for ASME SA-193, threaded studs with diameters of 5/8 inch and 1 inch, respectively. Item 3 was for 1/2 inch heavy hex bolts and item 4 was for ASME SA-194, Grade 2H, heavy hex nuts. The PO specified that the material must be manufactured and controlled under a quality program which satisfies the requirements of Section NCA-3700/3800 of Section III of the ASME Code.

For Items 1 through 3, PECO's document packages included ACCUTECH's certification that this material had been manufactured in accordance with their

QSC. The certification included ladle and check analyses, apparently transcribed from another document, tensile properties, and a description of the heat treatment. PECO's documentation also included ACCUTECH's certificate of compliance with the applicable PO, ASME Code and 10 CFR 50, Appendix B requirements, a statement that macroetch test results were acceptable, and a statement that ACCUTECH had conducted a satisfactory visual inspection (report attached). The data package for item 4 included similar information, except that proof load and hardness test results were reported instead of the tensile properties. There was no indication that a macroetch test had been performed on this material as required by the specification.

The inspector noted that, as discussed in Section b.1 above, additional documentation was not provided by the Material Organization and maintained at the plant site. The melting mill certifications, including heat analyses and, for item 4, macroetch certification were not provided with the material.

Failure to verify conformance with procurement document requirements was identified as an example of Unresolved Item 50-352;50-353/96-201-01.

b.3 PECO PO LS 157145, dated November 26, 1996, to Energy & Process Corporation (E&P).

Item 1 of this PO was for four 3 inch, NPS, butt weld, schedule 40, seamless SA-234, Grade WPB elbows, to be provided in accordance with ASME Code, Section III, Class 2.

The document package for this material included E&P Material Certification which certified compliance with the ASME Code and PO requirements and noted that the material was supplied in accordance with E&P's ASME QSC. Included with this certification was a Material Test Report from Ladish Company, dated March 23, 1994, containing chemical analyses and tensile properties of the elbows and noting that the starting material was seamless pipe. Also included with the certification was a Magnetic Particle Inspection Report from Gramlich Inspection Services.

Item 2 of this PO was for four 3/4 inch NPS, forged stainless steel, SA-182, Grade 316L couplings, to be supplied in accordance with ASME Code, Section III, Class 1.

In addition to E&P Material Certification, the documentation package included a Test Report Certification from Alloy Stainless Products Co., dated October 7, 1992, and a Liquid Penetrant Report from Gramlich Inspection Services.

Item 3 of this PO was for two 1 inch NPS, forged SA 105 couplings, to be supplied in accordance with ASME Section III, Class 2.

In addition to E&P Material Certification, the documentation package included a Certificate of Analysis from Capitol Manufacturing Co., dated September 2, 1992, containing chemical analyses and tensile and hardness test results.

Item 4 of this PO was for two 1 inch NPS, SA-105 tees, to be supplied in accordance with ASME Code, Section III, Class 2.

In addition to E&P Material Certification, the document package included a CMTR from Bonney Forge, dated September 30, 1981, containing chemical and tensile test results and certifying that the material was supplied in accordance with their ASME QSC, which expires on March 30, 1982. The certification stated that the material property data was either copied from material records furnished by the production mill or obtained from laboratory checks.

The inspector noted that, as discussed in Section b.1, above, additional documentation was not provided by E&P and maintained at the plant site. For Items 1 through 4, the material producing mill certifications, including heat analyses were not provided with the material.

Failure to verify conformance with procurement document requirements was identified as an example of Unresolved Item 50-352;50-353/96-201-01.

b.4 PECO PO LS 158648, dated February 7, 1997, to Nova Machine Products Corporation (Nova).

Item 1 of this PO was for ten 1 1/8 inch ASTM A-325, Type 1, heavy hex bolts, to be supplied in accordance with the applicable provisions of 10 CFR Part 50, Appendix B. The PO also required that a CMTR be provided, including the results of all required chemical analyses, tests, and examinations.

The document package for this material included Nova Certificate of Compliance which identified the material by heat number, described the heat treatment, and provided a quality program statement attesting compliance with the PO requirements and referencing Nova QA manual and their ISO-9001 Certificate. Attached to the Nova certification was a CMTR issued by Lake Erie Screw Corporation. This CMTR identified the steel producing mill and contained chemical analysis and mechanical properties of the material. It also provided heat identification, described the head markings, and certified that the material was produced in accordance with a QA program that had been surveyed and approved by Nova.

b.5 PECO PO LS 155212, dated August 28, 1996, to Nova.

Item 1 of this PO was for twenty four 3/8 inch ASTM A-307, Grade A, hex head bolts, to be supplied in accordance with a QA program meeting the applicable provisions of 10 CFR Part 50, Appendix B. A Certificate of Conformance was required to be provided for this material.

The document package for this material included a Nova Certificate of Compliance describing the material and attesting to compliance with the PO and the applicable specifications. It also included a quality program statement, certifying to compliance with the applicable portions of the Nova QA manual and stating that the material has been processed per ISO-9001(94), Certificate # GQC 211.

c. Conclusions

A review of documentation packages for material purchased to the ASME Code requirements identified several examples of apparent failure to assure that the Material Organizations supplying this material provided all of the Code-required documentation. Specifically, Paragraph NCA-3861(b) of Section III of the ASME Code requires the Material Organization to transmit all certifications received from other Material Organizations or approved suppliers to the purchaser at the time of shipment. Paragraph NCA-3862.1(b) states that, when the required chemical analyses, tests, examinations, heat treatment, etc., are subcontracted, the approved suppliers certification for the operations performed shall be furnished as an identified attachment to the CMTR. As discussed in the above examples, in several instances the approved supplier certifications were not furnished with the CMTR and were not included in the document package. This issue was identified as Unresolved Item 50-352;50-353/96-201-01.

Document packages for safety-related material purchased for non-Code applications were also reviewed. Such material was procured to 10 CFR Part 50, Appendix B requirements and was ordered to different quality levels, apparently determined by the service application or (for replacement material), the original equipment or design specification. Documentation requirements for these orders were defined by specific purchase clauses which are a part of the PO. It was noted that some material orders required only the supplier's certificate of conformance, while others required traceable CMTRs including the actual results of all required chemical analyses and examinations.

E7.4 Review of Procurement from Amer Industrial Technologies, Inc. (AIT)

a. Inspection Scope

The inspectors reviewed the April 26 through 30 and May 5, 1993, PECO triennial audit of AIT. The audit was performed using the NUPIC Audit checklist, Revision 4, for review of the supply of piping subassemblies, pressure vessels as nuclear service components and pipe shop fabrication. The PECO audit did not identify any audit findings and concluded that the AIT quality program for the scope of supply was adequate and being effectively implemented. An inspection conducted by the NRC's Special Inspection Branch in January 1996 (Report No. 99901292/96-01) identified numerous inspection findings in several areas of program implementation and came to significantly different conclusion on effectiveness of program than the PECO audit.

The purpose of the inspector's review was to determine why such a difference in the results of the PECO audit and the 1996 NRC inspection existed and to review PECO's evaluation of the NRC inspection findings for relevance to purchased material and what (if any) compensatory measures were taken by PECO in 1996 as a result of this evaluation.

p. Observations and findings

The inspectors reviewed the 1993 PECO audit and material supply history of AIT with PECO and determined that PECO had not identified problems with AIT supplied material received to date. The inspectors determined that AIT was downgraded from approved to bidder status (not qualified for nuclear purchases) on the PECO EVL based upon the negative results of an ASME survey conducted at AIT on June 26-28, 1995. When PECO became aware of the NRC inspection findings at AIT, they initiated different actions to determine: (1) how the findings affected delivered items and components such as the Limerick RHR heat exchanger, (2) a review of the NRC inspection report and the effectiveness of the PECO supplier audit process, and (3) an engineering evaluation of the NRC ASME findings for their effect on the operability of the Limerick RHR heat exchanger.

As a result of these actions PECO issued a Nonconformance report and requested that engineering perform an operability/reportability determination and a review of the ASME code issues identified in the NRC inspection report for the RHR heat exchangers at Limerick Unit 1. This evaluation determined that the RHR heat exchangers were acceptable for current and continued use for all modes of operation at Limerick Unit 1. This conclusion was based on the fact that PECO did not solely rely on the audit results at AIT for vendor qualification, but also subcontracted with UE & C Nuclear, for 18 source surveillance inspections at AIT between May 1993 and February 1994, during various stages of manufacturing and testing. Also, the NRC inspection report was reviewed to develop a lessons learned perspective and training sessions were conducted with all auditors on a lessons learned on vendor audit methodology covering areas such as ASME Code Section III upgrading of material as well as all of the areas where findings were identified with QA program implementation. Finally, on July 31, 1996, AIT was removed from the PECO EVL.

The inspectors also determined that previously, in August 1993, Corrective Action Request (CAR) Q-4320/Quality Evaluation Q0004320 was initiated by PECO to conduct a self-assessment to document and resolve problems associated with the NQA Assessment Section's acceptance of supplier audits. As a result of the activities associated with the CAR, 13 problem audits were identified. The PECO audit of AIT was one of the problem audits identified in the review. Each of the audits identified required either additional actions or compensatory measures to be taken. However, NQA personnel at the time of the audit, accepted these audits for use without imposing any additional actions or compensatory measures. A root cause analysis was performed utilizing the "events and Causal Factors" methodology and resulted in the identification of 14 causal factors. Several corrective measures were identified and implemented by PECO as a result of this CAR process to correct and improve the supplier audit process. PECO concluded that the corrective measures taken, combined with more rigorous management reviews of work activities, would eliminate all of the causal factors identified.

c. Conclusions

Based upon the self-initiated CAR supplier audit measures in 1993, the fact that the 1993 PECO audit of AIT was identified as a problem audit that required either additional actions or compensatory measures to be taken, and the actions taken by

PECO against AIT in 1996 addressing the concerns identified during the ASME survey and the NRC inspection at AIT, the inspectors concluded that PECO initiated appropriate actions to address the supplier audit issues at AIT.

E7.5 Review of Recent 10 CFR Part 21 Activities

a. Inspection Scope

The inspectors reviewed PECO's recent 10 CFR Part 21 activities as related to recent information which B&G Manufacturing Company (B&G) had provided PECO concerning potentially defective (inadequately heat treated) grade B7 fasteners. The fasteners had been originally supplied to PECO by Cardinal Industrial Products (CIP) prior to B&G's purchase of CIP in July of 1995. After the purchase of CIP, B&G had formed B&G-Cardinal and later renamed the company ACCUTECH.

b. Observations and Findings

B&G had informed PECO in a September 12, 1995, letter that CIP had provided PECO with 1-1/2" B7 fasteners from a lot which had subsequently been identified to contain defective product. PECO had initiated two Action Requests A09965059 (Limerick) and A0965060 (Peach Bottom) which recommended the removal of the fasteners from available stock. PECO determined that Limerick had not received any of the material. PECO determined that the fasteners had been issued from the Peach Bottom warehouse, had been used in eight work orders, and installed in several applications. PECO had discussed the properties of the inadequately heat treated fasteners with ACCUTECH, performed an engineering review of the fastener application, and determined that the fasteners were acceptable for the applications.

B&G had informed PECO in a September 29, 1995, letter that Cardinal Industrial Products (CIP) had provided PECO with 3" B7 fasteners from a lot which had subsequently been identified to contain defective product. PECO had initiated two Action Requests A09965045 (Limerick) and A0965046 (Peach Bottom) which recommended the removal of the fasteners from available stock. PECO determined that Limerick had not received any of the material. PECO determined that the fasteners had been issued from the Peach Bottom warehouse and had been installed in a pipe support application. PECO had discussed the properties of the inadequately heat treated fasteners with ACCUTECH, performed an engineering review of the fastener application, and determined that the fasteners were acceptable for the application.

Atwood and Morrill Co. Inc.(A&M), informed PECO in a April 12, 1996, letter that potentially defective B7 fasteners, manufactured by CIP, had been used in valves supplied by A&M to PECO. A&M indicated that it did not believe that the fasteners would cause a substantial safety hazard but recommended that the fasteners be replaced at a convenient time. PECO had initiated two Action Requests A1023118 (Limerick) and A1023119 (Peach Bottom). PECO determined that Peach Bottom had not received any of the material. PECO determined that the valves provided by A&M had been installed in a service water application at Limerick.

Nonconformance Report (NCR) LG 96-01975 was initiated, which documented the PECO operability determination that the valves were operable as is but that the fasteners should be replaced during a system outage. The NCR documented that all fastener replacements had been accomplished by January of 1997.

c. Conclusions

The inspectors concluded that PECO had taken adequate actions to review the supplied information concerning potentially defective fasteners, had performed the appropriate review of the suitability for application, had taken adequate corrective actions, and had adequately documented these actions.

V. Management Meetings

X1 Exit Meeting Summary

On March 14, 1997, the inspection team conducted an exit meeting with members of the PECO staff and management at PECO Nuclear offices. During the exit meeting the team summarized the inspection findings and observations.

PARTIAL LIST OF PERSONS CONTACTED

LICENSEE

D. Feters, Vice President - Station Support
T. Niessen, Director, Nuclear Quality Assurance
J. Cotton, Director, Engineering
H. Birch, Manager, Supply Management
W. Texter, Manager, Corporate Nuclear Quality Assurance
T. Baxter, Nuclear Quality Assurance
K. Borton, Licensing Section, Station Support Department
W. Bradley, Nuclear Quality Assurance
W. Strickland, Manager, Materials Management
C. Kembring, Procurement Supervisor
D. Schmidt, Engineer
J. Joneja, Engineer

NRC

G. Cwalina, Chief, Vendor Inspection Section, Special Inspection Branch
F. Rinaldi, Project Manager
J. Bednar, Foreign Assignee

APPENDIX A

Open Items

This report categorizes the inspection findings as unresolved items and inspection follow-up items in accordance with the NRC Inspection Manual, Manual Chapter 0610. An unresolved item (URI) is a matter about which more information is required to determine whether the issue in question is an acceptable item, a deviation, a nonconformance, or a violation. The NRC Region I office will issue any enforcement action resulting from their review of the identified unresolved items. An inspection follow-up item (IFI) is a matter that requires further inspection because of a potential problem, because specific licensee or NRC action is pending, or because additional information is needed that was not available at the time of the inspection.

<u>Item Number</u>	<u>Finding Type</u>	<u>Title</u>
50-352;50-353/96-201-01	URI	Failure to meet ASME NCA-3861 and 3862 (Section E7.5 - b.1, b.2, and b.3)

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

August 14, 1997

Mr. Jerry Ethridge, Project Manager
Tritium Target Qualification Program
Battelle Boulevard
P.O. Box 999
Richland, Washington 99352

Subject: NRC INSPECTION REPORT 99900541/97-01

Dear Mr. Ethridge:

On July 18, 1997 the U.S. Nuclear Regulatory Commission (NRC) completed an inspection of the Pacific Northwest National Laboratory (PNNL) activities relating to the Tritium Target Qualification Program. The enclosed report presents the results of the inspection.

During this inspection, the NRC inspectors found several instances where the implementation of your quality assurance program failed to meet certain NRC requirements. The specific instances are described in the enclosed report. However, as described in the report, the team noted that PNNL took action to correct the identified deficiencies prior to completion of the inspection. Therefore, a response to this letter is not necessary.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in NRC's Public Document Room.

Sincerely,

A handwritten signature in dark ink, appearing to read "Stuart A. Richards", is written over a horizontal line.

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No.: 99900541

Enclosure: Inspection Report 99900541/97-01

cc:

Max Clausen
Office of Commercial Light-Water
Reactor Production
Tritium Project Office
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

Stephen M. Sohinki, Director
Office of Commercial Light-Water
Reactor Production
Defense Programs
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

DP-60 Records Management
Office of Commercial Light-Water
Reactor Production
Tritium Project Office
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Mr. Paul L. Pace, Manager
Licensing and Industry Affairs
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, Tennessee 37381

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth Street, SW, Suite 23185
Atlanta, GA 30303-3415

County Executive
Rhea County Courthouse
Dayton, TN 37321

County Executive
Meigs County Courthouse
Decatur, TN 37322

Mr. Michael H. Mobley, Director
Division of Radiological Health
3rd Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

**U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION**

Report No: 99900541/97-01
Project 697

Organization: Pacific Northwest National Laboratory
Richland, Washington

Contact: Jerry Ethridge, Project Manager
Tritium Target Qualification Project
(509) 372-4991

Nuclear Activity: Fabrication of the Tritium Producing Burnable Absorber
Rods for the Tritium Target Qualification Project, Lead
Test Assemblies

Dates: April 29 - May 2, 1997
July 7-11, 1997
July 14-18, 1997

Inspectors: Steven M. Matthews, Team Leader, VIS/PSIB/DISP
Robert M. Latta
Kenneth C. Heck

Approved by: Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs

Robert A. Gramm, Chief
Quality Assurance Section
Quality Assurance and Maintenance Branch
Division of Reactor Controls and Human Factors

1 INSPECTION SUMMARY

During this inspection, the inspectors reviewed the implementation of selected portions of Pacific Northwest National Laboratory's (PNNL's) quality assurance (QA) program for supplying Tritium Producing Burnable Absorber Rods (TPBARs) for the Tritium Target Qualification Project (TTQP), Lead Test Assemblies (LTAs). The inspection was focused on the review of PNNL activities related to the design and manufacture of the TPBARs for their subsequent use in Westinghouse designed burnable poison rod assemblies (BPRAs).

The inspection bases were:

- Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50)
- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50
- 10 CFR Part 21, "Reporting of Defects and Noncompliance"
- Tennessee Valley Authority (TVA) Nuclear Quality Assurance (NQA) Plan (TVA-NQA-PLN89-A), Revision 6, dated August 31, 1995.
- PNNL's TTQP, "Quality Assurance Plan," ETD-003, Revision 4, dated July 1997

During this inspection, several nonconformances to NRC requirements were identified. However, PNNL's corrective actions, taken prior to the end of the inspection, resulted in the team closing the nonconformances. No issues remain open at the time of this writing.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of PNNL activities related to the TPBARs.

3 FINDINGS FROM THIS INSPECTION

3.1 Quality Program

During the weeks of April 29 through May 2, 1997, and July 7 through 11,

1997, the team evaluated the acceptability of the QA provisions established to control the design and fabrication of the TPBAR LTAs. This evaluation involved the review of TTQP project documents that described the quality and technical requirements imposed on PNNL by the host utility (TVA) for these safety-related components. The team also examined the adequacy and implementation of PNNL's quality assurance plan described in EDT-003, "Tritium Target Qualification Project, Quality Assurance Plan," Revision 3, dated July 1997. Sections 3.1.1 through 3.1.6 provide the team's assessment of controls consistent with TPBAR component safety classification; QA program adequacy; and QA program implementation in the areas of audits, corrective actions, training, and design control.

3.1.1 Safety Classification

On December 4, 1996, the Department of Energy (DOE) submitted for the staff's preliminary review PNNL topical report "Topical Report on the Evaluation of Tritium Producing Burnable Absorber Rod Lead Test Assembly" (PNNL-11419/UC-731), dated November 1996. The purpose of this report was to provide technical information related to the anticipated irradiation of TPBARs in a commercial light water reactor. In particular, the report provided a description of the TPBAR design and fabrication requirements, as well as general quality provisions and an evaluation of the safety issues associated with the irradiation of these assemblies in a commercial light water reactor.

Based on NRC's review of the report, a request for additional information (RAI) was forwarded to DOE on January 3, 1997. DOE responses to the RAI were provided in letters dated January 21 and February 14, 1997. Both responses asserted that the TPBARs did not perform a safety-related function and were, therefore, considered to be non-safety related. The topical report, however, indicated that PNNL would "voluntarily" comply with 10 CFR Part 21 provisions and would apply the PNNL QA program to those items which were considered to meet the requirements of 10 CFR 50, Appendix B. By letter dated February 13, 1997, the staff conveyed to DOE its position that the TPBARs were part of a basic component and that, as such, were subject to compliance with the provisions of 10 CFR Part 21 and the quality assurance requirements of 10 CFR Part 50, Appendix B.

In response to the staff's position regarding the safety classification of the TPBARs, PNNL forwarded a revised response to the staff's RAI on March 7, 1997, acknowledging that the design and fabrication of TPBARs would be accomplished under a quality assurance program that complies with the requirements of 10 CFR 50, Appendix B. However, the initial inspection of

PNNL, conducted from April 29 through May 2, 1997, found no apparent connection between the safety classifications described in TVA's Nuclear Quality Assurance Plan, TVA-NQA-PLN89-A, and PNNLs "importance factors" described in procedure TTQP-1-046, Revision 0.

Subsequent to a June 4, 1997, public meeting between the NRC and TVA, the staff provided further amplification on the specific safety function of the TPBARs in a letter to Mr. O.D. Kingsley (TVA) from Mr. F.J. Hebdon dated, June 14, 1997. This letter underscored the NRC's position that the fuel and control rod assemblies were required to be considered basic components subject to 10 CFR Part 21 that, by definition, were required to be designed and manufactured under a quality assurance program that complied with the requirements of 10 CFR Part 50, Appendix B, and that parts thereof (e.g., burnable poison rods and TPBARs) were regarded similarly because of their safety function. The letter further stated that the NRC has always considered burnable poison rods in their entirety to be safety-related and that, as such, this position includes the TPBARs in their entirety (end plugs, getter, cladding, plenum spring, etc.).

a. Inspection Scope

To evaluate the acceptability of PNNL's safety classification process for TPBAR components, the team reviewed PNNL's controlling procedure TTQP-1-046, "Tritium Target Qualification Project, TPBAR Component Characteristics and Related Importance Factors," Revision 3, dated July, 1997. The following paragraphs summarize the results of this review.

b. Observations and Findings

As determined by the team, procedure TTQP-1-046, Revision 3 had been revised to comply with the NRC's position that the TPBAR components were safety-related; TPBAR components were listed with corresponding safety functions, and controlling critical characteristics. Specifically, the TPBAR critical characteristics were defined as those important design, material and performance characteristics necessary to provide reasonable assurance that the item will perform its intended safety function. Table 1 of TTQP-1-046 designates those TPBAR components and critical characteristics, as either Category A or B.

As defined in TTQP-1-046, Category A characteristics are those that could affect the ability of the lead test assemblies (LTAs) to perform their safety function of maintaining the core in a safe condition. Category B

characteristics are defined as those that could (1) significantly affect the mechanical integrity of the TPBAR, or (2) result in incremental tritium releases and either onsite or offsite doses, or (3) result in localized core power peaking. During the review of TTQP-1-046, the team noted that the designated inspection criteria for Category A and B components appeared to be consistent with their relative importance to safety.

c. Conclusion

Based on subsequent reviews related to this area, the team determined that procedure TTQP-1-046 provided an adequate basis for controlling the design, procurement, fabrication, assembly and handling of the TPBAR LTAs and that appropriate provisions for the component safety classification had been implemented.

3.1.2 Quality Assurance Plan

PNNL's project quality assurance program is described in procedure ETD-003, "Tritium Target Qualification Project, Quality Assurance Plan," Revision 3. This TTQP project QA plan encompasses all quality activities related to the TPBARs, including design, procurement, process development, fabrication, inspection, testing, verification and assessment.

a. Inspection Scope

The team evaluated the adequacy of PNNL's quality assurance program with respect Appendix B to 10 CFR Part 50 requirements.

b. Observations and Findings

NRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," Revision 3, dated August, 1985 conditionally endorses ANSI/ASME NQA-1-1983 Edition, "Quality Assurance Program Requirements for Nuclear Facilities," as an adequate basis for complying with the quality assurance requirements of Appendix B to 10 CFR 50. The additional conditions that are imposed by the Reg Guide 1.28, Revision 3, involve qualification of inspection and test personnel, quality assurance records, and audits.

The team reviewed procedure ETD-003, which is based on ANSI/ASME NQA-1-1989, and the referenced documents which implement the TTQP project QA program. The team also examined PNNL's QA procedures with respect to the conditions imposed by Reg Guide 1.28. Additionally, the

team performed a comparison of the programmatic requirements contained in ANSI/ASME NQA-1-1983 versus those of ANSI/ASME NQA-1-1989 in order to determine if PNNL's quality assurance program appropriately addressed the Regulatory Positions contained in Reg Guide 1.28.

c. Conclusions

Based on the team's review of ETD-003, Rev 3, and the implementing QA program procedures, it was determined that PNNL's TTQP project QA program adequately addressed the requirements of 10 CFR Part 21 and Appendix B of 10 CFR Part 50.

3.1.3 Internal Audit Program

a. Inspection Scope

The team reviewed selected internal assessments and surveillances of PNNL activities controlled by the quality assurance program.

b. Observations and Findings

Criterion XVIII, "Audits," of Appendix B to 10 CFR 50 requires that a comprehensive system of planned and periodic audits be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. Contrary to this requirement, the team determined that not all aspects of the quality assurance program had been audited. Although, this deficiency had been previously identified during an internal PNNL assessment conducted in May 1996, the team concluded that adequate corrective actions had not been implemented in that internal audits had not been performed during fiscal year 1997 and none had been scheduled until after project completion.

Subsequent to the team's identification of this issue, PNNL promptly initiated corrective action report (CAR) 97-010. This CAR addressed this deficiency and, as documented by CAR 97-010, PNNL attributed the root cause to its reliance on internal assessments, surveillances, and on external audits conducted by Westinghouse and TVA to sufficiently evaluate QA program elements.

The team reviewed PNNL's corrective actions taken to resolve this deficiency and to prevent recurrence. This review included evaluating the following documents.

- (1) Section 18 "Audits" of the QA Plan (ETD-003) has been revised (Revision 4, July 1997) to include:
 - Management participation in audit scheduling, review of audit findings, and corrective actions taken to resolve them.
 - Responsibility for scheduling annual audits and assuring that they are performed and reported to management has been assigned to the project lead quality engineer.
- (2) Internal audit procedure TTQP-7-048 has been issued which describes the planning, scheduling, preparing, performing, and reporting of internal audits. The method for reporting audit-identified deficiencies, follow up action, and re-audit of deficient areas is also described in this procedure.
- (3) A fiscal year annual audit schedule has been issued, which assures that all aspects of the quality assurance program are audited. Additionally, audits of all activities, with the exception of Organization (Criterion I), QA Records (Criterion XVII), and Audits (Criterion XVIII) will be conducted prior to shipment of the TPBARs to the Watts Bar nuclear plant. Audits of the three excepted areas are scheduled for August 1997.
- (4) An annual audit schedule for fiscal year 1998 has also been issued.
- (5) An evaluation of the impact which the lack of formal audits may have had on the project has been performed. As determined by the team, this evaluation takes into consideration the conduct of internal assessments, surveillances, and external audits performed in fiscal year 1997. The evaluation concluded that there were no direct adverse impacts on the project as a result of not having performed internal audits because the subject areas had been alternatively evaluated.

c. Conclusions

Based on review of the above documentation, the team concluded that appropriate corrective actions had been implemented in response to this nonconformance. The team, therefore, closed this nonconformance.

3.1.4 Corrective Action Program

a. Inspection Scope

In order to evaluate the adequacy of PNNL's corrective action program, the team reviewed the status of actions taken to correct identified deficiencies. In particular, the team examined deficiencies that had been identified during an internal assessment conducted on May 1-9, 1996, deficiencies identified by an external audit conducted by Westinghouse on November 18-21, 1996, with a follow up audit conducted on June 2-5, 1997, and other deficiencies identified by an external audit conducted by TVA on March 24-27, 1997.

b. Observations and Findings

The internal assessment report dated May 14, 1996, indicated that PNNL's corrective action programs were not being used because the tracking system employed by the QA department was cumbersome.

The Westinghouse audit conducted in November 1996 found that identified problems were being tracked by a TTQP project-specific action tracking system, but that the tracking system was not described by the QA program. Additionally, the Westinghouse follow up audit determined that the corrective actions taken in response to this audit finding had not been effective.

The TVA audit which was performed in March 1997 also found that the corrective action program was not being effectively implemented. Nonconformance reports and deviation reports were not being used. Consequently, trending, root cause, and extent of condition evaluations were not being performed.

The team reviewed the actions that had been taken to correct the identified conditions and to implement an effective corrective action program. As a result of this review, it was determined that the following project procedures had been issued:

- TTQP-7-045, "Corrective Procedure," Revision 0, dated July 1997.
- TTQP-7-037, "Corrective Action Processing Procedure for Deficiency Reports and Corrective Action Reports for the Tritium Target Qualification Project," Revision 2, dated July 1997.

TTQP-7-040, "Nonconformance Reports" Revision 0, dated May 1997.

The team reviewed a recently developed TTQP project corrective action tracking system printout. Based on the review of this printout the team found that the actions responsive to TVA's audit finding on the corrective action program were being tracked under item number CAR-97-001. However, the status of TVA audit findings at the time of the inspection remained open, pending verification activities by TVA, as documented in a letter from the TVA Project Manager to the TTQP Project Manager, dated July 3, 1997.

The team reviewed the corrective actions procedures identified above, which had been revised or issued after the TVA audit. These procedures were determined to contain appropriate requirements, such as provisions for trending, root cause determination and extent of condition evaluation. The team also determined that the procedures prescribed appropriate requirements and responsibilities for identifying, documenting, tracking, evaluating and correcting deviations from established quality assurance requirements and program controls.

Additionally, the team reviewed the corrective action tracking status report and determined that deficiencies were being reported for all active project tasks, including material suppliers, design, and fabrication. Based on an examination of the issue dates for reported deficiencies, the team concluded that organizations performing active project tasks were using the system effectively.

c. Conclusions

Based on evaluation of the TTQP corrective action program, the team determined that it was generally acceptable. However, the team noted that PNNL was not taking full advantage of the trending program, since a trend analysis had not been performed.

3.1.5 Training

a. Inspection Scope

In order to determine the adequacy of the TTQP training program, the team reviewed the governing procedure TTQP-7-011, "Training and Qualification Plan for the TTQP-7-011 Project," Revision 1, dated February 1997.

Additionally, the team reviewed a surveillance conducted on training for fabrication personnel and examined selected training records.

b. Observations and Findings

A Westinghouse audit conducted in November 1996 reported weaknesses in maintaining training records for project personnel. To strengthen this area, responsibility for training had been transferred to the TTQP project office.

To verify the accessibility and retrievability of training records, the team requested the training records following issuance of TPQP-7-037 on June 25, 1997. The project office provided a listing of personnel and dates on which training had been completed. Based on a comparison of this list with a listing of TTQP personnel for June 1997, the team determined that training records were being adequately maintained.

The team examined surveillance report SR-2-05, which reviewed the training requirements for personnel in the fabrication facility. The surveillance report found that fabrication personnel had met all training requirements and were qualified to perform associated fabrication activities. The team also noted, during their inspection of the fabrication facility, that a matrix of training requirements and the current training status for fabrication personnel was posted prominently at the entrance.

c. Conclusion

Team inspection results confirmed that fabrication personnel satisfied the training requirements for job performance and that appropriate training records were maintained and retrievable.

3.1.6 Task 1 - Design

a. Inspection Scope

In order to confirm the adequacy of methods used by PNNL, TVA, and Westinghouse to exchange design information, the team reviewed current Design Interface Agreements, which were determined to appropriately designate single points of contacts for information used to transmit and develop formal design outputs.

b. Observations and Findings

The team reviewed the design verification process with the Task 1 Manager and a TVA project representative. Additionally, the team examined pertinent audits and assessments of design activities that had been conducted internally by PNNL (#97-01) and externally by Westinghouse and TVA. These audit and assessment reports were reviewed for content relative to specific findings and to determine the adequacy of the corrective actions related to design activities.

Task 1 procedures reviewed included:

- TTQP-1-017, "Design Analysis/Calculation and Associated Independent Review", Revision 1, dated June 1997
- TTQP-1-019, "Design Change Control", Revision 0, dated January 1997
- TTQP-1-021, "Design Interface Controls Process", Revision 1, dated June 1997
- TTQP-1-022, "Design Requirements", Revision 1, dated June 1997
- TTQP-1-058, "Design Change Impacts on Technical and Functional Requirements", Revision 0, dated March 1997
- PNNL-TTQP-1-580, "Functional Requirements for the TPBAR", Revision 2, dated June 1997

The team also examined TVA's letter to PNNL concerning, "Verification of TVA Plant Specific Design Information," dated May 27, 1997, and TVA's letter to PNNL concerning, "Westinghouse Changes from Cycle 2 Analysis," dated June 19, 1997.

Additionally, the team reviewed selected calculations related to the design and procurement of TPBAR LTA components, associated with the TPBAR cladding and end plugs in order to confirm the appropriate implementation of design control processes.

The team interviewed representatives from PNNL's Task 1 design organization in order to gain insights into the TPBAR design verification process. As a result of these discussions, it was ascertained that all PNNL design inputs and engineering documents have been reviewed for approval

by a Design Review Board (DRB) consisting of representatives from PNNL, TVA, and Westinghouse. The DRB review process, which was completed at the time of the inspection, concluded that an adequate design basis had been established for the TPBAR LTAs to be installed in the Cycle 2 core of the Watts Bar Unit 1 nuclear plant.

As determined from interviews, a design report was being prepared and was scheduled to be reviewed for adequacy by a design review team prior to the Cycle 2 reload. Subsequent to the approval of the design report, any future changes will be reviewed for adequacy and impact on the safety and operation of the Cycle 2 core by TVA.

The team reviewed the current status of the observations and findings that resulted from internal assessments and external audits of the design control program. As determined by the team, all corrective actions had been appropriately tracked and no significant design-related corrective actions remained open.

The team examined eleven calculation files selected from a list of engineering calculations related to the design and fabrication of TPBAR cladding and end plugs. As a result of this review effort, it was determined that all design input information had been properly verified and the team did not identify any deviations from either the administrative design requirements or the technical/functional requirements.

c. Conclusions

Based on the team's reviews of design control documents, audits and assessments, and interviews with engineering management personnel, it was determined that an adequate design control process had been established for the design of the TPBAR LTAs.

3.2 Procurement Activities

a. Inspection Scope

The team evaluated procurement activities for selected critical component parts of the TPBAR lead test assemblies (LTAs) to determine whether applicable regulations were imposed, material specifications were met, and procedures followed.

b. Observations and Findings

PNNL's procurement process is defined in Section 4.0, "Procurement," of its Commercial Light Water Reactor (CLWR), Tritium Target Qualification Project (TTQP), Quality Assurance (QA) Plan, documented in ETD-003. To evaluate the acceptability of a component part, the team used PNNL's critical characteristics as defined in procedure TTQP-1-046, Revision 3.

In order to evaluate the procurement activities, the team reviewed PNNL's acquisition of the stainless steel material for the TPBAR cladding tubes and end-plugs, and the Lithium Aluminate (LiAlO_2) pellets. The following paragraphs summarize the results of this review.

b.1 316 Stainless Steel Material

PNNL procured the stainless steel bar stock material from Westinghouse Hanford, that had originally procured the material for use in DOE's Fast Flux Test Facility (FFTF). The material was procured to material specification TTQP-1-003, "Specification for 316 Stainless Steel Seamless Cladding Tubes," Revision 1, dated May 1996. TTQP-1-003 complied with ASTM Standard A-771-88, "Standard Specification for Austenitic Stainless Steel Tubing for Breeder Reactor Core Components," that, according to PNNL, reflects the fabrication and technical data gained over two decades of cladding development and procurement for the FFTF.

The team determined that both ASTM A-771 and TTQP-1-003 require double vacuum melted feed stock, and chemistry and inclusion limits on the product. The products to be produced from the 316 bar stock procured from Westinghouse Hanford for PNNL fabrication was TPBAR clad tubing and end-caps.

PNNL verified the adequacy of the stainless steel material in accordance with TTQP-2-001, "Material Verification Procedure for the Tritium Target Qualification Project," Revision 0, dated May 1996. However, on the basis of the team's review of the certification of the starting bar stock material, the team determined that the Material Reverification Record, signed and certified by PNNL, was not complete and that procedure TTQP-2-001 failed to adequately establish requirements for the completion and certification of that document. The team identified this issue as a nonconformance.

Subsequently, PNNL issued TTQP Deficiency Report (DR) 07-066, dated May 1, 1997. The team reviewed PNNL's corrective actions that included revising procedure TTQP-2-001 to address the weaknesses identified and

correctly completing the Material Reverification Record. The team found these corrective actions adequate and closed the nonconformance.

During this review the team also determined that TTQP-1-046 failed to adequately describe the actual sample size PNNL used to confirm the chemistry and inclusions of the stainless steel cladding bar stock or the actual characteristics verified during PNNL's reverification process. In addition the team determined that material specification TTQP-1-004, "Specification for Target Rod End Cap Bar Stock Material," failed to agree with the importance factor sampling plans specified in TTQP-1-046 and that TTQP-1-046 did not adequately address the use of the ASTM standard sampling frequencies, where applicable, that PNNL actually used during its reverification plan.

Subsequently, PNNL issued DR 97-068, dated May 7, 1997. That DR defined the deficiency as follows:

The 316 stainless steel bar stock dedication activity which was performed in the fall of 1996 is inconsistent with the sampling requirements specified in TTQP-1-046. In addition, both the sampling requirements and nomenclature used to describe characteristics in TTQP-1-004, Revision 1, "Specification for Target Rod End Cap Bar Stock Material," are inconsistent with the sampling requirements and nomenclature used in TTQP-1-046.

On the generic basis of this nonconformance, PNNL required all of its material specifications to be compared to TTQP-1-046 to assure agreement in all cases.

The team reviewed the effectiveness of this corrective action and determined that the actions taken by PNNL were adequate to address the specific item issue and the generic implications of this issue. The team therefore, closed this nonconformance.

b.2 Lithium Aluminate Pellets

PNNL procured the Lithium Aluminate pellets from ICI Advanced Ceramics. The pellets were procured in accordance with TTQP-1-009, "Specification for Enriched, Annular LiAlO_2 Pellets," Revision 3, dated April 1997. That specification provided that the seller shall be capable of showing with 95% confidence, at least 95%, 90%, and 75% of the pellets in a lot meet the specifications for the characteristics defined in Table 3, "Classification of

Pellet Characteristics for Sampling Plans." That table listed the confidence/inspection level (95:95, 95:90, or 95:75) for each characteristic of the pellet to be verified. In addition, the specification required ICI to submit to PNNL the sampling plan for review and approval prior to use.

In a memorandum to PNNL dated May 1, 1997, ICI specified the sample sizes associated with the inspection levels as follows:

<u>Inspection Levels:</u>	<u>Sample Size:</u>
95:95	15 pellets
95:90	7 pellets
95:75	3 pellets

According to ICI, the sample sizes were verified by the QC curves found in ANSI/ASQC Z1.9-1993, "Sampling Procedures and Tables for Inspection by Variables for Percent Nonconforming." The team determined that ASQC Z1.9 sample sizes are based on the assumption that the data represents a normal distribution. Therefore, in order for PNNL to use this standard to determine sampling sizes, PNNL would have to show a documentation of ICI's past performance of complying with the critical characteristics as the basis to support PNNL's assumption that the data represents a normal distribution.

On the basis of its review of PNNL's procurement, receiving inspection, and acceptance of the Lithium Aluminate pellets, the team identified the following concerns that constitute a nonconformance:

- (a) PNNL failed to document its basis for the assumption that ICI's data represents a normal distribution and therefore the appropriateness of using the small sample sizes to verify critical characteristics of the pellets.
- (b) PNNL failed to include these sampling sizes in the Inspection/Test Instructions (ITIs) used by the QC inspectors to verify the adequacy and acceptance of the pellets for use in the TPBARs.

PNNL responded to the team's concerns by taking the following corrective actions:

- (a) In a memorandum dated July 18, 1997, PNNL adequately documented its previous procurements from ICI and established its basis for using the small sample sizes to accept the pellets.

- (b) The ITIs for receipt inspection for all pellets for LTA lots received were revised to address the sampling matrix and confidence/inspection levels specified in TTQP-1-009.

The team reviewed the effectiveness of these corrective actions and determined that PNNL's actions taken were adequate to address the issues. The team, therefore, closed this nonconformance.

- c. Conclusions

The team identified concerns with the procurements of the stainless steel material for the TPBAR cladding tubes and end-plugs, and the Lithium Aluminate (LiAlO_2) pellets. These concerns constituted nonconformances. However, PNNL responded with corrective actions that adequately addressed the team's concerns and resulted in the team determining that the nonconformances were closed.

3.3 TPBAR Fabrication Activities

- a. Inspection Scope

During this portion of the inspection, the team evaluated the material control for the TPBAR fabrication, handling and storage of cladding tubes, welding of the upper end-caps, and pencil assembly activities for the TPBAR LTAs to determine whether adequate quality assurance provisions were established and procedures followed. PNNL's fabrication process and quality plan is defined in TTQP-2-013, "Manufacturing and Quality Plan for Tritium Producing Burnable Absorber Rods for the Tennessee Valley Authority Lead Test Assemblies," Revision 1, dated June 1997, and in TTQP-2-014, "Tritium Target Rod Fabricating Process Plan," Revision 1, dated June 1997.

- b. Observation and Findings

To evaluate the acceptability of the fabrication process, the team initially reviewed the established procedure and QA plan for the process and then observed PNNL's activities in performing the process activity. The following sections describe the results of this review.

b.1 Material Control

PNNL's material control process is described in the following:

- TTQP-2-101, "General Receiving for the Tritium Target Fabrication Facility (TTFF)," Revision 0, dated February 1997**
- TTQP-2-102, "Handling of Miscellaneous Components," Revision 0, dated February 1997**
- TTQP-2-105, "Control and Inventory for Received Storage," Revision 0, dated January 1997**
- TTQP-2-106, "Control and Inventory for Accepted Storage," Revision 0, dated January 1997**
- TTQP-2-109, "Control and Inventory for Rejected Storage," Revision 0, dated January 1997**

The team reviewed the established measures for material control contained in TTQP procedures and interim change notice (ICN) 2-101-01, completed July 8, 1997, and determined that the measures established were adequate.

The team reviewed the Component Inventory Ledger and the Transaction Log Sheet for each of the material control cages (received, accepted, and rejected storage cages) and found that adequate controls were in place and that all entries reviewed matched with existing inventory. The team also verified that the colored tags found on many items in the TTFF were appropriately documented and controlled in accordance with established procedures.

b.2 Cladding Tubes

PNNL's process for handling cladding tubes were documented in TTQP-2-103, "Handling of Empty Cladding Tubes," Revision 0, dated February 1997, TTQP-2-104, "Handling of Loaded Cladding Tubes," Revision 0, dated February 1997, and TTQP-2-211, "Inspection/Test Instructions for Inspecting Cladding Tubes," Revision 0, dated March 1997. The team reviewed the established measures for the control of cladding tubes in TTQP-2-103 and -104 and determined that the measures established were adequate.

The team reviewed the Transaction Log Sheet and tags for the empty cladding found in the TTFF and determined that adequate controls were in place and that all documents and tags reviewed matched with existing records.

b.3 Welding

PNNL's process for welding qualification, performance, and inspection were documented in the following:

- TTQP-2-018, "Qualification Plan for End Plug Welding," Revision 1, dated June 1997
- TTQP-2-024, "Radiography Inspection Procedure Qualification Plan," Revision 0, dated June 1997
- TTQP-2-117, "Top and Bottom End Plug Welding," Revision 0, dated June 1997
- TTQP-2-303, "Weld Visual Inspection," Revision 0, dated May 1997
- TTQP-2-310, "Radiography Inspection," Revision 0, dated June 1997

The team reviewed the established measures for welding qualification, performance, and inspection and determined that the measures established were adequate.

The team witnessed the real-time radiography (RTR) of the 12 end-cap welds used to qualify the end-cap welding process and determined that the process was adequately controlled and performed in accordance with established procedures. The team also reviewed the radiographs taken of the 12 end-cap welds which were used by PNNL as additional information regarding the adequacy of the welding process. The radiographs were produced by X-raying the welds in accordance with Westinghouse Hanford Nondestructive Examination Procedure Manual WHC-CM-4-38, Section NDT-RT-4000, "General Radiographic Examination Procedure," Revision 2, dated January 15, 1994. Specifically, Appendix C, "Capsule, Fuel, and Absorber Pin Radiography," Revision 1, dated July 16, 1994, was followed to achieve the radiographs of the 12 end-cap welds using the beam-filtered tangential radiography technique. The team found the radiographs and the processes used to produce them to be adequate.

On July 17, 1997, the team reviewed the completed welding qualification report for the qualification of the end-plug welding qualification. That report was documented in TTQP-2-023, "Qualification Report for End Plug Welding," Revision 0, dated July 1997. The team found the report to be well documented and very thorough. Overall, the team found PNNL's welding qualification package to be excellent.

On July 18, 1997, the team witnessed actual production welding of the second set of 8 top end-cap welds. The activity was well controlled and the welding processes were performed in accordance with the established procedures.

b.4 Pencil Assembly

PNNL's process for performing the pencil assemblies were documented in TTQP-2-125, "Pencil Assembly Loading," Revision 0, dated May 1997, and TTQP-2-225, "Pencil Assembly Inspection," Revision 0, dated May 1997. The team reviewed the established measures for the control of pencil assembly and inspection and determined that the measures established were adequate.

The team witnessed the pencil assembly process for several pencils and determined that adequate controls were in place and that all documents and tags reviewed matched with existing records. The team noted and commented on the extreme care and precision exhibited by PNNL's staff performing the pencil assembly and inspection activities.

c. Conclusions

For the fabrication activities reviewed, the team found that PNNL performed those activities with great care and attention to detail and that the personnel involved followed established procedures. No adverse findings were identified by the team.

4 ENTRANCE AND EXIT MEETINGS

In the entrance meeting on April 28, 1997, the NRC inspectors discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with PNNL management. In the exit meetings, on July 11 and July 18, 1997, the inspectors discussed their findings and concerns.

PARTIAL LIST OF PERSONS CONTACTED

PNNL

J. Ethridge
G. Sorensen
R. Latorre
S. English
D. Sensor
C. Painter
L. Erickson
S. Bales
D. Rittenhouse
C. Thornhill
R. Guenther

TVA

J. Chardos

DOE

M. Clausen

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

August 10, 1997

Mr. Lew Goetz
President and CEO
SOR, Inc.
14685 West 105th Street
Lenexa, Kansas 66215-5904

SUBJECT: NRC INSPECTION REPORT 99900824/97-01 AND NOTICE OF NONCONFORMANCE

Dear Mr. Goetz:

On June 19, 1997, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the SOR, Inc. (SOR) facility. The enclosed report presents the results of that inspection.

The inspection was conducted to assess the adequacy of the corrective actions SOR took to correct manufacturing defects associated with certain safety-related pressure, vacuum, and temperature switches made by SOR and sold to licensees of nuclear plants. The inspector assessed specific attributes of the SOR quality assurance program and reporting of defects under 10 CFR Part 21. He also assessed licensee monitoring of the quality of SOR switches.

During this inspection, the inspector determined that in 1993 and 1994, SOR applied a heavy coating of epoxy on the insulated lead wires of switches it sold to licensees for safety-related applications. The epoxy hardened the insulation causing it to crack when it was bent during installation. In September 1994, Nebraska Public Power District (for the Cooper Power Station) and Connecticut Yankee Atomic Power Company (for the Haddam Neck Plant) reported cracked insulated switch wires in their plants. SOR issued a Part 21 response and repaired or replaced the affected switches. The NRC inspector concluded that SOR had failed to prescribe procedures or instructions to prevent the epoxy from being applied on the insulation. On this basis, the inspector concluded that SOR's quality assurance program had not met certain NRC requirements imposed upon it by NRC licensees.

This issue is cited in the enclosed Notice of Nonconformance (NON), and the circumstances surrounding it are described in detail in the enclosed report. In a June 19, 1997 letter, SOR reported steps it took to correct the nonconformance and prevent recurrence. No further response is required.

In addition, the inspector observed that during the January 1995 Nuclear Utilities Procurement Issues Committee audit, licensees did not evaluate SOR's action to correct two manufacturing defects that SOR reported to its customers.

L. Goetz

-2-

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC's Public Document Room.

Sincerely,

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No. 99900824

Docket No. 99900912

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99900824/97-01

NOTICE OF NONCONFORMANCE

SOR, Inc.
Lenexa, KS

Docket No.: 99900824

Based on the results of an NRC inspection conducted on June 16 through 19, 1997, it appears that certain of your activities were not conducted in accordance with NRC requirements.

Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions and procedures of a type appropriate to the circumstances, and shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

SOR Nuclear Quality Assurance Manual 8303-100, Section 5.2, "Instructions, Procedures, and Drawings," dated April 14, 1993, requires, in part, that instructions, procedures, and drawings must contain acceptance criteria to ensure compliance with customer quality requirements.

Contrary to these requirements, SOR did not prescribe instructions or procedures to ensure that epoxy was not applied on insulated lead wires of safety-related pressure, vacuum, and temperature switches, or that quality inspectors examine the switches properly. As a result, a heavy coating of epoxy was applied on the insulated wires during manufacture. The epoxy hardened the insulation causing it to crack when it was bent. In September 1994, Nebraska Public Power District (for the Cooper Power Station) and Connecticut Yankee Atomic Power Company (for the Haddam Neck Plant) identified cracked insulated lead wires of switches installed in their plants. The cracked insulation had the potential to reduce the wire insulation resistance or cause a short to ground. On October 14, 1994, SOR issued a 10 CFR Part 21 to inform the NRC and applicable customers of this defect and pertinent corrective action. SOR replaced or repaired the safety-related switches and took appropriate steps to prevent epoxy from being applied on the insulation of lead wires.

In a June 19, 1997, letter to the NRC, SOR reported the steps it took (in 1994) to correct the problem and prevent recurrence. Steps comprised alerting customers about the cracked insulation, repairing defective switches installed in plants, preparing instructions and procedures, and implementing new quality control inspections to prevent the use of epoxy on the insulation. No further response is required (99900824/97-01-01).

Dated at Rockville, Maryland
this 10th day of August 1997

Enclosure 1

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Report No: 99900824/97-01

Organization: SOR, Inc. (SOR)
Lenexa, Kansas

Contact: Colbert O. Turney
Vice President, Quality Assurance
913/888-2630

Nuclear Industry Activity: Pressure, vacuum, and temperature switches

Dates: June 16-19, 1997

Inspector: Anil S. Gautam, Senior Engineer

Approved by: Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs

Enclosure 2

1 INSPECTION SUMMARY

During this inspection, the NRC inspector assessed the adequacy of the actions taken by SOR to correct manufacturing defects associated with certain safety-related pressure, vacuum, and temperature switches (hereafter referred to as switches). The defects included (1) cracked insulation of lead wires for switches, (2) leakage of O-ring seals in switches exposed to radiation and elevated temperatures, and (3) leakage of epoxy seals in switches. The inspector assessed specific attributes of SOR's quality assurance program and reporting of defects under 10 CFR Part 21, and licensees' monitoring of SOR's control of quality.

The inspection bases were as follows:

- 10 CFR Part 21, "Reporting of Defects and Noncompliance"
- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50)
- SOR Nuclear Quality Assurance Manual (QAM) 8303-100, Revision 9, dated April 14, 1993, and associated implementing procedures

During this inspection, the inspector noted one instance in which SOR failed to conform to NRC requirements imposed upon it by NRC licensees. This nonconformance is discussed in Section 3.1 of this report. In addition, the inspector observed that during the January 1995 Nuclear Utilities Procurement Issues Committee (NUPIC) audit, licensees did not evaluate SOR's corrective actions regarding certain manufacturing defects reported by SOR in information notices to customers. Licensees' monitoring of SOR's control of quality is discussed in Section 3.2 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

Open Item 99900912/93-01-06 (Closed)

During a June 1993 NRC inspection of National Technical Systems (NTS), Inc., in Acton, Massachusetts, the inspector assessed qualification testing of SOR pressure switches and observed (1) a pressure leak in one sample during a high-energy-line break (HELB) test, and (2) excessive leakage current in another sample during a dielectric withstand test. The inspector found no documented evaluation by NTS or SOR of the root cause of the test failure nor pertinent corrective action. The inspector considered this an open item.

Following the NTS inspection, SOR gave the NRC documentation regarding the test failures. On the basis of the documents, the inspector determined that the pressure leak was due to a leakage path provided by unsealed mounting bracket screws for the microswitch (switching element) mounted in the switch housing. SOR believed that the screws had not been resealed after the microswitch was readjusted during factory calibration. Failure to reseat the screws allowed the switch diaphragm (seal) to be overpressurized during the test and caused it to leak. SOR stated that since other switches with the

same type of housings did not suffer a similar test failure, the test failure was attributable to a random occurrence, not to an inherent weakness in the design. SOR's corrective action consisted of (1) resealing the microswitch mounting screw threads if the microswitch was readjusted during factory calibration, and (2) applying a primer to the microswitch bracket screws to improve the curing of the thread sealant in stainless steel housings. SOR confirmed that applicable safety-related pressure switches installed in plants were not compromised because the corrective measures had been instituted before production. This issue is closed.

Regarding excessive leakage current in one sample: the inspector observed that the SOR pressure switch test specimens passed the dielectric withstand test at 1500 Vac for 1 minute, except for one sample that experienced about 2 milliamps (mA) of leakage current at 900 Vac. SOR could not find a root cause and believed that the 2 mA leakage current was a random anomaly and not indicative of a common failure mode. SOR's basis was that the other specimens (1) passed the test at 1500 Vac, (2) had adequate insulation resistance at 500 Vdc, and (3) had sufficient margin for service conditions because the switches were rated for 250 Vac and typically energized for 120 Vac or 125 Vdc applications. The inspector determined that on the basis of information provided by SOR, and because leakage current from moisture intrusion would have been higher than 2 mA, the anomaly was satisfactorily addressed. This issue is closed.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Quality Assurance Program

a. Inspection Scope

The inspector examined the adequacy of SOR's Part 21 evaluations, corrective actions, conformance to procurement documents, and self-assessment of performance.

b. Observations and Findings

The inspector observed that SOR's QA program was based on the policies and criteria of 10 CFR Part 50, Appendix B. The QA program staff was comprised of the Quality Assurance Vice President (QAVP) and 2 quality control (QC) inspectors. The QAVP reported directly to the SOR's President/ CEO. The QC inspectors were authorized to stop production of a nonconforming item until the nonconforming conditions were corrected.

The inspector observed that SOR had posted sections of the Federal Register, dated September 19, 1995, concerning the latest changes to 10 CFR Part 21 but had not posted the complete Part 21 regulation, as is required by 10 CFR 21.6. During the inspection, the QAVP posted copies of the complete regulation in appropriate locations. No further concerns were identified.

The inspector assessed SOR's Part 21 reports and corrective actions for manufacturing defects associated with SOR switches during the past 5 years. Defects included cracked lead wire insulation, leaking O-ring seals, and leaking epoxy seals in the switch conduit seal. The inspector's review is summarized below:

(1) Cracked Insulated Lead Wires

Insulated lead wires for the switch enter and exit an epoxy seal in the conduit adapter of the switch housing. The conduit adapter is potted (sealed) with epoxy to keep moisture from entering the switch housing.

In September 1994, Nebraska Public Power District (for the Cooper Power Station) and Connecticut Yankee Atomic Power Company (for the Haddam Neck Plant) notified SOR of eight defective switches that had cracks in the insulated lead wires. SOR determined that the cracks had been caused by SOR's misapplication of the epoxy on the insulation, subsequent hardening of the insulation, and cracking and tearing of the insulation when it was bent.

On October 14, 1994, SOR sent a 10 CFR Part 21 report to the NRC and customers about the cracking of the lead wire insulation in SOR's nuclear-qualified switches (the NRC also issued event notification 27902 on October 14, 1994, to inform licensees that the switches posed a potential risk of failure of safety-related equipment). Subsequently, approximately 11 licensees returned their switches to SOR for repair.

During this inspection, SOR provided a written response (Attachment 1) to the inspector, dated June 19, 1997. SOR stated that it inadvertently applied the epoxy on the insulation "due to poor workmanship" and that "the condition was undetected because SOR quality inspectors did not notice the coating of the epoxy on the (insulated) wires." SOR also told the inspector that it had not prepared instructions to ensure that epoxy was not applied to the wire insulation. The inspector concluded that SOR's failure to prescribe instructions or procedures to ensure that epoxy was not applied on the insulated lead wires of the switches, or that quality inspectors examined the switches properly, as required by Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, constituted Nonconformance 99900824/97-01-01.

The inspector observed that, in 1994, SOR revised its work procedures to preclude application of epoxy on insulated wires, and to reject any insulated wire that may have been covered with epoxy. SOR added shrink tubing to the insulated wires where they entered the conduit adapter epoxy seal to protect the wire insulation during shipping and handling. SOR also recommended not exceeding a minimum bend radius for the insulated wires. The inspector determined that SOR's actions to correct the misapplication of epoxy and prevent recurrence were adequate. No further response is required.

(2) Leakage of O-Ring Seal

The vacuum switch included an O-ring installed in a triangular gland to seal the process air or fluid between the vacuum screw and the piston, and between the vacuum piston and the primary diaphragm. In 1992 and 1993, SOR implemented a new program to qualify the vacuum switches (SOR test report 9058-102), and discovered that the O-ring seal in the vacuum switch was not capable of retaining the required maximum pressure after exposure to high radiation, high temperature, and hydrostatic pressure greater than 150 psi. SOR concluded that leakage between the vacuum screw and the O-ring had occurred during testing.

On April 1, 1993, SOR issued "Information Notice Concerning Vacuum O-Ring Seal in SOR Nuclear-Qualified Vacuum Switches" to applicable customers regarding potential leakage in vacuum switches, and suggested to customers that all switches be replaced if exposed to pressures greater than 150 psi. The affected switches were those designated by a 54N6, 54TA, 52N6, or 52TA in the first section of the model number and JJTTX6, JJTTX7, JJTTX13, or JJTTX14 at the end of the model number.

During this inspection, on June 19, 1997, SOR provided a written response (Attachment 2) to the inspector. SOR stated that it had not discovered the condition described above earlier "because of inadequate engineering testing and analysis of the vacuum switch." On May 20, 1993, SOR took corrective action to eliminate the leak path by (1) welding the vacuum screw to the vacuum piston and (2) replacing the triangular O-ring seal with a face seal of the same material. The face seal was qualified by analysis (SOR test report 9058-102, Section 14, Appendix 4, Analysis 8923-219) to retain a pressure of 750 psi after exposure to radiation and elevated temperatures. SOR reported that it replaced applicable switches sold to licensees. No further concerns were identified.

(3) Leakage of Conduit Seals

The switch lead wires pass through the outer nipple of the conduit seal connector, through the epoxy seal potted in the nipple, and through a glass seal which is soldered inside the nipple. In May 1994, during routine testing, SOR discovered leakage of pressure through the conduit epoxy seal of "NQ" switches. On June 10, 1994, SOR issued "Information Notice Concerning Conduit Seals in SOR Nuclear-Qualified Pressure, Vacuum, and Temperature Switches" to inform the NRC and customers about the potential leak. In the notice, SOR stated that the leak could lead to reduced insulation resistance or loss of function of the switch during or after a HELB. SOR suggested to customers that all switches be returned to the SOR factory for inspection if they were subject to HELB conditions during or after an event, or if subject to conditions in which condensate may form inside the conduit, or if subject to any other conditions in which moisture could penetrate the conduit seal.

During the inspection, on June 19, 1997, SOR provided a written response (Attachment 3) to the inspector. SOR stated that the problem went undetected because (1) the leakage was a random problem, (2) SOR's inspection steps were not adequate to identify the faulty condition seals, and (3) there was a manufacturing error in the heat cure of the epoxy because manufacturing personnel had not followed procedures. SOR also determined that the work format was not adequate because it did not require manufacturing personnel to record the actual heat cure temperature and the cure time for each batch.

In June 1994, SOR took measures to prevent recurrence by implementing more stringent testing on all conduit seals, including requiring (1) an insulation resistance test for conduit seals, (2) a 100 psi leak test for the completed conduit seal assembly, (3) a housing leak test for the conduit seal after completing all assembly steps and all thermal testing, and (4) test results to be approved by manufacturing and QA personnel for every order of switches. In addition, SOR took measures to record the cure temperature and time for the epoxy to ensure that the correct heat cure was used. This activity is required to be approved by manufacturing and QA personnel for every order of conduit seals. No further concerns were identified.

The inspector observed that SOR did not ask customers to identify any chemicals that the switch components would be exposed to during installation or operation to ensure that the switch was not compromised in the performance of its function. Chemicals in the process (e.g., ammonia) could degrade switch components (e.g., seals). SOR indicated that they assessed any process chemicals if identified by the licensee.

The inspector assessed SOR's implementation of licensee purchase order requirements in SOR's design documents (SOR assembly drawings 8520-264 Revision 2, 8520-506 Revision 1, and 8215-659 Revision 2). No concerns were identified.

The inspector observed that SOR's General Instructions did not address the protection of switch components during handling (e.g., debris entering the switch housing, damage to lead wires) which could affect the operation of the microswitch. The QAVP added a cautionary statement to the General Instructions.

The inspector assessed SOR's internal audit report 7701-128, revision 4, dated December 30, 1996. The audit, in part, assessed the results of nonconformance reports and corrective actions. No concerns were identified.

c. Conclusions

In general, SOR's QA manual and its implementation were in compliance with the requirements of Appendix B to 10 CFR Part 50, except for the nonconformance described herein. SOR took adequate corrective actions and steps to prevent recurrence of identified manufacturing defects.

3.2 Review of Licensee Monitoring of SOR

a. Inspection Scope

The inspector evaluated licensee monitoring of SOR's control of quality for safety-related items purchased by licensees, including Part 21 reports and associated corrective actions.

b. Observations and Findings

In January 1995, NUPIC - represented by Southern California Edison (SCE), Baltimore Gas & Electric Company, and Yankee Atomic Electric Company - audited SOR's QA program. Part of the scope of the audit was to verify whether SOR had established and effectively implemented a QA program in compliance with the requirements of 10 CFR Part 50, Appendix B and other industry standards. The NUPIC audit team identified SOR deficiencies in the areas of (1) control of purchased materials, (2) test control, (3) corrective actions, and (4) control of measuring and test equipment. NUPIC considered the findings to be "administrative" and believed there was no adverse impact on the quality of SOR's completed products. NUPIC accepted SOR's corrective actions for the above findings and closed the findings on April 11, 1995. NUPIC concluded that SOR's QA program was adequate and that implementation was satisfactory.

The inspector observed that during its audit, NUPIC reviewed SOR's 10 CFR Part 21 report, dated October 14, 1994, regarding cracking of the insulation of switch lead wires. NUPIC verified SOR's corrective actions by observing in-process assembly of pertinent switches and associated documentation. The inspector noted that during the NUPIC audit, licensees did not evaluate SOR's 1993-1994 corrective actions regarding leakage of O-ring seals in switches exposed to radiation and elevated temperatures, and leakage of epoxy seals in switches. These manufacturing defects were reported by SOR in information notices to customers (see section 3.1b of this report). After a telephone discussion with SCE's procurement quality staff, the inspector confirmed that NUPIC had not included these issues in the scope of the audit.

In February 1997, NUPIC, represented by Omaha Public Power District, examined the application of SOR's QA program to all phases of the design and manufacture of SOR switches. NUPIC noted that SOR had issued a 10 CFR 21 report in 1993 (no details are noted in the NUPIC report). NUPIC recommended, in part, that SOR should clearly document its methods of verification of critical characteristics, and develop a checklist of specific inspection criteria for items purchased. No findings were identified.

c. Conclusions

In general, licensee monitoring of SOR's quality was in accordance with proper criteria, procedures, and checklists. NUPIC did not evaluate SOR's corrective action for two manufacturing defects reported by SOR to its customers.

3.3 Entrance and Exit Meetings

At the entrance meeting on June 16, 1997, the NRC inspector discussed the scope of the inspection, outlined the areas to be inspected, and established interactions with SOR management. In the exit meeting on June 19, 1997, the inspector discussed his finding and observations.

4 PARTIAL LIST OF PERSONNEL CONTACTED

SOR

Lew Goetz, President and CEO
Colbert Turney, Vice President (VP), Quality
Lind Coutts, Coordinator, Nuclear Engineering
Joseph Modig, Engineer, Nuclear Engineering
Landen Tuggle, Director, Manufacturing
Harold Moddy, VP Sales
Charisse Smith, VP Finance
Tim Ceillesen, Product Manager
Richard Johnson, QC Engineer

Southern California Edison

Jeff Larson, Supervisor, Procurement Quality

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

99900824/97-01-01	Para 3.1 b	NON	inadequate instructions and procedures
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Closed

99900912/93-01-06	Para 3.10 of inspection report 99900912/93-01	Open Item	inadequate information regarding test anomalies
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14685 W. 105th Street • Lenexa, Kansas 66215-5964 USA
Tel. 913.888 2630 • Fax 913 888 0767

9001

Mr. Anil S. Gautam, NRC

Subject: 10CFR21 dated October 14, 1994

Dear Mr. Gautam:

In July of 1993 SOR began to manufacture pressure, vacuum, and temperature switches that were qualified by SOR test report 9058-102. In September of 1994 SOR was notified of a manufacturing defect by Nebraska Public Power (Ref. RGA 2125, seven defective units) and Connecticut Yankee (Ref. RGA 2117, one defective unit). In addition SOR assembly personnel had identified the same defect (Ref. MRR 1479, one defective unit). The defect was identified as a crack in the lead wire insulation. This prompted SOR to issue a 10CFR21 and investigate the cause of the defect. The cause of the cracked insulation was a heavy coating of epoxy on the wires outside of the potted area and was due to poor workmanship. This condition went undetected because SOR quality inspectors did not notice the coating of epoxy on the wires.

As noted in the Part 21 Notification, the following corrective action was taken in October 1994:

1. The Work Order formats for the conduit seals were revised to include specific instructions not to allow epoxy on the wires. In addition, there is an inspection step at the end of the Work Order that instructs the inspector to examine the wire and reject any that have epoxy on the wire. Each of these steps must be signed off on the Work Order by Manufacturing and QA personnel for each order of conduit seals.
2. Shrink tubing was added to the lead wires where they exit the conduit seal. The purpose of this tubing is to protect the wire insulation during shipping and handling. This step is signed off on the assembly procedure by Manufacturing and it is reviewed by QA personnel for every switch.
3. The wire manufacturers recommended minimum bend radius was added to the SOR General Instructions that are provided to the customer with each switch.

The conduit seals are manufactured as a sub-assembly in a separate environmentally controlled room. Therefore, there is no danger of epoxy contamination on any other parts of the switches.

Reference Corrective Action Report 0357.

Regards,

Colbert Turney

Colbert Turney, V.P. Quality

Joseph G. Modig 4/19/97
Joseph G. Modig, Engineer

14685 W. 105th Street • Lenexa, Kansas 66215-5964 USA
Tel. 913.888.2630 • Fax 913.888.0767

9001
REGISTERED

Mr. Anil S. Gautam, NRC

Subject: Information Notice of April 1, 1993

Dear Mr. Gautam:

Prior to April 1, 1993 SOR had been manufacturing nuclear qualified vacuum switches for approximately 10 years. These switches were qualified by a combination of testing and analysis as listed below:

- AETC Test Report 17344-82N-D, Rev.1
- AETC Test Report 18441-83N, Rev. 1
- AETC Test Report 17344-82N-C, Rev. 3
- AETC Test Report 18577-83N, Rev. 1
- AETC Test Report 18878-84N-2, Rev. 2
- SOR Analysis 8215-959

In 1992 and 1993 SOR underwent a new qualification program (SOR Test Report 9058-102) and discovered that the o-ring gland design on the vacuum piston was not capable of retaining maximum operating pressure after exposure to radiation, aging, and cycling. SOR informed the NRC and the utilities of this condition on April 1, 1993. The qualification test specimens were left in the test program with no modifications and continued to function properly and passed all tests with the exception of the hydrostatic test at the conclusion of the HELB and LOCA. SOR redesigned this seal to meet hydrostatic requirements and qualified it by analysis (Ref. SOR Test Report 9058-102, Section 14, Appendix 4, Analysis 8923-219).

This condition was not discovered earlier because of inadequate engineering testing and analysis of the vacuum switch.

As noted in analysis 8923-219, examination of the test specimens revealed that the o-ring was still sealing between the vacuum piston and the diaphragm, but leakage was occurring between the vacuum screw and the o-ring. This is attributed to the triangular gland design and a combination of compression set, volumetric swell, and shrinkage which occurs from exposure to elevated temperatures and irradiation. All of these factors contributed to the loss of the line of contact between the vacuum screw and the o-ring, and the resultant leakage at high hydrostatic pressures.

The redesign, which was released by an Engineering Order on May 20, 1993, eliminates the leak path mentioned above because the vacuum piston is now welded to the vacuum

screw. In addition, the triangular o-ring gland was changed to a face seal configuration. A face seal is utilized on the pressure port o-ring of the vacuum switch and has successfully retained 750 PSI hydrostatic pressure after exposure to radiation and thermal aging. The pressure port o-ring and the vacuum screw o-ring are made of the exact same material (Parker compound E740 for option "M9"; Parker compound V709 for option "M4") and differ only in size. The o-ring gland dimensions are in accordance with the Parker O-Ring Handbook for a static face seal gland.

Regards.

Colbert Turney
Colbert Turney, V.P. Quality

Joseph G. Modig 6/19/97
Joseph G. Modig, Engineer

14685 W. 105th Street • Lenexa, Kansas 66215-5964 USA
Tel. 913.888.2630 • Fax 913.888.0767

9001
REGISTERED

Mr. Anil S. Gautam, NRC

Subject: Information Notice of June 10, 1994

Dear Mr. Gautam:

In July of 1993 SOR began to manufacture pressure, vacuum, and temperature switches that were qualified by SOR test report 9058-102. In May of 1994 SOR discovered a potential leakage problem in the conduit seals of these switches during routine testing and reported this discovery to the NRC and the affected utilities on the subject Information Notice. This problem went undetected by SOR for three reasons:

1. The leakage was a random problem.
2. Inspection steps were not adequate to identify faulty conduit seals.
3. There was a manufacturing error in the heat cure of the epoxy. This error was due to manufacturing personnel not following procedures. In addition, the Work Order format was not adequate because it did not require manufacturing personnel to record the actual heat cure temperature and cure time for each batch.

As noted in the Information Notice, more stringent testing was instituted immediately (June, 1994). This included the following steps:

1. An insulation resistance test was added to the Work Order format for conduit seals. This step is signed off by the manufacturing personnel and reviewed and signed off by QA personnel for every order of switches.
2. The insulation resistance test procedure was changed to include testing of wire to wire (all combinations) in addition to the standard wires to case test.
3. A 100 PSI leak test was added to the Work Order format for the completed conduit seal assembly. This step is signed off by the manufacturing personnel and reviewed and signed off by QA personnel for every order of conduit seals.
4. A housing leak test was added to the assembly procedures in order to test the conduit seal after all assembly steps and all thermal testing is complete. The test pressure is equivalent to the HELB or LOCA pressure as applicable. This step is signed off on the assembly procedure by the manufacturing personnel and reviewed and signed off by QA personnel for every order of switches.

In addition to the above steps, the following corrective action was taken in August 1994:

1. The Work Order formats for the conduit seals were changed to require manufacturing to record the cure temperature and time for the epoxy. This will insure that the correct heat cure is used. This step is signed off by the manufacturing personnel and reviewed and signed off by QA personnel for every order of conduit seals.

Reference Corrective Action Report 0338.

Regards,

Colbert Turney
Colbert Turney, V.P. Quality

Joseph G. Modig 6/19/97
Joseph G. Modig, Engineer

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

July 22, 1997

Mr. Richard P. Bender
Vice President
Yuasa Exide, Inc.
2366 Bernville Road,
Post Office Box 14145,
Reading, Pennsylvania 19612

SUBJECT: NRC INSPECTION REPORTS 99900358/97-01 AND 99900359/97-01, NOTICE
OF VIOLATION, AND NOTICE OF NONCONFORMANCE

Dear Mr. Bender:

On March 4-7, 1997, and on April 28-May 2, 1997, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection at the Yuasa-Exide, Inc. (YEI), facilities at Reading, Pennsylvania, and Richmond, Kentucky, respectively. The enclosed report presents the results of those inspections.

During the inspections, the NRC inspectors found that certain of your activities appeared to be in violation of NRC requirements. Specifically, the inspectors determined that contrary to Section 21.21 of Part 21 of Title 10 of the *Code of Federal Regulations* (10 CFR 21.21), YEI failed to inform all applicable licensees that certain YEI GN type battery cells that were manufactured between October 1992 and December 1992 could potentially have less than the manufacturer's publicized rated capacity of 8-hours. YEI sent letters to Southern California Edison (San Onofre Nuclear Generating Station) and Cleveland Electric Illuminating Company (Perry Nuclear Power Plant) informing these affected licensees of this deviation in GN type battery cells pursuant to 10 CFR 21.21(b), but not to the Washington Public Power Supply System (Washington Nuclear Plant, Unit-2).

This violation is cited in the enclosed Notice of Violation (NOV), and the circumstances surrounding the violation are described in detail in the enclosed report. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed NOV when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In addition, the NRC found that the implementation of the YEI quality assurance program failed to meet certain NRC requirements imposed on you by your customers. YEI did not comply with its Quality Assurance Manual requirements regarding documenting nonconforming material upon the failure of 2GN-15 cells for the San Onofre Nuclear Generating Station. Also, the measures YEI established for review of suitability of application of purchased parts and materials to be used in Class 1E batteries and verification that those purchased parts and materials met the procurement specifications were inadequate. The specific findings and references to the pertinent requirements are identified in the enclosures of this letter.

Mr. Bender

- 2 -

The failure of the San Onofre cells has potentially generic implications. The apparent susceptibility of the 2GN cell to premature loss of capacity at elevated temperatures may be indicative of an inherent weakness in some aspect of the design or manufacturing process. Should this be the case, it could result in the inability of station batteries to maintain required voltage for the required time under certain design basis conditions. In particular, elevated ambient temperatures due to loss of air conditioning during design basis events such as station blackout may impact the station batteries' ability to perform their safety function. The NRC believes that this potential deviation from YEI's published product performance claims should be thoroughly investigated. If it is found to be a deviation, all affected licensees or purchasers should be informed in accordance with 10 CFR 21.21(b).

Please provide us within 30 days from the date of this letter a written statement in accordance with the instructions specified in the enclosed Notice of Nonconformance. We will consider extending the response time if you can show good cause for us to do so.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC's Public Document Room (PDR).

Sincerely,

Stuart A. Richards, Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket Nos. 99900358, 99900359

Enclosures: 1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99900358,359/97-01

NOTICE OF VIOLATION

Yuasa Exide, Inc.
Reading, Pennsylvania/Richmond, Kentucky

Docket Nos.: 99900358/99900359
Report No.: 97-01

During NRC inspections conducted at you facilities March 4-7, 1997, and April 28-May 2, 1997, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

Section 21.21, "Notification of failure to comply or existence of a defect and its evaluation," of Part 21 of Title 10 of the *Code of Federal Regulations* (10 CFR 21.21) became effective October 29, 1991. Section 21.21(b), states that if a deviation or failure to comply is discovered by a supplier of basic components, or services associated with basic components, and the supplier determines that it does not have the capability to perform the evaluation to determine if a defect exists, or if the failure to comply is associated with a substantial safety hazard, then the supplier must inform the purchasers or affected licensees within five working days of this determination so that the purchasers or affected licensees may evaluate the deviation or failure to comply, pursuant to 10 CFR 21.21(a).

Contrary to the above, YEI failed to inform all affected licensees that certain YEI GN type battery cells that were manufactured between October 1992 and December 1992 could potentially have less than the manufacturer's publicized rated capacity of 8-hours. YEI sent letters to Southern California Edison (San Onofre Nuclear Generating Station) and Cleveland Electric Illuminating Company (Perry Nuclear Station) informing them of this deviation in the affected GN type battery cells, but not to Washington Public Power Supply System (Washington Nuclear Plant, Unit 2), which had received eight GN type safety-related battery cells that were manufactured during the same time period. Violation 99900358,359/97-01-01.

This is a Severity Level IV violation (10 CFR Part 2, Appendix C, Supplement VII).

Pursuant to the provisions of 10 CFR 2.201, Yuasa Exide Inc., is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 22nd day of July, 1997

Enclosure 1

NOTICE OF NONCONFORMANCE

Yuasa-Exide, Inc. (YEI)
Reading, Pennsylvania/Richmond, Kentucky

Docket Nos.: 99900358/99900359
Report No.: 97-01

Based on the results of an NRC inspection conducted at YEI facilities in Reading, Pennsylvania, and Richmond, Kentucky, on March 4-7, 1997, and on April 28-May 2, 1997, respectively, it appears that certain of your activities were not conducted in accordance with NRC requirements as follows:

- A. Criterion XV, "Control of Nonconforming Material," of Appendix B, "Quality Assurance Requirements for Nuclear Power Generation Facilities," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix B), requires in part that nonconforming items shall be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures.

Section 15, "Nonconforming Material, Parts, or Components," of the YEI Quality Assurance (QA) Manual required that QA personnel prepare material review reports in cases of nonconforming material and disposition the nonconformances.

Contrary to the above, YEI did not document and disposition four test failures of the SCE batteries in accordance with its 10 CFR Part 50, Appendix B quality assurance program in July 1996.

Nonconformance 99900359/97-01-02.

- B. Criterion III, "Design Control," of 10 CFR Part 50, Appendix B, requires a review for suitability of application for safety-related structures, systems, and components.

Contrary to these requirements, the measures established by YEI-Richmond, Kentucky, for review for suitability of application (commercial grade dedication procedures prescribed in QAP 70.0 and individual technical evaluations) of purchased parts and materials to be used in the manufacture of Class 1E station batteries for nuclear power plants were not adequate as follows:

- Not all critical characteristics for certain items were identified, e.g., O-ring material and cure date,
- Not all verification methods or acceptance criteria were appropriate or correct, or consistent with design documents (drawings or bills of materials), or purchase specifications (which themselves were not always consistent with design documents), or expressed in technically correct terms.
- Not all engineering design drawing changes were incorporated into purchase specifications, technical evaluation or acceptance process attachments to QAP-70, or into incoming inspection report forms.

Nonconformance 99900358,359/97-01-03

Enclosure 2

- C. Criterion VII, "Control of Purchased Material, Equipment, and Services," of 10 CFR Part 50, Appendix B, requires that measures be established to verify, by review of suppliers and supplier documentation, and examination of products upon delivery, that the purchased material, equipment, or services, meet the procurement specifications.

Contrary to these requirements, the measures established by YEI-Richmond, Kentucky, for verification that purchased parts and materials to be used in the manufacture of Class 1E station batteries for nuclear power plants met the procurement specifications were not adequate as follows:

- For those critical characteristics that were identified, not all were adequately verified, e.g., material specified for intercell connector fasteners was not adequately verified and the wrong material was specified on incoming inspection report forms for post-seal caps.
- Commercial grade supplier surveys used to support verification of one or more critical characteristics for various items were broad-based programmatic reviews (not performance based) and without adequate specificity to verify that the supplier controls the critical characteristic of interest. Certificates of conformance were accepted from distributors whose ability to provide valid certificates was not verified.

Nonconformance 99900359/97-01-04

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance: (1) the reason for the nonconformance, or if contested, the basis for disputing the nonconformance, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further noncompliances, and (4) the date when your corrective actions will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 22nd day of July, 1997

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Enclosure 2

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

ORGANIZATION: Yuasa Exide, Inc. (YEI)
2366 Bernville Road,
Post Office Box 14145,
Reading, Pennsylvania 19612

DOCKET/REPORT NO(s): 99900358/97-01 (Reading, Pennsylvania)
99900359/97-01 (Richmond, Kentucky)

ORGANIZATION CONTACT: Richard P. Bender, Vice President, Quality Assurance
and Purchasing (610)208-1972
Lana West, Quality Assurance Manager (606)627-7346

NUCLEAR INDUSTRY: Manufactures and supplies stationary batteries and
ACTIVITY battery racks.

INSPECTION CONDUCTED: Reading, Pennsylvania, March 4-7, 1997
Richmond, Kentucky, April 28-May 2, 1997

TEAM MEMBERS: K.R. Naidu, Team Leader, NRR
J.J. Petrosino, NRR
S.D. Alexander, NRR
S.N. Saba, NRR

Approved By: G.C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Enclosure 3

1 INSPECTION SUMMARY

The purpose of this inspection was to evaluate the quality assurance (QA) program and its implementation in the design, qualification and manufacture of rectangular, vertical plate, lead-acid battery cells (Type "GN") by Yuasa-Exide, Inc. (YEI), used in Class 1E/vital station batteries at nuclear power plants.

Conducted at YEI's engineering facility in Reading, Pennsylvania, and its factory in Richmond, Kentucky, the inspection focused on: (1) the implementation of the manufacturer's process controls, (2) procedural adequacy (including consistency with established requirements) and procedural compliance, (3) procurement and acceptance of purchased parts and materials used in battery manufacture (including commercial-grade dedication of components and parts for resale as basic components, e.g., battery racks and replacement parts), and (4) purchase orders (POs) from NRC licensees and certificates of conformance (COCs) and associated documents provided to NRC licensees.

Inspection bases were:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50, Appendix B)
- 10 CFR Part 21, "Reporting Defects and Noncompliance"

The inspectors identified two minor violations of 10 CFR Part 21 (§21.21(a) and §21.6) (See Section 3.1); one Level IV violation of 10 CFR 21.21(b) (See Section 3.2); one nonconformance with respect to Criterion XV of 10 CFR Part 50, Appendix B (See Section 3.5); nonconformances with respect to Criterion III of 10 CFR Part 50, Appendix B (See Section 3.7); nonconformances with respect to Criterion VII of 10 CFR Part 50, Appendix B (Section 3.7); and one inspector followup item (Section 3.6).

2 STATUS OF PREVIOUS INSPECTION FINDINGS

There have been no NRC inspections conducted since Exide became Yuasa-Exide, Inc. (YEI) and after the company reorganized under new management.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 10 CFR Part 21 Procedure and Posting

a. Scope

The inspectors reviewed YEI's quality assurance procedure (QAP) for reporting in accordance with 10 CFR Part 21 (Part 21), QAP 80.0, "10CFR21 - Procedure for Reporting Non-Conforming Material," dated November 2, 1994. QAP 80.0 was developed by YEI to address Part 21 requirements at the two YEI facilities

which perform activities relating to "basic components." The inspectors also observed and reviewed the document that was posted at both facilities to comply with § 21.6 of Part 21, "Posting requirements."

b. Observations and Findings

The inspectors determined that QAP 80.0 included certain provisions of 10 CFR 21.21(c) (NRC notification procedures), which are not required by the current revision to 10 CFR Part 21 to be included in procedures adopted pursuant to the regulation. However, QAP 80.0 did not contain any of the required provisions of §21.21(a) (evaluation of deviations and failures to comply, interim reports, and notification of directors or responsible officers).

The inspectors also determined that the Part 21 posting at YEI's Reading facility was not in accordance with the requirements of 10 CFR 21.6. The inspectors observed that YEI-Reading, apparently opting for a §21.6(b) posting, had posted only a notice that indicated that a copy of 10 CFR Part 21 was available for review in its administrative office. However, the notice lacked the other information required by §21.6(b), i.e., a description of the regulation and the Part 21 procedures, the location where the procedures (as well as the regulation itself) may be viewed, and the name of the person to whom reports should be made. In addition, YEI-Reading did not post Section 206 of the Energy Reorganization Act of 1974 as required by both §21.6(a) and §21.6(b). At its Richmond, Kentucky, facility, YEI had complied with the posting requirements of §21.6(a).

The inspectors discussed with the YEI staff the provisions of §21.21(b), which require that deviations or failures to comply, discovered by a supplier of basic components, for which the supplier determines that it does not have the capability to perform the evaluation of §21.21(a)(1) to determine if a defect exists, must be reported to the purchasers or affected licensees within five working days of this determination so that the purchasers or affected licensees may evaluate the deviation or failure to comply. The inspectors explained that although §21.21(b) is not specifically required to be included in the procedures adopted pursuant to 10 CFR part 21, it is perhaps the most important provision for a particular vendor/supplier's disposition of deviations or failures to comply, because most vendors or suppliers do not have the capability to perform a §21.21(a)(1) evaluation. As also defined in Section 21.3, the §21.21(a)(1) evaluation is the process of determining whether a particular deviation constitutes a defect, i.e. whether it could create a substantial hazard or lead to exceeding a license technical specification safety limit, or determining whether a failure to comply (with the Atomic Energy Act of 1954, as amended, or any rule, regulation, order or license of the NRC) is associated with a substantial safety hazard. YEI agreed that this process would normally be beyond its capability because, although YEI can advise a licensee or purchaser of the effect of a particular deviation on the performance of the battery, it cannot determine the ultimate effect on plant operation, reliability or safety.

The inspectors also explained that nothing in the regulation should be construed as prohibiting a report to the NRC by anyone who is concerned that a deviation may be a defect or that a failure to comply may be associated with a

substantial safety hazard, even if they are not capable of performing a §21.21(a)(1) evaluation. However, a vendor who is not qualified to perform the §21.21(a)(1) evaluation should not perform the evaluation (in lieu of informing affected licensees or purchasers in accordance with §21.21(b)) and determine that a report to the NRC is not required because the deviation does not appear to the unqualified vendor to be a defect or because the failure to comply does not appear to be associated with a substantial hazard.

After discussing 10 CFR Part 21 responsibilities in detail with the YEI QA staff, the inspectors informed YEI that the failure to establish an adequate procedure and having an inadequate posting at its Reading facility were violations of 10 CFR Part 21. However, these failures constituted violations of minor significance and will be treated as a Non-Cited Violations, consistent with Section IV of the NRC Enforcement Policy. Subsequent to the Reading inspection, the YEI corporate staff drafted its revision to QAP 80.0 and the draft document was discussed during the Richmond inspection.

c. Conclusion

The inspectors concluded that YEI had not developed an adequate procedure to comply with the requirements of 10 CFR Part 21, and had not complied with §21.6 posting requirements at its Richmond facility. However, YEI had complied with the §21.6 posting requirements at its Richmond facility and has adequately revised QAP 80.0 to address the procedural requirements of 10 CFR 21.21(a) as well as including provisions to ensure compliance with §21.21(b).

3.2 10 CFR Part 21 - Informing Affected Licensees

a. Scope

The inspectors reviewed an April 19, 1996, YEI letter that was sent to Southern California Edison (San Onofre Nuclear Generating Station) (SONGS) and Cleveland Electric Illuminating Company (Perry Nuclear Power Plant) (Perry) regarding GN type battery cells that were supplied. The YEI author also provided a copy of the letter to the NRC staff for information. The inspectors reviewed the letter to determine the adequacy of informing its customers of deviations or failures to comply.

b. Observations and Findings

The YEI letter indicated that certain YEI GN type battery cells that were manufactured between October 1992 and December 1992 could potentially have less than YEI's publicized rated capacity of 8-hours. The letter stated to the two licensees and NRC staff that only Perry and SONGS received GN type batteries manufactured in the suspect time period.

During the Richmond facility inspection, the inspectors reviewed the letter and associated documents including YEI-Richmond's 1992 "custom order status log" (status log) for the suspect time period. The status log is an internal YEI-Richmond document that is used by YEI-Richmond quality control (QC) personnel and order entry personnel to maintain the status of all commercial and nuclear incoming battery cell orders that have special requirements

imposed. The inspectors noted that the status log reflected that both licensees that received the YEI letter had GN type cells manufactured in the subject time period.

However, the inspectors determined that there was a third licensee that received cells manufactured during this time period. It was noted that Washington Public Power Supply System (WPPSS) had also received eight GN type safety-related battery cells. Further review of YEI's records showed that the WPPSS Purchase Order (PO) 221020, dated October 1992, did in fact order eight GN battery cells and imposed nuclear safety-related requirements on YEI. YEI committed to informing WPPSS of the deviation by July 25, 1997.

The inspectors determined that YEI had failed to inform one of the affected licensees of the deviation in affected GN type battery cells within the time limit required by 10 CFR 21.21(b). This has been identified as Violation 99900359/97-01-01.

As discussed in Section 3.3 below, the inspectors determined that the GC and GN cells are similar in design and performance and their manufacturing process controls are nearly identical. Therefore, the inspectors noted that GC cells may be susceptible to the deviation discussed above. Although YEI has qualified only the GN cells for safety-related service, some licensees buy the commercial-grade GC cells and dedicate them. The inspectors found that Florida Power Corporation (FPC) and New York Power Authority (NYPA) purchased GC-Type cells manufactured during the period in question. FPC purchased 30 2GC-9 cells, assembled October 2, 1992 (FPC PO A7301166), and NYPA purchased four GC-33 cells (PO number undetermined), assembled November 23, 1992, and tested December 11, 1992.

c. Conclusion

The inspectors concluded that YEI did not perform an adequate review of its manufacturing records for the time period in question. Consequently, it did not identify the one other nuclear customer that received affected battery cells intended for safety-related applications. As a result, WPPSS was not informed of the deviation, a potential substantial safety hazard.

3.3 Battery Manufacturing Process

a. Scope

The inspectors observed processes in the various stages of manufacturing of YEI GN type battery cells. The inspectors examined items in production, reviewed logs and other process records and interviewed technicians to determine the adequacy and availability of the manufacturing instructions at the work stations and compliance with those instructions. The inspectors also reviewed production QA activities.

b. Observations

The inspectors observed various phases of cell manufacturing, including oxide mill operation, grid casting and assembly, flat plate manufacturing, paste mixing, machine pasting, positive and negative plate processing (paste curing,

trimming, etc.), cell assembly and sealing, leak testing (15 GC-25 cells, of which two cells had post seal-to-cover leaks and were put aside for repair and retesting), final formation of the cells (electrolyte filling and a specified sequence of initial charging, discharging, recharging), and capacity discharge testing. In general, manufacturing instructions were available and being followed. Operators were knowledgeable and concerned with ensuring quality. No concerns were identified in this area.

The inspectors observed QC personnel performing various prescribed routine checks during the manufacturing process, including inspecting the plates for hairline cracks, plate thicknesses, loose/missing pellets, etc. No concerns were identified in this area.

The inspectors noted that while the GN (nuclear qualified Class 1E) and GC (commercial) cells are generally similar in design and comparable in performance, there are some differences, including the following:

- The container material of GN is a type of polycarbonate. The GC container material is a type of styrene.
- The GN cover is polycarbonate; the GC cover is polyvinylchloride.
- The positive pastes of GN and GC cells have different amounts of lead.
- The outside negative plates of the GN are thinner than the negative GC plates.
- The GC cells have not been qualified by YEI for seismic requirements.
- The GN and GC plate separators are of different materials.

c. Conclusions

The inspectors concluded that the YEI manufacturing process controls observed were effectively implemented.

3.4 Qualification of YEI GN-Series Batteries for Class 1E Service

a. Scope

The inspector reviewed Wyle Laboratories Report 45001-1, Revision A, dated January 15, 1989, the environmental and seismic qualification report for YEI's "GN"-Series batteries and battery racks for Class 1E service in nuclear power plants and interviewed YEI's Manager of Engineering Support, Large Stationary Batteries, who had witnessed the Wyle testing. The inspector also reviewed the report prepared by Flight Dynamics, Inc., in which it documented the seismic qualification analysis for the YEI Series GN battery design change. In the design change, the number of terminal posts was changed from two positive and two negative posts per cell to one positive and one negative post per cell to conform to the current design of YEI's similar cells for non-nuclear applications.

b. Observations

Although Class 1E batteries are typically not subject to exposure to the harsh environment of a design basis event such as loss-of-coolant accident or high energy line break (therefore not required to be qualified in accordance with 10 CFR 50.49), the Wyle test program was intended to qualify them for the expected extremes of normal service conditions (in accordance with the applicable general design criteria of 10 CFR Part 50, Appendix A) and subject the test specimen batteries to seismic testing at an end-of-life condition. Accordingly, the various test specimen batteries underwent accelerated thermal and radiation aging to the equivalents of 10, 15 and 20 years of nuclear plant service (at Wyle, Huntsville) and the order of 10,000 rads Co-60 radiation exposure (at The Georgia Institute of Technology) before undergoing seismic testing at Wyle. No deficiencies were noted in this report.

GN-Series batteries originally had two positive and two negative posts per cell (a total of eight posts in the usual 2GN or 2-cell-per-jar configuration) for ampacity reasons. However, the non-nuclear line had larger single posts to save weight and cost without sacrificing ampacity. In order to qualify the improved design, YEI contracted Flight Dynamics, Inc., who used finite element analysis to show that (1) the stresses during design basis seismic excitation were actually lower in the larger single posts and (2) the single-post design was stronger than the 2-post design that had been qualified by the Wyle tests cited above. No deficiencies were noted in this report.

c. Conclusions

The original qualification test program and subsequent design-change reconciliation analysis for the GN-Series batteries and racks appeared to have been conducted in accordance with applicable requirements and guidelines in effect at the time including General Design Criterion 2 of Appendix A to 10 CFR Part 50, Regulatory Guide 1.100, Institute of Electrical and Electronic Engineers (IEEE) Standard 323-1974 (Class 1E equipment qualification), IEEE Std 344-1975 (seismic qualification of Class 1E equipment), IEEE Std 450-1987 (Class 1E battery qualification), and IEEE Std 535-1987 (Class 1E battery seismic qualification). The inspectors had no concerns in this area.

3.5 Purchase Orders

a. Scope

At the YEI Reading and Richmond facilities, the inspectors reviewed selected licensee purchase orders (POs) for YEI GN type batteries and associated records to determine whether NRC licensees imposed the necessary and appropriate technical and quality requirements on YEI for the procurement of basic components.

b. Observations and Findings

The inspectors noted that each of the licensee POs reviewed imposed the requirements of 10 CFR Part 50, Appendix B, and stated that 10 CFR Part 21 was applicable. Each of the YEI customer files (called PO packages) contained

documents such as test and manufacturing records, COCs, and Lab Test Assignment (LTA) sheets. The LTA sheet is a form that is used as a manufacturing, test and quality assurance function traveler. The LTA for each particular job contains all of the customer's specific manufacturing and test requirements. It is generated by Richmond QC personnel, in conjunction with the order entry department. The inspectors determined that all of the PO packages reviewed contained appropriate quality and test requirements from the licensees except for one package discussed in Section 3.6 below.

c. Conclusion

The inspectors concluded that YEI's customer order packages were well maintained, retrievable and reflected the required test results. With the exception of the test failures discussed in Section 3.6, no other anomalies were noted in this area.

3.6 Failure of Capacity Test Discharge by San Onofre Cells

a. Scope

The inspectors reviewed Southern California Edison (SCE) PO 6L225004, dated March 1, 1995 (with subsequent revisions in October 1995 and August 1996), and associated documentation for ten 2GN-15 replacement cells for the San Onofre Nuclear Generating Station (SONGS). Also reviewed were test data for several other batteries with similar cells (Type 2GC) that had been capacity discharge tested at the 8-hour rate.

b. Observations and Findings

The test data indicated that ten 2GN-15 battery cells tested in July 1996 failed to meet SCE's original testing requirements on four occasions. The cells finally passed with acceptance criteria modified by SCE.

Although the test failures were largely attributable to some weak cells (e.g., 50001), YEI explained that they felt that the elevated temperatures (as high as 79°F) were a significant contributing factor to the poor performance of the weakest cells. YEI explained that they had experienced difficulty in maintaining lower temperatures due to inadequate air conditioning in the test room and the July heat. This was YEI's justification for conducting the test three more times after the initial failure, all with similar unsatisfactory results.

Originally, the 10 cells were to be discharged at the temperature-corrected 8-hour rate, maintaining ambient temperature as close to 77°F as possible, to an average end voltage of 1.75 volts per cell [VPC] with a minimum of 90% capacity per IEEE 450-1987. The temperatures during the discharges conducted by YEI on July 1, 4, 11, and 17, 1996, varied between 74°-79°F. However, in each test, the average cell voltage reached 1.75 VPC before 90% of 8 hours (7 hours, 12 minutes). The worst results were from the July 17th test conducted at 79°F. When the test was terminated at the 7-hour reading, the individual cell voltages ranged between 0.64-1.67 volts. In this test at the highest temperatures, most of the cells suffered degraded performance; although, cell

50001 had exhibited the poorest performance consistently in the previous three test discharges. Cell 50002 was also weak, but not as bad and the other eight cells had performed consistently better during the previous tests run at lower temperatures. Thus, the data suggested that one or two cells were weak, but all the cells exhibited a sensitivity to high temperature.

Section 15, "Nonconforming Material, Parts, or Components," and Section 16, "Corrective Action," of YEI's QA Manual required that quality personnel prepare a material review report and disposition the nonconformances in accordance with the QA program. However, YEI did not generate any material review reports to document the deviation from the licensee's PO requirements and its disposition, as required by its QA procedures. Instead, according to YEI, after the fourth test failure, the YEI Richmond facility QC staff contacted YEI corporate engineering, and YEI engineering contacted the SCE engineering staff. After consultation between the YEI and SCE engineering staffs, SCE changed its testing requirements to 95% capacity at the 4-hour discharge rate. The cells met this requirement during the fifth test and were shipped to SONGS.

c. Conclusion

The inspectors concluded that YEI had established measures to assure that conditions adverse to quality, such as test failures, and deviations are promptly identified and corrected, and that YEI had procedures to assure that nonconforming items would be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures. However, YEI failed to document and disposition the four test failures of the SONGS cells in accordance with its 10 CFR Part 50, Appendix B quality assurance program. Accordingly, Nonconformance 99900359/97-01-02 with respect to Criterion XV of 10 CFR Part 50, Appendix B, was identified in this area.

3.7 Safety Implications of Failure of SONGS Cells

a. Scope

The failure of the San Onofre cells is of technical concern with potentially generic implications. In addition to concerns about the requirements for SONGS batteries, the inspectors were concerned that perhaps YEI's advertised 8-hour capability for the 2GN cells in its product literature was not always achievable, particularly under elevated ambient and cell temperature conditions. To try to resolve concerns raised by the failure of the SONGS cells to pass their 8-hour capacity test discharge, the inspectors reviewed test data for several other groups of similar cells for other plants to determine whether there was any inherent difficulty in YEI 2GN cells meeting the manufacturer's published 8-hour discharge capability.

b. Observations

The 8-hour-rate capacity test data for several other batteries with similar cell types (2GC) with no anomalous cells and no elevated temperatures showed that all the cells exhibited greater than 90-percent capacity at the 8-hour rate. Some, at the lowest temperatures, were above 100 percent. However,

there were no data readily available for other 8-hour discharges at temperatures above 77°F. Therefore, the inspectors could not rule out generic susceptibility to premature loss of capacity at elevated temperatures.

c. Conclusion

Based on the cell test data reviewed, the inspectors concluded that YEI cells of this type should be able to meet an 8-hour rate, 90-percent capacity requirement within the bounds of expected ranges of normal service conditions in nuclear plant battery rooms. The inspectors further concluded that a significant factor contributing to the failures of the group of 10 cells for SONGS was the poor performance of Cells 50001 and 50002. However, the apparent susceptibility of the 2GN cell to premature loss of capacity at elevated temperatures may be indicative of an inherent weakness in some aspect of the design or manufacturing process. For example, according to YEI, the 2GN cell is of a relatively low electrolyte volume design. This feature causes the cell to exhibit a capacity-versus-discharge-rate profile typical of lead-acid cells up to about the 4-hour rate. However, for longer discharges/lower rates, the cell appears to suffer from electrolyte depletion and starts to exhibit reduced capacity in very long discharges. The inspectors were concerned that should this become limiting, particularly at high temperatures, it could result in the unexpected inability of station batteries to maintain required voltage for the required time under certain design basis conditions. In particular, elevated ambient temperatures due to loss of air conditioning during design basis events such as station blackout may impact the station batteries' ability to perform their safety function. Accordingly, the inspectors strongly recommended that this be investigated and that should it be determined to be a deviation, affected licensees and purchasers should be informed in accordance with 10 CFR 21.21(b). This issue was identified as Inspector Followup Item 99900358,359/97-01-05.

Instead of ordering a replacement for the weak cells, SCE revised the test acceptance criteria. The inspectors were not able to determine at YEI what the basis for SCE's original specification was, whether the criterion of 95 percent at the 4-hour rate was appropriate, or whether the weak cells would adversely impact the performance of one of SONGS's Class 1E station batteries.

3.8 YEI Commercial Grade Dedication Program

GN-Series batteries are designed and manufactured under YEI's 10 CFR Part 50, Appendix B, QA program and accordingly, are supplied to NRC licensees as basic components as defined in 10 CFR 21.3. YEI uses the provisions of its commercial grade dedication program as a systematic means to verify that purchased material and components used in the manufacture of Class 1E GN-type battery cells are suitable for safety-related service. In addition, the seismically qualified battery rack systems, manufactured to YEI specifications by the KIM Company, are purchased by YEI as commercial grade items, dedicated by YEI and supplied to NRC licensees as basic components. An adequate and effectively implemented commercial grade dedication program, being an activity affecting quality, functions under the applicable controls of the vendor's 10 CFR Part 50, Appendix B, QA program. However, it must meet, in particular, the requirements of Criterion III, "Design Control," and Criterion VII, "Control of Purchased Material, Equipment, and Services," of 10 CFR Part 50, Appendix B.

a. Scope

a.1 Procedures and Technical Evaluation/Review for Suitability

In order to evaluate the effectiveness of the YEI QA program and its implementation regarding controls applicable to the review for suitability of application and design verification in accordance with Criterion III, the inspector reviewed YEI Procedure QAP-70.0, "Dedication of Commercial Grade Items (CGI) for Nuclear Safety Related Applications," revision dated April 20, 1994. The review included the attachments to QAP-70.0 which comprise the component technical evaluations and associated acceptance process sheets. For reference, component part manufacturers' technical information, YEI design documents (principally drawings), and YEI procurement specifications, were also reviewed.

a.2 Acceptance/Procurement Specification Compliance

To evaluate the effectiveness of the YEI QA program and its implementation regarding controls applicable to supplier selection and qualification, review of supplier documentation and examination of purchased material and components in accordance with Criterion VII for verification of satisfaction of procurement specification requirements, the inspectors reviewed acceptance process sheets, incoming inspection reports (IIRs), supplier audits and commercial grade surveys, independent laboratory material analysis reports, procurement documents and supplier certificates of conformance. The inspectors also interviewed technicians and QA/QC personnel and examined selected purchased component parts and materials.

b. Observations and Findings

b.1 Procedures and Technical Evaluation/Review for Suitability

The inspector determined that QAP 70.0 was not fully consistent with the requirements of Criteria III and VII of 10 CFR Part 50, Appendix B, and the provisions of Electric Power Research Institute (EPRI) Report NP-5652, "Guideline for the Use of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)." The procedures were also not consistent with certain provisions of NRC Generic Letter 89-02, "Actions to Improve the Dedication of Counterfeit and Fraudulently Marketed Products," issued March 21, 1989 and NRC GL 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs," issued April 9, 1991, in which the NRC promulgated clarifications of staff positions on key issues, later incorporated into the revision of 10 CFR Part 21 that became effective November 1995.

QAP 70.0 did not contain the restrictions from NRC Generic Letter 89-02 on the use of Acceptance Methods 2 (commercial grade surveys) and 4 (product and supplier performance history) alone of EPRI Report NP-5652. However, the inspector noted that QAP 70.0 and its attachments, which comprise the technical evaluation and acceptance worksheets for dedication of individual GN component parts and materials and for the battery rack components, did prescribe multiple acceptance methods for many items and relied predominantly on Method 1 (special tests and inspections) when a single method was employed.

QAP-70.0 defined critical characteristics as in EPRI NP-5652, i.e., measurable quantities that when verified provide reasonable assurance that the item received is the item specified. It then defined critical characteristics for design and acceptance as in EPRI NP-6406, "Technical Evaluation of Replacement Items (NCIG-11)." However, these definitions and the stated position (as in NP-6406) that critical characteristics for acceptance are a subset of critical characteristics for design are not consistent with the intent of GL 89-02, nor the explicit definition of and NRC staff position on critical characteristics as promulgated in GL 91-05, nor the definition of critical characteristics contained in the revision of 10 CFR Part 21 that became effective in 1995. The NRC position is that critical characteristics when verified provide reasonable assurance that the item will perform its safety functions (not necessarily all design functions) and not fail in a manner detrimental to safety under all design basis conditions.

During the inspector's review of selected QAP 70.0 technical evaluation and acceptance process worksheets for the battery racks and various battery components and materials, the lists of critical characteristics for acceptance and their associated verification methods and acceptance criteria were not all complete and consistent with each other or with plant application requirements. The technical basis or rationale for the selection of characteristics, verification methods and acceptance criteria was not apparent in some cases. For example, QAP 70.0, No. 100-Series Attachments, the technical evaluation and acceptance process worksheets for the battery rack and components did not address the integrity of the welds in the fabricated rack components, nor the integrity of the bolted joints (e.g., stiffness, bolt torque, etc.) of installed racks. In another instance, terminal post O-ring material and cure date were omitted. Since the battery racks must remain structurally sound, and the cell connections remain tight during a design basis earthquake in order to ensure the operability of the safety-related station batteries, weakness in the welds (or loose/broken connections) could lead to rack failure and battery failure and thus prevent the batteries from performing their safety functions.

In addition, the inspectors found that not all acceptance criteria were appropriate or correct, or consistent with design documents (drawings or bills of materials), or purchase specifications (which themselves were not always consistent with design documents), or expressed in technically correct terms. One reason for this was because not all engineering design drawing changes were incorporated into purchase specifications, technical evaluations, or acceptance process sheet attachments to QAP-70, or into IIR forms. The inspectors' sampling review indicated that for post seal caps, the wrong material, i.e., not in accordance with the latest revision of the design drawing, was given on the incoming inspection report form. In another example, the incoming inspection form specified the wrong durometer hardness value for post-seal O-rings. Although this had been corrected by a pen-and-ink change on the IIR forms, the QAP-70 acceptance process sheet had a number from the previous drawing revision (D) and the P-spec had yet another number. Having incorrect acceptance criteria may fail to detect components that are not capable of performing their safety functions under all application design basis service conditions.

b.2 Acceptance/Procurement Specification Compliance

For those critical characteristics that were identified, not all were correctly specified or adequately verified. The inspectors identified material as an example of inappropriate verification method and acceptance criteria that was seen throughout the selected job files reviewed. Material was specified on most of the acceptance process sheets, but the verification method, visual inspection upon receipt, did not actually verify material. The IIR forms (for which there was no procedure for preparation or for use and little training beyond OJT) where material was identified as a critical characteristic, would simply say "material" under one of the critical characteristic column headings. For example, IIRs for intercell connector fasteners, specifically, bolts, specified "SS-316," for the material and the blocks for each sample specimen would simply say "yes." In effect, entering a yes just reflected the belief that the material was correct based on markings and other factors. IIRs for post seal caps, showed that the material was similarly inappropriately verified.

The inspectors followed up on the question of intercell connector fastener material in more detail. The material specified for intercell connector fasteners, expressed as "SS-316," was supposed to be ASTM Type 316 stainless steel. YEI had Singleton Laboratories perform annual chemistry analyses on samples of various materials including these fasteners to confirm proper material. However, YEI's implementation audits/commercial grade surveys of the suppliers of these fasteners (who were distributors, not the manufacturers), Threaded Screw Products (TSP), Inc., and PM Fasteners, Inc., were not performance based or item and critical characteristic specific. They did not document objective quality evidence that the suppliers obtained valid, lot-traceable information (e.g., CMTRs) on material and fabrication from the fastener manufacturers, nor did they document objective quality evidence of the suppliers' commercial quality controls to ensure that substandard or fraudulent material was not commingled. Therefore, the audits were not usable as a basis for acceptance of the fasteners from lots other than the ones from which actual analysis samples were taken. YEI also did not maintain lot traceability on the fasteners. YEI did not know, for example, whether their suppliers obtained the fasteners from the same manufacturers (the markings to be verified were inadequately described) or whether the manufacturers had adequate material control, nor had YEI documented the history of the consistency of this attribute (material). Finally, the audits were not consistent with the restrictions on the use of EPRI Method 2 in NRC GL 89-02 (nor was this mentioned in QAP-70.0). Therefore, in view of these deficiencies, the significant instances of substandard or fraudulent fasteners on the market, many from Asia, in view of the Asian origin of the received material, and in view of the non-standard markings on the fasteners, the inspectors concluded that the yearly sample analysis was not adequate to ensure consistent material suitability. The inspectors also noted that the COCs from TSP were not in accordance with the PO requirements (B and N of QAP-500). Although the signature blocks had the typed name of a person presumably in authority, it appeared that none of the COCs were signed by the named person, but rather by two different subordinates who signed the name of the designated person in authority's name instead of signing their own name with the annotation "for" the named person. It was also not known whether the

actual signers had proper authority or whether the named person had ever reviewed or approved the COCs. The fasteners were confirmed to be non-magnetic, consistent with an austenitic stainless steel such as 316. In addition, YEI milled a bolt taken from stock to check for indication of irregularities such as welded-on bolt heads. No such indications were apparent by visual examination.

An example of a technically incorrect expression of a specification was the electrical resistance specification in the purchase specification for Amerace ACE-SIL plate separators. Inconsistent with the separator manufacturer's technical information, the YEI purchase specification, acceptance process sheet, and the IIR form expressed this parameter in terms of ohms/square inch/mil of separator web thickness. Separator resistance (as used in battery parlance) is actually conduction path length-specific resistance (resistivity normalized on conduction path cross sectional area) and is properly expressed in units of ohm-inch/mil of separator web thickness. Although the separator resistance was specified using incorrect units, the inspector's calculation confirmed that the numerical values specified were consistent with the manufacturer's specifications.

c. Conclusions

The deficiencies in the YEI commercial grade dedication program description documents is considered a weakness in the YEI QA Program with respect to conformance to Criteria III and VII of 10 CFR Part 50, Appendix B. In addition, the YEI definition of the term critical characteristic in QAP 70.0 was inconsistent with 10 CFR Part 21.

The inspectors concluded that contrary to these requirements, the measures established by YEI-Reading, and implemented by YEI Richmond, Kentucky, for review for suitability of application (CGI dedication procedures prescribed in QAP 70.0 and individual technical evaluations, acceptance process sheets, purchase specifications and prepared IIR forms) for purchased parts and materials to be used in the manufacture of Class 1E station batteries for nuclear power plants did not meet the requirements of Criterion III of 10 CFR Part 50, Appendix B. Accordingly, Nonconformance 99900358,359/97-01-03 was identified.

In addition, YEI's measures for verification that these purchased parts and materials met the procurement specifications did not meet the requirements of Criterion VII of 10 CFR Part 50, Appendix B. Accordingly, this was identified as part of Nonconformance 99900359/97-01-04.

3.9 Supplier Quality Audits

a. Scope

The inspectors reviewed YEI audit and commercial grade survey procedures and checklists and also reviewed audits performed by YEI on its vendors. The inspectors evaluated them to determine if YEI auditors verified critical characteristics of the items furnished by those suppliers, and if the audits were performance based.

b. Observations and Findings

According to the audit reports reviewed, YEI used checklists to perform the audit/surveys of Cobra Wire and Cable, Inc., Amerace-Microporous Products, Inc., I.E. Dupont De Nemours & Co, Threaded Screw Products, Inc., and KIM Engineering Company (KIM), who supplied cable, separators, acid, bolts & nuts, and battery racks respectively. YEI documented adverse conditions in Audit Corrective Action Requests (ACARs) and requested suppliers to return the ACARs with the proposed actions to correct them. YEI then closed the ACARs if the proposed corrective actions were acceptable.

The inspectors determined that the attributes in the audit checklist were common to all suppliers and covered a broad programmatic overview of the supplier rather than focusing on the specific item being supplied. Furthermore, the audit did not attempt to verify the control of the quality of the critical characteristics of the specific item being manufactured.

For example, the audit of KIM failed to specifically verify the qualifications of the welders who welded the steel components of the rack and the quality control welding inspectors (QCWIs) who inspected the welds. The audit did not reveal that the QCWIs were using a welding inspection checklist that contained all the attributes of an acceptable weld (such as, size, undercut, porosity, and length).

QAP 30.9, "Rack Welding requirements and Welder Qualifications," dated October 15, 1982, stated, in part, "Responsibility for the qualification and reexamination rests with Exide working with their suppliers as to requalification and certification of the welders." The YEI auditors did not verify this attribute. YEI management acknowledged this weakness and committed to take adequate corrective action by developing supplemental item-specific checklists.

Similar concerns regarding the audit/survey of Threaded Screw Products is discussed in Section 3.8 above.

c. Conclusion

The inspector's review of YEI's supplier quality audit and commercial grade survey procedures and checklists and selected supplier audit/survey reports revealed that they were a broad-brush, programmatic review, not performance based. Such surveys/audits could provide a reasonable basis for preliminary qualification of a commercial grade supplier (e.g., placing the supplier on an approved commercial grade suppliers list), and may be useful in managing supplier quality resources, but they were not critical characteristic-specific and item-specific, and inconsistent with NRC GL 89-02. The audits/surveys did not adequately verify that the suppliers' commercial quality programs were effectively implemented, and did not cover distributors' programs where applicable. The inspectors concluded that YEI's qualification of certain suppliers and examination of supplier documentation did not meet the requirements of Criterion VII of 10 CFR Part 50, Appendix B. Accordingly, this concern is identified as part of Nonconformance 99900358,359/97-01-04.

3.10 Seismic Racks

a. Scope

To evaluate the requirements that YEI established to ensure that the seismic racks are manufactured to meet or exceed the requirements of those racks that met the seismic qualification tests, the inspectors reviewed QAP 30.8, "Quality Assurance Control-Seismic Racks," dated November 16, 1994, QAP 30.9, "Rack Welding requirements and Welder Qualifications," dated October 15, 1982, and YEI procurement specification P-011, Section 07, "Welding Steel Racks," dated October 1, 1982.

b. Observations and Findings

QAP 30.8 referenced QAP 30.9 and P-011, Section 07. QAP 30.9 referenced American Welding Society (AWS) Standard D1.1, as the applicable standard for welding the racks and specified the following:

- Filler material to be AWS A5.18 Z 70S-3 or equivalent
- Welded material to be ASTM A-36 of specified minimum yield strength
- Welding to be done in horizontal position
- Weld profiles to be in accordance with AWS D1.1, Sections 2.7 and 3.6
- Weld Procedure Specification and Welder Qualification to be in accordance with AWS D1.1, Parts B and C
- Weld quality as required by AWS D1.1, Paragraphs 5.10.3 and 5.11.2
- Visual inspection in accordance with AWS D1.1, Paragraph 5.6.3
- Weld size and location as specified on Drawing MC-83860, "Frame - Steel - 2 Step (G)," Revision E, dated October 3, 1985

The inspectors noted that Drawing MC-83860 had no weld dimension tolerances and observed that the engineering specifications were scattered among various documents and not consolidated in the drawing to facilitate QC inspection of the welds to the applicable requirements. YEI committed to revise the drawing which is frequently used by welders as well as inspectors to ensure that all welds meet or exceed the quality of the qualified specimen.

c. Conclusion

The engineering specifications for the weld and assembly of the racks were adequate, but were found to be scattered in various places instead of being consolidated in one place. The lack of colocated specifications and the audits or surveys of the KIM Company not verifying welder qualifications were weaknesses in YEI's QA control of special processes which YEI committed to strengthen.

3.11 Internal Audits

a. Scope

In order to determine if QC performed internal audits to verify the effectiveness of the quality assurance program, the inspectors reviewed Quality Procedures Manual (QPM) 4.17, "Internal Quality Audits," dated October 7, 1996, as well as selected internal audit reports.

b. Observations and Findings

According to the records, the internal audit performed by QA on April 3, 1997, to verify compliance with QPM 4.3.1, "Contract Review" dated July 10, 1995, identified 19 nonconformances. The audit conducted March 31, 1997, to verify compliance with QPM 4.18, "Training," dated March 13, 1997, identified several nonconformances; three of them repeated. The inspectors noted that corrective action for the repeat nonconformances was being implemented, but full compliance had not yet been achieved. The inspectors determined that corrective action requests (CARs) were written to identify nonconformances in all instances and were sent to the cognizant manager to determine the root cause of the problem, and document the corrective action taken or recommend the proposed corrective action.

c. Conclusion

The inspectors concluded that plant QA auditors performed internal audits in accordance with procedures to verify that quality activities comply with the planned arrangements and to determine the effectiveness of the quality system, audits were legible and retrievable, and corrective action being taken was being monitored. No concerns were identified in this area.

3.11 Training

a. Scope

The inspectors reviewed selected training records of craftsmen to determine if they had received training for the activities they were performing.

b. Observations and Findings

The inspectors reviewed the training folders of five craftsmen working in the pasting, oxide, burning, and gluing areas. The folders contained training record sheets that documented the date and subject of the training and the name of the instructor. The sheets were signed by attendees, acknowledging the training received. The training records were legible, retrievable, and complete.

c. Conclusion

The inspectors determined that personnel were trained in the areas in which they were working.

4 PERSONS CONTACTED

I.C. Baeringer, Vice President, Engineering
R.P. Bender, Vice President, Quality Assurance and Procurement
C. Claypool, Quality Control Technician
J. Hall, Quality Control
E. Simpson, Quality Control
L. Rickman, Plant Manager
B.P. Lightner, Manager, Supplier Quality
M.A. Patel, Manager, Engineering Support, Large Stationary Batteries
S.J. Weik, Manager, Design Engineering and Document Control
L.R. West, Manager, Quality Assurance, Richmond

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

99900358,359/97-01-01	VIO	Informing Affected Licensees of Deviation
99900358,359/97-01-02	NON	Documenting/Dispositioning Test Failures
99900358,359/97-01-03	NON	Review for Suitability of Application
99900358,359/97-01-04	NON	Product Verification/Supplier Audits/Surveys

Closed

None. No prior open items.

Discussed

99900358,359/97-01-05	IFI	Elevated temperature effect on GC and GN cells
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Selected Generic Correspondence on the Adequacy of
Vendor Audits and the Quality of Vendor Products

<u>Identifier</u>	<u>Title</u>
Information Notice 97-45	Environmental Qualification Deficiency for Cables and Containment Penetration Pigtailes
Information Notice 97-53	Circuit breakers Left Racked Out In Non-Seismically Qualified Positions
Information Notice 97-59	Fire Endurance Test Results of Versawrap Fire Barriers