NUREG–0040
Vol. 19, No. 4

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
October – December 1995

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.

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Manuscript Completed: February 1996
Date Published: February 1996

Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from October 1995 through December 1995.

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INTRODUCTION

A fundamental premise of the U. S. Nuclear Regulatory Commission (NRC) licensing and inspection program is that licensees are responsible for the proper construction and safe and efficient operation of their nuclear power plants. The Federal government and nuclear industry have established a system for the inspection of commercial nuclear facilities to provide for multiple levels of inspection and verification. Each licensee, contractor, and vendor participates in a quality verification process in compliance with requirements prescribed by the NRC's rules and regulations (Title 10 of the *Code of Federal Regulations*). The NRC does inspections to oversee the commercial nuclear industry to determine whether its requirements are being met by licensees and their contractors, while the major inspection effort is performed by the industry within the framework of quality verification programs.

The licensee is responsible for developing and maintaining a detailed quality assurance (QA) plan with implementing procedures pursuant to 10 CFR Part 50. Through a system of planned and periodic audits and inspections, the licensee is responsible for ensuring that suppliers, contractors and vendors also have suitable and appropriate quality programs that meet NRC requirements, guides, codes, and standards.

The Vendor Inspection Section (VIS) of the Special Inspection Branch reviews and inspects nuclear steam system suppliers (NSSSs), architect engineering (AE) firms, suppliers of products and services, independent testing laboratories performing equipment qualification tests, and holders of NRC construction permits and operating licenses in vendor-related areas. These inspections are done to ensure that the root causes of reported vendor-related problems are determined and appropriate corrective actions are developed. The inspections also review vendors to verify conformance with applicable NRC and industry quality requirements, to verify oversight of their vendors, and coordination between licensees and vendors.

The VIS does inspections to verify the quality and suitability of vendor products, licensee-vendor interface, environmental qualification of equipment, and review of equipment problems found during operation and their corrective action. When nonconformances with NRC requirements and regulations are found, the inspected organization is required to take appropriate corrective action and to institute preventive measures to preclude recurrence. When generic implications are found, NRC ensures that affected licensees are informed through vendor reporting or by NRC generic correspondence such as information notices and bulletins.

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. Appendices list selected bulletins, generic letters, and information notices, and include copies of other pertinent correspondence involving vendor issues.

INSPECTION REPORTS

November 6, 1995

Mr. Henry G. McCullough, Manager, Quality
Crane Valves Nuclear Operations
104 North Chicago Street
Joliet, IL 60431

SUBJECT: NRC INSPECTION NO. 99901285/95-01

Dear Mr. McCullough:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Crane Valves Nuclear Operations (CVNO), Joliet and Romeoville, Illinois, conducted September 25-28, 1995. The NRC inspection team, led by David H. Brewer and other inspectors named in the report, conducted a performance-based evaluation of CVNO management, staff, and quality programs and the implementation of those programs related to the fabrication of valves and valve components. The inspection was conducted to provide a basis for confidence that CVNO products supplied to the U.S. nuclear industry in fuel assemblies would perform their safety function.

The NRC team (a) examined technical documentation, procedures, and representative records, (b) held discussions, and (c) made various observations. On the basis of this inspection, the NRC team determined that the implementation of the CVNO quality assurance program, documented in the CVNO Nuclear Assurance Manual, Edition 2, Revision 4, March 5, 1994, either met or exceeded the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B to 10 CFR Part 50). The enclosed inspection report contains a detailed discussion of the areas examined and the NRC team observations.

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

Should you have any questions concerning this report, we will be pleased to discuss them with you. Thank you for your cooperation during this process.

Sincerely,

Original signed by Don Norkin for:

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

Docket No.: 99901285

Enclosure: Report No. 99901285/95-01

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
DIVISION OF TECHNICAL SUPPORT

REPORT NO.:	99901285/95-01
ORGANIZATION:	Crane Valves Nuclear Operations (CVNO) 104 North Chicago Street Joliet, Illinois 60431
ORGANIZATIONAL CONTACT:	Henry G. McCullough Manager, Engineering
NUCLEAR INDUSTRY ACTIVITY:	Crane Valves Nuclear Operations manufactures, repairs, modifies or replaces ASME Section III, Division 1, Class 1, 2 and 3 (includes non-code nuclear safety-related, government products and nonsafety-related manufacture) valves, valve parts and appurtenances.
INSPECTION DATES:	September 25 through 28, 1995

LEAD INSPECTOR: David H. Brewer — 11/3/95
David H. Brewer, Date
Vendor Inspection Section (VIS)
Special Inspection Branch (PSIB)

OTHER INSPECTORS: Richard P. McIntyre, VIS/PSIB

REVIEWED BY: Gregory C. Cwalina — 11/3/95
Gregory C. Cwalina, Chief, VIS/PSIB, Date

APPROVED BY: 11-6-95
Robert M. Gallo, Chief, PSIB, Date

Enclosure

1 SCOPE AND SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspection team (team) evaluated CVNO management, staff, and quality programs and the implementation of those programs related to the manufacture, repair, modification and replacement of nuclear safety-related valves and valve parts. The inspection was conducted to provide a basis for confidence that these items and services supplied to the U.S. nuclear industry would perform their safety function. The inspection basis consisted of the following:

- CVNO Nuclear Quality Assurance Manual (QAM), Edition 2, Revision 4, March 5, 1994
- 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)
- Part 21, "Reporting of Defects and Noncompliance," of 10 CFR

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

No nonconformances were identified during this inspection.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of CVNO.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

Crane Company began manufacturing valves in 1855. In 1952 Crane manufactured their first nuclear valves and supplied them to the U.S. Navy. The first valves supplied to a commercial nuclear power plant went to Duquesne Power and Light, Shipppingport, PA, in 1956.

In 1959 Crane made the first of a series of acquisitions by obtaining the Chapman Valve line (tilting disc check valve). In 1985 Crane acquired Aloyco, a leading stainless steel valve manufacturer, and nuclear valve operations became known as Crane-Aloyco, a division of the Crane Company. Also in 1985 Crane obtained the license for original equipment manufacturer production of Walworth nuclear spare parts and valves and acquired the Mark Controls service centers (Romeoville, Houston, Gonzales, San Leandro, Signal Hill and Woodbury). In 1994 Crane acquired Mark Controls adding Pacific Valves, Flowseal and Centerline products to its manufacturing and supply capability.

In 1988 the Chapman facility, Indian Orchard, MA, was closed and safety-related products were transferred to Crane's Romeoville facility. In 1992 the Flowmatics control valve line was transferred from the Chempump division to Crane's Nuclear Operations facility in Romeoville.

In 1971 Crane became the first manufacturer in the United States authorized to use the Nuclear "N" symbol of the American Society of Mechanical Engineers, for both valves and weld fittings.

Crane-Aloyco became Crane Valves Nuclear Operations in 1994.

3.2 Entrance and Exit Meetings

During the entrance meeting in Joliet, IL, on September 25, 1995, the team met with members of CVNO management and staff, discussed the scope of the inspection, reviewed mutual responsibilities for handling proprietary information and established contact persons for the team within the management and staff of the CVNO organization.

During the inspection, the NRC team conducted a performance-based inspection of CVNO organizations through technically directed observations and evaluations of processes, activities, and documentation. The NRC team (a) examined technical documentation, procedures, and representative records, (b) conducted interviews, (c) held discussions, and (d) made various observations.

Inspection participants and contacts are listed in Section 4.

During the exit meeting on September 28, 1995, the team summarized the inspection results with CVNO management and staff.

3.3 Quality Assurance Manual Control

The team reviewed QAM, Section II-A, "Quality Assurance Program," Paragraph 4.0, "Manual Control," to determine the appropriate QA program requirements. The team reviewed implementation of the QAM revision process for the changes from Revision 2 to Revision 3. Revision 3 was made to incorporate changes requested by an ASME Survey conducted August 2-4, 1993. During the ASME review, numerous suggestions and comments were made which were incorporated into Revision 3 of the QAM. The team verified that QAM Revision 3, August 4, 1993, had been reviewed and approved by the QA manager and by the ASME Authorized Nuclear Inspector (ANI). The team also determined that Revisions 3 and 4 had been distributed to all listed holders of a controlled copy of the QAM.

3.4 Indoctrination and Training

The team reviewed QAM, Section II-A, "Quality Assurance Program," Paragraph 6.0, "Indoctrination and Training," and Procedure Number 02-101, "Indoctrination and Training," Revision 4, May 18, 1990, to determine the appropriate QA program requirements. The team chose three quality assurance and three design support engineers for a review of their training records.

For each position, one individual was relatively new to the position and one was the senior engineer in the group. The team verified through review of indoctrination and training records that each of the six engineers had attended indoctrination, formal training classes or had reviewed and signed for all the training required for their positions as listed on their personal Indoctrination and Training Matrix and as described and required in Procedure Number 02-101.

When reviewing and verifying the training records of the engineers for Engineering Bulletin, EB-013, "Preparation, Review and Approval of Design Calculations," Revision 0, November 24, 1994, the team questioned the circumstances under which an Engineering Bulletin was issued. CVNO personnel stated that Engineering Bulletins were typically issued to document and discuss technical issues that the engineers use in performing their jobs. Procedure Number 03-104 stated that an "Engineering Bulletin is a document issued by Crane-Aloyco's Engineering Department for the purpose of informing operating personnel and/or customers of the design, material, methods or practices utilized by Crane-Aloyco."

The team noted that EB-013 contained required processing information for the preparation, review and approval of calculations which are integral to, or which support engineering, design and analysis. Based on the definition of an Engineering Bulletin contained in Procedure Number 03-104, "Preparation and Issue of Engineering Bulletins," Revision 1, May 1, 1992, the team suggested that EB-013 would be more appropriately classified as a Procedure because it implemented the QAM. CVNO personnel agreed to make this change.

3.5 Design Control

CVNO manufactured gate valves, globe valves, check valves and butterfly valves and supplied replacement parts. They repaired, but did not manufacture, power operated relief valves and safety relief valves.

The team reviewed QAM, Section III, "Design Control," Paragraph 4.0, "Design Basis and Design Inputs," which stated that, "The Owner's [customer's] Certified Design Specification and the requirements of the [ASME] Code shall be the basis for CVNO's design review for complete valve assemblies. For valves 4 in. nominal size and less, CVNO does not provide its own Design Specification as allowed by the [ASME] Code, but uses the Owner's Design Specification as basis for construction."

The team reviewed Procedure Number 03-103, "Control of Design Interface," Revision 2, July 14, 1992. This document provided design guidance for internal design interfaces as well as customer interfaces.

The team reviewed Engineering Bulletin EB-004, "Valve Parts Classification," Revision 0, April 10, 1989, which contained CVNO baseline design for all valves and valve parts by specifying the classification of each component. EB-004 stated that CVNO uses ASME Boiler and Pressure Vessel Code Case N-62-4 as the primary basis for classifying components of "Code Nuclear" and "Non-Code Nuclear Safety-Related valves." EB-004 placed all valve components into one of three classifications; "N", Nuclear, ASME Code Section III, Class 1, 2 & 3; "Q", Non-code, Nuclear Safety Related; and "C", Non-code, Non-Safety-Related Nuclear Power Plant Use.

The QA manager stated that when a purchase order was received from a customer, CVNO personnel reviewed the classification of the components with the customer to ensure that the customer was fully aware of the classification. If the customer required one or more components to be upgraded from "C" to "Q", the purchase order was revised to reflect the change. If special testing (e.g., seismic), was required, it was stated in the purchase order and the test report was included in the documentation furnished with the valve or components on delivery. Certified material test reports (CMTRs) provided with the product identified components individually and stated their classification individually. If a component was "N" or "Q", the CMTR included the heat number or lot number to provide traceability. "C" items had no heat or lot number.

The team reviewed four purchase orders from nuclear utilities to determine CVNO adherence to EB-004 classification and determined that in each case there was compliance with the requirements of EB-004.

The team reviewed Engineering Bulletin No. EB-005, "Stem Material Standardization," Revision 0, June 29, 1989, which recommended the use of ASTM A-564, Type 630, Condition H1150 (17-4PH), in the place of 410 and 304 stainless steels for stem and hinge pin applications. EB-005 also instituted 17-4PH, H1150, as the CVNO standard for such applications. The team considered the recommendation of 17-4PH usage relative to NRC Information Notice 92-60, "Valve Stem Failures Caused by Embrittlement," August 20, 1992. The Information Notice alerted addressees to the fact that 17-4PH, H900 through H1150, could become excessively embrittled on exposure to temperatures of 600°F and higher for a period of several thousand hours or longer as the result of secondary aging. Power operated relief valves and safety relief valves mounted on the pressurizer of a pressurized water reactor are the valves subjected to this type of service. The team determined that CVNO did not produce this type of valve and although they provided replacement parts for such valves, the replacement parts were supplied to the design specifications of the owner of the valves as they were less than 4 inches in nominal diameter. The team determined that CVNO application of 17-4PH materials was consistent with internal procedures and regulatory guidance.

The team determined that CVNO had established measures for the selection and review for suitability of application of materials, parts and processes essential to the safety-related functions of the components they provide to the nuclear power industry. The team also determined that CVNO had established measures that identified and controlled the various design interfaces.

3.6 10 CFR Part 21 Program

The team observed the posting of required documents in prominent places in the CVNO Joliet, IL, office and in the Romeoville, IL, facilities at 12 East Devonwood and 720 North Parkwood.

The team reviewed Procedure Number 15-100, "Reporting of Defects and Non-conformances (10CFR21)," Revision 2, October 29, 1993, and determined that it complied with the requirements of 10 CFR Part 21.

Although Procedure Number 15-100 was first issued January 10, 1986, no issues were entered into the Part 21 log until January 1994. The team made the observation that it was unusual that no items were entered into the log during the eight year period between January 1986 and January 1994. CVNO personnel made the point that during that period they were almost exclusively involved with producing replacement parts for existing valves on a small lot basis and thereby not likely to encounter conditions that would require evaluation or reporting. The team told CVNO personnel that it was as important to document issues that were raised, evaluated and not reported to the NRC as it was to document those issues that were reported to the NRC.

4 PERSONS CONTACTED

The NRC staff participating in the inspection; CVNO personnel contacted during the inspection; and the personnel attending the entrance and exit meetings are listed below. A bullet (•) indicates that person attended the entrance meeting and a dagger (†) indicates that person attended the exit meeting.

Crane Valves Nuclear Operations:

•	†	McCullough, H.G.	Manager, Engineering
•	†	Bisesto, F.J.	Director, Customer Service
•	†	Carlson, J.	Director, Operations
•	†	Landholt, W.W.	President, Valve Group

U.S. Nuclear Regulatory Commission:

•	†	Brewer, D.H.	Metallurgical Engineer, VIS/PSIB/DISP/NRR
	†	Cwalina, G.C.	Chief, VIS/PSIB/DISP/NRR
•	†	McIntyre, R.P.	Senior Reactor Engineer, VIS/PSIB/DISP/NRR

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

December 4, 1995

Mr. Joe R. Harrell, Manager
GE Nuclear Energy
Nuclear Field Services/Power Delivery Services
640 Freedom Business Center
King of Prussia, PA 19406

SUBJECT: NRC INSPECTION REPORT NO. 99901001/95-01

Dear Mr. Harrell:

This letter addresses the inspection of your organization's activities at the Philadelphia Service Shop of the GE Apparatus Service Division and at the Specialty Breaker Plant in Philadelphia, conducted September 11-14, 1995, by Mr. Stephen Alexander and Mr. Ronald Frahm, Jr., of this office, and the discussion of their findings with the members of your staff identified in the enclosed report at the conclusion of the inspection and in subsequent telephone conversations. The inspection was conducted to assess corrective actions implemented by your organization in response to quality assurance audit by customers and GE Nuclear Energy, San Jose, problems with circuit breaker maintenance, and problems with dedication of commercial grade items for use in safety-related applications. The inspectors also reviewed GE Nuclear Energy procedures adopted pursuant to Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21). Areas examined during the inspection and our findings are discussed in detail in the enclosed report. Within these areas, the inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

During this inspection, we determined that the implementation of your quality assurance (QA) program did not meet certain NRC requirements, specifically, Criterion XIII, "Handling, Storage, and Shipping," of Appendix B to 10 CFR Part 50. We observed unsafe handling practices in moving Magne-Blast circuit breakers in the service shop using overhead travelling hoists and found that adequate procedures to control this activity had not been established. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter. You are requested to 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.

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.

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, Public Law No. 96-511.

The cooperation of your staff and that of the GE Philadelphia Service Shop, and GE Specialty Breaker Plant in this matter was greatly appreciated. Should you have any questions about the enclosed report, we would be glad to discuss them with you.

Sincerely,

Robert M. Gallo

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

Docket No.: 99901001

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99901001/95-01

cc w/encl:
Mr. Forrest Hatch, Manager NS&PO Quality, GE NE, San Jose
Mr. Normand Roux, Manager, Switchgear Services, GE Philadelphia Service Shop
Mr. Edward Dugan, Chief Design Engineer, GE Specialty Breaker Plant

NOTICE OF NONCONFORMANCE

GE Nuclear Energy, Power Delivery Services
King of Prussia, Pennsylvania

Docket No. 99901001
Report No. 95-01

Based on the results of an NRC inspection conducted on September 11-14, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

A. Criterion XIII, "Handling, Storage, and Shipping," of Appendix B to 10 CFR Part 50, states, in part: "Measures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration."

Contrary to the above, PDS did not establish adequate procedures to control handling of Magne-Blast circuit breakers being moved about the service shop in a manner that would ensure that they would not be damaged. As a result, unsafe handling practices were used when moving Magne-Blast circuit breakers (without arc chutes) in the service shop using overhead travelling hoists.

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 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 action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 4th day of December, 1995

Enclosure 1

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

ORGANIZATION INSPECTED:	GE Nuclear Energy (GE NE) Nuclear Field Services/Power Delivery Services (PDS) 640 Freedom Business Center King of Prussia, PA 19406 (NRC Docket No. 99901001)
INSPECTION CONDUCTED AT:	GE Apparatus Service Division Philadelphia Service Shop 1040 East Erie Avenue Philadelphia, PA 19124 (NRC Docket No. 99901147) GE Specialty Breaker Plant 6901 Elmwood Avenue Philadelphia, PA 19142 (NRC Docket No. 99900219)
REPORT NO.:	99901001/95-01
PRINCIPAL CONTACT:	George Sanders, Lead Engineer, GE NE PDS 610-992-6049
CORRESPONDENCE ADDRESS:	640 Freedom Business Plaza King of Prussia, PA 19406
NUCLEAR INDUSTRY ACTIVITY:	Medium-voltage switchgear manufacturing (SBP), equipment maintenance and repair services (ASD) for electric power generation and distribution equipment
INSPECTION TEAM LEADER:	Stephen D. Alexander Vendor Inspection Section (VIS) Special Inspection Branch (SIB) — 12/1/95 Date
OTHER INSPECTORS:	Ronald K. Frahm, Jr. Quality Assurance and Maintenance Branch
REVIEWED:	Gregory C. Cwalina, Chief VIS SIB — 12/4/95 Date
APPROVED:	Robert M. Gallo, Chief, SIB — 12/4/95 Date

Enclosure 2

1.0 SUMMARY OF INSPECTION FINDINGS

1.1 Inspection Basis:

The inspection basis comprised the following:

- 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"
- General Electric Nuclear Energy (GENE) Quality Assurance (QA) Program Documents and Procedures Applicable to Power Delivery Services (PDS) Scope of Activities at the GE Apparatus Service Division (ASD) Philadelphia Service Shop, and at the GE Specialty Breaker Plant (SBP) in Philadelphia.

1.2 Violations:

None.

1.3 Nonconformance:

(99901001/95-01-01) Contrary to the requirements of Criterion XIII of 10 CFR Part 50, Appendix B, GE NE PDS did not establish adequate procedures to control handling of Magne-Blast circuit breakers being moved about the service shop in a manner that would ensure that they would not suffer inadvertent damage or cause injury to shop personnel. As a result, unsafe handling practices were used when moving Magne-Blast circuit breakers (without arc chutes) in the service shop using overhead travelling hoists. (See Paragraph 3.5 of this report).

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous findings were addressed during this inspection.

3.0 INSPECTION DETAILS

3.1 Inspection Objectives:

The approved inspection plan for this NRC Vendor Inspection Section inspection included the following objectives:

3.1.1 Evaluate PDS QA at ASD with emphasis on corrective action and improvements prompted by a GE NE (San Jose) QA audit and customer-identified deficiencies such as incorrect trip units in serviced breakers for Boston Edison Company's Pilgrim Nuclear Station.

3.1.2 Evaluate resolution of Magne-Blast circuit breaker problems (switch problems, failures to latch) identified at Maine Yankee and Millstone.

3.1.3 Review the joint PDS/SBP Magne-Blast design reconciliation to identify any changes about which licensees may need to be informed.

3.1.4 Review resolution of failures of Magne-Blast trip cranks at the Watts Bar Nuclear Plant and Shoreham Nuclear Plant.

3.1.5 Evaluate GE NE PDS program at SBP for dedication of commercial grade items (CGIs) for use by SBP in building or ASD in servicing safety-related equipment.

3.2 Magne-Blast Circuit Breaker Problems

The NRC investigation of the circuit breaker problems at Maine Yankee, Millstone and Pilgrim discussed below revealed manufacturing deficiencies in switches manufactured by GE Electrical Distribution and Control (ED&C), Plainville, Connecticut, deficiencies in workmanship and QA at ASD, Philadelphia, and deficiencies in the dedication process at the GE NE PDS commercial grade dedication facility at the GE SBP in Philadelphia. These deficiencies resulted in audits of ASD by Yankee Nuclear Services Division for Maine Yankee and audit of ASD and the PDS dedication facility at SBP by GE NE, San Jose. In addition, the NRC visited Maine Yankee, ED&C, and ASD and GE NE to obtain further information and observe testing and inspection. These visits resulted in issuing Information Notices 94-54, "Failures of General Electric Magne-Blast Circuit Breakers to Latch Closed," (August 1, 1994) and 95-02, "Problems With General Electric CR2940 Contact Blocks in Medium-Voltage Circuit Breakers," (January 17, 1995). During this inspection, the NRC inspectors reviewed PDS corrective actions and programmatic improvements made as a result of the identified problems and audit findings.

3.2.1 Interlock Switch Problems

On March 23, 1994, Maine Yankee Atomic Power company, the licensee for Maine Yankee, reported that during a pump surveillance test, 4.16-kV Magne-Blast breaker (Maine Yankee ID No. 3-17) for the high pressure safety injection (HPSI) pump motor (P-14B) failed to close. The affected breakers, GE Type AM-4.16-250-9HB, vertical-lift, medium-voltage (4.16-kV), had been overhauled on site by a team from GE ASD Philadelphia in the summer of 1993. The licensee investigated the incident and determined that one of the two circuit breaker cubicle interlock limit switches (comprising Interlock Switch Assembly 52-IS) that had been replaced during the overhaul had not been properly manufactured and installed. The affected switch (which enables the breaker closing circuit when the vertical-lift breaker is fully elevated in its cubicle) was found loose on its mounting and misaligned.

The two normally-open (shelf state) configuration CR2940U310 contact blocks that comprise interlock switch assembly 52/IS are ganged together such that they are both held closed by a breaker cubicle interlock mechanism lever when the vertical-lift breaker is fully elevated in its cubicle. However, in this instance, the hole drilled in the head of one of the two special mounting screws in the upper contact block (fastening it to its mounting bracket) had no female threads tapped into it. Therefore, inappropriately using the existing fasteners, instead of controlling the nonconforming screw and replacing it with a new, properly made screw, the new lower contact block could was mounted onto the upper block using only one screw. During subsequent operations of the breaker, the lower block became loose. Consequently, its plunger could not be depressed by the moving contact carrier of the upper contact block, thus it remained open, disabling the breaker closing circuit.

Following the HPSI pump breaker's failure to close on March 23, 1994, the licensee discussed the material and workmanship concerns with GE PDS and initiated a comprehensive program to reinspect all the safety-related Magne-Blast breakers. During subsequent breaker inspections, Maine Yankee and PDS discovered additional problems with mechanisms, including non-standard interlock paddles (spring steel in stead of rigid), and a damaged set screw in one crankshaft. The licensee identified some additional loose mounted power switches, interlock switches, and close-latch monitoring switches on several breakers. Additionally, other minor deficiencies, such as interference with the close-latch monitoring switch return spring and adjustment on the top positive interlock switch gaps, were also identified. The licensee corrected these deficient conditions and the breakers were tested satisfactorily.

In addition, on September 26, 1993, while the plant was shut down in preparation for the loss of power testing, an emergency diesel generator (EDG-1A) output circuit breaker (MY ID# 3-12) failed to close. The breaker closing spring was found uncharged. Troubleshooting revealed that the closing spring charging mechanism had failed to recharge its closing spring automatically when the breaker was last closed. The contacts of interlock switch assembly 52/IS used to enable the closing spring charging circuit this time (as opposed to the breaker remote closing circuit failure discussed above) were stuck open. The licensee corrected this problem by replacing the defective CR2940U310 contact block. Based on the deficiency identified in this EDG breaker, the licensee initiated a work order (WO No. 93-3293) to verify that all safety-related breakers' charging circuitry and all breakers positive interlock switches were functional. The licensee also verified that the breaker functioned adequately overall and the connections in the breakers' components were tight. The failure of the charging motor circuit switch (by sticking open) was believed to be similar to the failure of a power switch discussed below. No additional concerns were identified during this verification.

The licensee determined that during overhaul of the 4.16-kV circuit breakers on the site, ASD had installed several defective limit switches. In addition to some contact sticking problems, several special fasteners in the contact blocks used for the Maine Yankee breaker overhaul were found with incomplete threading and/or shallow screwdriver slots. The manufacturer, GE-ED&C, determined that the problems occurred when a screw making machine shut itself down due to a broken tool. On this occasion, a number of incompletely machined screws that passed through during the shutdown sequence were not captured, but instead became mixed with that batch of finished fasteners. ED&C corrected the problem and confirmed that the use of the defective screws was limited to a few CR2940 contact blocks with MA3XX= (1993) date codes. ED&C has reported establishing tighter controls to prevent recurrence, inspecting its stock, and weeding out any additional defective fasteners.

3.2.2 Power Switch Failures

On December 6, 1993, while the plant was at full power. Emergency feedwater pump breaker (GE Serial no. 0224A4126-017, Maine Yankee ID No. 4-34) failed to close during a routine surveillance test. The breaker failed to close in about one-third of the attempts to close it during shop testing. PDS sent a technician to Maine Yankee to assist in the failure analysis. During the overhaul of the 4160-Vac Class 1E breakers at Maine Yankee, ASD had replaced the over 10-year-old original CR2940 contact blocks used for power switches (in addition to the CR2940 interlock switch replacement discussed above). Further evaluation on this breaker identified a normally closed set of contacts in power switch assembly 52/SM-LS in the closing circuitry would stick in the open position. Power switch assembly 52/SM-LS is made up of three CR2940 contact blocks (two U310, normally open, and one U301, normally closed) ganged together. The three contact blocks are operated simultaneously by the closing spring charging mechanism. Each of the three contact blocks has one set of two contacts (numbered 1 through 6). The normally closed contacts 5 and 6 of power switch assembly 52/SM-LS are supposed to return to their normal closed state as the contact block's plunger is released (when the closing spring is fully charged) and enable the breaker closing circuit. The contact blocks of power switch assembly 52/SM-LS open as soon as closing spring charging mechanism leaves the fully charged condition. During subsequent breaker testing, the licensee identified one other faulty power switch on a spare circuit breaker (4-32).

Internal examination of the failed switches revealed that the moving contacts (5 and 6) of the normally-closed contact block were sometimes stuck in the open position. This appeared to result from mechanical interference that prevented reclosure of the contacts by their internal return springs. These observations raised the question of the reliability of at least this batch of new CR2940s (their date codes, e.g., "MA316=" indicated 1993 manufacture). The NRC pursued this issue with the CR2940 manufacturer, ED&C, the breaker manufacturer, SBP, and GE NE/PDS. ED&C eventually reported its determination confirming the inspectors' suspicions that the affected contact blocks came from a batch or batches that had some excess plastic injection mould flashing that was apparently interfering with the movement of the moving contact bar or its carrier assembly. GE ED&C has reported instituting initial corrective actions and preventive measures at its factory in Puerto Rico, but at the time of this inspection, had not yet provided definitive information on the scope of the problem; although there have been no further occurrences of this particular problem reported.

According to the breaker manufacturer, SBP, power switch 52/SM-LS contacts 5-6 are provided along with another accessory function called [trip] latch checking when this feature is supplied as an option. In some applications, such as at Palo Verde, contacts 5 and 6 provide a white light indication on the switchgear panel that the closing spring is charged and also that control power is available (hence the term "white light switch"). However, contacts 5 and 6 only perform a needed breaker control/interlock function when used in conjunction with a so-called "automatic reclosure" feature not used at Maine Yankee. According to the technical evaluation (455-93) by staff of Maine Yankee Plant Engineering Department, done in consultation with PDS and SBP as part of a modification screening pursuant to 10 CFR 50.59, contacts 5 and 6 are not required for a close-permissive function to inhibit closing attempts during closing spring charging, because the mechanism's design does not permit breaker closure before the closing spring is fully charged. Hence, contacts 5 and 6 have no required function at Maine Yankee, yet they can disable the breakers' electrical closing circuit should they stick open. Therefore, in consultation with GE, Maine Yankee determined that under these circumstances, it should not impair any required function of the breaker to "jumper them out" (short across or bypass them). According to the licensee's WO 94-42, the licensee eventually jumpered out the 52/SM-LS contacts 5 and 6 of 16 safety-related breakers.

Other problems reported with the CR2940 contact blocks in Magne-Blast breakers have been contacts 3 and 4 of power switch 52/SM-LS in the charging motor circuit becoming welded shut when interrupting charging motor current. In addition, new contact blocks occasionally have been found inoperative during acceptance testing (sometimes as part of commercial grade dedication), some due to misalignment or incorrect installation of internal parts. Other problems with the special screws such as shallow screwdriver slots also have been found as discussed above. At the request of GE NE/PDS, SBP is looking into using a different switch that is more reliable for these Magne-Blast applications, at least for nuclear safety-related service, and is planning to issue a Service Advice Letter (SAL) that will address these problems.

3.2.3 Breaker Failures to Latch Closed

During troubleshooting of the interlock switch problems at Maine Yankee, GE PDS and the licensee discovered a potential latching problem with the 4.16-kV circuit breaker mechanism. For example, breaker 4-34 failed to latch numerous times during subsequent testing after its power switch failure episode discussed above. The affected 250-MVA capacity Magne Blast breaker models were AM-4.16-250-6, 7, 8 or 9-HB. The suffix "H" denotes that the breakers were equipped with an "ML-13" operating mechanism, and the suffix "B" denotes that they were of the so-called "high momentary" design to achieve a 75-kA close-latch rating. This capability was added to 250-MVA breakers in 1975 (350-MVA breakers were already high momentary) to meet current ANSI/IEEE Standard C37.20. In order to achieve the 75-kA close latch rating, the breakers were fitted with heavier main and arcing contacts and heavier closing springs. In order to keep the breaker in proper balance, this required adding a second set of opening springs as well. However, the additional opening springs, along with some other improvements, made the breakers more sensitive to the various adjustments and variable parameters being at or near nominal design values in order to latch reliably. According to the GE SBP design engineer, a second prop spring was added in 1972 as a means of simplifying the process of set up and adjustment of the closely balanced, high-momentary breakers in the factory. The second prop spring made the action much less sensitive to all adjustable parameters being close to nominal in order to achieve reliable latching. With the faster prop action, the various adjustments needed only to be within tolerance without the concern that random tolerance stackup might cause the breaker occasionally to fail to latch closed. The occasional failure to latch could also occur in those 350-MVA capacity, AM-4.16 breakers (Model AM-4.16-350-1-H) all of which have two opening springs, but equipped with only one prop spring. The problem apparently does not affect newer breakers that have two prop springs. Over the years, during the overhauling of the breakers, GE retrofitted some breakers with two prop springs.

After some problems with prop spring end hook breakage, SBP issued SAL 073-(SBP prefix) 348.1 in December 1990, which provided information for obtaining an improved (gold colored) main prop spring, less subject to breakage, and an improved design for the mounting bracket (chamfering or radiussing the hole edges) that would cause less wear on spring hooks. SAL 348.1 recommended that all breakers' be modified as described in the SAL during a scheduled outage with special attention being applied to breakers approaching 2000 operations. However, although SAL 348.1 addressed breakers already fitted with the second, auxiliary prop spring, it did not mention the potential problems associated with only one prop spring under some conditions, nor did it recommend adding the second prop spring. The inspectors noted that SBP had issued SAL 351.1, dated June 23, 1994, which provided information and recommendations (including ordering information for the modification kit) for installation of the second prop spring in ML-13 and ML-13A Magne-Blast breaker mechanisms. In addition, in August 1995, SBP issued SAL 354.1 which filled a long-standing need for definitive lubrication guidance for Magne-Blast breakers. SAL 354.1 also included information on the failure to latch problems in high-momentary breakers with one prop spring engendered by replacing the TUF-LOC prop bushings with aluminum-bronze in conjunction with overhaul and relubrication which require installation of a second prop spring. SAL 351.1, on second prop spring installation, is referenced.

Analysis of the failures at Maine Yankee (and some additional similar failures at Millstone), determined that the problem began to occur intermittently following about 35-50 operations of affected breakers after the original "TUF-LOC" prop bushings were replaced with the new aluminum-bronze bushings in breakers which have (1) two opening springs and one prop spring and (2) which already had the rest of the mechanism TUF-LOC sleeve bearings or bushings replaced with aluminum-bronze bushings in accordance with SAL 318.1. After all bushings, including the prop bushings, are replaced, and the breaker is cleaned and lubricated, and, as experience has shown, after the wearing-in period mentioned above, the timing and force balances shift enough to make a high-momentary breaker with one prop spring highly sensitive to the combinations of tolerances in adjustments such that a second prop spring is mandatory to ensure reliable latching. Because of the observed delayed onset, the problem may not present itself until after the completion of post overhaul testing and receipt inspection and/or post installation/pre-operational testing. However, this testing can involve sufficient operations or cycles of the breaker to render it susceptible to the latch failure shortly after its release for plant operation.

Under the conditions in question, breakers sometimes fail to latch closed because the prop does not move fast enough to be in the proper position under the prop pins of the mechanism linkage as they descend during the closing cycle (whether manually or electrically initiated). When the prop moves too slowly relative to the motion of the descending prop pins and the pins miss the prop - fall in front of it - instead of landing on it, the linkage collapses and the main contacts fall open again. This event is also called "going trip-free."

3.2.4 Tests and Inspections of Breaker 4-34 at GE's Philadelphia Service Shop

On April 5, 1994, ASD technicians performed tests on the breaker 4-34 under the supervision of the Lead Electrical Engineer from the PDS. NRC inspectors and licensee staff observed these inspections and tests. During the testing, the breaker exhibited erratic, unreliable latching (only latched 6 times out of twenty attempts. When it did latch, it had inadequate prop wipe. During the course of the inspection and testing phase, ASD installed the second prop spring modification using a standard kit. After installing the second prop spring, the breaker latched 8 out of ten times; much better, but still not fully reliable. Upon further inspection, the GE NE PDS field engineer (based in King of Prussia, PA) noted that at least one of the opening springs had excessive preload on it. This was adjusted to be just within tolerance (deliberately not adjusted to nominal) and the breaker latched approximately 20 out of twenty tries, with prop wipe now well within tolerance, a strong predictor of continued reliable latching. As an experiment, with spring adjustment back within tolerance, the breaker was tested further with its second prop spring removed. Predictably, it latched reliably, but with marginal prop wipe. On the basis of these observations, it was predicted that Maine Yankee would likely find excessive preload on the opening springs (or too little preload or weakened closing springs) of breaker 6-53 that had required a stronger second prop spring (of the gold-colored type intended to be used only as a main prop spring) to make it latch reliably. The inspectors concluded that the technical problems with Maine Yankee's Magne-Blast breakers have been adequately resolved. GE's deficiencies in QA, workmanship, and commercial grade dedication revealed by the problems at Maine Yankee are addressed later in this report.

3.3 Magne-Blast Design Reconciliation Project

The lack of a clear recommendation from GE SBP for adding a second prop spring, while arguably not necessary at the time the modification was instituted in production breakers, pointed up the need for a review of the design change history of the Magne-Blast line to identify any other changes about which it would be prudent to notify NRC licensees. Following discussions of these concerns, GE PDS and SBP undertook a joint, comprehensive design reconciliation project for Magne-Blast breakers, intended also to confirm that all design changes (including materials and process changes) and their cumulative effects did not invalidate design basis considerations such as seismic qualification.

The result of this project was SAL 352.1, "Latest Design Configuration: GE AM Type Circuit Breakers and Medium Voltage Switchgear," issued by SBP on July 7, 1995. SAL 352.1 describes and explains 33 modifications and improvements made to Magne-Blast breakers over the years and provides references (e.g., SALs, manuals, etc.), ordering information and applicability data. All nuclear plants with GE medium-voltage switchgear are on the standard distribution list for SBP SALs, unlike most other GE product department SALs that go only to customers of record (most of whom were/are distributors or architect-engineering firms). During this inspection, the inspectors reviewed the project documents to identify any modifications or process changes about which licensees should be notified. The inspectors found that the joint PDS/SBP review was comprehensive and thorough.

In order to ensure that future design changes in Magne-Blast breakers get a contemporaneous review for impact on safety-related applications, performance and qualification, GE NE and GE SBP have instituted a process in which GE NE is given an opportunity to review proposed changes before they are implemented. The inspectors reviewed the PDS procedure SBO-DP-02 (Revision dated January 10, 1995), "Desktop Instruction for Design Change Review at the Specialty Breaker Operations." The procedure called for review Magne-Blast (Type AM and AMH) design change notices (DCNs) for impact on safety related functional performance (including qualification), dedication and maintenance, requiring the maintenance activities to be notified. Although not stated in the procedure, PDS stated that affected customers would also be notified of changes via SALs, SILs or other correspondence.

3.4 Incorrect Trip Units Installed in Pilgrim Breakers

In addition to the ASD workmanship and PDS QA concerns involving work on Maine Yankee breakers cited above, the inspectors investigated the circumstances surrounding a 10 CFR Part 21 notification (NRC Part 21 Log No. 94-265) from Boston Edison Company (BECo) regarding incorrect trip units installed by ASD in two spare GE AK-2A-50 low-voltage, metalclad circuit breakers. Under PDS QA controls and supervision, ASD had overhauled, upgraded, and dedicated the spare breakers for safety-related service at BECo's Pilgrim Nuclear Power Station (Pilgrim) per BECo purchase order (PO) No. RRR001864 to PDS, dated February 22, 1994. The PO specified that RMS-9 trip units should be installed in Pilgrim's AK-2A-50 breakers with an "LST1" trip characteristic indicating that they would have long-time and short-time trip functions. During shop testing of the breakers at Pilgrim prior to installation, BECo discovered that ASD had installed RMS9 trip units with an "LSIT1" trip characteristic, indicating long-time, short-time and instantaneous trip functions. These breakers were to be used as feeder breakers from 480-Vac bus "MCC B2" to "MCC B6." Had any of the affected breakers with an instantaneous trip function been installed in this application, the result could have been a loss of breaker trip coordination. An electrical fault in a load powered from a switchboard fed by one of the affected feeder breakers could have caused the feeder breaker to trip before the affected load breaker, resulting in the unnecessary loss of other loads, including safety-related equipment, on that switchboard.

The inspectors found that ASD received and inspected the breakers (serial numbers 256A9428-207-3EL and 256A9428-217-5EL) on March 15, 1994. On June 26, 1994, PDS issued a work authorizing document called an apparatus requisition (AR) to ASD for the overhaul and upgrade of the breakers. This AR included a summary of the technical and QA requirements and correctly reflected BECo's PO requirement for the LST1 trip characteristic in the RMS9 trip units to be installed in the breakers. PDS issued another AR for the replacement parts to ED&C, the GE product department for low-voltage breakers, on July 7, 1994, but the parts AR incorrectly specified control (catalog) number TK503T1604 for an LSIT1 RMS9 with the instantaneous trip function instead of the correct control number, TK503T1606, corresponding to an RMS9 with the LST1 trip function. PDS QA personnel at ASD received and inspected the incorrect LSIT1 RMS-9 kits on August 15, 1994, but did not detect the error because of the erroneous control number on the parts AR, against which PDS receipt-inspected the trip unit kits. ASD installed the incorrect LSIT1 RMS9 trip unit kits and then tested them satisfactorily as if they were supposed to have instantaneous trip functions. PDS shipped the two breakers to Pilgrim with GE NE product quality certifications on August 31 and September 1, 1994 respectively. When BECo discovered the error on September 19th, it returned the breakers to ASD for rework. PDS had ASD install the proper replacement LST1 RMS9 trip units and, after satisfactory completion of post-installation/dedication testing, shipped the breakers back to Pilgrim on September 23, 1994.

PDS initiated nonconformance report number 1EH4M-05 on September 19, 1994, to document the deficiency. The cause of the nonconformance was identified as the "1" in "LST1" being mistaken by the person ordering parts (and by the ASD shop technicians) as an "I" for instantaneous, even though the I is supposed to come before the T in the standard trip characteristic designation format for a long-time, short-time, instantaneous trip unit, i.e., LSIT1. The disposition was to rework the breakers by installing the LST1 RMS9 trip units after dedicating them to the applicable dedication specification. The action to prevent recurrence was to establish the practice of ensuring that ASD personnel received legible copies of the actual customer POs. In addition, PDS completed a 2-hour training session on October 26, 1994, which included a discussion of attention to detail, verbatim compliance, and a review of the applicable sections of the QA manual.

BECo Supplier Finding Report 94-47, issued to PDS, addressed this issue to assure its resolution. In response to this finding, PDS committed to have the QC Supervisor verify both parts requisitions and work authorizing requisitions in the future to assure conformance to the purchase order requirements. In addition, the QC Supervisor would verify that parts conform to the PO requirements at receipt inspection. Also, the work authorizing documents would be more specific to detail the circuit breaker overhaul process. The work authorizing document is the basis from which the work controlling documents (travellers) are generated. Finally, the Project Manager and QC Supervisor's review of the travellers would require the documentation to be of sufficient detail to insure the purchase order requirements are appropriately delineated with a requirement for verification. PDS did not change any procedures to document this process because it was already described in existing procedures.

The inspectors reviewed several recent purchase orders and their associated support documentation (i.e. parts requisitions, work authorizing requisitions, travellers, and PQCs) and found them to be of adequate detail and to correctly reflect the purchase order requirements. The inspectors concluded that PDS had taken adequate corrective action and preventive measures, particularly in view of the training conducted on the lessons learned. Accordingly, the vendor's action is considered satisfactory for closing out the Part 21 Report.

3.5 QA Corrective Action

During the inspection of this area, the inspectors toured the nuclear shop area, segregated from the general shop floor and dedicated to work on nuclear safety-related low and medium-voltage breakers. While looking at jobs in progress, examining parts, equipment, and documentation, and interviewing technicians, the inspectors observed a Magne-Blast breaker frame and mechanism assembly (without its arc chutes) being lifted by an overhead travelling hoist and moved about the shop floor. The inspectors noted that the hooks on the ends of the chain lifting bridle were not engaged with the lifting holes provided in the upper horizontal frame members, but rather had simply been hooked under the edges of those frame members. When asked, the technician performing this operation stated that the assembly was out of balance without its arc chutes so that using the designed lifting points would not lift the assembly straight. The inspectors noted that if the otherwise unrestrained hooks were to slide along the frame members should the assembly tip during transit, they could trip at the ends, come loose, dropping the assembly or at least loosing control of its attitude. In either case, the breaker may be damaged and cause injury to personnel. No attempt had been made to use a third lifting point, for example with lifting strap(s) in conjunction with the designed safe main lifting points to compensate for the unbalanced condition while maintaining positive control over the assembly. This situation was brought to the attention of the shop supervisor and management.

The inspectors asked to see the procedural guidance governing this type of operation and were shown the only such written ASD guidance said to be available which was GE NE Plant Services & Projects Department Procedure SBO-002, "Shipping, Handling, & Storage" (Revision 1, June 7, 1993). Paragraph 4.2.1 of this procedure required that items be handled "...in a manner which will prevent physical damage and preserve the quality of the item and the container." Paragraph 4.2.4 stated, in part: "The weight, lifting point or center of gravity indicated on the crate, skid or package by the shipper shall be utilized to ensure proper handling during unload, transfer between carriers, and loading. Although the general admonishment of Paragraph 4.2.1 to handle items in a safe manner could be thought to apply to moving heavy objects around in the shop, the more specific requirement in 4.2.4 on use of proper lifting points clearly applies to other handling situations. There was no other guidance offered that addressed the unsafe practice observed. Therefore, the inspectors concluded that contrary to the requirements of Criterion XIII, "Handling, Storage and Shipping," of 10 CFR Part 50, Appendix B, GE NE PDS had not established adequate procedures, nor, in the absence of procedures specifically governing intra-shop handling practices, had PDS ensured proper interpretation of and compliance with the intent of existing handling, storage, and shipping procedure SBO-002, which could be deemed to be implicitly applicable, to control handling of Magne-Blast circuit breakers being moved about the service shop in a manner that would ensure that they would not be damaged. The inspectors noted that the unsafe handling practices observed were also an occupational safety and health concern. Accordingly, the unsafe handling practice is cited as Nonconformance 95-01-01.

3.6 Failures of Magne-Blast Trip Coil Cranks

Magne Blast breaker trip cranks had failed on several occasions at the Tennessee Valley Authority's (TVA's) Watts Bar Nuclear Plant (WBN) and one additional failure had been reported at the Shoreham plant. The failure reports to the NRC pursuant to 10 CFR Part 21 were being tracked under NRC Part 21 Report Log No. 93-304. The trip crank is a piece part of the Magne-Blast ML-13 operating mechanism, GE Part Number 105C9316G001. The crank is a stamped, formed and drilled steel plate with a steel pin socket-welded into it (See Figure 1 of Appendix A to this report). The function of the trip crank is to convert the linear force of the trip coil armature or plunger to the torque needed to rotate the breaker trip shaft. Failure of the trip crank could prevent any electrical trip of the breaker. The known failure mode is fracture of the weld that fastens the trip coil armature link pin to the plate portion of the crank under shock loading shear reaction stress on the pin.

In response to the original failures, GE SBP changed the manufacturing process for the crank by requiring some reinforcement (on the weld fastening the pin to the plate) to be left on the back side of the plate and not be ground off. Now specified on the ML-13 drawing (SBP Drawing No. 0105C9316/DCN 083-88-003, May 18, 1988, View "A" weld detail for Group 001 Trip Crank) is: "1/32 TO 1/16 WELD TO REMAIN AFTER GRINDING FLAT SURFACE, <u>DO NOT GRIND FLUSH</u>." The inspectors examined a crank that was in stock at ASD, but tagged for use in non-safety-related applications only. The crank plate was bent at the end and the pin was cocked about 10 to 15 degrees. It appeared that this crank would very likely fail in service, albeit presumably in non-safety-related service, but fail nonetheless.

The inspectors toured the SBP production floor, including the area where the pins are welded to the trip cranks. The inspectors examined the setup and interviewed one of the factory workers who performed the welding on this particular part (among others). The factory worker described and partially demonstrated how he did the welding in terms of setup, but admitted that he had no fixture to ensure that proper penetration and alignment of the pin was maintained and it was not clear that he did any inspection of the pins before use, but welded them into the cranks as they were obtained from a part bin. He intimated that the person who likely welded most of the failed trip cranks and ground the reinforcement flush had allowed some "cold welds," i.e. lack of fusion, and probably lack of adequate penetration as well, depending largely on the conformance of the pins to the drawing.

During the tour of the ASD nuclear work area, the inspectors noted that the ML-13 mechanisms of two AM-4.16 breakers undergoing initial inspection and disassembly had trip cranks that were susceptible to failure with their weld reinforcements ground off and the transition or interface between the pin and the plate clearly visible. The inspector noted that the technician performing the inspection and teardown had not documented this condition in the inspection record, but PDS stated that the technicians were aware of the problem and that in this case, the technician had not come to that portion of his inspection yet. PDS confirmed subsequent to the inspection that the trip cranks in question had been replaced and the customer notified with recommendations for inspection of other breakers.

To ensure that faulty trip cranks are not used in safety-related breakers, PDS is now relying on the 1988 design change and its dedication process to screen out improperly fabricated cranks, as evidenced by, for example, the fact that the unsatisfactory crank found by the inspector had been rejected for nuclear use. Although the amount of chamfer on the small end of the pin that is inserted into the plate and the penetration depth allowed by the depth to which the shoulder is machined into the pin both significantly affect the amount of pin surface area in contact with the weld filler material, these attributes are not specified to be checked during manufacture, as they cannot be seen afterward. The dedication at the time of this inspection relied solely on a 100-lbf static load test and the presence of 1/32 to 1/16 inch of pin weld reinforcement on the back side of the plate. With respect to faulty cranks in the field, SBP is considering issuing a SAL on this problem and in the mean time PDS has put out instructions to the service shops to inspect the cranks as breakers come through for service and replace the cranks as necessary. The inspectors concluded that the vendor's action thus far was satisfactory in resolving this issue, but that it would be complete only with the issuance of a SAL.

3.7 <u>Evaluation of GE NE PDS Program at SBP for Dedication of Commercial Grade Items</u>

GE NE PDS reported upgrading its process for dedicating the commercial grade components used in manufacturing and overhauling safety-related breakers. During this inspection, the inspectors addressed the problems with switch dedication by the PDS dedication facility at SBP.

3.7.1 Program Procedures Review

The program and process of dedication of commercial grade items (CGIs) at all GE NE dedication facilities in general is described and prescribed by a hierarchy of procedures under the GE NE nuclear quality assurance program. The highest tier GE NE procedure dedicated to dedication is Engineering Operating Procedure (EOP) 65.2.20, "Dedication of Commercial Grade Items. The Vendor Inspection Branch had reviewed this procedure extensively in the past during inspections in San Jose. Accordingly, during this inspection it was given a cursory review that confirmed no substantive changes since the last formal review by the NRC. Nevertheless, the inspectors reviewed the latest effective revision of EOP 65-2.20, Revision 5, dated July 27, 1994.

In the system in use at GE NE's dedication facility at SBP, the next tier of dedication procedures are called dedication specifications. The inspectors reviewed those pertinent to CGIs of particular focus during this inspection; namely (1) No. 24A1113, "Dedication of Magne-Blast Switchgear Parts" (Revision 2, dated November 30, 1994), (2) No. 24A1114, "Dedication Specification for Hardware and Materials" (Revision 0, dated June 4, 1993). In addition, local procedures for general dedication guidance were reviewed: (3) SBO-DP-01, "Desktop Instruction for Dedication of Material at the Specialty Breaker Operations" (revision dated May 3, 1993), (4) SBO-DP-03, "Desktop Instruction for Dedication/Inspection Tolerances at the Specialty Breaker Operation" (revision dated January 17, 1995), and (5) No. 24A1115, "Dedication Specification for ED&C Low Voltage Switchgear Parts," Revision 0, April 25, 1992.

3.7.2 Dedication package review

The inspectors reviewed several dedication packages at SBP for replacement parts to be supplied to ASD for installation in overhauled/modified circuit breakers or to GENE for direct shipment as spare/replacement parts to the licensees. They included switches, shafts, clutches, trip cranks, and secondary couplers. In general, the critical characteristics were found to be properly identified and verified (with the few exceptions noted below) to assure that the components would perform their intended safety function under all design basis conditions and not fail in a manner adverse to safety. In particular, the inspectors reviewed the dedication specifications, critical characteristics matrices and verification records for recently dedicated CR2940 contact blocks (both the NO and NC types), Magne-Blast breaker trip cranks, and secondary couplers. The verification records included: the initiating AR, and for each line item; a traveler, CC forms, dedication check sheets, instrument use and calibration log sheets, etc.. In addition, the dedication specification for ED&C low-voltage switchgear parts, No. 24A1115 (Revision 0, dated April 25, 1992) was reviewed for technical adequacy even though the recent problems with incorrect AK breaker trip units (Pilgrim) discussed above were related to identification of correct material during procurement as opposed to adequacy of critical characteristic verification which was not in question at Pilgrim.

3.7.2.1 Critical Characteristics Matrix Form No. 00181, Revision 1 (no revision date), for "microswitch (N.C.)," part no. Q0456A0866P006, was being used to specify and document the verification of critical characteristics for the dedication of CR2940U301 contact blocks (which are normally closed (NC) in their shelf state) for use in Magne-Blast ML-13 operating mechanisms. In Magne-Blast breakers, this NC contact block is used solely for power switch 52/SM-LS, contacts 5 and 6 discussed above. The inspector noted that the selection of critical characteristics appeared to be reasonably complete, but pointed out several weaknesses that could reduce the effectiveness or usefulness of the matrix.

- The description block described the item as a normally **open** control switch which it is not instead of normally closed which it is (also inconsistent with the brief part description in the header of the form).
- The function was described as "part of the breaker control circuitry" instead of, for example, "contacts 5 and 6 of power switch assembly 52/SM-LS as shown on "Typical Wiring Diagram," in Instruction Book GEI-88671. Returns to its normal closed state when its plunger is released when closing spring is fully charged and enables closing spring release circuit. Opens as soon as closing spring charging mechanism leaves fully charged condition.
- The failure mode and effect was characterized as "loss of control circuitry" instead of circuit function; or more specifically, an open causing inability to close the circuit breaker remotely, a short or ground possibly causing loss of 125-Vdc control power.

- Under the critical characteristic or inspection attribute of "General Appearance," the requirement was to "Verify [visually] the existence of internal threads and thread sizes;" although the fasteners or screws were not mentioned.

- Under "Check Number of Contacts-1," the method of examination was "visual;" yet the contacts are hidden from view. The only visible attribute indicating the number of contact (sets) is that the block should have only two terminals.

- Under "Operation-Continuity," the method of examination/test was given as "Operate switch and verify open/close function," but it was not stated how this was to be done.

- Dielectric strength was listed as a critical characteristic, but the inspector questioned the appropriateness or usefulness of a 1500-volt high potential test for one minute on a tiny contact block only designed for 120 Vac or 125 Vdc when insulation resistance measurement at 500 or 250 Vdc would be more than adequate. Also, testing terminals to ground and line to load while open (i.e., across open contacts) was not specified. In general, the acceptance criteria were given as part of the critical characteristic, but for dielectric strength, the acceptance criterion (typically given in terms of maximum allowable leakage current or stated as no breakdown) was not stated.

3.7.2.2 Critical Characteristics Matrix Form No. 00296, Revision 1 (no revision date), for "Crank," Piece Part of ML-13 Mechanism, GE Part Number 105C9316G001, was being used to specify and document the verification of critical characteristics for the dedication of CR2940U310 contact blocks for use in Magne-Blast ML-13 operating mechanisms.

- The failure effect was described as "could prevent any electrical trip of the breaker." A known failure mode, fracture of the pin weld under shock loading shear reaction stress on the pin, was not mentioned.

- No NDE except visual.

- Weld proof test was specified as a 100-lbf minimum static ("dead") load applied "normal to axis." The inspector questioned whether this test effectively simulated or was reasonably equivalent to the severe shock loading reaction stresses generated during operation. The breaker design engineer stated that he had also tested production trip cranks by operation on a shop test breaker and would discuss this method with the onsite PDS staff for dedication purposes.

- Also, the amount of chamfer on the small end of the pin that is inserted into the plate and the penetration depth allowed by the depth to which the shoulder is machined into the pin both significantly affect the amount of pin surface area in contact with the weld filler material, yet these attributes were not specified to be checked during manufacture, and they cannot be seen afterward. As stated above, the dedication relies solely on the pull test and the presence of 1/32 to 1/16 of weld reinforcement on the back side of the plate.

3.7.2.3 Critical Characteristics Matrix Form No. 00315, Revision 0 (no revision date), for "Secondary Disconnect," Piece Part of ML-13 Mechanism, GE Part Number 108B1931G001, was being used to specify and document the verification of critical characteristics for the dedication of secondary coupler blocks for connecting the control cable from Magne-Blast breaker cubicle to the ML-13 Magne-Blast operating mechanisms. The only comment on this matrix form was the lack of insulation resistance measurements or dielectric withstand test. PDS explained that at the time the revision of this matrix form was in use, PDS used material verification and surface finish as attributes to ensure that the electrical characteristics would be acceptable as an alternative to insulation resistance or dielectric withstand testing. Current matrices, however, have added the dielectric withstand or Hi-Pot test.

The inspectors concluded that with the improvements recently implemented (such as revising sample sizes), the commercial grade dedication activities of PDS at SBP (with the minor exceptions discussed above) were in general being conducted consistent with the intent of 10 CFR Part 50, Appendix B, NRC Generic Letters 89-02 and 91-05, EPRI Report NP-5652, and GE NE procedures.

3.8 Review of GE NE Part 21 Procedures

The inspectors asked to see the procedures adopted by PDS pursuant 10 CFR 21.21(a) and were given a copy of GE Nuclear Energy Policy and Procedures Manual, NEDE-31746, Procedure 70-42, Revision Dated August 1994, "Reporting of Defects and Noncompliances Under 10 CFR Part 21." The inspectors reviewed this procedure with the following comments:

3.8.1 After completing their review, the inspectors learned that this was the same procedure that had been the subject of considerable discussion during a previous NRC inspection AT GE NE, San Jose, and had been substantially revised to address NRC concerns. Although these particular inspectors had not seen this procedure before, they were familiar with the issues previously discussed and noted that the procedure appeared to have been adequately revised to address the concerns raised in previous NRC inspections of GE NE at San Jose. However, the inspectors did identify certain sections in which the language of the procedure was not consistent with the applicable requirements of Part 21.

3.8.2 Paragraph 3.1, "Discovery of a PSC," required employees who become aware of deviations **that may be potential safety concerns** (PSCs) to advise the Safety Evaluation Program (SEP) Project Manager in writing. However, §21.21 requires that deviations in basic components delivered to or offered for use at NRC-licensed facilities be evaluated to identify defects, not just deviations that may be potential safety concerns. That is, all deviations are supposed to be evaluated to determine if they are safety concerns. To add the qualification, "...that may be potential safety concerns," to this provision of the procedure may act effectively to screen out deviations from consideration by SEP to determine even if they are PSCs because it calls for a conclusion by the employee that he or she may not be qualified or empowered to draw.

3.8.3 Procedure 70-42 did not address **failures to comply** (as defined in 10 CFR Part 21, §21.3, and as distinguished from deviations) which must also be evaluated per §21.21(a) to determine if they are associated with a substantial safety hazard, or reported to affected licensees or purchasers per §21.21(b). A basic component delivered to or offered for use at an NRC-licensed facility by GE NE PDS could **fail to comply** in some respect with the Atomic Energy Act of 1954, as amended, or any rule, regulation, license, or order of the Commission (NRC) without being a deviation, per se, i.e., a departure from a technical procurement specification, yet still require evaluation under §21.21(a) or reporting to customers under §21.21(b).

3.8.4 Paragraph 4.3, "Transfer of Information," provided for the customer notification discussed in §21.21(b), but the language used in Paragraph 4.3 was not fully consistent with §21.21(b) and, in the judgement of the inspector, would not ensure that the intent of §21.21(b) would be met in all cases. Section 21.21(b) requires that affected licensees or purchasers (meaning all affected licensees or purchasers of whom the vendor should have reason to be aware) be notified of deviations and failures to comply that the vendor determines it is unable to evaluate per §21.21(a). However, the language of Paragraph 4.3: "...to **the licensee or purchaser who has knowledge of the application** of the deficient product," instead of simply, "affected licensees or purchasers," as stated in §21.21(b) effectively restricted the disbursement of the information to a narrower scope than that intended by §21.21(b). Although this provision (§21.21(b)) is not currently required by Part 21 to be part of the procedures adopted pursuant to the regulation, in order to be consistent with the intent of §21.21(b), the procedure would need to require the vendor to identify from its records all licensees or purchasers who could be affected by the deviation or failure to comply, and notify all of them within the five working days of discovery prescribed in the regulation.

3.8.5 Paragraph 4.7, "Reportable Condition," required that if the evaluation concludes that a **defect** exists, the Responsible Officer be notified within five working days, but it did not address notifying the responsible officer of a **failure to comply** that an evaluation might have determined was associated with a substantial safety hazard.

Previously this procedure addressed conditions characterized as those that could create a substantial safety hazard or lead to exceeding a technical specification safety limit, "i.e., a defect." Although failures to comply were not explicitly mentioned per se, the description of the conditions of interest could be deemed to include failures to comply in that the failures to comply that ultimately are to be reported to the NRC under Part 21 (within the scope of provisions required to be part of procedures by §21.21(a)) are those that could be associated with a substantial safety hazard. However, the inclusive nature of the language of the procedure was negated in Paragraph 4.7 by departing from the inclusive language used previously in the procedure and instead only mentioning defects. The problem that could arise is that following the procedure as written verbatim without reference to Part 21, could lead to a situation in which a condition (PSC or even [illegible] that could create a substantial safety hazard might not be considered, let alone reported to the responsible officer, because it did not start out as a deviation, i.e., a departure from a technical procurement specification, but might still have been in some respect in noncompliance with the Atomic Energy Act of 1954, as amended, or any rule, regulation, order, or license of the Commission.

In addition, the procedure then stated, in part: "If the PRC is determined to be a reportable defect by the Responsible Officer, the SEP Project Manager shall notify the NRC...." The language here implied that even after the evaluation has concluded that the PRC is a defect, and hence is by definition reportable to the NRC, the responsible officer would make a determination of reportability separate from the conclusion of the evaluation that the PRC was a defect. However, Part 21 requires that defects (and failures to comply associated with substantial safety hazards) as identified on the basis of an evaluation, be reported to a director or responsible officer who then must cause them to be reported, without need for further evaluation, to the NRC.

The inspector discussed his concerns with the language used in Procedure 70-42 with PDS representatives, with GE NE's Manager of NS & PO Quality, and with the staff of GE NE SEP cognizant over this procedure. The inspector explained that the observations discussed above were based on the NRC staff's positions on Part 21 requirements, and interpretations of those requirements where necessary by the NRC Office of General Counsel. The procedural deficiencies identified were considered **minor violations** and as such would not be cited as violations in accordance with the NRC's Enforcement Policy, NUREG 1600.

4.0 PERSONNEL CONTACTED

Charles Taylor, Project Manager, GE-ASD
Edward Dugan, Design Engineer, GE-SBP (215-726-2316)
Brian Kennedy, QA Supervisor, GE NE
Joseph La Clair, QA Engineer, GE NE, PDS
George Wetsell, Manager, Plant Services and PM&C Quality, GE NE
Forrest Hatch, Manager, NS&PO Quality, GE NE
George Sanders, Lead Engineer, GE NE PDS
Charlie Vickers, Project Manager, GE NE (ASD ext. 657/8)
Charles Rodgers, Field Representative, GE NE PDS
Brian Forrest, ASD Project Manager (215-289-0400)
Richard A. Thielking, ASD Switchgear Specialist
Tom Connolly, Manager, Northeast Region, GE NE PDS

Appendix

A. Trip Crank for GE Magne-Blast Circuit Breakers

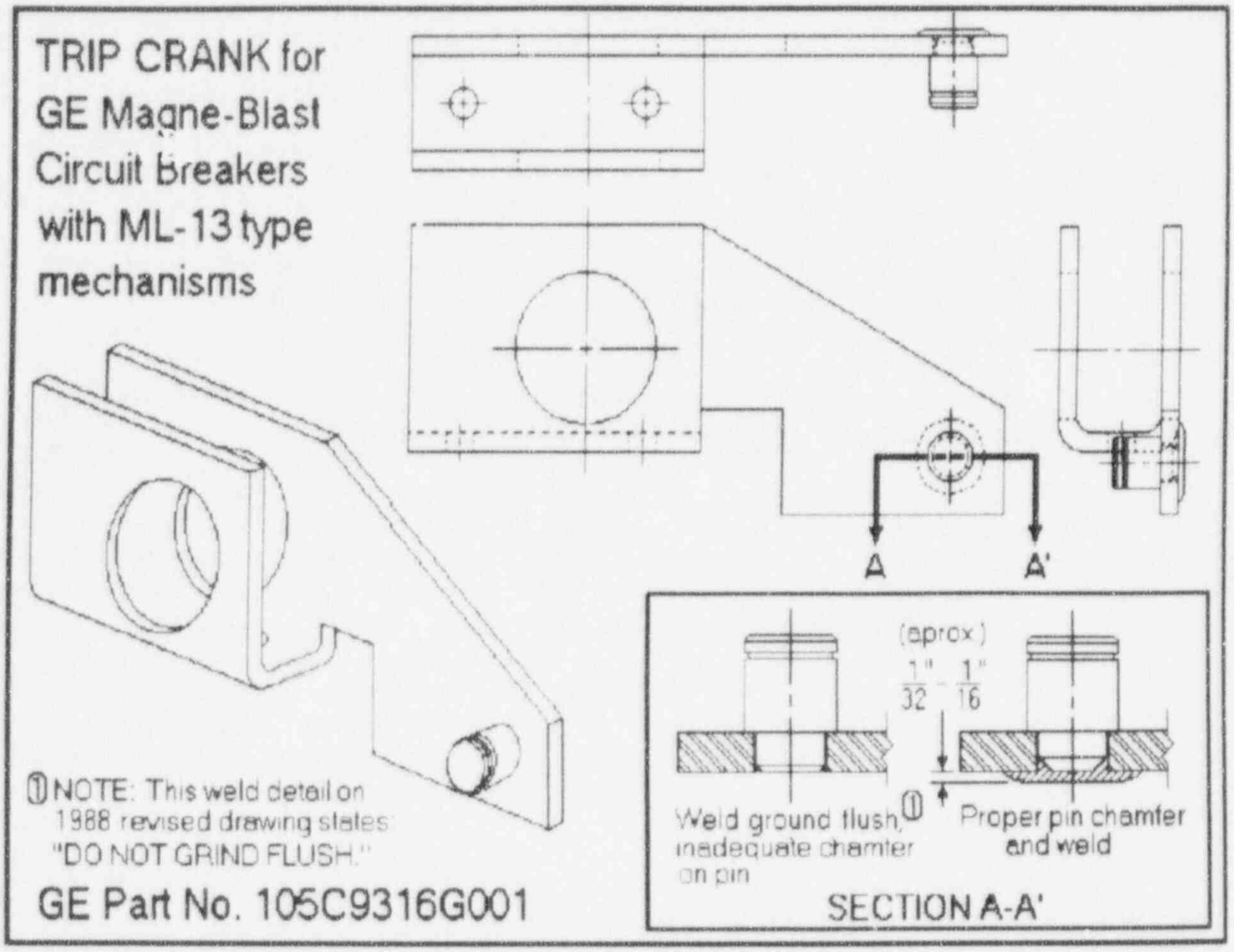
TRIP CRANK for
GE Magne-Blast
Circuit Breakers
with ML-13 type
mechanisms
A
A'
(aprox.)
1"/32 - 1"/16
①NOTE: This weld detail on
1988 revised drawing states:
"DO NOT GRIND FLUSH."
GE Part No. 105C9316G001
Weld ground flush, ①
inadequate chamfer
on pin
Proper pin chamfer
and weld
SECTION A-A'

Figure 1

19

November 22, 1995

Mr. Richard G. Knoblock, President
Pacific Scientific Company
HTL/Kin-Tech Division
22715 Savi Ranch Parkway
P.O. Box 87019
Yorba Linda, CA 92687-8719

SUBJECT: NRC INSPECTION NO. 99900255/95-01

Dear Mr. Knoblock:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Pacific Scientific Company, HTL Kin-Tech Division, conducted by Robert L. Pettis, Jr. and Billy Rogers, of this office, on September 25-28, 1995. The NRC inspection team conducted an evaluation of the Pacific Scientific quality program and the implementation of that program as it relates to the manufacture of mechanical shock suppressors and spare parts supplied to the nuclear industry as safety-related.

The NRC inspection team reviewed documentation, procedures, and representative records, conducted interviews and held discussions with members of your staff. On the basis of this inspection, the inspection team determined that the implementation of the Pacific Scientific quality assurance program failed to meet certain requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations and 10 CFR Part 21.

The enclosed inspection report contains a detailed discussion of the areas examined.

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

Should you have any questions concerning this report, we will be pleased to discuss them with you. Thank you for your cooperation.

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

Docket No.: 99900255

Enclosures: 1. Notice of Nonconformance
2. Inspection Report No. 99900255/95-01

NOTICE OF NONCONFORMANCE

Pacific Scientific Company
Yorba Linda, California

Docket No.: 99900255

Based on the results of an NRC inspection conducted at the Yorba Linda, California, facility of Pacific Scientific on September 25 through 28, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

I. Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

   Contrary to the above, Pacific Scientific Company (PSC) did not have a procedure in place for the replacement of commercial grade parts with safety-related. Such parts were routinely changed out during the process in which surplus snubbers, purchased from material brokers, were refurbished and sold as new to licensees.
   (95-01-01)

II. Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

   Pacific Scientific Standard Operating Procedure 07.104, "Supplier Approval," dated May 18, 1995, stated that all suppliers who furnish material or supplies for use in a saleable product shall be listed on the applicable PSC Approved Suppliers List (ASL).

   Contrary to the above, Walden Industrial Supply Company and Vabcor Inc., companies from which PSC purchased surplus market snubbers for resale to licensees as safety-related, were not listed on the PSC ASL. (95-01-02)

III. Criterion I of Appendix B to 10 CFR Part 50, "Organization," states, in part, that the authorities and duties of persons performing quality assurance activities be clearly established and delineated in writing. These activities include the quality assurance functions of assuring that an appropriate quality assurance program is established and effectively executed and verifying that activities affecting safety-related functions have been correctly performed.

Enclosure 1

PSC Quality Assurance Manual Section 0.20, "Management Statement," states that the Manager of Quality Assurance is given the responsibility and authority to maintain the Quality Assurance program and its implementation. Further, the Manager of Quality Assurance is given the freedom to identify quality assurance problems, initiate actions which results in solutions, verify implementation of solutions to those problems, and the authority to stop any work which violates any provisions of the Quality Assurance Manual.

Contrary to the above, during the period of approximately 1992 through 1994, the PSC activities involving the purchasing, receiving inspection, disassembly, parts replacement, reassembly, label plate replacement, and testing of snubbers (activities affecting safety-related functions) purchased on the surplus market and resold to licensees as safety-related, an activity affecting quality, occurred without the quality oversight or knowledge of PSC's Quality Assurance Manager and, as a result, the Quality Assurance Manager was not able assure that an appropriate quality assurance program had been established and effectively executed or verify that activities affecting safety-related functions have been correctly performed. (95-01-03)

IV. Criterion XVI of Appendix B to 10 CFR Part 50, "Corrective Action," states, in part, that 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.

Pacific Scientific QAM, Section 11 "Nonconformance and Corrective Action," Revision 6, dated December 15, 1993, requires the documentation of corrective action on a Failure Analysis and Corrective Action Report.

Contrary to the above, PSC did not complete a Failure Analysis and Corrective Action Report, as required by PSC procedure, for a PSA Model 100 snubber which failed an activation test. The snubber, purchased as surplus material from Vabcor Inc., was part of an order for Pennsylvania Power & Light's Susquehanna Steam Electric Station. (95-01-04)

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) a description of the steps that have been or will be taken to correct these items; (2) a description of the 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 day of November, 1995

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

REPORT NO.: 99900255/95-01

ORGANIZATION: Richard G. Knoblock, President
Pacific Scientific Company
HTL/Kin-Tech Division
22715 Savi Ranch Parkway
P.O. Box 87019
Yorba Linda, California 92687-8719

ORGANIZATIONAL CONTACT: Steve Palm
Quality Assurance Manager

NUCLEAR INDUSTRY ACTIVITY: Manufacturer of various model mechanical shock suppressors supplied as safety-related.

INSPECTION DATES: September 25-28, 1995

LEAD INSPECTOR: Robert Pettis Jr. 11/13/95

Robert L. Pettis, Jr., P.E. Date
Vendor Inspection Section (VIS)
Special Inspection Branch (SIB)
Division of Inspection
and Support Programs (DISP)
Office of Nuclear Reactor Regulation (NRR)

OTHER INSPECTORS: Billy H. Rogers, VIS/SIB/DISP/NRR

REVIEWED BY: Gregory C. Cwalina 11/17/95

Gregory C. Cwalina, Chief Date
VIS/SIB/DISP/NRR

APPROVED BY: Robert M. Gallo 11/22/95

Robert M. Gallo, Chief Date
SIB/DISP/NRR

Enclosure 2

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspection team evaluated the implementation of the Pacific Scientific Company (PSC) quality assurance (QA) program related to the manufacture of various model mechanical shock suppressors (snubbers) supplied to the nuclear industry as safety-related. The inspection was conducted to determine PSC's compliance with the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B) and the provisions of 10 CFR Part 21 (Part 21). The inspection team reviewed technical information, procedures and representative records, conducted interviews, held discussions and observed manufacturing activities.

1.1 Violation

Contrary to 10 CFR 21.21, which states that corporations subject to the regulations must adopt appropriate procedures to include specific requirements related to the length of evaluation of deviations, preparation and submittal of interim reports, and the informing of a director or responsible officer that a defect or failure to comply exists, PSC did not include the requirements of 10 CFR Part 21.21 in Standard Operating Procedure (SOP) 01.07, "Compliance With 10CFR21," dated August 10, 1993. (Non-Cited Violation)

1.2 Nonconformances

1.2.1 Contrary to Criterion V of Appendix B, "Instructions, Procedures, and Drawings," PSC did not have a procedure in place for the replacement of commercial grade parts with safety-related or dedicated parts used during the process of upgrading surplus secondary market snubbers to current production line standards. (95-01-01)

1.2.2 Contrary to Criterion V of Appendix B, "Instructions, Procedures, and Drawings," and PSC SOP 07.104, "Supplier Approval," dated May 18, 1995, PSC had not listed Walden Industrial Supply Company and Vabcor Inc., companies from which PSC purchased surplus snubbers which were resold to licensees as safety-related, on its Approved Suppliers List (ASL). (95-01-02)

1.2.3 Contrary to Criterion I of Appendix B, "Organization," PSC performed a portion of its safety-related activities without the quality oversight or knowledge of the PSC QA manager and these activities were solely conducted under the control of production quality control personnel and, as a result, the Quality Assurance Manager was not able assure that an appropriate quality assurance program had been established and effectively executed or verify that activities affecting safety-related functions have been correctly performed. (95-01-03)

1.2.4 Contrary to Criterion XVI of Appendix B, "Corrective Action," and PSC QA Manual, Section 11 "Nonconformance and Corrective Action," Revision 6, dated December 15, 1993, PSC did not perform a failure analysis and corrective action Report, as required by PSC procedure, for a PSA Model 100 snubber which failed an activation test. The snubber, purchased as surplus material from Vabcor Inc., was part of an order for Pennsylvania Power & Light's Susquehanna Steam Electric Station. (95-01-04)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

None were reviewed during this inspection.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

PSC manufactures snubbers which are designed to restrain piping systems and associated components from damage resulting from earthquakes and other shocks while enabling unrestricted movement for thermal growth. The snubber operates on the principle of limiting the acceleration of any pipe movement to a threshold level of .02 times the acceleration of gravity (g), which is the maximum acceleration that the snubber will permit the piping system to experience. Should the piping system experience an acceleration in either direction, a braking force will be applied within the snubber necessary to limit the acceleration to less than .02 g's, while not restricting thermal expansion.

This braking force is achieved by a ball screw and drum which attempts to angularly accelerate an inertia mass. The inertial resistance of the mass causes a resilient capstan spring to tighten around a hardened mandrel which is part of a structural tube which provides a restrain force against rotation of the ball screw. The snubber's performance is independent of the amount of force being applied and at no time does it lock to become a rigid strut.

Each snubber is built in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Subsection NF, and to an ASME Section III, Division 1, NCA-3800, approved Quality Assurance Program which complies with the requirements of Appendix B, American National Standards Institute (ANSI) ANSI N45.2, and the reporting requirements Part 21. Each snubber is production tested by PSC in accordance with established procedures and ASME Code rules to establish rated capacities and are designed and built to ASME Class 1 Code requirements regardless of customer specifications. PSC has served the power industry for more than 30 years and has supplied snubbers to over 100 nuclear power plants. PSC also maintains an ASME Quality System Certificate (QSC-527) as a material supplier.

3.2 Entrance and Exit Meetings

During the entrance meeting, held on September 25, 1995, the NRC inspection team met with members of PSC management and staff, discussed the scope of the inspection, and established organizational contacts. During the exit meeting, held on September 28, 1995, the inspection team summarized its findings with PSC management and staff. The Appendix of this report lists the persons contacted during the inspection.

3.3 10 CFR Part 21 Program

The inspectors reviewed PSC's 10 CFR Part 21 program including procedures and implementation. PSC had identified several potential deviations in products that they had supplied and consequently had performed evaluations in accordance with its Part 21 program.

In August of 1992, PSC discovered a group of PSA Model 110 pinion gears (part number 1801418-01), which did not appear to have been heat treated. PSC determined that one batch of gears had not been tested for hardness during the initial production of PSA 100 snubbers. PSC performed an evaluation and determined that a PSA 100 snubber using a non-heat treated pinion gear would still meet 100% rated load and that the use was not a defect.

In June of 1993, Arizona Public Service (APS) reported to PSC that snubbers which had passed PSC testing had subsequently failed during APS testing. PSC performed additional testing on the snubbers, with the oversight of APS, and concluded that the APS test failure (high drag) was caused by set up error due to a loose adaptor, and that the test error was conservative in that it would fail a good snubber but would not pass a bad snubber. PSC concluded that was not a deviation or a defect. The inspectors determined that PSC had performed adequate evaluations in accordance with their Part 21 program and did not identify any concerns in this area.

The inspectors reviewed PSC's Part 21 implementing procedure, SOP 01.07, "Compliance With 10CFR21," dated August 10, 1993, and determined that it did not include several items required to be proceduralized by Part 21. Section 10 CFR 21.21 requires that corporations subject to the regulations 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 (2), in all cases within 60 days of discovery; (2) ensure that if an evaluation of an identified deviation or failure to comply potentially associated with a substantial hazard cannot be completed within 60 days of discovery of the deviation or failure to comply, an interim report is prepared and submitted to the Commission through a director or responsible officer or designated person; and (3) ensure that a director or responsible officer is informed as soon as practicable, and, in all cases, within the 5 working days after completion of the evaluation in (1) or (2) if a defect or failure to comply associated with a substantial safety hazard exists.

The inspectors concluded, based on a review of the Part 21 evaluations performed by PSC, that although PSC had failed to proceduralize the 10 CFR 21.21 requirements, there had not been an occurrence where PSC's actions were not in accordance with 10 CFR 21. The failure to proceduralize the requirements specified in 10 CFR 21.21 constitutes a violation of minor significance and is being treated as a Non-Cited violation, consistent with Section IV of the NRC Enforcement policy (NUREG-1600).

In addition, SOP 01.07 also contained several instances where the terms deviation and defect were inappropriately interchanged. The use of these terms, and their definitions in Part 21, were discussed with PSC who indicated that the procedure would be modified to correctly use these two terms. The inspectors also reviewed PSC's posting as required by 10 CFR 21.6 and determined it to be in accordance with the regulation.

3.4 Processing Surplus Snubbers

Discussion with PSC personnel indicated that they had bought surplus snubbers (referred to as buybacks) from two material brokers, Walden Industrial Supply Company (WISC) and Vabcor Inc. (VI), performed certain modifications and tests and resold them to licensees as safety-related. PSC purchased the surplus snubbers, which had been originally sold to licensees as safety-related and had never been installed, from WISC and VI. During conversation with the inspectors, various PSC personnel stated that this practice had been ongoing for approximately ten years.

The inspectors reviewed the records associated with the two companies supplying the surplus snubbers. PSC provided purchase orders it had made to WISC and VI since 1992 but did not have any purchase orders to either company prior to 1992. In further discussion, PSC indicated that this was the earliest either company had been used for surplus snubbers. PSC was unable to provide information on any other companies which might have supplied surplus snubbers prior to 1992 (although PSC indicated it had purchased surplus snubbers since approximately 1985).

PSC indicated that the process used to prepare the surplus snubbers for safety-related sales to licensees had evolved over the period from 1992 to 1995. Initially the surplus snubbers were only visually inspected, tested and shipped without documentation of the activities. This had evolved to the current practice which included receiving inspection, disassembly, replacement of commercial grade parts with safety-related parts, ASME code reconciliation, reassembly, and testing. In addition, PSC replaced the original label plate on the surplus snubbers with a new label plate indicating the current production information including a new serial number and manufacturing date. Each model of snubber was processed to a manufacturing order (MO) specific to the model and the model specific MOs were in various states of revision with some models of snubbers requiring less work to be prepared for shipment than other models.

For the current production line, PSC had evaluated all the snubber designs, determined that the majority of components should be safety-related, and had established a policy and revised the applicable MOs for the various snubber models to dedicate the appropriate commercial grade parts used in the production of new snubbers. During the initial production of the surplus snubbers, PSC indicated that some commercial grade parts had possibly been used in the manufacture of the snubbers (if manufactured prior to the implementation of PCS's dedication policy). PSC's current practice in processing the surplus snubbers included replacing all commercial grade parts with new parts which were safety-related (dedicated) to bring the surplus snubbers up to current production line standards. However, PSC was unable to provide procedures requiring the replacement of commercial grade parts or documentation that this had occurred for various models and orders of surplus snubbers from 1992 through 1994.

PSC's failure to have a procedure in place for replacement of commercial grade parts with safety-related parts, or documentation that this had occurred, during the process of upgrading the snubbers to current production line standards, in the 1992 through 1995 time period, was identified as a Nonconformance with Criterion V of Appendix B (95-01-01).

3.5 Supplier Approval

The inspectors reviewed SOP 07.104, "Supplier Approval," dated May 18, 1995, which established the procedure for the qualification of PSC suppliers. Paragraph 3.1.1 stated that all suppliers who provide items for use in the nuclear product line would be qualified through surveys or audits. In addition, paragraph 3.4 stated that all suppliers who furnish material or supplies for use in saleable product shall be listed on the applicable PSC ASL. Discussion with PSC personnel and review of various purchase orders (POs) indicated that PSC had purchased numerous snubbers from two companies, WISC and VI. The snubbers were originally manufactured by PSC and sold to licensees who, in turn, sold them on the surplus market to material brokers such as WISC and VI. A review of PSC's ASL, Revision 3, dated August 17, 1995, did not identify either company as an approved supplier as required by SOP 07.104. The QA manager indicated that neither company had ever been on the ASL or had ever been qualified through survey or audit.

PSC's failure to place WISC and VI on the ASL, as required by SOP 07.104, was identified as a Nonconformance to Criterion V of Appendix B (95-01-02).

3.6 Quality Assurance Organization

The inspectors reviewed PSC's organization in the current QA manual which established that the QA Manager reported directly to the division President. Additional quality control personnel reported directly to the Commercial and Nuclear Director who was responsible for nuclear production. The QA Manager indicated that the functions of the position included administrative and QA aspects, policy and procedure, internal audits, document control, receiving inspection for new material (not the surplus snubbers) and calibration. Quality control personnel, who reported to the Commercial and Nuclear Director, responsibilities included supplier control, material review board, corrective actions, in-process inspection, final inspection, non-destructive testing, receiving inspection (for surplus snubbers), change control for drawings, contract review, and PO review.

PSC Quality Assurance Manual Section 0.20, "Management Statement," stated that the Manager of Quality Assurance is given the responsibility and authority to maintain the Quality Assurance program and its implementation. However, the inspectors determined that the arrangement of quality personnel's responsibilities, quality control personnel reporting directly to the Commercial and Nuclear Director with little oversight by the QA Manager, could effect the QA Manager's ability to assure that an appropriate quality assurance program was established and effectively executed or to verify that activities affecting safety-related functions were correctly performed. An example of this was determined during discussion with the PSC QA Manager, who indicated that for a particular period of time, approximately 1992 through 1994, the purchasing of surplus snubbers, receiving inspection of surplus snubbers, disassembly, parts replacement, reassembly, label plate replacement, and testing had been solely conducted under the control of production quality control personnel without the quality oversight or knowledge of the PSC QA Manager.

The inspectors concluded that the activities related to the purchasing, processing, and testing of surplus snubbers were activities affecting quality, and therefore should have been performed with the quality oversight and knowledge of the QA Manager to assure that an appropriate quality assurance program had been established and effectively executed and to verify that activities affecting safety-related functions were correctly performed. This was identified as Nonconformance to Criterion I of Appendix B (95-01-03).

3.7 Purchase Order Review

The NRC inspectors reviewed several customer POs to PSC for various safety-related snubbers. One such order, Pennsylvania Power & Light (PP&L) PO 3-50439-1, dated October 11, 1993 (PSC Order No. 0410003), ordered 11-PSA Model 100 snubbers for general use at the Susquehanna Steam Electric Station. The PO required that the documentation package include test reports and a certificate of conformance to the PO requirements. The PO also required that the items supplied be new and that refurbished and modified items would not be accepted without the approval of PP&L prior to shipment. The technical and quality requirements required that the units be designed and fabricated in accordance with the PP&L approved PSC quality program, which complies to Appendix B and Part 21.

PSC purchased all 11 units from WISC and VI as buyback inventory as follows: A total of nine units were purchased from VI under PSC POs 1124040 and 1110250 and the remaining 2 units were purchased from WISC under PSC PO 1106560. A review of the POs to both companies did not identify any quality or technical requirements.

Following receipt by PSC, all snubbers were tested under PSC IT 533, "Acceptance Test for 1801119 Shock Arrestor, PSA-100," dated January 29, 1975. Paragraph 6.0 of IT 533 identifies the following tests to be performed: Activation, Lost Motion and Drag/Breakaway. Design Report (DR) 1319, "Mechanical Shock Arrestors Standard Design Specification," dated April 17, 1975, requires in Paragraph 8.1 that each production unit shall be subjected to acceptance tests to verify functional compliance with applicable drawings and specification requirements.

PSC policy is that all snubbers, regardless of application, are designed and manufactured as Class 1 Linear Standard supports and are classified as material in accordance with the provisions of ASME III, Subsection NF-1214, as documented in DR 1319. In addition, all snubbers are designed in accordance with ASME Section III, Division 1, Subsection NF, Sub-article 3200 or 3300. A review of the documentation package identified that PSC did not complete a failure analysis and corrective action report for one of the four snubbers, purchased under PO 1110250, which failed the activation test on June 17, 1994. The unit was retested satisfactorily on October 28, 1994, however the package did not give a reason for the failure. The NRC inspectors verified the existence of a Certificate of Compliance, dated October 27, 1994, which certified compliance to the PO requirements for snubber serial numbers 2489 through 2499. The final inspection checklist for the failed snubber (serial number 2492) was reviewed and appeared satisfactory. PSC's failure to complete the failure analysis and corrective action report for snubber number 2492 was identified as a Nonconformance to Criterion XVI of Appendix B (95-01-04).

Several other POs were reviewed during the inspection. They included Commonwealth Edison PO 4W5147, dated January 19, 1995, for one Model PSA 35, and PP&L PO 20581-1, dated April 10, 1990, for 25-Model PSA 100s. The documentation package for these appeared satisfactory.

APPENDIX

PERSONS CONTACTED

The following persons were contacted during the inspection and except as noted, attended both the entrance and exit meetings.

Pacific Scientific:

	R. Knoblock	Division President
	J. Dowdy	Nuclear Sales and Product Manager
	S. Palm	Quality Assurance Manager
•	A. Camacho	Senior Product Engineer
•	N. Hergenreder	Field Services Manager
•	W. Scott	Quality Control Engineer
•	L. Wright	Quality Assurance Engineer

U.S. Nuclear Regulatory Commission:

R. Pettis	Team Leader
B. Rogers	Reactor Engineer

• Did not attended exit meeting

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

October 6, 1995

Mr. Nicholas J. Liparulo
Nuclear Safety and Regulatory Activities
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

SUBJECT: NRC INSPECTION NO. 99900404/95-02

Dear Mr. Liparulo:

This letter addresses the inspection at the Ente per le Nuove Tecnologie, L'Energia e L'Ambiente (ENEA) Valve and Pressurizer Operating Related Experiments (VAPORE) test facility in Casaccia, Italy, conducted by Richard P. McIntyre of the Nuclear Regulatory Commission's (NRC's) Special Inspection Branch, Alan E. Levin of the Reactor Systems Branch, Juan D. Peralta of the Quality Assurance and Maintenance Branch, Andrzej Drozd of the Containment Systems and Severe Accident Branch, and James H. Wilson of the Standardization Project Directorate, on July 24 through 26, 1995. The details of the inspection were discussed with your staff members during the inspection and at the exit meeting on July 26, 1995.

The purpose of the inspection was to determine if automatic depressurization system (ADS) testing activities performed at the VAPORE test facility to support design certification of the Westinghouse AP600 advanced reactor design were conducted under the appropriate provisions of WCAP-8370, Revision 12A, the most recent Westinghouse Quality Assurance Plan (topical report) that has been approved by the NRC. The pertinent provisions of WCAP-8370 were implemented at VAPORE by ENEA document AP600-GQ9402, "Quality Assurance Plan Description: AP600 Test Program Conducted at the VAPORE Plant in ENEA Cassacia (Phase B)," Revision 2.

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

The results of the inspection indicate that Westinghouse/ENEA, in general, were adequately implementing the AP600 Project quality assurance program requirements with the exception of one finding and an unresolved item. Specifically, the team identified a Nonconformance with program implementation with respect to the generation of facility as-built drawings for AP600 ADS Phase B testing at VAPORE. Also, the team identified an Unresolved Item concerning the fact that the ENEA quality assurance program does not include adequate measures to effectively control the calibration status of the reference instruments or standards used for instrument calibration. No provisions were in place to require re-calibration at the requisite intervals.

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

In accordance with 10 CFR Part 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.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Robert M. Gallo

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

Docket No.: 52-003

Enclosures:
1. Notice of Nonconformance
2. Inspection Report No. 99900404/95-02

cc w/encls: See Next Page

Mr. Nicholas J. Liparulo
Westinghouse Electric Corporation

Docket No.: 52-003
AP600

cc:

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Energy Systems Business Unit
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Pittsburgh, PA 15230

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Mr. Charles Thompson, Nuclear Engineer
AP600 Certification
U.S. Department of Energy
NE-451
Washington, DC 20585

NOTICE OF NONCONFORMANCE

Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

Docket No.: 52-003
99900404

Based on the results of a Nuclear Regulatory Commission (NRC) inspection of Westinghouse Electric Corporation's AP600 Automatic Depressurization System (ADS) Phase B test program at the Ente Nazionale Energia e Ambiente (ENEA) VAPORE Test Facility, conducted on July 24 through July 26, 1995, it was determined that activities supporting Westinghouse Electric Corporation's AP600 design certification program, were not conducted in accordance with NRC requirements as identified below:

A. Section 9.0, "Quality Assurance Requirements" of WCAP-14112, "Automatic Depressurization System Test Specification (Phase B1)," Revision 2, and Section 7.0, "As-Built Records" of ENEA document AP600-GQ9402, "Quality Assurance Plan Description: AP600 Test Program Conducted at the VAPORE Plant in ENEA Cassacia (Phase B)," Revision 2, provide for and require the preparation and maintenance of VAPORE test facility as-built drawings which pertain to the ADS Phase B tests that characterize the features which influence thermal-hydraulic and structural parameters for code validation and calculation methodology verification efforts.

   Contrary to the above, as-built drawings, as defined and stipulated in WCAP-14112 and AP600-GQ9402, had not been generated for AP600 ADS Phase B testing at VAPORE. (95-02-01).

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, 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 the steps that were or will be taken to correct these items; (2) a description of the steps that have 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____________ day of ____________, 1995

Enclosure 1

ORGANIZATION:	Westinghouse Electric Corporation Pittsburgh, Pennsylvania
REPORT NO.:	99900404/95-02
CORRESPONDENCE ADDRESS:	Mr. Nicholas J. Liparulo Nuclear Safety and Regulatory Activities Westinghouse Electric Corporation P.O. Box 355 Pittsburgh, Pennsylvania 15230
ORGANIZATIONAL CONTACT:	John Butler, Principal Engineer Advanced Plant Safety and Licensing (412) 374-5268
NUCLEAR INDUSTRY ACTIVITY:	Nuclear steam supply system components and services
INSPECTION CONDUCTED:	July 24 through 26, 1995
TEAM LEADER:	Richard P. McIntyre 10/5/95 Richard P. McIntyre Date Vendor Inspection Section (VIS) Special Inspection Branch (PSIB)
OTHER INSPECTORS:	Juan D. Peralta, HQMB Alan E. Levin, SRXB Andrzej Drozd, SCSB James H. Wilson, PDST
REVIEWED:	Gregory C. Cwalina 10/5/95 Gregory C. Cwalina, Section Chief, VIS Date
APPROVED:	Robert M. Gallo 10/6/95 Robert M. Gallo, Chief, PSIB Date
INSPECTION BASES:	10 CFR Part 50, Appendix B and 10 CFR Part 21
INSPECTION SCOPE:	To determine if activities performed to support the design of AP600 and, specifically, the automatic depressurization system (ADS) Phase B test program at the Ente Nazionale Energia e Ambiente (ENEA) VAPORE Test Facility in Casaccia, Italy, were conducted under the appropriate provisions of WCAP-8370, Revision 12A, the most recent Westinghouse Quality Assurance Plan that has been approved by the NRC.
PLANT SITE APPLICABILITY:	None

Enclosure 2

1 INSPECTION SUMMARY

1.1 Nonconformance

- Nonconformance 99900404/95-02-01 was identified and is discussed in Section 3.3 of this report.

1.2 Unresolved Item

- Unresolved Item 99900404/95-02-02 was identified and is discussed in Section 3.4 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous inspection has been conducted at the VAPORE test facility.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 AP600 Quality Assurance Program

Chapter 17 of the AP600 Standard Safety Analysis Report (SSAR) describes the Westinghouse Electric Corporation (W) quality assurance (QA) program for the design phase of the AP600 Advanced Light Water Reactor (ALWR) Plant Program. The QA program is identified as Westinghouse Topical Report WCAP-8370, "Energy System Business Unit-Power Generation Business Unit Quality Assurance Plan," Revision 12A, dated April 1992. WCAP-8370 applies to all W activities affecting quality of items and services, including the design certification process for AP600. Accordingly, WCAP-8370 establishes W's commitments to meeting the requirements of 10 CFR 50 Appendix B, ASME NQA-1 and NQA-2, and Regulatory Guide 1.28, Revision 3, "Quality Assurance Program Requirements (Design and Construction)"

Section 17.3 of the AP600 SSAR states that activities supporting the design and design certification phase of the project are performed in accordance with W topical report WCAP-8370 as supplemented by a project-specific Quality Plan. WCAP-12600, "AP600 Advanced Light Water Reactor Design - Quality Assurance Program Plan (QAPP)," dated December 1993, the project-specific QA plan, was developed by W to enhance WCAP-8370 in specific areas and to establish additional commitments needed to support the AP600 program. It also states that the AP600 design certification test programs and related analyses are within the scope of this QAPP.

WCAP-12601, "AP600 Program Operating Procedures," Revision 13, dated July 8, 1994, was developed by W to establish requirements and responsibilities for developing, approving, implementing, revising, and maintaining operating procedures to meet the QA and administrative requirements of the AP600 program. WCAP-12601 includes an "AP600 Program Procedure Matrix," Revision 15, dated April 4, 1995, which identifies the correlation between the W commitments to the QA requirements of (1) ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," 1983 Edition (as endorsed by

Regulatory Guide 1.28, Revision 3) and (2) ANSI/ASME NQA-1 1989 Edition through NQA-1b-1991 Addenda, and the corresponding implementing guidance embodied in WCAP-9565, "Nuclear and Advanced Technology Division (NATD) Quality Assurance Program," Revision 34, dated May 2, 1994, and in WCAP-12601. WCAP-9565 governs the implementation of all NATD activities related to areas within the scope of WCAP-8370.

During the inspection, the team assessed the implementation of the applicable QA criteria essential to support the AP600 Design Certification application, including Design Certification testing. Specifically, the team evaluated the effectiveness of the QA program and controls, as described above, including the soundness of the data obtained during the AP600 VAPORE ADS test program.

3.1.1 Ente Nazionale Energia e Ambiente (ENEA) VAPORE Facility QA Program

In the AP600 design, the automatic depressurization system (ADS) ensures that the reactor coolant system (RCS) is depressurized during pertinent transient and accident conditions, thereby initiating and maintaining long-term gravity injection. Under a technical cooperation agreement, W, ENEA, and ANSALDO S.p.A., combined resources to conduct testing at the ENEA VAPORE test facility in Cassacia, Italy, with two major objectives: (1) advance knowledge and understanding of passive safety system operations, and (2) conduct testing of the ADS to provide both design information and data for computer code validation efforts needed to support design certification.

During phase A of testing, information on the performance of a prototypic sparger was gathered. The ADS phase B tests comprised full-sized simulation of one of the two AP600 ADS flowpaths from upstream of the ADS valves to a sparger and was intended to simulate and/or conservatively bound the operating conditions of the AP600 ADS system configuration.

WCAP-14112, "Automatic Depressurization System Test Specification (Phase B1)," Revision 2, provides that testing, designed to demonstrate overall automatic depressurization system (ADS) performance verification, be conducted under a QA program that conforms to the requirements of the American Society of Mechanical Engineers (ASME) NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," 1989 Edition through NQA-1-1991 Addenda.

To this effect, ENEA developed AP600-GQ9402 (AP600 Document No. RCS-T1H-001), "Quality Assurance Plan Description: AP600 Test Program Conducted at the VAPORE Plant in ENEA Cassacia (Phase B)," Revision 2 (QAPD), which defines responsibilities, prescriptions and recommendations to govern the AP600 ADS Test Program/Phase B according to W requirements and pertinent ENEA procedures.

During the inspection, the team reviewed the pertinent documents to determine if design certification testing activities associated with the AP600 program and performed at the ENEA VAPORE test facility during the ADS test program were conducted in accordance with the appropriate provisions of W's 10 CFR 50, Appendix B, QA program (WCAP-8370). The team examined the performance of activities in specific areas within the scope of AP600-GQ9402, i.e. test control, test instrument calibration, facility and records configuration control to confirm that activities in these areas were conducted and accomplished under suitably controlled conditions by properly trained personnel and that the resultant test data collected during such activities was appropriately recorded and maintained.

Based on reviews of these areas, including the QA implementation audit at VAPORE conducted by W on June 6-9, 1995, and documented in W Audit Report No. QLA/ENA0006, dated June 19, 1995, the team concluded that the QA program set forth in AP600-GQ9402, in conjunction with the W's implementation of the pertinent criteria of WCAP-12601, provided sufficient evidence of overall QA implementation appropriate to design certification testing, except for certain areas as identified below.

3.2 Test Control

The team reviewed aspects of test control for the AP600 VAPORE Phase B1 test program. These tests involved prototype Stage 1, 2, and 3 automatic depressurization system valves in a full-scale piping network, to acquire thermal-hydraulic performance data for validation of computer models for accident and transient analyses of the AP600 plant. The test program was completed in 1994. Documentation reviewed included the ADS test specification, test procedures, test logs, test checklists, and test results as reflected by selected data and Day of Test reports. WCAP-14112, Revision 2, was developed by W, and the detailed test procedures were developed by ENEA and approved by W.

The test procedures were comprehensive and easy to follow, and included checklists for each type of test that was performed in the Phase B1 program. The procedures implement the W test specification requirements. The test procedure included requirements for the development of a test diary as well as the checklist. Documentation for each test, including completed and signed-off checklists, data plots, Day of Test reports, and other relevant information, was contained in a separate divider in the Design Record File (DRF). Each test was performed by a team of 2-4 individuals and a W resident engineer who was assigned to the VAPORE facility to oversee and participate in the conduct of the test program.

The Day of Test reports were prepared by W's on-site resident engineer for transmittal to W Test Engineering in Monroeville, Pennsylvania, and, as was found in the inspection of the SPES-2 program at SIET, provide an excellent contemporaneous record of important test results, "unexpected events," instrumentation condition and potential problems, and other relevant discussion. A list of critical instrumentation was also included as part of the test procedures, but functionality was not verified prior to performing each test. When the team questioned W and ENEA personnel about verification of critical instrumentation, they were informed that the verification was performed after the test, rather than before. It was noted that, although the primary objective of the testing was the acquisition of thermal-hydraulic data, a substantial amount of mechanical-related instrumentation was also included on the loop, including strain gages and accelerometers. These were not considered to be critical instruments for the Phase B1 tests. In general, functionality of thermal-hydraulic instrumentation (thermocouples, flowmeters, pressure transducers) was adequate.

The test data records were maintained in a well organized manner. The data acquisition system (DAS) software was developed according to British standards and checked independently by W. The test data was recorded on PROSIG system using CD ROM disks and W is custodian of all the data. As has been the case in other W AP600 test programs performed by outside organizations, the final review of the data to determine if test acceptance criteria were met, was performed by W personnel in Monroeville. As a result, there is little documentation of the actual data review itself, other than the initial assessments as contained in the Day of Test reports. In addition, deviations during the tests requiring disposition are also not recorded in ENEA's DRF. The team was informed that this documentation is in W's files in Monroeville and could not be verified during this inspection. Nevertheless, the design record file does contain correspondence between W and ENEA, specifically addressing acceptability of the VAPORE Phase B1 tests. In most cases, tests were considered to have met acceptance criteria; however, several tests were re-run as a result of W's determination that they had not met acceptance criteria. The team considered the documentation of W's review in this manner to be appropriate and adequate to ensure proper test control.

3.3 As-Built Drawings and Configuration Control

WCAP-14112, Section 9.0, "Quality Assurance Requirements," and AP600-GQ9402, Section 7.0, "As-Built Records," provide for the preparation and maintenance of the VAPORE test facility as-built drawings which pertain to the ADS Phase B tests that characterize the features which influence thermal-hydraulic and structural parameters. The thermal-hydraulic parameters are needed to allow adequate modelling of the facility for code validation efforts. AP600-GQ9402 also provides that specific as-built features to be recorded shall be as specified by W. Additional configuration control provisions in AP600-GQ9402 are specified under Section 7.2, "Features to be Recorded and Format," Section 7.3, "Method and Responsibilities," and 7.4, "Changes".

Modifications to the VAPORE test facility, necessary to support AP600 ADS design certification testing, were performed by ANSALDO S.p.A. under contract to W. On November 29, 1994, W placed a contract with ANSALDO (MB21177S Change Notice) to provide as-built documentation of the ADS test loop at the ENEA's VAPORE test facility. W stipulated that ANSALDO provide one full set of as-built drawings (comprising P&ID, line list of principal flow paths, valve list, ADS loop layout drawings, ADS loop isometric drawings, ADS loop platform, and ADS loop support drawings) covering both ADS Phases B1 and B2 configurations. W intended to include these drawings as part of the as-built records package for AP600 VAPORE Phase B testing.

During the inspection, however, the team found that as-built drawings, as defined and stipulated in WCAP-14112, and in AP600-GQ9402, had not been generated for AP600 ADS Phase B testing at VAPORE. This issue was identified as Nonconformance 95-02-01.

3.4 Procurement and Calibration of Test Instrumentation

WCAP-14112 requires, in part, under Section 9.0, "Quality Assurance Requirements," that the following measures be taken in the detailed test procedure(s): (1) Provisions for ensuring that calibration of test equipment is traceable to recognized national standards, and (2) Verification and documentation, to be submitted to W, by the testing organization that the instrumentation calibrations have been performed prior to testing. Section 6.0, "Instrumentation Management and Control," of the ENEA QAPD Document, AP600-GQ9402, implemented these requirements.

During the inspection, the team confirmed that all test instruments used in the ENEA VAPORE test facility had been calibrated, prior and after testing, using standards or reference instruments traceable to the Servizio di Taratura in Italia or, Italian calibration System (SIT). In Italy, under the auspices of the Western European Calibration Cooperation (WECC), national calibration standards equivalent to NIST are established and maintained by SIT.

During the inspection, the team reviewed the VAPORE test facility calibration records which provided evidence of traceability to the appropriate ENEA controlled SIT-certified standards. This review also provided evidence of the adequacy of the facility instrumentation calibration status during each testing phase. The team found, however, that the ENEA QA program does not include adequate measures to effectively control the calibration status of reference instruments or standards used for instrument calibration, as no provisions were in place to require re-calibration by SIT at the requisite intervals. This may have resulted in the introduction of uncertainties in the adequacy of calibration of test facility instrumentation which relied on these standards to establish and maintain their accuracy.

Pending confirmation by W that this lapse in the SIT-certified calibration interval for the ENEA standards did not undermine or adversely impact the VAPORE ADS test results, this issue will remain unresolved. This issue has been identified as Unresolved Item 95-02-02.

3.5 Quality Assurance Records

QA records and documents associated with the W AP600 ADS testing at the VAPORE facility were processed in accordance with the provisions in the ENEA QAPD (AP600-GQ9402). The QAPD lists all of the applicable ENEA procedures for implementation of QAPD requirements. Although ENEA does not have a separate QA department, one individual, the designated QA Responsible, has responsibility for all QA activities at the facility. The QA Responsible is charged with maintaining all QA records, including, QAPD and audits, procedures, test specifications and test matrix modifications, instrument calibration records, data acquisition, test results, training and informal meetings, ENEA surveys, and nonconformance reports.

ENEA maintains a single "dossier" document file of instrument calibration records for each instrument, as required by ENEA Document EIHE-94021, "Instrument Management." Each dossier contains an instrument card, calibration certificates and/or calibration control certificates, and any existing instruction manuals.

As part of its contract with W, ENEA has developed an overall design record file that archives all documents associated with AP600 ADS testing at the VAPORE facility. This design record file, conforming to the ENEA QAPD, is a deliverable from ENEA to W and serves as the official record of testing activities at VAPORE.

3.6 Instructions, Procedures and Drawings

Procedures and drawings were processed and maintained in accordance with the provisions of Section 5.0, "Documentation Management," of the ENEA QAPD Document, AP600-GQ9402. The QAPD requires that a copy of the procedures applicable to test activities be sent to the QA Responsible for approval and distribution to the relevant departments. Drawings and as-built records pertaining to the AP600 ADS tests at VAPORE are maintained to document the features that influence thermal hydraulic and structural parameters of the tests.

ENEA maintains all testing-related procedures and drawings in the design record file that archives all documents associated with AP600 ADS testing at the VAPORE facility. Based on the team review, it was determined that appropriate procedural controls had been developed and implemented to govern the conduct of AP600 quality-related test activities at VAPORE.

3.7 Audits

The ENEA QAPD does not include any requirements for conducting internal audits of work performed at the VAPORE facility. However, W has assumed this responsibility under their QA program and conducted a series of readiness assessments (in 1993 and 1994) and implementation audits (in 1991 and 1995) of the activities associated with the AP600 ADS development. In these assessments and audits, W treats ENEA as an approved supplier. The most recent W audit, WES 95-243, was conducted on June 6-9, 1995, and evaluated ENEA's implementation of the activities described under Criterion 7, "Procurement," of WCAP-12601. The audit report, QLA/ENA0006, dated June 19, 1995, contained 7 findings and 2 recommendations and requested a response from ENEA by July 20, 1995. The audit report was distributed to ENEA management and department heads. In a July 21, 1995, memorandum, the W lead auditor concluded that the actions taken by ENEA, as described in the July 20, 1995 response, were acceptable for each of the findings and the recommendations, and that the findings were closed.

The staff concluded that the W procedures for the conduct of internal audits were appropriately followed in the evaluation of QA activities at the VAPORE test facility and were effective in evaluating activities associated with AP600 ADS testing.

4 PERSONNEL CONTACTED

Westinghouse Electric Corporation

Eugene Piplica, AP600 Test Manager
David Alsing, AP600 Quality Assurance Manager
Robin Nydes, AP600 Senior Project Engineer
Robert Tupper, AP600 Project Engineer
Tim Bueter, AP600 Test Engineering

Ente per le Nuove Tecnologie, L'Energia e L'Ambiente (ENEA)

C. Kropp, Division Head, VAPORE Operating Group
Piero Incalcaterra, Test Responsible
Bruno Milana, Quality Assurance Responsible
L. Solaro, Instrumentation Responsible
G. Serafini, Maintenance Mechanical Responsible

Selected Generic Correspondence on the Adequacy of Vendor Audits and the Quality of Vendor Products

Identifier	Title
Information Notice 95-45	American Power Service Falsification of American Society for Nondestructive Testing (ANST) Certificates

CORRESPONDENCE RELATED TO VENDOR ISSUES

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

December 5, 1995

Mr. David Z. Hathcock, Quality Assurance Manager
Cardinal Industrial Products,
Limited Partnership
3873 West Oquendo
Las Vegas, NV 89118

SUBJECT: CARDINAL'S AUGUST 30, 1995 REPLY TO NOTICE OF NONCONFORMANCE 99901076/94-01-03 (NRC INSPECTION REPORT NO. 999001076/94-01)

Dear Mr. Hathcock:

Thank you for your letter of August 30, 1995, which supplemented the information contained in your May 30, 1995, and January 30, 1995, letters in response to Notice of Nonconformance 99901076/94-01-03.

We have reviewed the sampling rationale discussed your letter and noted that, while this approach can be expected to improve the overall assurance of product integrity, it places heavy reliance on visual and dimensional inspection to support the verification of lot homogeneity. We agree that 100% visual inspection can be a significant factor in the identification of mixed product lots by noting non-uniform markings, coloration differences, or other visual discontinuities. However, visual inspection 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.

With respect to your reference to the use of ASTM A-325 shipping lot sampling plan, please review the comments contained in our April, 24, 1995 letter to Mr. Scott Akers, Jr. In this letter, we also stated that, when product sampling is used as the basis for qualifying an item for unrestricted safety application in a nuclear power plant, the NRC staff has generally accepted a confidence level of 90-95% that no more than 5-10% of the sampled items are nonconforming. The sample size and rationale described in your August 30, 1995 letter does not appear to provide this level of confidence for random lots (unverified traceability) of material, especially for the verification of critical characteristics related to the physical properties of material. Recognizing the need for more specific guidance in this area, the NRC staff has initiated a generic study of industry practices and existing guidance related to material sampling. This study, when completed, may result in NRC's endorsement of specific sampling approaches.

The latest revision of 10 CFR Part 21 (Federal Register 48369, September 19, 1995 - enclosed) defines dedication as "an acceptance process undertaken to provide reasonable assurance that a commercial grade item to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR 50, Appendix B, quality assurance program." The regulation further states that reasonable assurance is achieved by identifying the critical characteristics of the item and verifying their acceptability by inspections, tests, or analyses. Critical characteristics of an item are defined as "those important design, material, and performance characteristics of a commercial grade item that, once verified, will provide reasonable assurance that the item will perform its intended safety function."

Determining the intended safety function of the item being dedicated is an integral part of the dedication process. The item's intended safety function is considered in both the identification and verification of critical characteristics, including the selection of an appropriate sampling plan. The safety function of the item is typically controlled by the end user of the item (licensee). If the licensee elects to procure such items as basic components, the licensee must invoke 10 CFR 50, Appendix B in the procurement documents. The supplier has the responsibility of assuring that it is supplying either an item which has been designed/manufactured under the applicable provisions of Part 50, Appendix B or of dedicating a commercial grade item under its own Part 50, Appendix B program. In either case, the licensee is responsible for reviewing and approving the suppliers quality assurance program controlling these activities. In order to achieve Part 50, Appendix B equivalency when dedicating items such as material, a supplier must consider all of the applicable specification requirements to be critical and verify that these specification requirements have been met using a sampling plan with a confidence level such as discussed above since, typically, the supplier does not know the end application (safety function) of the item.

The licensee may also opt to procure an item which has been subjected to specified inspections and tests by the supplier, but is being sold by the supplier as a commercial grade item. Under these conditions, the licensee is ultimately responsible for the item's dedication which would include identification of the critical characteristics of the item and verifying their acceptability. If the inspections and tests performed by the supplier are to be considered a part of the dedication process, the inspections and tests must be controlled in accordance with a quality assurance program that is consistent with the applicable portions of Part 50, Appendix B that has been audited or surveyed and approved by the licensee. The procured item would be considered a commercial grade item until completion of the dedication process by the licensee. When supplying items under these conditions, the supplier would certify that the item has been designed/manufactured in accordance with its licensee-approved quality assurance program but would not certify that the item complies with Part 50, Appendix B. Considering that a significant number of items supplied to nuclear plants may have a specialized or limited safety application, suppliers with quality assurance programs that do not demonstrate specification compliance with a confidence level as discussed above may, since the recent revision of Part 21, consider supplying such items as commercial grade under their quality assurance program. This option would need to be negotiated, of course, with the end user of the item (licensee). The licensee as part of its dedication of the item, would determine the acceptability of sampling procedures and the need for additional inspections or tests based on the safety related application of the item.

In accordance with Section 2.790 of 10 CFR, a copy of this letter will be placed in the NRC's Public Document Room.

No response to this letter is required.

Sincerely,

Robert M. Gallo

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

Docket No.: 99901076

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

December 4, 1995

Mr. Mark T. Capallo, President
Energy & Process Corporation
2146-B Flintstone Drive
Tucker, GA 30084-5000

SUBJECT: YOUR AUGUST 25, 1995 RESPONSE TO NOTICE OF NONCONFORMANCE (NRC INSPECTION REPORT NO. 99900866/95-01)

Dear Mr. Capallo:

Thank you for your letter of August 25, 1995, in response to our June 14, 1995, letter and the enclosed Notice of Nonconformance.

We have reviewed your letter and find the proposed corrective actions for nonconformances 99900866/95-01-01 and 99900866/95-01-03 satisfactory. We will review the implementation of these actions during a future inspection.

We find that your response to nonconformance 99900866/95-01-02 does not provide an adequate basis for substantiating that your material sampling plans for verifying critical characteristics provide reasonable assurance that the dedicated items meet all of the applicable specification requirements.

When sampling is used to qualify items for unrestricted safety applications in nuclear plants, the NRC staff generally expects such sampling to provide a confidence level of 90-95% that no more than 5-10% of the items sampled are nonconforming. Your sampling program does not provide a confidence level consistent with these expectations.

With respect to your comments concerning the lack of established guidelines for sampling of commercial grade items, the NRC staff has initiated a generic study of industry practices and existing guidance related to material sampling. This study, when completed, may result in NRC's endorsement of specific sampling approaches.

The latest revision of 10 CFR Part 21 (Federal Register 48369, September 19, 1995 - enclosed) defines dedication as "an acceptance process undertaken to provide reasonable assurance that a commercial grade item to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR 50, Appendix B, quality assurance program." The regulation further states that reasonable assurance is achieved by identifying the critical characteristics of the item and verifying their acceptability by inspections, tests, or analyses. Critical characteristics of an item are defined as "those important design, material, and performance characteristics of a commercial grade item that, once verified, will provide reasonable assurance that the item will perform its intended safety function."

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The licensee may also opt to procure an item which has been subjected to specified inspections and tests by the supplier, but is being sold by the supplier as a commercial grade item. Under these conditions, the licensee is ultimately responsible for the item's dedication which would include identification of the critical characteristics of the item and verifying their acceptability. If the inspections and tests performed by the supplier are to be considered a part of the dedication process, the inspections and tests must be controlled in accordance with a quality assurance program that is consistent with the applicable portions of Part 50, Appendix B that has been audited or surveyed and approved by the licensee. The procured item would be considered a commercial grade item until completion of the dedication process by the licensee. When supplying items under these conditions, the supplier would certify that the item has been designed/manufactured in accordance with its licensee-approved quality assurance program but would not certify that the item complies with Part 50, Appendix B. Considering that a significant number of items supplied to nuclear plants may have a specialized or limited safety application, suppliers with quality assurance programs that do not demonstrate specification compliance with a confidence level as discussed above may, since the recent revision of Part 21, consider supplying such items as commercial grade under their quality assurance program. This option would need to be negotiated, of course, with the end user of the item (licensee). The licensee as part of its dedication of the item, would determine the acceptability of sampling procedures and the need for additional inspections or tests based on the safety related application of the item.

In accordance with Section 2.790 of 10 CFR, a copy of this letter will be placed in the NRC's Public Document Room.

No response to this letter is required.

Sincerely,

Robert M. Gallo

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

Docket No.: 99900866

Enclosure: Federal Register

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

December 5, 1995

Mr. William Blackwell, President
Mackson, Inc.
2346 Southway Drive
Rock Hill, SC 29730

SUBJECT: MACKSON, INC's AUGUST 24, 1995 REPLY TO NOTICE OF NONCONFORMANCE 99901179/95-01-01 (NRC INSPECTION REPORT NO. 99901179/95-01)

Dear Mr. Blackwell:

Thank you for your letter of August 24, 1995, which supplemented the information contained in your April 25, 1995, letter in response to Notice of Nonconformance 99901179/95-01-01.

We have reviewed the additional information provided in your letter and agree that the increased sampling level, combined with the other qualitative factors, together with augmented visual inspection, should increase the level of confidence when the verification of critical characteristics is accomplished. These changes represent a significant improvement to your commercial grade dedication program.

However, as stated in our letter of July 19, 1995, when qualifying an item for unrestricted safety application in a nuclear plant, the NRC staff would generally expect sampling with a confidence level of 90-95% that no more than 5-10% of the items sampled are nonconforming. Your revised program does not provide a confidence level consistent with these expectations for destructive testing of commercial grade material obtained from unqualified suppliers or distributors. Recognizing the need for more specific guidance in this area, the NRC staff, as discussed in our July 19, 1995 letter, has initiated a generic study of industry practices and existing guidance related to material sampling. This study, when completed, may result in NRC's endorsement of specific sampling approaches.

The latest revision of 10 CFR Part 21 (Federal Register 48369, September 19, 1995 - enclosed) defines dedication as "an acceptance process undertaken to provide reasonable assurance that a commercial grade item to be used as a basic component will perform its intended safety function and, in this respect, is deemed equivalent to an item designed and manufactured under a 10 CFR 50, Appendix B, quality assurance program." The regulation further states that reasonable assurance is achieved by identifying the critical characteristics of the item and verifying their acceptability by inspections, tests, or analyses. Critical characteristics of an item are defined as "those important design, material, and performance characteristics of a commercial grade item that, once verified, will provide reasonable assurance that the item will perform its intended safety function."

Determining the intended safety function of the item being dedicated is an integral part of the dedication process. The item's intended safety function is considered in both the identification and verification of critical characteristics, including the selection of an appropriate sampling plan. The safety function of the item is typically controlled by the end user of the item (licensee). If the licensee elects to procure such items as basic components, the licensee must invoke 10 CFR 50, Appendix B in the procurement documents. The supplier has the responsibility of assuring that it is supplying either an item which has been designed/manufactured under the applicable provisions of Part 50, Appendix B or of dedicating a commercial grade item under its own Part 50, Appendix B program. In either case, the licensee is responsible for reviewing and approving the suppliers quality assurance program controlling these activities. In order to achieve Part 50, Appendix B equivalency when dedicating items such as material, a supplier must consider all of the applicable specification requirements to be critical and verify that these specification requirements have been met using a sampling plan with a confidence level such as discussed above since, typically, the supplier does not know the end application (safety function) of the item.

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In accordance with Section 2.790 of 10 CFR, a copy of this letter will be placed in the NRC's Public Document Room.

No response to this letter is required.

Sincerely,

Robert M. Gallo

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

Docket No.: 99901179

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0040
Vol. 19, No. 4

2. TITLE AND SUBTITLE

Licensee Contractor and Vendor Inspection Status Report
Quarterly Report
October - December 1995

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1996

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Quarterly

7. PERIOD COVERED (Inclusive Dates)

October - December 1995

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from October through December 1995.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Vendor Inspection

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS MAIL
POSTAGE AND FEES PAID
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PERMIT NO. G-67

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