Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
January – March 1995

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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

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ABSTRACT

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

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

March 1, 1995

Mr. Don Girard, President
Girard Development, Inc.
P.O. Box 338
Rancho Cordova, CA 95741

SUBJECT: NRC INSPECTION NO. 99901283/95-01

Dear Mr. Girard:

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection of your facility at Rancho Cordova, California, conducted by Messrs. U. Potapovs and B.H. Rogers of this office on January 24 and 25, 1995, and the discussions of their findings with you at the conclusion of the inspection. The inspection was conducted to evaluate your quality assurance program and its implementation in selected areas such as (1) control of purchased material and services, (2) material and traceability control, (3) supplier audits, (4) design control, and (5) a review of your program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR).

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

Although your quality assurance program implementation was generally satisfactory, the inspection identified that it did not meet applicable NRC requirements in the areas of corrective action, qualification of suppliers of safety related services, and certification of material. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter.

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

The responses requested by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Public Law No. 96-511. In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and the enclosed inspection report will be placed in the NRC Public Document Room.

D. Girard

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If there are any questions concerning this inspection we will be pleased to discuss them with you.

Sincerely,

A handwritten signature in black ink, appearing to read "Robert M. Gallo", with a long horizontal flourish extending to the right.

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

Docket No. 99901283

Enclosures:

1. Notice of Nonconformance
2. Inspection Report 99901283/95-01

NOTICE OF NONCONFORMANCE

Girard Development, Inc.
Rancho Cordova, California

Docket No.: 99901283/95-01

Based on the results of an NRC inspection conducted on January 24 and 25, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

- A. Criterion XVI, "Corrective Action," of Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, requires, in part, that measures shall be established to assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, Girard Development, Inc. (GDI) did not establish measures to assure that conditions adverse to quality were promptly corrected. Specifically, a 1989 customer audit finding concerning the lack of design review for all Girard tube clamps was not corrected until 1993, when another external audit identified the same deficiency. (Nonconformance 99901283/95-01-01)

- B. Criterion VII "Control of Purchased Material, Equipment and Services," of Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, requires, in part, that measures shall be established to assure that purchased material, equipment, and services conform to the procurement documents and that the effectiveness of control of quality by contractors and subcontractors shall be assessed at intervals consistent with the importance, complexity, and quantity of the product or service.

Contrary to the above, GDI did not establish measures to assure that purchased services conformed to the procurement documents and to assess the effectiveness of control of quality by subcontractors at intervals consistent with the importance, complexity, and quantity of the service. Specifically, GDI had contracted Pacific Consulting Engineers (PCE) to verify the adequacy of GDI's tube clamp design and had not taken action to assess PCE to assure the effectiveness of control of quality and to assure that the services conformed to procurement documents. (Nonconformance 99901283/95-01-02)

- C. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures appropriate to the circumstances and shall be accomplished in accordance with these instructions or procedures.

GDI's Quality Assurance Manual commits to compliance with the procedures and instructions contained in Subarticle NCA 3800 of Section III of the ASME Boiler and Pressure Vessel Code (Code).

Paragraph NCA 3820(a) of Section III of the Code states that a Material Manufacturer shall hold a Quality System Certificate (Materials) issued by the Society or have his Quality System Program surveyed, qualified, and audited by the Material Supplier holding a Quality System Certificate or by the Certificate Holder for the items.

Contrary to the above, GDI did not accomplish activities affecting quality in accordance with established procedures. Specifically, GDI did not accomplish the procurement activities associated with the supply of tube clamps under Georgia Power Company's purchase order 60191890000, which required compliance with Section III of the ASME Code, in accordance with the instructions and procedures contained in Subarticle NCA 3800 of Section III of the ASME Code, as required by GDI's Quality Assurance Manual. GDI was neither a Quality System Certificate holder nor had been qualified by a Certificate holder to supply material in accordance with NCA 3800, yet issued certification to Georgia Power Company stating that the tube clamps which they supplied met the requirements of Section III of the ASME Code.
(Nonconformance 99901238/95-01-03)

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 Technical Support, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed.

Dated at Rockville, Maryland
this 1st day of March, 1995

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

REPORT NO.: 99901283/95-01

ORGANIZATION: Girard Development, Inc.
P.O. Box 338
Rancho Cordova, California 95741

ORGANIZATIONAL
CONTACT: Don Girard, President

NUCLEAR INDUSTRY
ACTIVITY: Girard Development, Inc. provides split block tubing
clamps

INSPECTION DATES: January 24 and 25, 1995

LEAD INSPECTOR:

Uldis Potapovs
Vendor Inspection Section
Special Inspection Branch

2-22-95
Date

OTHER INSPECTORS:

Billy H. Rogers
Vendor Inspection Section
Special Inspection Branch

REVIEWED BY:

Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch

2/22/95
Date

APPROVED BY:

Robert M. Gallo, Chief,
Special Inspection Branch

3/1/95
Date

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspectors evaluated Girard development, Inc.'s (GDI) implementation of quality assurance measures for those activities which directly affected the quality and performance capability of their product. These activities included the control of purchased material and services, audits, material traceability, and design control. The team also reviewed GDI's program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR).

The inspection basis consisted of 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
- Section III of the ASME Boiler and Pressure Vessel Code (Code)
- Girard's Quality Assurance Manual, Revision 22

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

1.2.1 Nonconformance 95-01-01

This nonconformance, described in Section 3.4.7 of the report identifies an instance where GDI failed to establish measures for corrective action as required by Criterion XVI of Appendix B to 10 CFR 50 and did not take prompt corrective action when a condition adverse to quality was identified during a customer audit.

1.2.2 Nonconformance 95-01-02

This nonconformance, described in Section 3.4.2 of the report identifies an instance where GDI did not assess the effectiveness of the control of quality by a subcontractor as required by Criterion VII of Appendix B to 10 CFR 50. This subcontractor was hired to perform an independent design review of GDI's load capacity calculations.

1.2.3 Nonconformance 95-01-03

This nonconformance, described in Section 3.4.8 of this report, relates to GDI's failure to implement the commitments of their quality assurance manual in the procurement and certification of material supplied under the rules of the ASME Code.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of GDI

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance and Exit Meetings

During the entrance meeting on January 24, 1995, the NRC inspectors discussed the inspection scope and developed general information about GDI's products and activities. During the exit meeting on January 25, 1995, the NRC inspectors discussed their findings and observations with GDI management.

3.2 Description of Facilities

GDI supplies pipe and tube split block clamps of their own design in sizes ranging from 3/16 to 1 1/2 inches. These clamps are advertised as nuclear grade seismic tubing clamps, produced to ASME Code, Section III NF and NCA. The clamps are typically ordered and supplied to a catalogue part number which identifies a standard size and configuration. Although the clamps are designed to be used with standard Unistrut supports, neither the supports nor the bolts required to assemble the clamp are provided by GDI. The load capacity data sheets provided with these clamps are based on using them with ASTM A-354 Grade BD or equivalent bolts.

The scope of safety related work activity at GDI is presently limited to accepting customer orders and filling them, including packaging and shipping, from existing stock as well as maintaining an inventory of clamps in the warehouse. GDI also performs a limited receiving inspection on clamps supplied by the manufacturer. All other activities, including manufacturing, identification stamping, as well as any testing and dimensional inspections of the product are subcontracted.

GDI has an approved subcontractor, Southern Tool, Inc., Anniston, Alabama (STI), which produces the clamps using GDI owned tooling and performs all inspection/verification activities. The clamps are investment cast and require no machining operations.

3.3 10 CFR Part 21 PROGRAM

The inspectors verified that GDI had posted a current copy of 10 CFR Part 21 as required by 10 CFR Part 21.6. However the inspector's review of GDI's 10 CFR Part 21 implementing procedure indicated a weakness in that it contained several inaccuracies and did not appear to be based on the current version of 10 CFR Part 21 which GDI had posted. The inspectors reviewed the current version of 10 CFR Part 21 with GDI and discussed the important concepts related to evaluation of deviations and the reporting of defects. GDI indicated that the 10 CFR Part 21 implementing procedure would be updated based on this discussion. GDI also stated that it had not encountered any deviations which required a 10 CFR Part 21 evaluation.

3.4 Quality Assurance Program Implementation

GDI's Quality Assurance Program is described in a manual which committed to meet the applicable requirements of Appendix B to 10 CFR 50 and Subarticle NCA 3800 of Section III of the ASME Code. The manual was issued in 1980 and has been periodically revised with the current revision being Revision 22 dated March 15, 1992. Selected areas of the manual and its implementation were examined during this inspection as indicated below.

3.4.1 Design Control

The inspectors identified weaknesses in GDI's quality assurance manual in this area. The manual states that design control is achieved through compliance with ASME Section III requirements and the generation of a load capacity data sheet based on allowable stresses for an approved material after the product is designed. The manual also states that it is GDI's intent not to change the design, once it is established. The manual does not address design verification as required by Criterion III of Appendix B to 10 CFR 50.

After this issue was identified during several customer audits, GDI contracted with an engineering company in 1993 to perform an independent review of their load capacity data sheets. However, the quality assurance manual was not revised to include this requirement.

The inspectors also noted that Section III of the ASME Code contains requirements for design review which are not addressed in GDI's quality assurance manual. Specifically, Paragraph NCA 3551.2 requires load capacity data sheets to be certified by a Registered Professional Engineer who is qualified in accordance with ANSI/ASME N626.3. Without such certification, the product can not be considered as complying with the requirements of Section III of the Code.

The inspectors also questioned GDI's classification of the product as "Seismic Tube Clamps" and whether GDI had performed any seismic analyses of their product for a specific application. GDI stated that the use of the word "seismic" in the product's name was not to imply that the product had been seismically qualified to a specific set of seismic parameters but that a licensee could use the product's specified load capacities, along with data on the fastener and mounting surface in the licensee's seismic calculations to determine if the system (tube clamp, fastener, and mounting surface) would meet the site seismic requirements. Based on a review of the GDI product literature, customer purchase orders, and documentation provided to the customers, the inspectors concluded that, although the word "seismic" is a part of the product name, GDI did not represent their product as seismically qualified in their product literature or in documentation provided to the users at delivery.

3.4.2 Control of Purchased Material and Services

As noted in paragraph 3.2, GDI has an approved subcontractor, STI, who manufactures the clamps using an investment casting process. STI also performs all examinations and tests and heat treatment required by the material specification and issues a material test report and material certification to GDI. GDI has qualified and approved STI's quality assurance program as meeting the requirements of Appendix B to 10 CFR 50 and Subarticle NCA 3800 of Section III of the ASME Code. GDI contracted with an outside auditor to perform the audits of STI. The same auditor was used for the last three annual audits and adequate documentation was available to establish the auditor's qualifications to perform these audits.

The inspectors noted that, as corrective action for a 1993 customer audit, GDI purchased the services of Pacific Consulting Engineers (PCE) to review GDI's design calculations supporting the load capacity data sheets for seven types of tube clamps. PCE had documented this review in a letter to GDI dated June 17, 1993, which concluded that GDI's calculations were complete and correct, the sketches were labeled correctly with the calculations, the formulas used were per the applicable portions of the 1989 ASME code, Section III, Subsection NF and its addendum, allowable stresses used were per the ASME code and cut sheets for materials were per the ASME code. The letter also certified that the calculation process and math was done correctly and completely. However, discussion with GDI indicated that GDI had selected PCE for this service solely on the basis of the PCE engineer performing the review being a registered professional engineer and that GDI had not taken any actions to qualify PCE, or the performing engineer, as an approved supplier of a safety related service. This was identified as a nonconformance to Criterion VII, "Control of Purchased Material, Equipment, and Services," of 10 CFR Part 50 Appendix B. (Nonconformance 99901283/95-01-02)

3.4.3 Material Receiving, Identification, and Warehousing

GDI's quality assurance manual contains a receiving checklist to document verification of material to purchase order requirements. The clamps are received in boxes marked with the material heat code (one heat of material in a box). Additionally, each clamp half is individually stamped with the heat code. As a part of the receiving process, the heat code identification is verified on a sample basis and the chemical analyses/mechanical properties reported on suppliers certification is checked against the applicable material specification. A table containing the material specification requirements was posted in the receiving area. No dimensional inspections are performed at this time. GDI relies on the clamp manufacturer to perform this function. Material is stored on warehouse shelves (full containers only) or on a large table in the warehouse area (partial containers). Spot checking of heat code markings by the inspectors indicated adequate traceability control in this area.

3.4.4 Training

GDI has only one full time employee (the owner and president) and uses part time employees for several activities. GDI had a letter on file dated December 1, 1983, which stated that the listed employees had been trained in the Quality Assurance requirements of the company relative to their respective jobs. This letter had been updated on several occasion to reflect new part time employees and the work they performed. The last update had been in 1988 which GDI stated had added the personnel presently used by GDI. These activities included packaging, shipping, and secretarial duties. GDI stated that the president prepares packing lists and certification and the part time employees use the packing lists to assemble and box the material.

3.4.5 Internal Audits

GDI's quality assurance manual requires internal audits to be performed every odd year. Due to the size of GDI (one full time employee), GDI contracted with an outside auditor to perform the internal audits of GDI. GDI had used the same auditor for the previous three audits and had documentation onsite to establish his qualification for performing the audits. The inspectors reviewed the three most recent audits which were performed July 18, 1989, May 23, 1991, and May 25, 1993. The audit reports appeared to document well performed internal audits, however, the auditor failed to identify the lack of design review for the GDI tube clamps or the material certification issues discussed in Sections 3.4.6 and 3.4.8 of the report, respectively.

3.4.6 Audits by Customers

GDI has supplied numerous licensees with safety related tube clamps and, consequently, has been audited by many licensees and other organizations. The inspectors reviewed audit reports and related correspondence for a number of these audits including audits performed by Tennessee Valley Authority on February 2, 1988, Wisconsin Electric Power Company on February 16, 1989, Florida Power and Light Company on April 27, 1989, Louisiana Power and Light on June 27, 1989, Washington Public Power Supply System on August, 29, 1990, Flour Constructors International on February 4, 1992, a NUPIC Joint Utility Audit led by Southern California Edison Company on April 20, 1993, and Bechtel on September 8, 1993.

The inspectors noted that Wisconsin Electric Power Company (WEPC) documented the February 16, 1989, supplier audit of GDI in a letter dated March 16, 1989. This letter contained a comment stating that "Drawing and load capacity sheets have no independent review. This does not meet the requirements of 10 CFR Part 50 Appendix B nor standard engineering practices." Although the WEPC letter requested a written GDI response to all items, including comments, GDI had no documentation on file to indicate that a response to this comment had been provided. Since there was documentation on file related to other WEPC findings, it would appear that GDI had not provided WEPC with a written response to the comment.

Washington Public Power Supply System (WPPSS) had performed a supplier audit of GDI on August 29, 1990. The WPPSS letter to GDI dated October 2, 1990, documenting the audit, indicated that there was no objective evidence of an independent design review. The letter continued on to say "as GDI is a one person organization, the Supply System has agreed to perform this independent review."

NUPIC had performed a Joint Utility audit of GDI on April 20 through 22, 1993. The audit led by Southern California Edison, was documented in Audit Report No. GIRD-1-93, and transmitted with a letter dated May 20, 1993. The NUPIC audit covered areas such as purchase documents, design, procurement, material control, handling, storage and shipping, document control, and program compliance. The NUPIC audit identified one deficiency documented on Corrective Action Request (CAR) S-1428 which stated, in part, that no objective evidence was available to support that an independent verification of design adequacy was performed. Subsequently, Girard had its design calculations reviewed by Pacific Consulting Engineers (PCE) as a corrective action to the CAR issued by NUPIC. The inspectors also noted that the NUPIC audit report, in the section titled "Quality History," stated that GDI had been audited by WPPSS in August 1990, and that no deficiencies were identified. The inspectors observed that, although the 1990 WPPSS audit report did not characterize the lack of design verification by GDI as a deficiency, it did clearly point this out as a weakness in the GDI quality assurance system. The inspectors did not agree with NUPIC's characterization of GDI's quality history.

3.4.7 Nonconformances and Corrective Action

Although GDI's quality assurance manual discusses corrective action for defective hardware discovered before or after delivery to the customer, it does not address corrective actions for other identified nonconforming conditions such as component design.

The inspectors observed that a nonconforming condition in the tube clamp design had been identified to GDI in 1989 by WEPC and in 1990 by WPPSS. On both occasions the licensees decided to perform their own review of the GDI design to verify adequacy for their application. In each case, this review was applicable only to the utility performing the review, and did not suffice as a design verification supporting any additional safety related sales by GDI. GDI did not take any corrective action for the lack of design verification until the item was presented as a finding in the 1993 NUPIC audit which required corrective action prior to the purchase of the GDI product. The inspectors concluded that a condition adverse to quality had been identified to GDI by a licensee in 1989 and 1990 but GDI had not taken any corrective action until 1993. This was identified as a nonconformance to Criterion XVI, "Corrective Action," of 10 CFR Part 50 Appendix B. (Nonconformance 99901283/95-01-01)

3.4.8 Certification of Material

Selected customer purchase orders (PO) and related documents were reviewed to determine whether adequate controls were provided by GDI to assure compliance with the customer specified technical and quality requirements. The inspectors observed that the clamps were generally ordered and supplied to quality assurance requirements of Appendix B to 10 CFR 50 and that numerous orders also required compliance with the applicable requirements of the ASME Code.

One of the POs reviewed was Georgia Power Company's (GPC) order 60191890000, dated November 7, 1994, for 12 part number 3/8 T 2D clamps. The material was specified as SA 351 Grade CF8, to be supplied as meeting ASME Code, Section III, Class 2 requirements. Additionally, the purchase documents required the material to be provided in accordance with GDI's quality assurance program, revision 22 conforming to ASME Code, Section III, Subsection NCA 3800. The inspectors noted that GDI's files did not contain any documentation indicating that GDI had ever been audited and qualified by GPC.

GDI supplied this material along with their Certificate of Conformance (CoC) dated November 7, 1994, attesting compliance with the PO requirements and containing a statement that this material meets the requirements of ASME Code, Section III, Class 2. GDI also provided a material certification issued by STI, dated June 11, 1985, which certified that the material meets ASTM Specification A 351 and was produced in accordance with the ASME Code, Section III, Subsection NCA, Paragraph 3800, as applicable.

STI also certified that the material had been heat treated to 1900 degrees F and water quenched per ASTM A 743, and that the material was produced in accordance with STI's quality assurance manual, dated July 17, 1983, Revision 6. STI's certification included a test report of the chemical and physical properties of the material, provided to STI by Spectrochemical Laboratories, Pittsburgh, PA.

The GPC PO was typical of several other licensee orders which invoked ASME Code requirements. After a review of the documentation related to these orders, the inspectors concluded that several of these procurements may not fully comply with the ASME Code.

In general, unless the licensee is an ASME certificate holder, the licensee can not purchase material from suppliers that have not been accredited by ASME (GDI is in this category) and be in compliance with the ASME Code requirements. Material can be certified as meeting Code requirements only by organizations that possess Quality System Certificates (QSC), or N-Type Certificates issued by ASME. Material manufacturers not holding such certificates can, however, certify that the material has been produced in accordance with a quality assurance program which meets the requirements of Subsection NCA, subarticle 3800 of the Code. Under these circumstances, an ASME certificate holder can accept (and sell) this material and certify that the material meets the Code requirements, providing that this certificate

holder has qualified (by audit) the quality assurance program referenced in the manufacturer's certification as complying with NCA 3800. Since the procurement chain in several of the document packages reviewed apparently did not include any organizations accredited by the ASME, such material should not have been certified as meeting ASME Code requirements. Certification of material to the requirements of the ASME Code without fully complying with the applicable Code provisions was identified as a nonconformance to Criterion V, "Instructions, Procedures and Drawings" of 10 CFR Part 50 Appendix B. (Nonconformance 99901283/95-01-03)

4 PERSONS CONTACTED

The NRC staff participating in the inspection and GDI personnel contacted during the inspection are listed below. All individuals listed attended the entrance and exit meetings.

Girard Development, Inc.

Girard, D. President

U.S. Nuclear Regulatory Commission

Potapovs, U. Team Leader, VIS/TSIB
Rogers, B. H. Reactor Engineer, VIS/TSIB

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

March 22, 1995

Mr. William Blackwell, President
Mackson Inc.
2346 Southway Drive
Rock Hill, SC 29731

SUBJECT: NRC INSPECTION NO. 99901179/95-01

Dear Mr. Blackwell:

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection of your facility at Rock Hill, South Carolina, conducted by Messrs. U. Potapovs and R.P. McIntyre of this office on February 15 through 17, 1995, and the discussions of their findings with you and members of your staff at the conclusion of the inspection. The inspection was conducted to evaluate your quality assurance program and its implementation in selected areas such as (1) control of purchased material and services, (2) material and traceability control, (3) supplier audits, and (4) a review of your program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR).

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

Our review of your commercial grade item dedication activities indicated that, although your quality assurance program had established good controls in several areas, your material sampling practices did not fully meet NRC requirements. Specifically, your program did not contain a basis or objective evidence to show that your destructive sampling plan for verifying critical characteristics of material provides reasonable assurance that the material meets all of the applicable procurement document requirements. The inspection also identified certain weaknesses in the areas of ASME Code material upgrading and in the documentation of corrective actions. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter.

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

The responses requested by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Public Law No. 96-511. In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and the enclosed inspection report will be placed in the NRC Public Document Room.

W. Blackwell

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If there are any questions concerning this inspection we will be pleased to discuss them with you.

Sincerely,

A handwritten signature in dark ink, appearing to read "Robert M. Gallo", with a large, stylized loop at the end.

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

Docket No. 99901179

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99901179/95-01

NOTICE OF NONCONFORMANCE

Mackson, Incorporated
Rock Hill, South Carolina

Docket No.: 99901179/95-01

Based on the results of an NRC inspection conducted on February 15 through 17, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

- A. Criterion VII, "Control of Purchased Material, Equipment and Services," of Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, requires, in part, that measures shall be established to assure that purchased material conforms to procurement documents.

Contrary to the above, Mackson had not established a documented basis to substantiate that its sampling plan for verifying critical characteristics such as dimensions and chemical and mechanical properties provided reasonable assurance that dedicated commercial grade items supplied met applicable procurement document requirements.
(99901179/95-01-01)

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

Dated at Rockville, Maryland
this ____ day of _____, 1995

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

REPORT NO.: 99901179/95-01

ORGANIZATION: Mackson, Inc.
2346 Southway Drive
Rock Hill, South Carolina 29730

ORGANIZATIONAL
CONTACT: Thomas Sharp, Manager
Quality Assurance

NUCLEAR INDUSTRY
ACTIVITY: Mackson, Inc. is a supplier of threaded fasteners and
metal products. Fastener products comprise more than
90% of their business.

INSPECTION DATES: February 15 through 17, 1995

LEAD INSPECTOR:

Uldis Potapovs
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

3/21/95
Date

OTHER INSPECTORS:

Richard P. McIntyre, VIS/TSIB

REVIEWED BY:

Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch

3/21/95
Date

APPROVED BY:

Robert M. Gallo, Chief, TSIB

3/22/95
Date

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspectors evaluated Mackson Inc.'s (Mackson's) implementation of quality assurance measures for those activities which directly affected the quality and performance capability of their product. These activities included the control of purchased material and services, audits, material traceability, and commercial grade item dedication. The team also reviewed Mackson's program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR).

The inspection basis consisted of 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
- Section III of the ASME Boiler and Pressure Vessel Code (Code)
- Mackson's Quality Systems Manual, Revision 4, dated October 11, 1994

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

1.2.1 Nonconformance 95-01-01

This nonconformance, described in Section 3.4.2.2 of the report identifies the failure of Mackson's quality assurance program to establish a documented basis to substantiate that its sampling plan for verifying critical characteristics provides reasonable assurance that the dedicated items meet all of the applicable procurement specification requirements.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of Mackson

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance and Exit Meetings

During the entrance meeting on February 15, 1995, the NRC inspectors discussed the inspection scope and developed general information about Mackson's products and activities. During the exit meeting on February 17, 1995, the NRC inspectors discussed their findings and observations with Mackson management.

3.2 Description of Facilities

Mackson has been accredited by the American Society of Mechanical Engineers (ASME) to supply ferrous and nonferrous bars, threaded fasteners, castings, forgings, plate, seamless fittings, flanges, NPT stamped tubular products, and similar items. The scope of their Quality Systems Certificate (QSC) also includes the qualification of material manufacturers and suppliers and suppliers of subcontracted services as well as upgrading of stock material. Their QSC expires on March 3, 1997.

According to Mackson management, threaded fastener products currently comprise more than 90% of their nuclear sales. Most of their stock material larger than 1/4 inch diameter is reportedly purchased from qualified suppliers/manufacturers although some upgrading is performed. Typical upgrading and dedication activities including chemical analyses, mechanical testing, and nondestructive examination (NDE) are subcontracted to qualified vendors. In-house activities are limited to receiving inspection (visual), dimensional inspection, and warehousing. No machining operations are performed.

3.3 10 CFR Part 21 Program

3.3.1 Procedures and Implementation

The latest issue of 10 CFR Part 21 along with Section 206 of the Energy Reorganization Act of 1974 and Mackson's implementing Quality Procedure V-A, "Reporting 10CFR21 Discrepancies," Revision 2, dated May 31, 1994, were observed to be posted as required by the regulation. Review of the implementing procedure determined general compliance with the regulation. The inspectors noted that in several instances the term "defect" was used in place of "deviation." These terms are specifically defined in the regulation and can not be interchanged.

This weakness was discussed with the QA Manager who indicated that the implementing procedure would be revised to address the concern.

3.4 Quality Assurance Program

Mackson's Quality Assurance Program is described in their Quality Systems Manual (QSM), Revision 4, dated October 11, 1994, which is committed to meeting the applicable requirements of Appendix B to 10 CFR Part 50, Subarticle NCA 3800 of Section III of the ASME Code, and the American National Standards Institute (ANSI) Standard N45.2. The quality assurance program is implemented through the procedures identified in the Mackson Quality Procedure Manual (QPM), Revision 6, dated July 8, 1994. Selected areas of the manual and its implementation were examined during this inspection as indicated below.

3.4.1 Material Supplied to ASME Code Requirements

Mackson's QSM and its implementing procedures require that purchase orders (PO) for Code material and services be placed only with vendors appearing on

Mackson's Approved Vendor List (AVL). Vendors are placed on the AVL based on holding a valid ASME QSC or following the completion of a satisfactory survey by Mackson in accordance with the requirements of Subarticle NCA 3800 of ASME Code, Section III.

Mackson's procedures for upgrading of stock material were reviewed and determined to be in compliance with the applicable requirements of the ASME Code. These procedures require the performance of all tests specified in the applicable material specification on each piece of the stock material or, when traceability has been established, the performance of the required tests on one sample of each heat of the material. Chemical analysis is required on each piece of the material in either case.

Small products as defined in paragraph NX-2610 of ASME Code, Section III are supplied under the provisions of that paragraph only when authorized by the customer. Mackson's procedures require that material supplied under NX-2610 comply with the requirements of NCA-3867.4 and NCA-3866.6 with respect to certification and identification/markings, respectively. Mackson's procedures permit such material to be supplied under their commercial grade dedication program. Specifically, the material manufacturer's commercial grade quality program is reviewed by Mackson and the material is sampled for chemical composition and mechanical properties initially and during every fifth shipment thereafter. Material supplied in this manner is not certified under Mackson's QSC number.

3.4.1.1 Control of Subtier Vendors and Documentation of Basis for Material Certification

Mackson purchases most of the material to be supplied under their QSC from suppliers that have been qualified by survey as having quality assurance programs meeting the requirements of NCA-3800. Mackson may perform additional testing on this material before issuing a certified material test report (CMTR). To evaluate Mackson's quality assurance program implementation, the inspectors reviewed several recently shipped customer orders as summarized in the following paragraphs.

- Duke Power Co. PO F45588-K3, dated October 26, 1994, for 1 1/8-7-2A threaded rod conforming to SA 193-GR7, in accordance with ASME Sec. III.

Mackson's certification of this material was accompanied by a Certificate of Conformance (CoC) issued by Plainville Mfg. Co. (PM) who is on Mackson's AVL as a material manufacturer qualified in accordance with ASME NCA-3800. PM's CoC included chemical analysis, tensile test results, impact test results, hardness test results, magnetic particle test results, and contained a statement that the raw material was supplied by North Star Steel Co. The corresponding test reports and the mill heat analysis from North Star Steel Co. were provided with PM's CoC.

PM's CoC did not describe the material heat treatment, however, the file contained a heat treatment certificate issued by Copperweld Steel Co.

This certificate was issued to Hy Alloy Steel Co., while the material was shipped to A.M. Castle & Co., who apparently supplied the steel to PM.

None of the manufacturers/suppliers besides PM were listed on Mackson's AVL and it could not be determined how some of the lower tier vendors were qualified to provide material or services.

The inspectors noted that under the provisions of NCA-3800, material suppliers who do not possess QSCs may not qualify other material suppliers. Thus, unless all of the subtier vendors were on PM's AVL, the material obtained from A.M. Castle would need to be treated as stock material (each piece tested) and the heat treatment certification by Copperweld could not be considered as validated. The subtier vendor qualification issue was discussed with Mackson's management for potential followup with their suppliers.

- Carolina Power and Light PO 7N3894AS, dated April 01, 1994, for 1 1/4-8 threaded rod conforming to SA 193-GRB7, in accordance with ASME Sec. III, Class 2, 1974 Edition, 1976 winter Addenda.

Mackson's certification of this material was based on a CMTR issued by A&G Engineering, dated January 29, 1993, under their ASME QSC. (Note: A&G's QSC was terminated on March 18, 1993, for nonpayment of fees). A&G's CMTR reported the material's chemical analyses, tensile properties, impact test results, heat treated condition, and attested that visual and magnetic particle testing had been performed. Test reports supporting the information included in the CMTR were not available. The material producing mill was not identified on the CMTR and the original mill heat analysis report was not provided.

The inspectors noted that, while Mackson's procurement of this material based only on a CMTR issued under A&G's QSC appeared to have complied with the ASME Code requirements at the time of this procurement, the material traceability to the producing mill was not demonstrated by the available documentation. The inspectors also advised Mackson management that the NRC was preparing to issued an Information Notice advising licensees that A&G may have applied potentially nonconforming fasteners to the nuclear industry. The inspectors further noted that Mackson did not appear to implement a consistent policy in determining what quality records need to be provided by their suppliers. This concern is demonstrated in the two examples discussed above and was identified as a weakness in Mackson's quality program. (Also, see discussion in paragraph 3.4.4)

3.4.2 Non-Code Material Supplied to 10 CFR 50, Appendix B Requirements (Commercial Grade Item Dedication Program)

3.4.2.1 Methodology

The requirements for Mackson's commercial grade item (CGI) dedication process are prescribed in Section III-I, "Dedication of Commercial Grade Material," Revision 2, dated July 8, 1994, of the Mackson QPM. The following sections of

the QPM provide additional requirements for performing and documenting various dedication activities and were reviewed during the inspection:

- IV-A, "Visual Exam of Bolts, Screws, and Studs," Revision 1, dated January 7, 1992
- IV-B, "Visual Exam of Nuts," Revision 1, dated January 7, 1992
- IV-C, "Visual Exam of Threaded Rod," Revision 1, dated January 7, 1992
- IV-D, "Visual Exam of Washers," Revision 1, dated January 7, 1992
- IV-E, "Visual Exam of Piping Material," Revision 1, dated January 7, 1992
- IV-M, "Verification of Material to P.O.," Revision 1, dated January 7, 1992
- IV-N, "Material Shipped to Service Vendors," Revision 1, dated January 7, 1992
- IV-O, "Material For Chemical and Physical," Revision 1, dated January 7, 1992
- XII-B, "Proper Use of Thread Ring Gages," Revision 0, dated February 26, 1993
- XII-C, "Proper Use of Thread Plug Gages," Revision 0, dated February 26, 1993
- XII-D, "Proper Use of Calipers, Revision 0, dated February 26, 1993
- XII-E, "Qualification Requirements for Dimensional Inspector," Revision 0, dated February 26, 1993

Mackson selects critical characteristics for CGIs based on the testing requirements specified by the applicable ASME Boiler and Pressure Vessel Code, Section II, "Materials," or American Society for Testing and Materials (ASTM) material specifications and any other testing requirements specified by the customer. For purposes of testing, the critical characteristics of the material include the chemical, mechanical, and dimensional requirements of the applicable material specification. Testing is performed by testing laboratories who are listed on the Mackson Approved Vendor List. The verification of critical characteristics is performed using sampling plans as discussed in Section 3.4.2.2 of this inspection report.

Material to be dedicated by Mackson, is purchased, when possible, with CMTRs traceable to the material manufacturer. Material suppliers of CGIs are chosen from the Mackson list of "Suppliers Approved Based on Performance Record." Suppliers are placed on this list after the receipt and testing of six consecutive purchases identifies zero deficiencies with the critical

characteristics of the material. Once a supplier is placed on this list, the sample size required for visual and dimensional inspections is decreased by 50 percent. However, the accept/reject rate remains the same.

The total verification of critical characteristics for CGIs consists of a receipt inspection, visual and dimensional inspections, and a review of the results from chemical and mechanical tests performed on a sample of the material, and serve as the basis for Mackson to issue a certification statement that the dedicated CGIs meet the applicable material and customer PO requirements. The NRC inspectors concluded that Mackson's CGI dedication program addresses the essential elements of the dedication process and that sufficient guidance for performing activities such as inspection and testing are provided.

The NRC inspectors considered the following to be strengths in Mackson's dedication program:

- Mackson selects critical characteristics based on the testing and material requirements of the applicable material specification and its customers.
- Mackson attempts to obtain material certification for each item or lot of items supplied from the manufacturer (qualified and non-qualified manufacturers) and reviews the certification, when received, for conformance with the applicable material specification.

The NRC inspectors considered the following to be a weakness in Mackson's CGI dedication program:

- Mackson has no documented bases, (qualitative or quantitative) for its dimensional and destructive sample plan used in verifying critical characteristics (see Section 3.4.2.2 of this report).

3.4.2.2 Use of Sampling in the CGI Dedication Process

Material purchased as commercial grade from nonqualified suppliers or from suppliers included on the list of Suppliers Approved Based on Performance Record and dedicated as basic components, are inspected and tested in accordance with the following sample plans unless a customer specifies another sampling plan.

1. Visual inspection is performed as follows for nonqualified suppliers:

<u>LOT SIZE</u>	<u>SAMPLE SIZE</u>	<u>ACCEPT/REJECT*</u>
0-50	16	0/1
51-90	26	0/1
91-150	40	1/2
151-280	64	1/2
281-500	100	2/3
501-1200	160	3/4

1201-3200	250	5/6
3201-10000	400	7/8

2. Visual inspection is performed as follows for suppliers approved based on performance record:

<u>LOT SIZE</u>	<u>SAMPLE SIZE</u>	<u>ACCEPT/REJECT*</u>
0-50	8	0/1
51-90	13	0/1
91-150	20	1/2
151-280	32	1/2
281-500	50	2/3
501-1200	80	3/4
1201-3200	125	5/6
3201-10000	200	7/8

3. Dimensional inspection is performed as follows for nonqualified suppliers:

<u>LOT SIZE</u>	<u>SAMPLE SIZE</u>	<u>ACCEPT/REJECT*</u>
2-8	4	0/1
9-15	4	0/1
16-25	4	0/1
26-50	4	0/1
51-90	6	0/1
91-150	6	0/1
151-280	6	0/1
281-500	6	0/1
501-1200	10	0/1
1201-3200	10	0/1
3201-10000	10	0/1

4. Dimensional inspection is performed as follows for suppliers approved based on performance record:

<u>LOT SIZE</u>	<u>SAMPLE SIZE</u>	<u>ACCEPT/REJECT*</u>
2-8	2	0/1
9-15	2	0/1
16-25	2	0/1
26-50	2	0/1
51-90	3	0/1
91-150	3	0/1
151-280	3	0/1
281-500	3	0/1
501-1200	5	0/1
1201-3200	5	0/1
3201-10000	5	0/1

5. According to Mackson, destructive testing to verify critical characteristics such as chemical and mechanical properties are performed on fasteners (bolts, screws and studs) and other CGIs being dedicated in accordance with the following sample plan:

<u>LOT SIZE</u>	<u>SAMPLE SIZE</u>	<u>ACCEPT/REJECT*</u>
0-50	2	0/1
51-500	3	0/1
501-35000	5	0/1

* Accept or Reject lot on this number of defects per sample size.

It is also noted that Mackson states a lot shall be defined as a single material type, size and heat.

The NRC discussed the use of Mackson's sampling methodology in detail during the inspection and asked Mackson to provide the bases for its destructive sampling plan as providing reasonable assurance that the fasteners not tested had the required material chemistry and strength and other properties required by the applicable material specification. According to Mackson, the sampling plans discussed above are not based on statistical methods but are considered adequate for the required testing. Mackson uses its destructive sampling plan for dedicating CGIs regardless of the size or type of product, whether the CGIs were purchased directly from the manufacturer or through a distributor, the use of single or multiple production lots, or the historical performance of the manufacturer or distributor. The NRC inspectors expressed a concern to Mackson that such a sample size could include more than one production lot or heat of material.

Mackson informed the NRC inspectors that it had not established a quantitative basis for their sampling plan nor an associated confidence level. Mackson's dimensional and destructive sample plans appear to be based on undocumented qualitative factors such as engineering judgement and experience of its personnel in inspecting and testing fasteners, and not on using quantitative statistics. The inspectors stated that this sampling method does not meet the quality requirements of Appendix B to 10 CFR Part 50. (Nonconformance 99901179/95-01).

3.4.2.3 Review of Mackson CGI Dedication Program Implementation

The NRC inspectors reviewed several completed CGI dedications to determine if the critical characteristics for the items had been properly identified and verified, and if adequate procedural controls were in place. The NRC inspectors reviewed the following completed CGI dedications.

- Mackson Purchase Order (PO), C000387, dated January 17, 1995, was for the purchase of 1800 heavy hex nuts, 5/8-11, ASTM A563, Grade C, in accordance with Duke Power PO F28721-C4. Mackson purchased these nuts from Decker Manufacturing Corporation, Albion, Michigan, a nonaudited supplier that was on the list of Suppliers Approved Based on Performance

Record. Mackson performed a receipt inspection, visual exam (markings, identification and overall workmanship) on 250 of the 1800 items and performed a dimensional exam of selected dimensions on 10 of 1800 items. Mackson also reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements and subcontracted to an approved non-code vendor for performance of chemical analyses on 52 of the 1800 nuts, a proof load test and a hardness test on 52 of the 1800 nuts.

The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification. Based on the above dedication activities, Mackson certified that the nuts were supplied in accordance with customer PO requirements and the Mackson Quality System Manual.

- Mackson PO, C002829, dated July 22, 1994, was for the purchase of 50 heavy hex bolts, 1-8 by 6 inches long, ASTM A325, Type 1, and 25 1½-7 by 6½ inches long in accordance with Duke Power POs F23053-K3 and F30399-K3. Mackson purchased these bolts from a distributor, House of Threads, Charlotte, North Carolina, a nonaudited supplier that was on the Mackson list of Suppliers Approved Based on Performance Record. NUCOR Fasteners, St. Joe, Indiana, the manufacturer of the bolts, is listed on the Mackson Non-Code Approved Vendors List for traceability only. However, during the inspectors review of the 1993 Mackson survey of NUCOR, it was identified that NUCOR does not audit subtier vendors and therefore would question the ability to maintain traceability.

Mackson supplied a total of 17 (6 1-8s and 11 1½-7s) House of Threads bolts to Duke as part of these POs. The remaining bolts supplied to Duke were purchased by Mackson from an approved Part 50 Appendix B supplier. Mackson performed a receipt inspection, a visual exam (markings, identification and overall workmanship) on 8 of each order items and a dimensional exam of selected dimensions on 2 of each order items. Mackson also reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements, and subcontracted chemical analyses, and tests of mechanical properties on 2 of each order of bolts.

The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification. Based on the above dedication activities, Mackson certified that the bolts were supplied in accordance with customer PO requirements and the Mackson Quality System Manual.

- Mackson PO, C001732, dated April 29, 1993, was for the purchase of 200 socket head cap screws of 5 different sizes ranging from 5/16-18 by 2½ to 3/4-10 by 2½, A574 in accordance with 5 Duke Power POs. Mackson purchased these bolts from Georgia Fasteners, Augusta and Savannah, Georgia, another nonaudited supplier that was on the Mackson list of Suppliers Approved Based on Performance Record. Mackson performed a receipt inspection, a visual exam (markings, identification and overall workmanship) on 32 of 200 items for each of the 5 different sizes and a

dimensional exam of selected dimensions on 3 of 200 items for each of the 5 different size cap screws. Mackson also reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements. In this case Georgia fasteners did have the CMTRs from the steel mill. A Mackson approved non-code vendor performed chemical analyses and mechanical properties tests on 3 of 200 for each of the 5 different sizes bolts.

The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification. Based on the above dedication activities, Mackson certified that the bolts were supplied in accordance with customer PO requirements and the Mackson Quality System Manual.

3.4.3 Audits

3.4.3.1 Internal Audits

The inspectors reviewed Section 11, "Audits," of Mackson's Quality System Manual, and Section III-A, "Vendor Audit" of the Quality Procedure Manual. The inspectors reviewed the lead auditor qualification and certification records for the Mackson QA Manager and also for two outside subcontractors who have performed internal audits of Mackson. All documentation appeared current. The team also reviewed related internal audit documentation such as annual audit plans, audit checklists, audit reports and responses to audit report findings. The team verified that appropriate corrective action was performed by the Mackson QA Manager for internal audit report findings.

3.4.3.2 External Audits

External vendor audits are predominantly performed by the QA Manager, but the President of Mackson is also an certified lead auditor and performs some external audits along with the QA Manager. Mackson maintains a separate Approved Vendor List for ASME code vendors and non-code vendors. The inspectors reviewed the most recent audits of Plainville Manufacturing Company and Texas Bolt Company (ASME code vendors) and JLK Industries, Inc. and NUCOR Fasteners (non-code vendors). All audits reviewed included appropriate documentation of objective evidence for the areas reviewed. However, the inspectors did identify that the JLK and NUCOR audits only verified traceability control at the audited vendor, and did not verify sub-tier vendor traceability.

The team concluded that Mackson was implementing an appropriate internal and external audit program per their QSM requirements.

3.4.3.3 Audits by Customers

To evaluate the effectiveness of audits by Mackson's nuclear customers, the inspectors reviewed the latest audit performed by the Nuclear Utilities Procurement Issues Committee (NUPIC). The audit was conducted on March 16-19, 1993, with South Carolina Electric & Gas Company providing the lead auditor. The audit resulted in two findings: failure to verify thread dimensions as a

part of the dedication process, and failure to present a Nuclear Quote Sheet to the Quality Assurance Manager for Approval. Both findings were satisfactorily corrected. Mackson's commercial grade material dedication procedure (III-I, Revision 2) was reviewed and determined to be satisfactory. The report did not address sampling plans used in the commercial grade material dedication process.

3.4.4 Nonconforming Material and Corrective Actions

Section V-B of Mackson's Quality Procedures Manual provides guidance for the control and disposition of items and activities that are found to be in nonconformance with specified requirements. Additionally, the Quality Assurance Manager maintains a nonconformance log which provides the status of nonconforming conditions. The inspectors reviewed the nonconformance log and noted that the majority of the entries had been properly dispositioned in a timely manner. A review of selected nonconformance reports indicated that, in general, adequate documentation was provided to describe the nonconformance, identify the root cause and support the method of disposition. However, the review identified several concerns with the evaluation and disposition process as summarized below.

- NCR 2161, dated July 27, 1994 was issued to document apparently incorrect material (1/4 inch hexagonal cap screws) supplied by Non Ferrous Bolt & Manufacturing Co. The specified material was ASTM-A 193, Grade B8M. The material (heat number 701522) was supplied with a CMTR from Carpenter Technology certifying the Carbon content less than .03%. Verification tests performed by Mackson's contractor showed the Carbon content of this material to be .064%.

Replacement material, certified to the same heat number, was tested and found to be within specification suggesting that cap screws from a different heat of material than identified in the accompanying CMTR had been provided in the initial shipment.

The inspectors noted that, although the replacement material had been accepted in August, 1994, the NCR had not been closed and the root cause of this occurrence had not been identified on the NCR form.

- NCR 2167, dated November 15, 1994, was issued to document material with impact values below specified requirements received from Texas Bolt Co. Texas Bolt had supplied 20 SA-194 GR7 nuts with a CMTR showing impact values below the specified minimum. Mackson identified the discrepancy during receiving inspection.

The nuts were returned to the vendor with a request for corrective action/root cause identification. The vendor responded by stating that the impact values were low because the test samples had been taken from finished nuts rather than from a test coupon. Corrective action included reheat treating the nuts with a test coupon from the same heat of material and retesting. The results were acceptable and the NCR was closed on November 18, 1994.

The inspectors noted that the vendor's root cause analysis appeared to be inadequate in that it did not address the question of why the vendor shipped material with a CMTR showing impact test values below the specified requirements.

- NCR 2114, dated July 11, 1992, was issued to document visually identified cracking of SA-193 GR8 hexagonal cap screws supplied to Mackson by Texas Bolt Co. The defects were identified by Carolina Power and Light Co. during receiving inspection and the material was returned to Mackson.

Mackson's root cause analysis determined that the material should not have passed vendor's in-process examination and should not have been shipped in this condition. Corrective action was requested from the vendor. The inspectors noted that the root cause analysis did not address the failure to identify these defects during Mackson's receiving inspection.

- NCR 2170, dated December 20, 1994, was issued to document an instance where the material manufacturer (US Bolt Co.) was unable to provide a liquid penetrant test report for SA 193 GRB8 threaded rod after Mackson's customer requested a copy of the test report (the CMTR certified that this testing had been performed)

As a part of the corrective action, Mackson performed liquid penetrant testing on this material utilizing the services of a qualified contractor.

The examples of inadequate root cause analyses and timeliness of NCR disposition were discussed with Mackson's management as potential weaknesses in the implementation of the corrective action program. The inspectors noted that the last example appeared to be directly related to a previously identified weakness (paragraph 3.4.1.1) concerning the lack of consistent policy in defining what quality records must be provided by Mackson's suppliers.

4 PERSONS CONTACTED

The NRC staff participating in the inspection and Mackson personnel contacted during the inspection are listed below. All individuals listed attended the entrance and exit meetings.

Mackson Inc.

Blackwell, W.	President
Sharp, T.	Quality Assurance Manager
Muse, H.	Vice President, Operations

U.S. Nuclear Regulatory Commission

Potapovs, U.	Team Leader, VIS/TSIB
McIntyre, R.	Sr. Reactor Engineer, VIS/TSIB

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

January 31, 1995

Mr. Kaydell C. Bowles
Manager, Quality Assurance
Sandvik Special Metals Corporation
P.O. Box 6027
Kennewick, Washington 99336-0027

SUBJECT: NRC INSPECTION NO. 99900764/94-01

Dear Mr. Bowles:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Sandvik Special Metals Corporation (SSM), Kennewick, Washington, conducted on December 5 through 7, 1994. The NRC inspection team, led by Steven M. Matthews and comprising the other inspectors named in the report, conducted a performance-based evaluation of the SSM management, staff, and quality programs and the implementation of those programs related to the manufacture of zirconium alloy nuclear fuel clad tubing. The inspection was conducted to provide a basis for confidence that SSM zirconium alloy fuel clad tubing supplied to the U.S. nuclear industry in fuel assemblies would perform its safety function.

The NRC inspection team (a) examined technical documentation, procedures, and representative records, (b) conducted interviews, (c) held discussions, (d) listened to presentations, and (e) made various observations. On the basis of this inspection, the NRC determined that, in one instance observed, the implementation of the SSM quality system, documented in the SSM Quality Assurance Manual (QAM), Revision 1, dated April 7, 1994, did not meet the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B to 10 CFR Part 50). Therefore, we are issuing a Notice of Nonconformance (Enclosure 1) to SSM. The report (Enclosure 2) contains a detailed discussion of the areas examined and our findings.

The NRC determined that SSM had not established adequate measures to control the identification of tube specimens after burst testing used to characterize the biaxial mechanical properties of zircaloy tubes. Specifically, the identification of tube specimens was not maintained in a manner that ensured SSM the ability to perform further analysis of the failed tube specimens, where the results of further analysis of the failed tube specimens may affect the quality of the zircaloy fuel clad tubing. In the instance observed, post-testing failure analysis was both requested by the fuel vendor and necessary to assign root cause for the reduced ductility of the tube specimens and to ensure that the corrective actions taken by SSM were adequate to preclude other failed tube specimens.

The NRC identified strengths in the SSM problem analysis and corrective actions planned and taken to prevent further instances of the "whitish" discolorations observed on the outside diameter tube surfaces, documented in SSM Technical Report No. 94-009, "Evaluation of Paper Divider Trays," dated September 26, 1994, which was thorough, competent, and conclusive as to the root cause of the discolorations.

For the nonconformances cited in the NRC report of its previous inspection of SSM (documented in NRC Report No. 99900764/87-01), the NRC determined that SSM corrective actions were adequate and had been effectively implemented. The enclosed report documents closure of those nonconformances.

Please respond to this letter following the instructions specified in the enclosed notice. Your response should document the specific actions taken and any additional actions you plan in order to prevent recurrence. Please send us written response within 30 days from the date of this letter.

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

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,

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

Docket No. 99900764

Enclosures:

1. Notice of Nonconformance
2. Report No. 99900764/94-01

NOTICE OF NONCONFORMANCE

Sandvik Special Metals Corporation
Kennewick, Washington

Docket No. 99900764
Report No. 94-01

On the basis of the results of the U.S. Nuclear Regulatory Commission (NRC) inspection of Sandvik Special Metals Corporation (SSM), Kennewick, Washington, conducted on December 5 through 7, 1994, it appears that certain of your activities were not performed in accordance with NRC requirements.

Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B to 10 CFR Part 50), requires, in part, that activities affecting quality shall be prescribed by documented instructions and shall be accomplished in accordance with these instructions.

Section QAM-2-5, "Instructions, Procedures, and Drawings," of the SSM Quality Assurance Manual (QAM), Revision 0, dated December 1, 1993, required, in part, that activities and operations affecting quality of tubing shall be controlled through the use of formally approved instructions and procedures.

Contrary to the above, SSM did not control the identification of tube specimens that had failed the burst test (used to characterize the biaxial mechanical properties of zircaloy tubes) through the use of formally approved instructions or procedures, such that SSM could not perform further analysis of the failed tube specimens necessary to assign root cause for the reduced ductility of the failed specimens. Specifically, Laboratory Procedure 1300.21, "Burst Testing Zirconium Base Alloy Tubing (Closed and Open End Testing)," Revision 4, dated August 21, 1985, did not prescribe documented instructions to accomplish the identification of failed tube specimens where further analysis of the specimens may affect quality (94-01-01). (Report Section 3.5.1)

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 Technical Support, 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 the following: (a) a description of steps that have been or will be taken to correct this item; (b) a description of steps that have been or will be taken to prevent recurrence; and (c) the dates your corrective actions and preventive measures were or will be completed.

Dated at Rockville, Maryland
this 31st day of January, 1995

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

REPORT NO.: 99900764/94-01

ORGANIZATION: Sandvik Special Metals Corporation (SSM)
P.O. Box 6027
Kennewick, Washington 99336-0027

ORGANIZATIONAL
CONTACT: Kaydell C. Bowles
Manager, Quality Assurance

NUCLEAR INDUSTRY
ACTIVITY: SSM provides zirconium alloy fuel clad tubing used in
nuclear fuel assemblies supplied to the nuclear power
industry.

INSPECTION DATES: December 5 through 7, 1994

LEAD INSPECTOR:

Steven M. Matthews
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

1-25-95
Date

OTHER INSPECTORS:

David H. Brewer, VIS/TSIB
Carl J. Czajkowski, Brookhaven National Laboratory

REVIEWED BY:

Gregory C. Cwalina, Chief, VIS/TSIB

1/25/95
Date

APPROVED BY:

Robert M. Gallo, Chief, TSIB

1/31/95
Date

1 SUMMARY OF INSPECTION FINDINGS:

During this inspection, the NRC inspection team (team) evaluated SSM management, staff, and quality programs and the implementation of those programs related to the manufacture of zirconium alloy nuclear fuel clad tubing. The inspection was conducted to provide a basis for confidence that SSM zirconium alloy fuel clad tubing supplied to the U.S. nuclear industry in fuel assemblies would perform their safety function.

The inspection basis consisted of 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)
- Part 21, "Notification of Failure to Comply or Existence of a Defect," of 10 CFR
- SSM Quality Assurance Manual (QAM), Revision 1, dated April 7, 1994, which described a quality system that, according to SSM, was intended to meet the requirements of Appendix B to 10 CFR Part 50

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

1.2.1 Nonconformance 94-01-01

This nonconformance, described in Section 3.5.1 of this report, identifies an instance where the documented instructions for certain activities affecting quality and the performance of those activities failed to comply with the requirements of Criterion V of Appendix B to 10 CFR Part 50 and Section QAM-2-5 of the SSM QAM.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

During an inspection of SSM conducted on March 16 through 19, 1987, the NRC inspection team determined that certain SSM activities were not conducted in accordance with NRC requirements contractually imposed on SSM by purchase orders from nuclear fuel manufacturers. The following nonconformances, issued by the staff on April 27, 1987, and the associated SSM corrective actions, were evaluated by the team during this inspection. Based on the evaluation of SSM corrective actions, the team determined that each of the nonconformances was closed.

2.1 Nonconformance 87-01-01 (CLOSED)

Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50 and Sections 3.4.1 and 3.5.2 of QA-GA-5, Revision 14, dated February 7, 1986, the inside diameter (ID) of 19 tube hollows from ingot number 224190Q were inspected with the wrong go-no-go gauge.

During its evaluation of the SSM tube reduction process, the team found that, for the inspections observed, SSM had appropriately implemented its inspection requirements with the correct equipment. Therefore, the team concluded that SSM corrective actions taken appeared adequate to ensure that the appropriate inspection equipment was used to determine the acceptability of tube hollows.

2.2 Nonconformance 87-01-02 (CLOSED)

Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50 and Section 2.1.1.8 of Z121, "Alkaline Cleaning," Revision 10, dated June 3, 1985, the west end of Tank No. 3 did not exhibit the required identification concerning the filling of the tank with fresh cleaning solution.

During its evaluation of the SSM tube reduction process, the team found that, for the process observed, SSM had appropriately identified and labeled all production process parameters. Therefore, the team concluded that SSM corrective actions taken appeared adequate to ensure that the appropriate identification for filling the tank with fresh cleaning solution.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

According to SSM, the company was founded in 1966 as a result of the Atomic Energy Commission diversification program for the Hanford Atomic Products Facility in Richland, Washington. SSM is a wholly-owned subsidiary of Sandvik AB (formerly Sandvik Jernverks) of Sweden. Zirconium alloy (for this inspection, zircaloy, or Zr2 and Zr4) fuel clad tubing for the nuclear industry was first produced by SSM in 1968.

SSM supplies nuclear fuel manufacturers zircaloy fuel clad tubing used in fuel assemblies for commercial power reactors in the U.S., Japan, Korea, and Europe. SSM's domestic clients for fuel clad tubing include ABB Combustion Engineering Nuclear Operations (ABB CENO) for its pressurized-water reactor (PWR) fuel designs, Siemens Power Corporation - Nuclear Division (SPC-ND) for its boiling-water reactor (BWR) fuel designs. Zircaloy sleeve material was supplied to Babcock and Wilcox Company (B&W). In addition to fuel clad tubing, SSM also supplied the nuclear power industry zircaloy material for the following fuel assembly or fuel rod applications:

- instrument tubes
- guide tubes
- water rods
- plenum spacers

3.2 Entrance and Exit Meetings

During the entrance meeting in Kennewick, Washington, on December 5, 1994, the team met with members of SSM management and staff, discussed the scope of the inspection, and established contact persons for the team within the management and staff of the SSM organization.

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

Inspection participants and contacts are listed in the Appendix.

During its exit meeting on December 7, 1994, with SSM management and staff, the team summarized the inspection findings.

3.3 Tubing Production

3.3.1 Tube Reduction

To manufacture fuel clad tubing, SSM received from approved zircaloy suppliers (e.g., Teledyne Wah Chang (TWC), Albany, Oregon) either (a) extruded tube shells, 3.2 inch outside diameter (OD) x 0.790 inch wall thickness, or (b) tube reduced extrusions (TREXs) which are extruded tube shells that have been pilgered, or reduced, once by the zircaloy supplier to a 2.5 or 2.0 inch OD. The tubing is produced by reducing the tube shells or TREXs through the basic reduction steps to form a final tube hollow. The final tube hollow is then reduced to meet the final dimensional requirements for the fuel clad tubing.

3.3.1.1 Basic Reduction Steps

The basic reduction steps used by SSM, as observed by the team, to produce fuel clad tubing from either extruded tube shells or TREXs were as described below. These steps would normally be repeated until the required OD and wall thickness for the final tube hollow were achieved.

- (1) Vacuum annealing – tube shells and TREXs were annealed at 1300 - 1400°F (704 - 760°C) for at least 1-1/2 hours.
- (2) Pilgering – the process of cold reduction of the extruded tube shells or TREXs to produce tube hollows. By working the tube hollows through several pilgering passes, SSM produces fuel clad tubing to the specified dimensions. The pilgering process was referred to interchangeably as "cold pilgering," "tube reducing" or "rocking." This process relies on tube elongation by rolling back and forth (hence the term "rocking")

between two grooved dies. During this process, the tube is rotated and advanced by small amounts and pulled over a tapered stationary mandrel. Both the tube diameter and wall thickness are reduced by this process.

- (3) Alkaline cleaning – where the zircaloy tube hollow was cleaned.
- (4) Vacuum annealing – the tube hollow annealing process heats the tube hollow to a specified temperature to promote both recrystallization and softening.
- (5) Inspection and Conditioning – an inspection of the surfaces of the tube hollow (both inner and outer surfaces) for small cracks or tears. The conditioning process removed these discontinuities to prevent propagation during subsequent pilgering passes, or reductions.
- (6) Straightening – allowed the tube hollow to flow more easily during subsequent pilgering process.

3.3.1.2 Final Tube Hollow Reduction

At this stage of the tube reduction process, the final tube hollow would be ready to be rocked (pilgered) for the last time; also known as final hollow reduction (FHR). The following steps, as observed by the team, would normally be performed during the FHR process:

- (1) Final pilger – the last cold reduction pilgering pass performed on the tubing which produces the required OD and wall thickness specified for the final fuel clad tubing.
- (2) Final cleaning – the zircaloy tubing grit blast cleaned on the inner surface, followed by inner surface etching performed in a flowing acid etch bath with a proprietary rinsing system. This combination reduces residual fluoride and produces a very smooth ID surface. SSM stated that the produced finish provides iodine stress corrosion cracking resistance superior to other finishes.
- (3) Pre-anneal inspection – the ID surfaces of the tubing were inspected for flaws prior to final anneal.
- (4) Final thermal treatment – on the basis of the intended service conditions of the tubing, as determined by the fuel manufacturer, the final thermal treatment performed by SSM can be one of the following:
(a) a full recrystallization anneal, or (b) a stress relieving operation. According to SSM, the majority of tubing lots were recrystallized to produce specified mechanical properties and ductility requirements.
- (5) Belt grinding and polishing – the tubing was belt polished on the OD surface to remove all evidence of the old metal. At this stage, tube specimens were sent to the SSM laboratory for testing.

- (6) Final cleaning – as described above. After this step, the lot of tubing was sent to final inspection.

3.3.2 Beta Quench

The team observed that SSM procured material to produce fuel clad tubing from TWC. All of this material had been "early" beta quenched by TWC as forged logs. SSM stated that approximately 2% of its fuel clad tubing production received "late" beta quenching and all of this material was supplied to SPC-ND for use in BWR applications.

The team evaluated the qualification reports on the development of two methods of "late" beta quenching process performed by SSM; one used induction heating of 10 kilo hertz (kHz) which heated the tubing through the wall thickness to the required temperature; the second used induction heating of 450 kHz which heated the tubing from the outside circumference partially through the wall.

SSM personnel stated that somewhat more than 1% of their fuel clad tubing production used the first method while somewhat less than 1% was produced by the second method.

The "late" beta quench was performed on the tubing immediately prior to the last two pilgering passes. The temperature of the tubing was controlled by a two-color infrared optical pyrometer. This type of pyrometer takes information from two different portions of the infrared spectrum and thereby factors out the emissivity variations that have made single color infrared pyrometers inaccurate unless emissivity was carefully measured and controlled.

As the tubing passed through induction heating, the system that provided the through-wall beta quench maintained the temperature in the beta range for 10 seconds while the partial-wall beta quench maintained the beta range temperature for 5 seconds. Coolant was passed through the ID of the tubing during both processes; nitrogen gas was used as the coolant for the through-wall beta quench process and water for the partial-wall process.

3.4 Final Inspection

The team observed the following final inspection operations:

- (1) OD surface inspection – performed in accordance with SSM procedure ETA 641, "Post Grind OD Surface Inspection," Revision 0, dated August 30, 1991, the automated electro-optical evaluation system responded to any instantaneous reflectivity changes occurring on tubing OD surfaces caused by pits, rocked-in metal, and grinder fatspots.
- (2) Dimensional inspection – the OD and wall thickness dimensions of the tubing were verified.

- (3) Ultrasonic examination - a tube standard was utilized with four notches (on the ID and OD surfaces, oriented in both the axial and circumferential directions) in order to calibrate the ultrasonic equipment. Four transducers were utilized to detect structural discontinuities in the tubing.
- (4) Tube Cutting - the tubing was cut to the length specified by the PO.
- (5) Straightness verification - the tubing straightness was verified.
- (6) Final Visual inspection - final visual inspection was performed.
- (7) Tube marking - final identification marking for each unique lot number was electro etched near one end of each tube.
- (8) Boxing for shipment - various spacers were inserted to prevent tube-to-tube fretting of the tubes in transit.

3.5 Testing

During its review of the testing lab practices, the team observed the following testing operations: (a) tensile and elevated tensile testing and (b) burst testing and circumferential elongation measurement. A random sampling of calibration stickers was performed on the following equipment: tensile tester, micrometer, digital calipers, and balance; all were within calibration.

3.5.1 Burst Test

The burst test (used to characterize the biaxial mechanical properties of zircaloy tubes) required that a tube sample be pressurized to failure. To begin the test, the wall thickness of each end of the tube sample was measured at 6 places, 60° apart, and all measurements were recorded. A rod was inserted into the tube to prevent bending of the specimen during deformation. The tube sample was installed on a test fixture and gradually hydraulically pressurized to failure (rupture). After failure, a piece of brass shim stock was wrapped around the tube sample at the rupture location (from one fracture surface circumferentially to the other fracture surface), in accordance with ASTM B353, "Standard Specification for Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service," and Annex A.1, "Recommended Close-End Burst Testing Procedure for Zirconium Alloy Tubing." The shim stock was creased at each fracture surface and the length of the shim stock between the creases was measured using calibrated dial calipers and the measurements were recorded.

These measurements were then entered into a computer program that calculated the ultimate hoop stress, total circumferential elongation (TCE %), and eccentricity. The burst test practice was prescribed in SSM Laboratory Procedure 1300.21, "Burst Testing Zirconium Base Alloy Tubing (Closed and Open End Testing)," Revision 4, dated August 21, 1985. The burst test is performed on a sample of two specimens per tubing lot. The burst test characterized the biaxial mechanical properties of the Zr4 fuel clad tubing.

The team evaluated the burst test practices for tubing lot DHS12, supplied to ABB CENO. The initial burst test TCE for this tubing lot failed on the basis that, in part, of the two specimens tested, one failed to meet TCE minimum requirements ($TCE \geq 12\%$), as specified in ABB CENO Specification No. 00000-PD-301, "Specification for Zircaloy 4 Fuel Rod Cladding Tubes," Revision 4, dated July 17, 1991. The burst hoop strength was acceptable. SSM performed a retest of four tube sample specimens from the lot (twice the original number tested). Of the retest tube sample specimens, one specimen failed the TCE requirements, and, therefore, the entire tube lot failed the TCE specification requirements.

According to the SSM request for approval or review 9491063-002, dated June 15, 1994, before reworking lot DHS12R (R standing for reworked), the tubing contained surface imperfections on the OD of the Zr4 fuel clad tubing that had caused the reduced ductility during burst testing. SSM described the imperfections as shallow axial OD microcracks attributed to parallel platelet microstructure and pilgering conditions. In its approval of the SSM rework procedure, ABB CENO stated that SSM would provide ABB CENO the root cause of the defects and the steps taken by SSM to avoid these defects in the future.

ABB CENO also requested (a) photographs of the visual appearance of the surface defects and (b) a fracture surface examination of low and high ductility burst test specimens to aid in characterization of the defects prior to its approval of the rework procedure.

After receiving approval from ABB CENO to rework the tubing lot, SSM reworked the tubing to remove the surface imperfections and additional tube specimens were tested. The rework consisted of belt-polishing the tube lot to remove approximately 0.0006 inch of the OD surface, considered by SSM to be sufficient to remove the discontinuities which caused the tube lot's failure to meet the TCE specification requirements. All tube samples of lot DHS12R tested met the TCE specification requirements with values that ranged from 15.1 to 62.6% TCE.

Note:

From its previous inspection of ABB CENO (documented in NRC Report No. 99900002/94-01; 99900102/94-01), the team noted that as of October 7, 1994, (a) the SSM root cause analysis requested by ABB CENO had not been performed, (b) ABB CENO had approved the lot of fuel clad tubing for use, and (c) SSM shipped the lot of tubing to ABB CENO without determining the cause of the microcrack condition and submitting to ABB CENO a root cause analysis report. The team had identified the ABB CENO evaluation of the failed burst test specimens as an open item and requested that NRC be notified by ABB CENO when it completed its problem analysis of the failed burst test specimens, identified possible causes, evaluated the possible causes, confirmed the true cause, and identified corrective actions to be taken. On October 20, 1994, ABB CENO gave the team a root cause analysis report that, according to ABB CENO, was the final report of the SSM analysis.

However, on December 7, 1994, the team was advised by SSM that its evaluation of the failed tube specimens was not complete and that it would be submitting to ABB CENO a final problem analysis with assigned root cause. Some confusion, therefore, appeared to exist regarding the status of the analysis report reviewed by the team on October 20, 1994.

The team was told by SSM that the identity of the failed test specimens was lost after the testing process and, therefore, the failed tube specimens were not available for further analysis. The tube specimens were originally identified with "indelible" ink. However, the oil used in the burst test process dissolved the ink and washed away the tube specimen identification. Because the test operator only performs the test and records the results without knowing whether tube specimens met the TCE requirements, after the test was completed and the test results recorded, all tube specimens were discarded into a bucket with no apparent method of retrieval (because identification on each tube specimen was inadvertently washed off) if required.

Without the failed tube specimens to evaluate, SSM was not able to
(a) photograph the surface defects and perform a visual examination or
(b) examine the fracture surfaces of the failed tube specimens or the low and high ductility tube specimens. Because the SSM test lab practices for conducting the burst test permitted lost identification of the tube specimens, the SSM final problem analysis was not able to evaluate the failed tube specimens.

The team concluded that the assigned root cause of the failed tube specimens was not conclusive. The SSM failure to control the identification of failed tube specimens in a manner that ensured the ability to perform further analysis and assign true root cause for failed tube specimens was an activity that affected quality and the adequacy of the corrective actions taken to preclude reduced ductility in other tube specimens. The team reviewed SSM Laboratory Procedure 1300.21 and determined that the procedure also failed to address the continued identification of tube specimens necessary to provide for post-testing failure analysis. By not establishing measures in Laboratory Procedure 1300.21 to control the identification of failed tube specimens that may affect quality and by not utilizing test practices to identify and retain failed tube specimens, where further analysis may affect quality, SSM failed to comply with Section QAM-2-5, "Instructions, Procedures, and Drawings," of the SSM QAM that required, in part, that activities and operations affecting quality of tubing shall be controlled through the use of approved instructions and procedures and was, therefore, a nonconformance. As a result, Nonconformance 94-01-01 was identified during this part of the inspection.

3.6 Corrective Actions

During its inspection of ABB CENO (documented in NRC Report No. 99900002/94-01; 99900102/94-01), the team reviewed two concerns that ABB CENO had developed regarding surface indications on fuel clad tubing supplied by SSM. During this inspection of SSM, the team evaluated the SSM problem analysis of the issues identified on ABB CENO Corrective Action Requests (CARs) for the concerns described below.

3.6.1 CAR V-20-94

Discussions with ABB CENO indicated that since July 1994, ABB CENO had experienced problems with SSM tubing involving blue, pink, black, gray, and brown stains and white deposits on various surfaces. There was evidence that the problems were being addressed as required by the ABB CENO QA programs, but the corrective action had not been successful in preventing the recurrence of new but similar indications of nonconforming tubing for fuel cladding. On November 8, 1994, ABB CENO issued SSM CAR V-20-94 that identified cleanliness and surface condition problems on the following SSM lots of tubing:

- DHU51
- DHU21
- DHT82
- DHT14
- DHW42
- DHY62

Although the SSM problem analysis and corrective action was not complete as of December 7, 1994, during this inspection the NRC team reviewed the SSM evaluation and corrective actions taken for the "whitish" discolorations observed on the tube OD surface. The results of the SSM evaluation were documented in SSM Technical Report No. 94-009, "Evaluation of Paper Divider Trays," dated September 26, 1994. For its evaluation of the "whitish" deposits, SSM utilized both scanning electron microscopy (SEM) and energy dispersive spectroscopy (EDS).

EDS is an analytical technique, capable of performing elemental analysis of microvolumes, typically on the order of a few cubic microns in bulk samples and considerably less in thinner sections. Analysis of x-rays emitted from a sample is accomplished by crystal spectrometers which use energy dispersive techniques permitting analysis by discriminating among x-ray energies. The feature of electron beam microanalysis that best describes this technique is its mass sensitivity. For example, it is often possible to detect less than 10^{-16} grams of an element present in a specific microvolume of a sample. The minimum detectable quantity of a given element or its detectability limit varies with many factors, and in most cases is less than 10^{-16} grams/microvolume.

SSM determined that the glossy coating on the back of its vendor supplied paper divider trays (*Rondo*TM trays), used in shipping fuel clad tubing, was responsible for the "whitish" deposits found on the OD surfaces of some tubes. SSM corrective action was to extend the paper separating the layers of tubing beyond the tube ends. SSM also altered their purchase specification to specify a type of paper backing which will not produce stains.

For the various color stains that were observed by ABB CENO, the SSM preliminary evaluation of the problem indicated that, in part, the illumination in the packaging and final visual inspection area would be increased.

Additionally, the ink identification marking practices and the cleaning practices would be revised, as they are prescribed respectively in SSM procedure QA-GA-21, "Additional Lot Identification by Color Coding of Tubes," Revision 5, dated June 8, 1994, and Process Specification No. 2449, "Belt Grinding Zircaloy - Niederberger Grinders," Revision 17, dated April 17, 1994.

On the basis of its evaluation of the SSM problem analysis and corrective actions, the team concluded that the SSM evaluation appeared to be thorough, competent, and conclusive as to the root cause of the discoloration.

3.6.2 CAR V-21-94

During the inspection of ABB CENO referenced in Section 3.5.1 above, the team observed apparent indications of pitting on the OD surface of two Zr4 fuel clad tubes from SSM lot DHU51. As of November 2, 1994, ABB CENO had neither completed its problem analysis of the surface indications to identify possible causes, evaluate the possible causes, and confirm the true cause, nor had it identified corrective actions to be taken. The team identified this to ABB CENO as an open item and requested that NRC be notified by ABB CENO when the corrective action had been completed.

On November 8, 1994, ABB CENO issued SSM CAR V-21-94 that identified two tubes from lot DHU51 as exhibiting a spiral/helix condition on the OD surface of the tube; ABB CENO added that the marks were indicative of pitting or areas that did not clean up at belt grinding.

Although the SSM problem analysis and corrective action was not complete as of December 7, 1994, during this inspection the team reviewed the SSM evaluation and corrective actions taken for the spiral pitting observed on the tube OD surface. The team examined the spiral pitted tube sections with a 40x magnification and the Manager, Quality Assurance put forth an analysis that the pitting was actually fretting that occurred while the tubes were in transit from SSM to ABB CENO.

Fretting occurs when two metal surfaces move against one another with an oscillating motion, and forms abrasive particles. These metallic particles react with the air, are compacted, and eventually are impressed into the metal surfaces.

The team also witnessed the spiral pitted tube section processed through the automated visual inspection process to determine whether the automated electro-optical evaluation system would detect the spiral pitting. The test proved to the team's satisfaction that the automatic visual inspection process, performed in accordance with SSM procedure ETA 641, "Post Grind OD Surface Inspection," Revision 0, dated August 30, 1991, would detect the spiral pitted surface condition and reject the tubing. The proposed SSM corrective actions included revising Quality Assurance Procedure NDT-V-57, "Final Visual and Packaging," Revision 15, dated December 8, 1993, to enhance the final packaging inspections and the possibility of adding packing material in certain instances.

On the basis of its evaluation of the SSM problem analysis and corrective actions, the team concluded that the SSM evaluation appeared to be thorough, competent, and conclusive as to the root cause of the fretting.

APPENDIX

PERSONS CONTACTED

The NRC staff participating in the inspection of SSM; SSM's 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.

Sandvik Special Metals Corporation:

- † Banko, W.W. Vice President, Operations & Plant Manager
- † Bowles, K.C. Manager, Quality Assurance
- Bradley, Dr. E.R. Senior Development Metallurgist
- † Brewer, T.M. Quality Assurance Engineer
- Erickson, P. Technician (Burst Test)
- † Galbraith, K.P. President & CEO
- Hollibaugh, C.L. Internal Auditor
- Hoxie, D. Technician (Tensile Test)
- Johnson, R.T. Plant Superintendent
- Knapp, G.R. Coordinator, Total Quality
- King, C. Manager, QA and Lab Services
- Luebke, J.T. Manager, Facilities Engineering
- † Martenson, Dr. A.J. Vice President, Research & Engineering

U.S. Nuclear Regulatory Commission:

- † Brewer, D.H. Metallurgical Engineer, VIS/TSIB
- † Czajkowski, C.J. Metallurgical Engineer, Brookhaven
National Laboratory
- † Matthews, S.M. Quality Assurance Specialist, VIS/TSIB

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

January 31, 1995

Mr. Albert E. Riesen, President
Teledyne Wah Chang
P.O. Box 460
Albany, Oregon 97321-0460

SUBJECT: NRC INSPECTION NO. 99901229/94-01

Dear Mr. Riesen:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Teledyne Wah Chang (TWC), Albany, Oregon, conducted November 28 through December 2, 1994. The NRC inspection team, led by Steven M. Matthews and comprising other inspectors named in the report, conducted a performance-based evaluation of TWC management, staff, and quality programs and the implementation of those programs related to the fabrication of zirconium alloys in various product forms. The inspection was conducted to provide a basis for confidence that TWC zirconium alloy 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) conducted interviews, (c) held discussions, (d) listened to presentations, and (e) made various observations. On the basis of this inspection, the NRC team determined that the implementation of the TWC quality assurance program, documented in the TWC Quality Manual, Revision 1, dated March 8, 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.

The NRC team identified the following strengths in TWC's production operations: (a) the products were custom fabricated in accordance with the processes and specifications established by the customer and all aspects of the production process were documented, from order entry through the execution of each process step by operators and quality control personnel; (b) training and requalification of personnel was in order and process operating personnel were considered competent, in most instances having long experience with TWC; and (c) technical problem analysis and modifications to equipment and processes to improve quality were performed by competent engineering personnel.

A. Riesen

- 2 -

For the nonconformances cited in the NRC report of its previous inspection of TWC (NRC Report No. 99901229/91-01), the NRC determined that TWC corrective actions were adequate and had been effectively implemented. The enclosed report documents closure of those nonconformances.

No response to this letter or its enclosure is required.

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,

A handwritten signature in dark ink, appearing to read "Robert M. Gallo", with a stylized flourish at the end.

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

Docket No.: 99901229

Enclosure: Report No. 99901229/94-01

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

REPORT NO.: 99901229/94-01

ORGANIZATION: Teledyne Wah Chang (TWC)
1600 North East Old Salem Road
P.O. Box 460
Albany, Oregon 97321-0460

ORGANIZATIONAL
CONTACT: Robert C. Bartcher
Director, Quality Assurance

NUCLEAR INDUSTRY
ACTIVITY: TWC produces zirconium alloys for use in fuel
assemblies by the nuclear power industry.

INSPECTION DATES: November 28 through December 2, 1994

LEAD INSPECTOR: Steven M. Matthews 1-25-95
Steven M. Matthews Date
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

OTHER INSPECTORS: David H. Brewer, VIS/TSIB
John R. Weeks, Brookhaven National Laboratory

REVIEWED BY: Gregory C. Cwalina 1/25/95
Gregory C. Cwalina, Chief, VIS/TSIB Date

APPROVED BY: Robert M. Gallo 1/31/95
Robert M. Gallo, Chief, TSIB Date

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspection team evaluated TWC management, staff, and quality programs and the implementation of those programs related to the fabrication of zirconium alloys in various product forms. The inspection was conducted to provide a basis for confidence that TWC zirconium alloy products supplied to the U.S. nuclear industry in fuel assemblies would perform their safety functions.

The inspection basis consisted of the following:

- TWC Quality Manual (QM), Revision 1, dated March 8, 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, "Notification of Failure to Comply or Existence of a Defect," of 10 CFR

No violations or nonconformances were identified during this inspection.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

During an inspection of TWC conducted on October 7 through 10, 1991, the NRC inspection team determined that certain TWC activities were not conducted in accordance with NRC requirements contractually imposed on TWC by purchase orders (POs) from nuclear fuel manufacturers. The following nonconformances, issued by the staff on December 4, 1991, and the associated TWC corrective action, were evaluated by the NRC team during this inspection. Based on the evaluation of TWC corrective actions, the NRC team determine that each of the nonconformances was closed.

2.1 Nonconformance 91-01-01 (CLOSED)

Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, the TWC Quality Control Manual (QCM), Revision 1, dated February 15, 1991, had not stated that activities affecting quality were to be prescribed by documented procedures and performed in accordance with those procedures.

To evaluate the TWC corrective action that addressed this nonconformance, the NRC team reviewed the TWC implementation of (a) Section F, "Instructions, Procedures, and Drawings," of the TWC Quality Manual, Revision 1, dated March 8, 1994; (b) Quality Control Instructions (QCI), document QCI-F-1, "Work Instructions," Revision 6, dated June 6, 1994; and (c) QCI-F-4, "Process Flow Outlines," Revision 1, dated August 29, 1994. The NRC team determined that TWC corrective action was adequate to ensure that activity affecting quality was documented and performed in accordance with those procedures.

2.2 Nonconformance 91-01-02 (CLOSED)

Contrary to Criterion XV, "Nonconforming Materials, Parts, or Components," of Appendix B to 10 CFR Part 50, and Section P.3 of the TWC QCM, dated October 23, 1989, TWC had not issued a Product Condition Information Request (PCIR) to document deviations in samples from Zircaloy-2 heat number 22881 following nodular corrosion testing.

The TWC QCM in effect during the October 1991 inspection had been replaced. To evaluate the TWC corrective action that addressed this nonconformance, the NRC team reviewed the TWC implementation of QCI-P-2, "Production Condition Information Request," Revision 9, dated November 24, 1993. The NRC team determined that TWC corrective action was adequate to ensure that a PCIR was issued for a material deviation.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

Wah Chang (a Chinese phrase meaning "great development") was founded in New York in 1916 for international ore trading, engineering and tungsten extraction. In 1956, to operate a government zirconium plant, Wah Chang expanded to Albany, Oregon. Wah Chang was purchased in 1967 by Teledyne, Inc. Zircaloy-2 (Zr2) and Zircaloy-4 (Zr4) (collectively, zircaloy), used in U.S. nuclear fuel assemblies, were produced by TWC in strip for channel sections and grid spacers, rod for fuel rod end caps, and extruded tube hollows for fuel clad tubing. To accommodate the various needs among customers for these products, TWC custom fabricates them to meet customer negotiated requirements.

3.2 Entrance and Exit Meetings

During the entrance meeting in Albany, Oregon, on November 28, 1994, the NRC team met with members of TWC management and staff, discussed the scope of the inspection, and established contact persons for the NRC team within the management and staff of the TWC organization.

During the inspection, the NRC team conducted a performance-based inspection of TWC 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, (d) listened to presentations, and (e) made various observations.

Inspection participants and contacts are listed in the Appendix.

During the exit meeting on December 2, 1994, the NRC team summarized the inspection findings with TWC management and staff.

3.3 Order Entry

During this inspection the NRC team evaluated the following documentation:

- (1) PO 100 751, dated November 17, 1993, from Sandvik Special Metals Corporation (SSM) for Zr4, Grade R60804, 3.20 inch outside diameter (OD) x 1.62 inch inside diameter (ID) x 0.790 inch wall extrusions (tube shells), produced in accordance with SSM's specification SSM 1021, Revision 10, and TWC Quality Control Plan and Process Flow Outline (PFO), PFO-271, "Zircaloy 2 and 4 Extruded Tube Shells (3.2" OD x 0.79" W)," Revision 1, dated December 3, 1993.
- (2) PO 9392494-D5602, dated September 23, 1993, from ABB Combustion Engineering Nuclear Operations (ABB CENO) for Zr4 centerless ground bar material produced in accordance with ABB CENO's specification 00000-PD-304, "Zircaloy 4 Bar Material for Fuel and Burnable Poison Rod End Caps," Revision 3, dated July 17, 1991, and TWC PFO-251, "Zircaloy Extruded Plug Stock Cold Worked, Annealed, and Centerless Ground," Revision 1, dated October 22, 1992.
- (3) PO R/073263, dated April 7, 1994, from Siemens Power Corporation - Nuclear Division (SPC-ND) for Zr4 strip, 0.17 inch thick x 9.5 inch wide, produced in accordance with SPC-ND's specification EMF-S35081, "Formable Zircaloy 4 Strip or Sheet for PWR Applications," Revision 4, dated November 14, 1991, and TWC PFO-147, "Zircaloy Spacer Strip-PWR," Revision 6, dated October 26, 1989.
- (4) PO 027266, dated August 8, 1994, from Carpenter Technology Corporation, Special Products Division, for Zr4, Grade R60804, triple melted, alpha annealed, 0.083-0.086 inch thick x 20.572-20.582 inch wide strip produced in accordance with Carpenter's specification S-70000-1, "Zircaloy Strip for BWR Channels," Revision 11, dated November 30, 1993, and TWC PFO-250, "Zircaloy Channel Strip," Revision 0, dated June 9, 1992.

The NRC team evaluated these packages and determined that TWC customers established the purchase specifications which included sampling and testing to be performed and the requirement to approve all revisions to applicable PFOs. The NRC team also determined that TWC material certifications, for the items reviewed from the POs listed above, were detailed in their description of the following items:

- heat number
- ingot chemistry analysis
- ingot impurities analysis
- product chemistry analysis
- ingot ultrasonic test
- ingot hardness
- grain size
- tests, as specified by the customer
- cumulative anneal parameter range
- metallographic tests

3.4 Zirconium Alloy Manufacturing Process

TWC produced zirconium, niobium, hafnium, titanium, and various refractory metals and alloys. Zirconium metal and alloys were produced using zircon sand (zirconium orthosilicate) from Australia, the major supplier.

The NRC team observed various stages of the TWC production process which is described here in some detail through completion of ingot production.

After separating zircon from several other naturally occurring minerals, the zircon was mixed and ground with an appropriate amount of carbon in the form of coke to produce an intimate mixture having a specified particle size range.

The zircon sand/coke mixture was reacted with chlorine at about 1200°C (2200°F) to produce zirconium tetrachloride and silicon tetrachloride. Zirconium tetrachloride and silicon tetrachloride were separated by partial condensation of the chlorinator off-gas.

Zirconium tetrachloride, which contained approximately two percent hafnium tetrachloride, was dissolved in water and filtered to remove suspended solids (fine sand and carbon). The zirconium and hafnium were separated, for although they are similar chemically, they are quite different in their nuclear properties. Hafnium absorbs large quantities of thermal neutrons while zirconium does not. Separation was accomplished by a series of resin columns.

The aqueous zirconium tetrachloride was precipitated with sulfuric acid to form hydrous zirconium sulfate and filtered. The filter cake was treated with ammonium hydroxide to form hydrous zirconium oxide and filtered and rinsed again to remove sulfate ions. The hydrous zirconium oxide was calcined at approximately 1000°C (1832°F) to produce zirconium oxide.

Zirconium oxide and the appropriate amount of carbon in the form of coke were intimately blended and reacted with chlorine to produce zirconium tetrachloride. Aluminum, iron, and titanium and others were removed as impurities. Zirconium tetrachloride was reacted with metallic magnesium at elevated temperature to produce zirconium metal and magnesium chloride. The zirconium metal was heated to distill excess magnesium and produce a zirconium regulus, a porous sponge-like material.

The zirconium sponge regulus was broken into chunks suitable for crushing to pass a three-quarter inch screen. Fines were removed. To achieve ingot chemistry requirements, sponge from various runs was blended together. Blended zirconium sponge and alloying elements were combined and pressed into 14" diameter by 5" thick compacts.

Based on chemical analysis, computer selected combinations of sponge compacts and recycled billets were electron beam welded to form electrodes. The electrodes were melted into ingots by the consumable vacuum arc melting process. Multiple melts decreased impurities and improved homogeneity. TWC technical staff stated that, with a few minor exceptions, all zirconium alloys

supplied to the commercial nuclear power industry were triple arc melted. Ingots were given a heat number by which they were identified through subsequent processing.

Ingots were ultrasonically tested to determine the extent of pipe, hardness tested, and sampled for chemical analysis at multiple locations along the length to assure homogeneity and compliance with customer specifications.

The NRC team reviewed a qualification report researched and written during the early period of plant history demonstrating that the triple arc melting process produces ingots that are uniform in chemical composition throughout within a few percent of the mean value. On this basis, TWC considers chemical samples from various locations on one side of the ingot as representative of the whole.

Ingots meeting customer criteria were identified as suitable for assignment to the customer's requirements. The circumference of the ingots was machined and ingots were fabricated into wrought product by forging, hot rolling, cold rolling, or extrusion. The choice of working methods and heat treatment sequence depends on the alloy, final product form, and customer requirements.

The NRC team observed processes in which (a) sponge was crushed, sorted, and blended in preparation for assembling compacts, (b) sponge compacts and recycled billets were electron beam welded into an electrode suitable for arc melting, and (c) vacuum arc melting was performed, producing an ingot.

TWC demonstrated to the NRC team its ability to control various alloying elements within a relatively narrow band of the range specified by ASTM B 350-93, "Standard Specification for Zirconium and Zirconium Alloy Ingots for Nuclear Application." With this capability and the ability to perform a range of thermal and mechanical treatments, TWC customers specify the combination of chemical composition, thermal, and mechanical treatments that will produce material properties that best suit their needs. TWC is, therefore, a custom fabricator, negotiating product and process details with each customer.

3.5 Beta Quenching

In discussion with the NRC team the TWC technical staff stated that beta quenching was the single most significant metallurgical factor relating to all product forms in both Zr2 and Zr4. The reason for this significance was the affect beta quenching has on improving the corrosion resistance of the materials.

3.5.1 Fundamentals

Beta quenching is a thermal treatment in which the workpiece is heated above about 1000°C (1832°F), held in the beta phase temperature range, and rapidly quenched in water. At the temperature of about 1000°C, the crystalline structure changes from hexagonal close packed to body centered cubic. This

change in crystalline structure is referred to as the alpha to beta transformation. Above the transformation temperature, alloying elements dissolve from particles of intermetallic compounds and go into solid solution more rapidly.

The effect of this thermal treatment is to more uniformly disperse alloying elements throughout the microstructure. TWC personnel stated that evidence of the effectiveness of the beta quench process had been demonstrated microscopically by differences observed in the size and distribution of intermetallic compound particles dispersed in the solid solution matrix.

The uniformity of the temperature to which the workpiece is heated, typically between 1016-1200°C (1861-2192°F), is not critical provided it is above the transformation temperature.

The upper temperature limit depends on the time at temperature. If acceptable limits of time at temperature were exceeded, too much of the alloying elements would go into solid solution. This would result in insufficient sites of intermetallic compounds available for nucleation and growth during quenching. Although in the alpha phase after quenching, precipitation would occur at prior beta grain boundaries, resulting in areas of parallel platelet microstructure which TWC personnel stated would be readily observable in metallographic examination. Parallel platelet microstructure is not as ductile as solid solution with islands of intermetallic compounds. A microstructure containing areas of parallel platelets is subject to cracking during subsequent hot and cold working and is to be avoided.

TWC stated that all zircaloy they produced received an "early" beta quench and, in addition, some Zr2 received a "late" beta quench performed by customers believing that enhanced corrosion resistance would result.

3.5.2 Corrosion Resistance

Research^{1 2} referenced by TWC personnel and reviewed by the NRC team, showed how different heat treatments must be performed on Zr2 and Zr4 to obtain optimum corrosion resistance for each alloy in light water reactor applications.

¹ Garzarolli, F., Stehle, H., Steinberg, E., and Weidinger, H., "Progress in the Knowledge of Nodular Corrosion," *Zirconium in the Nuclear Industry: Seventh International Symposium*, ASTM STP 939, R. B. Adamson and L. F. P. VanSwam, Eds., American Society for Testing and Materials, Philadelphia, 1987, pp. 417-430.

² Garzarolli, F., Steinberg, E., and Weidinger, H. G., "Microstructure and Corrosion Studies for Optimized PWR and BWR Zircaloy Cladding," *Zirconium in the Nuclear Industry: Eighth International Symposium*, ASTM STP 1023, L. F. P. VanSwam and C. M. Euker, Eds., American Society for Testing and Materials, Philadelphia, 1989, pp. 202-212.

At the ingot stage and during subsequent open die forging, alloying elements segregate on a microscopic basis by growing intermetallic compound precipitates. Beta quenching of slabs, logs, or extrusion billets homogenizes the chemistry by dispersing the alloying elements into solid solution. The chemical homogenizing resulting from beta quenching is essential to the enhanced corrosion resistance of Zr2 and Zr4 as described in the references.

Zircaloy-2, generally used in boiling-water reactors (BWRs), is more subject to nodular corrosion. The best corrosion protection for this alloy is produced by an "early" beta quench performed in the billet reduction process. Subsequent thermal treatments are performed in a way to control heat input below a specified value, as cited in the referenced research, to ensure optimum corrosion resistance.

TWC stated that some of their customers that manufacture Zr2 fuel clad tubing perform a "late" beta quench prior to the final mechanical forming processes. Thermal and mechanical processes performed on the material after the "early" beta quench promote additional precipitation of alloying elements from solid solution. The metallurgical effect of the "late" beta quench is to return the optimum amount of the alloying elements to solid solution. Higher concentrations of alloying elements in solid solution for Zr2 have demonstrated enhanced resistance to nodular corrosion in BWR service.

In contrast, Zr4, generally used in pressurized-water reactors (PWRs), is more subject to uniform corrosion. The best corrosion protection for this alloy is produced by an "early" beta quench performed in the billet reduction process. Subsequent thermal treatments are performed to expose the material to a relatively higher range of accumulated heat input than for Zr2, as cited in the referenced research. The metallurgical effect of subsequent mechanical and thermal treatments is to precipitate more intermetallic compounds and grow them larger, leaving less alloying elements in solid solution. Lower concentration of alloying elements in solid solution for Zr4 has demonstrated enhanced resistance to uniform corrosion in PWR service. No "late" beta quench is performed which would increase the alloy content of the solid solution.

3.5.3 Furnaces

TWC used several electrical resistance and induction furnaces to heat zircaloy for beta quenching. Electrical resistance furnaces were used to heat material that had been open die forged to slab (for plate, sheet, and strip products), and material rotary forged to log (for bar, rod, wire, and about 95% of the material for fuel clad tubing or other tubing applications). Induction furnaces were used to heat extrusion billets³ only; about 5% of the material for fuel clad tubing or other tubing applications.

³ An extrusion billet is very much like a log that has been reduced to a round cross section by various forging operations but with a hole trepanned down the centerline.

The NRC team noted that a variety of induction furnaces were available for installation on the beta quench tower. Each induction furnace was a different size depending on the size of the billet it was to heat. Each furnace had a ceramic sleeve mounted between the induction coil and the workpiece space. The sleeve prevented the workpiece from contacting the coil and maintained it in the approximate center of the coil.

An induction furnace heats a workpiece more rapidly than an electrical resistance furnace. For this reason the induction furnace may be considered more desirable for heating Zr2 for nodular corrosion resistance considerations. Induction heating produces heat at or near the surface of the workpiece while power is flowing in the induction coil. When power is turned off to the coil, heat induced when power was on conducts to the interior of the workpiece. TWC personnel selected furnace operating parameters to produce the desired heating characteristics. Furnace operating parameters were documented in the TWC profile report evaluated by the NRC team.

3.5.4 Temperature Measurement and Control

The NRC team observed a variety of temperature measuring and indicating methods used by TWC; thermocouples, optical pyrometers, Tempilsticks, and temperature indicating crayons. However, only thermocouples with calibration traceable to the National Institute for Standards Technology (NIST) were observed being used to control beta quenching processes in both the electrical resistance and induction furnaces.

The NRC team observed a single color optical pyrometer in the area where induction heating was occurring but it was not in use. TWC personnel stated that it was accurate only under tightly controlled circumstances. The emissivity of a workpiece surface influenced the temperature reading produced and since workpiece emissivity varies considerably depending on surface finish and oxidation thickness, TWC used it only where temperature measurement accuracy was not considered critical.

Calibration of the thermocouple temperature measuring and controlling equipment had been performed periodically as required by MIL-STD-45662A, "Calibration Systems Requirements," dated August 1, 1988. A similar method was used to calibrate the thermocouples and controller used in the induction furnaces. The NRC team observed that all thermocouple measuring and controlling equipment used in both electrical resistance and induction furnaces were current in calibration.

3.5.4.1 Electrical Resistance Furnace Temperature Surveys

Furnace temperature surveys are performed to determine the working zone or volume within the furnace that maintains the required temperature control. Furnace surveys are performed by placing a number of thermocouples throughout the furnace to collect temperature information as the furnace operates. The temperature information is evaluated to determine the working zone of the furnace within which acceptable temperature limits are maintained. Subsequent heat treatment activity is required to be performed within the working zone.

The NRC team reviewed the furnace surveys for electrical resistance furnaces used for beta quenching. Furnace surveys had been performed on the furnaces periodically as required by MIL-STD-45662A, "Calibration Systems Requirements," dated August 1, 1988. In each case the NRC team determined that the acceptable working zone was being used and the surveys were current.

3.5.4.2 Induction Furnace Temperature Profiles

For an induction furnace a temperature profile performed using a profile billet is comparable to performing a temperature survey on an electrical resistance furnace.

TWC personnel stated that a temperature profile was performed on each furnace before and after beta quenching each lot of material. This process verified consistent furnace performance during beta quenching of the lot.

The NRC team reviewed documentation of a temperature profile of the Mark II Induction Furnace performed September 27, 1994. The temperature profile had been performed using a "profile" billet of Zr2 having dimensions like the billets that were to be heated. The "profile" billet was instrumented with six thermocouples embedded; two, $\frac{1}{2}$ inch from the top of the billet, one of these $\frac{1}{2}$ inch in from the OD and the other $\frac{1}{2}$ inch out from the ID. The other four were similarly placed; two, $\frac{1}{2}$ inch from the bottom of the billet and two midway along the axis. The NRC team observed that the calibration of each thermocouple was verified and documented by TWC personnel. Temperature profile documentation reviewed by the NRC team was within specified limits with the exception of one control thermocouple which was noted to be out of proper position. TWC personnel stated that they considered the control thermocouple out of position as insignificant since the profile thermocouples had registered within acceptable limits. The NRC team reviewed the discrepant condition and agreed that for temperature profiling purposes, it was not significant.

3.5.4.3 Electrical Resistance Temperature Control

Just as induction furnace profiles compare and contrast to electrical resistance furnace surveys, so control thermocouples have similarities and differences between the two types of furnaces. While both types of furnaces use the same type of control thermocouples and their calibration is checked the same way, the way they are used is different. In the electrical resistance furnace the control thermocouple is permanently mounted at a location within the working zone.

3.5.4.4 Induction Furnace Temperature Control

In the induction furnace three process control thermocouples were placed in the ID of each billet quenched; two, $\frac{1}{2}$ inch from each end and one midway. Billet soak temperature was specified as 1050 to 1200°C (1922 to 2192°F) and soak time as 2 to 5 minutes. The wide range on soak temperature was specified to allow for variations in heating that can occur in the induction heating process. TWC personnel stated that standard practice had not produced such a large range of variation.

The temperature data from the September 27, 1994, profile reviewed by the NRC team showed the maximum temperature recorded by the six embedded thermocouples ranged from 1096 to 1167°C (2005 to 2133°F). Temperature variation occurred throughout the billet as the induction furnace cycled on and off. This temperature variation was considered normal by TWC. The NRC team reviewed the temperature variation and found it within the limits discussed in Section 3.5.1 of this report.

3.5.5 Quenching Operations

According to TWC, selection of the beta quenching process used on a customer's product was made by that customer. Since each process produces various advantages and disadvantages, the selection depends on customer requirements for its product.

TWC strived for a quenching rate of 20-40°C (68-104°F)/second. This parameter was neither specified in controlling documentation nor measured. To accomplish this quenching rate however, the following conditions were specified in a number of specifications for quenching from electrical resistance and induction furnaces.

- Water temperature at the beginning of quenching less than 37°C (99°F)
- The volume of quenching water greater than twenty times the volume of the metal being quenched
- Except for the pit tank used to quench slabs, recirculating capability in all quenching tanks

All beta quenching operations observed by the NRC team complied with these requirements. All thermocouples indicating the temperature of water in the quench tanks were observed to be in calibration. Quenching water was taken from a nearby river and contained no chlorine. Water from all quench tanks was recirculated through a cooling tower.

The NRC team observed beta quenching operations at various times during the inspection involving the following materials.

- (1) Material from Zr2 heat number 236861Q, open die forged to 4 inch thick slab, was electrical resistance furnace heated for Carpenter Technology Corporation, Special Products Division, to produce channel strip for use in a German nuclear reactor. The specified soaking time at the 1052°C (1926°F) set temperature was 20 to 60 minutes; soaking time was observed to be 35 minutes. Quenching was performed in the pit tank located in the floor immediately in front of the furnace. A motor lift with a pincer arm pulled the slab from the furnace and dropped it into the quench tank. The specified maximum transfer time (from opening of the furnace door to immersion in the water) was 25 seconds; actual transfer time as recorded by the operator's stopwatch was 16.9 seconds.

- (2) Material from Zr4 heat number 236830Q, extruded to 3.2 inch diameter bar, was electrical resistance furnace heated for SSM to produce fuel clad tubing. The specified maximum soaking time at the 1052°C (1926°F) set temperature was 60 minutes; soaking time was observed to be 40 minutes. Quenching was performed in the recirculating quench tank located approximately 12 meters (40 feet) from the furnace. A motor fork lift carried two logs from the car bottom furnace to the quench tank for immersion. The logs were retained on the fork of the lift during immersion. The specified maximum transfer time was 60 seconds; actual transfer time as recorded by the operator's stopwatch was 39.8 seconds.
- (3) Material from Zr2 heat number 236746Q, rotary forged and trepanned to an extrusion billet approximately 6-7/8 inch OD X 2.4 inch ID X 18-1/4 inch long, was heated in the Mark II Induction Furnace Quench Tower for Kobe Steel to produce the outer shell of barrier coextrusion⁴. Specified soaking time in the 1050 to 1200°C (1922 to 2192°F) temperature range was 2 to 5 minutes; soaking time was observed to be approximately 3 minutes 15 seconds. Quenching was performed by lowering the support platform and sliding it from under the billet, allowing the billet to fall into the quench tank. Transfer time was observed to be less than 5 seconds. The quench tank had an alignment cone made of a rod assembly to direct the billet to a position in which forced circulation was provided through the hole in the axis.

TWC did not produce fuel clad tubing but supplies an intermediate product form to customers that do. The following customers required tower quenched product.

- GE Nuclear Energy, Nuclear Fuel and Components Manufacturing (USA), for Zr2 coextrusion outer shell
- Nuklearrohr-Gesellschaft MBH (Germany) for Zr4 tube hollows
- Kobe Steel (Japan) for Zr2 coextrusion outer shell

The NRC team reviewed documentation produced in the process of using the Mark II Induction Furnace Quench Tower for beta quenching 33 billets from heat number 236531, Zr2, on shop order 3773, item 1, reference numbers 34 through 66. Each billet quenched was instrumented with three thermocouples to produce a record of its temperature profile. Each temperature profile was reviewed to determine that the billet had received the required thermal treatment. Two billets failed to receive the proper temperature profile prior to the first quench as shown on the induction furnace controller/recorder. These billets were reheated and requenched and the proper temperature profiles were recorded.

⁴ Barrier coextrusion is a tubular product with a commercially pure zirconium liner placed inside an outer shell of Zr2.

Each billet was allowed a maximum of three attempts to obtain the proper heat treatment and quenching. TWC data reviewed by the NRC team did not show any billet beta quenched three times; TWC stated that in general less than 5% of the billets required beta quenching twice to obtain an acceptable thermal profile.

3.5.6 Operator Qualification

The NRC team observed TWC personnel performing beta quenching operations and reviewed the training records of those persons. The training recorded was considered appropriate and personnel qualifications were current.

3.5.7 Corrosion Resistance Testing

Corrosion resistance testing was performed by TWC according to the customer purchase specification requirements. These requirements generally coincided with the requirements of a specified revision of ASTM G 2, "Standard Practice for Aqueous Corrosion Testing of Samples of Zirconium and Zirconium Alloys."

The sampling plan for selecting corrosion resistance specimens was also specified by the customer purchase specification. A typical sampling plan randomly selected one to three specimens from the tail end of extrusions comprising a lot, typically one third of an ingot, about 2724 kg (6000 lbs). One corrosion resistance test specimen failure typically required two retest specimens. If one of the retest specimens failed the lot was rejected.

3.6 Cracking Analysis

The NRC team reviewed the TWC report dated June 13, 1994, regarding evaluation of cracks observed by SPC-ND in grid spacer components produced by Caran of Paramount, California, from strip supplied by TWC. The report made five major recommendations for changes in material processing, grid spacer forming tooling, and grid spacer design. Changes that could be affected between TWC and SPC-ND were being negotiated. The NRC team considered the quality of the evaluation and recommendations contained in this report as an instance of the strength of the TWC technical staff.

3.7 Laboratory Facilities

The NRC team visited production and research laboratory facilities. TWC had extensive metallography capabilities for polishing, etching, viewing, and photographing metallographic specimens. Numerous examples of award winning metallographic work were exhibited. Heat tinting and color photomicrography were demonstrated to be particularly effective in identifying metallurgical phases, intermetallic compound compositions, and grain texture developed by various forming schedules.

Numerous autoclaves were observed for performing a variety of corrosion resistance tests discussed in Section 3.5.7 of this report. Chemical analysis equipment was present for determining gaseous and metal elements in the alloys produced by TWC. Tensile and bend testing facilities were also observed.

PERSONS CONTACTED

Teledyne Wah Chang:

- U.S. Nuclear Regulatory Commission:**

- A-1 -

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

January 12, 1995

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

SUBJECT: NRC INSPECTION NO. 99900404/94-01

Dear Mr. Liparulo:

This letter addresses the inspections at the Oregon State University (OSU) Advanced Plant Experiment (APEX) Test Facility in Corvallis, Oregon and at the Societa' Informazioni Esperienze Termoidrauliche (SIET) SPES-2 Test Facility in Piacenza, Italy, conducted by Richard P. McIntyre and Uldis Potapovs of the Nuclear Regulatory Commission's (NRC's) Special Inspection Branch, George Thomas and Alan E. Levin of the Reactor Systems Branch, Robert M. Latta of the Quality Assurance and Maintenance Branch, David E. Bessette and Harold H. Scott of the Office of Nuclear Regulatory Research, and David T. Tang of the Standardization Project Directorate. The inspection at OSU APEX was conducted August 30 through September 1, 1994 and the inspection at SPES-2 was conducted October 11 through 14, 1994. The details of the inspection were discussed with management at each test facility and with your staff members present during the inspection and at the exit meetings on September 1, 1994, at OSU and on October 14, 1994, at SPES-2.

The purpose of the inspections was to determine if testing activities performed at the OSU APEX facility and at the SIET SPES-2 facility to support design certification of the Westinghouse AP600 advanced reactor 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. This was implemented at OSU APEX by LTCT-GAH-001, "AP600 Long Term Cooling Test Project Quality Plan" and at SIET SPES-2 by 00006-QQ-92, "Quality Plan Relative to Nuclear Orders."

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, in general, was adequately implementing the AP600 Project quality assurance program at OSU and SIET, with the exception of a few findings in certain areas. Specifically, the team identified Nonconformances with program implementation with respect to (1) the calibration of instrumentation, (2) the accuracy of the facility as-built drawings, and (3) the lack of adequate control of drawing revisions.

Mr. Nicholas J. Liparulo

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January 12, 1995

Also, an Unresolved Item concerning the acceptance of test results by both OSU and SIET that failed to meet established test acceptance criteria, without an evaluation and disposition being included in the test design record file was identified. Westinghouse stated that it, not the testing organization, make the final determination as to what test results are acceptable and documentation of this evaluation and disposition would reside in the official design record file at Westinghouse offices in Monroeville. The NRC will review Westinghouse design record files in Monroeville during a future inspection to determine if Westinghouse appropriately evaluated and dispositioned the test results that did not meet the acceptance criteria.

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,

R. W. Borchardt, Director
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Docket No.: 52-003

Enclosures:

1. Notice of Nonconformance
2. Inspection Report No. 99900404/94-01

cc w/encs: See Next Page

Mr. Nicholas J. Liparulo
Westinghouse Electric Corporation

Docket No.: 52-003
AP600

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NOTICE OF NONCONFORMANCE

Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

Docket No.: 52-003
99900404

Based on the results of Nuclear Regulatory Commission (NRC) inspections conducted on August 30 through September 1, 1994, at Oregon State University's (OSU's) Advanced Plant Experiment (APEX) low pressure integral system test facility and on October 11 through 14, 1994, at SIET's SPES-2 full height, full pressure test facility, related to the AP600 certification activities, it appears that certain activities were not conducted in accordance with NRC requirements as identified below:

- A. Section 7.5.2 "Methods of Acceptance" of Section 7, "Control of Purchased Items and Services" of Westinghouse Electric Corporation Energy System Business Unit and Power Generation Business Unit Quality Assurance Plan, (WCAP-8370, REV. 12A), dated April 1992, which has been accepted by the NRC as implementing the requirements of 10CFR50, Appendix B, states, that when a Certificate of Conformance is used for acceptance of items or services, verification of the supplier's certification process is performed during audits or surveillance, or both, at intervals commensurate with the supplier's past performance.

Contrary to the above, The Industrial Company which performed thermocouple calibration at the OSU APEX Test Facility, subcontracted the calibration of their primary standards to Industrial Instruments, Inc. and accepted their calibration certificates without having performed audits or surveillance of that company. Also, OSU failed to verify the validity of the flowmeter calibration certificates received from Foxboro Company, an unaudited commercial supplier. (94-01-01)

- B. Section 2, "Instructions, Procedures and Drawings" of WCAP-8370 REV. 12A, states that activities affecting the quality of items and services are accomplished in accordance with documented instructions, procedures or drawings that include appropriate quantitative and qualitative means of verifying quality.

Procedure DP-5.0, Rev. 3, of Westinghouse Electric Corporation Nuclear and Advanced Technology Division Quality Assurance Program (WCAP-9565) states that instructions, procedures and drawings shall include or reference the appropriate quantitative or qualitative acceptance criteria for determining that prescribed activities have been satisfactorily accomplished. Procedure DP-5.0 also identifies organizational responsibilities for the preparation, review and approval of instructions and procedures.

Enclosure 1

Contrary to the above:

1. No procedures or instructions were available or utilized to identify methods to be used or the accuracy and the acceptance criteria for determining the as-built elevations and dimensions of the Oregon State University (OSU) APEX Test Facility.
2. The Industrial Company (TIC) Calibration Procedure TIC C.P. #19, which was used to calibrate thermocouples for the long term cooling tests at the OSU facility, did not identify who originated, reviewed, or approved this document.
3. No instructions or procedures were available at the SIET SPES-2 Test Facility to verify critical dimensions or configuration of commercial manufacturing drawings before accepting these drawings as representing the as-built design condition and placing them under the SIET quality assurance system.
4. The procedures for determining system and component elevations and arrangement at the SPES-2 test facility did not prescribe the required accuracy or include any acceptance criteria for these measurements. (94-01-02)

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 Technical Support, 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 12th day of January, 1995

ORGANIZATION: Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

REPORT NO.: 99900404/94-01

CORRESPONDENCE ADDRESS: Mr. Nicholas J. Liparulo
Nuclear Safety and Regulatory Activities
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P.O. Box 355
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ORGANIZATIONAL CONTACT: John Butler, Principal Engineer
Advanced Plant Safety and Licensing
(412) 374-5268

NUCLEAR INDUSTRY ACTIVITY: Nuclear steam supply system components and services

INSPECTIONS CONDUCTED: August 30 through September 1, 1994 at OSU APEX
October 11 through 14, 1994 at SIET SPES-2

TEAM LEADER: Richard P. McIntyre 1/9/95
Richard P. McIntyre Date
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

OTHER INSPECTORS: Uldis Potapovs, TSIB
Robert M. Latta, TQMB
Alan E. Levin, SRXB (SIET only)
George Thomas, SRXB (OSU only)
David E. Bessette RPSB/RES
Harold H. Scott, RPSB/RES (OSU only)
David T. Tang, PDST (SIET only)

REVIEWED: Gregory E. Cwalina 1/10/95
Gregory E. Cwalina, Section Chief, VIS Date

APPROVED: Robert M. Gallo 1/10/95
Robert M. Gallo, Chief, TSIB 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, testing activities conducted at the Oregon State University (OSU) Advanced Plant Experiment (APEX) test facility in Corvallis, Oregon, and the Societa' Informazioni Esperienze Termoidrauliche (SIET) SPES-2 test facility in Piacenza, 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

1 INSPECTION SUMMARY

- Nonconformance 99900404/94-01-01 was identified and is discussed in Section 3.5.1. of this report
- Nonconformance 99900404/94-01-02 was identified and is discussed in Sections 3.4.1.4, 3.4.2.3, and 3.4.2.4 of this report.
- Unresolved Item 99900404/94-01-03 was identified and is discussed in Sections 3.4.1 and 3.6.2.2 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous inspections have been conducted in this area.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Quality Assurance Program

As a supplier of items and services to the nuclear industry, Westinghouse Electric Corporation (W) has established a single quality assurance topical report that addresses the applicable regulatory and industry requirements. The provisions of W's NRC approved topical report which are identified in WCAP-8370, Rev. 12A, "Energy Systems Business Unit, Power Generation Business Unit Quality Assurance Plan," apply 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 Standard Safety Analysis Report (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. The AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan (QAPP), WCAP-12600, Rev. 2, affirms the commitments established in WCAP-8370 for the AP600 program and states that WCAP-12600 does not replace the WCAP-8370 document but establishes supplemental commitments for the AP600 Program. It also states that the AP600 design certification test programs and related analyses are within the scope of this QAPP.

3.1.1 OSU APEX

Specific quality provisions applicable to the OSU facility are identified in the AP600 Long Term Cooling Test Project Quality Plan (PQP), AP600 Document Number LTCT-GAH-001, Revision 1. This specific PQP establishes controls for the design and construction of the test facility as well as the conduct of testing activities.

The inspection team reviewed the above documents to determine if testing activities performed at the OSU facility associated with the design of the AP600 advanced reactor were conducted in accordance with the appropriate provisions of W's 10 CFR Part 50, Appendix B, quality assurance program.

In particular, the inspection team examined the items and services covered by the quality assurance program to confirm that these activities were accomplished under suitably controlled conditions by properly trained personnel and that the pertinent test data used to furnish documentary evidence of activities affecting quality were appropriately recorded and maintained.

Based on these document reviews and discussions with project management personnel it was ascertained that the organizational structure defined in the PQP had been effectively implemented and that the overall quality assurance program administered by W provided for appropriate organizational independence and authority to identify conditions adverse to quality. Facility personnel involved with managing and performing activities affecting quality were properly trained and knowledgeable of the quality elements embodied in the PQP and the associated W test control documents.

3.1.2 SIET SPES-2

The quality related activities associated with the AP600 full height, full pressure integral system tests performed at SPES-2 were conducted under the auspices of a cooperative agreement between W, ENEA, ENEL and SOPREN-Ansaldo. The purpose of this test program is to provide thermal hydraulic data for computer code validation and to simulate the operation of the AP600 plant passive safety systems. Specifically, the SPES-2 facility, which is operated by SIET, incorporates a full height, full pressure (FHFP), 1:395 volume scale simulation of the AP600 reactor vessel, reactor coolant system, and the passive safety injection system.

The general features of SIET's quality assurance system are described in their Quality Manual, 00001-QQ, Revision 3. Additional quality plan provisions specific to AP600 testing are also detailed in Procedure 00006-QQ-92, "Quality Plan Relative to Nuclear Area Orders," Revision 2, which includes a cross reference of European quality standards to the Criteria of NQA-1. In order to conduct these confirmatory tests, SIET has implemented an internal quality system which incorporates the requirements of both the International Organization for Standardization (ISO) Code Series 9000 (UNI-EN 29000) and the Italian Code UNI-8450 (1983). As described by SIET's Quality Assurance Manager, these codes were derived from ASME NQA-1-1983 and IAEA No. 50-C-QA, Revision 1.

The inspection team reviewed SIET's quality system practices described in the Quality Manual and Procedure 00006-QQ-92 as well as other specific QA implementing procedures. During the conduct of these reviews the inspection team was somewhat hampered by the unavailability of english translations of the subject documents (with the exception of Procedure 00006-QQ-92). However, using the interpretive services available at SIET, the inspection team successfully completed the examination of these documents.

Specific design requirements and quality elements associated with the SPES-2 test facility are delineated in W WCAP-14053, "AP600 FHFP Integral Systems Test Specification." These requirements include facility and test article design test matrix control, actuation setpoints, initial test conditions, test oversight and document approval, and quality assurance requirements. Under the provisions of this cooperative effort ENEA contracted with SIET to perform SPES-2, AP600 test facility modifications/construction, instrumentation calibration, QA plan development, facility operations and maintenance, and data acquisition/transmittal.

Based on the results of the team review it was generally determined that appropriate procedural controls had been established at the SPES-2 test facility which properly incorporated the applicable quality provisions of 10 CFR 50 Appendix B. It was also concluded that organizations participating in the QA program had appropriately assigned functional responsibilities and that quality related activities were accomplished under suitably controlled conditions.

3.2 Instructions, Procedures and Drawings

3.2.1 OSU APEX

The following W quality documents utilized for the AP600 Long Term Cooling test project were reviewed by the inspection team:

- WCAP-9565, Nuclear and Advanced Technology Division (NATD) Quality Assurance Program Procedures (DPs)
- WCAP-12600, Revision 2, AP600 Quality Assurance Program Plan
- WCAP-12601, Revision 13, AP600 Program Operating Procedures (APs)
- LTCT-GAH-001, Revision 1, AP600 Long Term Cooling Test Project Quality Plan
- PXS-TIP-001, Revision 2, AP600 Long Term Cooling Test Specification
- DP-2.0, Revision 6, Quality Assurance Program
- DP-3.2.6, Revision 8, Preparation and Control of Drawings
- DP-5.0, Revision 3, Instructions, Procedures and Drawings
- A-01, Revision 2, Testing Administration Procedure

Based on the review of these documents it was determined that appropriate procedural controls had been developed and implemented to govern the conduct of quality related test activities at the OSU APEX facility.

The inspection team also reviewed the following selected sample of test procedures to determine if these documents conformed to the provisions of

procedure DP-5.0 and that they contained appropriate quantitative or qualitative acceptance criteria for determining if quality related test activities were satisfactorily accomplished:

<u>Test ID No.</u>	<u>Test Procedure Title</u>
94001	Pressurizer Volume Determination
94002	Primary Sump and Secondary Sump Volume Determination
94003	BAMS Moisture Separators Volume
94004	Incontainment Reactor Water Storage Tank (IRWST) Volume
94005	Steam Generator 1 & 2 Secondary Side
94007	Reactor Volume Determination

As a result of these reviews it was concluded that the listed test procedures properly described test parameters including acceptance criteria. However, it was noted by the inspection team that the final acceptance of these documents which were characterized as in process records was to be conducted at W's Monroeville offices in accordance with Procedure AP-6.2, Revision 1, "Technical Document Release and Control."

3.2.2 SIET SPES-2

The following quality documents used by SIET for the SPES-2 integral systems test were reviewed by the inspection team:

<u>Document No.</u>	<u>Title</u>	<u>Revision No.</u>
00001-QQ	Quality Assurance Manual	3
00002-QQ	Document Control Procedure	2
00003-QQ	Instrument Management	1
00004-QQ	Quality Plan for Back Flow Prevention Test	3
00005-QQ	SIET Organization	2
00006-QQ	Quality Plan Relative to Nuclear Area Orders	2
00007-QQ	Internal Audits Procedure	0
00008-QQ	Plant Instrumentation and Data Acquisition System Set-up Procedure	1
00009-QQ	Measurement Chain Check and Modification Procedure	0

As previously noted all of the above documents except for Procedure 00006-QQ, which was provided with an English translation, were prepared in Italian. As a result, the inspection team focused on examining SIET's site quality activities by conducting interviews with responsible personnel and evaluating the contents of each of the above documents which governed the SPES-2 test program. This review indicated that, although some implementing procedures were still under development, the programmatic aspects involving quality related activities were properly prescribed, performed, and documented in accordance with established instruction, procedures, and drawings. Results within this area were generally acceptable however, one concern was identified

involving the lack of appropriate procedures and acceptance criteria for the confirmation of critical as-built dimensions. This concern is discussed in detail in Section 3.4 of this report.

To gain additional insights into the implementation and control of quality related activities for the SPES-2 test program, the inspection team reviewed a selected sample of site design record files (DRFs), which for a typical matrix test, contained an organized set of information including the following items.

- Pretest QA checklist
- Pretest checklist
- Test procedure
- Deviation form
- Deviation report
- Day-of-test report

Based on these evaluations it was determined that test information was properly recorded, reviewed, and approved for matrix tests in accordance with the commitment described in WCAP-14053, "AP600 FHFP Integral Systems Test Specification." With respect to design considerations and document control this review indicated that test procedures had been properly reviewed and approved by W's Monroeville office. Additionally, in order to determine the adequacy of administrative controls for testing activities at the SPES-2 facility the inspection team performed a detailed review of the following procedures:

<u>Procedure No.</u>	<u>Title</u>
00226PP93	Cold Leg Break Plan and Procedures
00293PP94	Cold Leg-Core Makeup Tank Balance Line Test Plant and Procedures
00227PP93	Direct Vessel Injection Break Plan and Procedures
00342PP94	Steam Line Break Plan and Procedures

As a result of these reviews, the inspection team concluded that the above procedures properly described test parameters, including acceptance criteria and QA activities. It was also noted that the final acceptance and closure of the design record files would be performed at W's Monroeville office in accordance with Procedure AP-6.2, Rev. 1, "Technical Document Release and Control."

3.3 Document Control

3.3.1 OSU APEX

The programmatic aspects of document control for activities affecting quality are described in Section 6.0 of the PQP. In order to determine the appropriate translation of these requirements into implementing procedures the inspection team reviewed the following procedures:

- DP-6.0, Rev. 4, "Document Control"
- AP-6.1, Rev. 0, "AP600 Document Numbering"
- AP-6.2, Rev. 1, "Technical Document Release and Control"
- AP-6.3, Rev. 0, "Preparation, Review and Approval of AP600 Documents"

As a result of these reviews and examination of the document control process it was determined that an effective system had been implemented to control the distribution and revision of quality related documents. In particular, the OSU records retention facility was efficiently operated with excellent controls over the distribution and maintenance of project documents including instructions, procedures and drawings. The effective administration of information resources of OSU was identified as a project strength.

3.3.2 SIET SPES-2

In order to evaluate the adequacy of the document control system at SIET the inspection team examined the administrative controls specified in Procedure 00002-QQ, Revision 2 "Document Control Procedure" and Procedure 00006-QQ-92, Section 4. The inspection team also evaluated the implementation aspects associated with the preparation, issuance, and revision of documents which specify quality requirements or prescribe activities affecting quality.

Based on the results of these reviews and the examination of SIET's document control facility it was ascertained that appropriate procedural provisions had been developed for the identification and distribution of controlled documents including the proper assignment of responsibility for preparing, reviewing, approving and issuing quality related documents.

3.4 Design Control

3.4.1 OSU APEX

The OSU APEX low pressure integral system test facility is a 1:4 linear scale model of the AP600 system, including a simulated reactor vessel and internals, two simulated primary coolant loops, a pressurizer and surge line, four simulated coolant pumps, two accumulators, and other associated piping and components. The facility is designed to operate at 400 psi pressure and 450 degrees F.

Design control was reviewed by tracing the facility concept, scaling approach, generation of design drawings, facility construction, and characterization testing. Although the design and construction of the test facility was accomplished to commercial quality standards, the scaling analysis documented in OSU-NE-9204 identifies certain critical attributes of the facility which would directly impact on the ability of the matrix test data to be used for benchmarking the computer codes developed for AP600 certification. The critical attributes are listed in Section 3 of the "OSU Implementation of the AP600 Long Term Cooling Test Project Quality Plan (LTCT-GAH-001)," document, dated May 5, 1994. These attributes are geometrical properties of the facility that must be measured to ensure that the important phenomena are adequately modeled in the facility and include the facility component (test

article) volumes, certain line resistances and the hot and cold leg piping inside diameters. Additionally, the W Long Term Cooling Test Specification states that all test articles and interconnecting piping shall be sized and located at the proper elevations based on the scaling study. The PQP implementation plan states that the procedures to measure these critical attributes will be developed and implemented by W. The following critical equipment volumes are used as input parameters for computer codes:

- Core Makeup Tank (CMT) Volume
- Accumulator (ACC) Volume
- IRWST Volume
- Primary Sump Volume
- Secondary Sump Volume
- Automatic Depressurization System (ADS) 4 Moisture Separator Volume
- Large Break Separator Volume
- Pressurizer (PZR) Volume
- Steam Generator (SG) shell side volume
- Reactor Volume

The team reviewed the results of several tests conducted to verify that the critical equipment volume tests were properly performed. The objective of these tests was to measure the volume of the major components of the test facility. The components were filled with water and drained. The drained water was weighed and the component volume was calculated. The calculated values agreed within 3% of the design values. These test results also agree with the values specified in the scaling analysis report (Ref: OSU-NE-9204). The team concluded that the tests were properly conducted and the test procedures were adequately implemented. The following is a list of the test procedures reviewed:

- OSU-V-01, Accumulator Volume Determination
- OSU-V-02, CMT Volume Determination
- OSU-V-03, PZR Volume Determination
- OSU-V-04, IRWST Volume Determination
- OSU-V-05, Primary Sump and Secondary Sump Volume Determination
- OSU-V-06, SG #1 and #2 Secondary side Volume Determination
- OSU-V-07, ADS Moisture Separators and BAMS Moisture Separator Volume Determination

The team also reviewed the documentation of the testing that measured the line resistances of the OSU model to verify that the tests were properly performed. The objective of these tests was to characterize the differential pressure versus flow for the piping connecting the major components of the test facility. The review consisted of: (1) determining if the tests had met the acceptance criteria (two had not); (2) checking a sample of the documentation of differential pressure (dp) measurement calibrations and verifying size marking on a sample of installed orifices; and (3) checking a sample of the hand-written values in the procedure tables with the typed numbers in the report (several typed values were incorrect). In two instances, W reported that characterization tests had not met the acceptance criteria. The first instance was a test of ADS-1, -2, and -3 flow. The second was a test of

primary sump injection flow. W stated that it, not OSU, make the final determination as to when test results are considered acceptable and that documentation of this evaluation and disposition would reside in the official design record file at W offices in Monroeville. The NRC will review W design record files in Monroeville during a future inspection to determine if W appropriately evaluated and dispositioned the test results that did not meet the acceptance criteria and if sufficient supporting documentation exists. The team identified these failures to meet acceptance criteria without sufficient supporting documentation as part of Unresolved Item 94-01-03.

3.4.1.1 Facility Scaling

The program objective for the facility were originally specified by W in PXS-T1P-001, "Long Term Cooling Test," dated August 20, 1992. This document describes the background that led to the facility development and the objectives of the testing program. This report is a reasonable requirements document to guide the facility's conception. Following this document, OSU proceeded to develop the facility scaling.

The facility scaling basis is described in "Scaling Analysis for the OSU AP600 Integral System and Long Term Cooling Facility" (OSU-NE-9204) Draft 1993. This report describes the basic rationale for designing a scaled test facility representative of the AP600 at 1:4 height and 1:192 volume scale. This report has been previously reviewed in some detail by NRC staff and the ACRS. Revisions and additions are still being made based on comments received and the final staff review must await receipt of the final report.

The scaling report describes the general scaling methodology and its relationship to Ishii-Kataoka scaling (NUREG/CR-3267) and Hierarchical Two Tier Scaling (NUREG/CR-5809). The report was discussed with its author, the OSU Program Manager. The report addresses the scaling approach for natural circulation and depressurization processes, which are relatively distinct portions of the experiments that comprise the test program. Nondimensional similarity scaling groups are identified and compared between the scaling facility and the prototype plant. The deviation of the ratio from one is a measure of the degree to which the process is distorted.

Most facility dimensional requirements follow from three basic decisions made to fix the facility design. The first is the 1:4 height decision and is a qualitative, engineering judgement. It takes into consideration the minimum height required to retain reasonable representation of gravity driven natural circulation forces, the size of the existing building to house the facility, and the cost. The second decision is the diameter scale ($\approx 1:7$), which is determined by the minimum dimension required to obtain representative flow regime transitions in the hot and cold legs. The l/d (length to diameter) ratio of the hot and cold leg was adjusted slightly (shortened and fattened). Similarly, the surge line l/d ratio was adjusted to the extent reasonable. The third decision was the pressure scale (400 psi) which is an economic decision.

3.4.1.2 Facility Design

Following these basic decisions, the next set of design decisions follow from piping friction resistances or Friction Number scaling. The chief designer was present during the inspection and the design process was discussed, however, the associated files could not be examined since the work was performed by W in Monroeville. The design was independently verified by the constructor (TIC) for the purposes of meeting Section VIII of the ASME code and State of Oregon requirements.

The reactor coolant pump is not fully representative of the prototypic pump since it has a different stopped rotor resistance. This resistance is, however, a relatively small component of the total loop flow resistance, which is dominated by the core and the steam generator. The core design has been simplified to reduce cost, but the resistance was matched.

The facility dimensional tolerances were generally specified to be ± 0.01 inches. Standard sizes were used for piping and tubing. The tolerance for placement of components was 1/4 inch in the vertical direction and 1 inch in the horizontal direction. The placement of components was verified by an as-built review.

The basic objectives and rationale for instrumenting the facility is given in PXS-TIP-001, "Long Term Cooling Test Specification." The instrumentation is intended to provide redundant means to determine mass and energy balances. Generally, there are two flow meters in series to provide a check. Discharge mass flow is measured by a flow meter and by a load cell on the catch tanks. The discharge flow passes through a separator, with the liquid flowing to the catch tank and the vapor to atmosphere. Both flow paths are instrumented to measure flow rate and temperature. The discharge flow measurement as it is instrumented would appear to provide good measurement of mass and energy loss to determine system mass and energy balances. The mass balance was stated to be within $\pm 2\%$.

3.4.1.3 Facility Construction

The OSU low pressure integral system test loop was procured as a commercial grade facility and constructed to the requirements of Section VIII of the ASME Code. The facility constructor, The Industrial Company (TIC), Northwest is an ASME Section VIII Certificate holder and provided on-site QA/QC manager to follow the construction quality. All process control and test instrumentation was also procured to commercial quality standards. Approximately 750 instruments for the measurement of flow, pressure, heat flux, power, temperature, and other parameters input into the data acquisition system.

3.4.1.4 Verification of As-built Dimensions and Elevations

According to the W Project Manager, the as-built dimensions of the facility were determined during a system walkdown by W contract personnel (Jacobsen Engineering) under W's supervision. The as-built dimensions were documented on isometric drawings which became a part of the system turnover package.

Selected isometrics were reviewed to assess the as-built verification process. Drawing No. ADS 1-3, Rev. 1, Automatic Depressurization System, for example, was an isometric sketch of this system with detailed dimensional information on the specific vertical and horizontal pipe runs, outside and inside pipe diameters, and specific notations regarding the type of connections. The Title Block contained several sign-off spaces which were initialed or signed, however, the title or organizational affiliation of the respective signers was not identified. Most of the layout dimensions were measured to fractions of inches with the smallest fraction being 1/4 inch. Piping dimensions were generally indicated as nominal pipe size although some inside diameters were recorded to three decimal places. It was not indicated whether the inside diameter dimensions were obtained by actual measurement. A general arrangement drawing (GAS-AP600-01) showed system elevations in fractions of inches with the smallest fraction being 1/16 inch.

No procedures or instructions had been prepared to provide guidance for the verification of the as-built dimensions. The failure to define significant parameters such as the measurement technique and the required accuracy or acceptance criteria for the as-built dimensional verification was identified as an example as part of Nonconformance 94-01-02.

As-built drawings were also reviewed to verify that the OSU model "actual" values assumed in the scaling report OSU-NE-9204 are correct. Major components (such as CMT and IRWST) physical dimensions such as height and diameter that are given in the scaling report were compared with the values from the as-built drawing. The as-built drawing dimensions were found to be within a reasonable range of values given in the scaling report.

A sample of facility items that have been changed to keep current with the AP600 design changes were also checked on the drawings, e.g., the direct vessel injection nozzle, core bypass holes, the rerouted cold leg pressure balance line, and piping orifices. One orifice (ORI-451) was marked correctly on the drawing, but the value in the database was different. Also, as-built drawings show orifices that are not installed, however, the database appeared correct, i.e., only installed orifices are listed.

3.4.1.5 Facility Characterization

Facility characterization testing was performed to verify line resistances, component volumes and to measure facility heat losses. The facility heat losses were measured in integral terms by heating the facility to full operating temperature and adjusting core power until a steady state balance was achieved. The measured value of 30 KW can be compared to the maximum facility power of 600 KW, which represent 5 percent of scaled full power. Therefore, heat losses will be in the range of 5 to 15 percent during the course of an experiment and should result in an acceptable level of distortion from the standpoint of providing an experimental representation of a transient.

During startup testing, some differential pressure instruments were found to be improperly ranged. The ranging was based initially on NOTRUMP

calculations. A substantial effort was required during startup to ensure grounding of instruments and to eliminate signal disturbances.

3.4.2 SIET SPES-2

SPES was originally commissioned by ENEA to simulate a W 312 (3-loop) pressurized water reactor with Italian specific design features utilizing a volume scale of 1:427, full height, and full pressure. For SPES-2 testing, the facility was modified to model the AP600 design using, when possible, the components of the existing plant (for example, steam generators). Beginning in early 1992, planning began to modify the facility for AP600 testing. Construction was completed in mid-1993, after which a period of shakedown and characterization testing commenced. Matrix testing began in early 1994 and was completed in October 1994.

3.4.2.1 Facility Scaling

The SPES-2 facility is 1:395 scaled volume with respect to the AP600. The scaling criteria maintained selected significant parameters including the fluid thermodynamic conditions, vertical elevations, power to volume and flow rate ratios, transit time of fluid, and thermal flux. Additional scaling criteria were applied to the design of selected components such as lines in which friction pressure drops were relevant.

SPES-2 is full height and so its diameter scale is approximately 1:20. This ratio distorts the surface to volume ratio and, therefore, heat loss to the environment. The scaled heat loss is about 8 kW while the actual heat loss is about 150 kW, therefore, the distortion is approximately 140 kW. This is not a very severe problem for larger LOCAs (i.e. greater than break sizes of about 3 inches) but it is a problem for smaller LOCAs. This is because small LOCAs proceed longer and decay heat decreases to less than the facility heat loss. Therefore, in order to run sensible small break LOCAs, heat loss is compensated by adding the heat loss to the core power.

The heat transfer area in the SPES steam generator is underscaled by approximately 50% compared to AP600. To obtain the correct heat transfer and primary system conditions, it is necessary to lower the secondary side pressure during the start of the experiments.

3.4.2.2 Facility Design

The facility design is described in WCAP-14053, "AP600 SPES-2 FHFP Integral Systems Test Specification." The design parameters were then converted into design requirements described in WCAP-13277, Rev. 1, "Scaling, Design, and Verification of SPES-2." WCAP-13277 requirements were then used for preparation of design drawings, procurement activities, and facility construction. The completed facility, then, is described in WCAP-14073, "SPES-2 Facility Description Report," (May 1994). Several updates to this last report were noted in the master copy at SIET, but had not been reissued.

The design modifications to the facility were prepared interactively by SIET and W. SIET drafted drawings and sent them to W, where they were reviewed and

adjusted. The iterations continued informally for some period of time until the design drawings reached a final draft stage, at which time they were placed under change and approval control. W checked the final drawings against the scaled dimensional requirement to assure conformance.

Much of the previously existing SPES facility was retained in SPES-2. Two of the steam generators were retained, as was the reactor vessel, pressurizer, and reactor coolant pumps. The elevation difference from the steam generator tube sheet to the hot leg and cold leg centerlines was preserved. The pressurizer elevation was changed to represent proper scaling of its bottom elevation with respect to AP600. The pressurizer is 60% of full height, which provides some distortion in volume versus elevation. Its aspect ratio was based on the Wilson bubble rise model and should preserve level swell. A new surge line was added representing the correct AP600 geometry. The size of this line (1 1/2 inch schedule 160) is small in terms of maintaining flow regimes. New components added to the SPES-1 facility included core makeup tanks, accumulators, passive residual heat removal (PRHR) system, and automatic depressurization system.

A number of design changes and their rationale were reviewed. Much of the scaling and design work associated with modifying the facility consisted of maintaining elevations and vertical dimensions at 1:1 in order to preserve gravity driving force and to maintain piping friction resistances, or Friction Number scaling. By obtaining the correct driving force and the correct resistance, natural circulation should be well represented.

New hot legs and a new surge line were added for correct geometric representation. The hot leg was scaled according to Froude number to preserve the transition to stratified flow. The upper plenum/upper downcomer region of the reactor vessel was replaced. It includes the arrangement of two hot legs and four cold legs, with center line elevations preserved. The direct vessel injection lines also enter this part of the vessel. The upper downcomer region is represented as an annulus. The annular region includes fins to represent the circumferential flow resistance present in the full scale plant, however, matching the flow resistance between adjacent cold legs was not an objective.

At the lower end of the upper plenum/upper downcomer, the downcomer converts from an annulus to an external pipe, which parallels the core region of the vessel and reenters in the lower plenum. Natural circulation flow patterns that may result from this geometry were calculated using a computational fluid dynamics code. The results were used to help determine locations for thermocouples. The results were also compared with a RELAP calculation, with reasonable agreement obtained.

The ADS4 valves have been oversized by about a factor of three in SPES compared to the scaled value from AP600. This was done based on analyses performed by Ansaldo using RELAP5. The calculations indicated that, because the metal mass of SPES is substantially overscaled, the resultant distortion in vapor generation during the approach to IRWST injection was large.

The inspector reviewed the "master copy" of the facility description report for SPES-2. This copy includes a penciled-in list of corrections and modifications to the original report. The pencil corrections did not appear to have been incorporated as yet into the as-built design drawings. Additionally, the SPES-2 facility description report contains a list of instrumentation. From examining the list of instruments, one can infer that the underlying rationale is to attempt to track mass and energy in the system during experiments. In addition, the relation between the processes and states to be measured and the method of measurement is described.

The team noted that, for all design documentation, such as reports and drawings, relevant information was properly archived and easily retrievable by SIET personnel.

3.4.2.3 Facility Construction

The SPES test facility was designed and constructed to commercial standards. The existing plant modifications were performed by commercial contractors or SIET also using commercial standards.

During the early phases of the SPES facility modification, component and construction drawings were not covered by the SIET quality assurance program and design changes to manufactured components or plant system arrangement were not controlled under the SIET quality system provisions. Several revisions of the facility drawings were made during this period. These revisions were recorded by letter designations (A,B,C, etc.). After the facility design was "stabilized", the drawings were placed under the SIET QA program controls and further revisions were issued with numerical revision designations. According to SIET management, the first numerical revision of each drawing was verified by the technical manager for conformance to design requirements by checking as-built dimensions. However, there was no formal record of such verification and this process was not controlled by any procedures or instructions.

All revisions of Drawing No. 25.01.02, SPES-2 Annular Downcomer, were reviewed to assess the traceability of modifications through the construction phase of the facility and to verify the validity of the as-built dimensions. This drawing encompassed three sets of letter revisions: B (October 20, 1992), C (December 3, 1992), and D (October 22, 1993). The revisions incorporated significant changes to the drawings which could be identified by examining file copies of marked up drawings. These changes, however were not specifically identified in the revised drawing either by summarizing or describing them in the appropriate space in the drawing title block or by highlighting the actual changes on the drawing.

The first numerical revision of drawing No. 25.01.02 (Revision 0) was issued on October 29, 1993, and was marked "issued" in the revision block. According to the system described above, Revision 0 was issued under the SIET QA system and represented the as-built status of the facility as of that date. It could not be determined by examining Revision 0 and the previous revision (Revision D) which dimensions of the drawing were actually verified before placing this drawing under the SIET QA system as representing the as-built status of the facility. Several dimensions on a copy of Revision D of the drawing were

circled, others were underlined, and some were both circled and underlined. The significance of these marks was not explained and it could not be determined which dimensions were actually verified or whether the verified dimensions included the critical dimensions of this particular assembly. Revision 1 of this drawing was issued on November 3, 1993, and appeared to be identical to Revision 0. The revision block contained the notation "As-Built" with no indication whether it incorporated any changes from the previous revision (Revision 0). The failure to provide instructions or procedures for the verification of critical as-built dimensions when placing uncontrolled drawings under the QA system was identified as an example as part of Nonconformance 94-01-02.

3.4.2.4 Verification of As-built Dimensions and Elevations

The AP600 SPES-2 FHFP Integral System Test Specification (WCAP-14053) states that all test articles and interconnecting piping shall be sized and located at the proper elevations based on a scaling study using prototypical AP600 dimensions as the basis. Since the test facility was constructed to commercial standards without significant QA involvement, conformance with critical system elevations, dimensions, and volumes was established by performing verification of these attributes under the SIET QA system. The determination of critical dimensions was accomplished by utilizing a combination of as-built drawing dimensions and measured layout/elevation dimensions. According to the SIET Technical Manager, critical elevations were determined by establishing a reference level (the top of the steam generator tube plate), stringing horizontal wires at that elevation, and using plumb line/liquid level techniques to obtain other reference planes from which specific elevation measurements were taken.

Although a general SIET procedure describing different measurement techniques (00216P093) was available, this procedure did not identify specific measurement methods to be used for a particular application or prescribe the measurement accuracy required for each type of measurement. A review of the system elevation drawings and isometrics shows linear dimensions and elevations in millimeters with an apparent accuracy to one millimeter over a distance of 20,000 millimeters. No tolerances were indicated for these measurements. The failure to specify accuracy requirements or include any acceptance criteria for the verification of the as-built facility dimensions was identified as an example as part of Nonconformance 94-01-02. The team reviewed several design drawings for the SPES-2 facility, including preliminary, issued, and as-built versions. All drawings appeared to be appropriately signed off by SIET personnel. Some were missing the "final approval" signature; SIET's explanation for this omission was that the final approval was done by W, and the final approval is thus documented by correspondence, but not shown on the drawing itself. The team was able to find the records of communications between W and SIET indicating review and approval of drawings in the appropriate volumes of the DRF.

3.5 Procurement and Calibration of Test Instrumentation

3.5.1 OSU APEX

The OSU PQP states that the procurement of instrumentation and calibration services shall include documentation of calibration traceable to national standards. It also states that OSU shall be responsible for the development of appropriate calibration procedures and that these procedures shall be approved by the OSU Program Manager and submitted for Quality Assurance review and Project Manager information. Section 5.0 of the PQP also states that activities affecting quality shall be performed in accordance with documented instructions or procedures and that such instructions or procedures shall be prepared in accordance with DP-5.0 which establishes specific responsibilities and requirements for the preparation, review, and approval of such procedures.

All instrumentation for the OSU test facility was procured by either OSU or TIC to commercial standards. Acceptability of this equipment for quality related activities was typically demonstrated through on-site calibration after delivery. Instrumentation is carefully arranged and fully labeled, including channel number, serial number, and calibration sticker. TIC performed the initial instrumentation calibration and installation. Each type of instrument was procured from a single manufacturer. For example, the pressure and differential pressure instruments (p and dps) are from Rosemount, thermocouples are from Gordon, and flow meters are from Foxboro. Record keeping of instrumentation calibration was thorough. The current record of each instrument is kept in a well organized notebook. Prior records are stored in a file cabinet. The test equipment is identified and the next calibration due date is noted. The test equipment is checked annually by Transcat or Fluke. Selected records of procurement and calibration activities were reviewed as summarized below.

- **Thermocouple calibration**

Most of the thermocouples used in the test facility were calibrated by TIC at ambient and two elevated temperatures (225F and 450F) and checked for linearity of signal prior to installation using certified calibration equipment. The calibration procedure (TIC C.P. #19) consisted of a half page document which was not dated and did not identify who originated, reviewed, or approved the document. According to this procedure, the ambient temperature calibration was to be accomplished using a certified temperature calibrator while the elevated temperature calibrations were to be obtained using a dry well calibration device. The procedure did not identify specific well sizes to be used for different thermocouple configurations but stated that the "proper well size" was to be used. The failure to identify and record the preparation, review and approval chain on calibration procedure TIC C.P. #19 was identified as an example as part of Nonconformance 94-01-02.

Review of the calibration records for the temperature calibrator and the dry well calibrator indicated that the calibration certificates for these devices were issued by Industrial Instruments, Inc. of Sandy, Utah. These certificates stated that the instruments had been calibrated using standards

whose accuracies meet or exceed acceptable calibration ratios and are traceable to either The National Institute of Standards and Technology (NIST) or to nationally accepted consensus standard. The certificates were signed by a metrologist and QA technician and were stated to be effective for one year from the calibration date (December 1993).

TIC did not invoke any QA requirements for the calibration/certification services performed by Industrial Instruments, Inc. and had not performed any audits or surveillance of that company. Section 7.5.2 of WCAP 8370, which is referenced in the PQP as applicable to activities important to the test data, states that "When a CoC is used for acceptance ... verification of the supplier's CoC and certification process is performed during audits or surveillance." The acceptance of calibration certifications from an unaudited company was identified as an example as part of Nonconformance 94-01-01.

- Flow Meters and Pressure Transmitters

Review of selected purchase orders for flow meters and pressure transmitters indicated that these devices were procured by OSU as commercial grade equipment to a catalogue model number without referencing any special quality or technical requirements. While such practice may be appropriate when OSU can independently assure the desired output accuracy (calibration) through on-site qualification, it was noted that the flowmeter calibration was also performed by a commercial supplier (Foxboro). The failure to verify the validity of the calibration certificates received from an unaudited commercial supplier is identified as an example as part of Nonconformance 94-01-01 as discussed above.

Review of the OSU procedure for the calibration of commercial grade Rosemount pressure transmitters indicated that this procedure had been prepared in accordance with the applicable PQP requirements. It contained detailed technical instructions as well as the appropriate reviews and approvals.

3.5.2 SIET SPES-2

The calibration of all measuring instruments at SIET is controlled in accordance with procedure 00003-QQ. Each instrument is individually identified with a registration number. The technical data pertaining to this instrument, including all calibration information is reported on an instrument card traceable to the instrument through the registration number. The instrument configuration for each test is reported in each test procedure.

The primary test instruments at the SPES-2 facility include thermocouples, pressure transmitters, and flowmeters. The flowmeters are generally venturi or calibrated orifice type although turbine meters are used in some applications. These instruments are typically procured to commercial standards (UNI) and calibrated by SIET. The instruments are required to meet Italian national consensus standards and are also calibrated to accuracy ranges established by these standards for specific instrument types. The accuracy ranges prescribed by these standards have reportedly been reviewed by SIET and found to meet or exceed the accuracy requirements stated in the FHFP Integral Systems Test Specification.

SIET has a functional metrology laboratory which is capable of calibrating all test instrumentation used in the integral systems tests. All primary standards used in the metrology laboratory are periodically calibrated to Italian national standards under the Sistema Italiano Taratura (SIT). This system includes three organizations classified according to the type of calibration (mechanical measurement, electrical measurement, etc.) which are empowered to accredit specific calibration laboratories. For example, Institute Meterologico Gustava Colonnetti is the organization empowered to accredit laboratories performing calibrations of mechanical and thermal measurement devices. Laboratories accredited under this system are identified in a register of accredited laboratories. SIET uses only laboratories listed in this register to calibrate their primary calibration standards. The SIT system, in turn, is audited by the Western European Calibration Cooperation (WECC). No user audits are performed by organizations utilizing this system.

Thermocouples are calibrated by the SIET metrology laboratory using a comparison method. Three constant temperature baths (water, oil, or molten salt) are available depending on the temperature range of interest. The primary standard is a resistance thermometer calibrated by an accredited laboratory with the calibration traceable to Italian national standards. The calibration is performed in accordance with an approved procedure (00165P092) specific to this activity. Pressure transmitters are calibrated using a dead weight method to SIET procedure 00118P091, while flow meters are calibrated to SIET procedure 00211PP92 using a gravimetric method. Since these procedures were written in Italian, a detailed review of their contents was not made. A limited translation, combined with observation of calibration activities and a review specific calibration records indicated that the calibration process was adequately controlled.

3.6 Test Control

3.6.1 OSU APEX

Westinghouse Engineering Procedure AP-3.11, "AP600 Testing," provides Specific AP600 test engineering instruction for implementing the requirements of the AP600 QAPP (WCAP-12600) and the NATD QA Program (WCAP-9565). AP-3.11 requires the preparation of a test specification for all safety-related AP600 tests. The team reviewed PXS-TIP-001, Rev.2, "AP600 Long-Term Cooling Test Specification," to verify that provisions of AP-3.11, including purpose, test objectives, test facility requirements, test articles, instrumentation requirements, data acquisition, test operation, test reports, data requirements, and quality assurance requirements required for the tests, were adequately covered in the test specification. The team determined that the PXS-TIP-001 test specification included all the appropriate provisions and was found to be adequate for the tests.

AP-3.11 also gives guidance for the preparation of the test procedure used for the conduct of the tests. Appendix C of AP 3.11 is a sample format given for the preparation of the test procedure. The team reviewed the test procedures for the tests SB-2, SB-3, SB-4, and SB-5, to verify that the test procedures used for the conduct of the tests were prepared according to Appendix C of AP-3.11. The team determined that test objectives, prerequisites, initial

conditions, instructions, acceptance criteria, and post test activities for the conduct of the tests were included in the test procedures. The test procedures used for the test were found to be acceptable.

3.6.1.1 Test Matrix

Following completion of facility characterization, a matrix testing program comprised of 14 experiments was conducted. These tests investigate a range of break sizes in the cold leg, direct vessel injection line, and pressure balance line. Then, a series of 12 additional tests were defined with seven having been performed at the time of the inspection. The remaining five were to be completed by the end of September, 1994. The W rationale for the additional 12 tests was not reviewed, however, the test matrix in total was reviewed.

The test matrix appears to provide a reasonable coverage of varying break sizes, break locations and break orientations. Breaks are included in the CMT cold leg, the PRHR cold leg, the PRHR hot leg, the direct vessel injection line, and the pressure balance line. A two inch break in the bottom of the CMT cold leg was chosen to perform a number of parametric variations in operation of the ADS and non-safety systems.

3.6.1.2 Test results and Data Processing

The instrumentation system operates normally with data recorded at eight second intervals. It can operate in the "burst mode" to record data more frequently, however, the interval over which this can occur is limited because of the capacity for data storage. While this may be adequate when conditions are changing slowly, the limited recording frequency is a concern because it may not catch higher frequency phenomena. The data must be reviewed carefully to determine whether there is evidence that this may have occurred. A data uncertainty evaluation was performed by W. This was not examined by the inspection team.

OSU performs the data reduction from the data acquisition system (DAS) to optical discs. OSU developed a software program called REDUCE to process the data. The reduction converts voltage signals to engineering units (with the exception of thermocouple data which is recorded on the DAS directly as temperature) and parses the data in about 1/2 hour. Temperature compensation of level differential pressure cell to produce a true collapsed liquid level is performed by W and not OSU. The software was subjected to a verification process that appeared to be very thorough. First, one channel from each measurement type was performed. Then every channel was verified for several recording points at the beginning and end of a test, which basically amounted to a complete channel by channel verification. A check was made for roundoff error as part of the software validation, with none found. The software flags any reading that indicate a clearly failed instrument, i.e. very high or very low. Differential pressure zero points are checked before and after each test. Thermocouples are checked against each other.

The OSU APEX operations team does not perform any formal data analysis. This work is performed by W in Monroeville so, therefore, was not examined. The

testing is conducted in a production mode, with normally two tests performed per week. The timing restricts the extent to which the data can be reviewed while the testing is in progress. This permits limited feedback from data analysis to the testing program. Often, data review can highlight the need for additional instruments or tests to evaluate a phenomena or process. The production mode of testing permits limited opportunity for such feedback. In addition, the data review will continue after the current test operation disperses. Thus, there may be some concern over whether W analysts will be able to consult with operations personnel when questions arise on suspect data indications.

Nevertheless, some feedback is occurring. Additional thermocouples are being placed in the cold legs to measure thermal stratification. Also, instruments are being placed in the cold legs, lower plenum and upper plenum to measure condensation events which have occurred in several of the tests. The instruments are fast response pressure transducers with a high frequency recording system. Currently, there are apparently no plans to instrument the PRHR to measure condensation despite the probable occurrences of condensation events in APEX.

3.6.1.3 Staff Training and Test Conduct

The operations team at OSU is comprised of OSU, TIC and W personnel. The team experience was stated to exceed 100 years. The training records of the OSU operators were reviewed. Each has extensive experience and exams were administered covering a wide range of subjects including instrument theory and practice. The instrumentation personnel were experienced with calibration and troubleshooting.

Test acceptance criteria are simple but consistent with the contractual obligations of OSU. The tests are run according to the test procedure and the data is recorded. Repeat of tests has been limited to when problems occurred with a DAS computer such that the data was not recorded.

Early in the testing program, problems were encountered with failures of magnetic flow meters in the cold legs. The cause was thermal stratification that normally occurs during experiments due to operation of the PRHR. The thermal stratification cracks allowed water to leak into the meter's electronics. While the instrument was tested to withstand thermal shock causing uniform change in flow temperature, it evidently cannot withstand highly non-uniform temperature. Seven magnetic flow meters failed before W decided to forego the measurement. The manufacturer (Foxboro) is currently trying to resolve the problem. Aside from the cold legs, the magnetic flow meters have been used elsewhere in the system without problems and have provided excellent data (when conditions are nearly single phase).

3.6.1.4 Witness of Matrix Test SB-31

The team witnessed Matrix Test SB-31 on August 31, 1994. The objective of the test was to obtain thermal hydraulic data with spurious "S" signal, but without actuation of the ADS valves. A copy of the test procedure was made available for use at the facility. The team briefly reviewed the test

procedure OSU-MATRIX-SB31, Revision 0, "Spurious 'S' Signal Without ADS Actuation and with No Break Simulation, Long-Term Cooling Operation." The test supervisor gave a short pre-test briefing on procedures and expected system behavior.

The "S" signal scrams the facility, opens the CMT isolation valves and the PRHR outlet valves. Since no pipe break was simulated, ADS stays closed. Cold CMT water drains into the core, expands, and consequently increases the Pressurizer level. Meanwhile, PRHR cools the core and shrinks the pressurizer level. Consequently, the pressurizer level was oscillating. From the control room observations, the system interactions and the system behavior were as expected. The test was performed for about one hour and 30 minutes.

Prior to the conduct of the test, the inspection team witnessed the completion of the test prerequisites including the briefing of operations personnel. The inspection team determined that the specified test parameters were properly established and that initial conditions and system line-ups were appropriately performed. Throughout the test the operations personnel demonstrated excellent communications and the resultant test data was accurately recorded. No discrepancies were identified by the inspection team and the observed activities indicated appropriate test control measures.

3.6.2 SIET SPES-2

3.6.2.1 Test Matrix

The test matrix for SPES-2 was specified by W. The test procedures are approved by the Test Engineer and the Test Requester. The procedures state the test objectives, specify initial conditions with acceptable tolerances, and specify critical instruments that must be operable. The procedure contains a check list with signature blocks for the key steps. The test procedure also includes test termination criteria, although in practice, a test may be continued longer. A day of test report is issued normally within a day of the conduct of the test. W reviews the test outcome to determine whether the test was valid or must be repeated. Three tests have had to have been rerun. The first matrix test was repeated due to AP600 design changes, the pressure balance line break (test 7) was repeated due to a valve being inadvertently left closed at the start of the test, and the main steam line break (test 12) was repeated due to a leak that developed. Dispositions for these tests were located in the respective design record files.

All the experiments are run with initial conditions representative of normal full power conditions in the AP600. No tests were planned to explore how variations in initial conditions might affect the results. Pre-test predictions are performed by Ansaldo for W using the RELAP5 code. The calculated results are used to help determine how to operate SPES-2 in order to best match the target initial conditions. The calculations are available during the test and can be checked against the experimental data as the test proceeds. This was done during the one inch cold leg break test that was witnessed on October 15, 1994.

3.6.2.2 Test results and Data Processing

Similar to the arrangement between W and OSU, SIET is not responsible for data analysis, which is performed instead by W. In addition to quick look reports for each experiment, W plans to issue: a Final Test Report that describes the experimental data (March 1995); a Data Analysis Report that compares the code with the experimental data (May 1995); and a Code Validation Report that support the application of the code to AP600 analysis (September 1995).

The team reviewed the part of the DRF called "Approvals." Clear evidence was found of close communication between SIET and W, and that W had direct approval authority for test procedures for all preoperational and matrix tests. Final acceptance of completed tests also is W's responsibility, but there was no indication in the SIET DRF of final acceptance. W stated that this information should be included in the official W DRF in Monroeville. Documentation of tests in the SIET DRF includes a DOT report, indicating pertinent information about the test, and other documentation, such as signed-off procedures checklist; instrumentation listing; and pre-test acceptance criteria. According to W's test engineer, SIET is responsible for completing DOT reports only for those tests that are initiated (although they may not be completed). Any test for which the transient is initiated is assigned a unique number. For tests that are aborted due to a system failure (e.g., leaks, power problems, malfunctioning equipment) prior to initiating the test, no unique number is assigned, and no DOT reports are required. However, any problems, maintenance to the facility, and so forth, occurring since the previous numbered test, are supposed to be recorded on the DOT report. In addition, for tests that are initiated but not completed, or for tests that are found on post-test examination not to meet acceptance criteria, test deviation reports are required by QA procedures.

The inspector noted some inconsistency in documentation of completed tests in the DRF. Indication of the reasons for aborting tests, either prior to or after test initiation, was not always documented and deviations also were not always documented on the DOT report or subsequently in the DRF. Some "deviations" are indicated on the DOT reports as "unexpected events" or "observations." While most problems were documented somewhere in the DRF, it appeared that the procedures for indicating deviations and documenting their resolution were unclear and the practice was inconsistent. The following are examples of this practice.

- Test S00605 (volume on DVI line breaks): Stage 4 ADS valves failed to open automatically as specified. They were opened manually about 90 seconds later than the automatic signal. This was not documented as a "deviation," but rather as an "unexpected event."
- Test S00401 (volume on cold leg breaks): Stage 4 ADS valves failed to open automatically as specified. They were opened manually about 4 minutes after the automatic signal. This was not documented as a "deviation," but rather as an "observation."
- Test S00807 (volume on pressure balance line breaks): This test was repeated because a manual valve on the break line was inadvertently left

in the closed position during the beginning of the test. Shortly after the test began, anomalous behavior was observed, and the closed valve was determined to be the cause. The valve was opened manually and the test continued to completion, but it was determined (apparently by W Test Engineering) that the closed valve was sufficient cause to rerun the test. The Day of Test report does not indicate why the test was declared to be invalid, and the closed valve is not indicated as a "deviation," but rather as an "unexpected event."

The above examples are identified as part of Unresolved Item 94-01-03 and the NRC will review the W design record files in Monroeville during a future inspection to determine if W appropriately evaluated and dispositioned the test results.

3.6.2.3 Witness of Matrix Test

On October 15, 1994, conduct of a one inch cold leg break test was observed. This test was a variation on Matrix Test 1, and was intended to evaluate the effect of PRHR cooling. The latter test included full (three tubes) PRHR cooling while the former included reduced PRHR capacity (one tube). The use of three tubes is representative of "best estimate" plant conditions while one tube is used because it represents approximately the scaled PRHR heat transfer assumed by W in its AP600 SSAR analyses of most design basis events. Pre-test calculations were performed by Ansaldo for three cases of heat loss compensation: 150 kW, 100 kW, and 50 kW. Examination of the results indicated that overall, the 150 kW compensation appeared to be the most prototypic, so the experiment was run accordingly. The agreement of the Ansaldo RELAP code predictions with the test data was very good for the data channels observed. The design record file includes a folder with all the Ansaldo pre-test calculations.

SIET delineated the functions of test engineer, test conduct, and data analysis and is responsible for qualifying the data. Attaining initial conditions was complicated by the need to supply steam to the feedwater heaters from the steam line, since the adjacent fossil plant, which normally supplies steam to SPES-2 was out of service. The use of a different steam supply and procedures was properly executed/managed and the experiment was successfully performed.

Through various discussions and observations, the subject personnel were considered very competent and the test was performed smoothly and professionally. No discrepancies were identified by the inspection team and the observed activities indicated the use of appropriate test control measures.

3.7 Corrective Actions

3.7.1 OSU APEX

The procedural controls for the corrective action program at OSU are described in Section 13.0 of the site PQP. Specific requirements regarding the identification and resolution of conditions adverse to quality are delineated

in Procedure DP-15.3, Revision 3, "Quality Performance Feedback." This document establishes the provisions for the control of non-conforming items including the description of functional/organizational responsibilities and identification of the management process for assessing appropriate resolution of discrepancies.

In order to evaluate the effectiveness of the corrective action program at OSU the inspection team examined a selected sample of deviations identified during the design, construction, and operation of testing activities. As a result of these reviews it was generally concluded that appropriate provisions had been established and documented to assure that conditions adverse to quality were properly identified and corrected. It was also determined that appropriate process controls had been implemented for the documentation and reporting of significant adverse conditions to appropriate levels of management.

3.7.2 SIET SPES-2

SIET's corrective action program is described in Section 16 of their Quality Manual. The inspection team evaluated the pertinent portions of the Quality Manual which addressed nonconforming equipment and test samples, technical concerns, and the corrective action processes. Additionally, the applicable sections of Procedure 00003-QQ involving nonconforming instruments and the relevant aspects of Procedure 00007-QQ associated with internal audit findings were examined.

Collectively the administrative provisions of these procedures combined with the training received by department personnel involving test deviation documentation indicated that appropriate provisions had been established for the prompt identification of conditions adverse to quality. Relative to the identification of root cause determination and corrective actions for significant conditions adverse to quality, the inspection team reviewed the results of Quality Status Reports 00269-VV-93 Revision 0 and 00333-VV-94 Revision 0.

Based on the results of these reviews it was generally concluded that appropriate provisions had been established for the documentation of conditions adverse to quality and that corrective actions had been initiated by appropriate levels of management regarding significant conditions adverse to quality. However, one potential area of concern was identified involving deviation reports which did not properly establish the basis for resolving discrepancies in test parameters. This issue is discussed in detail in Section 3.6.2.2 of this report.

3.8 Quality Assurance Records

3.8.1 OSU APEX

Quality Assurance records are prescribed in Section 14.0 of the PQP and describes typical quality records associated with the Long Term Cooling Test project. Additionally, requirements involving the retention of quality records at the test site pending transfer to the AP600 Control File are delineated in the PQP. The inspection team reviewed OSU's implementing

procedures A-01, Rev. 2, "Testing Administration" and A-03, Rev. 0, "Control of Test Data" to determine the adequacy of program controls.

Based on the inspection teams reviews within this area, it was generally determined that the inprocess records associated with test activities were properly stored and controlled and were easily retrievable. Procedure A-03 contained detailed instructions related to the development and control of electronically recorded test data and provisions for the proper identification and maintenance of documenting evidence of activities affecting quality had been established.

3.8.2 SIET SPES-2

Quality assurance records requirements associated with the AP600 integral system test program at the SPES-2 facility were instituted for the collection, identification, and preservation of test records pending their transmittal to W. The administrative controls associated with these activities are delineated in Section 12.0 of the Quality Manual and Section 4.0 of Procedure 00006-QQ. The inspection team examined the referenced administrative controls and determined that appropriate provisions had been developed for the assimilation of documents that had been designated to become test records. Specifically, provisions had been established in SIET's document plan for the proper maintenance of test related records which identified the document title, identification number, revision status, initiating organization, approval authority, and effective issue date. Additionally, the document plan properly specified the requirements for external control and distribution of project documents.

In order to evaluate the implementation of SIET's document control plan the inspection team reviewed a selected sample of QA documents including test reports, instrumentation calibration reports, plant log reports, and the facility description. The results of these reviews indicated that documents which were designated as QA records were legible, accurate and complete, and that they were properly indexed and readily retrievable. No discrepancies were identified and it was determined that the QA records control process was well established and properly implemented.

3.9 Audits

3.9.1 OSU APEX

OSU performed no internal audits of the test activities at the APEX test facility. This function was conducted by W for OSU APEX. The team reviewed the 1994 W readiness review activities and reports as well as the August 1994 W corporate Projects QA audit at OSU. The team concluded that it appeared that W conducted an appropriate QA overview sight of OSU activities at the APEX test facility.

OSU stated that no external supplier audits were performed of equipment suppliers since all equipment was purchased as commercial grade from known suppliers. This area will be reviewed further at W Monroeville.

3.9.2 SIET SPES-2

The team reviewed Section 17, "Internal Quality System Audits," of SIET's Quality Manual, 00001-QQ, and implementing Procedure 00007-QQ, "Internal Audits Procedure." The team reviewed the lead auditor qualification and certification records for the SIET QA Manager and his staff, and all documentation appeared current. The team also reviewed related internal audit documentation such as: annual audit plans, audit checklists, audit reports and responses to audit report findings. The team verified that appropriate corrective action was performed by the responsible personnel for audit report findings.

External supplier audits are not within the scope of the testing contract with W for SPES-2. SIET stated that purchased equipment (such as pipe and valves) was procured from reputable, known Italian suppliers and accepted for use with an in house receipt inspection. SIET maintains a list of supposedly reputable, internationally known suppliers of instrumentation (e.g. Honeywell) who are used as suppliers if they provide the instrumentation with calibration certificates. SIET also purchases thermocouples from local suppliers who maintain an ISO 9000 QA program based on this certification.

The team concluded that SIET was implementing an appropriate internal audit program for SPES-2 test activities. The conduct of audits external to the SIET organization was not examined.

4 PERSONNEL CONTACTED

Oregon State University (OSU) APEX

- * Jose N. Reyes, Program Manager, OSU
- * John Groome, Facility Operating Manager, OSU
- * Moshe A. Mahlab, OSU Test Project Manager, W
- * Carl Dumsday, OSU Test Manager, W
- * Louis Lau, Senior Engineer, W
- * Bob Tupper, AP600 Project Engineer, W
- * Kenneth A. Kloes, Quality Assurance Engineer, W

Westinghouse Exit Meeting Participants on Conference Call

- * Brian McIntyre, AP600 Licensing Manager
- * Eugene Piplica, AP600 Test Manager
- * David Alsing, AP600 Quality Assurance Manager
- * John Butler, AP600 Licensing Engineer

Societa' Informazioni Esperienze Termoidrauliche (SIET) SPES-2

- ♦ Carlo Medich, Technical Manager, SIET
- ♦ Gustavo Cattadori, Head of Component Qualification Division, SIET
- ♦ Marina Bacchiani, Nuclear Reactor Division, SIET
- ♦ Alberto Musa, Quality Assurance Manager, SIET
- ♦ Marco Rigamonti, Test Engineer, SIET

- ♦ Eugene J. Piplica, AP600 Test Manager, W
- ♦ Larry E. Conway, AP600 Test Engineer, W
- ♦ Kenneth A. Kloes, Quality Assurance Engineer, W

GE Nuclear Energy

- ♦ Paul F. Billig, SBWR Test Engineer

Nuclear Regulatory Commission

- * ♦ Richard P. McIntyre, Team Leader, Special Inspection Branch (TSIB)
- * ♦ Uldis Potapovs, TSIB
- * ♦ Robert M. Latta, Quality Assurance and Maintenance Branch (TQMB)
- ♦ Alan E. Levin, Reactor Systems Branch (SRXB)
- * George Thomas, SRXB
- * ● David E. Bessette RPSB/Research (RES)
- * Harold H. Scott, RPSB/RES
- ♦ David T. Tang, Standardization Project Directorate

- * Attended the exit meeting at OSU on September 1, 1994
- ♦ Attended the exit meeting at SIET on October 14, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

February 6, 1995

Wyle Laboratories
Attn: Sherwyn Hyten,
Senior Vice President
7800 Governors Drive
Huntsville, AL 35807

SUBJECT: NRC INSPECTION NO. 99900902/95-02

Dear Mr Hyten:

This letter refers to the inspection led by Mr. K.R. Naidu, of this office on January 25-28, 1995. The inspection included a review of activities conducted at your facility in Huntsville, Alabama. At the conclusion of the inspection, the inspectors discussed their findings with Dr. C. Thibault and other members of your staff identified in Section 4 of the enclosed report.

Areas examined during the inspection and our findings are discussed in the enclosed report. The inspectors reviewed the actions taken to correct a violation and nonconformances identified in Inspection Report 99900902/92-01, and evaluated selected areas of the program established and executed by Wyle to implement the provisions of Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) for reporting defects and noncompliance, and Appendix B to 10 CFR Part 50. The inspectors also observed selected seismic tests being performed to qualify 4.16-kV switchgear for installation at Pacific Gas and Electric Company's Diablo Canyon Power Plant. Within these areas, the inspection consisted of an examination of procedures and representative documents, interviews with personnel, and observation by the inspectors.

Within the scope of this inspection, we found no instance where you failed to meet NRC requirements. Therefore, no response to this letter is required.

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

Sincerely,

A handwritten signature in black ink, appearing to read "Robert M. Gallo", is written over the typed name.

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

Docket No.: 99900902

Enclosure: As stated

cc: See Next Page

cc: Mr. Gregory M. Rueger
Nuclear Power Generation
B 14A
Pacific Gas & Electric Company
77 Beale Street, Room 145
P.O. Box 77000
San Francisco, CA 94106

Mr. D. R. Michaud
Products Manager
NTS
533 Main Street
Acton, MA 01720

Mr. J. L. Bachmann
Project Manager
Power Distribution Services, Inc.
9870 Crescent Drive
West Chester, Ohio 45069

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

REPORT NO.: 99900902/95-02

ORGANIZATION: Wyle Laboratories
7800 Governors Drive
Huntsville, Alabama 35807

ORGANIZATIONAL CONTACT: C. Thibault, Director,
Nuclear Engineering Services Division
(205) 837-4411 ext. 255

NUCLEAR INDUSTRY ACTIVITY: Third party qualification of equipment manufactured by others for use in nuclear power plants.

INSPECTION DATES: January 25-28, 1995

LEAD INSPECTOR:

Kamalakar R. Naidu, Team Leader
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

2/2/95
Date

OTHER INSPECTORS:

Stephen D. Alexander, Reactor Engineer, VIS/TSIB
Rogelio Mendez, Reactor Engineer, VIS/TSIB

REVIEWED BY:

Gregory C. Cwalina, Chief, VIS
TSIB

2/3/95
Date

APPROVED BY:

Robert M. Gallo, Chief, TSIB

2/6/95
Date

1 INSPECTION SUMMARY

During this inspection, the inspectors reviewed the actions taken by Wyle to correct one violation and three nonconformances identified in Inspection Report 92-01, and found them acceptable. The inspectors also observed the implementation of Wyle's quality assurance program in selected areas during the seismic qualification tests of 4.16-kV circuit breakers and drawout cubicles supplied by National Technical Systems (NTS), Acton, Massachusetts. Pacific Gas and Electric Company (PG&E) issued a purchase order to NTS to supply the assembly for installation at its Diablo Canyon Power Plant to replace existing circuit breakers. Power Distribution Services (PDS), West Chester, Ohio, manufactured vertical lift drawout cubicles containing the 4.16-kV circuit breakers (with Sulfur Hexafluoride (SF-6) interrupters) manufactured by Yasgawa Company, Ltd., of Japan.

The inspection basis consisted of 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"
- Wyle Quality Assurance Manual and Procedures

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Violation 99900902/92-01-01 (Closed):

Contrary to Section 21.21 of 10 CFR Part 21, Wyle Scientific and Systems Group Laboratories (Wyle) failed to incorporate the new time limits and other new requirements that were specified in the revision of 10 CFR Part 21, that became effective October 1991.

The inspectors reviewed Wyle's Quality Directive (QD) No. XIX-1, Revision A, "Reporting Defects and Noncompliance Per 10 CFR Part 21," and identified the need for some clarification. The Wyle QA manager revised the procedure before the conclusion of the inspection. The inspectors reviewed the final draft of QED No. XIX-1, Revision B, dated January 31, 1995, and determined it to be acceptable.

2.2 Nonconformance 99900902/92-01-02 (Closed):

Contrary to Criterion I of Appendix B to 10 CFR Part 50, Wyle failed to adequately delineate in writing the duties and authority of the QC Level I and Level II inspectors in its QD II-2, "Engineering, Verification, Quality Assurance and Testing Personnel Qualification Program," dated August 1, 1988.

As stated in its response letter of September 30, 1992, Wyle revised the following Nuclear Environmental Qualification (NEQ) procedures that delineate the responsibilities of Levels I and II Third Party Qualification (TPQ) inspectors: NEQ No. 403, Revision C, "TPQ Receiving and Component

Homogeneity," dated May 6, 1994, and NEQ No. 409, Revision D, "The Wyle Third Party Qualification Process," dated May 6, 1994. Additionally, Wyle revised QD II-2. Revision E of QD II-2, dated November 15, 1993, delineates the general requirements of Levels I, II, and III personnel who perform functions in the disciplines of engineering, verification, quality assurance and testing.

The Wyle response stated that on May 22, 1992, personnel from TPQ Engineering, Inspection, and Quality Assurance and Testing were given training on the duties and authority of the Quality Control Levels I and II inspectors. Wyle staff informed the inspectors that the training was verbal. On the basis of interviews with members of Wyle staff and a review of some of their personal notes, the inspectors concluded that the reported training had been completed, but not documented. Before the conclusion of the inspection, the Director of Nuclear Engineering Services repeated and documented the training for his entire staff on the following subjects:

- Duties and authority of TPQ Verification Level I and II Inspectors.
- NEQ 404, Qualification by Similarity.
- Timeliness of issuing a certificate of conformance.
- Requirement to issue a Nonconforming Material Report (NMR) when components are observed to be nonconforming and are rejected.

2.3 Nonconformance 99900902/92-01-03 (Closed):

Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 5, "Instructions, Procedures and Drawings," of Wyle's quality assurance manual (QAM), two examples were identified where Wyle's procedure did not ensure that activities affecting quality were adequately prescribed in QA program documents.

During this inspection, following Wyle's formal written response to this nonconformance that had been forwarded to the NRC in a Wyle letter of September 30, 1992, the inspectors reviewed Wyle's implementation of corrective actions and preventive measures described in the letter. The inspectors found Wyle's corrective actions and preventive measures acceptable with the exception of documentation of retraining that was reported completed. As stated previously, before the conclusion of the inspection, the Director of Nuclear Engineering Services repeated and documented the training.

2.4 Nonconformance 99900902/92-01-04 (Closed):

Contrary to 10 CFR Part 50, Appendix B, Criterion V, Wyle personnel did not follow documented procedures. Three examples were cited.

The inspectors reviewed the corrective action taken by Wyle as outlined in its formal written response to this nonconformance dated September 30, 1992, and found it acceptable.

3 INSPECTION FINDINGS

3.1 Entrance and Exit Meeting

During the entrance meeting on January 25, 1995, the NRC inspection team discussed the scope of inspection, the areas to be reviewed, and established the persons to contact within Wyle management and staff. During the exit meeting on January 27, 1995, the NRC inspection team summarized its findings and concerns. Persons contacted during this inspection are identified in Section 4.

3.2 Review of Procurement Documents

Pacific Gas and Electric (PG&E) issued purchase orders (PO) Nos. 068497 and 068517 to National Technical Systems (NTS) for the supply of vertical lift drawout cubicles with 4.16-Kv, 350-MVA (million volt-ampere) circuit breakers manufactured by Yasgawa Company, Ltd., of Japan, for installation at the Diablo Canyon Power Plant (DCPP). The drawout cubicles are designed to replace the existing drawout cubicles that were made for GE AM-4.16, 250-MVA circuit breakers. PG&E selected NTS as the project manager to oversee the design, manufacture and qualification of the drawout cubicles on which the Yasgawa circuit breakers and other auxiliary equipment will be mounted for Class 1E applications. Power Distribution Services, Inc. (PDS), was selected to manufacture the drawout cubicles. NTS issued PO No. 37796, and supplement 37796A, dated June 28 and November 28, 1994, respectively, to Wyle to subject three 4.16-Kv drawout cubicles containing Yasgawa 4.16-Kv circuit breakers to triaxial seismic multi-frequency random tests. The tests were designed to simulate plant specific seismic conditions in accordance with NTS Test Procedure No. 60431-95N-ST, "Triaxial Seismic Multi-frequency Random Test on 1200/2000 Amp SF-6 Conversion Breaker for Use in the Diablo Canyon Site for Pacific Gas & Electric Company." The inspectors reviewed the NTS PO and its supplement, which were designated safety-related services, and determined that they required compliance with the following:

- A quality assurance program meeting the requirements of 10 CFR Part 50, Appendix B.
- Wyle's reporting requirements to meet 10 CFR Part 21; Wyle to notify NTS if a reportable condition is observed.
- Requirements of MIL-STD-45662A for the calibration of measuring and test equipment.
- Wyle's responsibility for notifying NTS of any nonconforming items and actions taken to remedy defects, or suspected defects found in the switchgear during testing.

The inspectors did not identify adverse findings in this area.

3.3 Review of NTS Test Procedure

The inspectors reviewed NTS's Test Procedure No. 60431-95N-5T and determined that it contained the following:

- a. The test specimen was described as follows:

The test specimen consisted of three GE stationary metal-clad cubicles similar to those that are installed at the DCP. Each stationary cubicle was equipped with a PDS-fabricated, vertical lift drawout cubicle on which a Yasgawa 350-MVA interrupting capacity circuit breaker was mounted and suitably modified to replace the existing GE Magne-Blast 250-MVA circuit breaker. Two of the three 4.16-Kv circuit breakers in the drawout cubicles were rated for 1200 amperes (A), and one was rated for 2000 A continuous current. PDS also mounted auxiliary equipment, including a potential transformer, miscellaneous relays, terminal blocks, dummy loads in the stationary cubicles to simulate the switchgear installed at DCP.

- b. The steps to seismically qualify the drawout cubicles and breakers were:

A visual inspection and functional tests were performed to obtain baseline information. The seismic test procedure and test sequence addressed the various positions and operations of the circuit breaker during six operating basis earthquakes (OBEs) and one safe shutdown earthquake (SSE). Full scale shock spectrums for the test runs and locations were specified: i.e., vertical, side to side, and front to back for OBE, SSE, and fragility levels.

The inspectors did not identify adverse findings in this area.

3.4 Review of Wyle Test Documentation

The inspectors reviewed the test control record for the qualification of DCP's 4.16-kV drawout cubicles and breakers prepared by Wyle and determined that it reflected the NTS test procedure requirements and included the following:

1. Receipt inspection of the 4.16-kV switchgear shipped from PDS. This step had been completed. Wyle's receipt inspection documented minor shipping damage to the side of one of the stationary cubicles.
2. Test Set-Up and Equipment Sheet.
3. Resonance search.
4. Qualification Level (Random Multi-Frequency (RMF)) tests for the OBE and SSE.

5. Fragility Level tests.
6. Post Test Review (Data Review).

Wyle QA imposed hold points on Items 1, 2, and 6. A hold point requires a QA representative to witness and acknowledge the satisfactory completion of the line item. The inspectors observed that a QA inspector signed the hold point for item 1, concurring with the identification of the shipping damage and completion of the receipt inspection.

The inspectors did not identify adverse findings in this area.

3.5 Observation of Seismic Qualification Tests

The inspectors noted that 42 response accelerometers were attached to the test specimen (stationary and drawout cubicles with breakers) at 14 locations. Records indicated that NTS had performed operability tests on the various auxiliary equipment mounted on the switchgear test specimen to establish the baseline conditions. PDS had also manufactured a control panel to operate the circuit breakers from a remote location during the tests.

Wyle welded a structural steel frame to the triaxial shaker table and the test specimen was welded to the frame to simulate the switchgear mounting at DCPD.

Contact chatter detection devices to monitor the chatter of the auxiliary contacts of the breaker and selected relays were connected.

The inspectors observed a total of seven tests conducted on the test specimen. Tests 1, 2, and 3 consisted of low level (approximately 0.2g) sine sweeps from 1 to 100 Hz and back to 1 Hz in each orthogonal axis to establish major resonances of the test specimen. Tests 4, 5, 6, and 7 consisted of multi-frequency random tests to simulate an OBE. The inspectors observed that NTS personnel were documenting the anomalies and that PG&E personnel will participate in their evaluation. The inspectors observed that the tests were performed in accordance with the approved procedures and did not identify any adverse findings in this area.

3.6 Review of PDS Instrument Calibration

The inspectors reviewed the calibration data of the instruments used during the seismic testing of the breakers and found no problems with instrumentation involved directly with the breaker qualification tests; however, the inspectors noted potential problems with the use of measuring and test equipment (M&TE) by NTS/PDS.

On January 25, 1995, the inspectors observed NTS/PDS using a Multi-Amp test instrument with a January 11, 1995, calibration due date. The Multi-Amp instrument was used for verifying protective relay baseline data (obtained at PDS) prior to and upon completion of the breaker seismic tests. The NTS/PDS representatives informed the inspectors that they had discussed this issue with PG&E and obtained their approval for using M&TE with an expired calibration date because it would not affect the results of the seismic

testing of the breakers. The NTS/PDS representatives further stated that the relay data were for information only and not directly related to the seismic testing of the breakers. However, the inspectors were concerned because NTS testing procedure No. 60431-95N-ST stated that the relays were to be calibrated to specified settings. The inspectors also noted that this Multi-Amp test instrument was not the same as the instrument used for taking the baseline test data.

In addition, the inspectors reviewed the calibration data of 18 protective relays taken using the Multi-Amp set with the expired calibration and observed that 5 of the relays were outside the tolerances specified in the test procedure. All 18 protective relays had been originally set to within the tolerances specified in the test procedure. The inspectors could not determine whether the Multi-Amp unit with the expired calibration date used at the test site caused the relays to appear to be out of calibration, whether there was a drift in the relay set points, or whether the Multi-Amp test equipment used for the baseline data was out of calibration.

The relay calibration issue does not necessarily invalidate the seismic testing of the circuit breaker and drawout cubicle design; although, excessive relay contact chatter experienced during the testing severely hampered the qualification effort. Nevertheless, this issue will be re-examined if future activities by NTS or PDS utilize the relay information.

4 PERSONS CONTACTED

Wyle Laboratories

	T. Brewington	Director of Nuclear Services, International
	D. Compton	Test Technician
	G. Duthi	Test Technician
*†	R. Francis	Manager, Electrical Equipment Retrofit
	S.D. Hyten	Senior Vice President
*†	J.N. Matzkiw	Manager, TPQ Components & Production
	R.B. Pinkerton	Senior Test Supervisor
*†	E.R. Schum	Manager, EQ/TPQ Assemblies
†	T. Stinson	Director of Administration
	D. Smith	Manager, Nuclear Qualification Testing
	P. Stover	Test Technician
*†	C. Thibault	Director, Nuclear Engineering Services
*†	R.G. Thomas	Manager, Quality Assurance
	P. Wadsworth	Test Technician

* Attended the entrance meeting on 1/25/95

† Attended the exit meeting on 1/27/95

National Technical Systems

M. Lilly	Quality Assurance
D.R. Michaud	Products Manager

Pacific Gas & Electric Company

M. Basu	Senior Electrical Engineer
S.J. Float	Electrical Engineer
M.R. Khan	Consulting Mechanical Engineer
T.W. Packy	Lead Auditor
B. Supremo	Electrical Engineer

Power Distribution Services, Inc.

J.L. Bachmann	Project Manager
J.E. Bachmann	Quality Assurance
B. Kogin	Project Engineer
D.R. Robling	Project Engineer

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

<u>Identifier</u>	<u>Title</u>
Information Notice 95-02	Problems with General Electric CR2940 Contact Blocks in Medium-Voltage Circuit Breakers
Information Notice 95-12	Potentially Nonconforming Fasteners Supplied by A&G Engineering II, Inc.
Information Notice 95-19	Failure of Reactor Trip Breaker to Open Because of Cutoff Switch Material Lodged in the Trip Latch Mechanism
Information Notice 95-20	Failures in Rosemount Pressure Transmitters Due to Hydrogen Permeation Into the Sensor Cell

CORRESPONDENCE RELATED TO VENDOR ISSUES

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

January 19, 1995

Mr. Joseph M. Tate
General Manager
ABB Service Company
Regional Service Center
5311 Commerce Parkway West
Cleveland, OH 44130

SUBJECT: ABB RESPONSE TO A NOTICE OF VIOLATION (NOV) AND NONCONFORMANCE (NON)
CONTAINED IN NRC INSPECTION REPORT 99901281/94-01

Dear Mr. Tate:

This letter addresses the ABB Service Inc. (ABB), January 5 and 13, 1995, response letters to the U.S. Nuclear Regulatory Commission (NRC) regarding the Notices of Violation (NOV), Nonconformance (NON), and unresolved items issued to ABB on December 5, 1994, with NRC Inspection Report 99901281/94-01.

The NRC review of your January 5, 1995, NOV response and discussions with your Mr. John Webb on January 11, 1995, found that the ABB's basis for disputing Violation 94-01-01 appeared reasonable and the NRC staff has withdrawn the violation.

The NRC review of your January 5 and 13, 1995, responses to the NON and Unresolved Items found that ABB's statements were responsive to our concerns. We will review the implementation of your corrective actions for the Nonconformances and Unresolved Items during a future NRC inspection to determine the adequacy of your corrective actions and ensure that full compliance has been achieved and will be maintained.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter will be placed in the NRC Public Document Room. Please have your staff contact Mr. Kamal Naidu (301) 504-2980 if you have any comments or questions.

Sincerely,

A handwritten signature in black ink, reading "Robert M. Gallo", is positioned above the typed name.

Robert M Gallo, Chief
Special Inspection Branch
Division of Technical Support
Office of Nuclear Reactor Regulation

Docket No. 99901281

cc: See attached list

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

February 7, 1995

Mr. Richard B. Kelley, Vice President
Power Generation Services
2146-B Flintstone Drive
Tucker, Georgia 30084-3101

SUBJECT: REQUEST FOR INFORMATION: 10 CFR PART 50, APPENDIX B, APPLICABILITY
FOR EXPORT TRADING FIRMS

Dear Mr. Kelley:

By letter dated November 22, 1994, you requested the U. S. Nuclear Regulatory Commission (NRC) to provide a ruling, accompanied by comments, precautions, procedures, and references, for the applicability of Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) to a U. S. commercial firm, representing a U. S. manufacturer, where the commercial firm is an export trading company involved with only commercial aspects of transactions licensed for export. In your letter, you asked the following specific questions.

1. Can the commercial firm solicit safety related orders in the name of the manufacturer?
2. Can the commercial firm accept orders from customers issued in the name of the manufacturer and re-issue the same order to the manufacturer unaltered, except for financial terms of sale?
3. Can the commercial firm make export shipping arrangements for safety related goods shipped directly from the manufacturer to the customer's import broker?

On December 20, 1994, Messrs. Larry Campbell, Robert Gramm, and Uldis Potapovs of this office contacted you to obtain additional information and the specific circumstances relating to your questions. The following example was discussed: A nuclear power plant owner in Spain places an order for ASME Section III, Class I, nuclear grade pipe (and invokes the requirements of Appendix B to 10 CFR Part 50) with a commercial firm in Spain. This commercial firm then places an order with a U. S. import/export commercial firm. The U. S. firm then places the order (invoking the requirements of Appendix B to 10 CFR Part 50 and the ASME Section III Code) with a U. S. manufacturer accredited by ASME for the manufacture and supply of nuclear grade piping. During this conversation, you raised the question: When a commercial import/export firm, located in another country and representing a foreign commercial nuclear power plant owner, places an order with a commercial import/export firm in the U. S. and the order invokes the requirements of Appendix B to 10 CFR Part 50 and the ASME Section III Code, what NRC requirements and regulations apply to this U. S. commercial firm?

R. Kelley

-2-

NRC RESPONSE

Although contractually the U. S. commercial firm may have an obligation to meet the quality and technical requirements invoked by the foreign commercial firm, the manufacturing and supply of safety related items for use by an owner of a nuclear power plant in another country is not within the jurisdiction of the NRC. Potential enforcement actions by the NRC, resulting from noncompliance to Appendix B to 10 CFR Part 50 or the ASME Section III Code, would not be appropriate for items supplied under the circumstances described in your letter and supplemented in your December 20, 1994, conversation with the NRC staff.

Should you have any further questions regarding this matter, please feel free to contact Mr. Larry L. Campbell at (301) 504-2976 or Mr. Robert Gramm at (301) 504-1010.

Sincerely,

Robert A. Gramm for /

Suzanne C. Black, Chief
Quality Assurance and Maintenance Branch
Division of Technical Support
Office of Nuclear Reactor Regulation

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. 1

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Licensee Contractor and Vendor Inspection
Status Report
Quarterly Report
January - March 1995

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1995

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6. TYPE OF REPORT

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January -
March 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 Technical Support
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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 January through March 1995.

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