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Vol. 19, No. 3

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
July – September 1995

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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

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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 July 1995 through September 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

September 14, 1995

Mr. Kenneth R. Shaw, President
Continental Disc Corporation
3160 West Heartland Drive
Liberty, MO 64068

SUBJECT: NRC INSPECTION NO. 99901287/95-01

Dear Mr. Shaw:

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection of your facility at Liberty, Missouri, conducted by Messrs. Bill Rogers and Robert Pettis of this office on June 27 through 29, 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 control of purchased material and services, supplier audits, manufacturing control and a review of your program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations.

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 control of purchased material, equipment, and services. 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.

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

Sincerely,

Robert M. Gallo

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

Docket No. 99901287

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99901287/95-01

NOTICE OF NONCONFORMANCE

Continental Disc Corporation
Liberty, Missouri

Docket No.: 99901287

Based on the results of an NRC inspection conducted at the Liberty, Missouri, facility of Continental Disc Corporation on June 27-29, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

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

Contrary to the above, Continental Disc Corporation Quality Assurance Manual, General Revision 1, issued April 1, 1990, did not address the quality requirements of non-pressure boundary parts, exempt from American Society of Mechanical Engineers (ASME) Section III, Division 1, NCA-1275, "Rupture Disc Devices," used in the manufacture of nuclear safety-related rupture discs, an activity affecting quality. (95-01-01)

II. Criterion VII of Appendix B to 10 CFR Part 50, "Control of Purchased Material, Equipment and Services," states, in part, that measures shall be established to assure that purchased material, equipment, and services conform to the procurement documents and include provisions for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery.

Contrary to the above, Quality Control Instruction No. 1014, Revision C, dated March 20, 1995, stated in paragraph 8.2 of Section 8.0, "Supplier Approval Criteria," that suppliers could be qualified on the single basis of "completion of a Quality System Questionnaire." Continental Disc Corporation's method of qualification for certain suppliers was limited to the supplier completing a Quality System Questionnaire which did not provide objective evidence of quality to demonstrate that purchased material conformed to procurement documents.

Consequently, as documented by the review of the qualification activities for Teledyne Rodney, Metal Goods, Castle Metals, Joseph T. Ryerson & Son, and Coulter Steel & Forge, Continental Disc Corporation did not adequately qualify suppliers furnishing materials or services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders. Additionally, as documented by review of the qualification activities for Teledyne Rodney, Metal Goods, and Sherry Laboratories, Continental Disc Corporation accepted material test reports from suppliers furnishing material and test services without a basis for accepting such documentation and subsequently certified to its customers that the material complied with the purchase order requirements. (95-01-02)

Enclosure 1

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

Criterion VII of Appendix B to 10 CFR Part 50, "Control of Purchased Material, Equipment and Services," states, in part, that measures shall be established to assure that purchased material, equipment, and services conform to the procurement documents.

Paragraph 4.5 of Section 4.0, "Purchasing Documents," of the Continental Disc Corporation Quality Assurance Manual stated that when regulatory requirements, design bases or other requirements necessary to assure adequate quality are incorporated in a Continental Disc Corporation customer's purchase order, these requirements shall be referenced in the purchase order to Continental Disc Corporation suppliers and apply to supplier and sub-tier supplier performance.

Continental Disc Corporation Quality Control Instruction No. 1003, "Purchasing and Supplier Quality Assurance Policy," Revision F, dated March 20, 1995, stated in Paragraph 6.7 of Section 6.0, "Supplier Quality Policy," that suppliers shall flow down all purchase order requirements to any authorized sub-tier suppliers.

Contrary to the above, as documented by review of CDC purchase orders No. 49912 to Teledyne Rodney, dated May 18, 1995; No. 49874 to Castle Metals, dated May 16, 1995; No. 47600 to Joseph T. Ryerson, dated October 26, 1994; No. 47107 to Coulter Steel & Forge, dated September 20, 1994; and No. 49909 to Metal Goods, dated May 18, 1995; Continental Disc Corporation did not impose customer required 10 CFR Part 50, Appendix B, quality requirements in purchase orders to its suppliers furnishing materials or services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders. (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 Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance: (1) a description of the steps that have been or will be taken to correct these items; (2) a description of the steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed.

Dated at Rockville, Maryland
this 14th day of September, 1995

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

REPORT NO.:	99901287/95-01
ORGANIZATION:	Continental Disc Corporation 3160 Heartland Drive Liberty, Missouri 64068
ORGANIZATIONAL CONTACT:	Dean Dachenhausen, Director Quality Assurance
NUCLEAR INDUSTRY ACTIVITY:	Continental Disc Corporation provides rupture discs and rupture disc holders to the nuclear industry.
INSPECTION DATES:	June 27 - 29, 1995

TEAM LEADER: Bill Rogers 8/29/95
Bill H. Rogers, Date
Vendor Inspection Section
Special Inspection Branch

OTHER INSPECTORS: Robert L. Pettis, Jr.
Vendor Inspection Section
Special Inspection Branch

REVIEWED BY: Gregory C. Cwalina 8/30/95
Gregory C. Cwalina, Chief, Date
Vendor Inspection Section
Special Inspection Branch

APPROVED BY: Robert M. Gallo 9/14/95
Robert M. Gallo, Chief, Date
Special Inspection Branch

Enclosure 2

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspectors evaluated Continental Disc Corporation's (CDC) 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, and production control. The team also reviewed CDC's program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR Part 21).

1.1 Violations

1.1.1 Contrary to 10 CFR 21.31, which requires that purchase orders to suppliers specify that the provision of 10 CFR Part 21 apply when applicable, CDC Quality Control Instruction No. 1003, "Purchasing and Supplier Quality Assurance Policy," Revision F, dated March 20, 1995, (which required that suppliers shall flow down all purchase order requirements to any authorized sub-tier suppliers), and Quality Control Instruction No. 1020, "Nuclear Safety Related Materials," Revision F, dated February 6, 1990, (which required that compliance with 10 CFR Part 21 shall be mandatory on all shop orders and all purchase orders to suppliers when imposed by the customer purchase order), CDC did not impose the requirements of 10 CFR Part 21 in its purchase orders to suppliers furnishing material or services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders to fill customer purchase orders from licensees specifying that the requirements of 10 CFR Part 21 applied. (Non-Cited violation)

1.2 Nonconformances

1.2.1 Contrary to Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings," the CDC Quality Assurance Manual, General Revision 1, issued April 1, 1990, did not address the quality requirements of non-pressure boundary parts, exempt from American Society of Mechanical Engineers (ASME) Section III, Division 1, NCA-1275, and used in the manufacture of nuclear safety-related rupture discs. (95-01-01)

1.2.2 Contrary to Criterion VII of Appendix B to 10 CFR Part 50, "Control of Purchased Material, Equipment and Services," and Sections 7.2 and 7.8 of the CDC Quality Assurance Manual, CDC did not adequately qualify suppliers furnishing materials or services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders. CDC's method of supplier qualification was limited to the supplier completing a Quality System Questionnaire which did not provide sufficient objective evidence to demonstrate that the supplier is effectively implementing its quality program.

Additionally, CDC did not independently verify material test reports and certificates submitted to them from these suppliers. Such documentation was supplied by CDC to its customers purchasing nuclear safety-related rupture discs and rupture disc holders. (95-01-02)

1.2.3 Contrary to Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings;" Criterion VII of Appendix B to 10 CFR Part 50, "Control of Purchased Material, Equipment and Services;" Paragraphs 4.5 and 4.6 of Section 4.0, "Purchasing Documents," of the CDC Quality Assurance Manual; Paragraph 6.7 of Section 6.0, "Supplier Quality Policy of CDC Quality Control Instruction No. 1003, Revision F, dated March 20, 1995; and Paragraph 5.1 of CDC Quality Control Instruction No. 1020, Revision F, dated February 6, 1990; CDC did not impose customer required 10 CFR Part 50, Appendix B, quality requirements in purchase orders to its suppliers furnishing materials or services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders. (95-01-03)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of CDC.

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance and Exit Meetings

During the entrance meeting on June 27, 1995, the NRC inspectors discussed the inspection scope and developed general information about CDC's products and activities. During the exit meeting on January 29, 1995, the NRC inspectors discussed their findings and observations with CDC management.

3.2 Background

CDC supplies rupture discs and rupture discs holders to the nuclear industry as well as for many general industrial applications. Rupture discs, used to provide instantaneous relief of overpressure conditions, are manufactured in a variety of designs of various metals and combinations of metals and teflon liners. The rupture discs can also be designed to resist vacuum, be non-fragmenting, and provide indication when rupture has occurred.

3.3 10 CFR Part 21 Program

The inspectors reviewed CDC's 10 CFR Part 21 program including procedures and implementation. CDC had not identified any deviations in products that they had supplied and consequently had not performed any evaluations in accordance with their 10 CFR Part 21 program.

The inspectors reviewed CDC's 10 CFR Part 21 implementing procedure, Quality Control Instruction No. 1013, "Compliance With 10-CFR-21," Revision A, dated March 30, 1995. The procedure was adequately written with the exception of several instances were the terms deviation and defect were inappropriately interchanged. The use of these terms, and their definitions in 10 CFR Part 21, and other minor procedural discrepancies were discussed with CDC who indicated that the procedure would be modified.

The inspectors reviewed CDC's posting as required by 10 CFR 21.6 and determined it to be in accordance with the regulation. CDC had posted Section 206 or the Energy Reorganization Act and a notice which described the regulations and applicable procedures, including the name of the individual to whom the reports could be made, and where the regulations and procedures could be examined.

The inspectors identified one area where implementation of the CDC 10 CFR Part 21 program was not adequate. CDC Quality Control Instructions required that suppliers flow down all purchase order requirements to any authorized sub-tier suppliers and that compliance with 10 CFR Part 21 shall be mandatory on all shop orders and all purchase orders to suppliers when imposed by the customer purchase order. The inspectors identified several examples where CDC did not comply with these requirements. This issue is discussed in detail in section 3.11 of this report.

3.4 Internal Audits

Internal audits were required to performed by Section 18, "Audits," of the CDC Quality Assurance Manual. The inspectors reviewed the implementing procedure Quality Control Instruction No. 1001, "Quality Assurance Internal Audit," Revision H, dated December 2, 1994. The procedure required that internal audits be performed annually with the frequency to be shortened if determined necessary based on previous results, corrective actions, nonconformances or customer feedback. The procedure required the Director of Quality Assurance to prepare the audit plans, train auditors, assign audit personnel, and evaluate the results. An audit plan and audit checklist were used for the internal audits. Corrective actions were to be provided by the managers of the department where the finding was identified.

The inspectors reviewed the reports for the internal audits which had been performed December 20-21, 1994, June 20-21, 1994, November 22-23, 1993, and December 16-21, 1992. The audit reports specified the areas reviewed for the specific audits which included material control, material traceability, manufacturing, document control, design control, and work in process and did not identify any findings. The inspectors concluded that the audit reports documented a thorough process which was in accordance with the requirements of the implementing procedure.

3.5 Rupture Disc Production Process

The inspectors reviewed the applicable procedures, and their implementation, for selected portions of the rupture disc manufacturing process. Rupture discs come in a variety of combinations including metal rupture discs, teflon seals, and those with backpressure supports. The inspectors reviewed Quality Control Instructions, Manufacturing Procedures and Production Operating Procedures applicable to the production of rupture discs and disc holders and observed portions of the production process.

Quality Control Instruction No. 1023, "Control of Rupture Disc Production Processes," Revision A, dated June 4, 1991, described the control and documentation procedures applicable to rupture discs, the conditions under which rupture disc manufacturing and testing were performed, and the documentation maintained to provide evidence of compliance with the CDC quality assurance program. The procedure included production personnel training and qualification, work instructions, material purchases, material controls, production flow, lot identification control, shipping department control, final inspection, packing, delivery, and documentation.

Manufacturing Procedures (MP) were used to provide instruction on the specific steps of rupture disc manufacturing. MPs included instructions on the order of precedence of referenced documents (shop order having the highest precedence), testing of rupture discs using teflon liners, and preparing for the final burst tests. The inspectors discussed the use of MPs with CDC personnel and observed the manufacture of several rupture discs.

Production Operating Procedures (POP) provided instruction on additional activities related to the production of rupture discs such as parts forming, parts etching, and teflon forming. For example, POP 1000, Initial Issue, dated January 26, 1995, provided specific quality requirements on the process of teflon forming such as no burn holes or pin holes were allowed in the teflon seals. The inspectors observed a successful demonstration of teflon forming and the subsequent check for pin holes.

CDC used a rupture disc burst test to demonstrate the quality of each lot of rupture discs manufactured. CDC made rupture discs of varying lot sizes based on the number of rupture discs in the order. Quality Control Instruction No. 2000, "Final Lot Rupture Disc Burst Test Procedure," Revision E, dated February 3, 1995, required a sample of all lots manufactured to be burst tested (a destructive test) to verify that the rupture discs manufactured in the lot would perform as designed. Lot size was based on Mil-Std-105E and CDC MP 2003 and required a minimum sample size of two burst tests for any order or lot manufactured. For example, if the order required one rupture disc, CDC would manufacture three rupture discs and burst test two of the three. In addition, if ordered quantities were broken down into smaller lots for control purposes the burst test sample size would be based on the smaller lots. The sample to be burst tested was selected by a burst test witness at random from the rupture discs or rupture disc components and were to include all rupture disc components. The inspectors observed the successful burst tests of several rupture discs and also witnessed a failure of rupture disc material (premature rupture) where the material was rejected and appropriately dispositioned.

The inspectors concluded, based on a review of the applicable procedures, discussion with personnel, and observations of portions of the production process, that the CDC quality assurance program was well implemented in the production area and that production testing was adequate to ensure the quality of the delivered rupture discs.

3.6 Customer Audits of CDC

The NRC inspection team reviewed CDC document "Customer On-site Quality Audits/ Approvals," dated June 12, 1995. The document listed all customer audits performed at CDC, including the audit criteria, audit date and results. CDC had been recently audited by the following customers: Entergy on June 3, 1992, Astro Nuclear/Dynamics on June 14, 1994, Public Service Electric & Gas on June 7, 1995, Commonwealth Edison on November 15, 1991, and Philadelphia Electric on February 10, 1992.

3.7 Customer Purchase Orders to CDC

The NRC inspection team reviewed selected customer purchase orders to CDC which specified the requirements of 10 CFR Part 50, Appendix B, and 10 CFR Part 21 for nuclear safety-related rupture discs and holders since 1990. The purpose of the purchase order review was to determine if CDC had properly implemented its quality assurance program, particularly in the area of raw material supplier qualification. The inspectors reviewed the following purchase orders:

3.7.1 Commonwealth Edison Company

Commonwealth Edison Company (CECo) ordered six, safety-related, 6-inch rupture discs on purchase order No. 4T8711, dated March 21, 1994, for its LaSalle Station No. 1. The purchase order requested certified material test reports (CMTR) and a Certificate of Compliance to the purchase order requirements which required compliance to 10 CFR Part 50, Appendix B, 10 CFR Part 21 and ASME Section III, Division 1, Class 2, 1974 Edition. The purchase order also stated in Statement 0013 that when subtier vendors are utilized, the appropriate quality assurance program requirements shall be incorporated in the procurement documents.

CDC processed the order using material furnished by Teledyne Rodney (CDC purchase order No. 41291) and an unidentified supplier (CDC purchase order No. 33535) which supplied an Inconel 600 vacuum support. The vacuum support material was sent to Metlab Testing Services, Inc., which verified on October 2, 1991, that the chemical analysis conformed to Inconel 600. CDC passed on the Metlab chemical analysis certificate (No. 91-6179) and a Material Test Certificate from Teledyne Rodney (No. 18141) under a CDC Certificate of Conformance which certified that the materials were furnished in strict accordance with the requirements and applicable specifications of the purchase order.

3.7.2 Niagara Mohawk Power Corporation

Niagara Mohawk Power Corporation (NMPC) ordered sixty, safety-related 1-inch rupture discs of various pressure and temperature ratings on purchase order No. 13874, dated October 14, 1993, for its Nine Mile Point, Unit 2, nuclear plant. The purchase order requested CMTRs for the disc material and a certificate of compliance to the purchase order requirements which required compliance to 10 CFR Part 50, Appendix B, 10 CFR Part 21, ASME Code Section III, Division 1, Class 2, 1980 Edition through the Winter 1980 addenda and for CDC to extend to sub-suppliers all appropriate technical and quality requirements.

CDC processed the order using material furnished by Teledyne Rodney (CDC purchase order not identified). CDC passed on a Material Test Certificate (No. 10706) from Teledyne Rodney under a CDC Certificate of Conformance which certified that the materials were furnished in strict accordance with the requirements and applicable specifications of the purchase order.

3.7.3 PECO Energy Company - Peach Bottom

PECO Energy Company (PECO) ordered eight, 16-inch, safety-related rupture discs on purchase order No. BW230572, dated August 9, 1994, for its Peach Bottom Atomic Power Station. The purchase order requested CMTRs and a certificate of compliance to the purchase order requirements. Additional purchase order requirements included CDC's compliance to 10 CFR Part 21, furnish materials in accordance with CDC's Quality Assurance Manual, Revision 1, dated April 1, 1990, ASME Code Section III, Division 1, Class 2, 1986 Edition through the Winter 1987 addenda and extend to lower tier suppliers all applicable quality assurance program requirements.

CDC processed the order using material furnished by Teledyne Rodney (CDC purchase order No. 42200) and Castle Metals, Inc. (CMI)(CDC purchase order not identified). CDC passed on a Material Test Certificate (No. 29099) from Teledyne Rodney and a Certificate of Test from CMI under a CDC Certificate of Conformance which certified that the materials were furnished in strict accordance with the requirements and applicable specifications of the purchase order.

3.7.4 PECO Energy Company - Limerick

PECO ordered a 16-inch, safety-related rupture disc on purchase order No. LS237030, dated February 2, 1994, for its Limerick Nuclear Generating Station. The purchase order requested CMTRs and a certificate of compliance to the purchase order requirements which included compliance to 10 CFR Part 50, Appendix B, 10 CFR Part 21, ASME Code Section III, Division 1, Class 2, 1986 Edition through the Winter 1987 addenda and to extend to lower tier suppliers all applicable quality assurance program requirements.

CDC processed the order using material furnished by Teledyne Rodney (CDC purchase order No. 41207) and Joseph T. Ryerson & Son (CDC purchase order No. 43240). CDC passed on a Material Test Certificate (No. 26736) from Teledyne Rodney and a Metallurgical Test Report (No. 50302) from North American under a CDC Certificate of Conformance which certified that the materials were furnished in strict accordance with the requirements and applicable specifications of the purchase order.

3.7.5 Georgia Power Company

Georgia Power Company (GPC) ordered two, safety-related 16-inch rupture discs on purchase order No. 60179000000, dated September 1, 1994, for its Edwin I. Hatch Nuclear Plant. The purchase order requested CMTRs and a certificate of compliance to the purchase order requirements which included compliance to 10 CFR Part 50, Appendix B, 10 CFR Part 21, ASME Code Section III, Division 1, Class 2, 1980 Edition through the Summer 1981 addenda and provide access to lower tier suppliers for quality assurance inspection or audit by GPC. CDC processed the order using material furnished by Teledyne Rodney (CDC purchase order No. 42000), CMI (CDC purchase order No. 46244) and Metal Goods (CDC purchase order No. 46331).

CDC passed on two Material Test Certificates (Nos. 27140 and 30547) from Teledyne Rodney and a Certificate of Test from Allegheny Ludlum Steel (through CMI) and a Test Report (No. 50302) from J&L Specialty Products Corporation (through Metal Goods) under a CDC Certificate of Conformance which certified that the materials were furnished in strict accordance with the requirements and applicable specifications of the purchase order.

3.7.6 Public Service Electric & Gas Company

Public Service Electric & Gas Company (PSE&G) ordered four, 12-inch, safety-related rupture discs on purchase order No. P2-0699501, dated June 9, 1994. The purchase order requested CMTRs and a certificate of compliance to the purchase order requirements which included compliance to 10 CFR Part 50, Appendix B, 10 CFR Part 21 and ASME Code Section III, Division 1, Class 2, 1977 Edition.

CDC processed the order using material furnished by Teledyne Rodney (CDC purchase order No. 41291) and CMI (CDC purchase order No. 42341). CDC passed on a Material Test Certificate (No. 16866) from Teledyne Rodney and a CMTR from Inco Alloys International (through CMI) under a CDC Certificate of Conformance which certified that the materials were furnished in strict accordance with the requirements and applicable specifications of the purchase order.

3.8 Classification of Components

ASME Code Section III, Division 1, NCA-1275, "Rupture Disc Devices," stated that the rupture disc holder was the only portion of the rupture disc considered part of the pressure boundary and therefore was the only component designated "Code material" and subject to material controls. The CDC Quality Assurance Manual did not address the quality requirements of exempt, non-pressure boundary parts. CDC stated that since the rupture disc itself was not considered a pressure boundary part under ASME Code, CDC had not classified the rupture disc material as safety-related. The inspection team pointed out that although the ASME Code exempts the disc from consideration as Code material, such non-pressure boundary safety-related items must be processed and controlled in accordance with the applicable requirements of 10

CFR Part 50, Appendix B. Although the rupture disc material is designed to fail (in a specified and predictable manner) materials are often chosen for use due to other considerations (such as resistance to corrosion) and could therefore have importance to the overall safety function of the rupture disc assembly. In addition, the rupture disc material is often specified by the customer on the purchase order and is therefore a technical requirement requiring verification and documentation. CDC's failure to properly classify material and components and accordingly the failure to take appropriate actions to verify that the material and components have been manufactured and controlled under an acceptable quality assurance program which has been properly implemented has been identified as Nonconformance 95-01-01.

NRC Information Notice (IN) 88-95, "Inadequate Procurement Requirements Imposed By Licensees On Vendors," dated December 8, 1988, discussed inadequate procurement requirements being imposed by licensees on vendors supplying components under the ASME Code which may result in the vendor's failure to implement critical portions of 10 CFR Part 50, Appendix B, quality assurance requirements. Specifically, the IN stated that compliance with ASME Section III satisfies 10 CFR Part 50, Appendix B, requirements for items covered by the Code, however, this is not sufficient to ensure that safety-related items exempt from Code requirements comply with 10 CFR Part 50, Appendix B. In addition, IN 90-03, "Malfunction Of Borg-Warner Bolted Bonnet Check Valves Caused By Failure Of The Swing Arm," dated January 23, 1990, discussed an event which occurred at the Comanche Peak Steam Electric Station (CPSES) where a 4-inch 150 pound BW/IP (formerly Borg-Warner Nuclear Valve Division) bolted bonnet swing check valve, installed in the service water system at the CPSES, Unit 1, exhibited excessive backleakage. The NRC concluded that BW/IP's classification of the swing arm as a non-pressure boundary valve internal, and exempt under the requirements of ASME III, resulted in the failure of BW/IP to impose nuclear quality assurance requirements on the swing arm manufacturer. Inadequate heat treatment by a supplier who was not on BW/IP's approved suppliers list was identified as the cause of the radial fracture of the swing arm. The disc separated from the swing arm which connected the disc and stud assembly to the clevis.

3.9 Qualification of Material Suppliers

The inspectors reviewed Section 7.2 of the CDC Quality Assurance Manual which stated that purchases shall be made from an approved suppliers list jointly maintained by the Purchasing Department and the Quality Assurance Department. The February 6, 1995, CDC approved suppliers list, defined five categories of suppliers, Codes I through V. The inspectors reviewed documentation for selected Code I (suppliers of metal and plastic raw materials used for rupture discs and rupture disc holders) and Code IV (suppliers of quality assurance services, calibration, and testing) suppliers.

Quality Control Instruction No. 1014, Revision C, dated March 20, 1995, stated in paragraph 8.2 of Section 8.0, "Supplier Approval Criteria," that Code I through Code IV suppliers may achieve quality approval status by one or more of the following means: 1) completion of a Quality System Questionnaire, 2) performance of an on-site quality survey, 3) quality certification provided by the supplier documenting use of an ISO 9000 standard, an ASME Quality System Certificate or authorization stamp, 4) the review and approval by the CDC Quality Department of the supplier's quality manual. Quality Control Instruction No. 1014 also stated that an implementation audit or survey of the supplier was not required for those suppliers which hold an ASME Quality System Certificate. The inspectors concluded that although "performance of an on-site quality survey," used alone could be an acceptable method of qualification, any of the other three methods, "completion of a Quality System Questionnaire," "quality certification provided by the supplier documenting use of an ISO 9000 standard, an ASME Quality System Certificate or authorization stamp," or "the review and approval by the CDC Quality Department of the supplier's quality manual," used alone, would not be adequate to ensure the effective implementation of a supplier's approved quality assurance program in order to supply safety-related material to be used without further verification.

Typical methods of qualifying companies for placement of safety-related, 10 CFR Part 21, purchase orders (material or services to be used without further verification) are 1) ASME supplier - inclusion on the approved suppliers list based on an ASME Quality System Certificate followed by a review of the supplier's quality assurance program with an associated implementation audit or 2) Non-ASME supplier - review of the supplier's quality assurance program with an associated implementation audit. In addition, for suppliers that would not accept safety-related, 10 CFR Part 21, purchase orders other means of ensuring quality could be used including test and measurement, source surveillance, commercial grade surveys, and performance history (used in combination with one or more of the other methods).

The inspectors reviewed the documentation for the qualification activities for several companies that CDC had purchased material and services from. These companies were Teledyne Rodney, Metal Goods, Castle Metals and Joseph T. Ryerson & Son. The suppliers were listed on CDC's approved suppliers list, dated February 6, 1995, as Code I suppliers, and had been used by CDC since 1977. All of these suppliers were qualified by CDC only on the basis of providing a satisfactory response to a CDC Quality System Questionnaire issued in January 1994 (with the exception of Teledyne Rodney who also had maintained an ISO 9000 quality assurance program). In addition, CDC had passed on to its customers material test reports and certifications submitted to them from Teledyne Rodney and Metal Goods without performing an independent verification of the basis of such documentation. Since the quality assurance programs of these suppliers were never verified through the performance of an implementation audit, survey or other appropriate means to objectively assess quality, CDC had no documented basis for accepting and supplying such material test reports to its customers and furnishing certification that CDC was in full compliance with all purchase order requirements.

The inspectors reviewed section 7.8 of the CDC Quality Assurance Manual which stated that pressure boundary nuclear safety-related materials shall be purchased only from suppliers holding current ASME authorizations or certificates. Paragraph 7.8.1 stated that CMTRs or certifications furnished by suppliers of pressure boundary nuclear safety-related materials shall be reviewed by quality assurance personnel for code compliance and adequacy. CDC Quality Control Instruction No. 1020, "Nuclear Safety Related Materials," Revision F, dated February 6, 1990, required in Paragraph 5.5 of Section 5.0, "Requirements," that purchase orders for all pressure boundary and flange bolting materials shall be issued only to approved suppliers holding current ASME stamps or Quality System Certificates for the material required. The NRC inspection team identified that only Coulter Steel & Forge appeared on CDC's approved suppliers list as a Code I supplier holding a Quality System Certificate and was the only supplier of rupture disc holder material. However, their qualification basis as a Code I supplier was solely based on a satisfactory response to a CDC furnished Quality System Questionnaire. The NRC inspection team discussed with CDC that possession of a Quality System Questionnaire is acceptable to place a supplier on the approved suppliers list for programmatic aspects of the suppliers quality assurance program, but prior to accepting material or services from the supplier, CDC would have to perform an implementation audit to verify that the supplier is effectively implementing its approved 10 CFR Part 50, Appendix B, quality assurance program.

CDC had used a testing laboratory (Sherry Laboratories) to analyze rupture disc holder material; however, CDC had not taken action to verify that the laboratory had an adequate quality assurance program which was being effectively implemented (see section 3.10).

CDC's failure to ensure that suppliers furnishing material used in the manufacture of nuclear safety-related rupture discs and rupture disc holders were qualified and effectively implementing the approved quality assurance programs has been identified as Nonconformance 95-01-02.

NRC IN 86-21, "Recognition Of ASME Accreditation Program For N Stamp Holders," dated March 31, 1986, discussed that the NRC's recognition applied only to the programmatic aspects of the ASME Accreditation Program and that licensees and their subcontractors were still responsible for ensuring that the supplier is effectively implementing its approved quality assurance program.

3.10 Qualification of Testing Laboratory

The NRC review identified that some material furnished by Teledyne Rodney and Metal Goods had been sent to Sherry Laboratories (formerly Metlab Testing Services, Inc.), for spectrographic examination. However Sherry Laboratories, a Code IV supplier listed on CDC's approved suppliers list since 1990, was qualified by CDC based on "performance," which is defined by CDC as historical satisfactory performance. However, the "Approval Criteria" section of the approved suppliers list for Code IV suppliers of quality assurance services, including calibration and testing, did not recognize the "performance" approach to supplier qualification (the approved suppliers list stated "testing laboratories have no pre-qualification requirements"). This is contrary to Paragraph 8.2 of Quality Control Instruction No. 1014, "Supplier Selection and Quality Assessment," Revision C, dated March 20, 1995, which did not recognize supplier "performance" as a means of supplier qualification.

In addition, CDC passed on to its customers material test reports and certifications submitted to them from Sherry Laboratories without performing an independent verification of the basis of the documentation. Since the quality assurance program had not been verified through the performance of an implementation audit, survey or other appropriate means to objectively assess quality, CDC had no documented basis for accepting and supplying such material test reports to its customers and furnishing certification that CDC was in full compliance with all purchase order requirements. CDC's failure to ensure that the supplier furnishing metallurgical testing services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders was qualified and effectively implementing an approved quality assurance program has been identified as a second example of Nonconformance 95-01-02.

3.11 Review of CDC Purchase Orders to Suppliers

The inspectors reviewed paragraph 4.5 of Section 4.0, "Purchasing Documents," of the CDC Quality Assurance Manual which stated that when regulatory requirements, design bases or other requirements necessary to assure adequate quality are incorporated in a CDC customer's purchase order, these requirements shall be referenced in the purchase order to CDC suppliers and apply to supplier and sub-tier supplier performance.

CDC Quality Control Instruction No. 1003, "Purchasing and Supplier Quality Assurance Policy," Revision F, dated March 20, 1995, required in Paragraph 6.7 of Section 6.0, "Supplier Quality Policy," that suppliers shall flow down all purchase order requirements to any authorized sub-tier suppliers. Additionally, CDC Quality Control Instruction No. 1020, "Nuclear Safety Related Materials," Revision F, dated February 6, 1990, required in Paragraph 5.1 of Section 5.0, "Requirements," that compliance with 10 CFR Part 21 shall be mandatory on all shop orders and all purchase orders to suppliers when imposed by the customer purchase order.

During the NRC's review of safety-related customer purchase orders to CDC, discussed in Section 3.7 of this report, it was noted that in all cases the customer purchase orders referenced that the provisions of 10 CFR Part 21 applied and that any material or items specified in the purchase order be supplied and certified in accordance with CDC's Quality Assurance Manual. Paragraph 1.4 of Section 1.0, "Introduction," of the CDC Quality Assurance Manual stated compliance to several recognized quality standards and specifications, including that of 10 CFR Part 50, Appendix B, and 10 CFR Part 21.

The NRC inspection team reviewed purchase orders to the suppliers identified above in Section 3.7 which had furnished material or services used by CDC in the manufacture of safety-related rupture discs or rupture disc holders. The suppliers included Teledyne Rodney, Castle Metals, Joseph T. Ryerson & Son, Coulter Steel and Forge and Metal Goods. In all cases the NRC review identified that the quality requirements imposed on CDC (10 CFR Part 50, Appendix B) in licensee purchase orders had not been passed down by CDC to its suppliers of safety-related materials and services. The following CDC purchase orders to the above suppliers were reviewed:

- CDC purchase order No. 49912 to Teledyne Rodney, dated May 18, 1995, ordered one hundred pounds of fully annealed, cold rolled, 316L stainless steel coil material. The material specified was 0.010-inches thick and 24-inches wide which was required to conform to the American Society for Testing and Materials (ASTM) ASTM A240 material specification.

- CDC purchase order No. 49874 to Castle Metals, dated May 16, 1995, ordered 144 inches of 7/8-inch, 304 stainless steel hex bar stock which was required to conform to ASTM A479 material specification.

- CDC purchase order No. 47600 to Joseph T. Ryerson, dated October 26, 1994, ordered sixteen pieces of 1-inch round bar stock which was required to conform to ASTM A479 material specification.

- CDC purchase order No. 47107 to Coulter Steel & Forge, dated September 20, 1994, ordered 12-inches of 3 1/4-inch round bar stock which was required to conform to ASME SB 160-200 nickel, minimum yield/tensile strength of 35,000/60,000 pounds per square inch, respectively.

- CDC purchase order No. 49909 to Metal Goods, dated May 18, 1995, ordered two pieces of 1 3/8-inch, 304 stainless steel round bar stock which was required to conform to ASTM A479 material specification.

Failure of CDC to impose customer required 10 CFR Part 50, Appendix B, quality requirements and in purchase orders to its suppliers furnishing materials or services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders has been identified as Nonconformance 95-01-03.

In addition, CDC Quality Control Instruction No. 1003 required that suppliers pass down all purchase order requirements to any authorized sub-tier suppliers and Quality Control Instruction No. 1020 required that compliance with 10 CFR Part 21 be mandatory on all shop orders and all purchase orders to suppliers when imposed by the customer purchase order. All of the customer purchase orders from licensees discussed in Section 3.7 specified that the requirements of 10 CFR Part 21 applied, however, CDC did not impose the requirements of 10 CFR Part 21 in its purchase orders to the suppliers which furnished material or services (as discussed previously in this section). Since CDC had not specified 10 CFR Part 50, Appendix B, on the purchase orders to the suppliers the requirements of 10 CFR Part 21 were not required (by regulation) to be included. However, CDC had placed a procedural restriction on itself which did require the inclusion of 10 CFR Part 21. As written, CDC's procedures did not allow them to purchase material or services as commercial grade (without passing down 10 CFR Part 21) and to take additional actions which would enable them to supply or use the material or services as safety-related. The inspectors concluded, based on discussion with CDC, that CDC was aware that if the requirements of 10 CFR Part 50, Appendix B, had been specified on CDC's purchase orders to suppliers that the requirements of 10 CFR Part 21 would have also been required to be specified by regulation. Although CDC's actions were ultimately consistent with 10 CFR Part 50, Appendix B, (not specifying 10 CFR Part 50, Appendix B, and therefore not specifying 10 CFR Part 21) the actions were not in accordance with the 10 CFR Part 21 requirements specified in Quality Control Instructions No. 1003 and No. 1020. CDC's failure to follow the procedural requirements related to 10 CFR Part 21, as specified in Quality Control Instructions No. 1003 and No. 1020, when purchasing material and services used in the manufacture of nuclear safety-related rupture discs and rupture disc holders, has been identified as a violation of 10 CFR 21.31, which requires that purchase orders to suppliers specify that the provision of 10 CFR Part 21 apply when applicable (in this case, when required by CDC Quality Control Instructions No. 1003 and No. 1020). This failure constitutes a violation of minor significance and is being treated as a Non-Cited violation, consistent with Section IV of the NRC Enforcement policy (NUREG-1600).

4 PERSONS CONTACTED

The NRC staff participating in the inspection and CDC personnel contacted during the inspection are listed below.

Continental Disc Corporation

Kenneth R. Shaw, President
*#Dean Dachenhausen, Director Quality Assurance

U.S. Nuclear Regulatory Commission

*#Bill Rogers	Team Leader, VIS/PSIB
*#Robert Pettis	Senior Reactor Engineer, VIS/PSIB
#Gregory Cwalina	Section Chief, VIS

*Attended the Entrance Meeting
#Attended the Exit Meeting

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

July 7, 1995

Ms. S. Kay Fisher
Manager, Quality Assurance
Divesco, Inc.
5000 Highway 80 East
Jackson, Mississippi 39208

SUBJECT: NRC INSPECTION NO. 99901117/95-01

Dear Ms. Fisher:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Divesco, Incorporated, conducted on February 8 and 9, 1995, by Mr. Stephen Alexander of this office. The inspection was conducted to provide a basis for assessing the validity and completeness of the list that you provided to General Electric Nuclear Energy (GE NE) of items from the American Heavy Trading Black Fox inventory that you supplied to GE NE. The inspection was also conducted to determine from your records the disposition of the remainder of the Black Fox inventory items, including those supplied to D-Tech (formerly OMTECH, Inc., and TEMCO, Inc.).

Areas examined during the inspection and our findings are discussed in detail in the enclosed report. Within these areas, the inspection consisted of selective examination of procedures and representative records, review of technical documentation, interviews with personnel, and observations by the inspectors. The major areas reviewed included (1) implementation of your quality assurance (QA) program based on Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50, Appendix B) with respect to quality assurance records and (2) implementation of your program for reporting of defects and noncompliance pursuant to 10 CFR Part 21.

Based on the results of this inspection, one part of your 10 CFR Part 21 implementation program appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (Notice). The violation of 10 CFR Part 21 is related to your procedures adopted pursuant to the regulation. However, the inspector found no instances in which your other practices or records were not in accordance with 10 CFR Part 21; nor did the inspector identify any instances in which potential Part 21 issues were not properly dispositioned. The specific findings and references to the pertinent requirements are identified in the enclosed Notice and inspection report.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. After reviewing your response to the Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

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

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

Docket No. 99901117

Enclosures: 1. Notice of Violation
2. Report No. 99901117/95-01

cc w/encl: Mr. Forest Hatch, Manager S&P Quality
GE Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

Mr. Dick Tettman, President
D-Tech, Inc.
15040 Los Gatos Boulevard
Los Gatos, CA 95032

NOTICE OF VIOLATION

DIVESCO, Incorporated
Jackson, Mississippi

Docket No. 99901117
Report No. 95-01

During an NRC inspection conducted February 8 and 9, 1995, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1992), the violation is listed below:

Section 21.21(a) of Part 21, "Reporting of Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR Part 21) requires, in part, that each individual, corporation, or entity subject to the regulations in this part adopt appropriate procedures to ensure the proper evaluation of deviations and failures to comply and the reporting of defects and failures to comply related to a substantial safety hazard to a director or responsible officer. Part 21 requires that (1) the evaluation of deviations and failures to comply in delivered basic components must be completed within 60 days of discovery, (2) reports to a director or responsible officer of defects and failures to comply related to a substantial safety hazard must be made within 5 working days of completion of the evaluation, and (3) an interim report must be made to the NRC within 60 days of discovery of the deviation or failure to comply if the evaluation cannot be completed within the required time. Section 21.21(b) requires that when a supplier determines that it is not capable of evaluating the deviation or failure to comply, then it must notify affected licensees or purchasers of the deviation or failure to comply within 5 working days of making this determination.

Contrary to the above, as of February 9, 1995, the effective revision of Divesco's Nuclear Quality Assurance Manual, Procedure No. 10, Revision 00, dated June 25, 1987, "Part 21 Evaluation and Notification," which constituted the Divesco procedures adopted pursuant to 10 CFR 21.21, would not, as written, ensure evaluation and reporting in accordance with the regulation as follows: The procedure called for notification to purchasers of deviations as if the procedure were established to follow §21.21(b), yet contrary to §21.21(b), the procedure required evaluation of those deviations to determine safety impact prior to, or as a prerequisite for, customer notification as would otherwise be performed under §21.21(a). However, having called for an evaluation of the type required by §21.21(a), the procedure did not provide for notification of a director or responsible officer within 5 working days should the evaluation identify a defect or failure to comply associated with a substantial safety hazard as required by §21.21(a). The procedure also did not contain the interim reporting requirement and time limit provisions added by the version of the regulation that became effective on October 29, 1991. (95-01-01)

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

Enclosure 1

Pursuant to the provisions of 10 CFR 2.201, Divesco is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Technical Support, Office of Nuclear Reactor Regulation (Mail Stop: O-9A1), within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Dated at Rockville, Maryland
this 7th day of July, 1995

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

REPORT NO.:	99901117/95-01
ORGANIZATION:	Divesco, Incorporated 5000 Highway 80 East Jackson, Mississippi 39208
ORGANIZATIONAL CONTACT:	S. Kay Fisher Manager, Quality Assurance
NUCLEAR INDUSTRY ACTIVITY:	Divesco (Formerly NSSS-Divesco) supplies (directly or as a consignment agent) surplus equipment, components, and piece parts (obtained primarily from canceled nuclear plant projects) to the nuclear industry.
INSPECTION DATES:	February 8 and 9, 1995

INSPECTOR: Stephen D. Alexander, Vendor Inspection Section (VIS), Special Inspection Branch (TSIB) — Date

REVIEWED BY: Gregory C. Cwalina, Chief, VIS/TSIB — Date 6/6/95

APPROVED BY: Robert M. Gallo, Chief, TSIB/DOTS — Date 7/7/95

Enclosure 2

1.0 SUMMARY OF INSPECTION FINDINGS

The inspection was conducted to provide a basis for assessing the validity and completeness of the Divesco list provided at NRC request to General Electric Nuclear Energy (GE NE) of items sold to GE NE from the American Heavy Trading, Incorporated (AHT), consignment inventory at Divesco. This inventory, on consignment with Divesco as agent for AHT from 1985 to 1989, consisted of material purchased by AHT from Public Service of Oklahoma's (PSO's) cancelled Black Fox Nuclear Station (Black Fox). The inspection was also conducted to determine from Divesco records the disposition of the remainder of the AHT consignment Black Fox inventory items, including those supplied to another surplus material dealer called D-Tech (formerly TEMCO, or OMTECH, Inc.) in Los Gatos, California, or to others, if any. During this inspection, the NRC inspector reviewed Divesco records and evaluated the Divesco system of record keeping to accomplish the above objectives.

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, Appendix B)

- Part 21, "Reporting of Defects and Noncompliance," of 10 CFR

- Divesco Nuclear Quality Assurance Manual (NQAM), Procedure No. 10, Revision 00, dated June 25, 1987, "Part 21 Evaluation and Notification"

1.1 **Violation** (95-01-01) Contrary to the requirements of 10 CFR 21.21, Divesco procedures adopted pursuant to the regulation (1) called for evaluating deviations to determine safety impact before notifying affected purchasers or licensees, but did not provide for notifying a director or responsible officer should the evaluation identify defects or failures to comply associated with a substantial safety hazard, and (2) did not contain certain reporting provisions and time limits required by the version of the regulation that became effective on October 29, 1991 (see S[illegible] 3.4 of this report).

1.2 Nonconformances

None

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous findings were reviewed during this inspection.

3.0 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

During an inspection of GE NE in San Jose, California, in April of 1994, the NRC requested GE NE to research its records and provide information for NRC review on the procurement, handling, and disposition of equipment, components, and parts belonging to the consignment to Divesco by AHT. GE NE transmitted to the NRC a list of seven procurements of material said to be traceable to the AHT Black Fox consignment through Divesco. Table 1 of Appendix A to this report lists these seven procurements. During a subsequent inspection at GE NE in January 1995, GE NE explained that it did not have its surplus material procurement records computerized or organized in a manner conducive to efficient searches of the kind requested by the NRC. Instead, GE NE had requested Divesco to provide the information. During the January 1995 inspection, the inspector reviewed the information on the handling and disposition of material on the list of the seven procurements of AHT/Black Fox material from Divesco and found that it appeared to have been procured, inspected, and supplied in accordance with GE NE procedures which the NRC has extensively examined in the past (refer to NRC Inspection Report Nos. 99900403/ 89-01, 90-01, and 94-02 or to 1989, 1990 and 1994 volumes of NUREG-0040). No discrepancies were noted with the handling or disposition of the material in the Divesco list of seven procurements during this February 1995 NRC inspection at Divesco.

During the January 1995 inspection at GE NE, the inspector learned that GE NE had not conducted an independent search of its own records as expected. Therefore, the inspector searched the records at GE NE and identified procurements of material from Divesco that were not listed on Divesco's list. These procurements are listed in Table 2 of Appendix A to this report. The inspector also identified several procurements from another surplus equipment dealer called OMTECH that had had some dealings with Divesco and AHT. This company was later called TEMCO and now is called D-Tech. The source of the material was not evident from these records, but it was possibly the AHT Black Fox consignment in question. Accordingly, this information was pursued at Divesco during this February 1995 inspection and also at D-Tech in March 1995.

3.2 Entrance and Exit Meetings

During the entrance meeting, the inspector met with the Divesco Quality Assurance (QA) Manager and discussed the scope of the inspection. During the exit meeting on February 9, 1995, with the Divesco Quality Assurance Manager, the inspector summarized the inspection findings.

3.3 Procurement/Receiving/Sales Records

The Divesco database (for sales in 1986 and later) contained all the sales to GE NE that Divesco had identified to GE NE following the April 1994 NRC inspection of GE NE (Table 1, Appendix A). Among the GE NE purchase orders (POs) to Divesco from 1986 and later (selected from GE NE files in January 1995) that had not been identified to GE NE by Divesco as having been filled from the AHT Black Fox consignment inventory (Table 2, Appendix A), one item, an Agastat relay, purchased on GE NE PO 205-86R686, possibly came from material originally supplied to PSO for Black Fox by GE NE, but not from the material included in the AHT Black Fox consignment. The time frame of this procurement, the type of item, and the way it was identified in material receiving reports suggested that it was from another of Divesco's sources, but it was examined as if it had been part of the AHT consignment.

Also among the POs listed in Table 2 of Appendix A were two other items purchased by GE NE in 1985 that were from the AHT Black Fox consignment. These were two pressure transducers for Perry (PO 205-85E949) and a hydraulic hand pump for Clinton (PO 205-85J769). The Divesco QA Manager explained that Divesco had not included any of the 1985 sales of Black Fox material in its original list to GE NE (Table 1, Appendix A) because Divesco had overlooked the fact that some of the pertinent information might be in its separate database containing records of 1985 sales only. One additional procurement identified among GE NE records, some terminal boards for Perry (PO 205-85E813), was not listed in the Divesco database. It was also not found during a review of Divesco paper sales records. Divesco has not yet traced these items to the AHT Black Fox consignment. The rest of the GE NE procurements in Table 2, Appendix A, selected at GE NE for review at Divesco were traceable to non-Black Fox sources, i.e., TVA, Chism Company, or GE Allens Creek.

Finally, review of the Divesco 1985 sales database revealed two procurements by GE NE of two control cards (P/N 204B7215G001) for Perry (PO 205-85N643) and three GE SBM-type switch handles for Clinton, PO 205-85N91_ (truncated PO number). According to Divesco Invoice B11225-1, listed in the Divesco database for 1985 sales for this line item, the referenced GE PO number was 205-85N911, but this PO, obtained from GE NE, is for ASCO solenoid-operated valves and the PO was issued to "OMTECH," now D-Tech. Although the source of the control cards was GE Allens Creek and the source of the SBM handles was listed as GE NE itself, these two procurements were not found among the surplus material procurement records at GE NE. Divesco is working with GE NE to resolve this apparent discrepancy.

With regard to the remainder of the AHT Black Fox consignment, the inspector also reviewed Divesco's records of all sales of AHT Black Fox consignment material to identify any other parties to whom the material may have been sold. According to Divesco records, there were several procurements by various utilities directly from Divesco of material from the AHT Black Fox consignment: 8 in 1985 and 31 more from 1986 on. Divesco certified to meeting the PO, handling and storage in accordance with applicable QA requirements, and traceability to the Black Fox consignment. No deficiencies were identified in the handling or disposition of this material. NRC licensees are responsible for review of the material for suitability of application and verification that it meets applicable requirements.

The only other party with whom Divesco's records indicate it had dealings in Black Fox-traceable material was TEMCO (now D-Tech). The inspector reviewed Divesco's records of transactions with TEMCO to determine which, if any, of the items involved came from the AHT Black Fox consignment inventory. Divesco explained that the material transactions between Divesco and TEMCO (listed as "sales" in the Divesco database) were not outright sales, but transfer of consignment goods, owned by TEMCO and warehoused by Divesco, for which Divesco would receive commission. This information was later confirmed by the inspector during the March 1995 inspection at D-Tech. Divesco's database does not list any such transfers to TEMCO in 1985. Of the numerous transfers to TEMCO from December 1987 through the last of them in 1992 listed in the Divesco database, all the material came from non-AHT/Black Fox suppliers (i.e., GE Allens Creek, GE NE, TVA (IRP), TVA/Chism, or TEMCO itself). Sales to TEMCO in 1986 were not listed in the Divesco database, although Divesco had paper records of them and was in the process of entering them into the database. This data was also reviewed by the inspector to confirm the source and disposition of the material during the inspection at D-Tech in March 1995.

In addition, the database showed several procurements by GE NE from Divesco (one in 1986 and three in 1988) for which Divesco records showed TEMCO as the supplier. During the inspection at D-Tech in March 1995, all the material identified as being possibly traceable to Black Fox was pursued to determine or confirm its origin and its disposition by D-Tech.

The inspector also reviewed Divesco records of sales to TEMCO to determine if Divesco (and ultimately Black Fox) was the source for any of the material supplied by TEMCO to GE NE which had been identified by the inspector at GE NE in January 1995 during the record search that also identified the procurement from Divesco listed in Table 2 of Appendix A. This search in January 1995 had identified nine procurements by GE NE from TEMCO (all 1986 to 1987). None of these were listed in Divesco records as of February 1995, suggesting TEMCO's source was other than Divesco for this material. During the March 1995 inspection at D-Tech, this was confirmed except for one procurement. GE NE PO 205-86R630 to TEMCO for a Rosemount temperature element (S/N 40249) was procured from Divesco in 1987. According to D-Tech records reviewed by the inspector in March 1995, this item came from Divesco in August 1987, but it did not appear in the Divesco computer database record of sales to TEMCO, which went back only to December 1987. The transfer of this item from Divesco to TEMCO was later reported to the NRC as confirmed by Divesco after review of its paper records.

In 1987, Divesco and AHT dissolved their consignment/joint venture agreement. However, Divesco retained the inventory for the time being and sold several items to GE NE and some items directly to nuclear utilities from time to time under special agreements in each case with AHT. In January 1988, Divesco purchased several items from the Black Fox consignment inventory outright (listed in a bill of sale, dated January 24, 1988, and signed by the president of AHT). The inspector reviewed the records of items from that purchase that have been sold. Finally, according to a receipt and release document on file with Divesco, dated July 25, 1989, signed by AHT's president, AHT acknowledged receipt of the remainder of its consignment inventory, which was to be removed from Divesco's site within 30 days. Divesco stated that the material known to belong to AHT was removed by AHT.

To summarize, the material in question is all surplus material from PSO's cancelled Black Fox project. Material from Black Fox was sold to both GE NE and AHT. Some of the items sold by PSO directly to GE NE were sold by GE NE directly to NRC licensees. Others were sold by GE NE to TEMCO, which placed those items, along with other surplus material from GE NE, in storage at Divesco and one other location. TEMCO sold some of this material directly to NRC licensees, and some to GE NE for sale to NRC licensees. In either case, TEMCO would get the material back from Divesco (or one other warehouse of its own) and ship it (or have it shipped direct) to its (TEMCO's) customer (either GE NE or a utility). The inspector found no evidence that any of this material was commingled with material from the AHT Black Fox consignment.

The Black Fox material purchased by AHT from PSO was placed on consignment with Divesco in accordance with a joint venture agreement between AHT and Divesco in 1985 and shipped directly from the Black Fox site to Divesco's warehouse. Over the next few years, Divesco sold some of the AHT Black Fox consignment to GE NE and some directly to utilities as agent for AHT. As discussed above, in 1989, AHT removed what both parties agreed to constitute the remainder of AHT's Black Fox consignment inventory from the Divesco site (except for certain items bought outright by Divesco from AHT). In telephone conversations with the inspector, AHT has stated that AHT has offered various items of the material it removed from the Divesco site to several utilities on the basis that it be used only for training aids or other non-safety-related purposes. The NRC has not confirmed the disposition of the material returned to AHT except that the inspector has seen photographs said to be of this material lying in an open field reportedly located somewhere in the Jackson, Mississippi area.

The NRC has received no substantive evidence that any of this material is, or has been, commingled with used, fraudulent, or refurbished material. The NRC also has no evidence that any of it is substandard. However, the material returned to AHT has not been maintained in 10 CFR Part 50, Appendix B, QA storage, and has apparently suffered some degradation from exposure to the elements.

As discussed above, the NRC has inspected GE NE's and Divesco's QA programs (including procurement), particularly scrutinizing their handling and disposition of the material in question. Neither Divesco nor D-Tech has certified as to the quality or suitability of the material for any plant applications other than that its condition has been maintained and that it is the material specified in customer POs, with records of traceability to the Black Fox consignment. GE NE has certified to the quality of the material in those instances in which the material has been determined to be traceable to material and QA records originally supplied to Black Fox and on the basis of receiving inspection or, as was the case with the MSIVs for NMP2, based on recertification by the original manufacturer.

Licensee procurement programs are required under 10 CFR Part 50, Appendix B, to determine whether any items procured from any supplier, including GE NE, TEMCO, Divesco, or another utility, are suitable and of adequate quality, condition, and reliability for their plant applications. In addition, 10 CFR Part 50, Appendix B, requires licensees to establish measures to control nonconforming material to prevent its inadvertent installation or use. Further, 10 CFR Part 50, Appendix B, requires licensees to establish measures to detect and correct conditions adverse to quality. Finally 10 CFR Part 21 requires any entity that supplies a basic component to an NRC-licensed facility to evaluate any deviation or failure to comply in that basic component of which they become aware and to report it either to the NRC or to affected licensees or purchasers. In addition to working with licensees and industry groups to improve the quality of industry procurement practices, the NRC has inspected the procurement programs of numerous licensees, including some of those that have procured some of this material.

Should the NRC become aware of any substantive evidence that any of the material in question may be in some way substandard, defective, unreliable, or otherwise unsuitable for service in a NRC-licensed facility, the NRC would follow up on the specific information in a manner appropriate to the circumstances. Absent such information or evidence, this matter is considered closed.

In order to investigate the concerns raised by AHT that material from its Black Fox inventory was being sold to NRC licensees without traceability and hence was of indeterminate quality, the inspector needed first to establish the disposition of this material. Based on review of Divesco records, the inspector concluded that the disposition of the material in the Black Fox inventory has been determined as far as Divesco is concerned.

3.4 10 CFR Part 21 Implementation

The inspector reviewed Divesco Nuclear Quality Assurance Manual (NQAM), Procedure No. 10, Revision 00, dated June 25, 1987, "Part 21 Evaluation and Notification," and made the following observations:

(1) The stated purpose (Paragraph 1.0) of the procedure was to "identify the requirements for evaluating deviations for potential safety impact and informing the purchaser of the deviation to satisfy the requirements of 10CFR21." The inspector noted that although evaluating deviations was addressed, notification of the NRC in accordance with 10 CFR 21.21 was not, nor was evaluation of failures to comply as defined in Part 21. Evaluation of deviations and reporting to the NRC were addressed in a posted company policy statement. The inspector noted that both the procedure language and the posted policy statement were inconsistent with Divesco's practice, as described by the Vice President and the QA Manager, of not performing the 10 CFR 21.21(a) evaluations, but of simply informing affected licensees or purchasers of all deviations (and failures to comply) involving basic components supplied by Divesco of which Divesco becomes aware as provided by §21.21(b).

(2) Paragraph 2.0, "Applicability," stated that the Divesco program was limited to the evaluation of deviations (identified by Divesco or reported to Divesco) to determine if a potential safety problem exists and to the notification of the purchaser so that a 10 CFR Part 21 evaluation can be performed. Here again, the procedure described the Divesco practice as if it were carried out pursuant to 10 CFR 21.21(b), yet provided for an evaluation for safety impact as a prerequisite for informing customers. Section 21.21(b) is applicable when the basic component supplier has determined that an evaluation will not be performed. The language of this paragraph in the Divesco procedure mandated an evaluation pursuant to 10 CFR 21.21(a), yet there were no subsequent provisions for notifying a director or responsible officer who would then effect NRC notification.

(3) Paragraph 4.3, under "Requirements" (Paragraph 4.0), called for review of deviations (but did not include failures to comply) for "potential impact on safety." Paragraph 4.4 stated the requirement for "followup

notification to purchaser of a defect." In the case that Divesco elects to perform an evaluation and determines that a defect does exist or that a failure to comply could create a substantial safety hazard, Divesco procedures would then be required by §21.21(a) to provide for notification of a Divesco director or responsible officer (who, as stated above would then effect notification of the NRC), yet the procedure contained no such provisions, only purchaser notification.

Divesco confirmed that it would not normally be in a position to know either the intended plant application of its supplied basic component or the impact of any deviation or failure to comply on the plant, system or parent component safety function. However, this practice was not consistent with the language of the procedure. The inspector explained that nothing in the regulation should be construed to discourage or prohibit reporting under Part 21. However, unless Divesco is fully qualified to determine that a given deviation **does not** constitute a defect as defined in §21.3 or that a given failure to comply **could not** create a substantial safety hazard, then, contrary to the language of the Divesco procedure, the performance of the evaluation and the determination that a so-called potential safety impact exists should not be a prerequisite to informing all affected licensees or purchasers as required by 10 CFR Part 21.21(b).

(4) The procedure did not contain the additional provisions and new time limits promulgated in the version of 10 CFR Part 21 that became effective on October 29, 1991. According to the minutes of a Divesco QA meeting held on November 21, 1991, Divesco had reviewed what it believed to be the current revision of Part 21 because this version, dated October 31, 1989, was included as Attachment 1 to NRC Information Notice 91-31, issued June 17, 1991. This was also the version of the regulation posted pursuant to §21.6(a). However, the revision of the regulation containing the new provisions and time limits was first published in the Federal Register (56FR 36081) on July 31, 1991. It was then announced in NRC Information Notice 91-76, "10 CFR Parts 21 and 50.55(e) Final Rules," dated November 26, 1991. The inspector provided Divesco with a copy of the current revision of 10 CFR Part 21.

(5) Paragraph 5.2 stated, in part: "...procurement documents shall specify, when applicable, that 10CFR21 [sic] requirements are imposed on the supplier." It then stated: "When the source of the material is an ex-licensee, 10CFR21 [sic] provisions are not imposed." However, this statement in the procedure is inconsistent with §21.31. If the material to be procured is a basic component, then §21.31 requires invoking Part 21 in procurement documents, regardless of the status of the supplier. Although the language of Paragraph 5.2, as written, would allow violation of §21.31, the inspector did not identify any instances (within the restricted scope of this inspection) in which Divesco had failed to comply with §21.31.

Upon completion of the review of Divesco NQAM Procedure No. 10, the inspector concluded that it would not, as written, ensure proper evaluation and reporting, if required, of deviations or failures to comply in accordance with

10 CFR 21.21 in that it (1) called for evaluation of deviations to determine safety impact prior to notification to purchasers of deviations or failure to comply, yet did not provide for NRC notification, and (2) did not contain reporting provisions and time limits required by the current version of the regulation. These deficiencies were cited as Violation 99901117/95-01-01.

4 PERSONS CONTACTED

Divesco, Incorporated:

Westbrook, T.	Vice President
Fisher, S. Kay	Manager, Quality Assurance

APPENDIX A

TABLE 1: DIVESCO LIST OF SALES OF AHT/BLACK FOX ITEMS TO GE NE (1986 ON)

GE NE PO#	DATE	QTY	ITEM	DRAWING	GE NE CUSTOMER	SOURCE
205-86K317	04/28/86	1	FC valve	112D1459P001	Perry (CEI)	AHT/BF
205-86J142	05/14/86	2	MSIV blowers	213A3762P001	Clinton (IP)	AHT/BF
205-86R638	10/02/86	1	Temp Element	159C4520P005	River Bend(GSU)	AHT/BF
205-86R687	10/06/86	8	MSIVs	B21-F021 - 28	NMP2 (NMPC)	AHT/BF
205-86R689	10/06/86	1	MSIV Blower	47B518664	NMP2 (NMPC)	AHT/BF
205-86R874	11/17/86	2	Relief Valves	21A9508P001	Hatch (GP&L)	AHT/BF
205-87C630	03/13/87	1	Temp Element	159C4520P005	Clinton (IP)	AHT/BF

TABLE 2: GE NE PURCHASES FROM DIVESCO FROM NRC SEARCH OF GE NE RECORDS

GE NE PO#	DATE	QTY	ITEM	DRAWING	GE NE CUSTOMER	SOURCE
205-85E813*	unkn	unkn	Terminal Bd	147D7614G004,5	Perry (CEI)	*AHT/BF
*Not listed on Divesco's 1985 sales records database						
205-85E949	07/05/85	2	Press XDCR	MPL:C85N001	Perry (CEI)	AHT/BF
205-85J769	11/25/85	1	Hyd Hand Pump	131C8966G001	Clinton (IP)	AHT/BF
205-86R686	10/03/86	1*	Agastat	145C3217P041	River Bend	*AHT/BF
(3 of 4 rtnd, 4th: *S/N 77231248 retained, baring GE DWG #, [illegible] gastat part number, E7024PB002, as were the two, S/Ns 85170022,3 from Control Components. Therefore, the one finally kept by GE could have been from AHT/BF)						
205-85N648	09/12/85	58	Anlg Isol	204B6220AAG002	Perry (CEI)	Allens Ck
205-87C632	02/03/87	4	CKT Cards	272A8614P101,02,12,20	RB(GSU)	Chism

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

August 29, 1995

Mr. K.J. Cummings, Plant Manager
Eaton Corporation
9 South Street
Danbury, CT 06810

SUBJECT: NRC INSPECTION REPORT 99901290/95-01

Dear Mr. Cummings:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Eaton Corporation (Eaton), Danbury, Connecticut, conducted by Messrs. K.R. Naidu and I. Ahmed on August 8-10, 1995. The inspection was conducted to provide a basis for NRC staff confidence that the components manufactured by Eaton to upgrade the existing engineered safeguards actuation system (ESAS) for Northeast Nuclear Energy Company's Millstone Nuclear Power Station, Unit 2 (MP-2), would perform their intended safety functions. On September 10, 1995, at the conclusion of the inspection, the inspectors discussed their findings with you and other members of your staff.

During this inspection, the team evaluated the Eaton quality assurance program that was established to implement the provisions of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), Appendix B, and the provisions of 10 CFR Part 21, in selected areas during the design, manufacture, and installation of the ESAS upgrade for MP-2. Within these areas, the NRC team (a) examined technical documentation, procedures and representative records, (b) held discussions, (c) listened to presentations and (d) observed Eaton technicians' activities.

During the evaluation of your activities at Danbury, the inspectors noted the proactive approach of your staff to acknowledge weaknesses in the existing quality program and willingness to correct them. The inspectors noted that the employees who were interviewed during the inspection exhibited good technical expertise and positive attitudes.

The procedure adopted by you to implement 10 CFR Part 21, which was developed in 1978 by Consolidated Controls Corporation, was last revised in 1981 and failed to meet the current requirements. This failure constitutes a violation of minor significance and is being treated as a Non-cited Violation, consistent with section IV of the NRC Enforcement Policy. We understand that you are in the process of revising the current 10 CFR Part 21 Procedure to reflect the latest regulations, and that you are reorganizing the quality assurance manual to reflect the practices of your current organization.

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. No response to this letter or its enclosure is required. 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

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

Docket No.: 99901290

Enclosure: Inspection Report 99901290/95-01

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

REPORT NO.:	99901290/95-01
ORGANIZATION:	Eaton Corporation 9 South Street Danbury, Connecticut 06810
ORGANIZATIONAL CONTACT:	G A. DeRome (203) 798-3216
NUCLEAR INDUSTRY ACTIVITY:	Instrumentation and control systems for safety and nonsafety-related applications.
INSPECTION DATES:	August 8-10, 1995

LEAD INSPECTOR: Kamalakar R. Naidu, Team Leader, Vendor Inspection Section (VIS) — 8/29/95 Date

OTHER INSPECTORS: Iqbal. Ahmed, Senior Engineer, Instrumentation and Control Branch

REVIEWED BY: Robert M. Gallo for Gregory C. Cwalina, Chief, VIS, Special Inspection Branch (PSIB) — 8/29/95 Date

APPROVED BY: Robert M. Gallo, Chief, PSIB, Division of Inspection and Support Programs — 8/29/95 Date

Enclosure

1 SUMMARY OF FINDINGS

During this inspection, the inspectors evaluated the implementation of the quality assurance program adopted by Eaton Corporation (Eaton) in selected areas relating to the supply of material and services for upgrading the existing engineered safeguards actuation system (ESAS) panels for Northeast Nuclear Energy Company's (NNECO's) Millstone Nuclear Power Station, Unit 2 (MP-2). The inspectors also reviewed the actions taken by Eaton regarding to a 10 CFR Part 21 item.

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 50, Appendix B)

- Part 21, "Reporting Defects and Noncompliance," of 10 CFR.

The inspection identified a violation of minor significance that is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. (Paragraph 3.7.1)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of this vendor.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

During the entrance meeting on August 8, 1995, the NRC inspectors discussed with Eaton staff the scope of the inspection, areas to be reviewed, and established the persons to contact within Eaton management and staff. During the exit meeting on August 10, 1995, the NRC inspectors summarized their findings and concerns to the management and staff of Eaton. Persons contacted during this inspection are identified in Section 4.

3.2 Background Information

Eaton, formerly Consolidated Controls Corporation, designs, manufactures, and provides field services to install new systems and upgrades for instrumentation and control systems for safety and nonsafety-related applications in commercial and military nuclear power plants.

Consolidated Controls Corporation, designed the original ESAS panels to Specification 7604-M-480 for NNECO's MP-2. The purpose of the ESAS is to continuously monitor the operation of the plant to detect accident conditions and to actuate the safeguards systems. In the early 1980's, Eaton acquired Consolidated Controls Corporation, and continued to service the equipment supplied. Between June 1991 and May 1992, NNECO issued several purchase orders (POs) to Eaton to replace all ESAS modules including the sequencer and the actuation modules, and the power supplies with an upgrade design. For instance, the original ESAS design for MP-2, which utilized +15-Volt Series 300 High Noise Immunity Logic, is obsolete and Eaton could not supply the spare parts required to maintain it. Eaton submitted a proposal to MP-2 with a system upgrade using +5-Volt Series 74 HC Complementary Metal Oxide Semiconductor Logic to replace the existing +15-Volt logic. The benefits of the upgrade were smaller size, lower power dissipation, shorter propagation delay periods, and extended service life. During the installation of the upgrade, MP-2 experienced a number of unrelated problems including a partial loss of normal power. To resolve these problems, NNECO re-evaluated the design capabilities, and vulnerability of various ESAS components to electromagnetic interference (EMI) and radio frequency interference (RFI). These studies and various tests of the ESAS resulted in several modifications to the design, and change orders to POs for the procurement of components and services.

3.3 Review of NNECO Purchase Orders to Eaton

NNECO issued six POs to Eaton, including several change orders, for the supply of equipment for ESAS. Because Eaton was not an approved vendor of NNECO, NNECO took compensatory measures by imposing selected provisions of its quality assurance program, including establishing hold points, and conducted quality control surveillances to witness hold points and acceptance tests.

The following table summarizes the NNECO POs:

Date	NNECO PO	Eaton Sales Order	Brief Description
1973	N/A	N/A	Original contract
June 1991	881661	35-1936	Module upgrade
May 1992	885480	35-2809	Automatic Test Insertion (ATI) added
May 1992	886009	35-2822	Power supply upgrade
July 1992	886476	35-2827	Field Services to install modules
July 1993	277176	35-3827	Field Services and Miscellaneous items
July 1993	278294	35-3829	Test rack and additional tests to observe impact of EMI and RFI.

The inspectors reviewed the procurement documents and determined that Eaton had prepared detailed proposals to meet the specific technical requirements for each item and that NNECO had issued POs based on the proposals.

3.4 Review of Eaton's Design Review Process

The Eaton engineering department developed Standard Procedure Instructions (SPI) for "Class 35 Power Industry Controls," to implement the provisions of the Eaton Quality Assurance Manual, Revision 11, dated February 6, 1995, relative to Section III, "Design Control." Eaton engineers followed these instructions during the design review of the modules intended for MP-2. Eaton's "Design Review Committees" met on several occasions to review the adequacy of the preliminary and final designs of the MP-2 modules. Eaton's "Design Review Committee" minutes, dated August 28, 1991, indicated that the layout and electrical design of the replacement modules (6N636,-37, -38, -40, and 5N636, -37, -38, and -40) for MP-2 were reviewed and found acceptable to replace the existing modules. A design review meeting was held on December 4, 1991, to review the final design on bistable module 6N636-1, isolation module 6N637-1 and block module 6N640-1. The committee also conducted the intermediate design review of ATI module 6N639-1 and actuation module 6N638-1, and initial design review of sequencer module 6N641-1 and U/V input module 6N642-1.

The inspectors determined that even though engineers followed the SPIs during the design of the MP-2 equipment, the engineers could not readily retrieve the necessary documents to demonstrate adherence to SPIs at various stages of review and approval. This is an indication of weakness in the implementation of the quality assurance program, While acknowledging this weakness, Eaton's management assured the inspectors that it will make appropriate enhancements to the quality program to ensure that documents generated during the review and approval cycles are readily retrievable.

3.5 Review of NNECO PO No. 277176

The inspectors reviewed in detail the NNECO PO No. 277176 to Eaton to examine the effectiveness of the procurement and installation process. This NNECO PO required Eaton to procure and install current/current (I/I) converters, noise suppression equipment, and auctioneered power supplies in ESAS Sensor Cabinets, and to reconfigure the sump recirculation actuation signal (SRAS) logic and ATI alarm. The inspectors selected two items (I/I converters and noise suppression equipment) to verify that Eaton developed design output documents, such as the field change procedures and respective drawings as required by the PO. During this review, the inspectors noted two discrepancies between the design drawings (and the material supplied) and the as-built configuration.

The first discrepancy concerned the voltage rating of the I/I converter. The NNECO PO, which was based on Eaton's Proposal No. "Mar 381," dated May 11, 1993, specified a 125-Vdc I/I converter. Eaton's proposal did not mention the voltage rating, but Eaton's schematic Drawing No. SGN548-13, Revision B, identified the input voltage rating of these I/I converters to be 18 to 60 Vdc. The inspectors could not find any documentation either from Eaton or from NNECO reconciling this discrepancy in the PO requirement. NNECO's only comment in its letter to Eaton dated March 24, 1994, was a request to add the NNECO drawing number on the Eaton drawing.

The second discrepancy concerned the addition of noise suppression devices to the coils of all output relays in the ESAS and to the sequencer inputs. Eaton illustrated the specified noise suppression devices consisting of a series combination of voltage-regulating Zener diode and a resistor (part number KLK2900-1) in Drawing No. KLK2900-1, and provided the instructions to install them in the Field Change Procedure, FCP KRH 136. FCP KRH 136 required a noise suppressor assembly to be soldered across pins 13 and 14 of each output relay socket. Both Eaton and NNECO engineers informed the inspectors that instead of this noise suppressor assembly, the as-built configuration consisted of a general purpose diode (without the resistor) soldered onto the relays across the relay coil instead of being soldered to the socket pins as required by the FCP. The inspectors could not find any documentation to indicate that NNECO evaluated this deviation from the manufacturer's design in the installation, or that Eaton either acknowledged this change with comments or concurred with the change. Eaton engineers informed the inspectors that their technicians performed the installation in accordance with verbal instructions from NNECO personnel.

The inspectors informed Eaton that lack of formal documentation on the changes to the voltage rating of the I/I converters, and the noise suppression devices was a weakness in the quality assurance program.

3.6 Process to Manufacture Printed Wiring Boards (PWBs)

Typically, Eaton design engineers prepare the schematics for the modules' design and submit them to NNECO for review and approval. After NNECO's approval, computer assisted designers generate artwork (silk screen, component side and circuit side of the PWB), and send it to a subcontractor for the fabrication of PWBs. Eaton populates (inserts components) the PWBs according to design drawings, sends them through the wave soldering machine to solder the components, and builds a prototype module. Only after the prototype successfully passes the tests at both Eaton and MP-2, does Eaton commence the manufacturing of production modules.

The inspectors reviewed the circumstances that led to the failures of 6N638-1 actuation modules and determined that the prototype modules successfully passed the tests at Eaton and MP-2. However, when the 6N638-1 production modules were tested at Eaton in the presence of NNECO quality control inspectors, they failed the insulation resistance tests during Hypot testing because the spacing between copper conductors on the PWB (clad runs) was inadequate.

Eaton's Work Order Instruction 1936-990 dated June 9, 1993, indicated that the PWBs that failed the tests were not shipped to MP-2 and that they were scrapped. Eaton redesigned the 6N638-1 actuation modules, increasing the distances between clad runs. PWBs manufactured from the revised drawings successfully passed the insulation resistance tests.

The records of the identification and disposition of nonconforming items were not readily retrievable. For instance, the documentation relating to the failure of the actuation modules during the Hypot tests, the corrective action taken to redesign the modules, and the disposal of the nonconforming PWBs was not readily retrievable. The inspectors identified this matter as a weakness in the implementation of the established quality assurance program. Eaton quality control personnel did not document the insulation resistance failure of the actuation modules 6N638-1 during the Hypot tests in a discrepancy report (DR). Eaton could have used the DR to document subsequent actions, such as the investigation of the failure, the root cause (inadequate spacing between the clad runs), the action taken to correct the unacceptable spacing (redesign the module by increasing the spacing), and the final disposal of the failed PWBs (scrapped). The DR could have been a readily retrievable quality assurance document with adequate description on the problem. When the inspectors pointed out this weakness to the Eaton Quality Manager, he responded that he will correct this weakness during the next revision of the quality assurance manual.

The inspectors found that the actions taken by Eaton were acceptable even though there was a weakness in the documentation on the dispositioning of the nonconforming PWBs.

3.7 Review of 10 CFR Part 21 (Part 21) Program

The inspectors reviewed the program established to implement the reporting requirements of Part 21 as discussed in the following sections.

3.7.1 The inspectors reviewed Revision A to Standard Procedure Instruction (SPI) No. 1563-031, "Reporting of Defects and Noncompliance," dated September 14, 1981. The procedure had not been revised to reflect that Eaton was the current entity responsible for implementing the reporting requirements of Part 21. Furthermore, the procedure did not reflect the current requirements of Section 21.21, "Notification of failure to comply or existence of a defect and its evaluation," of Part 21 which requires Eaton to adopt appropriate procedures for evaluating and reporting defects and failures to comply.

Contrary to the above, SPI No. 1563-031 did not have provisions that would implement the above requirements. The inspectors informed Eaton engineers that failure to have a procedure to implement these provisions of Part 21 constituted a violation of minor significance and would be treated as a Non-cited Violation, consistent with Section IV of the NRC Enforcement Policy (NUREG 1600).

Section 21.21(b) of Part 21 states that if a supplier of basic components determines that it does not have the capability to perform evaluations to determine if a defect exists, then it must inform the purchasers or licensees within five working days of making this determination. Section 21.21(c) of Part 21 requires a director or responsible officer to notify the commission when he or she obtains information reasonably indicating a failure to comply associated with substantial safety hazards, or a defect. However, Part 21 does not explicitly require these provisions to be included in procedures adopted pursuant to the regulations. Nevertheless, the inspectors expressed their concern that either insufficient or incorrect guidance may fail to prevent or even lead to violations of Part 21.

3.7.2 In a letter dated May 10, 1994, Eaton notified the NRC pursuant to Part 21, of a problem with its design of the 6N642 module. The design caused a higher-than-normal failure rate of an integrated circuit (IC) on the 6N642 electronic module assembly. The 6N642 electronic module is part of the ESAS at MP-2. Eaton experienced failures of the IC (Part U7) on this module. Eaton corrected the problem with the following actions:

1. Upgraded the drawings for the schematic and artwork

2. Upgraded the spare unit which was being manufactured.

3. Upgraded the two existing spare units in the possession of Millstone-2.

4. Prepared and released "Field Change Procedure for Correction of U7 of 6N642-1," drawing KRT 136 so that the other units that had been supplied could be upgraded on site.

The inspectors concluded that despite procedural weaknesses, this Part 21 issue had been satisfactorily dispositioned. No instance of unsatisfactory handling of Part 21 issues was identified.

3.8 <u>Quality Control Training</u>

Eaton has established an acceptable training program for all employees. To qualify personnel performing quality control (QC) inspection and testing activities, Eaton provides the "Study Guide for Inspector/Tester Qualification Program (Ref. QCR-82)," to its supervisors so that new employees are trained on appropriate information which is required to pass the Inspector/Tester

qualification program test. Eaton Quality Assurance demonstrated through records that inspection personnel were qualified to perform assignments. In addition to training documented in these qualification records, QC personnel receive additional training in a variety of subjects.

The inspectors reviewed the records maintained by Eaton's "Training Facilitator" and observed examples that Eaton had trained quality control department employees (Classified 0481) in the following technical areas:

- Quick response training (QRT) terminal board soldering

- SPI 571-2, Revision AN (list of manufacturing procedures)

- Use of MIL - standards at the workbench.

* QRT Securing Capacitors
* Quality Auditor Workshop
* Fundamentals of purchasing
* Principles of Materials Management
* Nuclear Coatings Seminar

The training program for quality control and purchasing personnel did not provide guidance to detect the various fraudulent, or otherwise unacceptable products that have entered the nuclear industry and did not mention the numerous generic communications issued by the NRC on this subject. Eaton personnel concurred with the inspectors and committed to upgrade the training program accordingly.

4 PERSONS CONTACTED

Eaton Corporation

•	K.J. Cummings	Plant Manager
• *	G.A. DeRome	Manager, Power Industry Controls
*	A. Emanuele	Customer Service
	W. Herrity	Design Engineer
• *	R. Magner	Quality Control Engineer
• *	A. Mancini	Senior Marketing Engineer
•	N.J. Tarasovic	Quality Advancement Manager
	D. Tuck	Training Facilitator

Applied Energy Services, Overland Park, Kansas

• *	S.A. Yousif	Senior Project Manager

* Attended the entrance meeting on 8/8/95
• Attended the exit meeting on 8/10/95

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

July 7, 1995

Dr. Stephen R. Specker
Vice President and
General Manager
General Electric Nuclear Energy
175 Curtner Avenue
San Jose, CA 95125

SUBJECT: NRC INSPECTION REPORT 99900403/95-01

Dear Dr. Specker:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection conducted January 17-19, 1995, at the GE Nuclear Energy (GE NE) facility in San Jose, California. The inspection was conducted by Mr. S.D. Alexander of this office, and the findings were discussed with the cognizant members of your staff identified in the report at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed report. The inspection was conducted to provide a basis for assessing the validity and completeness of your list of items from the American Heavy Trading Black Fox inventory supplied to GE NE by Divesco, Incorporated; to determine from your records the disposition of Black Fox inventory items, including those supplied to GE NE by D-Tech (formerly Temco, Inc., and OMTECH, Inc.).

The inspectors also reviewed the actions taken by your staff to correct inspection findings identified in Inspection Report 99900403/94-02. Within these areas, the inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

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

Sincerely,

Robert M. Gallo

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

Docket No.: 99900403

Enclosure: Inspection Report 99900403/95-01

cc: See next page

cc w/encl: Mr. Dick Tettman, President
D-Tech, Inc.
15040 Los Gatos Boulevard
Los Gatos, CA 95032

Ms. S. Kay Fisher
Quality Assurance Manager
Divesco, Inc.
5000 Highway 80 East
Jackson, MS 39208

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

REPORT NO.:	99900403/95-01
ORGANIZATION:	GE Nuclear Energy 175 Curtner Avenue San Jose, California 95125
ORGANIZATIONAL CONTACT:	Kenneth W. Brayman Manager, Quality Assurance Systems
NUCLEAR INDUSTRY ACTIVITY:	GE Nuclear Energy's activities within the scope of this inspection include supplying replacement parts and equipment to the nuclear industry.
INSPECTION DATES:	January 17-19, 1995

LEAD INSPECTOR: [signature] 7/6/95
Stephen D. Alexander — Date
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

OTHER INSPECTORS: None

REVIEWED BY: Gregory C. Cwalina 7/7/95
Gregory C. Cwalina, Chief, VIS/TSIB — Date

APPROVED BY: Robert M. Gallo 7/7/95
Robert M. Gallo, Chief, TSIB/DOTS — Date

Enclosure

1 SUMMARY OF INSPECTION FINDINGS

The inspection was conducted (1) to provide a basis for assessing the validity and completeness of the list of items from the American Heavy Trading Black Fox inventory supplied to General Electric Nuclear Energy, (2) to determine the disposition by GE Nuclear Energy (GE NE) of the Black Fox inventory items, including those supplied to GE NE by Divesco, Inc., D-Tech (formerly Temco, Inc., and OMTECH, Inc.), or others, if any, and (3) to close out previous Nonconformance 94-02-01.

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, "Reporting of Defects and Noncompliance," of 10 CFR

1.1 Violations

None

1.2 Nonconformances

None

1.3 Open Item

(99900403/95-01-01) Review of GE NE policy, procedure, and practice regarding QA, QC, and supervisory review of test data records or other documents associated with activities affecting quality (See Paragraph 3.2 of this report).

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Nonconformance 99900403/94-02-01: (Closed)

Contrary to the requirements of Criterion V, "Instructions, Procedures, and Drawings" of 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, Appendix B), GE NE Test Instruction (TI) 4389, used for dedicating molded-case circuit breakers for safety-related applications did not have appropriate acceptance criteria for determining, during the instantaneous magnetic trip test, that the breaker would not trip below the lower tolerance limit of the design magnetic trip band. Such criteria were also not found in the test equipment operating instructions. Consequently, for example, in GE NE dedication Work Order 93554, the hold current value (the test current pulse for which the breaker does not trip) was not recorded.

As a result of the GE NE response to Nonconformance 94-02-01 (GE NE Letter dated October 7, 1994), the inspector reviewed the dedication documents and test instructions (TIs) associated with Work Order (WO) D93554 again.

Specifically reviewed were: (1) Selected Item Drawing (SID) DD213A9893, Revision 6, dated December 20, 1993 (the item tested under WO D93554 was a TEC36100SST12RS molded-case circuit breaker (MCCB) which is Part 4 on SID DD213A9383), (2) Test Instruction TI 4353, Revision 7, dated October 22, 1993, (3) TI 4389, Revision 2, dated September, 22, 1993, and (4) Work Order D93554 itself, completed on February 1, 1994. The WO also referenced the above mentioned SID and TI 4353. Step 4.1.10 of TI 4353 requires performing the magnetic trip test using the PS-600 test set in accordance with TI 4389.

The GE NE response stated that GE NE had determined that "TI 4389 did require all pertinent data to be recorded in Paragraph 2.11." However, Paragraph 2.11 in the version reviewed during the April 1994 inspection (Revision 2, dated September 22, 1993) merely stated: "Record the pertinent [sic] data required for dedication for the following tests:" without specifying what data (and in what form) was pertinent and expected to be recorded. Therefore the criticism of TI 4389 in the inspection report, i.e., that it was not specific enough to ensure that all required data would always be recorded, remained valid.

Also stated in the GE NE response was that TI 4389 had been revised for clarification, specifically addressing recording of "hold point" test currents. Presuming that this meant that hold current (i.e., a test current value at which the MCCB does not trip instantaneously) would be recorded, the response stated further that the test data sheet, Form QC 348, "now requires" hold data to be recorded. However, in reviewing the QC 348 form that is Attachment 1 to TI 4389, as well as the QC 348 that is also Attachment 3 of TI 4353, the inspector found that Form QC 348 already provided for recording hold current. Therefore, the language of the GE NE response raised the following questions:

(1) Was the text of Paragraph 2.11 of TI 4389 revised to require recording hold current or was it revised to require recording the specific data required by the test (for which blocks are already provided on the QC 348 that is an attachment to TI 4389)?

(2) The language of the response implied that the QC 348 form had been revised, presumably by a revision to TI 4389, to which the QC 348 Form in question is an attachment. However, as is GE NE practice, the QC 348 attached to WO D93554 was Attachment 3 to TI 4353, the dedication procedure for a type of MCCB, not TI 4389, the detailed test instruction. Therefore, were the QC 348s that are Attachment 3 to TI 4353, TI 4337, TI 4271, and any other TIs in which Form QC 348 is an attachment, also revised, either as a revision to the standard form or by revisions to all the procedures to which the form is an attachment?

Finally, the GE NE response, stated that TI 4389 sets up the PS-600 test set to perform the tests by ramping up the test current until the MCCB trips. However, the Multi-Amp "Instruction Manual for Circuit Breaker Test Set Model PS-600," Revision 2, dated August 15, 1991, and the PS-600 settings given in TI 4389, indicate that an incremental pulse method is used. Specifically, for the breaker tested under WO D93554, the PS-600 would have been set up to put out a series of pulses of 12 cycles duration (JOG ON CYCLE setting of 12) with a one second pulse interval (JOG OFF SECONDS setting of 01) with each successive pulse incremented in magnitude by the test set as a preset function of the JOG OFF pulse interval setting. The pulse series would be terminated by either a breaker trip or by the limit settings in the PS-600.

The inspector agreed with GE NE's conclusion that in this case, the performance of the magnetic trip of the MCCB tested under WO 93554 was satisfactorily verified. This was based on the inspector's knowledge of how the test set applies test pulses. However, it was not clear from knowledge of the test set operation and of its display and from discussion with the test technician that the assertion made in the GE NE response that "any current value below the trip value is the hold current value" is valid. If this were true, the test technicians who performed the other tests, documented in other WOs reviewed by the inspector, could have arbitrarily selected and recorded any value below the trip value captured by the test set display as the hold current. Whereas, the test technician explained that the values recorded for hold current are determined by noting the captured displayed trip value, determining the pulse amplitude increment for the JOG-OFF setting in use, and subtracting the applicable increment from the trip value, or if possible noting the pulse amplitude current value displayed for the pulse preceding the one for which the MCCB tripped.

During this inspection, the inspector reviewed the revised version of TI 4389 (revised since the April 1994 inspection) and found that contrary to what was stated in the GE NE response to Nonconformance 94-02-01, TI 4389 had in fact been revised in a manner responsive to our original concerns. Therefore to address the first question stated above, the inspector determined that the TI was revised to include instructions for recording specific data and appropriate acceptance criteria. With regard to the second question, GE NE explained that it had not intended to imply in its response to the April 1994 inspection report that the QC 348 form itself had been revised (which it had not) and that admittedly, the phrase "now requires" was an inappropriate choice of words. Neither the QC 348 form itself nor the attachments of Form QC 348 to other TIs were, or needed to be, revised.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Black Fox Parts

During an April 1994 inspection (See NRC Inspection Report 99900403/94-02, dated September 13, 1994), the NRC requested GE NE to research its records and provide information on the procurement, handling, and disposition of equipment, components, and parts belonging to the consignment to NSSS Divesco, Inc. (Divesco), from the cancelled Public Service of Oklahoma (PSO) Black Fox Nuclear Plant (Black Fox) Project. Subsequently, GE NE transmitted to the NRC a list of seven procurements by GE NE of material traceable to Black Fox through Divesco. These procurements are listed in Table 1 of Appendix A to this report. During this inspection in January 1995, GE NE explained that it did not have its surplus material procurement records computerized or organized in a manner conducive to efficient searches of the kind requested by the NRC. Instead, GE NE had requested Divesco to provide the information. During this inspection, the inspector reviewed the information on the handling and disposition of material on the list of the seven procurements of AHT/Black Fox material from Divesco and found evidence that it was procured, inspected, and supplied in accordance with GE NE procedures which the NRC has extensively examined in the past (Refer to NRC Inspection Report Numbers 9990403/89-01, 90-01, and 94-02 or to 1989, 1990 and 1994 volumes of NUREG-0040). No discrepancies were noted with the handling or disposition of the material in the Divesco list of seven procurements during this inspection.

Also, because GE NE had not conducted an independent search of its own records as expected, the inspector searched the records at GE NE during this (January 1995) inspection and identified procurements of material from Divesco and also another surplus equipment dealer formerly called OMTECH, Incorporated, then later, TEMCO, and now D-Tech, that were not listed on the Divesco list of seven procurements discussed above. These procurements are listed in Table 2 of Appendix A to this report. It was not evident from the surplus material procurement records at GE NE what the source of the material was, but it was possibly from the AHT Black Fox consignment in question. Accordingly, this information was pursued at Divesco during a February 1995 inspection (See NRC Inspection Report 99901117/95-01) and also at D-Tech in March 1995.

During the visit to D-Tech in March 1995, conducted as part of the January 1995 inspection of GE NE, the inspector determined that the material supplied to GE NE by D-Tech had been originally sold by GE NE to D-Tech and was being bought back by GE NE for resale to other GE NE customers. Some of it may have at one time belonged to the Black Fox project because some of the Black Fox surplus inventory was bought back from PSO by GE NE when Black Fox was cancelled. However, the records indicated that this material was not part of the American Heavy Trading consignment of Black Fox material to Divesco.

During the inspections of Divesco, as discussed in Inspection Report No. 99901117/95-01, and at GE NE and D-Tech, as discussed above, the inspector found no evidence that surplus material from the cancelled Black Fox project traceable to the AHT Black Fox consignment to Divesco was commingled with other material or procured, handled, or resold in a manner inconsistent with NRC regulations or detrimental to safety.

3.2 QC Review

In April 1994 and during this January 1995 inspection, the inspector noted that the GE NE test technicians who performed tests documented in dedication WOs would routinely sign those WOs in the QC review block. The inspector also noted that this was the case on a WO reviewed in connection with a 10 CFR Part 21 notification by an NRC licensee regarding some MCCBs that failed during onsite testing. The GE NE WOs for the dedication of these MCCBs clearly showed that one of the test results was out of tolerance, yet the test technician erroneously signed the block at the bottom of the data sheet indicating that the QC review was complete and presumably that no discrepancies had been identified. The inspector learned that there is no other required, routine or random review of these WOs by independent QC, QA personnel or supervisors which would (or should) provide an opportunity to detect errors of this sort, i.e., missing data, as in the case cited in the previous nonconformance, incorrect data, or missed out-of-tolerance data as in the Part 21-reported case. The issue of GE NE policy, procedure, and practice regarding independent QA, QC, and supervisory review of test data records or other documents associated with activities affecting quality (i.e., by someone other than the technician who performs the tests) will be addressed in a future NRC inspection. This issue is designated Open Item 95-01-01.

4 PERSONS CONTACTED

4.1 GE NE

Forest Hatch, Manager, Services & Projects Quality
Kenneth W. Brayman, Manager, Quality Assurance Systems
Noel Shirley, Principal Licensing Engineer, Safety Evaluations Project
Elanor Schock, Program Manager, Safety Evaluations Project
Newell Metras, Dedication Testing Supervisor
Robert Thomas, Procurement Engineer (Retired)

4.2 D-Tech

Dick Tettman, President

APPENDIX A

TABLE 1: DIVESCO LIST OF SALES OF AHT/BLACK FOX ITEMS TO GE NE (1986 ON)

GE NE PO#	DATE	QTY	ITEM	DRAWING	GE NE CUSTOMER	SOURCE
205-86K317 AHT/BF	04/28/86	1	FC valve	112D1459P001	Perry (CEI)	
205-86J142 AHT/BF	05/14/86	2	MSIV blowers	213A3762P001	Clinton (IP)	
205-86K638 AHT/BF	10/02/86	1	Temp Element	159C4520P005	River Bend(GSU)	
205-86R687 AHT/BF	10/06/86	8	MSIVs	B21-F021 - 28	NMP2 (NMPC)	
205-86R689 AHT/BF	10/06/86	1	MSIV Blower	47B518664	NMP2 (NMPC)	
205-86R874 AHT/BF	11/17/86	2	Relief Valves	21A9508P001	Hatch (GP&L)	
205-87C630 AHT/BF	03/13/87	1	Temp Element	159C4520P005	Clinton (IP)	

TABLE 2: GE NE PURCHASES FROM DIVESCO FROM NRC SEARCH OF GE NE RECORDS

GE NE PO#	DATE	QTY	ITEM	DRAWING	GE NE CUSTOMER	SOURCE
205-85E813* *AHT/BF	unkn	unkn	Terminal Bd	147D7614G004,5	Perry (CEI)	

*Not listed on Divesco's 1985 sales records database

GE NE PO#	DATE	QTY	ITEM	DRAWING	GE NE CUSTOMER	SOURCE
205-85E949	07/05/85	2	Press XDCR	MPL:C85N001	Perry (CEI)	AHT/BF
205-85J769	11/25/85	1	Hyd Hand Pump	131C8966G001	Clinton (IP)	AHT/BF
205-86R686	10/03/86	1*	Agastat	145C3217P041	River Bend	*AHT/BF

(3 of 4 rtnd, 4th: *S/N 77231248 retained, baring GE DWG #, vice Agastat part number, E7024PB002, as were the two, S/Ns 85170022,3 from Control Components. Therefore, the one finally kept by GE could have been from AHT/BF)

GE NE PO#	DATE	QTY	ITEM	DRAWING	GE NE CUSTOMER	SOURCE
205-85N648	09/12/85	58	Anlg Isol	204B6220AAG002	Perry (CEI)	Allens Ck
205-87C632	02/03/87	4	CKT Cards	272A8614P101,02,12,20	RB(GSU)	Chism

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

July 10, 1995

Mr. Charles L. Perry, General Manager
ITT Barton
ITT Fluid Technology Corporation
900 South Turnbull Canyon Road
City of Industry, CA 91749-1882

SUBJECT: NRC INSPECTION NO. 99900113/95-01

Dear Mr. Perry:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of ITT Barton at City of Industry, California, conducted by Mr. R.C. Wilson of this office on June 12-15, 1995. The purpose of the inspection was to review activities conducted under your 10 CFR Part 50, Appendix B, quality assurance program and 10 CFR Part 21 reporting program. The inspection consisted of an examination of procedures and records, interviews with personnel, and observations by the inspectors.

The NRC inspectors found no instances where the implementation of your quality assurance program failed to meet certain NRC requirements. No response to this letter is required.

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.

Sincerely,

Donald P. Norkin for

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

Docket No. 99900113

Enclosure: Inspection Report 99900113/95-01

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

REPORT NO.:	99900113/95-01
ORGANIZATION:	ITT Barton ITT Fluid Technology Corporation 900 South Turnbull Canyon Road City of Industry, California 91749-1882
ORGANIZATIONAL CONTACT:	Jerald E. Anderson Director of Quality Assurance 818/961-2547
NUCLEAR INDUSTRY ACTIVITY:	Instrumentation such as Pressure, Level, and Flow Transmitters and Indicating Switches, and Valve Actuators
INSPECTION DATES:	June 12-15, 1995

INSPECTOR:

RCWilson — 7/10/95

Richard C. Wilson, Senior Engineer — Date
Vendor Inspection Section (VIS)
Special Inspection Branch (TSIB)

REVIEWED BY:

Gregory C. Cwalina — 7/10/95

Gregory C. Cwalina, Chief, VIS/TSIB — Date

APPROVED BY:

7-10-95

Robert M. Gallo, Chief, TSIB — Date

Enclosure

1 SCOPE OF INSPECTION:

ITT Barton supplies a variety of pressure, differential pressure, level, and flow transmitters and indicating switches, as well as valve actuators. The commercial nuclear portion of sales varies with the specific type of instrument, generally amounting to about 10% or less. Barton Industrial Sales in Glenwood, Illinois, also manufactures and supplies nuclear safety-related differential pressure units; that facility was not covered in this inspection.

The NRC inspector reviewed the implementation of selected portions of Barton's quality assurance (QA) program for supplying safety-grade components, and reviewed Barton's 10 CFR Part 21 program including reports that have been submitted to the NRC. The inspection bases were 10 CFR Part 50, Appendix B, and 10 CFR Part 21.

No violations or nonconformances resulted from this inspection. Within the inspection scope, the inspector found that adequate programs were in place, and that some improvements were being incorporated.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

No open findings remained from previous NRC inspections of Barton.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on June 12, 1995, the NRC inspector discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with Barton management and staff. In the exit meeting on June 15, 1995, the inspector discussed his findings and concerns with Barton management and staff.

3.2 Quality Assurance Program and Organization

The inspector selectively reviewed the Barton QA program established by the Quality Program Manual, Edition 1, Revision 2, dated August 17, 1994. The manual provided the top level requirements, with implementation covered in lower tier QA instructions. Although the QA manual was organized in ISO-9001 format, it was readily auditable and appeared to satisfy Appendix B to 10 CFR Part 50. Barton performed all activities under the same QA program, whether or not safety-related, and manufactured all instruments of a particular model in the same manner. A limited amount of additional documentation (e.g., testing and certification) was provided for nuclear safety-related POs.

Based on the review of the QA manual and instructions and discussions with the QA manager, the inspector determined that the QA organization had sufficient authority and organizational freedom to identify quality problems, initiate solutions, and verify implementation of the solutions.

3.3 Translation of Purchase Order Requirements

The NRC inspector selectively reviewed files for eight purchase orders (POs) to determine if the PO requirements were correctly translated into documented specifications, procedures, drawings, testing requirements, and products. The selected POs covered a variety of transmitter and indicating switch models, a valve actuator, and three types of replacement piece parts.

Barton assigned a Register Number (comparable to a shop order or work order number) to each PO, and identified each specific instrument by a Product Identification Code (PIC) number. The PIC numbers contained two-digit fields identifying the applicable version for each of the parts of the instrument, such as housing, bellows, fill, and range spring. In one PO, a Model 288A differential pressure indicating switch with a model 352 remote sensor was defined by an 18-field PIC number.

When Barton received an inquiry for a replacement instrument, the PIC number for the original equipment was retrieved as the definitive description of the replacement assembly. In an example reviewed by the NRC inspector, a 1995 inquiry covered a replacement for an indicating switch supplied in 1983. While reviewing the 1983 PIC number, Barton found a drawing revision that changed the original range. That information was provided to the customer for review, before PO placement. As a result, the original issue of the 1995 PO specified the proper PIC number and range for the instrument to be supplied in 1995. The NRC inspector considered Barton's practice of ensuring the accuracy of the original PO to be a positive feature of their program.

Barton engineering was preparing a detailed compilation of the PIC numbers covered by specific environmental qualification test reports. When completed, the list will replace the present practice of specifically reviewing the detailed characteristics of a specific configuration during the engineering review of customer inquiries. This approach will facilitate determination of environmental qualification pedigrees.

The NRC inspector concluded that Barton's controls effectively ensured that customer PO requirements were correctly incorporated into finished products.

3.4 Manufacturing and Testing

The NRC inspector witnessed various material handling and manufacturing operations, but no activity specific to safety-related POs was in progress during the inspection. The inspector witnessed the accuracy and repeatability portion of final calibration testing of a model 753 pressure transmitter for foreign use. The test procedure was the same as specified on the Certificate of Processes and Procedures for the PO. The inspector verified that parameters such as range agreed with the register sheet. All observed operations were in accordance with the calibration sheet and procedure. Barton personnel pointed out that the calibration procedure required previous elevated temperature testing; thus, each harsh environment instrument is actually exposed to its design basis temperature during final acceptance testing. The inspector concluded that Barton exercised appropriate procedural control over final acceptance testing activities.

3.4 Accuracy and Calibration of Test Equipment

The NRC inspector examined the calibration traceability of a model 200 differential pressure indicator, serial number 88063, shipped on April 13, 1995, under Duquesne Light Company PO No. D193001. This is the same instrument described in Section 3.3 above, where the range had changed since the original unit was shipped in 1983. The accuracy was specified on Barton's calibration certification sheet as ±½ % of full scale. This sheet showed the final acceptance test data for an ascending and a descending calibration run, and specified the applicable test instrument as Barton # 93-3-53. The calibration report for # 93-3-53 identified it as a Heise model CMM pressure gage, with NIST [National Institute of Standards and Technology] traceability through # 95-41-16, and provides data from a current calibration. The calibration report for # 95-41-16 covered a Heise model 179E/QBT transfer standard, which in turn was calibrated against a Ruska model 2465-751 deadweight tester. The deadweight tester was calibrated by the Ruska Instrument Corporation under Barton PO 37873. The Ruska calibration certificate identified the NIST test numbers for the Ruska standards used to calibrate the piston and masses of Barton's deadweight tester. The NRC inspector reviewed the report of an audit of Ruska by ITT Barton on June 23, 1992, and considered it adequate to dedicate Ruska's commercial grade calibration services by the standards of that time. The NRC inspector concluded that Barton's documentation adequately documented the calibration of the delivered pressure indicator.

The NRC inspector also reviewed Barton's report of a June 11, 1992, audit of SIMCO Electronics, which performs most of Barton's external calibrations, and had no concerns. Since Barton's procedures require triennial audits of calibration service suppliers, the inspector inquired about plans for future audits of these vendors. Barton stated that future vendor audits will be contracted out to EGS Corporation of Huntsville, Alabama. Barton mentioned that EGS has been audited by a licensee group.

3.5 10 CFR Part 21 Program

The NRC inspector reviewed Barton's procedure for reporting in accordance with 10 CFR Part 21: QA Manual Procedure QU-121, "NRC Regulations to 10CFR, Part 21," Revision 0, dated June 1, 1994. The procedure satisfied the requirements of 10 CFR Part 21, but it focused on evaluating deviations and failures to comply with the technical requirements of procurement documents, and only briefly addressed the identification and reporting of such concerns. The inspector suggested revising the procedure to emphasize such reporting. The inspector also pointed out that Barton would rarely have the plant-specific information necessary to perform the required evaluation of deviations, and suggested instead that the procedure concentrate on the five-day notification of customers addressed in 10 CFR 21.21(b), so that customers can perform the evaluation. The suggested changes would result in a shorter procedure that would better address the Part 21-related activities that Barton normally performs.

The NRC inspector noted that an October 1994 licensee group audit of Barton reported a finding related to a Barton internal deviation report: that Barton failed to complete evaluation of a deviation within the 60-day evaluation period required by 10 CFR Part 21, and failed to submit a timely interim report. In fact, Barton was still investigating the possible occurrence of a deviation, and had not yet determined that a deviation had occurred. As noted in the previous paragraph, the inspector discussed this confusion with Barton personnel.

Barton had recently initiated a charge to customers for safety-related equipment labelled a "10 CFR 21 configuration control engineering charge." The inspector pointed out to engineering and QA personnel that the configuration control activities actually apply to meeting the QA requirements of Appendix B to 10 CFR Part 50 as imposed by licensee POs for safety-related equipment or services, and not to the reporting requirements of 10 CFR Part 21. Even though Part 21 provides dedication guidance, the dedication activities are subject to Appendix B.

Review of selected specific issues, as detailed below, indicated that Barton's Part 21 reporting program was functioning properly. The inspector noted that copies of QU-121, 10 CFR Part 21, and section 206 of the Energy Reorganization Act of 1974 were properly posted. The inspector concluded that, subject to the clarifications discussed above, Barton's activities with respect to 10 CFR Part 21 appeared to be acceptable.

3.6 Review of Specific Part 21 Reports

- Switch chatter in Model 288A and 289A differential pressure switches - Interim Report dated December 19, 1994, and Final Report dated February 15, 1995 (NRC Log Nos. 94-324 and 95-052)

Barton began an engineering evaluation of mild environment equipment qualification in late 1994 as a result of a licensee group audit. During review of 1980 and 1986 seismic test reports, Barton determined that switch chatter may have occurred that was not detected. The specific instrumentation used to monitor contact chatter was not identified in the test reports, but was suspected of being an incandescent lamp. Barton conducted additional seismic tests early in January 1995, using instrumentation capable of measuring contact chatter as rapid as two milliseconds. The new testing showed that higher G levels, and setpoints very close to actual parameter values, produced the most chatter; there was no chatter at 4 G. Barton provided all affected customers with a table showing the duration of contact chatter as a function of G level and the proximity of the trip setpoint to the actual differential pressure value.

The NRC discussed preliminary test results with Mr. Anderson in a February 7, 1995, telephone call. After reviewing the seismic test report, the NRC again discussed the concern with Messrs. Anderson and Larson in a May 17, 1995, telephone call. The inspector briefly reviewed the concern during the inspection. The review of data from earlier seismic tests, and conducting additional tests, revealed a possible concern that had gone unnoticed for several years. The inspector considered Barton's activities including notifications to be acceptable, and no further action is required.

* Possibly unqualified relays in Model 288A differential pressure indicating switches - Interim Report dated October 5, 1994, and Final Report dated October 17, 1994 (NRC Log Nos. 94-266 and 94-271)

In January 1994 Barton supplied a model 288 indicating switch containing relays as a replacement for a unit sold in 1972. The sales department did not act on notification from engineering that the new relays were different than the original relays. The discrepancy was discovered during processing of a repeat order later in 1994 in accordance with procedure. Barton conducted additional seismic testing, which showed that the new relays constitute qualified replacements for the 1972 models.

The NRC addressed this concern during the May 17, 1995, telephone call and during this inspection. Barton personnel stated that several 1972 tests were customer- and lot-specific. Barton did not initiate a configuration control program for indicating switches containing relays until 1978. Subsequent 1980s seismic testing of model 288 switches did not include relays. The 1994 testing demonstrated equivalence of the new relays to those supplied in 1972, but Barton personnel stated that they still do not sell a qualified model 288A indicating switch containing relays. (The relays were sometimes added to increase the power handling capability of the output microswitches, or to provide more contacts.) An isolated failure of sales to act on engineering's review of a purchase inquiry caused the concern. The inspector considered Barton's subsequent actions including notifications to be satisfactory.

* Qualification limitations on all Series 200 differential pressure indicators - Barton Industry Advisory dated March 13, 1995, and transmitted to all affected customers (No formal Part 21 report to NRC and no NRC Log No.)

During Barton's engineering evaluation of mild environment equipment, engineering identified the possibility that confusion might exist concerning the various bellows fill fluids used in Model 200, 227A, 288A, and 289A differential pressure indicators and switches. The Industry Advisory stated that for applications below 40° F and less than 1 Mrad gamma, Barton had recommended an aqueous solution of ethylene glycol (B-fill). Qualification reports for these instruments, which did not include B-fill samples, showed a 3 Mrad radiation limit. The purpose of the industry advisory was to notify all affected customers that the 3 Mrad limit did not apply to B-fill, which was known to disassociate above 1 Mrad into gases which prevent proper operation.

After notification by a licensee on April 10, 1995, the NRC addressed this concern in an April 11 telephone call with Barton, and also during this inspection. Barton personnel stated that the qualification report clearly identified that the 3 Mrad limit covered testing of D-fill and M-fill units, and did not mention B-fill. Certifications specifically cited the test report. Records showed no instance of supplying B-fill where the PO specified a radiation requirement. The inspector considered Barton's actions including notifications to be satisfactory.

- Other Recent Part 21 reports -

The inspector considered other Part 21 reports submitted by Barton in the past five years and previously reviewed by the NRC, together with those discussed above, and concluded that no underlying root cause remained unaddressed.

3.7 Commercial Grade Item Dedication

Three of the eight licensee PO files reviewed by the NRC inspector covered dedication of commercial grade microswitches, O-rings, and bezel gaskets for nuclear safety-related use. In each case Barton had worked with the supplier to establish the desired parts characteristics, which essentially became the critical characteristics lists. The Barton microswitch source control drawing defined the desired characteristics of the purchased switch, which had a unique manufacturer's part number. Barton then selected switches which have a limited range of actuation force, and assigned a unique Barton part number to that group. All of the specified characteristics were verified by testing.

The elastomer supplier is audited triennially for the specific part numbers used. Barton is able to order custom production runs of O-rings because a distributor accepts excess quantity from production runs beyond Barton's needs. An oil-exposure test is used to verify lots of EPT (ethylene propylene terpolymer) elastomer, and a receipt inspection for dimensions is performed.

Barton was also improving the process for dedication of commercial grade piece parts. The basis of the new program is failure modes and effects analysis (FMEA) reports prepared for each instrument, covering all of the variations of all parts. The NRC inspector briefly reviewed the 36-page report covering model 764 differential pressure transmitters. The critical characteristics identified in this process, that are not verified by either the assembly process or an existing functional test of the part or a higher level assembly, are defined as "Barton Critical Characteristics." These characteristics will then be listed on the part drawing, and QA is procedurally responsible to define and perform verifications of them. At the time of the inspection Barton considered this process to be 85% complete, including all harsh environment equipment. The inspector considered the FMEA method of dedication evaluation to be a strength of Barton's program.

A licensee group audit of Barton in October 1994 identified the incomplete FMEA and critical characteristics lists for the series 200 differential pressure indicators used in mild environments, and also found errors in the FMEA report for the NH-90 Series actuators. This licensee activity appears to be adequately addressing the implementation of Barton's dedication program. The inspector also reviewed two individual licensee surveillances in 1995 that had no findings.

The NRC inspector considered Barton's dedication activities to be adequate.

4 PERSONNEL CONTACTED

+ * C.L. Perry, General Manager
+ * J.E. Anderson, Director of Quality Assurance
+ * T.W. Holdredge, Quality Manager
+ * D.L. Norman, Quality Assurance Administrator
+ * R.L. Krechmery, Director of Engineering
+ * J.K. Meyer, Engineering Manager, Nuclear Products
+ * M.K. Larson, Senior Staff Engineer, Nuclear Products
+ * M.P. Loo, Contracts Manager
+ J.E. Incotri, Marketing Manager
+ * T.E. Roide, Fabrication Manager
+ * G.M. Busch, Materials Manager
+ * L.F. Dropulic, Product Manager for Differential Pressure Units
+ * R. Einem, Product Manager for Actuators
+ * S. R. Goldberg, Product Manager for Electronics
R.W. Pownell, Metrology Engineer
T. Tran, Electronics Technician

+ Attended the entrance meeting on June 12, 1995
* Attended the exit meeting on June 15, 1995

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

August 23, 1995

Mr. R. Nim Evatt, President
and Chief Executive Officer
Liberty Technologies, Inc.
555 North Lane
Conshohocken, PA 19428-2208

SUBJECT: NRC INSPECTION REPORT NO. 99901225/95-01

Dear Mr. Evatt:

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection of your facility at Conshohocken, PA, conducted by Messrs. J.B. Jacobson and T. Scarbrough of this office on August 8 and 9, 1995, and the discussion of their findings with you at the conclusion of the inspection. The inspection was conducted to evaluate your actions with regard to open items identified during a previous NRC inspection (99901225/91-01), to review current technical issues pertaining to the use of the "VOTES" valve operation test and evaluation system, and to review Liberty Technology's implementation of requirements delineated in Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10CFR).

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.

The inspectors determined that you have taken appropriate actions with regard to previous NRC open items and have implemented an effective program for meeting the requirements of 10 CFR Part 21. Weaknesses, were however identified in your corrective action program for meeting the requirements of Criterion XVI "Corrective Action" of Appendix B to 10 CFR Part 50. Specifically, the inspectors determined that potential safety issues and nonconformances are not being uniformly documented within your quality program. Although evaluations are apparently being performed as safety issues arise, complete documentation of the evaluations was not avaible for review in some instances.

The inspectors also performed a limited review of your actions taken to validate the accuracy with which your new "Motor Power Monitor" equipment can predict motor actuator thrust at torque switch trip. With regard to this equipment, the inspectors identified a weakness in not comparing Motor Power Monitor readings against a known accurate source other than the VOTES equipment.

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 response 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. 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. If there are any questions concerning this inspection please contact Mr. Jeffrey B. Jacobson at (301) 415-2977.

Sincerely,

Robert M. Gallo

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

Docket No.: 99901225

Enclosures: 1. Notice of Nonconformance
2. Inspection Report No. 99901225/95-01

NOTICE OF NONCONFORMANCE

Liberty Technologies, Inc.
Conshohocken, PA

Docket No.: 99901225

Based on the results of an NRC inspection conducted on August 8 and 9, 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, states 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 properly identified and corrected.

   Liberty Technologies Quality Assurance Procedure No. QA-NCR-004, "Processing Safety Concerns," Revision 0, states in part, that any individual that discovers a condition that is, or is suspected of being, a safety concern shall complete a Safety Concern Evaluation.

   Liberty Technologies Quality Management System Process No. QMS-QA-06, "Nonconformance/Corrective Action Control," dated May 4, 1995, states in part, that all nonconformances to established procedures and errors in software, services, and management systems, be documented, processed and resolved correctly. The individual identifying the nonconformance shall issue a Nonconformance/Corrective Action Report.

   Contrary to the above, Liberty Technologies, Inc. did not initiate a Safety Concern Evaluation or a Nonconformance/Corrective Action Report for an issue involving a software virus nor for an issue involving potential inaccuracies of Votes equipment at low thrust values. (Nonconformance 99901225/95-01)

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

Dated at Rockville, Maryland
this 23rd day of August, 1995.

Enclosure 1

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

REPORT NO.:	99901225/95-01
ORGANIZATION:	Liberty Technologies, INC. 555 North Lane Conshohocken, PA 19428
ORGANIZATIONAL CONTACT:	Susan Yankanich, Quality Program Manager (215) 834-0330
NUCLEAR INDUSTRY ACTIVITY:	Liberty Technologies, Inc. supplies systems for testing and diagnosing the condition of motor operated valves.
INSPECTION DATES:	August 8 and 9, 1995

LEAD INSPECTOR: Jeffrey B. Jacobson, Special Inspection Branch — Date: 8/21/95

OTHER INSPECTORS: Thomas Scarbrough, NRR

REVIEWED BY: Donald P. Norkin, Section Chief, Special Inspection Branch — Date: 8-22-95

APPROVED BY: Robert M. Gallo, Chief, Special Inspection Branch — Date: 8/23/95

Enclosure 2

1 SUMMARY OF INSPECTION

1.1 Scope

This inspection was conducted to review Liberty Technology's response to open items identified during NRC inspection #99901225/91-01, to review current technical issues pertaining to the use of the "VOTES" valve operation test and evaluation system, and to review Liberty Technology's implementation of requirements delineated in Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10CFR).

1.2 Violations

No violations were identified during this inspection.

1.3 Nonconformances

1.3.1 Nonconformance 95-01-01

This nonconformance, described in sections 3.1.2 and 3.1.3 of the report identifies weaknesses in Liberty Technology's programs for implementing the requirements of Criterion XVI, "Corrective Action," of Appendix B to 10CFR 50.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Open Item 91-01-01 (CLOSED)

During a previous NRC inspection of Liberty Technologies (99901225/91-01) the inspectors identified that Liberty had not (1) validated particular uncertainty terms by testing, (2) provided clear overall system error information in the VOTES User's Manual, (3) included uncertainties in the use of MOV diagnostic equipment (such as rate-of-loading and torque switch repeatability effects) in the VOTES User's Manual, and (4) validated the overall system error by testing. These items were tracked as Open Item 91-01-01.

During this inspection the staff reviewed Liberty's actions in response to this open item. The previously identified weaknesses and the corresponding actions taken by Liberty since the 1991 inspection are summarized below:

1. Validation of Uncertainty Terms

   During the 1991 inspection, the inspectors noted that several uncertainty terms used in the error analysis for the VOTES diagnostic equipment had not been verified.

   With respect to the assumed error for machining tolerances of valve stems in the determination of effective stem diameter, Liberty conducted a testing program of valve stems with threads machined to the limit of the allowable tolerance. Using a finite element model, Liberty performed a study to determine the sensitivity of the effective stem diameter to thread tolerance. The VOTES software version 2.3 provides revised effective stem diameters. Liberty also has included a discussion of effective stem diameter in Addendum 5, "Stem Material Constants and Torque Correction," of the VOTES User's Manual.

With respect to the assumed error in the values selected for the modulus of elasticity and Poisson's ratio for various stem materials, Liberty performed a stem material study that indicated that the ratio of these values used in the VOTES software needed to be revised. On October 2, 1992, Liberty notified the NRC in accordance with 10 CFR Part 21 of this problem and the action taken to alert VOTES users. Liberty has included a discussion of the proper assumptions for modulus of elasticity and Poisson's ratio in Addendum 5 of the VOTES User's Manual.

With respect to the assumed effects on the modulus of elasticity and Poisson's ratio as a result of temperature changes, Liberty reviewed existing literature and increased the assumed error resulting from temperature changes. Liberty has revised the discussion of the VOTES Error Analysis in Addendum 4 of the VOTES User's Manual to address the increased error. During this inspection, it was identified that Liberty inappropriately combined the error to the assumed ratio of the modulus of elasticity and Poisson's ratio resulting from temperature changes with other errors through a square-root-of-sum-of-squares methodology. The error resulting from temperature changes is a biased error and therefore, should not be applied as a random error with other errors. The overall effect on the error analysis was found to be negligible.

With respect to the assumed error for changes in the modulus of elasticity of the yoke material caused by temperature changes, Liberty reviewed existing literature to provide additional support for the assumed error resulting from yoke temperature changes. This error also appears to have been combined as a random error with other errors. Again the overall effect on the error analysis was negligible.

Liberty has taken several actions to address the uncertainty intended to account for the non-linearity of the yoke, torsional effects of the yoke, stem directional effects, and other effects. Liberty has revised the VOTES software to allow for calibration of the yoke sensor using a curve-fit analysis to minimize yoke non-linearity effects. Liberty has prepared guidance for calibration of the yoke sensor when the stem is in tension (valve opening direction). In the October 2, 1992, Part 21 notice, Liberty discusses the increased error that could result from torque effects of the valve stem when the yoke sensor is calibrated based on strain of the threaded region of the stem. Liberty has revised the VOTES software to include a torque correction factor and discusses this issue in Addendum 5 of the User's Manual.

2. Overall System Error Information

During the 1991 inspection, the inspectors identified that the VOTES User's Manual did not discuss the basis for statistical uncertainty of the overall VOTES system error of 9.2 percent as determined by Liberty. Since then, Liberty has included a discussion of statistical uncertainty of the VOTES system error in Section 30-6, "Overall Thrust Measurement Accuracy," of the VOTES User's Manual.

3. Torque Switch Repeatability and Rate-of-Loading Uncertainties

During the 1991 inspection, the inspectors identified that the VOTES User's Manual did not discuss uncertainties resulting from torque switch repeatability and rate-of-loading effects when using the VOTES equipment. Since then, Liberty has included a discussion of these uncertainties in Section 30-7, "Factors Affecting Thrust at Torque Switch Trip," and in Addendum 4 of the VOTES User's Manual.

4. Validation of Overall VOTES System Error by Testing

During the 1991 inspection, the inspectors identified a lack of overall testing to verify the calculated VOTES system error. Following the 1991 inspection, Liberty participated in a testing program conducted by the MOV Users Group (MUG) of nuclear power plant licensees. The results of the MUG testing program supported Liberty's determination of the overall VOTES system error.

The inspectors concluded that Liberty had adequately addressed the issues identified in Open Item 91-01-01.

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Review of VOTES Technical Issues

The inspectors reviewed Liberty's evaluation of potentially significant MOV diagnostic equipment issues. As part of this review, the staff evaluated the implementation of the Liberty 10 CFR Part 21 program and Liberty's notification to VOTES users of significant information concerning the use of their MOV diagnostic equipment. The specific issues reviewed were as follows:

3.1.1 Extrapolation of Open Thrust Data

In May 1993, Liberty became aware of potentially large errors associated with thrust readings obtained from the VOTES equipment beyond the calibration range in the valve opening direction. Liberty initiated a "safety concern" evaluation to determine the safety impact on nuclear plant operation, the cause of the problem, and possible corrective action. Liberty considered the issue to not be safety significant, and not requiring a 10 CFR Part 21 notice because the torque switch is bypassed in the valve opening direction. However, Liberty believed that, in the long term, structural or motor output capability might be affected. Liberty issued Customer Service Bulletin (CSB)

31 (November 19, 1993), "Stem Tension," and CSB 31 Addendum (February 25, 1994), "Extrapolation Errors at 09 Quantified," to address the issue and to provide guidance to VOTES users. The inspectors concluded that this method of notification was appropriate for this issue.

3.1.2 Accuracy of VOTES equipment at Low Thrust Levels

At recent industry meetings, the reduced accuracy of MOV diagnostic equipment at low thrust levels compared to its accuracy at high thrust levels has been discussed. On June 16, 1994, Liberty issued Customer Service Bulletin 34, "Running Load Differences Between VOTES and Packing 'nForcer," that discussed observed differences in thrust at low thrust levels between VOTES equipment and their Packing 'nForcer equipment that is used for packing load measurements. At the end of the bulletin, Liberty stated that "even if a significant running load error were to exist in a VOTES trace, the affect on the static thrust margin is almost always negligible, and thus no corrective actions are deemed to be required."

During the inspection, Liberty agreed that the inaccuracy of the VOTES equipment at low thrust levels could be greater than the published value of 9.2 percent but was unable to provide documentation regarding the safety evaluation of this issue. The inspectors were concerned that additional inaccuracies at low thrust levels could be significant for MOVs with minimal thrust margin. The inspectors identified Liberty's failure to initiate a Safety Concern Evaluation or Nonconformance/Corrective Action Report as Nonconformance 95-01-01.

3.1.3 VOTES Virus

In 1994, Liberty discovered a virus in the VOTES software that prevented infected computers from operating. Liberty alerted VOTES users to the problem through the nuclear computer network and a problem report letter. Liberty has improved its software and procedures to reduce susceptibility to virus attack. Although Liberty responded to the virus problem, Liberty did not implement their procedures for evaluating potential safety concerns or nonconformances. This was cited as another example of Nonconformance 95-01-01.

3.2 Review of Motor Power Monitor

Liberty presented a summary of the features of its new Motor Power Monitor (MPM) diagnostic equipment. This equipment is designed to non-intrusively calculate actuator thrust output at torque switch trip under static conditions, by measuring motor current and voltage from the motor control center. The published accuracy of this equipment is 15 percent. In determining the accuracy of the MPM, Liberty compared thrust data obtained with the MPM to data obtained using its VOTES diagnostic equipment, for 230 valve strokes on 22 motor operated valves. The VOTES diagnostic equipment has a published accuracy of 9.2 percent. Review of the data seemed to support Liberty's claims of the MPM being accurate to within 15 percent; however, the inspectors saw a weakness in not comparing MPM readings against a known accurate source other than the VOTES equipment. The inspectors noted that MPM users will be expected to justify the accuracy of the MPM when used to make decisions regarding the operability of safety-related MOVs.

4 PERSONS CONTACTED

The NRC staff participating in the inspection and Liberty personnel contacted during the inspection are listed below. An (*) indicates individuals whom attended the exit meeting.

NAME	TITLE
* Jeffrey B. Jacobson	Inspection Team Leader, NRC
* Thomas Scarbrough	Senior Mechanical Engineer, NRC
* R. Nim Evatt	President and Chief Executive Officer, Liberty
* Susan Yankanich	Manager, Quality Programs, Liberty
* Paul J. Schott	Mgr., Nuclear Marketing & Int'l. Sales, Liberty
* Michael J. Delzingaro	Manager, Service Engineering, Liberty
Robert L. Leon	V.P. and Chief Technical Officer, Liberty

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

September 27, 1995

Mr. E.A. George, Jr., Vice President
Mid-South Nuclear, Inc.
40-B Sayerton Drive
Birmingham, AL 35202

SUBJECT: NRC INSPECTION NO. 99901270/95-01

Dear Mr. George:

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection of your facility at Birmingham, Alabama, conducted by Mr. U. Potapovs of this office on August 22 through 25, 1995, and the discussions of his 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) upgrading of material purchased from non-qualified sources (commercial grade item dedication), and (3) the implementation of your corrective action commitments resulting from the NRC inspection which was conducted on January 25 through 28, 1994.

The inspection was accomplished through objective evaluation of selected procedures and records, discussions, and observations by the inspector. The specific areas examined during the NRC inspection and the findings are discussed in the enclosed inspection report.

Our review of your activities in these areas indicated that, although significant improvements have been achieved in defining the methods of identification and verification of critical characteristics of commercial grade material, these methods do not always assure that material supplied and certified to ASME Code or 10 CFR Part 50, Appendix B quality programs complies with the procurement document requirements. 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.

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

Sincerely,

Robert M. Gallo

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

Docket No.: 99901270

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99901270/95-01

NOTICE OF NONCONFORMANCE

Mid-South Nuclear, Inc.
Birmingham, Alabama

Docket No.: 99901270/95-01

Based on the results of an NRC inspection conducted on August 22 through 25, 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 10 CFR Part 50 requires, in part, that measures shall be established to assure that purchased material conforms to procurement documents.

Section 3 of Mid-South Nuclear (MSN) Quality System Program, Revision 2, dated April 12, 1995, requires, in part, that applicable provisions necessary to meet customer purchase order (PO) requirements shall be included in appropriate documents or instructions and that the material to be supplied shall be processed in accordance with these documents.

Contrary to the above, the inspection identified the following examples where the established measures did not assure that material was supplied in accordance with the customer purchase order requirements. (Nonconformance 99901270/95-01-01)

1. TVA (Sequoyah) PO 95N5F-133214, dated April 21, 1995, for 24 feet of 1 1/2 inch diameter ASME SA-479, Type 316 bar to be supplied as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 2 material required the vendor to provide documentation that his quality program meets ASME Section III, Division 1, NCA 3800 current edition and addenda, and that the material has been supplied in accordance with the quality requirements of that program.

   MSN did not supply this material in accordance with their NCA 3800 quality assurance program. Instead, MSN certified the material as being supplied in accordance with ASME Section III, Article NC 2610 which exempts the material from most of the NCA 3800 quality assurance provisions. Additionally, paragraph NC 2610 limits the nominal crossection of bar stock that can be supplied under that paragraph to less than one square inch. The bar supplied exceeded this dimension.

2. TVA (Sequoyah) PO P95N5F-129724, dated February 22, 1995, for 32 internally threaded, one inch, Class 3000, ASME SA-105 pipe caps, required these caps to be supplied as ASME Section III, Class 2 material with the same quality program applicability statement as discussed in example 1, above.

Enclosure 1

MSN did not supply this material in accordance with their NCA 3800 quality assurance program. Instead, MSN certified this material as supplied in accordance with ASME Section III, paragraph NC 2610. Additionally, material hardness test records in the sales order file indicated that the hardness level of the caps supplied under this PO exceeded the maximum values permitted by the applicable material specification.

3. TVA (Sequoyah) PO P95N5-135199, for 80 feet (20 foot lengths) of ASTM A-36 angle iron (6 x 6 x .375 inch), included the provision that commercial material, procured from unqualified source and dedicated by the supplier must have all critical attributes (e.g. chemical, tensile, hardness) required by the applicable material specification independently verified.

   MSN certified the material provided under this PO as meeting the stated requirements without independently verifying that the tensile properties of the material complied with ASTM A-36 requirements.

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

Dated at Rockville, Maryland
this 27th day of September, 1995

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

REPORT NO.:	99901270/95-01
ORGANIZATION:	Mid-South Nuclear, Inc. 40-B Sayerton Drive P.O. Box 10063 Birmingham, Alabama 35202
ORGANIZATIONAL CONTACT:	E. A. George, Jr., Vice President
NUCLEAR INDUSTRY ACTIVITY:	Mid-South Nuclear. Inc. is a supplier of metal products to the nuclear industry.
INSPECTION DATES:	August 22 through 25, 1995

INSPECTOR:
Uldis Potapovs — 09-21-95 Date
Vendor Inspection Section (VIS)
Special Inspection Branch (PSIB)

REVIEWED BY:
Gregory C. Cwalina, Chief — 9/21/95 Date
Vendor Inspection Section
Special Inspection Branch

APPROVED BY:
Robert M. Gallo, Chief, PSIB — 9/27/95 Date

Enclosure 2

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspector evaluated Mid-South Nuclear Inc.'s (MSN) Commercial Grade Item (CGI) dedication process and assessed the effectiveness of MSN's corrective actions for nonconformances identified during the previous (January 25-28, 1994) NRC inspection. The nonconformances related to procedural and implementation deficiencies in MSN's CGI dedication program and improper application of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section III, "Rules for Construction of Nuclear Power Plant Components," (Section III) paragraph NX 2610(b) which provides for the exclusion of small parts from certain quality assurance program requirements. The evaluation included the review of selected sales orders and related documentation for safety related material processed after the 1994 inspection.

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)

- Section III of the ASME Code

- MSN's Quality Systems Program Manual and implementing procedures.

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

1.2.1 Nonconformance 95-01-01

This nonconformance, described in Sections 3.3.1 and 3.3.2 of the report identifies three examples where the implementation of MSN quality system program did not assure that material conformed to customer purchase order (PO) requirements due to: (1) Improper application of ASME Code, Section III paragraph NC 2610, (2) supply of material with hardness level in excess of specification limit, and (3) failure to perform tensile testing as required by the PO.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Violations

2.1.1 Violation 94-01-01 (Closed)

Contrary to Section 21.21, "Notification" of Title 10 of the Code of Federal Regulations (10 CFR), MSN failed to adopt a procedure to implement the provisions of 10 CFR Part 21 that were effective October 29, 1991.

By letter dated March 31, 1994, MSN advised the NRC that it had obtained a current copy of 10 CFR Part 21 and revised procedure SOP-601 to include the applicable requirements. No additional discrepancies were identified in this area.

2.2 Nonconformances

2.2.1 Nonconformance 94-01-02 (Closed)

Contrary to the requirements of 10 CFR 50, Appendix B and Section 3 of the MSN's quality system program, neither the MSN critical characteristics forms nor the sales orders for certain materials identified adequate critical characteristics and verification methods to ensure that the items being supplied met the customer procurement document requirements.

As discussed in letters dated March 31, 1994 and May 22, 1995, MSN has revised their procedure SOP-701 "Dedication of Commercial Grade Items" and most of the material critical characteristics forms to provide additional guidance for the identification and verification of critical characteristics. Review of this guidance indicated significant improvement, however weaknesses were identified in the revised program requirements (see discussion in paragraph 3.3.2)

2.2.2 Nonconformance 94-01-03 (Closed)

Contrary to the requirements of ASME Code Section III, Article NC 2610, MSN issued a certificate of compliance indicating that ASME SA-213, Type 304 tubing had been furnished to TVA in accordance with the requirements of NC 2610 without the required involvement of a Certificate Holder.

By letters dated March 31, 1994 and May 22, 1995, MSN advised the NRC that it had obtained their customer's (Certificate Holder's) consent to use the provisions of NC 2610 and that training had been provided to employees to review the requirements of "Certificate Holder" consent for customer orders under ASME Section III when utilizing paragraph NC 2610. The review of recent sales order files during this inspection, however, identified instances of misapplication of or improper certification to Section III, paragraph NC 2610. (See examples 1 and 2 of Nonconformance 99901270/95-01-01 and Section 3.3.1 of this report)

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance and Exit Meetings

During the entrance meeting on August 22, 1995, the NRC inspector discussed the inspection scope and developed general information about MSN's products and activities. During the exit meeting on August 25, 1995, the NRC inspector discussed his findings and observations with MSN's management.

3.2 Description of Facilities

MSN has been accredited by the ASME as a Material Organization, authorized to manufacture and/or supply ferrous and nonferrous bars, threaded fasteners, castings, forgings, plate, seamless fittings, flanges, NPT stamped tubular products, structural shapes, welding material, and similar items. The scope of their Quality Systems Certificate (QSC) also includes the qualification of material manufacturers and suppliers of subcontracted services and upgrading of stock material.

According to MSN management, approximately 85% of their products are supplied for safety related nuclear applications with carbon steel structural shapes providing the highest volume of material processed. MSN provides material under their ASME QSC as well as non-Code material under 10 CFR Part 50, Appendix B quality assurance requirements. Most of the Appendix B material is purchased as commercial grade and dedicated.

MSN does not perform any material manufacturing operations at their facility and does not warehouse material. MSN has on-site capability to perform hardness testing, hydrostatic testing, limited flattening tests, and visual and dimensional inspections. Chemical and mechanical testing is subcontracted to qualified laboratories.

3.3 Quality Assurance Program Implementation

3.3.1 Material Supplied to ASME Code Requirements

MSN's program for supplying ASME Code material is described in their Quality System Program Manual (QSPM), Revision 2, dated April 12, 1995 which is committed to meeting the requirements of ASME Section III, NCA 3800 as well as the applicable portions of NQA-1 and 10 CFR Part 50, Appendix B. The program controls are described in the manual and in referenced implementing procedures.

The NRC inspector reviewed several recent customer purchase orders and accompanying data packages and determined that, in most instances, the material appeared to have been supplied in accordance with the applicable QSPM provisions. Sufficient documentation was generally available to demonstrate compliance with customer purchase order and ASME Code requirements. However, the review identified inconsistencies in the processing and certifying material supplied under ASME Section III, paragraph NX 2610, including instances of improper application of the provisions of this paragraph. The review also identified inconsistencies and apparent contradictions in customer POs related to the acceptance criteria for material supplied under NX 2610 as illustrated in the following examples:

3.3.1.1 TVA (Sequoyah) PO 95N5F-133214, dated April 21, 1995, for 24 feet of 1 1/2 inch diameter ASME SA-479, Type 316 bar to be supplied as ASME Code Class 2 material required the vendor to provide documentation that his quality program meets ASME Section III, Division 1, NCA

3800 current edition and addenda, and that the material has been supplied in accordance with the quality requirements of that program.

MSN procured this material from an unqualified supplier in two pieces, removed test coupons from each piece and sent the coupons to a qualified laboratory for chemical and mechanical testing. The chemical analysis was performed on each coupon while tensile testing was done on coupon representing only one of the two bars. According to MSN, this practice is consistent with their CGI dedication program which permits one test on unverified heat lot based on the supplier's performance history. The material was certified as provided in accordance with ASME Section III, paragraph NC 2610 (small parts exclusion) which exempts the material from most of the NCA 3800 quality assurance requirements. MSN processes such material in accordance with their CGI dedication program.

The inspector noted that processing such material under MSN's CGI program, while consistent with the requirements of NC 2610, did not appear to satisfy the PO requirement that the material must be supplied in accordance with the requirements of their NCA 3800 quality program. It was also noted that the material processed under this PO exceeded the maximum size limit (one square inch cross section) of material that can be supplied under paragraph NC 2610 and, therefore, should have been supplied and certified under MSN's QSC (NCA 3800 program) in order to comply with the applicable ASME Code requirements. Improper application of the ASME Code, Section III, paragraph NC 2610 was identified as example 1 of Nonconformance 99901270/95-01-01.

Before the completion of the inspection, MSN requested that TVA clarify whether the statement in their POs which requires certification that material is supplied in accordance with MSN's NCA 3800 quality program precludes the use of paragraph NX 2610. TVA responded that the statement in question does not preclude the use of ASME Section III paragraph NX 2610 where applicable. According to TVA, the referenced PO paragraph is intended to indicate that the supplier is required to maintain a quality assurance program that meets ASME Section III, NCA 3800 requirements and that the material is supplied with all the required documentation.

The inspector noted that acceptance of the NX 2610 small parts exclusion while requiring certification to NCA 3800 program requirements appeared contradictory and that the maintenance of an NCA 3800 quality program had no effect on the material supplied if the material is not required to be supplied in accordance with that program. It was also noted that, according to accepted ASME practice, material supplied in accordance with NX 2610 can not be certified as produced under the Material Organization's QSC (NCA 3800 program).

3.3.1.2 TVA (Sequoyah) PO 95N5F-129724, dated February 22, 1995, item 2, for 32 Class 3000 ASME SA-105 internally threaded (NPT 1-inch) pipe caps specified that this material was to be supplied in accordance with ASME Code, Section III, Class 2 requirements.

MSN certified this material as supplied in accordance with ASME Code, Section III, paragraph NC 2610. MSN obtained the material through a distributor (Dodson Steel Products) from an unqualified manufacturer, Bonney Forge (BF) with Certified Material Test Reports (CMTR) from two heat lots. The material was upgraded using MSN's CGI dedication program. The upgrading consisted of performing chemical analysis of one sample from each heat lot and a hardness test (Rockwell B) on each piece. ASME SA 105 allows hardness testing (Brinell method) as an acceptable alternate for verifying tensile properties on forgings too small to permit obtaining a subsize tensile specimen when such forgings are produced on equipment unsuitable for the production of separately forged test bars. SA-105 specifies an acceptable hardness range of 137 to 187 Brinell (HB).

The inspector noted that the MSN test report indicated measured hardness levels of 93-96 and 92-96 Rockwell B, respectively, for the two heat lots of material supplied under this order. These ranges convert to 200-216 and 195-216 HB, which is significantly higher than the 187 HB maximum hardness permitted by ASME SA-105. Failure to assure compliance with applicable procurement document requirements was identified as example 2 of Nonconformance 99901270/95-01-01.

It was also noted that, as discussed in paragraph 3.3.1.1, above, although the customer's PO required the material to be supplied in accordance with the quality requirements of NCA 3800, MSN certified the material as supplied under NC 2610 which exempts the material from most of the NCA 3800 requirements.

3.3.1.3 TVA PO 95N2T-148585 for 66 feet of 2 inch, schedule 160, ASME SA-106, grade B pipe, specified this material to be supplied in accordance with ASME Code, Section III, Class 2 requirements. The PO also required the vendor to supply documentation that his QA program meets ASME Section III, NCA 3800, current edition and addenda, and that the material was supplied in accordance with this program.

MSN obtained this material by commercial grade purchase from M&R pipe supply, who purchased it from Texas Pipe & Supply Co. Inc., who, in turn, obtained the pipe from Koppel Steel Co. The material was supplied in three pieces with Koppel Steel CMTR which stated that Koppel had performed bend and hydrostatic tests on this material.

MSN upgraded this material in accordance with their CGI dedication program by performing chemical analysis and hardness test on each of the three pieces. Consistent with their CGI dedication program requirements for SA-106 material, no tensile testing was done. The results of Koppel Steel (unqualified vendor) hydrostatic and bend testing were accepted without validation. According to MSN, chemical analyses were performed on each piece of the material because MSN did not have a documented performance history of this vendor.

MSN provided a Certificate of Compliance (COC) for this material which indicated by "X" marks that the material was manufactured and processed in accordance with requirements which included:

ASME Section III NC 2610, 1989 edition
MSN QA Program, Revision 2, dated April 12, 1995
QSC 560, Expiration date May 5, 1998

The inspector noted that the COC was contradictory and misleading, since it certified that the material was processed in accordance with their ASME QSC (NCA 3800) while the supporting documentation shows that the material was processed under MSN's CGI dedication program. The inspector also noted that this was another example where the customer's PO required the material to be provided in accordance with MSN's NCA 3800 program but was supplied under paragraph NC 2610.

3.3.2 Commercial Grade Item Dedication Program

MSN's program for purchasing and dedicating commercial grade material is described in Procedure SOP-701, Dedication of Commercial Grade Items. The current revision is Rev. 6, dated September 8, 1994. This procedure is used for supplying ASME Code material under the small parts exemption of Section III paragraph NX 2610 and for supplying all safety related material to the requirements of 10 CFR Part 50, Appendix B. The procedure has been revised since the last NRC inspection to address some of the concerns identified during that inspection. Additional guidance is provided for the identification and verification of critical characteristics. The procedure references Form 701, "Material Critical Characteristics Form" for the identification of critical characteristics and verification methods applicable to different materials and product forms. SOP-701 also requires justification for the selected critical characteristics to be identified on Form 701. This is accomplished by referencing a justification code. Engineering evaluations for all justification codes are compiled on Form 701B, "Critical Characteristics Selection Engineering Justification Code."

Although the revised procedure required additional testing to verify material conformance to the applicable specification, certain materials and product forms were permitted to be dedicated based on an "indirect verification" method. This method utilizes hardness testing to verify that material tensile properties conform to the specification requirements. The method was limited to mild steel products for which approximate hardness versus tensile strength relationships are shown in ASME SA-370. Additionally, MSN has compiled extensive test data to support this relationship.

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

August 22, 1995

Mr. William A. McCloy, President
Power Distribution Services
9870 Crescent Park Drive
West Chester, OH 45069

SUBJECT: NRC INSPECTION REPORT 99901286/95-01

Dear Mr. McCloy:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Power Distribution Services, Inc. (PDS), West Chester, Ohio, conducted by Messrs K.R. Naidu, J.L. Knox, and R. Mendez on June 19-22, 1995. The inspection was conducted to provide a basis for NRC staff confidence that the switchgear manufactured by PDS for 4.16-kV Yaskawa circuit breakers would perform their intended safety function. This report also discusses an observation made during a January 25-28, 1995, inspection at Wyle Laboratories, Huntsville, Alabama, when the qualification testing activities related to the 4.16 kV switchgear manufactured by PDS were in progress. On June 22, 1995, at the conclusion of the inspection, the inspectors discussed the findings with you, other members of your staff, and representatives of National Technical Systems (NTS).

During this inspection, the team evaluated the NTS/PDS quality program that was established to implement the provisions of 10 CFR Part 50, Appendix B, and the provisions of 10 CFR Part 21 in selected areas during the manufacture of the switchgear cubicles. Within these areas, the NRC team (a) examined technical documentation, procedures and representative records, (b) held discussions, (c) listened to presentations and (d) observed PDS technicians working activities.

During the evaluation of your activities at West Chester, the team noted the proactive approach being taken by your staff to correct adverse customer findings. The team noted positive PDS employee attitudes and technical expertise that were shown by the personnel who were interviewed during the inspection. However, the team observed that PDS personnel are experiencing difficulties adapting to a written quality program.

Based on the results of the inspection, the inspectors found that the implementation of the NTS/PDS program failed to meet NRC requirements as specified in the enclosed Notice of Nonconformance. Specifically, the team identified an inadequacy in the control of purchased materials.

Please provide us within 30 days from the date of this letter a written statement in accordance with 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 response 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 Paper Reduction Act of 1980, Public Law No. 96-511.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's public document room.

Should you have any questions regarding this matter, please do not hesitate to call.

Sincerely,

Robert M. Gallo

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

Docket No.: 99901286

Enclosures: 1. Notice of Nonconformance
2. Inspection Report

cc: Mr. Gregory M. Ruegger
Nuclear Power Generation
B14A
Pacific Gas & Electric Company
77 Beale Street, Room 145
P.O. Box 77000
San Francisco, CA 94106

Mr. M. Basu
Electrical Project Eng.
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Mr. D.R. Michaud
Product Manager
NTS
533 Main Street
Acton, MA 01720

NOTICE OF NONCONFORMANCE

Power Distribution Services, Incorporated
West Chester, Ohio

Docket No.: 9901286
Report No.: 95-01

Based on the results of a U.S. Nuclear Regulatory Commission (NRC) inspection conducted at the Power Distribution Services, Incorporated (PDS), West Chester, Ohio, facility on June 19-22, 1995, it appeared that one of your activities were not conducted in accordance with NRC requirements.

> Criterion VII, "Control of Purchased Material, Equipment, and Services" of Appendix B to Part 50 of Title 10 of Code of Federal Regulations, (10 CFR 50) states, in part, "Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source and examination of products upon delivery."
>
> Contrary to the above, PDS failed to establish appropriate measures to control purchased material in Standard Operating Procedure (SOP) No. 02, Revision 1, "Purchasing Materials & Services for Nuclear Orders." Specifically, there were no provisions to utilize the same technical description of material equipment or services in the purchase order that had been approved by the National Technical Services (NTS)/PDS staff in the "Nuclear Purchase Requisition." Furthermore, the measures did not require that purchase orders for safety-related items be issued only to vendors listed in the NTS approved vendors list.

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include (1) a description of steps that have been or will be taken to correct this item, (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 22nd day of August, 1995.

Enclosure 1

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

REPORT NO.:	99901286/95-01
ORGANIZATION:	Power Distribution Services, Inc. 9870 Crescent Park Drive West Chester, OH 45069
ORGANIZATIONAL CONTACT:	Mr. J.L. Bachman (513) 777-4445
NUCLEAR INDUSTRY ACTIVITY:	Fabricating 4.16 kV switchgear cubicles and reconditioning low voltage metal-clad circuit breakers
INSPECTION DATES:	June 19-22, 1995

LEAD INSPECTOR: [signature] 8/17/95
Kamalakar R. Naidu, Team Leader — Date
Vendor Inspection Section (VIS)

OTHER INSPECTORS: Rogelio Mendez, Region III
John L. Knox, NRR/EELB

REVIEWED BY: Gregory C. Cwalina 8/24/95
Gregory C. Cwalina, Chief, VIS — Date
Special Inspection Branch (PSIB)

APPROVED BY: Robert M. Gallo 8/22/95
Robert M. Gallo, Chief, PSIB:DISP — Date
Division of Inspection and Support Programs (DISP)

Enclosure 2

1.0 SUMMARY OF FINDINGS

During this inspection, the inspection team evaluated the National Technical Services,/Power Distribution Services, Inc. (NTS/PDS) quality program and its implementation during the fabrication of 4.16 kV retrofit switchgear which includes 4.16-kV, 350-MVA (million volt-amperes), SF_6 Gas Fluopac Series, Rotary-arc, circuit breakers manufactured by Yaskawa Electric Corporation (Yaskawa), Japan, intended for Pacific Gas and Electric Company's (PG&E's) Diablo Canyon Power Plant (DCPP). The Yaskawa circuit breakers have a higher short circuit interrupting capacity than the existing GE breakers (350 versus 250 MVA), require less maintenance, and are compact enough to fit into the existing stationary GE cubicles.

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, Appendix B)
- Part 21, "Reporting of Defects and Noncompliance," of 10 CFR.

One nonconformance was identified and is discussed in Paragraph 3.4.2 of this report.

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of this vendor.

3.0 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

During the entrance meeting on June 19, 1995, the NRC inspection team discussed with PDS and NTS staff the scope of the inspection, the areas to be reviewed, and established the persons to contact within PDS and NTS management and staff. During the exit meeting on June 22, 1995, the NRC inspection team summarized its findings and concerns to the management and staff of PDS and NTS. Persons contacted during this inspection are identified in Section 4.

3.2 Background

NTS issued Purchase Order (PO) No. 36986 dated March 20, 1994, to PDS for the project management, engineering, quality assurance, production testing, manufacturing and technical labor associated with the supply of vertical-lift drawout cubicles with 4.16-kV, 350-MVA, SF_6 Yaskawa circuit breakers. PDS fabricates metal enclosures to permit the installation of 4.16-kV Yaskawa circuit breakers and other components into the existing stationary GE cubicles at DCPP. NTS provides the quality assurance (QA) coverage. In January 1995, NTS contracted Wyle Laboratories, Huntsville, Alabama, to subject a representative switchgear bay to DCPP site-specific seismic qualification tests. The test specimen consisted of three GE stationary cubicles with retrofit drawout cubicles manufactured by PDS with 4.16-kV Yaskawa circuit breakers.

In addition to the work being performed for DCPP, PDS has received three 480 Volt GE metal-clad circuit breakers from the Waterford Nuclear Station (Waterford) and one similar circuit breaker from the River Bend Nuclear Station (River Bend) for complete reconditioning. PDS informed the inspectors that it has submitted the procedures that it had developed to perform the required re-conditioning to Waterford and River Bend and is awaiting necessary approvals.

PDS fabricates electrical device enclosures using material, components, and sub-assemblies from other equipment manufacturers. It also assembles and supplies remanufactured low, medium and high voltage switchgear for various commercial power generation and distribution companies. PDS also provides a variety of services for non-nuclear electric utility companies including testing and maintaining protective and power apparatus (circuit breakers, starters, transformers, network protectors, relays, and electrical conductors from 600 V through 765 kV).

3.3 10 CFR Part 21 Program

PDS personnel informed the inspectors that PDS and NTS entered into a Teaming Agreement, which was expressly developed to enable PDS to manufacture the vertical-lift drawout retrofit cubicles for DCPP. According to the agreement, PDS implements the NTS/PDS quality program during the manufacture of the retrofit switchgear, and NTS maintains the 10 CFR Part 50, Appendix B, QA program and 10 CFR Part 21 reporting responsibilities as defined in the NTS Quality System for safety-related activities performed by PDS personnel on NTS nuclear orders. The NTS/PDS quality program requires NTS to review all Nonconformance/Corrective Action Reports (NCARs) initiated by PDS during the manufacture of the DCPP switchgear for Part 21 reportability.

The team reviewed the location and the adequacy of documents posted at the PDS facility pursuant to 10 CFR 21.6. PDS posting consisted of Section 206 of the Energy Reorganization Act of 1974 and a notice in accordance with 10 CFR 21.6(b). The inspectors determined that the posting was acceptable.

3.4 QA Program Implementation

The NTS/PDS Quality Manual (QM) provided programmatic guidelines to supplement the NTS Quality Assurance Manual as applicable to PDS activities. NTS/PDS developed standard operating procedures (SOPs) to implement the written program. NTS trained selected PDS individuals in quality control inspection techniques to implement the NTS/PDS Quality Program and the SOPs and certified them as PDS Quality Control Inspectors. NTS personnel perform quality assurance functions. The inspectors selected the following SOPs for review.

3.4.1 SOP No. 1, Revision 1, dated April 17, 1995, "Nuclear Control Reviews," describes the methodology by which PDS reviews, and approves NTS nuclear orders. The inspectors observed that this procedure does not explicitly state that change orders to purchase orders (POs) should receive the same degree of review and control as the original PO. PDS initiated NCAR 95-29 on June 22, 1995, to revise the SOP clarifying the requirement.

3.4.2 SOP No. 2, Revision 1, dated April 17, 1995, "Purchasing Materials & Services for Nuclear Orders," describes the preparation, review, approval, issuance and verification process of procurement documents by PDS for NTS nuclear orders. The inspectors observed that there were no provisions in SOP No. 2 requiring the purchasing agent to transcribe the same technical description that NTS approved in the "Nuclear Purchase Requisition" (NPR) prepared by PDS into the PO.

According to the SOP, when a NPR is prepared, the technical description of the material is reviewed and approved by the NTS/PDS staff. However, there is no requirement for the same technical description to be transcribed into the purchase order. Paragraph III.B of SOP No. 2 states, in part, "Completed and PDS approved requisitions are forwarded to NTS for review and approval... NTS will review the requisitions per the relevant supplier file at PDS," assuring that the description of the purchased material is controlled. Paragraph III.C of the procedure which discusses the preparation of the PO states, in part, approval of the resulting PO and all associated paperwork by NTS is indicated by signature and/or quality stamp and date on the hard copy of the resulting P.O....a copy of the NTS approved requisition and PO must be filed in "P.O. Requirements Review Sheet" and indicate approval of each requisition by signature, initials, or quality stamp and date for all nuclear purchases. A copy of the "P.O. Requirements Review Sheet,"...will be sent to Purchasing for the preparation of the actual hard copy P.O. However, the SOP does not require the transcription of the technical description that had been previously approved in the NPR into the PO. The inspectors informed the PDS staff that they were concerned that the intent of the review and approval of the technical description of a component or material in the NPR is defeated if the same technical description is not restated in the PO.

Additionally, Paragraph H of SOP No. 2 did not require the purchase of safety-related material from the NTS approved vendors list (AVL). Instead, Paragraph H only discusses the control of the NTS AVL.

As noted above, the inspectors were concerned that the purchase of materials and services cannot be adequately controlled if safety-related material is purchased from a vendor not listed on the NTS AVL, and if the PO does not use the same description of material and services that was specified and approved in the NPR. The inspectors identified to NTS/PDS an instance where PDS issued a purchase order for cable to a vendor not listed on the NTS/PDS AVL, and the technical description in the PO was different from the NPR. Details of the procurement of the cable are discussed in Paragraphs 3.7 and 3.8.6 of this report. The inspectors identified the failure to establish adequate measures to control purchased materials and services as a nonconformance. (Nonconformance 95-01-01)

3.4.3 SOP No. 3, Revision 1, dated April 17, 1995, "Standard Receiving, Handling, Storage and Shipping," describes the methodology to assure PDS purchased materials and services are properly received, handled, inspected, and stored. The inspectors were concerned that the procedure did not provide guidance on accepting certificates of conformance (CoCs) (from manufacturers instead of distributors,) to detect the various fraudulent, or otherwise unacceptable products that has entered the nuclear industry and did not mention the numerous information notices issued by the NRC on this subject. For instance, during receipt inspection of cable, the PDS Receipt Inspector did not identify that the CoC was unacceptable because it was from a cable distributor instead of the cable manufacturer. PDS concurred with the inspectors and initiated NCAR 95-29 in which the corrective action recommends indoctrination on NRC information notices on fraudulent or otherwise unacceptable products.

3.5 Control of Measuring and Test Equipment

During a January 25-27, 1995, inspection at Wyle Laboratories, (Wyle) Huntsville, Alabama, an NRC team observed some of the qualification tests being performed on the 4.16 kV switchgear manufactured and supplied by PDS/NTS. During the testing, the inspectors observed that NTS/PDS used a relay test set (RTS) which was not within its current calibration schedule. The calibration due date on the RTS had expired in December 1994. NTS/PDS used this RTS to check the calibration of the protective relays mounted on the breaker cubicles. The PG&E representatives stated that the purpose of the seismic testing was to specifically qualify the PDS retrofit drawout cubicles with Yaskawa breakers and not the relays mounted on the stationary cubicles. However, subsequent discussions indicated that PG&E had intended to seismically qualify the entire breaker cubicle including the instruments and relays mounted on the cubicle. However, due to the poor performance of the induction relays during seismic testing, the DCPP licensee decided to use solid state protective relays. The PG&E representative informed the inspectors that the PDS switchgear successfully withstood the qualification tests at Wyle.

During the current inspection, the inspector reviewed control of measuring and test equipment (M&TE) and concluded that PDS had an acceptable M&TE program. The inspectors noted that the M&TE was of the proper range, type, accuracy and tolerance. In addition, M&TE was calibrated, utilizing standards traceable to the National Institute of Standards and Technology.

Even though the M&TE program was generally acceptable, the inspectors observed a weakness related to the root cause analysis of a calibration problem. During a calibration check the calibration laboratory (GE Electronic Services) found that a AC/DC power supply (manufactured by Phenix Technologies) had exceeded its ± 1.0% tolerance. The calibration of the DC portion required an accuracy of ± 1.0% of full scale voltage. NTS/PDS initiated NCAR 95-14 to document that the accuracy of the power supply exceeded the tolerance. In the disposition, NTS/PDS stated that exceeding the ±1.0% tolerance for the DC voltage range was acceptable because it was within the ±3.0% accuracy listed in the M&TE master equipment list and took no further action. In independently reviewing the root cause, the inspectors observed that the power supply had four different power supply functions, each with its own calibration accuracy. PDS had erroneously selected the ±3.0% tolerance which was applicable to the AC power supply range even though it does not use this function to take measurements. At the inspectors request, NTS/PDS personnel re-examined this matter and concurred with the inspectors that the original disposition of NCAR 95-14 was incorrect because it extracted the erroneous ±3.0% tolerance from the M&TE master equipment list. After the discovery, NTS/PDS issued a revision to NCAR 95-14 requiring a review of previous jobs where the power supply with the incorrect accuracy was used to determine if there were any adverse affects. Additionally, before the conclusion of the inspection, PDS corrected the M&TE master equipment list specifying the accuracy of this power supply as ±1.0% of the full scale voltage and reopened NCAR 95-14, Revision 1, which initially identified the incorrect calibration of the power supply.

The inspectors identified no problems other than a weakness in the investigation of the root cause of a problem.

3.6 Pacific Gas and Electric Company (PG&E) Audit of PDS

The inspectors reviewed the results of an audit performed at PDS by PG&E on February 21-24, 1995. Replying to a question from the inspectors regarding the timeliness of the PG&E audit, the PG&E project engineer stated that PG&E wanted to audit PDS after the completion of the seismic qualification tests of the prototype retrofit breakers and before PDS commenced the manufacture of the safety-related switchgear. The inspectors considered PG&E's reply acceptable. PG&E conducted the audit to verify that PDS had effectively implemented the NTS/PDS quality program during the fabrication of the specimen retrofit 4.16-kV switchgear that was tested at Wyle. PDS was contracted to manufacture a total of 105 identical Class 1E drawout cubicles with 350-MVA, 4.16-kV circuit breakers rated for 1200 and 2000 Amperes which will meet or exceed the quality of the cubicles that successfully withstood the seismic qualification tests. The audit focused on the following areas:

- dedication and fabrication of stock material
- dedication of parts and components
- receiving inspection and test of Yaskawa circuit breakers
- welding and assembly of the NTS/PDS circuit breaker cubicles
- production testing of the NTS/PDS circuit breakers.

The audit was very comprehensive and identified eight findings. In a letter dated April 12, 1995, NTS acknowledged PG&E's audit findings and responded to them outlining actions planned to correct them. Actions taken to correct PG&E's adverse audit findings included revising the NTS/PDS Quality Manual and the standard operating procedures (SOPs) that PDS uses to implement the program, and reassigning specific quality functions to NTS personnel.

No problems were identified in this area.

3.7 Review of Purchase Orders

The team selected the following purchase orders (POs) issued by PDS to examine the implementation of the NTS/PDS Quality Program in areas related to the control of purchased materials. The receipt inspections for these purchased items are discussed in Section 3.8.

- PO No. 10472-HQ, dated May 18, 1995, to Hillman Fastener, Cincinnati, Ohio, for the supply of various hardware.

- PO No. 9819-HQ, dated February 17, 1995 to Monti, Cincinnati, Ohio, for various shapes of metallic components, fiberglass-reinforced polyester angle type GPO3.

- PO No. 10764-HQ, dated June 19, 1995, to Century Springs, Los Angeles, California, for 110 Type ASTM A 227 springs.

- PO No. 9826-HQ, to Copper & Brass Sales, Detroit, Michigan, for the supply of several pounds each of 3/8" x 3" rectangular ASTM B 187, Alloy 110, full round copper bus bar, and 1/8" x 1-1/2" rectangular ASTM B 187, C 110, full round edge bus bar.

- PO No. 9818-HQ, dated February 17, 1995, to Central Steel & Wire Company for the supply of various sizes and shapes of bus bar material.

- PO No. 9384, dated December 20, 1994, to Anixter Southern, Cincinnati, Ohio, which stated, "600 Volt Tefzel (EFTE) Insulated Wire 14 AWG, Single Conductor 14/19 black."

SOP No. 2 is the applicable procedure for the preparation and issuance of these POs. The inspectors reviewed the implementation of this procedure. According to SOP No. 2, the first step is for PDS to prepare a "Nuclear Purchase Requisition," (NPR) with the technical description of the item. The second step is for NTS/PDS to review the NPR for the adequacy of the technical description of the item and, if acceptable, approve it. The next step is for PDS to transcribe the technical description of the item into the PO.

The inspectors determined that the technical description of the cable in the NPR, which was reviewed and approved on December 15, 1994, was different than the description stated in the PO. The NPR stated "#14 AWG, 600 V w/flame retardant per [attached] description." The attached description stated "Control cables shall have adequately sized stranded conductors, no less than #14 AWG, and at least 600 V insulation with highly flame retardant characteristics. Tefsel [sic] or specially flame retardant type SIS insulating and jacketing compounds of neoprene, hypalon, or flame retardant XLPE/XLPO are acceptable. The cables shall be approved by PG&E prior to wiring by the supplier."

The description in the PO, which was prepared from the above NPR, stated "600 VOLT TEFREL (EFTE) INSULATED WIRE 14 AWG SINGLE CONDUCTOR 14/19 BLACK." This PO was approved by NTS QA. The PO also contained standard instructions to the effect that the items checked on the "Attached Purchase Order Requirements" sheet were an integral part of the PO, and that the material ordered under this purchase order was classified as a "critical item" and required a receiving inspection beyond the standard receiving criteria. This material is not to be "accepted" (or tagged as such) until the receiving inspection of critical items has been satisfactorily completed in accordance with NTS/PDS SOP No. 3.

In the "Purchase Order Requirements Sheet" attached to the PO, the annotated requirements stated "items must be new, not used, refurbished, altered or repaired and must be free from defects, all supplied items must be received in standard manufacturer packaging which is unopened and unaltered, all supplied items shall have a uniform configuration and appearance that is in accordance with any applicable manufacturer specifications or PDS drawings/purchase order." The PO required a certificate of conformance (CoC), signed by a vendor authorized individual other than someone in sales, marketing, or customer service, attesting to the quality of all supplied items.

Other than inadequate measures to control purchased materials, which has been identified as a nonconformance in Paragraph 3.4.2, no further problems were identified in this area.

3.8 Control of Purchased Material

The inspectors reviewed the process through which NTS/PDS controlled purchased materials. PDS personnel performed receipt inspections on material received using SOP No. 3, Revision 1, and NTS Work Procedure 60431-95N-1466-FAS, Revision 1, June 2, 1995, to inspect the material and document the results of the receipt inspection in NTS/PDS "Standard Receiving Report." Accepted material is identified and kept in a pending status awaiting detailed inspections to accept or reject it. Acceptable material is then transferred to its designated permanent location. NTS Quality Assurance inspectors performed detailed inspections and dedicated the commercial-grade items for use in safety-related applications. During the detailed inspections, NTS inspectors establish a sample size depending on the lot size utilizing guidance provided by the Electric Power Research Institute (EPRI Report No. NP-7218, "Guideline For The Utilization of Sampling Plans For Commercial-Grade Item Acceptance (NCIG-19)."

3.8.1 Regarding the hardware received from Hillman Fasteners, PDS performed a receipt inspection and documented the observations in a "NTS/PDS Standard Receiving Report" dated June 20, 1995. The attributes verified were:

a. The material received appeared new, uniform, unused, not altered, not tampered with, and not repaired or refurbished.

b. The packing slip establishes traceability of the received material to the point-of-manufacture.

c. The adequacy of the packaging, cleanliness, identification/marking, workmanship and vendor documentation was also verified and documented.

The inspectors verified that for bolts, the PDS inspectors examined the marking on the head of the bolt, and used a Go-No-Go thread gauge to verify the correct size of the threads on the samples. NTS/PDS sent some specimens to Massachusetts Material Research for special tests, such as chemical composition and hardness.

3.8.2 For various shapes of fiberglass-reinforced polyester angle type GP03, procured from Monti, the critical characteristic was dielectric strength and the failure mode for these components was identified as pinholes. The acceptance is verified when the installed components successfully withstand a dielectric strength voltage of 19+.5 kV for 60 seconds.

3.8.3 For the steel springs received from Century Springs, the acceptance criteria is provided in Paragraph 4.1.3.5, "Helical Springs" of NTS Procedure 60431-95N-1466-Bar, Revision 0, "Receipt Inspection and Sampling Procedure for Safety Related Bar Stock and Components For the PG&E Units." The technical specifications for the 110 steel springs stated in the PO are: outside diameter: 0.625", inside diameter: 0.510", length: 3.00", 15 coils, 9.16 pounds per inch spring rate, 1.496" deflection, closed ends, zinc plated. When the springs are received, PDS quality control inspectors verified the overall length of the spring. NTS quality assurance personnel used Procedure No. 60431-95N-3, Revision 1 to dedicate the springs. The procedure focusses on the compression strength on a random sample of the springs to provide assurance that the springs will have equivalent performance with those specimens that successfully withstood the seismic tests.

3.8.4 For the various shapes of copper received from Copper & Brass Sales, PDS personnel verified that the dimensions for each piece of bar stock met the PO requirements, measured the resistance, and hardness (Rockwell B [HRB 75]). NTS identified the critical characteristic of the bar stock to be resistivity, and NTS established the acceptability of the round copper bus bar to ASTM B 187, by correlating the hardness numbers to the resistivity. NTS quality personnel measured the hardness of the copper and compared them to the acceptance values established by NTS.

3.8.5 For the various bus bar components received from Central Steel and Wire Company, PDS/NTS determined the acceptability by measuring the hardness and comparing them with predetermined values.

3.8.6 PDS received 600 Volt Tefzel 14 AWG, single conductor 14/19 black cable from Anixter Southern with a certificate of conformance (CoC) from Basic Wire and Cable, Chicago, Illinois. The CoC was addressed to Anixter Southern, and stated "It is herewith certified that all articles in the quantities as called for in your purchase order No. 850-120498-861 are in conformance with requirements, specifications and drawings listed on that order." In the NTS/PDS "Standard Receiving Report" for the cable, the PDS receipt inspector identified no unacceptable findings and the NTS quality assurance person noted that the vendor was not on the NTS approved vendors list (AVL). In paragraph 3.4.2 of this report, the inspectors identified a nonconformance relative to the inadequate control of purchased material.

The inspectors were also concerned that SOP No. 2 does not provide guidance to distinguish between a distributor and an equipment manufacturer and does not provide sufficient guidance on scrutinizing the authenticity of certificates of conformance (CoCs). NTS/PDS initiated NCAR 95-29 and included in it a corrective action to revise SOP No.2 and to indoctrinate the staff so that personnel who issue purchase orders can precisely define the type of CoCs that are acceptable to help quality control inspectors to recognize genuine CoCs.

The inspectors identified that PDS procured cable from a vendor not listed on the NTS AVL and accepted a CoC from a vendor who was a distributor and not the manufacturer of the cable.

Other than inadequate measures to control purchased materials and services, which has been identified as a nonconformance in paragraph 3.4.2, no further problems were observed in this area.

3.9 Review of the Circuit Breaker Dedication

PDS purchased a total of 132 SF_6 Rotary-Arc Yaskawa circuit breakers. Of these, eight 2000-Ampere (A)-rated and ninety seven 1200-A rated circuit breakers are intended to perform safety-related functions at DCPP. PDS used the remaining breakers for seismic testing, manufacturing non-Class 1E cubicles for DCPP, and spares.

The inspectors reviewed the following NTS Procedures to assess the effectiveness of the implementation of the NTS/PDS quality program in the areas of electrical design, receipt inspection, and production testing requirements and to assess their conformance with ANSI/IEEE recommended practices for circuit breakers.

- NTS Procedure No. 60431-95N, Revision 1, of May 1, 1995, "Dedication/Acceptance Basis for Class 1E Retrofit Circuit Breakers, 4 kV, 350 MVA for Diablo Canyon Power Plant Units 1 & 2 Pacific Gas & Electric Company."

- NTS Procedure No. 60431-95N-1466-RI, Revision 1, "Receipt Inspection/Test Procedure for Yaskawa Circuit Breakers, Type: 5GYB1-1200-350, 5GYB-2000-350."

- NTS Procedure No. 60431-95N-1466-CPT, Revision 2, "Conversion Production Test Procedure for PDS SF Retrofit Circuit Breaker, Types: 5GYB1-1200-350 and 5GYB1-2000-350."

The inspectors reviewed the adequacy of the receipt and production test programs with respect to their demonstrating the functional capability of selected component parts of the procured Yaskawa SF_6 circuit breaker. The breaker's expulsion membrane, and pressure switch were selected for review. PG&E indicated that the essential purpose of these components is to maintain the integrity of the SF_6 insulating medium for the circuit breakers' main contacts. This purpose is demonstrated by performance of an insulation dielectric test. NTS/PDS performs dielectric tests on the circuit breaker as part of the final production tests. PG&E also indicated that dielectric tests would be repeated as part of Diablo Canyon site receipt and periodic test programs.

The inspectors found the program procedures to be consistent with ANSI/IEEE recommended practices and PG&E's requirements. The inspectors identified no concerns in this area.

3.10 Observation of Assembly Activities In Progress

The inspectors toured the PDS fabrication areas where the DCPP switchgear was being assembled. During the tour, the inspectors observed incoming material staging, and storage areas. In these areas, the inspectors examined Yaskawa SF_6 circuit breakers, completed receipt inspection/test data sheets for the Yaskawa circuit breakers, prototype enclosures that had been used for seismic testing which contained the adapted Yaskawa SF_6 circuit breakers for use at Diablo Canyon.

During the tour, the inspectors noted that electrical control cables on the prototype enclosures were routed next to sharp edges and in the vicinity of moving parts. The inspectors expressed concern that this routing could, over time, cause chafing and failure of the cable's insulation system. In response, PDS stated that the control cables when installed in the production enclosures would be reconfigured, routed, and tie-mounted to the enclosure such that the cable's insulation system will not be exposed to chafing from sharp edges or moving parts of the converted Yaskawa SF_6 circuit breaker. Even though a completed production enclosure (with the proposed cable routing installed) was unavailable for inspection, the inspectors concluded that the proposed reconfiguration, routing, and tie-mounting is feasible and is standard industry practice, and can be performed such that the cable and the cable's insulation system will not be subjected to conditions or stresses for which they are not designed. In addition, given the passive nature of cable systems, the inspectors concluded that the proposed control cable reconfiguration, routing, tie-mounting (although different from that used in the prototype enclosures) will not affect the seismic qualification of either the prototype or production enclosures.

The inspectors examined the welds on two drawout cubicles and observed that the size, length and location of the welds met the drawing requirements. The inspectors reviewed the qualifications of the weld procedure and the welders and determined them acceptable. The inspectors reviewed the NTS/PDS procedure No. 11A, "Procedure for Structural Welding and Weld Inspection," and observed that the procedure implied that welders themselves could evaluate rejected welds. Paragraph IV.D of the procedure states in part, "The welder and inspector shall disposition welds as 'accept', 'reject' or 'rework', record the results, sign and date the Weld data sheet." The NTS/PDS representatives concurred with the inspectors that the deletion of "the welder" from the sentence of this paragraph would minimize confusion. On June 22, 1995, NTS/PDS issued NCAR 95-29 to delete "the welder" from the procedure.

Other than a weakness in the NTS/PDS procedure No. 11A, which was being corrected, the inspectors did not identify any unacceptable findings in this area.

3.11 Review of Training Records

SOP No. 10 describes the control of qualification activities for personnel at PDS. The inspectors reviewed the training documents and observed the following weaknesses:

- The agenda of the training was not detailed.

- It was not clear if the inspectors were trained on what constituted a valid certificate of conformance (CoC) or what was required for a CoC to be valid.

- The topics that were used to discuss the numerous ways to detect fraudulence were not documented, and the generic communications that the NRC had issued on fraudulence, (e.g. Bulletin 88-10) were not mentioned.

NTS/PDS informed the inspectors that even though they had discussed these issues they had not documented them in an auditable form. Before the conclusion of the inspection PDS initiated NCAR 95-29 to document the topics discussed during training sessions.

The inspectors informed PDS personnel that NRC issued Generic Letter 89-02 stressing the importance of personnel performing safety-related activities being trained in the detection of fraudulent material. There were no records at PDS to indicate that such information was collected and used in training sessions on fraudulent material known to have been previously supplied to the nuclear industry to educate the individuals on the significance of CoCs, to stress the importance of verifying the authenticity of CoCs, and to enable the inspection personnel to detect fraudulent or otherwise unacceptable material during receipt inspections. NTS/PDS informed the inspector that action to enhance training requirements will be included in NCAR 95-29 and records will be developed to reflect the training.

Other than some weakness in the depth of training, the inspectors did not identify any unacceptable findings in the training area.

4.0 PERSONS CONTACTED

	Name	Title
Power Distribution Services, Inc. (PDS)		
† *	J.E. Bachmann	Quality Control Inspector
† *	J.L. Bachmann	Assistant to the President
*	E.J. Kuehne	Senior Vice President
† *	J.P. McCloy	Vice President Operations
† *	W.A. McCloy	President
† *	T. Miracle	Plant Manager
† *	D.R. Robling	Manager, Technical Services
National Technical Services (NTS)		
† *	F.W. Bean	Quality Representative
† *	W.E. Copeland	Quality Technical Specialist
† *	J.E. Dozier	Quality Manager
† *	D.T. Grand	Site Engineer

† *	M.E. Lilly	Quality Representative
†	D.R. Michaud	Division Program Manager
† *	M.P. Saniuk	Engineering Manager

Pacific Gas and Electric Company (PG&E)

† *	M. Basu	Project Engineer
†	R.A. Carvel	Supplier Assessment Auditor

* Individuals who attended the entrance meting on June 19, 1995.
† Individuals who attended the exit meeting n June 21, 1995.

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

July 5, 1995

Mr. Mark Van Sloun
Vice President and General Manager
Rosemount Nuclear Instruments, Incorporated
12001 Technology Drive
Eden Prairie, MN 55344

SUBJECT: NRC INSPECTION NO. 99900271/95-02

Dear Mr. Van Sloun:

This letter transmits the report of the inspection conducted by Mr. Stephen Alexander of this office and Mr. S.V. Athavale of the Instrument and Control Branch from April 5 to 7, 1995, at your facilities at Eden Prairie and Chanhassen, Minnesota. At the conclusion of the inspection, the findings were discussed with you and the members of the Rosemount staff identified in the enclosed report. In telephone conversations and telefax messages subsequent to the inspection, your staff provided additional information relevant to the inspection that is documented in the report.

Areas examined during the inspection are identified in the report. They included (1) an assessment of the validity and comprehensiveness of the methods by which Rosemount researched its records and determined the serial numbers of the nuclear transmitter sensor modules that potentially contained Monel isolators (initially from the lot used in the failed transmitters from St. Lucie) and the customers to whom these modules or transmitters containing these modules were supplied, (2) a review of Rosemount's supplemental measures taken to determine if the isolator lots identified in the initial search were the only Monel isolators to be inadvertently used in nuclear transmitters, (3) an examination of the isolator assembly manufacturing process and the circumstances surrounding the original error in selecting Monel foil strip stock to make 1152/3/4 foil disc assemblies, (4) an examination of the circumstances surrounding the identification and documentation of the apparent error and the ultimate inappropriate disposition of the discrepancy report, (5) a review of the quality control measures subsequently established that would minimize the probability of such errors, and (6) a review of testing and root cause analysis thus far and of design information relating to the exclusion of Monel from applications with a high hydrogen concentration environment. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

During this inspection, we determined that the implementation of your quality assurance (QA) program failed to meet certain NRC requirements. The nonconformance cited was for failure to prevent inadvertent use of certain nonconforming parts and failure to take adequate corrective action by inappropriate disposition of a discrepancy report identifying the use of the nonconforming parts. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter.

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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.

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

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

Sincerely,

Robert M. Gallo

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

Docket No. 99900271

Enclosures: 1. Notice of Nonconformance
2. Inspection Report No. 99900271/95-02

cc w/encl: Paul Blanch
135 Hyde Road
West Hartford, CT 06117

Ernest Hadley, Esquire
414 Main Street
Post Office Box 3121
Wareham, MA 02571

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

REPORT NO.:	99900271/95-02
ORGANIZATION:	Rosemount Nuclear Instruments, Incorporated 12001 Technology Drive Eden Prairie, Minnesota 55344
ORGANIZATIONAL CONTACT:	J. Valley Quality Assurance Manager
NUCLEAR INDUSTRY ACTIVITY:	Rosemount manufactures and supplies nuclear qualified pressure and differential pressure transmitters to most of the commercial and government nuclear facilities market.
INSPECTION DATES:	April 5 through 7, 1995
LOCATION(S):	Engineering offices at the Eden Prairie Facility; sensor cell manufacturing facility at Chanhassen, Minnesota
LEAD INSPECTOR:	Stephen D. Alexander Vendor Inspection Section (VIS) Special Inspection Branch (TSIB)
	6/29/95 Date
OTHER INSPECTORS:	S.V. Athavale Instrument and Control Branch Division of Engineering
	7/31/95
REVIEWED BY:	Gregory C. Cwalina, Chief, VIS/TSIB
	6/30/95 Date
APPROVED BY:	Robert M. Gallo, Chief, TSIB/DOTS
	7/5/95 Date

Enclosure 2

NOTICE OF NONCONFORMANCE

Rosemount Nuclear Instruments, Incorporated — Docket No. 99900271
Eden Prairie, Minnesota — Report No. 95-02

Based on the results of an NRC inspection conducted on April 5-7, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

A. Criterion XV, "Nonconforming Material, Parts or Components," of 10 CFR Part 50, Appendix B, states: "Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, disposition, and notification to affected organizations."

Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, states, in part: "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, measures established by Rosemount to control certain nonconforming material did not prevent its inadvertent use. In addition, action taken to correct a condition adverse to quality was inadequate in that Discrepancy Report 491585, written October 30, 1989, was dispositioned inappropriately by the Material Review Board. The discrepancy report documented the use of strip stock material that was not in accordance with the bill of materials to make Lot 20 of Part No. 01153-0252-0042 disc assemblies. The Material Review Board inappropriately dispositioned the discrepancy by directing that the strip stock part number and lot number on the traveller be corrected. As a result, Lot 16 of C10181-0014 Monel foil strip stock, documented as having been used to make Lot 20 of the disc assemblies, was actually so used, yet the traceability data on the traveller was erroneously changed to read Lots 23 and 24 of Part No. C09851-0011 (316L stainless steel foil strip stock) (95-02-01).

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Chief, Special Inspection Branch, Division of 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) the reason for the nonconformance, or if contested, the basis for disputing the nonconformance, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further noncompliances, and (4) the date when your corrective action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 5th day of July, 1995

Enclosure 1

1.0 SUMMARY OF INSPECTION FINDINGS

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, "Reporting of Defects and Noncompliance," of 10 CFR

- Rosemount Quality Assurance (QA) Program Documents and Procedures

The inspectors reviewed historical documents relating to the manufacturing process used for replacement sensor cells provided to licensees for the fill oil loss problem. The inspectors examined Rosemount's factory procedures and process for receiving, inspecting, and punching metal foil strip stock to be used for sensor cell isolator diaphragms; handling, cleaning, and welding of discs to weld rings; and testing of finished isolator assemblies. The inspectors interviewed engineers, QA and quality control (QC) personnel, and factory workers (instrument builders) to gather data relating to past manufacturing and QA/QC errors. The inspectors also reviewed followup documentation; Rosemount's methodology of identifying, scoping, and bounding the lots of suspect transmitters; Rosemount's efforts to obtain the services of the Southwest Research Institute (SwRI); and Rosemount's efforts to provide required support to its affected customers.

The inspectors determined that Rosemount's methodology to bound the problem with the affected transmitters acceptable for transmitters made with isolators from the same lot as those that failed at St. Lucie. The inspectors concluded that production control measures subsequently established by Rosemount should have precluded such errors since the incident in question and should continue to do so in the future. The inspectors noted that Rosemount was supporting replacement of the suspect sensor cells on a first priority basis. With stepped up production and diversion of resources to this project from those with less urgent needs, Rosemount estimated that lead time for replacement units could be reduced from its normal 12-week period to as low as 2 weeks.

1.1 Violations

None

1.2 Nonconformances

(95-02-01) Contrary to the requirements of Criterion XV of 10 CFR Part 50, Appendix B, Rosemount's measures to control nonconforming parts did not prevent the inadvertent use of certain transmitter sensor cell isolator diaphragms of the wrong material. Contrary to the requirements of Criterion XVI, Rosemount took inadequate corrective action in that the disposition of a discrepancy report, written October 30, 1989, that identified the use of the wrong material cited above, was inappropriate in that the disposition stated was to change the part number on the traveller rather than verify what material was actually used and to take the steps necessary to capture any incorrect material and prevent its use in transmitters designed for nuclear safety-related service (hereinafter referred to as nuclear transmitters). (See Section 3.5 of this report.)

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous findings were reviewed during this inspection.

3.0 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

3.1.1 Identifying Event

On November 22, 1994, Florida Power and Light Corporation's (FP&L's) St. Lucie Unit 1 Nuclear plant (St. Lucie), suffered an inadvertent safety injection (SI) event when two of the four Rosemount Model 1153 nuclear pressurizer pressure transmitters failed with high outputs during repressurization of the reactor coolant system following a full depressurization for an outage. The two high outputs removed the manual SI block (imposed during shutdown) and the two normal outputs, transmitting the actual (low) system pressure, caused SI initiation. Gas entrapped in the fill oil cavities of the transmitter sensor cells was the apparent cause of the failures. To address this concern, Rosemount undertook a root cause analysis, and also performed examination and testing of the failed transmitters using the services of SwRI which had the capability of handling potentially radiologically contaminated material.

3.1.2 SwRI Tests and Analysis

The tests and analysis indicated that (1) the gas entrapped in the sensor cells was pure, diatomic (molecular) hydrogen, (2) there was no evidence of process inleakage, (3) there was no evidence of fill oil decomposition, and (4) the failed transmitters had Monel Alloy 400 isolating diaphragms instead of Type-316L stainless steel (316L) isolating diaphragms that are supposed to be used for the nuclear grade (Types 1152, 1153, and 1154) transmitters.

3.1.3 Postulated Failure Mechanism

The SwRI test results suggested the following postulated failure mechanism: The entrapped hydrogen gas came out of solution upon depressurization of the plant. Upon repressurization, the coalesced hydrogen bubbles displaced fill oil from the process side cavity of the sensor cell into the center chamber, deflecting the sensing diaphragm (capacitor plate) and causing the high output signal. In addition, the hydrogen dissolved in the fill oil altered the fill oil dielectric constant, also causing a high output signal. The postulated source of the entrapped molecular hydrogen was recombination of monatomic hydrogen that most likely diffused through the Monel isolating diaphragms. Monel is known to be highly permeable to hydrogen. The source of the externally produced monatomic hydrogen was not yet conclusively established at the time of preparing this report. However, the two prevailing theories are

(a) that the monatomic hydrogen was in the process fluid (reactor coolant) in the pressurizer as a minority constituent of the relatively large amount of hydrogen expected to be in the pressurizer of a pressurized water reactor plant such as St. Lucie, or (b) that the monatomic hydrogen was generated on the isolator surface as a product of general corrosion of the wetted metal surfaces of the isolator and weld ring, and/or galvanic corrosion in a cell formed of the Monel isolator diaphragm (cathode) and the stainless steel weld ring with the coolant acting as electrolyte. It is also possible that both of these postulated phenomena can occur.

3.1.4 Rosemount Internal Investigation

Prompted by SwRI's finding that the entrapped gas was hydrogen, and confirmed by the finding that the St. Lucie isolator diaphragms were made of Monel, Rosemount's in-house investigation revealed that due to a manufacturing error compounded by a QA/QC error in 1989, sensor cells having isolation diaphragms manufactured using Monel instead of 316L were used for as many as 451 Type 1152, 1153, and 1154 nuclear grade transmitter sensor modules of Range Codes 6 through 10. The modules were being supplied to nuclear utilities or being used to repair transmitters returned by utilities for various reasons, including correcting the fill oil loss problem (as was the case with St. Lucie). Rosemount identified affected manufacturing lots of the modules, and issued a notification to all affected licensees or purchasers pursuant to 10 CFR 21.21(b) on March 21, 1995. On March 22, 1995, the NRC issued Information Notice 95-20, "Failures in Rosemount Pressure Transmitters Due to Hydrogen Permeation Into the Sensor Cell." Appendix A to this report gives the names of the affected organizations in the Rosemount Part 21 notification, and Appendix B to this report contains the chronology of events from the incident at St. Lucie until the inspection documented in this report.

3.1.5 Regulatory Response Groups' Responses

On April 4, 1995, in support of the NRC's examination and tracking of this issue, and in preparation for this Rosemount inspection, the inspector reviewed the responses to this issue submitted by the Regulatory Response Groups (RRGs) of the Westinghouse Owners Group (WOG), the Boiling Water Reactor Owners Group (BWROG), the Babcock & Wilcox Owners Group (B&WOG), and the Combustion Engineering Owners Group (CEOG). These responses were submitted in reply to NRC activation of the RRGs and questions on affected Rosemount sensor modules and transmitters. Some safety assessments were based on assumptions as yet unconfirmed regarding the primary source of entrapped hydrogen and its solubility in the transmitter fill oil. The B&WOG response in particular pointed out that one explanation why transmitters had not yet failed in a similar application to St. Lucie (i.e., pressurizer pressure transmitters at Florida Power Corporation's (FPC's) Crystal River Plant) was that Crystal River had much longer instrument lines (and hence, longer hydrogen diffusion lengths) on its pressurizer pressure instruments than those on the St. Lucie pressurizers. The inspector noted that underlying this explanation for a failure of the type in question not yet occurring at Crystal River is the assumption that the coolant in the pressurizer is the primary source of monatomic hydrogen operative in this failure mode. In addition, without documenting its basis, this same utility asserted that 500 psig was the pressure threshold for hydrogen coming out of solution in major pressure excursions or transients during plant operation. These questions were addressed with Rosemount during the inspection.

3.2 Entrance and Exit Meetings

During the entrance meeting for this inspection, held April 5, 1995, at Rosemount's Eden Prairie, Minnesota, facility, the inspectors met with Rosemount management and discussed the scope and objectives of the inspection. During the exit meeting on April 7, 1995, with Rosemount management, the inspectors summarized the inspection findings.

3.3 Inspection Details

3.3.1 Rosemount Root Cause Analysis

3.3.1.1 SwRI Results

The inspectors reviewed Rosemount's root cause analysis. SwRI test reports which were issued on March 21, 1995, indicated that SwRI performed analyses to identify the types of gases trapped under the diaphragm, determined moisture content of the oil and performed electron micrography of both inner and outer surfaces of high pressure diaphragms showing the center and middle areas including weld beads, and performed dark field micrography of cross sections of both diaphragms. In addition, electron dispersive spectroscopy (EDS), semi-quantitative elemental analysis of high and low pressure side diaphragms along with the weld ring material and inductively coupled argon plasma (ICP) spectrography on high pressure side diaphragms were performed to identify and analyze the materials. Results of the SwRI tests were that (1) gas trapped under the isolator diaphragm was identified to be pure molecular hydrogen and no other gases were found; (2) moisture content was not found in the fill oil nor other evidence of process in-leakage, (3) there was no evidence found of oil decomposition, and (4) the elements (and their quantities) identified in the weld rings were consistent with 316L stainless steel and the elements (and their quantities) in the diaphragms were consistent with Monel Alloy 400.

3.3.1.2 Source of Hydrogen

There are several postulated sources of the entrapped hydrogen. The two most prevalent theories postulate an external source of the hydrogen entrapped in the transmitters: (1) hydrogen in the process fluid (coolant) diffusing or leaking into the sensor cell and (2) hydrogen generated by corrosion reactions. In addition, the possibility of internally generated hydrogen by fill-oil decomposition reactions or reactions with coolant that might have leaked into the transmitters were investigated by SwRI for Rosemount.

If the primary source of monatomic hydrogen available for diffusion through the Monel isolating diaphragm was from the process fluid, then the first occurrence of this particular failure being in transmitters exposed to coolant from the pressurizer is consistent with the expected hydrogen content of that coolant. Monatomic hydrogen would be expected to be present in the coolant as a minority constituent of the diatomic (molecular) hydrogen in the coolant. The molecular hydrogen (H_2) comes from chemical addition to the coolant and radiolytic decomposition of the coolant water. In this case, coolant in the pressurizer and especially in the steam space during operation would be the only plant location expected to contain sufficient excess hydrogen to cause significant hydrogen diffusion and resulting transmitter failure. Therefore, transmitters with Monel isolating diaphragms in other locations in the plant with less hydrogen or none at all in the process fluid, would not be expected to fail as a result of process hydrogen intrusion. The coolant as the primary source of the entrapped hydrogen is also consistent with the single reported instance of failures thus far because the rate of hydrogen diffusion through Monel is a function of temperature (relatively hotter at or near the pressurizer) as well as external hydrogen concentration and the time to failure is a function of the hydrogen diffusion length through the process fluid in instrument line dead legs.

However, if the primary source of monatomic hydrogen operating in this failure mode is from galvanic action and general corrosion, both producing hydrogen atoms on the surface of the Monel isolator, then other locations in the plant could be susceptible. The galvanic action could result from the intimate contact of dissimilar metals - in this case, the Monel diaphragm and the 316L weld ring - in the presence of an electrolyte, the coolant. Such a galvanic cell would produce about 0.5 volt, with the Monel being the cathode. Rosemount's reported commercial experience with noticeable detrimental hydrogen diffusion through Monel has primarily been in environments with relatively high hydrogen concentrations as opposed to the relatively small amounts that may be produced by a galvanic cell and by general corrosion.

The inspectors discussed with Rosemount the possibility that as moisture may intrude into the fill oil at a high pressure, a chemical reaction of water molecules with the silicon oil would yield pure hydrogen and silicon dioxide. The reaction would be driven to completely consume the limiting reactant, the water. Tests for moisture therefore should include looking for silicon dioxide precipitate on internal surfaces, particularly the diaphragm. The inspectors inquired whether SwRI testing revealed any amount of silicon dioxide deposits in oil or on the inner surface of the diaphragm. Prompted by this question, Rosemount reported having contacted SwRI during this inspection and discussed the issue with the SwRI test technician who had inspected the internals of the sensor cells. The SwRI technician was reported to have told Rosemount that although SwRI did not look specifically for silicon dioxide; if present, it would most likely have shown up in the tests conducted by SwRI. Rosemount considered this further evidence (and the inspectors agreed) that moisture did not penetrate into the oil in the failed St. Lucie transmitters. Therefore, the pure hydrogen found in the oil was not likely to have been generated internally by reaction of the oil with moisture.

SwRI's dark field micrographs indicated some small fractures in the interior of the weld in addition to the heat-affected grain boundary zones. No corrosion, other fractures, or other indications of possible leakage paths were found. The inspectors questioned whether these fractures might have provided a path for hydrogen leakage into the fill oil, but Rosemount determined that such a leakage path was highly unlikely because the weld fractures did not appear to penetrate the weld. Furthermore, if weld fractures could provide an in-leakage path, there likely would have been evidence of moisture intrusion and certainly a penetrating crack would also provide a leakage path out. Rosemount concluded (and the inspectors agreed) that leakage out would be inconsistent with the amount of entrapped gas being sufficient to cause the deformation or distention of the isolator diaphragms observed on the failed transmitters at St. Lucie.

The SwRI testing also revealed that since refurbishing of the transmitters, specific quantities and material properties of the fill oil of the failed transmitter had changed very little, compared to a fresh sample of the fill oil, indicating that fill oil breakdown did not occur. Therefore, SwRI and Rosemount concluded that the hydrogen found in the oil was not a result of the fill oil breakdown.

3.3.1.3 Comparison with Similar Applications

With regard to the few cases in which transmitters identified with Monel diaphragms did not fail, such as the third transmitter on the affected pressurizer at St. Lucie or transmitters at Crystal River, Unit 3, which were also exposed to operating conditions similar to the failed transmitters at St. Lucie, Rosemount stated that if transmitters in similar situations did not exhibit signs of failure, that does not mean that they may not be close to failure. Rosemount further explained that there may be other factors specific to individual installation configurations which influence the failure rates of transmitters otherwise exposed to similar operating conditions. An example of these factors, cited by Florida Power Corporation in its RRG response, would be the longer instrument lines at Crystal River as compared to the corresponding shorter lines at St. Lucie.

3.3.1.4 Factors Affecting Failure Mode and Probability

The inspectors inquired whether Rosemount had generated a mathematical model to simulate conditions of hydrogen permeation in the sensor cell to provide information about the direction and amount of signal shift for each of various connection configurations and about any precursors of failure. Rosemount responded that development of mathematical models to predict transmitter failure probability was not being considered at the time of the inspection. With respect to solubility of hydrogen (both H and H_2) in the Dow-Corning 704 silicone-based oil that Rosemount uses as transmitter fill oil, Rosemount stated that Dow-Corning had solubility data for helium in this oil, but not for hydrogen.

However, Rosemount had performed an analysis to determine the direction of transmitter drift for various transmitter configurations. The conclusion was that for pressure (absolute or gauge) transmitters, entrapped gas would always cause transmitter output to fail high because (1) gas displacement of fill oil during repressurization would cause hydraulic deflection of the sensing diaphragm (in the center of the cell) in a direction that would produce high output, and (2) hydrogen in the dielectric fill oil would reduce the effective dielectric constant of the oil, producing the same effect. In differential pressure transmitters, the sign or direction of the error introduced by entrapped gas would depend on which side of the sensor cell had more hydrogen and on the relative pressures. To address the question of the possibility for differential pressure cells having isolators of different materials on either side, Rosemount stated that in its manufacturing process, a single lot of sensor cells may be built using isolator assemblies from different lots (as was the case with the mixed lots), but that individual cells are supposed to be built from only one lot. In other words, the last isolator assembly in a lot would not be welded to one side of a cell; rather use of a new isolator assembly lot would be started and the single isolator scrapped. This practice would preclude having a cell with a Monel isolator on one side and another material on the other. Therefore, Rosemount concluded that different isolator materials in the high and low pressure sides of a differential pressure transmitter sensor cell was not a factor to be concerned with.

3.3.1.5 Selection of Isolator Diaphragm Materials

The inspectors also reviewed design information relating to selection of 316L for hydrogen environment applications and to the basis for excluding Monel from those environments. The inspectors concluded that although 316L was not selected for nuclear applications on the basis of low hydrogen permeability, Monel was normally excluded from high hydrogen environments because of known high hydrogen permeability. The inspectors determined that Rosemount has considerable commercial experience with Monel in hydrogen environments and this experience compelled Rosemount to avoid using Monel in such applications. Rosemount's problems with Monel in hydrogen environments is documented in Rosemount Technical Report 28210B, "Transmitter Damage by Hydrogen Generation and Diffusion," dated March 10, 1982. In addition, although it is not certain what references, research papers, or other information, formed the basis for the original design decision not to use Monel in hydrogen applications, the current design engineer has several such references in his files which confirm the commonly held notion or conventional wisdom (also cited in Rosemount Report 28210B) that nickel alloys tend to have a high hydrogen permeability. Rosemount's collective knowledge on corrosion (including hydrogen problems) is also published in Rosemount Technical Data Sheet (TDS) 3045A00, "Corrosion and Its Effects" (current edition dated January 1995).

3.3.2 Root Cause Conclusions and Recommendations

At the time of this inspection, Rosemount was not planning any experiments or other research into the mechanisms of these transmitter failures with the exception of a technical evaluation of potential stress cracking at the Monel disc-to-stainless steel weld ring weld due to corrosion and/or differential expansion. At the time of preparation of this inspection report, Rosemount had not been able to obtain solubility data for hydrogen in the silicone oil used in Rosemount nuclear transmitters, nor has it obtained any new information on sources of hydrogen most likely to cause the type of failures experienced at St. Lucie.

Rosemount maintained that its root cause analysis, as it had been developed at the time of the inspection, was consistent with that provided in the Rosemount notification to its affected customers pursuant to 10 CFR 21.21(b), i.e. that monatomic hydrogen (whether from the coolant or generated by corrosion on the isolator surface) diffused through the Monel isolators, recombined to form molecular hydrogen, became trapped in the fill oil, and following a depressurization and repressurization, resulted in failed-high transmitter output signals. Rosemount believed this to be sufficient basis to recommend that its customers replace all potentially affected transmitters (i.e., those with known or suspected Monel isolator diaphragms), but it could not yet provide any concrete information that would enable licensees to justify leaving certain transmitters in service longer than others in order to prioritize the replacements. Rosemount further stated that its current strategy was (through increased production and some stock diversion) to focus on supporting replacement of sensor modules with Monel diaphragms thus far identified as needed by customers.

3.3.3 Rosemount Methodology for Scoping and Bounding the Problem

To aid in understanding the search methodology employed by Rosemount to bound the problem (i.e., identify all the potentially affected sensor modules and transmitters), the basic construction of a Rosemount pressure or differential pressure transmitter and the manufacturing process (and its documentation) for a sensor cell is described below.

3.3.3.1 Basic Transmitter Construction

A complete transmitter consists of a sensor module, an electronics module, and two process flanges. A sensor module consists of a cylindrical stainless steel housing containing a sensor cell and printed circuit boards with discrete electronic components. The sensor cell is welded into the module housing such that the housing totally encloses the cell and the electronics, except that the isolator diaphragms (which for the units in question were made of Monel) and the adjacent inner surfaces of the weld rings that surround the isolator diaphragms remain exposed at either end face of the housing cylinder. The sensor module is sandwiched between the two transmitter process flanges that are bolted together using metal O-rings to seal the process chambers in flanges to the weld rings around the isolator diaphragms in the faces of the sensor module. The electrical leads from the sensor cell are connected to the circuit boards, and wire leads from the module originate at the circuit boards and extend from the threaded neck of the module housing. The leads are then connected to a terminal block in one chamber of the transmitter electronics module housing after the housing is screwed onto the threaded neck of the sensor module.

3.3.3.2 Search Methodology and Manufacturing Process Documentation

The inspectors walked through the search methodology Rosemount employed to bound the problem thus far. The incoming information was the two transmitter serial numbers from St Lucie, 408929A and 411711A. The A suffix is used for transmitters with a serial number under 500000 that have been repaired (sensor module replaced) for the oil loss problem. To begin the search, this number (without the suffix) was entered into the repair records database which yielded among other information, repair house order (HO) number 767765.

Microfilm records were searched for this repair HO number and the associated documents filed under the HO number. These documents include all the production or manufacturing travellers and traceability information.

A manufacturing traveller is a document that accompanies the parts being built and lists the various attached drawings (DWGs), bills of material (BOMs), and manufacturing instructions (MIs) required to produce the part number identified as the finished assembly level designated on the traveller. There are signoffs for step completion and blocks to record traceability data. Traceability data consists of the part numbers and lot numbers or, if applicable, serial numbers or heat numbers, of the raw stock, purchased parts, or Rosemount-built parts used in building the assembly designated on the traveller.

The production of a sensor cell follows two principal paths from raw materials to finished cell. In the primary path, bar stock of a special alloy used by Rosemount is fabricated through several process steps into the so-called cell cups which then go through the glassing process and other steps. Finished cell halves are welded together with a center (sensing) diaphragm and are then ready to have the two isolator assemblies welded onto each side of the cell.

The secondary path, in which the isolator assemblies are fabricated, was of particular relevance to this inspection. Accordingly, the inspectors walked through this process in the factory at Chanhassen, Minnesota, and examined it in detail. In this path, rolls or coils of metal foil strip stock, of various materials and thicknesses (each thickness of each material with a unique part number) are received by the receiving clerks, who inspect them for damage and packaging compliance with the invoice or packing slip, and assign each box (containing a single roll) a sequential lot number (sequenced for that part number), recorded in a computer database. The foil strip stock then undergoes receiving inspection, which, for nuclear part numbers, is done in accordance with Nuclear Engineering Department drawings and procedures. After the foil strip stock successfully completes the various examinations and tests during receiving inspection, the receiving inspectors must affix a stock tag or label to each box. It is also usually the practice, although not specifically required by procedures, to mark the reel flange inside the box as well. The boxes of inspected strip stock are then stored in a segregated area and loaded periodically onto the designated ready service slide-down racks (one for each part number) next to the first production station area. Individual reels of strip stock are taken from these racks (as called for by the bill of materials for the part number of the disc assembly being produced) and placed on the disc punching or "blanking" machine. The bill of materials is attached to the traveller along with the manufacturing instruction and the drawing for blanking or punching out the foil discs (called "disc assemblies") from the strip stock.

For this path in the production of a nuclear (1152, 1153 or 1154) sensor cell, the first traveller is for the disc assembly. The traveller for a Part No. 01153-0252-0042 disc assembly, used in various nuclear transmitter models, including those affected by the problem in question, governs fabricating (blanking) disc assemblies from 2-inch-wide foil strip stock, made of type 316L stainless steel; in this case, for disc assemblies using 0.004-inch thick diaphragms among Range Codes 6 through 10. The next assembly level is the isolator assembly for which the principal production steps consist of cleaning and polishing the foil disc, laser welding it to a weld ring (a purchased part), which is supposed to be of the same material, an inspection, and a helium leak check of the assembly. Upon satisfactory completion of these operations, the part is now an isolator assembly, which for the 1153 transmitter sensor modules in question is designated Part No. 01153-0262-1042. The final sensor cell assembly level is where an isolator assembly is welded to the outside of each of the two cell cup halves of the sensor cell.

3.3.3.3 Detailed Record Search Specifics

During the inspectors' walkthrough of the Rosemount record search, the inspectors noted that the serial numbers of the failed 1153 transmitters, 408929A and 411711A, led to records that indicated that these two transmitters had been refitted with Part No. 01153-0221-0192 sensor modules, Serial No. 2328086 and 2328094 respectively. The travellers for sensor modules 2328086 and 2328094 indicated that the modules had been assigned these serial numbers, as is standard Rosemount procedure, because they were built with sensor cells of the same serial numbers. During manufacture at Rosemount's Chanhassen, Minnesota, factory, sensor cells are identified by part number, by lot number, and by the heat number of the bar stock of the proprietary alloy from which the cell cup halves are made. Once a cell has been completed, it is assigned a complete cell part number (for each range code) and a unique serial number. When nuclear part number sensor cells are received at Rosemount's Eden Prairie, Minnesota, facility, each cell is filled with fill oil, sealed, and initially tested. It is then used to build a sensor module, which is assigned the same serial number as the cell it contains.

The travellers for sensor cells 2328086 and 2328094 were retrieved from microfilm, printed, and reviewed. However, to initially determine the lot numbers of isolator assemblies used on those cells, Rosemount (and the inspector) reviewed another quality record called the Weld Log which also listed other sensor cells using the same lots of isolator assemblies. The Weld Log showed that these sensor serial numbers were among those of sensor assembly Lot 75, which consisted of Serial No. 2328077 consecutively through 2328104. The Weld log showed that this entire lot (Range Code 9) had been built from Lot 55 of Part No. 01153-0262-1042 isolator assemblies (weld ring and foil disc or diaphragm). Next, the traveller for isolator assembly Lot 55 was retrieved, printed, and reviewed. It showed that the entire Lot 55 of these isolators assemblies had been made from Lot 20 of Part No. 01153-0252-0042 foil disc assemblies.

Finally, the traveller for Lot 20 of these foil disc assemblies, dated May 30, 1989, was reviewed. The traceability block data showed that strip stock, Part No. C10181-0014, Lot (coil) No. 016 had been recorded (and presumably used). This part number was for the correct size (Dash No. -0014 indicates 0.004-inch thickness by 2.00-inch width), but the incorrect material (C10181 indicates commercial Monel Alloy 400). However, this original part and lot number had been lined out and Part No. C09851-011 (indicating 316L, 0.004" X 2.00"), Lot 23 and 24, written in, initialed by "V.K." and dated October 30, 1989. The annotation in the margin cited DR 491585, also dated October 30, 1989, and was initialed by the Nuclear Document Coordinator, as authority for the part number and lot number correction. Rosemount stated that their initial investigation had led them to this point when questioned as to the manufacturing history of these sensors by FP&L.

3.3.3.4 Confirmation of Incorrect Material

Review of the DR, which indicated that the Material Review Board had apparently concluded that a paperwork error had been made and corrected, reportedly did not suggest to Rosemount at the time that there was a definite material problem. However, as the chronology of events (Appendix C) confirms, after the Rosemount technician found the distended isolators at St. Lucie and after SwRI identified the entrapped gas as pure hydrogen, Rosemount ordered the material analysis of the isolator diaphragms and weld rings because, as explained by Rosemount, these facts strongly suggested that the isolators were actually made of Monel, the part and lot number of which, C10181-0014 and Lot 16, the blanking machine operator had recorded. Rosemount further stated that the identification of the hydrogen also prompted it to continue its search of records to determine the scope of the problem and the use of Monel was shortly thereafter confirmed by the SwRI analyses.

3.3.3.5 Identification of Affected Disc Assembly Lots

Once the material of the Lot 20 disc assemblies was confirmed, Rosemount reviewed the travellers for five disc assembly lot numbers above and below Lot 20 to determine which, if any, also may have been made with Monel strip stock, particularly from Lot 16. The inspector repeated this review and found, as had Rosemount, no other disc assemblies for nuclear transmitters in this series that did not have the correct part number recorded. None of the disc assembly lots in this series (for the disc assembly part number in question) used 316L strip stock Lots 22, 23, 24, or 25. Therefore, it was not clear on what basis the person with the initials V.K. had selected Lots 23 and 24 of the C09851-0011 316L strip stock that had been annotated on the Lot 20 disc assembly traveler to correct the discrepancy noted in DR 491585. Rosemount's only plausible explanation was that the person who selected 316L strip stock Lots 23 and 24 most likely chose 316L lots in use at about the same time frame. The inspectors reviewed the receipt inspection records and found that Lots 23 and 24 of the 316L strip stock had been received on April 18, 1989, and June 20, 1989, respectively, and both coils had been purchased under Rosemount purchase order (PO) No. EF3082. The also inspectors reviewed the certified material test reports for these lots of 316L strip stock with no discrepancies noted.

However, because Lots 22 through 25 of the 316L (C09851-0011) strip stock were evidently not used for Lot 20 disc assemblies, the inspectors' asked Rosemount to find out where its records showed these lots were used. Subsequent to the inspection, Rosemount reported that it had found that Lots 22 through 25 had been used for other disc assembly part numbers in process at the time. For example, Lots 23 and 24 were used for disc assemblies intended for Model 1151, non-nuclear transmitters (See Appendix C to this report).

The receipt inspection record for Lot 16 of the Monel strip stock indicated it had been received on December 8, 1988, on PO ED1909 and had initially been rejected for slightly out of specification hardness on Rejection Document No. 46902. This discrepancy was dispositioned "use as is" because the hardness was only slightly out of specification, all other physical properties and chemical composition were within tolerances and hardness itself was not considered a critical attribute for the application, but was used as a consistency overcheck for chemical composition and other physical properties. The inspectors concluded, particularly on the basis of the timing involved, that it was not likely that this discrepancy contributed to the inadvertent use of this Monel strip stock instead of 316L.

3.3.3.6 Identification of Affected Isolator Assembly Lots and Sensor Modules

The remainder of the Rosemount scoping review (repeated by the inspectors) consisting primarily of searching through the Weld Log, identified all the lots (serial number ranges) of sensor cells that were made from Lot 55 of the isolator assemblies which had been made with Lot 20 disc assemblies. Then the microfilm records of the isolator assemblies were searched to identify all of the other lots, if any, of isolator assemblies that may have been made, wholly or in part, with Lot 20 disc assemblies. This search, also repeated by the inspectors, revealed that one other isolator assembly lot, Lot 54, was built using Lot 20 Monel disc assemblies. Another review of the Weld Log identified all the groups of sensors (serial number ranges) that were built with Lot 54 isolator assemblies. While determining which sensors had been made from Lot 54 isolator assemblies, Rosemount noted that two sensor groups within Lot 65, Range Code 7, had been made from both Lot 54 and Lot 55 isolator assemblies.

However, during this review, Rosemount discovered that some of the sensors of sensor Lot 76, Serial Nos. 2336503 through 2336530, were made from Lot 55 isolator assemblies with Monel discs (diaphragms) and some from Lot 56 isolator assemblies, which records (traveller traceability data) showed were made with 316L stainless steel discs as required. Traceability data for this so-called mixed lot of sensors indicated that Lot 56 isolators were made from Lot 21 disc assemblies, which were made from Lot 25 of 316L foil strip stock).

In addition, at the lower end (by serial number) of the range of affected sensors (those potentially made with Lot 54 and 55 isolators), the Weld Log showed that in Sensor Lot 73, Serial Nos. 2290445 through 2290472, some sensors were made from Lot 54 Monel isolator assemblies and some from Lot 53 isolator assemblies, which, like lot 56, records confirmed to be made of 316L as they were supposed to be. Therefore, among the approximately 450 sensors potentially affected, subsequent review of the individual travellers in the mixed lots indicated that about 50 of them actually had 316L isolators and not Monel. However, Rosemount reported all the 450 as suspect because its records did not indicate which serial numbers within these mixed groups used Lot 54 or 55 (Monel) and which used Lot 53 or 56 (316L).

Material analysis by X-ray fluorescence (XRF) was performed by Rosemount on three returned suspect (Part 21-listed) sensors from Northern States Power. The XRF identified two sensor modules as having stainless isolators and one as having Monel isolators. In fact, it was this analysis that led Rosemount to discover the second mixed lot earlier than their methodical search would have. Finally, Rosemount indicated in its compiled list of affected serial numbers of sensor modules those that were scrapped before shipment and others that were not logged as having completed pressurization aging and so also would not have been shipped. Appendix A to this report was taken from Rosemount's list showing the names of potentially affected licensees and other purchasers and the quantities supplied.

Subsequent to the inspection, in May 1995, the Rosemount QA Manager informed the inspector that Rosemount had completed a review of all travellers for Part No. 01153-0252-0042 disc assemblies from 1981 to December 1994 (which represents 49 documents), and in a different database, from January 1995 to the present, during which time 10 travellers were on file. The Rosemount QA Manager explained that the apparent difference in the rate of traveller use was because during the previous period, lot sizes were generally more than 1000; whereas, recently, lot sizes have been typically fewer than 100. The corresponding travellers of the other nuclear transmitter foil disc assembly part numbers were still being reviewed. Rosemount reported that no discrepancies were identified during this review and that no other DRs had been identified for 01153-0252-0042 disc assembly travellers.

There are five other configurations with unique part numbers used in nuclear transmitters of other range codes or for special applications. The Rosemount QA Manager later reported a review of travellers of the other disc assembly part numbers used in nuclear transmitters from 1981 to the present (a total of 123 documents). Two DRs had been written. One in 1987 identified the wrong dash number recorded in the traceability block on a traveller, a part number suffix that indicated that the wrong thickness of strip stock (although correct material) may have been used. The disposition of this DR was, appropriately, to check the thickness of completed diaphragms from affected lots, which were found to be correct, indicating that the error had been in recording the wrong dash number used. The other DR, written in 1988, indicated that the wrong strip stock material had been used for a lot of disc assemblies, and all affected parts were scrapped. No other discrepancies were reported by Rosemount.

3.4 Investigation of Original Error

To learn more about the original error, the inspectors examined in detail the processes of receiving, inspecting, and issuing strip stock, and the manufacturing processes for isolator assemblies at the factory in Chanhassen, Minnesota. The NRC had initially believed that the original error was welder selection of the incorrect disc assemblies to make isolator assemblies. However, the inspectors determined that the original detectable error actually occurred when a punch press or blanking machine operator used the incorrect material spool (Monel instead of 316L stainless steel) with which to make Lot 20 of Part No. 01153-0252-0042 disc assemblies. According to Rosemount procedures, machine operators are supposed to verify that the material called out on bills of materials attached to travellers is used, and the traceability data from the Rosemount stock tag or stock label (part number and lot number), and only from the stock tag or label, is to be recorded on the traveller in the block provided.

In this case, the blanking machine operator correctly recorded the information on the material used, but failed to verify that the material used was correct according to the bill of materials. In doing so, the non-nuclear part number, C10181-0014, for strip stock of Monel Alloy 400, lost its material identity and became, for all anyone further down the manufacturing line would know, part of a nuclear transmitter sensor cell subassembly. The welder later selected Lot 20 disc assemblies identified with the correct part number for the Lot 54 and 55 isolator assemblies being made, but did not know that the Lot 20 disc assemblies were made of the wrong material. The inspectors also learned however, that the welder may have compounded the error by failing to recognize certain anomalies in the welding process (discussed later).

It could not be determined why or how the wrong (Monel) strip stock had been used initially to make the Lot 20 disc assemblies. However, the inspectors determined that due to the part numbering system employed since 1990, using "N" numbers for material to be used in nuclear transmitter parts (there are no Monel "N" numbers), the error would be much less likely to occur after that time. Also, according to Rosemount records, there were no returns or complaints other than from St. Lucie attributable to this problem. These facts support the conclusion that Rosemount's search methodology was adequate and that the probability of Monel isolators being used inadvertently in nuclear transmitters since the time of this incident is very low.

In examining boxes of material on the rack next to the punch press, the inspector noted that the Rosemount computer-printed stock label was placed on the outside of the box, but not on the spool flange holding the coiled foil strip stock. As discussed above, it was the practice sometimes, although not required by procedure, to mark the spools as well with part number, lot number and sometimes heat number. The inspector asked for copies of the purchase orders for the material in question to determine what marking requirements were imposed on the vendor. The inspector reviewed Rosemount Purchase Specification PS-25, called out on the drawings for the strip stock (C101, 81 for Monel 400 and C09851 for 316L). The inspectors found that PS-25 required that all packaging (taken by Rosemount to mean exterior) be marked with the Rosemount part number.

After reviewing PS-25, the inspector noted that some packaging previously observed by the inspector may not have been marked in accordance with PS-25. One box of N09851-0011 strip stock had a printed white label taped onto the box with only the PO number hand written in. The part number was on the Rosemount stock tag (label) put on by the receipt inspectors, but the inspector did not recall noticing the part number elsewhere on the box. Other boxes examined (e.g., for Monel and commercial 316L (C09851)) had the Rosemount part number stencilled on the box by the vendor in addition to heat number, and purchase order number. The possibility that some markings on boxes of strip stock may not conform to PS-25 was pointed out to the accompanying cognizant Rosemount staff for investigation and appropriate disposition.

At the time of the inspection, Rosemount was interviewing supervisors and reviewing records. Subsequent to the inspection, Rosemount reported that it had identified the blanking machine operator from the initials on the traveller in question. This operator was an employee who had about 11 years experience at the time, but is no longer with the company. The inspectors did not attempt to locate and interview this person during this inspection. Rosemount has not yet been able to determine the root cause of the mistake. The inspectors were told that the ready service supply racks next to the blanking machine were the same as they were in 1989. The inspectors noted that they were marked with standard locations for each part number, and the Monel and stainless part sections are not adjacent. The inspectors did not identify any apparent working condition or situation that would have been (or would now be) conducive to selecting incorrect material. The inspectors concluded however, that the use, since just after this incident, of N part numbers for "raw" materials to be used in assemblies that have nuclear (i.e., 1152, 1153 or 1154 prefix) part numbers would be the most effective means of preventing use of non-nuclear parts (such as Monel) in nuclear transmitters.

The inspectors concluded that Rosemount's failure on May 30, 1989, to follow Blanking Procedure 01153-3036 constituted a nonconformance with respect to the requirements of Criterion V, "Instructions, Procedure, and Drawings," of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). Criterion V requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and that they be accomplished in accordance with these instructions, procedures, and or drawings. Instructions, procedures, or drawings are required to include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Rosemount Manufacturing Instruction 01153-3036, "Blanking Procedure," instructed the operator: "Obtain the proper stock coil material for the part number to be blanked." The bill of material (BOM) attached to the traveller for Part Number 01153-0252-0042 foil disc assemblies, Lot 20, specified using Part Number C09851-0011, 316L stainless steel foil strip stock.

However, while blanking Lot 20 of Part Number 01153-0252-0042 foil disc assemblies, the operator apparently used Part Number C10181-0014 Monel Alloy 400, Lot 16, strip stock for this process instead of the required Part Number C09851-0011 316L; although the operator did document use of Lot 16 of the Monel in the traceability data block on the traveller. When the parts were then redesignated as 01153-0252-0042 disc assemblies, the identity of the material was lost and was henceforth, as would be expected, presumed to be the correct material. This discrepancy was not detected during subsequent operations, despite the apparent considerable difficulty in successfully welding the Monel discs to 316L weld rings. The difficulty was indicated in part on the basis of discussions with factory personnel (instrument builders) who explained that (1) the laser welding machine would have had to be significantly adjusted from its nominal settings for stainless steel in order to get welds of the 316L weld rings to Monel disc assemblies to pass the leak test, (2) laser welding Monel produces a characteristic green glow, and (3) the actual recorded yield from Lots 54 and 55 totaled only 800 isolator assemblies; whereas, the potential yield available was 2500 as derived from the number of disc assemblies (2500) in Lot 20. The discrepancy was caught by the Nuclear Department Document Coordinator review of the document packages containing the travellers. Because this error was caught and properly documented by the Rosemount QA system's routine review and screening process, this error is not being cited as a nonconformance.

3.5 Review of Discrepancy Report Disposition

The inspectors reviewed Discrepancy Report (DR) No. 491585 written by the Nuclear Document Coordinator on October 30, 1989, when she discovered the error during final record review. The coordinator, who still held this position at the time of the inspection, stated that she turned the report over to the Material Review Board (consisting of the production supervisor, the cognizant design engineer, and the cognizant quality engineer) for disposition. Subsequent to the inspection, the Rosemount QA manager informed the inspector that two of the three members of the Material Review Board that dispositioned DR 491585 (the quality engineer and the design engineer) had been interviewed and had submitted written statements regarding their recollections of the rationale for their disposition of DR 491585. The inspectors noted that current Rosemount procedures required written justification for DR dispositions, but this had not been required at the time the DR in question had been inappropriately dispositioned.

Rosemount reported that the quality engineer first explained that he had believed that use of Monel was highly unlikely because to the best of his recollection, all the correct piece parts were (and are) kept in a locked cabinet near the isolator assembly welding station. The inspector and Rosemount QA Manager noted that this indicated a lack of understanding on his part of how the mistake occurred. The quality engineer further stated his belief at the time that Monel discs could not have been successfully welded to stainless steel weld rings. He also stated that the welds are examined and leak checked with helium and it was believed that isolator assemblies with stainless steel weld rings and Monel discs would not have passed the test. The quality engineer finally added the rationale that no punch marks were visible on the weld rings, indicating that they were 316L. According to this logic, if Monel had been used for the discs, the weld rings would have also been Monel and would show the characteristic identifying punch marks. The inspector and the Rosemount QA Manager both noted that this was circular reasoning because the weld rings would only be expected to be Monel and show Monel punch marks if Monel was being used intentionally for the discs. This reasoning, again, showed lack of full understanding of the process and the problem by the quality engineer. The QA Manager stated that he would examine this issue further, including talking with the quality engineer about the discrepancies in his reasoning. In addition, the QA Manager would attempt to find out if the instrument builders involved had been interviewed to determine if they had noted unusual welding settings or adjustments being required to successfully weld the Monel discs to the stainless weld rings, if anyone had noticed the characteristic green glow of Monel being laser-welded, or if anyone had questioned the apparently unusually high difference between the number of disc assemblies and the yield of isolator assemblies indicating a high scrap rate from inspection and tests of finished welds.

Rosemount reported that the statement of the design engineer Material Review Board member indicated that she had concurred in the disposition of simply "correcting the part numbers on the traveller" because she stated that the discrepancy was characterized on the DR as "incorrect part number recorded" and the QA Manager added that he had been told that there were "a large number of paperwork errors at the time." The inspector noted that the design engineer's recollection of the language of the DR was inaccurate, suggesting a misinterpretation. In fact, the DR stated "wrong part number...used to build product." The QA Manager agreed to discuss with the design engineer the basing of her concurrence on (1) simply noting that the production supervisor and quality engineer had concurred and (2) an inaccurate understanding of the implication of the DR and nature of the mistake.

Rosemount later reported (June 1995) that the third member of the Material Review Board, the production supervisor, had been interviewed, but could not recall his rationale for the disposition of the DR; although he did not disagree with the statements of the other two board members. On the basis of the reported statements by the Material Review Board, the inspectors concluded that the inappropriate disposition of the DR resulted from incomplete understanding of the problem and a perfunctory review. Discussions with factory personnel gave the inspectors the sense that a climate of low tolerance for identifying problems that may have existed at the time of this incident (which it was emphasized no longer existed) and perhaps some production pressure, may have contributed to the lack of recognition of unusually high scrap rates during welding as well as quick acceptance by the Material Review Board of an easy explanation for the wrong part number issue identified in the DR.

Criterion XV, "Nonconforming Material, Parts or Components," of 10 CFR Part 50, Appendix B, states: "Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, disposition, and notification to affected organizations."

Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, states, in part: "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. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition...."

Contrary to the above, measures established by Rosemount to control certain nonconforming material did not prevent its inadvertent use. In addition, action taken to correct the related documented condition adverse to quality was inadequate in that Discrepancy Report 491585, written October 30, 1989, was dispositioned inappropriately by the Material Review Board. The discrepancy report documented the use of strip stock material that was not in accordance with the bill of materials to make Lot 20 of Part No. 01153-0252-0042 disc assemblies. Instead of investigating the condition and verifying what material was actually used in making Lot 20 of these disc assemblies, then taking the steps necessary to capture any incorrect material and prevent its use in transmitters designed for nuclear safety-related service, the Material Review Board dispositioned the discrepancy by directing that the strip stock part number on the traveller be "corrected." As a result, Lot 16 of C10181-0014 Monel, documented as having been used in the data block on the traveller for Lot 20 of the disc assemblies, was actually so used, yet the traceability data on the traveller was erroneously changed to read Lots 23 and 24 of the part number called out on the bill of materials, C09851-0011 (316L stainless steel) (95-02-01).

3.6 Review of Rosemount Material Identification Tests

The inspectors reviewed traces of X-ray fluorescence (XRF) nondestructive examination (NDE) of isolators returned from Northern States Power (NSP). In the printout from XRF equipment, XRF traces of material known to be Monel or 316L stainless steel are very distinctive, both qualitatively (pattern) and quantitatively (amplitudes, distribution) Traces for XRF tests on isolators of returned modules were consistent with those known to be either Monel or stainless respectively. However, instead of two of the three returned NSP modules having Monel isolators and one possibly having stainless isolators (due to its being from one of the mixed lots discussed above) as expected, the XRF analysis indicated that two had stainless steel and one had Monel isolators. This led Rosemount to re-review the Weld Log and thus led to the identification of the second mixed lot described in Paragraph 3.3.3.6 above, this one made from both Lot 54 (Monel) and Lot 53 (316L). Further review of the travellers of these modules confirmed the information in the Weld Log. The inspectors concluded that material analysis performed thus far has confirmed the reliability of Rosemount records.

3.7 Discrepancies Between Rosemount and Customer Records

At the time of this inspection, one customer had reported a discrepancy between the data in the Part 21 report and its own records. Commonwealth Edison Company (CECo) reported that the storeroom records at its Dresden Nuclear Station (Dresden), which were being reviewed in response to the Rosemount Part 21 report, indicated that the serial number of an affected replacement transmitter provided by Rosemount had serial number 413060A; whereas, the Part 21 report listed this transmitter as having serial number 415060A. Rosemount had determined, and the inspectors confirmed, that this was an error in transferring the serial number from Rosemount production records, which agree with Dresden procurement/installation records, to the list published with the Part 21 customer notification. At the time of preparing this report, there have been no other similar occurrences reported.

3.8 AMS Signal Analysis Results From St. Lucie

Just before this inspection the NRC learned that special signal analysis equipment (from a company called "AMS") at the St. Lucie plant that is normally used for trending the performance of transmitters, specifically their ability to track with minute variations in reactor system pressure, appeared to have the capability of identifying transmitters with Monel isolators. Further, St. Lucie believed, on the basis of data from this AMS equipment, that it may have Monel isolators in transmitters that were not on Rosemount's Part 21 list. This information had the potential for expanding the scope of the problem beyond the bounds initially established by Rosemount. During this inspection, the inspectors asked for the results of material analyses that Florida Power & Light was performing on certain transmitter isolators to confirm their theory. However, based on the latest report from St. Lucie, the material of the isolators in one transmitter not listed by Rosemount that had exhibited a so-called "Monel-like" AMS signature was evaluated as stainless steel using what the licensee described as a gamma-backscatter test. This result renders the AMS method for detecting Monel inconclusive at present.

3.9 Customer Complaints/Returns Involving Other Than Oil-Loss Symptoms

Rosemount reported reviewing customer complaints/returns documentation and stated that prior to the St. Lucie event, it has had no returns or complaints regarding nuclear (1152,1153, or 1154) transmitters attributable to the Monel/hydrogen intrusion problem. The review was described as a computer-aided search of the listed final disposition failure modes field of the customer return/complaint database, using pertinent keywords such as gas, intrusion, isolator, Monel, and hydrogen. The inspectors did not make an independent check of this process during this inspection.

4.0 PERSONS CONTACTED

Rosemount Nuclear Instruments, Incorporated

Mark Van Sloun, VP & General Manager
Ken Ewald, Business Unit Manager
Jerry Valley, Quality Assurance Manager
Stuart C. Brown, Engineering Supervisor
Timothy J. Layer, Product Marketing Manager
Paul Roepke, Receiving Department Technician
Lori Majerus, Receiving Department Inspector
Esther Pollard, Receiving Inspection Supervisor
Jeff Bracken, Nuclear Inspector
Bonnie Strawberry, Production Line Supervisor
Jan Bockman, Instrument Builder

APPENDIX A

ORGANIZATIONS IN THE U.S. TO WHOM ROSEMOUNT REPORTED SENDING TRANSMITTERS OR SENSOR MODULES WITH MONEL ISOLATORS

Organization	Facility	Quantity
Arizona Public Service	Palo Verde	9
Baltimore Gas & Electric	Calvert Cliffs	2
Bechtel		1
Boston Edison	Pilgrim	13
Carolina Power & Light	Brunswick, Harris, Robinson	4
Commonwealth Edison	Byron, Braidwood, Quad Cities LaSalle, Dresden, Zion	2
Consumers Power	Palisades, Big Rock Point	16
Duke Power	Oconee, Catawba, McGuire	4
Duquesne Light Company	Beaver Valley	1
Ellis & Watts		1
Florida Power Corp.	Crystal River	16
Florida Power & Light	St. Lucie, Turkey Point	33
Georgia Power	Hatch, Vogtle	2
GPU	Oyster Creek, Three Mile Island	2
Gulf States Utilities	River Bend	5
Houston Lighting & Power	South Texas Project	3
Illinois Power	Clinton	1
Maine Yankee Atomic Power Company	Maine Yankee	2
New Hampshire Yankee, Inc.	Seabrook	8
New York Power Authority	Fitzpatrick, Indian Point 3	5
Niagara Mohawk Power Corp.	Nine Mile Point	6
Northern States Power	Monticello, Prairie Island	12
Omaha Public Power District	Ft. Calhoun	4
Pacific Gas & Electric	Diablo Canyon	2
Pennsylvania Power & Light	Susquehanna	1
Philadelphia Electric Company	Limerick, Peach Bottom	8
Portland GE	Trojan	5
Public Service Electric & Gas	Salem, Hope Creek	7
South Carolina Electric & Gas	Summer	4
Southern Cal. Edison	San Onofre	2
Systems Energy	Grand Gulf, Waterford	7
Toledo Edison	Davis-Besse	2
TU Electric	Comanche Peak	2
TVA	Watts Bar, Sequoyah, Browns Ferry	5
Vermont Yankee	Vermont Yankee	1
Virginia Power	Surry, North Anna	1
Washington Public Power Supply System	WNP-2	3
Westinghouse		5
Wolf Creek NOC	Wolf Creek	3
Yankee Atomic		2

APPENDIX B

CHRONOLOGY
ROSEMOUNT DIAPHRAGM ISSUE

11/22/94 2137	Failure of two Rosemount pressurizer pressure transmitters generate an SI signal during repressurization from cold shutdown at St. Lucie 1
11/23/94 0125	ESF actuation reported by FP&L under 10 CFR 50.72; no mention of root cause
12/15/94	Rosemount tech at St. Lucie finds no oil loss, but identifies gas in fill oil as probable failure mode due to distention/depressibility of diaphragms.
12/94-02/95	Rosemount continues internal investigation and searches for laboratory capable and willing to handle tests on radioactively contaminated sensor modules.
02/03/95	Failed sensor modules sent from St. Lucie to SwRI for examination and testing.
02/22/95	Hydrogen identified as only entrapped gas by SwRI. SwRI finds no evidence of corrosion, water leakage, or fill oil breakdown.
03/14/95	SwRI identifies diaphragm material as Monel.
03/17/95	Number of transmitters potentially involved identified by Rosemount.
03/20/95	Rosemount transmits to NRC preliminary list of affected organizations and the serial numbers of potentially affected modules provided to them along with preliminary technical evaluation and root cause analysis.
03/21/95	Rosemount issues 10 CFR 21.21(b) notification to affected customers.
03/22/95	NRC issues Information Notice 95-20
03/23/95	NRC activates Regulatory Response Groups of owners groups (Westinghouse, BWR, B&W, CE) to obtain information on transmitter location, operability, safety assessment, corrective action, etc.
03/31/95-04/03/95	RRGs submit written responses to NRC
04/05-07/95	NRC conducts inspection at Rosemount.

APPENDIX C
TRACEABILITY DATA SUMMARY

Foil Strip Stock or Strip Stock (FSS)

Part Number C10181-0014: C=commercial, 10181=Monel alloy 400, -0014=coiled, spooled metal foil strip stock, 0.004" thick x 2.00" wide
Lots of Interest: 16 (assigned by computer during receiving)

Part Number C09851-0011: commercial, 09851=316L, 0011=0.004" X 2.00" Lots 22 through 25 not used for D/As of interest, D/A Lots 15-19, used 316L FSS Lots 11, 12, 12, 13, and 21 respectively, D/A Lots 21 and 22: 316L FSS Lots 26 & 27, D/A Lots 23,24 1st to use nuclear N09851-0011, Lots 001 and 003, D/A Lot 25: NFSS Lots 003 & 005

Part Number N09851-0011: N=nuclear grade, 09851=316L, 0011=0.004" X 2.00" (N-numbers used for part numbers involving 01152,3,4 since 1989

Disc Assembly (D/A)

Foil Disc "Assembly" (punched out disc) Part Number: 01153-0252-0042
Lot 19 traceable to 316L (C09851-0011) strip stock Lot 21
Lot 20 traceable to Monel (C10181-0014) strip stock Lot 16 (erroneously determined to be traceable to 316L Lots 23 and 24)
Lot 21 traceable to 316L (C09851-0011) strip stock Lot 26

Isolator Assembly (I/A)

(consisting of a cleaned, polished disc "assembly" welded to a weld ring I/A Part Number 01153-0262-1042, Lots of interest: I/A Lot 53 traceable to D/A Lot 19 -Identified as last I/A Lot to use Lot 19 D/As. (Checked back to Lot 49 to confirm Lot 19 or below and not Lot 20), I/A Lot 54 traceable to D/A Lot 20 traceable to FSS Lot 16-identified by search of records of I/A travellers to see which I/A Lots used Lot 20 D/As, I/A Lot 55 traceable to D/A Lot 20 traceable to FSS Lot 16-identified through Weld Log by module serial numbers 2328086 and 2328094 (same as sensor serial numbers) installed in FP&L (St Lucie) transmitters 408929A and 411711A respectively according to Rosemount records filed under repair HO # 767765. I/A Lot 56 traceable to D/A Lot 21 - first I/A Lot to use Lot 21 D/As. (Checked forward to Lot 62 to confirm use of D/A Lots 21 and above and no more 20s)

Sensor Cells

Sensor cells and modules have same serial numbers. Sensor cells for range codes 6-10 of interest IAW Weld Log only these RCs used I/A Lots 54 or 55 Part Numbers, e.g., 01153-0264-0092 for range code (RC)-9. Lots of Interest: Lots 73, 63, 66, 74, 14, 42, 43, 65, 75, 68, 76 made entirely or in part with Lot 54 or 55 I/As. Lot nos. contain serial number ranges in Part 21 report.

Transmitters

Model number, e.g., 1153GD9PB for RC-9, guage pressure, records trace cells/modules to XMTRs in which used (some supplied as modules only).

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

August 17, 1995

Mr. Martin R. Benante
President and General Manager
Target-Rock Corporation
1966E Broadhollow Road
Farmingdale, NY 11735-0917

SUBJECT: NRC INSPECTION NO. 99900060/95-01 AND NOTICE OF NONCONFORMANCE

Dear Mr. Benante:

This letter addresses the inspection of your facility at Farmingdale, New York, conducted by Mr. Stephen Alexander and Mr. Paul Narbut of this office on July 11-12, 1995, and the discussion of their findings with the members of your staff identified in the enclosed report at the conclusion of the inspection and in subsequent telephone conversations. The inspection was conducted to examine Target Rock's actions and conclusions regarding three of its pilot-operated main steam safety relief valves (SRVs) which failed to operate when the valves were tested at Nebraska Public Power District's Cooper Nuclear Station.

Areas examined during the inspection and our findings are discussed in the enclosed report. The inspection included a review of the circumstances surrounding the SRV failure to provide a basis for assessing the validity and completeness of your root cause analysis and your investigation to determine the extent of the problem with corroded SRV pilot valve solenoids in the industry. The inspectors also reviewed your procedures adopted pursuant to Section 21.21 of Part 21 of Title 10 of the Code of Federal Regulations and your actions regarding the solenoid valve failures in question prescribed by those procedures. Within these areas, the inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

During this inspection, we determined that the implementation of your quality assurance (QA) program failed to meet certain NRC requirements. The failure to prevent inadvertent supply of nonconforming components (i.e., solenoid valves apparently containing residual hydrostatic test water) to an NRC-licensed facility constituted a nonconformance with respect to the requirements of 10 CFR Part 50, Appendix B. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter. You are requested to provide us within 30 days from the date of this letter a written statement in accordance with the instructions specified in the enclosed Notice of Nonconformance.

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

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

The cooperation of your staff in this matter was greatly appreciated. Should you have any questions about the enclosed report, we would be glad to discuss them with you.

Sincerely,

Robert M. Gallo

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

Docket No.: 99900060

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99900060/95-01

NOTICE OF NONCONFORMANCE

Target Rock Corporation
Farmingdale, New York

Docket No. 99900060
Report No. 95-01

Based on the results of an NRC inspection conducted on July 11-12, 1995, it appears that certain of your activities were not conducted in accordance with NRC requirements.

A. Criterion XV, "Nonconforming Material, Parts or Components," of 10 CFR Part 50, Appendix B, states: "Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures."

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

Contrary to the above, Target Rock failed to properly control certain activities to prevent the inadvertent supply of nonconforming material to an NRC-licensed facility in that procedures or procedural compliance were inadequate to ensure that three Model 1/2-SMS-S-02 solenoid pilot valves for safety relief valves supplied to the Cooper Nuclear Station were properly and completely dried following a hydrostatic test. As a result, the solenoid valves failed to operate due to corrosion in the core tube caused by storage for about 1-1/2 years prior to installation with residual hydrostatic test water trapped in the core tubes. (95-01-01)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance

Enclosure 1

(1) the reason for the nonconformance, or if contested, the basis for disputing the nonconformance, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further noncompliances, and (4) the date when your corrective action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 17th day of August, 1995

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

REPORT NO.:	99900060/95-01
ORGANIZATION:	Target Rock Corporation 1966E Broadhollow Road Farmingdale, NY 11735-0917
ORGANIZATIONAL CONTACT:	James D. White Sales and Service Manager
TELEPHONE:	(516) 293-3800, EXT 647
NUCLEAR INDUSTRY ACTIVITY:	Target Rock's activities within the scope of this inspection include manufacturing and supplying valves, primarily relief valves, and replacement parts and service to the nuclear industry.
INSPECTION DATES:	July 11-12, 1995

LEAD INSPECTOR:	G C Cwalina for S Alexander Stephen D. Alexander Vendor Inspection Section (VIS) Special Inspection Branch (PSIB)	8/16/95 Date
OTHER INSPECTORS:	Paul P. Narbut Special Inspection Section, PSIB	
REVIEWED BY:	Gregory C Cwalina Gregory C. Cwalina, Chief, VIS/PSIB	8/16/95 Date
APPROVED BY:	Robert M Gallo Robert M. Gallo, Chief, PSIB/DISP	8/17/95 Date

Enclosure 2

1.0 SUMMARY OF INSPECTION FINDINGS

The inspection was conducted to examine Target Rock Corporation's (Target Rock's) actions and conclusions regarding three of its pilot-operated safety relief valves (SRVs) which failed to operate when the valves were tested at Nebraska Public Power District's (NPPD's) Cooper Nuclear Station (Cooper).

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, "Reporting of Defects and Noncompliance," of 10 CFR.

1.1 Violation-

None

1.2 Nonconformance (99900060/95-01-01)

Contrary to the requirements of Criteria V and XV of 10 CFR Part 50, Appendix B, Target Rock failed to prevent the supply of nonconforming material (i.e., SOVs containing residual water) to an NRC-licensed facility (Cooper) due to inadequacy of and/or noncompliance with procedures governing manufacture and testing of the material (See Section 3.1.3 of this report).

1.3 Open Item (99900060/95-01-02)

Review of Target Rock's actions regarding a problem it identified involving damage to SRV main disc return springs cause by repeated SRV functional testing using the standard reduced flow test setup. The cycle-dependent damage includes apparent spring relaxation spring end chipping. (See Section 3.2 of this report).

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous findings were reviewed during this inspection.

3.0 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Failed Target Rock SOVs at Cooper

3.1.1 Background

On February 10, 1995, NPPD declared a Notice of Unusual Event at Cooper when three Target Rock Model 7457F-600 main steam SRVs failed to open upon receipt of a manually initiated electric signal during startup testing. Cooper has eight SRVs, six associated with the automatic depressurization system (ADS) and two with the Low-Low setpoint safety relief logic. NPPD and Target Rock attributed the failure of the two non-ADS SRVs to open upon command to a failure of the SRVs' solenoid-operated pilot air valves (SOVs) to port air or nitrogen to the SRVs' pneumatic operating cylinders. NPPD disassembled and examined one of the failed SOVs and found internal corrosion had caused binding of the solenoid actuating mechanism. (REF: PNO-IV-95-005)

3.1.2 NRC Preliminary Inquiries

During separate NRC telephone conversations with Cooper and Target Rock on February 15, 1995, the NRC was told that the three SOVs, two of which later failed (Model 1/2-SMS-S-02, Serial Numbers 376, 377, and 378), were delivered to Cooper with three replacement SRV's, Model 7567F-600, purchased from GE Nuclear Energy (GE NE) by NPPD for Cooper. The SRVs were obtained by GE NE from uninstalled spares at the cancelled Shoreham Power Plant (Shoreham) and were refurbished by Target Rock at its facility in Farmingdale, New York, under contract with GE NE. The SRVs had not been in service at Shoreham. Target Rock installed new, improved-design, Model 1/2-SMS-S-02, SOVs on the three SRVs at its facility, then shipped the completed SRVs to Cooper.

Target Rock concluded that only the three SOVs supplied for Cooper were affected, because they were the only ones ever built entirely with new parts by field service personnel, based on personnel recollections of the field service manager. Additional information in support of this conclusion was reviewed by the inspectors during this inspection at Target Rock.

Initially, Target Rock stated that it did not intend to issue a report to the NRC pursuant to 10 CFR 21.21(a) because it had determined that it did not have the capability to evaluate this deviation for existence of a defect. Target Rock stated its intention to identify potentially affected licensees or purchasers (although Target Rock believed the problem to be isolated to Cooper) and to inform them of the deviation pursuant to 10 CFR 21.21(b).

The Target Rock Sales and Service Manager stated during another telephone conversation with the NRC, on March 30, 1995, that he had traveled to Cooper and disassembled and inspected the second failed solenoid. He found rust in the bonnet tube which would inhibit the plunger movement and concluded that the assembly had been improperly or ineffectively dried (due to installed internals) at the time of manufacture. Since only three such assemblies were made in the manner described above, and only those three required the second hydrostatic test, Mr. White concluded that the problem was limited to the three assemblies provided to Cooper. Target Rock has sent a letter to Cooper to reporting these conclusions.

The initial NRC assessment, based on the above information, was that a generic concern did not exist and immediate notification of industry was not required.

3.1.3 Inspection at Target Rock

The inspectors examined the actions that Target Rock had taken since the February 15 and March 30, 1995 telephone discussions. Target Rock had not changed its conclusions regarding that the problem was limited to the three valves at Cooper. Target Rock had been to the Cooper site and examined a second of the failed valves. The inspectors examined photographs of the disassembled valve internals. The rust was evident and the Target Rock representative stated that the movable core was restricted from movement by rust and that the rust was in the area which would contain the post-hydro water. The core showed a rust pattern that was consistent with the storage position of the valve. Target Rock did not consider disassembly of the third valve to be necessary since the evidence from the first two valves was consistent.

As is occasionally Target Rock's practice in the case of special orders, the work on these SRVs, including new SOV shell assembly, hydrostatic testing (for strength of pressure-retaining parts and fasteners), final assembly, installation and final tightness and functional testing, was done at the Farmingdale facility by Target Rock field service personnel instead of production assembly and test personnel. Although field service personnel routinely service SRVs in the field, including disassembly, cleaning, software renewal and reassembly of the SOVs, they do not perform hydrostatic tests for strength in the field because all pressure retaining parts, including any new parts they use, have all been hydrostatically tested during production at Farmingdale. Field service personnel only perform leak tightness tests on reassembled SOVs using air or nitrogen gas in the field. In this case however, the field service personnel were building new SOVs at Farmingdale from new factory parts that had not yet been hydrostatically tested. The hydrostatic test and its associated preparations and restorations are included in Section 3.2 of Target Rock Procedure TRP-4754 (currently Revision D, dated August 3, 1993). Accordingly, the field service technicians reportedly assembled the pressure-retaining parts including a valve body, bottom plate, bonnet/core tube assembly, and solenoid assembly (includes the bolting flange) into what is called a shell assembly (without the internals, i.e., plunger and adjusting rod) in accordance with Step 3.2.1 of TRP-4754 in preparation for performing the hydrostatic test. After performing the hydrostatic test itself per Step 3.2.2, the procedure step then called for cleaning and drying the shell assembly (without internals) in accordance with TRP-1595. The traveller had a signoff block for completion of the hydrostatic test, but since the drying requirement was included with the hydrostatic test step, there was no separate signoff on the traveller for drying.

The work performed by the field service personnel on the SOVs in question was not documented completely on the factory traveler. Target Rock stated that the field service personnel apparently used the field procedure instead as overall guidance. Target Rock records from October 1993 show that the hydrostatic test had to be repeated on the SOVs because the records of the original hydrostatic tests were misplaced, although they were found later. Target Rock concluded that the SOVs were not properly dried by the field service personnel after the second set of tests, most probably because the circumstances strongly suggested that the internals (solenoid plunger and adjusting rod assembly) were not removed for the second hydrostatic tests. Hydrostatic testing of SOVs with their internals installed is not prescribed by factory procedures because it would trap water in the air gap between the plunger and the fixed core at the end of the core tube and along the annulus formed by the plunger and core tube. Target Rock's standard drying procedure, using compressed air or nitrogen on SOVs without their internals (i.e, so-called shells), would not be effective in removing all moisture from a fully assembled SOV. The records do not explicitly state that the drying step was completed, but even if it had been, normal drying methods would not likely have been able to remove all the trapped water.

Target Rock concluded that the internals were not removed because that would have meant having to reperform all the functional adjustments and tests on the SOV. However, performing the hydrostatic tests on the SOVs with the internals installed would be contrary to Target Rock procedures which would constitute a nonconformance with respect to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B (quality assurance) to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50, Appendix B). In addition, measures established to ensure that nonconforming material is not used were not adequate. Although Target Rock's postulated root cause scenario fits all the evidence and is the most logical explanation for the failure of the SOVs, it is not known for certain that the events took place in the manner postulated. Nevertheless, nonconforming material (i.e., SOVs containing residual water) was supplied, later used, and failed in an NRC-licensed facility. Therefore, the activities affecting quality related to the preparation of this material for supply were not adequately controlled due either to procedural inadequacies, or noncompliance, or both (95-01-01).

As corrective action, Target Rock prepared a problem report in accordance with its procedures. The inspectors reviewed the problem report, PR-018 dated February 16, 1995. Target Rock concluded that the problem was caused by the failure of the field personnel to follow the procedure which was routinely used successfully by its factory personnel. To prevent recurrence Target Rock identified four actions:

- Revise the field procedure TRP 3959 to specify that work done in the factory would be done fully in accordance with established factory procedures regardless of who was doing the work, production personnel or field service personnel.

  Target Rock personnel stated that they would require factory work to be done only by factory procedures and not a combination of field and factory procedures. Target Rock reported subsequent to the inspection that the procedural corrective actions were completed as of August 1, 1995.

- Train/Retrain field service personnel for proper assembly and test of production hardware (most field service personnel have had factory experience).

  The inspector reviewed the completed training record dated June 2, 1995, and found it acceptable.

- Revise the assembly and test procedure, TRP 4574, to add a specific sign-off for cleaning and drying.

  The inspector reviewed the procedure change and found it acceptable.

* Advise all affected customers to perform a "click" test of the Model 1/2-SMS-S-02 valves prior to installation of the valves in plant. A click test consists of a bench energization of the SOV and listening for a click, i.e., the sound of its actuation indicating free movement of the SOV's solenoid plunger/stem and disc assembly.

  The inspector reviewed a sample letter dated May 1, 1995, typical of the letters sent to all Target Rock customers, advising them of the problem, Target Rock's view of its limited nature, and the recommendation for a click test and found it acceptable.

The inspectors considered the Target Rock problem report corrective actions to be appropriate and adequate.

The inspectors noted that the Target Rock evaluation of reportability in the problem report concluded that the problem was not reportable under 10 CFR Part 21. The inspectors agreed that the problem was not necessarily reportable to the NRC under 10 CFR 21.21(a), but may still be reportable to all affected licensees or purchasers under §21.21(b). The inspectors considered that the rationale for Target Rock's reportability determination was in error. The stated rationale was that the failure of the solenoid did not affect the "automatic or safety function" of the relief valve. The inspectors pointed out that although the automatic function of the valve (pressure relief on high pressure) was not affected, the valve's other safety function, to respond to a manual signal (or in some installations, to the automatic depressurization system signal, although not at Cooper) to open and quickly reduce system pressure (in certain reactor event scenarios) was affected. Therefore, it was not accurate for the problem report to conclude that the problem was not reportable for the reason of its safety function not being affected. The inspectors pointed out that 10 CFR 21.21(b) requires the supplier to notify purchasers when the supplier cannot perform the safety evaluation. Since Target Rock had notified its customers, it had met that requirement. Further, since only Cooper had been affected and since Cooper had notified the NRC, no further action except documentation appeared to be required under Part 21.

The inspectors further questioned Target Rock personnel regarding their rationale for considering that the problem was limited to the three valves supplied for Cooper. Target Rock representatives stated that they had interviewed the field service supervisor and his recollection was that no other similar job, requiring field service personnel to hydrostatically test a solenoid valve, had been performed. Further, in their field work, which involved disassembly of about ten solenoids a year at different facilities, field service personnel reportedly had seen no other evidence of corrosion in the solenoid valves. The inspectors also interviewed the Field Service Supervisor who confirmed this information. Additionally, Target Rock reported that it had experienced no other test failures during its routine performance of periodic relief valve testing for the facilities with the type of MSSRVs in question and stated it had received no other reports from customers regarding any other failures to operate. The field service supervisor reported reviewing slightly more than one year's worth of work records to further verify that no other solenoid valves had been hydrostatically tested by field personnel. Also, consultation with cognizant NRC staff members and review of NRC historical information did not reveal any other instances of similar failures of Target Rock SRVs. Therefore, the inspectors concluded that the Target Rock rationale for considering the problem to be an isolated case was reasonable.

3.2 Relief Valve Main Spring Relaxation and Tip Breakage

The inspectors also discussed with Target Rock representatives a problem with relief valve main spring apparent relaxation and spring tip breakage. The problem was described in Target Rock Problem Report PR-20, dated June 14, 1995. The problem involved apparent spring relaxation and tip breakage that was observed after extensive testing. The limited steam generation capacity of the Target Rock test facility requires that the SRV discharge flow be restricted in order to maintain the pressure necessary to confirm proper SRV operation through its operating cycle. The discharge flow is reduced to within the steam flow capacity of the test facility by installing a restrictor in the discharge port of the SRV under test. The problem revealed itself when an SRV under test (after about 3000 actuations) failed to close. Disassembly showed that the main disc return spring free length was reduced, the normally circular cross section spring had slightly flattened faces between coils, and some small pieces of the spring tip (on the end away from the disc) had broken off. Target Rock examined the circumstances under which the failure occurred and determined that under test conditions, without the normal flow, the differential pressure normally felt across the main disc is significantly reduced, and in fact immediately goes nearly to zero as soon as the disc first lifts off its seat. With little flow-generated differential pressure to retard its motion, the disc, stem and main operating piston assembly literally slam open with the high-pressure opening force across the piston. The piston's motion is terminated when it contacts the head of the main operating cylinder. The spring is not normally completely collapsed because of an annular recess provided for it in the cylinder head, but the tremendous momentum imparted to the return spring coils or turns during testing causes them to bunch up at the non-piston end of the spring until most of them slam into one another. According to Target Rock, this caused the apparent relaxation (reduction of free length), flattened inter-coil surfaces, and tip breakage experienced on the affected test SRV.

The Target Rock engineer determined that without the discharge flow restriction (as in the plant) the differential pressure across the valve generated by unrestricted flow would allow the valve to close normally in an operating environment. He further stated that spring tip fragments might cause scoring in the operating cylinder, but that the phenomenon had not been seen or reported. During the Target Rock testing in which the problem was first observed, a fragment of a spring tip was passed through the pneumatic actuator of the SRV under test, but the Target Rock engineer conceded that it was possible for such a fragment to become lodged in the valve internals during testing. In this case, the Target Rock engineer explained, although a lodged fragment would not be likely to affect automatic self actuation of the valve on high pressure in the plant (the safety relief function), and it would not affect reclosing under normal flow conditions as it did under the restricted flow during bench testing, it could possibly affect the ability of the valve to be opened remotely (electro-pneumatic operation), which could impact operability of a valve used for the ADS.

Target Rock initially identified this potential problem several years ago, but had not yet reported it because Target Rock had determined that it was aware of the condition of all its valves because Target Rock services the valves and monitors their condition in the field and the condition had not manifested itself until after 3000 test operations of an SRV using reduced flow. The Target Rock problem report on this issue indicated that Target Rock planned to issue a service bulletin to all potentially affected facilities by August 16, 1995, to alert them to the problem. To be conservative, the bulletin will recommend replacement of SRV main disc springs after 100 actuations (bench test, restricted flow actuations). Target Rock explained that it had decided to issue the service bulletin (and possibly also to recommend that GE NE issue a service information letter (SIL) pursuant to the requirements of 10 CFR 21.21(b). The NRC will follow up on Target Rock's action on this issue and may review it in a future inspection (Open Item 99900060/95-01-02).

3.3 10 CFR Part 21 Procedures

The inspector reviewed the effective revision of procedures adopted pursuant to 10 CFR 21.21(a), contained in QCI-1306, Revision C, dated August 10, 1994, and found that QCI-1306 was generally up to date and consistent with the requirements of the regulation. However the inspector noted some weaknesses in the procedure, which left uncorrected, could fail to prevent violation of certain provisions of 10 CFR Part 21. In one instance, the weakness would have previously constituted a Severity Level V violation of 10 CFR 21.21(a) in accordance with Supplement VII of Appendix C to 10 CFR Part 2, the previous enforcement guidance of the NRC's Rules of Practice. However, under the new NRC enforcement policy (as promulgated in NUREG 1600) that became effective on June 30, 1995, this weakness constituted a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the new NRC Enforcement Policy. The specific weaknesses noted in QCI 1306 were as follows:

3.3.1 Paragraph 6.2 requires reading and understanding QCI-1306, Section 206 of the Energy Reorganization Act of 1974, and 10 CFR 21.21. However, other sections of Part 21, also relevant to Target Rock scope of activities are omitted.

3.3.2 Apparently pursuant to 10 CFR 21.6(b), Paragraph 7.1(a) of QCI-1306 calls for posting a notice that describes the instruction, identifies those to whom reports should be made, and states where the instruction and its so-called "implementing" reports may be found and examined. However, including in the notice the location of implementing reports is not consistent with 10 CFR 21.6(b) which requires that the notice posted in addition to Section 206 and in lieu of the regulation and the procedures adopted pursuant to the regulation include a description of the regulation as well as the procedures. In addition, §21.6(b) requires that the notice state where "they" may be examined. The language of Paragraph 7.1(a) of QCI-1306 indicates that Target Rock interpreted the word "they" in §21.6(b) to be referring to its immediate antecedent noun, i.e., the reports. The purpose of the notice allowed by §21.6(b), in addition to identifying persons to whom reports should be made, is to make employees aware of 10 CFR Part 21 and the procedures adopted pursuant to it, the purpose and function of the regulation and the procedures and where the regulation and the procedures may be examined. The notice states where the procedures and regulation may be examined because the notice is posted in lieu of the entire regulation and the entire procedures when posting them is not practicable. Making employees aware of where Part 21 reports may be examined was not intended to be a required function of the notice. Finally, it was not clear what was meant by the term "implementing reports."

3.3.3 Paragraph 7.3.2 of QCI-1306 required that the General Manager or designee(s) be "informed...within 5 working days after completion of the evaluation." Although it appeared that the General Manager was to be informed of the conditions stated in Paragraph 7.3.1, it was not stated what the General Manager was to be informed of, nor if the evaluation being referred to was the one described in Paragraph 7.3.3. While it is within Target Rock's prerogative to require that its General Manager be informed of so-called potential defects, deviations, or failures to comply, QCI-1306 lacked a provision required by §21.21(a) to be included in procedures adopted pursuant to the regulation, to implement §21.21(a)(3) which explicitly required that a **director or responsible officer** be informed of a **defect** or a failure to comply **associated with a substantial safety hazard** within 5 working after completion of the evaluation **described in §21.21(a)(1)**.

In response to this concern, Target Rock explained that its General Manager is also its President (and also a Vice President of Curtis-Wright Corporation, of which Target Rock is a wholly-owned subsidiary). The General Manager being also the President of Target Rock satisfied the director or responsible officer requirement, but the inspector pointed out that using the words "or designees" without making it clear that such designees must also meet the Part 21 requirement of being themselves directors or responsible officers may lead to failure to comply with §21.21(a)(3) should a defect be reported only to an ineligible designee.

The inspector also noted that other language in QCI-1306 was quite effective, particularly Paragraph 7.3, and noted the significant strength that Target Rock's QA procedures governing control of nonconforming materials and corrective action were integrated or fed into the Part 21 process. The inspectors found no instances in which Target Rock had failed to handle and document deviations or failures to comply in a manner inconsistent with the requirements of Part 21.

Finally, the inspectors had noted in the review of problem reports, the one associated with the MSSRV issue in particular, that Target Rock made a practice of making what amounted to a recommendation to its customers regarding reportability of a given issue pursuant to Part 21. The inspectors pointed out that the statement of this type in the MSSRV solenoid problem report was misleading because the evaluation that formed the basis for the conclusion of nonreportability did not consider all possible safety functions of the electropneumatic controls of the MSSRV. Although the failed valves at Cooper were not in the ADS group, the inspectors pointed out that, in general, Target Rock as a vendor (other than an NSSS or A/E), would only be expected to provide affected licensees or purchasers with all pertinent information it has relative to a deviation or failure to comply as provided for in §21.21(b) and would not be expected to suggest that something is not reportable unless Target Rock were sure that it is qualified to perform the evaluation described in §21.21(a).

4.0 PERSONS CONTACTED

Andrew L. Szeglin, Senior Design Engineer

Edward Champey, Jr., Director, Quality Assurance

Robert E. Glazier, Manager, Quality Engineering

James D. White, Sales and Service Manager

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

September 14, 1995

Mr. Ronald H. Koga
General Manager
Commercial Nuclear Fuel Division
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: NONPROPRIETARY VERSION OF NRC INSPECTION REPORT 99900005/95-01

Dear Mr. Koga:

This letter transmits the nonproprietary version of the U.S. Nuclear Regulatory Commission's (NRC's) Inspection Report No. 99900005/95-01. Our letter to you dated July 25, 1995, transmitted the original (proprietary) version of the report. On the basis of our discussions and review of the information in your August 23, 1995, letter (NTC-NRC-95-4535), and its enclosures (Application For Withholding Proprietary Information From Public Disclosure AW-95-874 and Affidavit AW-95-874), we have concluded that the specific values identified in your letter could be regarded as proprietary and, as such, were removed from the inspection report. In the revised nonproprietary (public) version of the report, the NRC has briefly summarized the deleted text.

Your response to either this letter or our letter dated July 25, 1995, and their enclosures 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 (a) of the NRC "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room. If you have any questions concerning this matter, please contact Steven M. Matthews at (301) 415-3191.

Sincerely,

Robert M. Gallo

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

Docket No.: 99900005

Enclosure: Report No. 99900005/95-01

REVISED NONPROPRIETARY VERSION

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

REPORT NO.:	99900005/95-01
ORGANIZATION:	Westinghouse Electric Corporation Energy Systems Business Unit Commercial Nuclear Fuel Division Pittsburgh, Pennsylvania
ORGANIZATIONAL CONTACT:	Michele M. DeWitt, Manager Quality Policy Deployment Quality and Strategic Management Energy Systems Business Unit
NUCLEAR INDUSTRY ACTIVITY:	Westinghouse Commercial Nuclear Fuel Division provides pressurized-water reactor reload core designs and reload safety analysis, fuel assemblies, fuel-related core components, zirconium alloy fuel clad tubing, and fuel-related inspection services to the U.S. nuclear industry.
INSPECTION DATES:	CNFD/Westinghouse Energy Center - February 5-10, 1995 CNFD Specialty Metals Plant - February 6-10, 1995 CNFD Columbia Plant - February 27-March 10, 1995 CNFD Western Zirconium Plant - March 20-24, 1995
LEAD INSPECTOR:	Steven M. Matthews — 9/11/95 Date Vendor Inspection Section (VIS) Special Inspection Branch (PSIB)
OTHER INSPECTORS:	David H. Brewer, VIS/PSIB Daniel W. Brite, Battelle Pacific Northwest Laboratory Dr. John F. Carew, Brookhaven National Laboratory Ramon L. Cilimberg, Parámeter, Inc. Rodney L. Grow, Parámeter, Inc. Edward D. Kendrick, Reactor Systems Branch, Division of Systems Safety and Analysis Fatrick S. Lacy, Parámeter, Inc. Roger Mendez, VIS/PSIB Dr. Partha Neogy, Brookhaven National Laboratory
REVIEWED BY:	Gregory C. Cwalina, Chief, VIS/PSIB — 9/11/95 Date
APPROVED BY:	Robert M. Gallo, Chief, PSIB/DISP — 9/13/95 Date

ENCLOSURE

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1 SCOPE AND SUMMARY OF INSPECTION FINDINGS:

During this inspection, the NRC inspection team (team) evaluated the Westinghouse Electric Corporation (W), Energy Systems Business Unit (ESBU), Commercial Nuclear Fuel Division (CNFD) management, staff, and quality programs and the implementation of those programs related to pressurized-water reactor (PWR) reload core designs and reload safety analysis, fuel assemblies, fuel-related core components, zirconium alloy (zircaloy) fuel clad tubing, and fuel-related inspection services. These inspections were conducted to provide a basis for confidence that these items and services supplied to the U.S. nuclear industry would perform their safety function. The inspection basis consisted of the following:

(a) General Design Criterion (GDC) 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50).

(b) Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

(c) Part 21, "Notification of Failure to Comply or Existence of a Defect," of 10 CFR.

(d) W ESBU Topical Report, documented in WCAP-8370, "Quality Assurance Plan," Revision 12A, dated April 1992, approved by the NRC on April 23, 1992, as meeting the requirements of Appendix B to 10 CFR Part 50, and as amended by the updated organizational charts submitted by W on June 14, 1994, hereafter referred to as the "QA topical report."

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

No nonconformances were identified during this inspection.

1.3 Weaknesses and Observations

A few weaknesses, primarily in the procedural conformance of certain activities that affect quality, were identified during this inspection. Neither the weaknesses nor the observations described in the inpsection report require any specific action by or written response from CNFD.

1.4 Open Item

1.4.1 Open Item 95-01-01

As described in Section 3.5.10.2, "Chemical Laboratory," of this report, the team observed weaknesses in certain calibration practices of CNFD COLA. The team requested CNFD to notify the NRC when its analysis of the calibration practices, and corrective actions taken if any, have been completed.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 CNFD Specialty Metals Plant - 99900005/86-01

During an inspection of CNFD Specialty Metals Plant (SMP) conducted on August 18-20, 1986, an NRC inspection team determined that certain CNFD SMP activities were not conducted in accordance with NRC requirements. The following nonconformances, issued by the staff on October 3, 1986, and the associated corrective actions taken by CNFD SMP were evaluated by the team during this inspection. On the basis of its evaluation of the CNFD SMP corrective actions, the team determined that each of the nonconformances was closed.

2.1.1 Nonconformance 86-01-01 (CLOSED)

Contrary to Section 17 of WCAP-8370/7800, Revision 10A/6A, dated August 1984, the standard used in ultrasonic testing (UT) to validate tube ovality was not serialized and maintained under the equipment calibration control system nor were the dimensions of the standard traceable to the National Bureau of Standards.

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

2.1.2 Nonconformance 86-01-02 (CLOSED)

Contrary to Section 17 of WCAP-8370/7800, Revision 10A/6A, dated August 1984, the data for vertical linearity, horizontal linearity, and calibrated attenuation were not recorded on the calibration data reports, dated February 24, 1986, and March 27, 1986, for two UT flow detectors (Nos. 33 and 40).

During its evaluation of the CNFD SMP tube reduction process, the team observed that CNFD SMP had appropriately implemented its inspection requirements with the correct UT data recorded on calibration data reports. Therefore, the team concluded that CNFD SMP corrective actions appeared adequate to ensure that the appropriate UT data were recorded on calibration reports to support the acceptability of tube hollows.

2.1.3 Nonconformance 86-01-03 (CLOSED)

Contrary to Section 17 of WCAP-8370/7800 Revision 10A/6A, dated August 1984, Quality Services Lab Procedures QS-213, QS-249, QS-261, and QS-262 contained references to QS-118, which became an obsolete procedure on July 7, 1985, when it was superseded by Produce Assurance (PA) procedure PA-103.

The team determined that CNFD SMP maintained current documents in computer files, thereby permitting the documents to be searched for specified words or phrases by the search routine in the program. This technique provided the basis for searching current documents for references that had been changed. Based on evaluation of CNFD SMP document management, the team considered this nonconformance closed.

2.1.4 Unresolved Item 86-01-01 (CLOSED)

Paragraph 4.5.3 of material specification NFD-31008, "Seamless Zircaloy-4 Tubing," Revision 28, dated May 16, 1986, required, in part, that tube ovality not exceed 0.0013-inch total indicator reading (TIR). Paragraph 6.4 of procedure QC-301, "Final Inspection - Ultrasonic Dimensional Setup and Calibration," Revision J (QC-301), outlines the method used to certify that this dimensional requirement is met. Although procedure QC-301 is accurate, it is unclear whether the actual requirement of TIR is being met.

The team determined that Revision L of procedure QC-301, dated July 6, 1989, requires that the TIR reading be made in a helical pitch rather than a stationary plane. This customer requirement clarifies this concern; the team had assumed that the TIR reading was to be taken in a stationary plane.

2.2 <u>CNFD Columbia Plant - 99900005/92-01</u>

During an inspection of CNFD Columbia Plant (COLA) conducted from January 13-17, 1992, an NRC inspection team determined that certain CNFD COLA activities were not conducted in accordance with NRC requirements. The following nonconformance, issued by the staff on February 6, 1992, and the associated corrective actions taken by CNFD COLA were evaluated by the team during this inspection. On the basis of its evaluation of the CNFD COLA corrective actions, the team determined that the nonconformance was closed.

2.2.1 Nonconformance 92-01-01 (CLOSED)

Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 7 of CNFD COLA Administrative Procedure CA-006, "Columbia Plant Training Policy," Revision 3, dated October 12, 1990 (CA-006), an operator performed the pre-plug/pre-weld operation for several weeks in accordance with an outdated revision to the governing procedure before acknowledging the correct revision to the procedure.

As part of the corrective actions, CNFD COLA removed references to the 5 day rule. The revised procedure CA-006 required that operations personnel review and sign-off a new or revised procedure before performing the operation covered by the procedure. In addition, the team was informed by the operation supervisor that no revision to a procedure would take effect until the Thursday of any week, allowing sufficient time to inform operators of pending changes to the procedures. During the inspection, several minor observations were noted regarding the sign-off of some procedures as discussed in this report; however, product quality was not affected. On the basis of its evaluation of the CNFD COLA corrective actions, the team determined that the nonconformance was closed.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

Westinghouse is a diversified, technology-based corporation founded in 1886. The CNFD, a charter recipient of the Malcolm Baldrige National Quality Award, is one of several divisions and business areas of the W ESBU. Other organizations within the W ESBU that perform fuel-related activities evaluated during this inspection are the Nuclear Technology Division, Nuclear Services Division, and Quality and Strategic Management.

In addition to completed fuel assemblies, CNFD supplies zircaloy fuel clad tubing, fuel-related core components, and engineering services and training to licensees, other fuel vendors, and utilities worldwide.

Over 2,000 people were employed by CNFD at four locations and CNFD was the only fully integrated supplier of nuclear fuel products and services in the United States. At the CNFD Western Zirconium Plant (WZ) in Ogden, Utah, zircon sand was converted to zircaloy, which was used to fabricate fuel clad tubing and other fuel-related core components. Extruded zircaloy from CNFD WZ was shipped to the CNFD SMP in Blairsville, Pennsylvania, for finishing. Finished fuel clad tubing was then shipped from CNFD SMP to CNFD COLA in Columbia, South Carolina, where completed fuel assemblies and fuel-related core components were fabricated. The CNFD reload core design and reload safety analysis engineering personnel were located at the division's headquarters, the Westinghouse Energy Center, in Monroeville, Pennsylvania.

The plant managers of CNFD WZ, CNFD SMP, and CNFD COLA all report to the general manager of CNFD. CNFD was managed with a "one-roof manufacturing" philosophy; that is, product flowing from one plant site to the other was treated as though it came from another department within the same organization. As implemented by CNFD, this philosophy meant that product parameters were not reinspected when products arrived at the subsequent plant site; material was checked for shipping damage and placed in production.

The "one-roof manufacturing" philosophy was implemented in several other ways. For example, the Fuel Performance Technology group (located at the Westinghouse Energy Center) of the Produc /Process Development & Design (P/PD&D) group of CNFD (P/PD&D management and most of its staff are located at CNFD COLA) includes a Product Design/Development group that consists of materials engineers and mechanical engineers that spend most of their time at CNFD SMP. The P/PD&D Product Dr jn/Development group was responsible for material specifications, maintained at CNFD SMP, and drawing definition, maintained at CNFD COLA. This group interfaced with other organizations at CNFD SMP and CNFD COLA by regularly scheduled, informal telephone conferences, in which the P/PD&D Product Design/Development group collectively dealt with questions arising at any of CNFD plants.

The Error Free Performance Team (EFPT), described in Section 3.3.4 of this report, was another way in which the CNFD "one-roof manufacturing" philosophy was implemented. Although EFPT management was located at the Westinghouse Energy Center, the EFPT membership and participation was from CNFD WZ, CNFD SMP, and CNFD COLA.

3.2 Entrance Meetings, Interim Exit Meetings, and Final Exit Meeting

For each of the following inspections, the team conducted an entrance meeting on the first day of the inspection.

- February 5-10, 1995 — CNFD/Westinghouse Energy Center
4350 Northern Pike
Monroeville, Pennsylvania
- February 6-10, 1995 — CNFD Specialty Metals Plant (SMP)
Westinghouse Road, R.D.4
Blairsville, Pennsylvania
- February 27 - March 10, 1995 — CNFD Columbia Plant (COLA)
5801 Bluff Road
Columbia, South Carolina
- March 20-24, 1995 — CNFD Western Zirconium Plant (WZ)
10,000 West 900 Street
Ogden, Utah

During each of the entrance meetings, the team met with members of the CNFD management and staff, discussed the scope of the inspection, reviewed the team's and CNFD's responsibilities for handling proprietary information, and established contact persons for the team within the management and staff of the applicable CNFD organization.

During the inspection periods described above, the team conducted a performance-based inspection of CNFD through technically directed observations and evaluations of processes, activities, and documentation. The team (a) examined technical documentation, procedures, and representative records, (b) interviewed CNFD personnel, (c) held discussions with CNFD personnel, (d) listened to presentations by CNFD personnel, and (e) made other observations. The specific areas examined, the documentation reviewed, and the team findings are described in this report. The persons who participated in and who were contacted during this inspection are listed in Appendix A to this report.

On the last day of each of the inspection periods described above, the team conducted an interim exit meeting to outline to CNFD management and staff major concerns, weaknesses, strengths, and observations identified by the team during that portion of the inspection.

During its closing exit meeting at the Westinghouse Energy Center in Monroeville, Pennsylvania, on April 13, 1995, with CNFD management and staff, the team summarized the inspection findings, weaknesses, strengths, open items and observations.

3.3 CNFD/Westinghouse Energy Center

In inspecting the CNFD activities at the Westinghouse Energy Center, the team evaluated (a) the reload core design and reload safety analysis process, (b) the fuel mechanical design process, and (c) fuel-related inspection services.

3.3.1 Reload Core Design and Reload Safety Analysis Process

CNFD produces 30 to 40 reload core designs and related engineering services to support licensing and plant operations each year; and this process was facilitated through a high degree of automation, including an automated Calc Note system that was geared towards error reduction and uniform documentation. Each reload core design requires a complete core design and safety analysis. The results of the reload core design and reload safety analysis were documented in the Reload Safety Evaluation Report (RSE) provided to the licensee. The reload analyses were performed with methods that were documented in NRC-approved W topical reports. Both full-scope and split-scope reload design evaluations were performed. In split-scope evaluations, the licensee performs selected parts of the reload core design (typically the core neutronics analysis) and CNFD performs the remaining analyses required to complete the reload core design. CNFD had released some of its core design computer codes and methods to licensees and provided the training required for the proper application of these methods.

In inspecting the CNFD reload core design and reload safety analysis process, the team evaluated the activities of (a) the CNFD, performed by the Core Engineering group located at the CNFD/Westinghouse Energy Center and by the Product/Process Development and Design (P/PD&D) group located at CNFD COLA, and (b) the Nuclear Technology Division (NTD) performed by the Nuclear Safety Analysis (NSA) group. Where the activities of these groups related to the reload core design and the reload safety analysis process, the team conducted a detailed evaluation of those activities by selecting certain reload core design packages. The team evaluated the design inputs, design processes, software controls, design verifications, design change controls, interface controls, and documentation and records. The evaluation of selected reload core design packages also covered the steady-state neutronics, thermal-hydraulics (T/H) design, transient analysis, fuel mechanical performance, core monitoring, and set point analysis.

Evaluation of the INCORE code (used to perform on-line core surveillance using in-core flux measurements) included the input preparation, transmittal of the INCORE data sets to the licensee, comparisons of predicted and measured neutron flux distributions, and licensee feedback. The performance of a reload core is evaluated during the operating cycle by monitoring the results of startup tests, critical boron concentration and core power distribution measurements, and coolant chemistry data obtained and transmitted by the licensee. These comparisons and test results were used to evaluate the reload core design. The team also evaluated the RSE report and the related Core Operating Limits Report (COLR). The team evaluated the reload core design analysis computer codes and verified that NRC-approved codes were being used and that NRC-developed Safety Evaluation Report (SER) restrictions and limitations were being observed. The team found that the reload core design activities were generally being adequately performed, with the few exceptions noted below.

In order to optimize the reload core design evaluation, CNFD employed a bounding analysis approach in which the cycle-specific core design was bounded by a previously analyzed reference core design. In this approach, many of the cycle-specific safety analyses were not required and the reload core design evaluation was greatly simplified. Where a bounding reference analysis could not be identified for a particular reload core design, a cycle-specific analysis was performed for those aspects that were not bounded by a reference core design.

The team examined certain reload core design packages by evaluating the reload core design and reload safety analysis process, starting from the end products (deliverables to the licensee). The two key deliverables examined were the RSE and COLR. From these documents, results were selected and traced to their source to determine if the analysis process was performed in accordance with the procedures and was adequately documented.

In inspecting other reload core design packages, the team began at the front end of the design process by interviewing the cognizant project engineer, and then examining the project interface documents. These interface documents included those internal to the CNFD, the NTD, CNFD COLA, and the NRC licensees.

To select the plant-specific reload core designs to be evaluated, the team reviewed current reload core design issues of special importance. The team identified more than 20 reload core design issues of special interest, among them the following: (a) vendor/licensee interface concerns (e.g., split-scope designs), (b) recent operational problems, (c) special-purpose fuel designs (e.g., flux suppression fuel assemblies), (d) new fuel designs, (e) recent licensing issues (e.g., asymmetric rod cluster control assembly (RCCA) withdrawal), and (f) issues identified in recent Licensee Event Reports (e.g., misalignment of wet annular burnable absorber (WABA) rods). These issues and their treatment in recent reload core designs were discussed during the initial meetings with CNFD. On the basis of these discussions, the team selected five reload core design packages to evaluate the reload core design and reload safety analysis process, and CNFD's response to the most significant reload issues:

- South Carolina Electric & Gas Company, Virgil C. Summer Nuclear Station Cycle 9
- Wisconsin Electric Power Company, Point Beach Nuclear Plant Unit 2 Cycle 21
- Commonwealth Edison Company, Zion Station Unit 1 Cycle 14
- Public Service Electric & Gas Company, Salem Nuclear Generating Station Unit 2 Cycle 9
- Houston Lighting & Power Company, South Texas Project Unit 2 Cycle 4

3.3.1.1 Reload Core Design Process

The CNFD reload core design process comprised four steps:

(a) Core Design and Steady-State Analysis (utilizing the ALPHA code for Automated Linkage of the PHOENIX-P and Advanced Nodal Code (ANC), the PHOENIX-P code used to generate cross-sections, the ANC code used for two-dimensional (2D) radial and three-dimensional (3D) core analysis, and the PHIRE post-processing code for PHOENIX-P data banks);

(b) Operational Strategy and Analyses (utilizing the APOLLO code for 1D (axial) core analysis), the VENUS code used to perform peaking factors synthesis for Constant Axial Offset Control (CAOC) analysis and Relaxation of Constant Axial Offset Control (RAOC) analysis, the ALUCARD code used to generate INCORE constants, and the INCORE code);

(c) Fuel Management (utilizing the Advanced Loading Pattern Search (ALPS); and

(d) Core Monitoring (utilizing the SPNOVA code and BEACON, the W On-Line Core Monitoring System code).

The core design process was described in detail in the Methods Communication manual (METCOM) as well as in the individual code manual. The team found the four volume METCOM document to be comprehensive and rich in analytical and procedural detail. The METCOM documented the design and quality objectives, responsibilities, and requirements for the reload process. The manual included summaries of certain codes, associated detailed modeling instructions, and selected reactor systems data. In addition, detailed procedures are provided for determining the accident-specific input to the RSAC. The METCOM manual provides substantially more detail than is included in the W topical reports.

The METCOM manual is reviewed every 3 years and updated as necessary (typically every 6 months), as required by Engineering Procedure (EP) procedure EP-105, "Design Manuals," Revision 5, dated February 1, 1993 (EP-105). In response to this requirement, CNFD established a METCOM team that continuously reviews and updates the manual and responds to users' needs. This activity typically results in approximately 10 METCOM revisions per year. During the course of the team's review, several minor METCOM omissions or errors were identified and discussed with the Core Engineering staff. Westinghouse CNFD stated that these would be evaluated and considered for possible inclusion in future METCOM updates.

The reload core design process began when CNFD received the Reload Schedule and Energy Requirements (RSER) document from the licensee. Core Engineering determined the fuel enrichment, the integral fuel burnable absorber (IFBA) design, the fuel rod design, burnup limits, number of fuel assemblies, and loading patterns. Fuel rod design limits were confirmed by the Fuel Analysis group in Core Engineering, and this confirmation was documented in the RSE. The Core Engineering, Core Design group determined the boron concentration, as documented in the Boron Design Requirements (BORDER). The Core Design group determined the peaking factors and power shapes and provided them to the Core Engineering, Fuel Analysis group for T/H design. The core design process also involved an extensive physics database including design data and hot zero-power (HZP) and hot full-power (HFP) data. The Fuel Analysis group provided the departure from nucleate boiling ratio (DNBR) and fuel temperature inputs to the Reload Safety Analysis Checklist (RSAC), an input to NTD. The Core Design group also determined the reactivity parameters, RCCA rod worths, and kinetics parameters, all of which were also inputs to the RSAC. The CNFD P/PD&D group prepared the Design Evaluation Verification List (DEVL) and the Fuel Parameters Checklist (FPC) and provided these documents to Core Engineering. The Core Engineering, Fuel Licensing Integration group summarized any mechanical design changes identified in the DEVL and FPC and provided that data to NTD. Also, the reload-specific RSAC document was completed and transmitted to NTD by Core Engineering. The CNFD groups that provided input to the RSE were Core Engineering, Core Design, P/PD&D, and Core Engineering, Fuel Analysis (T/H analysis and fuel rod design). The Fuel Licensing Integration group was responsible for completing the final RSE and incorporating the inputs from CNFD and NTD.

3.3.1.2 Reload Safety Analysis Process

The CNFD reload safety evaluation methodology was based on WCAP-9272, "Westinghouse Reload Evaluation Methodology," dated March 1978, and made extensive use of the RSAC. The methodology provided the basic methods for routine evaluation of reload core safety. It utilized a perturbation approach to determine whether key safety parameters for the reload core design (i.e., design parameters that have non-negligible impact on the safety performance of the core) were bounded by values used in the reference reload safety analysis. The approach tended to minimize the effort spent in reanalysis at the expense of being somewhat over-conservative. Uncertainties were included in most, though not all, key safety parameter values. Because the overall conservatism of the reload safety evaluation methodology, the team judged that the neglect of explicit accounting of uncertainties in a few safety parameters did not alter the net conservatism of the approach. The RSAC did not contain explicit values of the key safety parameters for the reload core design, only a determination that the values were bounded (or not bounded) by those assumed in the reference safety analysis. CNFD argued that this process ensured that the appropriate safety margins were managed and controlled by a single group (the CNFD Core Engineering group). The team felt that not recording the current values of the key safety parameters on the RSAC increased the chances of an error in the comparison to the reference values. However, the team did not uncover an instance where such an error had been made. This team observation requires no specific action nor written response.

The reload safety analysis process began when the Reload Safety and Licensing Checklist (RSLC) document was received from the licensee. A Reload Initialization Questionnaire (RIQ) document was prepared by CNFD and transmitted to the licensee to confirm the current status of the plant, specifically with respect to safety-related operations and design input values. NTD Fluid Systems group, using the BORDER input from CNFD Core Engineering, confirmed that the boron system design requirements were met. Using the identified fuel mechanical design changes and RSAC inputs from CNFD Core Engineering, NTD NSA performed any necessary loss-of-coolant accident (LOCA) analysis and Non-LOCA transient analysis, confirmed the safety analysis, and defined Technical Specification changes, if any. NTD NSA also provided its input to the RSE through CNFD Core Engineering Fuel Licensing Integration group.

Many of the reload safety analyses evaluated by the team appeared to be the result of comparing the current reload core design parameters with earlier bounding analysis values. In many instances, the earlier bounding analysis was completed a number of years ago, in a different culture with a less controlled process, by engineers who are no longer with NTD. Thus, the team observed the potential for interface gaps between past work and present work; current work may be based on a weak understanding of the earlier work. This team observation requires no specific action nor written response.

3.3.1.3 Virgil C. Summer Nuclear Station Cycle 9

In evaluating South Carolina Electric and Gas Company (SCE&G), Virgil C. Summer Nuclear Station (Summer) Cycle 9 reload core design and reload safety analysis, the team began by evaluating the end products; the Reload Safety Evaluation (RSE) and the Core Operating Limits Report (COLR). The team examined these documents to determine key or representative results to be traced to their source. Summer Cycle 9 was a split-responsibility reload core design with the licensee, SCE&G, having responsibility for the nuclear design.

The CNFD engineer functioned as verifier for the nuclear design Calc Notes. Interface documents were examined to determine the control of the split responsibility and the flow of information to and from the licensee. There were no Technical Specification changes for Summer Cycle 9 reload.

The technology transfer and the control of software was examined by first reviewing the applicable procedures in the CNFD Design Engineering Procedure Manual and the Software Engineering Methodology manual, Revision 11, dated October 25, 1994. The process was inspected by examining (a) the original technology transfer to SCE&G, (b) the most recent technology transfer to SCE&G, (c) the development and release of a new code, (d) the updating of an old code, and (e) examples of error reporting. All material inspected was found to be in compliance with the relevant procedures. A demonstration of the STATEPOINT software for configuration monitoring by the supervisor engineer of CNFD Core Engineering Technology Product Services group showed that the configuration control process had been automated.

(1) Thermal-Hydraulic Analysis

The Thermal-Hydraulic (T/H) Design documented in Section 2.3 of the RSE was examined by discussions with the fuel analysis engineer and evaluating the relevant Calc Notes. A weakness was identified as a result of this inspection.

CNFD Engineering Procedure (EP), as documented in EP-302, "Documentation and Verification of Design Analyses," Revision 5, dated November 1, 1992 (EP-302), Revision 42, November 30, 1994, required that analysis such as the T/H analysis for the reconstituted fuel, be documented. The team observed that the RSE for Summer Cycle 9 stated that a T/H evaluation for the fuel rod reconstitution had been performed. However, the relevant CNFD Calc Note, T/H 94-109-0, did not contain any discussion or analysis to support the Cycle 9 reconstituted fuel. The responsible engineer had relied on a topical report without documenting the rational and analysis that showed the relevance of the topical to the current reload core design. The team observed that the referenced topical report, WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology," dated July 1993, was applicable to the Summer Cycle 9 reload core design. The reliance on topical reports without cycle-specific justification was judged by the team to be a poor method that did not conform to CNFD engineering practice.

This weakness was discussed with CNFD personnel and they responded by adding to Calc Note T/H 94-109-0, Part 14 which identified the reconstituted fuel, the relevant CNFD Core Engineering methodology, and the engineering analysis and rationale for the Cycle 9 reconstituted fuel. The CNFD actions taken during the inspection satisfied the team's concerns.

(2) Fuel Mechanical Design

The Mechanical Design, documented in Section 2.1 of the RSE, was examined by discussions with the fuel analysis engineer, and the Calc Note was reviewed relative to the Fuel Rod Design Procedure Manual, Revision 4, dated April 1993. This review identified a weakness.

The team observed that no single document showed that the 11 fuel rod design criteria, specified in the Fuel Rod Design Procedure Manual, were satisfied for the Summer Cycle 9 reload core design. In contrast, the team found that checklists were used frequently in most other aspects of the reload core design evaluation process. Further inspection showed that three different W organizations had performed calculations for this specific reload and that collectively the 11 design criteria were satisfied.

This lack of a single document, such as a checklist, that could be referenced in the RSE approval documentation was a weakness discussed by the team with Core Engineering personnel. CNFD responded by creating a fuel design criteria checklist and stated that the checklist would be included in the next revision of the Fuel Rod Design Procedure Manual. Memo FA-95-052, dated February 9, 1995, was issued instructing engineers to utilize the new checklist and to reference the checklist in the RSE sign-off documentation. The CNFD actions taken during the inspection satisfied the team's concerns.

3.3.1.4 Point Beach Nuclear Plant Unit 2 Cycle 21

In evaluating Wisconsin Electric Power Company (WEPC), Point Beach Nuclear Plant Unit 2 (Point Beach) Cycle 21, the team began at the front end of the reload core design process by interviewing the cognizant project engineer. This interview led to an examination of project interface documents. On the basis of its evaluation of these documents, the team determined that the Point Beach Cycle 21 reload core design and reload safety analysis was challenging in several ways:

(a) Cycle 21 energy requirements increased from 11.2 gigawatt-days per metric tonne of initial uranium metal (GWD/MTU) to 11.8 GWD/MTU. This required changing the reload core design during the Cycle 21 design process from 28 feed fuel assemblies to 29 assemblies.

(b) The design of integral fuel burnable absorber (IFBA) (uranium dioxide (UO_2) fuel pellets coated with zirconium diboride (ZrB_2)) used a longer length fuel rod.

(c) Some of the IFBA loaded fuel assemblies have an asymmetrical IFBA loading pattern with a "half moon" design.

(d) Fuel rods with IFBA loadings had a lower weight percent (w/o) U_{235} enrichment than the surrounding fuel rod enrichment; however, of the two fuel assembly designs with the same enrichment one did not include IFBA and the other included IFBA.

(e) The reload pattern included 12 fuel assemblies with hafnium (Hf) flux suppressor rods and four fuel assemblies with water displacement rods. Each of the four water displacement fuel assemblies contained 12 dummy zircaloy rods in the guide tubes. The Hf flux suppressor rods to be used were already at the Point Beach Unit 2 site, having been used in Cycle 20. The

water displacement rods were to be shipped by CNFD COLA with the reload fuel assemblies. The licensee had the responsibility to load both of these fuel-related core components (Hf flux suppressor and water displacement rods) per the loading plan.

(f) The existence of a quadrant power tilt which was observed during Cycle 21 startup. The size of the tilt varied with Cycle 21 burnup. A tilt had also been observed during Cycle 20 but of a significantly smaller size (< 2%).

The team evaluated a number of Calc Notes and related documents, using the 12 requirements for Calc Notes stated in procedure EP-302. The team determined that all Calc Notes evaluated, with two exceptions, were prepared and verified in accordance with the requirements of procedure EP-302.

However, the team identified three weaknesses. One of the two exceptions noted above, Calc Note T/H-94-086-0, contained fuel rod design results, did not contain a checklist. The team examined this Calc Note to follow up the Summer Cycle 9 reload core design evaluation and confirm whether the same weakness existed in the Point Beach Cycle 21 reload core design. It did not because of the CNFD action taken (Memo FA-95-052) as a result of the team evaluation of the Summer Cycle 9 reload core design. No further action was taken with regard to this weakness.

The team evaluation of the related Calc Notes determined that all of the unusual design features were properly addressed in the reload core design and reload safety analysis.

Examination of Calc Note WII-94-056-0, "Point Beach Unit 2 Cycle 21 Reevaluation of the RSE due to a HFP tilt of 3.3%," confirmed that measured results obtained during startup and early-cycle operation were evaluated for impact on the pre-operation RSE results. This reevaluation identified potential causes of the tilt, e.g., the steam generator plugging imbalance between the two steam generators and a small burnup difference in reinserted fuel assemblies. Corrective actions for Point Beach Cycle 22 were described in the Calc Note. One of these was based on the recognition that small burnup differences in fuel assemblies located symmetrically across the core could exacerbate a small tilt driven by steam generator imbalance or could mitigate the tilt.

(1) Beta-effective

Beta-effective (βeff) is the isotopic and importance-weighted delayed-neutron fraction in core (delayed neutrons are neutrons emitted by fission products sometime after a fission). The team evaluation of Calc Note WII-94-017-0, "Beta Effective for WIS21 RSAC," identified the following procedural and technical weaknesses:

(a) The βeff uncertainty at the lower band was applied incorrectly. METCOM procedure 6.7, "Beta-effective and Prompt Neutron Lifetime," paragraph 6.7-1, clearly required that minimum βeff is calculated by multiplying the best-estimate value by **[Deleted pursuant to 10 CFR 2.790 - Document describes**

a specific value]. However, the Calc Note divided the best-estimate value by **[Deleted pursuant to 10 CFR 2.790 - Document describes a specific value]**, thereby obtaining a slightly larger minimum Beff. Although this methodology was judged by the team to be nonconservative, the nonconservatism was not significant because the nonconservatism in the error (0.25%) was small compared to the available margin.

(b) The verifier noted on the checklist that, contrary to the requirements of METCOM procedure 6.7, the I-factor multiplier of 0.97 on Beff was not applied. However, the team determined that the multiplier had been applied but that the verifier had not closed out this mistakenly observed defect with the author of the Calc Note but rather left the defect unresolved.

(c) The total number of pages were not noted and the last two pages of the Calc Note were not numbered, including the checklist page. As required by EP-302, these provisions were intended to ensure that the Calc Note is complete and auditable.

The team discussed these weaknesses with the author and verifier, and again later with the author, verifier, and manager. CNFD Core Engineering responded to these weaknesses by correcting the Calc Note and reissuing it as Revision 1. The CNFD actions taken during the inspection satisfied the team's concerns.

(2) Doppler Effect

The team evaluation of the Point Beach Cycle 21 inputs into the rod ejection accident evaluation led to investigating the Doppler effect, which is negative reactivity insertion due to an increase in neutron absorption by fuel. This reactivity effect occurs when the fuel temperature is elevated, thus increasing the absorption cross-section of fuel. The Doppler effect is the primary core physics inherently negative fast feedback to power changes and plays a large role in fast reactivity transients. CNFD methodology emphasized the Doppler defect representation as opposed to the Doppler coefficient. The METCOM and other documents such as topical report WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," specified uncertainties on safety-related parameters. An internal memo, WIN 249-5142, "Recommended Design Limits for Reactivity Coefficients for Safety Analysis," provided for an uncertainty to be applied to the Doppler effects. However, METCOM procedure 6.6, "Doppler Coefficient and Defect," Revision 0, dated February 1994, did not provide a Doppler defect uncertainty factor.

The best estimate value of the Doppler defect for this reload core design, if adjusted in the conservative direction by **[Deleted pursuant to 10 CFR 2.790 - Document describes a specific value]** in accordance with the value in WIN 249-5142, would be 3% from the bounding analysis value for the rod ejection reactivity insertion accident. This was not significant for the Point Beach Cycle 21 reload core design since margin (3%) would have remained and the team considered **[Deleted pursuant to 10 CFR 2.790 - Document describes a specific value]** uncertainty to be conservative. However, it was postulated by the team that future reload core designs could have smaller Doppler defects which, if adjusted by an uncertainty factor, would not meet the bounding analysis value.

The team determined that this may not be apparent if an uncertainty factor was not applied since it may not be clear that the Doppler defect value provided by CNFD to NTD was a best estimate and not a value including uncertainty.

The team discussed this weakness with CNFD. It was agreed that the **[Deleted pursuant to 10 CFR 2.790 - Document describes a specific value]** uncertainty factor value in WIN 249-5142 was larger than currently warranted. CNFD also stated that METCOM procedure 6.6 would be modified to make designers aware that Doppler defects calculated by the ANC code were best estimate values without uncertainty factors incorporated. For future evaluations, the designer will factor this into evaluations of margin for Doppler affected transients. CNFD assigned METCOM work item 95-06 to address this issue. The CNFD actions taken during the inspection satisfied the team's concerns.

3.3.1.5 Zion Station Unit 1 Cycle 14

The fuel projects engineer was the W interface for the Commonwealth Edison Company (CEC), Zion Station Unit 1 (Zion) Cycle 14 reload core design and reload safety analysis and provided the Contract and Technical Data (CATD), RSER, and the RSLC to Core Engineering for the core design analyses.[1] The Core Engineering reload core design activities were performed using methodologies that were documented in (a) the METCOM, (b) the Software Engineering Methodology manual, and (c) the Engineering Services Manuals, as required by procedure EP-105. The Zion Cycle 14 reload core design was carried out following the METCOM procedures found in Volumes 1-4 and documented in the Calc Notes Report prepared by the core designer. As part of the review of the Zion Cycle 14 reload core design process, the METCOM procedures, and their application, as documented in the Zion Cycle 14 Calc Notes, were reviewed in detail.

Westinghouse CNFD had established a special F-configuration classification for computer codes that received sufficient verification and validation to be used in the reload core design analyses. All computer codes used in the Zion Cycle 14 reload core design analysis were F-configured. There appeared to be close interaction and good communication between the computer software developers and the Core Engineering code users. This was illustrated in the identification and timely correction of the erroneous xenon (Xe) yield data in the PHOENIX-P library. The effect of this modification on the F-configured PHOENIX-P data were appropriately evaluated and incorporated in the Zion

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[1] The Zion Cycle 14 reload core design was initially a single-scope design with CNFD providing the complete reload core design and reload safety analysis. However, after the initial CNFD reload core design was completed, the licensee, CEC, requested a split-scope and assumed responsibility for the reload core design only. The team evaluation described in this report addresses the initial CNFD reload core design. After the CNFD initial reload core design was completed, CEC performed the reload core design analysis with the W PHOENIX-P code and the ANC code system (installed on a computer system similar to that used by CNFD), with the METCOM manuals, and with training provided by W. For several cycles, CEC has also provided the cycle-specific startup data and constants for the INCORE code.

Cycle 14 reload core design analysis (RSAC Calc Note, CWB-93-013-0, "Evaluation of the Impact of the Xe Yield Error on the Zion 1 Cycle 14 Models").

The Zion Cycle 14 reload core design analysis, documented in the four volumes of Calc Notes, was reviewed in detail. The Zion Cycle 14 calculations of the reload core design parameters input to the RSAC (e.g, reactivity coefficients, power peaking factors, RCCA bank worths) were also reviewed. These analyses were carried out and documented in a manner consistent with the METCOM calculational procedures and requirements. Each Calc Note was verified by an independent reviewer and all comments satisfactorily resolved, as required by METCOM procedure 1.9.

(1) Vessel Fluence Reduction

CEC's core design specification, documented in letter ZIC141003, "Final Energy Specification for Zion 1 Cycle 14," dated September 22, 1992, required that the reload core design employ the strategy of the L4P (low leakage loading pattern) plus Hf flux suppressor rod flux reduction as the vessel fluence reduction option for Zion Cycle 14. The average fuel assembly power for the peripheral fuel assemblies should approximate, or be bounded by, the L4P target assembly powers. The review of the Zion Cycle 14 Calc Notes found no documentation of the calculation made to demonstrate that the proposed Cycle 14 reload core loading provided the required vessel fluence reduction. The lack of documentation confirming the ability of the Zion Cycle 14 reload core design loading pattern to provide the required vessel fluence reduction is considered a weakness in the Calc Note system.

When this omission was brought to the attention of the CNFD staff, a new Calc Note, CWB-95-001-0, dated February 10, 1995, was developed which demonstrated that the Cycle 14 core loading pattern provided the vessel fluence reduction required by the CEC core design specification. The CNFD actions taken during the inspection satisfied the team's concerns.

Several Zion Unit 1 operational issues were discussed during meetings with the CNFD Core Engineering staff. A relatively small core power tilt (≈2% at HFP) had been observed at Zion Unit 1. Several possible causes of the tilt were suggested, including (a) steam generator tube plugging, (b) pump maintenance, or (c) asymmetric fuel shuffle.

During the startup tests for Zion Cycle 13, a quadrant power tilt was observed and ultimately traced to the failure to load the Hf flux suppression rods in the selected peripheral fuel assemblies. The correct core loading pattern, including the location of the IFBA and the Hf flux suppression rods, was transmitted by W CNFD to CEC in Figures 2 and 9, respectively, of the Fuel Loading Pattern Letter, 93 CW-G-0030, "Commonwealth Edison Company Zion Nuclear Power Plant, Zion 1 Cycle 14 Burnable Absorber Requirements and Candidate Loading Pattern," dated March 1, 1993. However, the locations of the Hf rod inserts were apparently not adequately communicated to the Zion Unit 1 site personnel responsible for the core loading. The team reviewed the core loading pattern in Figure 9 and concluded that the indication of the Hf flux suppression rods was adequate. Nevertheless, CNFD stated that the core loading maps are produced by the ANCHOR code (used to aid in verifying ANC loading patterns) and, that to improve the CNFD - CEC interface, ANCHOR had been modified to simplify and improve the presentation of the data on the core loading maps. The team concluded that the flexibility and responsiveness indicated by this corrective action was a strength in the CNFD Core Engineering reload process.

(2) Thermal-Hydraulic Analysis

The Zion Cycle 14 T/H analysis was performed by the CNFD Core Engineering Fuel Analysis group and documented in Calc Note T/H-93-012-0. The analysis included the determination of fuel rod densification and temperatures, DNBR limits, and axial offset limits. Analyses of the loss-of-flow, locked-rotor, rod misalignment and steamline break transients are also included. These analyses used the THINC-IV (thermal-hydraulic interaction, analysis, code) and PAD (fuel rod performance code). The procedures documenting the methods, bases, and assumptions for these analyses were given in the Thermal-Hydraulics Design procedure manual. The Zion Cycle 14 T/H analyses generally followed these procedures. However, in reviewing the T/H Calc Note, the team noted that reload core design information required for the steamline break T/H analysis was obtained in a telephone conversation with CEC. No followup documentation of this communication and data transmittal was available.[2] This undocumented transmittal of design input data did not allow independent verification and was not consistent with the T/H design procedures. The team considered this instance a weakness in the Calc Note system.

(3) LOCA and Non-LOCA Transient and Safety Analyses

The reload safety analyses were performed by the NTD. The non-LOCA transient analyses were performed by the Transient Analysis group and were controlled by a set of Safety Analysis Standards. The LOCA analyses were performed by the Safeguards Engineering group and were controlled by a set of Safeguards Engineering Standards. The reload transient analyses were performed using a bounding analysis approach in which the cycle-specific input parameters, with a significant effect on the transient were compiled in the RSAC and compared to the parameters for the precalculated reference or bounding analysis. If the input parameters were not conservative relative to the reference analysis, a cycle-specific transient evaluation was required. The RSAC was initiated by the Transient Analysis group and the cycle-specific parameters were contributed by the responsible engineering groups. Zion Cycle 14 was a split-scope reload, and the licensee, CEC, provided the RSAC parameters for reload core design neutronics in letter Z1C14/016, "Neutronics Only SPIL Transmittal - Zion Unit 1 Cycle 14: Revised," dated October 7, 1993.

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[2]It was noteworthy that letter Z1C14/016, "Neutronics Only SPIL Transmittal - Zion Unit 1 Cycle 14: Revised," dated October 7, 1993, provided information concerning the CEC reload core redesign relative to the current Zion Unit 1 reference analysis limits; however, the Zion Cycle 14 T/H analysis was based on a comparison of the CEC reload core redesign to the original CNFD Zion Cycle 14 analysis.

As part of the reload procedures evaluation, the Safety Analysis Standards were reviewed with representatives of the Transient Analysis group. The RSAC listing of transient parameters used for the Zion Cycle 14 reload was found to be in agreement with the approved listing in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," dated July 1985.

To evaluate the safety analyses performed when the reference analysis was not bounding, the team reviewed the Zion Cycle 14 rod ejection accident (REA) analysis performed for the VANTAGE-5 upgrade. The REA analysis in Calc Note CN-TA-90-282 documented the methods and assumptions of this analysis. The calculations were performed with F-configured versions of the TWINKLE (core transient) code and the FACTRAN (fuel rod performance) code. The stand-alone calculations of the REA rod motion and doppler weighing factor were reviewed and found to be correct. The TWINKLE moderator temperature coefficient (MTC) input was determined by the ANC code, and the fuel rod heat flux and temperatures were determined by the PAD code. The team concluded that the Zion Cycle 14 REA analysis was carried out in a manner consistent with the procedures given in Safety Analysis Standard 14, "Rupture of a Control Rod Drive Mechanism."

(4) Fuel Rod Design Analysis

The Zion Cycle 14 fuel rod design was performed by the Fuel Analysis group. The analysis included an evaluation of centerline temperature, internal pressure, clad strain, oxidation, corrosion, clad flattening, swelling, and gap conductance. The P/PD&D group in CNFD COLA provided input to these analyses, including the DEVL and the Bill of Materials/Key Sheet (BOM/KS). The Fuel Analysis group provided the rod back-fill pressure to the CNFD COLA. The methods and criteria used in these analyses were documented in the Fuel Rod Design Procedure Manual. The analysis was performed with F-configured versions of the PAD code and various stand-alone versions of the PAD modules.

The Zion Cycle 14 fuel rod analysis documented in Calc Note T/H-93-067-0 and the three subsequent revisions were reviewed in detail. Revision 1 of T/H-93-067-0 used a cycle-specific fluence versus burnup correlation to determine rod growth; Revision 2 incorporated the final CEC split-scope reload core design; and Revision 3 incorporated the revised Region-14A fluences (resulting from the Hf rod inserts) in the rod growth calculation. In Revision 1 to T/H-93-067-0, the growth calculation required a special-purpose computer calculation, and the team confirmed that this calculation was verified by an independent hand-calculation. The fuel design analyses satisfied all the performance criteria up to the Zion Cycle 14 fuel burnup limit. These analyses were carried out, documented, and verified in a manner consistent with the methods and requirements of the Fuel Rod Design Procedural Manual.

3.3.1.6 Salem Nuclear Generating Station Unit 2 Cycle 9

A designated project engineer acted as the contractual and technical interface between the licensee, Public Service Electric & Gas Company (PSE&GC), and the CNFD for the Salem Nuclear Generating Station Unit 2 (Salem) Cycle 9. The team noted that the reload design process begins as an iterative process in which the licensee, CNFD, NTD, and the CNFD COLA participate. The RSLC plays a key role in the interactions of these groups. Throughout the reload core design process, the RSAC played a key role in ensuring that the Core Engineering group and the NTD NSA group together produced a reference safety analysis that is valid for the reload core being designed.

The team reviewed the interfaces and documentation of the reload analysis process by inspecting the key interfacing documents (i.e., the RSER, the RSLC, the Enrichment Requirement Letter, and the RSAC) and their transmittal letters to gauge the adequacy of the process. The team judged that for this reload the external and internal interfaces governing the reload core design and reload safety analysis process, the technical directions provided for the process, the responsibilities assumed by the individual engineers in the execution of their tasks, and the engineers responsiveness to licensee inputs and requirements were adequate.

Furthermore, the team noted that in the instance described below, the strength of the CNFD organization was demonstrated through the effective interactions between members of CNFD and NTD under conditions of considerable time constraint. After failed fuel was detected during ultrasound inspection of fuel to be reinserted into the Salem Cycle 9 reload core design, the Cycle 9 reload core was redesigned. As documented in RSAC, "Evaluation for Salem Unit 2 Cycle 9 Redesign," CDB-94-253, FA-94-294, dated November 11, 1994, the redesign included replacing all region 8 fuel assemblies with a history of baffle placement. The redesign also necessitated a re-evaluation of the LOCA analysis (Calc Note SEC-SAII-4570-C2, "RSAC-PNJ-Cycle 9 Reload Evaluation redesign," Revision 0, dated November 14, 1994). The redesign and the associated analyses were nonroutine activities. On the basis of its evaluation, the team determined that these activities were carried out in a thorough manner meeting existing procedural requirements. These activities also reflected effective interfacing between members of CNFD and NTD to successfully execute a difficult task under considerable time pressure.

(1) Asymmetric RCCA Withdrawal

On May 27, 1993, at Salem Unit 2, a single failure in the rod control system caused a single rod to withdraw 15 steps from the core while an insert signal was being applied. On June 17, 1993, PSE&GC requested an emergency license amendment involving the rod control system at Salem Units 1 and 2. The emergency license amendment request noted that a potential single failure could cause a single (or multiple asymmetric) RCCA withdrawal, and that explicit analyses determined the single RCCA withdrawal at power event to be bounded by a multiple RCCA withdrawal of two adjacent D-Bank RCCAs. On June 21, 1993, the NRC issued Generic Letter 93-04, "Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54." In response to this issue, W CNFD published WCAP-13803-A, "Generic Assessment of Rod Cluster Control Assembly Withdrawal," Revision 1 (original version dated August 1993; approved version dated November 1994).

Westinghouse CNFD had previously considered the occurrence of a multiple RCCA withdrawal event due to a single failure to be incredible. Consequently, the METCOM design manual did not analyze this event. The Calc Notes documenting the analysis of this event, therefore, identified the method used as a nonstandard method since it is not described in the design manual. The use of nonstandard methods was governed, in part, by CNFD procedure EP-302. Two sections of procedure EP-302 were relevant to the documentation of nonstandard calculations:

(a) Section 7.1.II.5: document assumptions, identified those assumptions that must be verified as design proceeds.

(b) Section 7.1.II.8: document deviation from standard methods (defined in a design manual) or the use of nonstandard methods (not defined in a design manual) in sufficient detail to support verification.

Calc Note PSK-93-001-0 (creation date June 5, 1993) documented the accident analysis for single and multiple rod withdrawal accidents. However, the team determined that Calc Note PSK-93-001-0 did not conform to Sections 7.1.II.5 and 7.1.II.8 of procedure EP-302 since the assumptions regarding bounding asymmetric rod configuration had not been documented and verified and since the use of nonstandard methods had not been documented in sufficient detail to support verification. Since the Calc Note provided the bases for the emergency license amendment request, conformance with procedure EP-302 was necessary to ensure that these calculations meet the appropriate CNFD technical and quality standards. The team concluded that CNFD failure to document the assumptions and the deviations from standard methods did not comply with the provisions of EP-302. As a result, a potential nonconformance was identified during this part of the inspection.

In response to the team determination that Calc Note PSK-93-001-0 did not conform to procedure EP-302, the following corrective actions were taken by Core Engineering while the team inspected CNFD/Westinghouse Energy Center:

(a) the coauthor modified a page of the Calc Note and added three pages to clarify the methodology used in analyzing the asymmetric rod withdrawal accident,

(b) a clarification of the methodology referred to WCAP-13803-A, Revision 1, which provided the analyses needed to identify the bounding asymmetric control rod configuration; and

(c) the additional material provided bore the signatures of the coauthor and the verifier.

As a result of the additional material being incorporated into Calc Note PSK-93-001-0 and the corrective actions taken by Core Engineering, the team determined that its concern regarding compliance with procedure EP-302 had been satisfied and that the potential nonconformance was closed.

(2) **LOCA and Non-LOCA Transient and Safety Analyses**

The team assessed transient, safety, and set point analysis processes by examining of the standards utilized in these analyses, by verifying that NRC-approved computer codes were used in these analyses in conformance with SER-specified limitations and restrictions, and by examining individual Calc Notes pertaining to transient and safety analyses.

CNFD currently utilized the LOFTRAN code for systems transient analyses, the FRACTRAN code for fuel rod heat transfer calculations, the THINC code for DNBR analyses, the TWINKLE code for fast transient analyses, and the OPTOAX code for set point analyses. The WFLASH and NOTRUMP codes were used for small-break LOCA analyses; and the LOCTA, SATAN, COCO, WREFLOOD, BART, and BASH suite of codes were used for large-break LOCA analyses.

In the reload safety evaluation process, the values of the key safety parameters for the reload core were determined by CNFD and transmitted to NTD via the RSAC. Westinghouse NTD determined whether all "reload values" of the key safety parameters were bounded by the current limits. If they were not, violations were resolved through redesign (interactively with CNFD) or through reanalysis of the affected transients. The team noted that owing to the conservatism inherent in the reference analyses, most reload cores have parameter values that are bounded by the current limits. Violations, when they did occur, were usually resolved through a relatively minor redesign of the core. Reanalysis of a set of transients was rarely necessary. The LOCA analyses for the reference core tend to be bounding for reload cores unless a new fuel design was introduced, in which case the LOCA analyses were redone. As a result, while CNFD routinely performs between 30 and 40 core design analyses a year, complete sets of transient and safety analyses were much rarer. This fact was reflected in the much smaller staffing in the transient and safety analysis area compared to core engineering.

Based on an examination of the appropriate standards, computer code documentation, topical reports on recent transient and safety analyses, and selected Calc Notes, and based on discussions with the engineers involved, the team determined that activities in the safety and transient analyses area were excellently performed.

3.3.1.7 South Texas Project Unit 2 Cycle 4

The evaluation of the CNFD design process for the Houston Lighting and Power Company (HL&P) South Texas Project Unit 2 (STP) Cycle 4 reload core design and reload safety analysis started with the examination of the contractual and schedular requirements. The overall process was outlined by the fuel licensing engineer. The project engineer correspondence file was examined to define the scope and length of the project. The minutes of the Design Initialization meeting on July 15, 1992, were reviewed for design constraints. The minutes of the Production Initialization meeting on October 30, 1992, were also reviewed for consistency with the interface documents (DEVL) and the original manufacturing schedule received from CNFD COLA. STP Cycle 3 was originally scheduled to shut down on February 28, 1993, with STP Cycle 4 to start up on April 25, 1993. Due to a lengthy outage, the actual STP Cycle 4 start up (Mode 2) was on April 24, 1994. The STP Cycle 4 reload subsequently underwent several redesigns from July 1992 to February 1994 to respond to changing licensee conditions (i.e., reduced energy requirement, fuel vibration problems, and rotated grid issue). A STP mid-Cycle 4 redesign effort was also reviewed, regarding a STP plant-specific evaluation which was submitted for NRC review in April 1993 and approved for incorporation during Cycle 4. Setpoint changes for the midcycle redesign were discussed in a meeting with HL&P on September 14, 1994, in which the RIQ was revised, and the RSE was scheduled for December 1994 to allow midcycle incorporation by April 7, 1995.

The detailed STP Cycle 4 inspection began with a review of the deliverables to the licensee: the final nuclear design report (NDR), RSE, and COLR documents. These were evaluated to determine the key reload design parameters, which were then followed backwards through the design process to the initiating contract and licensee requirement documents. The process and documentation for the review and approval of these documents were also examined.

(1) **LOCA and Non-LOCA Transient and Safety Analyses**

Section 3.2 of the RSE documents the Accident Evaluation analyses. It was reviewed through discussions with the responsible NTD Safeguards Engineering group and Transient Analysis group engineers. The analyses process and the Calc Notes for the STP Cycle 4 reload LOCA analysis were reviewed relative to the Safeguards Engineering Standards. The referenced standards were RSAC-01, "Overview of the Reload Process," Revision 3, dated June 13, 1994, and RSAC-02, "Reload Safety Analysis Checklist (RSAC) Parameters and their Review," Revision 4, dated March 18, 1994. RSAC-01, which defined the CNFD and customer interfaces with NTD, were examined for appropriate documentation. RSAC-02 defined the key LOCA-related parameters and listed the code models used.

The non-LOCA analyses were reviewed relative to the Safety Analysis Standards (SAS). The application of SAS-17, "RSAC Preparation and Evaluation," Revision 5, dated March 21, 1994, to specify the allowed reload key safety parameter ranges which define the current limits of RSAC was examined for consistency with WCAP-9272-P-A. Other standards referenced in the Calc Notes were examined.

(2) **Thermal-Hydraulic Analysis**

Section 2.3 of the RSE, "Thermal and Hydraulic Design," was evaluated by discussions with the responsible fuel analysis engineer. The STP Cycle 4 Calc Notes (T/H-92-179) were reviewed relative to the Thermal Hydraulic Design Procedure Manual. Previous Calc Notes (T/H-91-124 and T/H-92-084) were also referenced as unchanged for the analyses of record. The Design Initialization meeting minutes, including the RIQ, was the primary input interface document for the T/H design. Fuel Design Data List (FUDDL) memo CDC-93-014, dated January 15, 1993, documented the reload redesign. The fuel rod design effort involved review of the mechanical design against the RSAC limit list to confirm applicability for the DNB events or to determine re-analysis, as for the steamline break.

(3) Nuclear Design

Section 2.2 of the RSE, "Nuclear Design," was reviewed through discussions with the responsible CNFD Core Engineering Core Design engineer. STP Cycle 4 became partially a split-scope reload core design, with the licensee participating in the nuclear design to reduce the number of feed bundles required. The primary interface documents were reviewed to determine the control of the split responsibility and the flow of information to and from the licensee. Calc Notes supporting the key parameters for the NDR, the RSE and the COLR were reviewed. The Technical Specification changes for the reload were reviewed and representative interface documents were examined and found satisfactory. The eight volumes of Calc Notes were reviewed for consistency and completeness with respect to the significant redesign, the extended schedule, and personnel turnover. The final reload core design was reviewed in detail with the currently responsible Core Engineering Core Design engineer, including the Cycle 4 final design model summary/checklist from METCOM Table 1.7-9. The overall flow of information between CNFD and HL&P was reviewed with the project engineer.

(4) Fuel Mechanical Design

Section 2.1 of the RSE, "Mechanical Design," was evaluated by discussions with the fuel analysis engineer and by the review of the Calc Note relative to the Fuel Rod Design Procedure Manual, Revision 4, dated April 1993. STP used the VANTAGE-5 fuel assembly design referenced in WCAP-10444-P-A, with a 14-foot active fuel length. The 36 Cycle 4 Region 6 reload fuel assemblies specified in the final core design incorporated the following features differing from the previous Region 5 reload:

- low-pressure drop (LPD) Zircaloy mid-grids,
- IFBAs,
- modified top grid assembly, and
- modified top nozzle assembly.

The 36 Region 7 fuel assemblies comprising the remainder of the Cycle 4 reload also incorporated the following additional features:

- extended burnup bottom grid,
- fuel rod repositioning,
- keyless top nozzle assembly, and
- rotated alternate mixing vane LPD mid-grids.

As was observed for the Summer Cycle 9 reload, no single document existed which showed that all fuel rod design criteria, as specified in the Fuel Rod Design Procedure Manual, were satisfied for the core design.

(5) Technical Specifications

Section 4.0 of the RSE references the STP Technical Specification changes required for Cycle 4 operation. The Technical Specification changes involved the implementation of increased boron concentrations in the refueling water storage tank and the safety injection accumulator, and it was verified that these changes were accounted for in the reload design process.

(6) Core Operating Limits Report

Section 5.0 of the RSE references the STP COLR for Cycle 3 and the updated Cycle 4 COLR that was delivered to the licensee along with the RSE. It was verified that the values of MTC, control rod insertion limits, the peaking factors (F_Q and $F\Delta H$), and the allowable axial flux difference as listed corresponded to those determined in the reload design process.

3.3.2 Fuel Assembly Mechanical Design

The W fuel assembly mechanical design functions were performed by the Product/Process Development & Design (P/PD&D) group of CNFD, located at CNFD COLA. Although P/PD&D was located at CNFD COLA, the team evaluated P/PD&D inputs to and interface with the reload core design and reload safety analysis process during this portion of the inspection at CNFD/Westinghouse Energy Center. Therefore, in the interest of both overall readability and convenience, this portion of the inspection report described the team evaluation of the P/PD&D fuel assembly mechanical design functions that were performed at both the Westinghouse Energy Center and CNFD COLA and evaluated during both inspection periods, as described in Section 3.2 above.

Westinghouse CNFD provided a wide range of fuel designs, including the W 14x14, 15x15, 16x16, and 17x17 lattice fuel assembly arrays and fuel assembly designs for nuclear power plants designed by both ABB Combustion Engineering and Babcock & Wilcox. The primary fuel mechanical design responsibilities of P/PD&D included (a) new hardware product development and testing, (b) process development to support fabrication of new hardware product designs, (c) reload-by-reload specification of the hardware product design for manufacturing to meet licensee-specific design requirements, (d) design support for manufacturing, and (e) collection, evaluation, and dissemination of product performance data. Both the Fuel Performance Technology and the Product Performance groups of P/PD&D were located at the Westinghouse Energy Center, and the following P/PD&D groups were located at CNFD COLA: (a) Product Development & Testing, (b) Thermal-Hydraulic Testing Analysis, (c) Materials & Mechanical Process Development, (d) Design Specification & Drafting, and (e) Product Design.

3.3.2.1 Mechanical Design Process

The Fuel Projects Organization provided the Job Order and CATD document, which included licensee fuel and component design and operating data. In a typical reload core design, the P/PD&D Design Specification and Drafting group received design and fabrication data from Core Engineering and produced the BOM/KS, which compiled the fabrication specifications for the fuel assemblies and core components. The reload-specific fuel and component designs were selected by the Design Specification and Drafting group from the W Current Design List (CRDL), which was a compilation of available compatible fuel-related core components (e.g., pyrex glass burnable absorbers, WABAs, holddown assemblies, and thimble plugs) and design features (for Westinghouse VANTAGE-5, VANTAGE-5H, and VANTAGE+ fuel assemblies, these design features include axial and radial blankets, IFBAs, intermediate flow mixer (IFM) grids, low-pressure drop (LPD) zircaloy grids, removable top nozzles (RTNs), debris-filter bottom nozzles (DFBNs), certain assembly modifications, and ZIRLO™ fuel clad tubing).

The Core Engineering inputs to the BOM/KS included the (a) fuel and IFBA enrichment, (b) pellet stack length and IFBA pattern, (c) core loading plan, and (d) the fuel rod pressure. The P/PD&D Product Design group evaluated design deviations and special designs. The P/PD&D Product Development & Testing group evaluated major redesigns and new fuel designs and did special mechanical testing (e.g., grid compressibility, clad burst, and vibration tests). These design evaluations were led by a project manager; the size and composition of the design team were determined by the scope of the design. Recent examples of designs evaluated include the Maine Yankee Atomic Power Company, Maine Yankee Atomic Power Station (Maine Yankee) Cycle 15 reload core design (ABB Combustion Engineering designed plant), Hf flux suppression rod designs, and fuel assembly vibration compensatory designs. These design evaluations may require qualification of special processes (e.g., debris-resistant coating) and T/H testing. The P/PD&D Thermal-Hydraulic Testing group performed pressure drop tests and evaluated DNB test data. The multirod-array DNB tests were performed at Columbia University and the side-by-side tests were performed in Canada. The results of this process included the DEVL, prepared by CNFD P/PD&D and the FPC, as described in Section 3.3.1 of this report, which were provided to CNFD Core Engineering. The CNFD Core Engineering Fuel Licensing Integration group summarizes any mechanical design changes identified in the DEVL and FPC and gave that data to NTD.

To evaluate the CNFD fuel mechanical design process, the team selected the following reload design packages.

(1) Zion Station Unit 1 Cycle 14

The Zion Cycle 14 reload fuel consisted of 44 assemblies at 3.6 w/o and 32 assemblies at 3.4 w/o VANTAGE-5 fuel assemblies without IFM grids. The fuel rod analysis for the reload core design employed NRC-approved methods[3].

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[3]Weiner, R.A., et al., WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," dated August 1988; and Davidson, S.L., (Ed.), et al., WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," dated December 1985.

The mechanical design activities for the Zion Cycle 14 reload fuel were controlled by a detailed design specification contract process. The specific design tasks were initiated by Job Orders, and the schedule was determined by the Project and Design Milestone Schedule System (PDMS). The initial BOM/KS were based on the information provided in the CATD for Zion Unit 1, dated March 2, 1992, which was provided by the project engineer. The available and compatible VANTAGE-5 fuel assembly design features, part numbers, and build instructions were selected from the CRDL 15A, "Current Design List 15x15 Fuel Rod," Revision 9, and PELS100, "Assembly, Pellet, Stack, Fuel," Revision 14. A Production Initialization meeting (analogous to the Design Initialization meeting held by Core Engineering) was held prior to the release of the DEVL. The Zion Cycle 14 fuel rod pressure was provided in the "Generic Backfill Pressure Update," dated February 2, 1993.

The core loading pattern and burnable absorber requirements were provided by Core Engineering in "Zion 1 Cycle 14 Burnable Absorber Requirements and Candidate Loading Pattern," dated March 1, 1993. This transmittal did not include the axial blanket stack height, but this data was provided later in "CWBQ Axial Blanket Length," letter dated March 19, 1993. This omission was corrected in future reload core loading pattern transmittals, as part of the corrective actions for the Florida Power and Light Company, Turkey Point Units 3 and 4, WABA axial misslocation event described in Section 3.3.2.2 below. The shipping requirements for the WABA core components was given in the Component Loading Chart, dated June 24, 1993. The final BOM/KS, giving the Zion Cycle 14 part numbers for fabrication, were then issued for the fuel assemblies and core components. These documents were reviewed and found to be in conformance with procedure.

The changes to the fuel assembly design included a change to a controlled fuel rod gap and changes to the top nozzle engraving. The controlled fuel rod gap was described in Engineering Change Notice 26754, dated June 22, 1993, and the engraving was described in the Waiver Request T93-021-01, dated May 25, 1993. The treatment of these design changes was in conformance with procedures. The final Zion Cycle 14 reload safety evaluation was transmitted to the licensee (CEC) in a letter dated July 15, 1994.

(2) Maine Yankee Atomic Power Station Cycle 15

The Maine Yankee Cycle 15 reload core design and fuel assemblies were being provided by W CNFD. Since this is the first nuclear fuel designed and manufactured by W CNFD for Maine Yankee, this Cycle 15 reload core design and fuel assemblies required a series of reload-specific analyses. The required analyses were simplified by the fact that the licensee, Maine Yankee Atomic Power Company, performed the nuclear analysis, most of the T/H analysis, and the reload safety analysis.

A member of the Project Manager group acted as the project engineer for the Maine Yankee Cycle 15 reload core design, and the P/PD&D Product Development and Testing group performed the mechanical design. The cycle-specific fuel data requirements were included in "Maine Yankee Purchasers' Fuel Data Requirements for Licensing," dated November 29, 1994, and "Maine Yankee (MYCQ) Fuel Design Drawing List - Revision-01," dated March 31, 1994. The P/PD&D

Product Development and Testing group design activities included (a) component procurement, (a) vibration testing, (c) bulge testing of thimble tubes to the grid sleeves, (d) fabrication drag tests of the fuel rods through the grids, (e) top nozzle-to-fuel handling device, (f) grid-crushing tests, and (g) RTN joint testing.

The test evaluation reports[4] were evaluated by the team and found to be adequately documented and verified, in conformance with procedures. The Maine Yankee final design review, documented in PDT-94-098, "Maine Yankee Fuel Design Assembly Final Design Review Package," dated May 5, 1994, and the acceptance of the resolution of the four action items by the designated reviewers were also evaluated by the team and found to be in conformance with procedure.

3.3.2.2 Mechanical Design Issues

During the inspection of the CNFD fuel mechanical design process, the CNFD staff briefed the team on certain mechanical design issues and events that were of special interest to the team. Of these issues discussed, the team selected the following reload fuel mechanical design issues to evaluate the P/PD&D response to and evaluation of the issues.

(1) **Fuel Assembly Vibration**

In April 1993, fuel rod failures resulting from grid-to-rod fretting (GRF) were observed during the Salem Unit 2 Cycle 7 outage and the Duquesne Light Company, Beaver Valley Power Station Unit 1 (Beaver Valley) Cycle 9 refueling outage. The GRF occurred between the third and sixth grids (mid-grids) and in most cases was found in fuel assemblies that had spent at least one operating cycle adjacent to the reactor core baffle. Westinghouse CNFD had not experienced prior mid-grid failures; however, a foreign utility had similar failures, involving only a few rods, in fuel assemblies close to the baffle. Based on a detailed evaluation, the foreign utility concluded that the responsible mechanism was a self-induced assembly vibration having a sharp amplitude peak at the characteristic (close to rated) flow (W_v).

It was noteworthy that the Salem Unit 1 and Beaver Valley Unit 2, plants similiar to Salem Unit 2, had not observed these type of failures. The GRF occurred in the first 17x17 zircaloy grid with a specific diameter rod design and in the reactor core regions with lead fuel assemblies having the LPD zircaloy grid. Of the eight nuclear power plants with this design, only the Salem Unit 2 and Beaver Valley Unit 1 plants had observed the GRF.

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[4]PDT-94-067, "Maine Yankee Bottom Grid Bulge Joint Strength Test Report," dated March 18, 1994, and PDT-94-167, "C-14 RTN Joint Test Report and Strength Capability," dated July 11, 1994.

Westinghouse P/PD&D believed that the plants experiencing the GRF had rated flows that were close to the characteristic assembly flow, where W_v = 1700-1900 gallons per minute (gpm), at which flow, this vibration was excited. The vibration frequency had been measured at the CNFD test facilities to be ≈ **[Deleted pursuant to 10 CFR 2.790 - Document describes specific values]**. The team noted that, since the vibrating fuel assemblies were located close to the core boundary, this frequency might be confirmed by a frequency analysis of the measured excore detector response at Salem Unit 1 and Beaver Valley Unit 1. Westinghouse CNFD P/PD&D had performed extensive vibration tests and finite-element analyses and had concluded that rotation of alternate grids will eliminate the fuel assembly vibration. Westinghouse has tested all of its grid designs; only this grid design experiences a significant vibration at operating flows. The vibration amplitude was believed to be larger for larger fuel assembly-to-reactor baffle gaps.

Although rotating the grids appeared to eliminate the assembly vibration and the potential for GRF, tests performed at Columbia University indicated that the grid rotation also reduced the margin to DNB when the grid-span was 10 inches. However, only two W-designed plants currently have the 10-inch grid-span, and CNFD P/PD&D was evaluating these plants and the effects on DNB.

After evaluating the CNFD P/PD&D response to the fuel assembly vibration issue, the team determined that the analysis and testing of the vibration mechanism and potential fixes were excellent and demonstrated the broad capabilities of CNFD in responding to fuel-operational problems.

(2) Axial Mislocation of WABA Rods

As part of its evaluation of reload fuel mechanical design issues, the team evaluated the axial mislocation of the WABA rods at Turkey Point Units 3 and 4, as documented in Turkey Point LER-001, dated January 15, 1993. The Turkey Point Cycle 13 reload core design fuel assemblies included the debris-resistant fuel rod design. This design had a solid fuel rod end-cap that shifted the active fuel upward 1.368-inches. This design change required a corresponding axial shift in the WABA rods.

The Core Engineering group identified these changes and correctly incorporated them into its core neutronics and T/H design analyses. However, the Core Loading Pattern letter from Core Engineering to P/PD&D did not explicitly identify the axial shift in the WABA rods and in this sense was incomplete. The Production Initialization meeting CNFD COLA also did not identify the necessary design change in the WABA rods, and the subsequent Mechanical Design Review (89-02) did not correct this design error.

In response to this event, CNFD established a Corrective Action Committee to review the event and define appropriate corrective actions. The committee identified the root cause as a failure to treat the relationship of the fuel stack to the burnable absorber stack as a specific design criterion. The corrective actions involved reviewing and updating the various data transmittals, such as the Core Loading Pattern letter, the DEVL, and the Core Engineering METCOM. In addition, the Corrective Action Committee considered training and transmittal distribution lists.

Although the team concluded that the corrective actions would prevent the reoccurrence of this particular design error, the team was not reasonable assured that these corrective actions (a) adequately addressed the inability of the Product Initialization Meeting and the Mechanical Design Review to identify the WABA design error and (b) provided reasonable assurance that more general design changes in the fuel and core components would be identified in future reload core designs. The team considered this to be a weakness in the CNFD problem analysis and made the following observations:

(a) the Production Initialization meetings and Final Mechanical Design Reviews could benefit from Core Engineering (e.g., neutronics and thermal-hydraulics engineers) participation, and

(b) the Design Initialization meeting could benefit from P/PD&D participation in these meetings.

These team observations require no specific action by or written response from CNFD.

3.3.3 Error Free Performance Team

The team examined error reporting in depth. The inspection of training material documented in FA-94-131, "Error Reporting Seminar Package," Revision 1, dated September 16, 1994, showed that adequate training was provided in software error reporting and the error-handling process. The Engineering Non-conformance Log for 1994 was reviewed and from this log several recent software error reports were traced through the Reportable Technology Error, Non-conformance, and Request/Problem Report (R/PR) process. The examples evaluated were found to be in compliance with the procedures.

The notebook, "EFPT Root Cause Analysis and Support Information," Volume 1, was reviewed to determine the functioning of the Error Free Performance Team (EFPT). The documentation showed that the EFPT was established in May 1993, and discussions with cognizant engineers revealed that the EFPT had been fully functional for about 6 months. The team evaluation of an incident pertaining to Summer from the Incident Status List showed that the root-cause analysis and corrective action followup was adequately performed and documented. The history of the EFPT was too short to determine its effectiveness, but the process was found to be sound.

3.3.4 Training

The training and qualification of the engineering staff were assessed by the team during the course of this inspection through discussions with individual members of the staff. The adequacy of the training programs was assessed through discussions with the training course administrator and engineering managers and through examination of selected training course materials and training records of individual engineers.

The team determined that CNFD had a formalized, comprehensive training program. At the center of the CNFD training program was a 3-week course in Nuclear Design Technology and Methods. All new members of the engineering staff were required to take the course. In addition, technical seminars were offered to the entire staff when a new methodology or a new set of computer codes was introduced. The instructors were certified by a training program, and student evaluations provide feedback on their performance.

The team reviewed a sample of training records of training activities for approximately 10 engineers from both CNFD and NTD. The work assignments for the engineers reviewed were consistent with the described training. The CNFD external training was the same as the internal training. CNFD required a certain level of training before an engineer was qualified to be a sole author of a Calc Note. An additional level was required for an engineer to be a verifier.

The training program at NTD was less formalized and exhaustive than CNFD's. Newly hired staff members participated in an informal Mentor Program for a year and then qualify as analysts and reviewer/engineers on the basis of qualifications, experience, and on-the-job training. NTD maintains a Training Matrix which listed analysis topics or reactor events such as rod withdrawal accident. This matrix was checked off as the engineer completed training activities. The NTD offers periodic Training Seminars to its staff to ensure that staff members remain current in methods and techniques.

Based on the its assessment described above, the team determined that the training and qualifications of the CNFD and NTD staff were adequate for the activities in which they were engaged.

3.3.5 Fuel-Related Inspection Services

This portion of the CNFD inspection at the Westinghouse Energy Center also included an evaluation of the quality program and design activities for selected fuel-related inspection services, specifically the design of fuel-handling tools. These activities are performed by the Reactor Cavity Service Engineering group of the Nuclear Services Division (NSD) of the W ESBU.

The team evaluated NSD's design activities for a self-aligning 17x17 removable top nozzle (RTN) installation and removal tool. The team found that the initial design review, tool qualification testing, and final design review were extensive and precisely documented. The final design and functional specification for the RTN tool was also well documented. In addition to this multi-phase design review and tool qualification process implemented by NSD, the team found the operating procedure for the RTN tool to be controlled and well documented.

To train and qualify its fueling operations technicians, NSD used competency-based training to develop the skills and knowledge needed to accomplish a given fuel-related inspection activity. According to NSD, the end result of its competency-based Fueling Operations Training Program was that training was efficiently matched to the needs for a particular job, and job performance was directly improved by such training. To become a W-certified fueling operations technician, each W fueling operations trainee must successfully complete a three-phase program: Primary Classroom Instruction, Secondary Classroom Instruction, and Practical Applications. In addition, candidates for senior fueling operations technologist must demonstrate proficiency in all phases of fueling operations during a specified minimum number of full or modified full-scope fueling operation days at a nuclear plant site, actively perform fueling operations during a specified minimum number of fueling operations, and pass a review by a Qualification Board.

3.3.6 Conclusions

The team conducted a performance-based audit of CNFD to provide a basis for confidence that CNFD products will provide their intended safety functions. In order to reach that conclusion, the team evaluated the organization, staffing, training, and qualification of the engineering staffs of Core Engineering, P/PD&D, and safety and transient analyses. At the end of the audit, the team determined that the performance of the CNFD staff, processes, and products in each one of these areas was excellent. The team identified one instance where the strength of the W ESBU organization was demonstrated through the effective interactions between members of CNFD and NTD under conditions of considerable time pressure.

The team observed that a strong quality culture existed within CNFD Core Engineering and that good quality assurance (QA) practices were routinely performed. The team based this observation on the following items:

(a) Excellent engineering procedures were utilized in the various functional areas. The procedures were rich in methodology and rationale. The procedures also utilized good interface control documentation such as checklists.

(b) The METCOM contained core analysis methodology as well as procedures for model inputs. The METCOM was updated frequently and update training was provided and revisions were issued with recipient sign-offs on the updates.

(c) Procedures were revised frequently suggesting that changes to the process were being made when required. An illustration of this was found in the Software Engineering Methodology manual, which originated in 1989 and currently was in Revision 11, dated October 25, 1994.

(d) The technology transfer to licensees was controlled and well documented via the Software Engineering Methodology manual. Transmittal packages for the initial technology transfer to a licensee, a technology update, the most recent technology transfer, and the transfer of a new code were examined. All packages were found complete per the applicable QA requirements.

(e) Software error handling was traceable through the Reportable Technology Errors, Non-conformance reporting, and R/PR process as required by the error reporting procedures.

(f) The EFPT was recently initiated at CNFD. The Root Cause Analysis and Support Information documentation was reviewed and an inspection of a recent incident from the Incident Status List showed that the root cause analysis and corrective action followup was documented.

(g) The initial design review, fuel-related tool collocation testing, and final design review by NSD and the operating procedure for the RTN tool were extensive, controlled and well documented; and the competency-based Fueling Operations Training Program for W-certified fueling operations technicians was judged to be comprehensive in the subjects covered and appeared to ensure demonstrated proficiency by certified fueling operations technicians.

3.4 CNFD Specialty Metals Plant

The CNFD Specialty Metals Plant (SMP) in Blairsville, Pennsylvania, manufacturers zirconium (Zr) alloy (zircaloy) tubing for use in the nuclear power industry. Although production at the plant in Blairsville began in 1955 (e.g., nuclear fuel pellets through 1960, stainless steel turbine blades, and forged bar and strip products), the manufacturing of zircaloy tubing started in 1967, and the manufacturing of inconel steam generator tubing began in 1968. In 1985, the manufacturing of inconel was discontinued and the CNFD SMP was committed completely to zirconium-alloy-based nuclear-grade (a) tubing for fuel rod cladding, (b) tubing for discrete burnable absorber rod cladding, and (c) tubing for thimble tubes, instrumentation tubes, sleeves, spacers and connectors. According to CNFD SMP, it has produced over 70 types and sizes of zirconium alloy tubing. Zircaloy-2 for boiling-water reactors, Zr4 for PWRs, ZIRLO™ for longer operating cycles and higher burnups, guide thimbles, and burnable absorber tubes are typical products.

3.4.1 Product Assurance

The team conducted this inspection, in part, by interfacing with personnel performing specific tubing production operations and with the Product Assurance (PA) organization, described in Department Charter BA-700, "Product Assurance Charter," Revision 7, dated September 23, 1993. The PA organization consisted of the following groups: (a) PA Engineering, (b) PA Operations, (c) Customer Projects, and (d) Equipment Reliability. The PA Operations group was responsible for the finishing inspection activities for all shifts, and for the operations of Laboratories Physical Testing and Laboratories Gages. The PA Engineering group was responsible for document control activities, engineering, and UT instrumentation.

The team determined that PA personn were functioning as expected; activities were performed by trained people ac rding to approved written procedures. The team concluded that the PA organization works well and meets QA requirements. The team identified no current weaknesses or concerns.

3.4.2 Customer Requirements

Customer requirements were imposed on CNFD SMP through standard material and design specifications and the CNFD quality program invoked in customer or licensee purchase orders (POs). Each customer required the CNFD quality program to meet Appendix B to 10 CFR Part 50. CNFD SMP was responsible for tubing production from the delivery of the tube reduced extrusion (TREX) to CNFD SMP from CNFD WZ through the delivery of finished tubing to CNFD COLA. The Zr4 and ZIRLO™ TREXs received by CNFD SMP were produced by CNFD WZ in accordance with CNFD standard material and design specifications. CNFD SMP also produced tubing in accordance with CNFD standard material and design specifications. Custom specifications were not used.

On the basis of its evaluation, the team determined that the quality program implemented by CNFD SMP met the requirements of Appendix B to 10 CFR Part 50 and provided effective control over activities affecting quality.

3.4.3 Fuel Clad Tubing Production

CNFD SMP received Zr4 and ZIRLO™ TREXs from CNFD WZ. Fuel clad tubing was produced by reducing the outside diameter (OD) and the TREXs wall thickness through the basic reduction steps to form a final tube hollow. The final tube hollow was then reduced to meet the final dimensional requirements for the fuel clad tubing.

The team observed that CNFD SMP produced fuel clad tubing from TREXs by the basic reduction steps described below. These steps would normally be repeated until the required OD and wall thickness for the final tube hollow were achieved. All activities were performed in accordance with Follower Cards (travellers) and written procedures.

(a) The first tube hollows were produced by cold reducing TREXs through cold pilgering. This process accomplishes tube elongation and wall reduction by rolling TREXs back and forth between two grooved dies. During this process, the tubes were rotated and advanced in small increments over a stationary mandrel. Both the tube diameter and wall thickness were reduced by this process.

(b) The first tube hollows were cut to lengths and deburred.

(c) The first tube hollows were cleaned and pickled.

(d) The first tube hollows were vacuum annealed. The tube hollow annealing process heated the material to a specified temperature to achieve recrystallization and reduction of the stresses introduced by pilgering.

(e) The first tube hollows were cold-reduced to produce the final tube hollows by cold pilgering.

(f) The final tube hollows were cut to lengths and deburred.

(g) The final tube hollows were cleaned and pickled.

(h) The final tube hollows were vacuum annealed.

(i) Following the second recrystallization anneal, the third and final pilgering pass was performed. Contractile strain ratio (CSR) and hydride orientation were developed at this stage from the amount of reduction produced in diameter and wall thickness. Subsequent measurement of CSR and hydride orientation, described in Section 3.4.4 of this report, confirmed the effectiveness of this step.

(j) After pilger reduction, the tubing was cut to specified lengths by removing a specified amount from the trailing end and the remainder from the leading end. This step removed material that had not received cold work consistent with the rest of the tube. The weight of the tube was measured and recorded and the follow card signed by the operators.

(k) The tube was then cleaned. The weight was measured and recorded and the follow card signed by the operator.

(l) Thermal stress relief was then performed and the furnace number recorded on the follow card. This operation retained the metallurgical texture of the microstructure developed during the final pilgering pass while producing a more uniform stress level within the structure. The weight was measured and recorded and the follow card signed by the operators.

(m) Straightening was performed, the weight was measured and recorded, and the follow card signed by the operators.

(n) The Inside Diameter (ID) of the tube was grit-blasted, the weight was measured and recorded, the follow card signed by the operators.

(o) The tube was cut to length, and the ends were faced, deburred, and checked for squareness. The cutoff machine number was recorded, the weight measured and recorded, and the follow card signed by the operator.

(p) The OD of the tube was polished, and the polisher number was recorded. The number of pieces accepted or scrapped was recorded and the follow card signed by the operator.

(q) Final cleaning was performed by alkaline cleaning, rinsing, and drying. The conductivity of the final rinse was controlled to a specified maximum value to assure the purity of the final rinse. The number of pieces accepted or scrapped was recorded and the follow card signed by the operator.

(r) Alloy verification was performed to assure that the proper Zr alloy was being supplied to the customer. Each tube was identification marked to provide traceability throughout product life. The number of pieces accepted or scrapped was recorded and the follow card signed by the operator.

On the basis of its evaluation, the team determined that the metallurgical implications for the final tube hollow reduction process were the same for Zr4 and ZIRLO™; that is, CSR and hydride orientation were affected in a similar manner by the final pilgering parameters, and the stress equalization achieved was the same in both alloys during thermal stress relief. CNFD SMP produced no Zr2 tubing at this time. The team also determined that the manufacturing and inspection activities produced fuel clad tubing suitable for its application.

(1) Beta-Quench

CNFD SMP performed no beta quenching. The Zr4 and ZIRLO™ TREXs provided to CNFD SMP by CNFD WZ had been suitably beta-quenched by CNFD WZ. Zircaloy-2 products produced by CNFD WZ were delivered to tubing producers other than CNFD SMP.

(2) Nondestructive Examinations

The team reviewed nondestructive examinations (NDE) of fuel clad tubing. CNFD SMP performs UT for OD, ID, wall thickness, and flaw detection on 100% of the Zr4 and ZIRLO™ finished tubing produced for nuclear application. The team observed UT Level I qualified personnel setup and calibrate UT machine No. 18. UT inspection of Zr4 fuel clad tubing was performed by Level I qualified personnel. Sorting stations were used to segregate the material tested according to flaw and dimensional characteristics requiring further action. One station received tubing with acceptable characteristics. Unacceptable material was processed in accordance with procedure QC-318, "Dispositioning of Fuel and WABA Tubing After Ultrasonic Dimensional and Flaw Inspection," Revision 37, dated November 15, 1994.

(3) Final Inspection

Final inspection consisted of the following steps.

(a) The ID surface was examined by visually examining the tube ID against a lighted background.

(b) The finish on the ends was examined. The number of pieces accepted, reworked, and scrapped was recorded and the Follower Card signed by the operator.

(c) Length and end squareness were checked.

(d) Straightness and ID at the ends were checked. The number of pieces accepted, reworked, and scrapped was recorded and the Follower Card signed by the operator.

(e) The OD surface was visually examined and a mechanized OD surface examination was performed to check for surface roughness.

(f) Tube packaging was performed. The number of pieces accepted, reworked, and scrapped was recorded and the Follower Card signed by the operator.

(4) Handling, Storage, and Shipping

The team evaluated the packaging and shipping of tubing with respect to the protection of the metal surface condition during shipping. Full sheets of styrofoam contoured to match the geometry of the tubing were used to separate the full length of each layer of tubing within heavy wooden boxes lined with thick brown paper. The packaging appeared effective and had not resulted in any reported shipping damage.

3.4.4 Quality Services Lab

The requirements for laboratory physical testing were contained in specifications imposed on CNFD SMP by its customers, such as the CNFD COLA. The team reviewed specification NFD-31008, "Seamless Improved Zircaloy-4 Tubing," Revision 38, dated July 24, 1992, to determine the requirements for longitudinal tensile properties, CSR, corrosion resistance, and hydride orientation. Samples for testing were pulled from production.

(1) Tensile Testing

The team observed that tensile testing was performed. Review of Lot Certification Laboratory Test Reports for zircaloy tubing determined that room temperature tensile properties met the requirements of CNFD specification NFD-31008.

(2) Contractile Strain Ratio

A relationship has been demonstrated between the contractile strain ration (CSR) and the crystallographic orientation or texture of the grain structure in zircaloy. Texture affects zircaloy tubing in several ways. Texture influences yield strength, hydride orientation, iodine stress corrosion cracking, and thermal expansion. The relationships between texture and these properties are known, and how a material will respond can be predicted. By controlling variables in the tubing reduction process, the texture of the material can be controlled, thereby producing desired characteristics or avoiding undesirable characteristics. The CSR measurement provides a method for characterizing texture and thereby assuring that desired metallurgical properties have been produced.

The team observed CNFD SMP performing CSR testing. A tube sample was marked with circumferential lines over a specified gage length and various measurements were taken. The sample was then strained in the prescribed manner and remeasured. From these measurements, the radial and circumferential strains were calculated and the contractile strain was determined. By reference to a graph showing the relationship between CSR and a radial texture parameter (f_r), the fraction of grains exhibiting radial texture can be determined. Radial texture exists when the basal plane of the zirconium hexagonal crystal is normal, or perpendicular to the radius of the tube.

(3) Corrosion Testing

The team found that corrosion testing was performed. Review of Lot Certification Laboratory Test reports for zircaloy tubing determined that corrosion test results met the requirements of CNFD specification NFD-31008.

(4) Hydride Orientation

Hydride formation in the zirconium hexagonal crystal has been shown to prefer the basal plane. Therefore, a radial crystal texture produces a circumferential hydride platelet orientation. Highly circumferential hydride platelet orientation has been shown to be necessary to produce good ductility in tubing after corrosion on reactor service. Hydride platelets at angles smaller than 40° to the radial direction are classified as radial and others as circumferential. The ratio of radial platelets to total platelets is defined as the hydride fraction (f_n). Since f_n can be controlled through the tube reduction process, a limit can be set on its value. The limit may be the subject of negotiations between the buyer and the supplier.

CNFD SMP used procedure QS-503, "Determination of Hydride Orientation of Tubing," Revision 6, dated June 8, 1993, to make the hydride orientation determination. A tube sample was carefully cut to avoid introducing additional internal stress, flash pickled and hydrided in a controlled furnace for a specified time. It was then cooled and carefully prepared for metallographic examination; avoiding the introduction of additional internal stress. A photomicrograph of a selected area was prepared and f_n was determined. A determination was made regarding the acceptability of the product represented by the sample and the report was returned to manufacturing operations for appropriate material disposition.

3.4.5 Instrument Lab

The team observed that production control and test instruments such as furnace controllers and UT and inspection devices were identified with calibration instrument numbers traceable to the gage laboratory and were within the calibration due dates. Review of documentation and discussions with CNFD SMP staff who were responsible for the Instrument Lab, indicated that activities were being performed by qualified personnel in accordance with approved procedures. QC technicians calibrated the dimensional standards used to perform final inspection, as described in Section 3.4.3 of this report. This calibration was performed in accordance with Product Assurance Procedure PA-212, "Dimensional Standards For Zirconium Alloy Tubing," Revision 5, dated August 22, 1994.

3.4.6 Corrective Actions

The team evaluated the findings of CNFD SMP internal audit ESBU-94-08, issued June 8, 1994, and the implementation of corrective actions. The team considered the concerns cited in the audit to be an indication of a thorough audit. CNFD SMP internal audit ESBU-94-08 itemized four issues requiring corrective action, three of which dealt with customer product processed during the audit. CNFD SMP implementation of the corrective and preventive actions taken in response to the audit findings were observed to have been performed in a timely manner.

3.4.7 Conclusions

The team conducted a performance-based audit of CNFD SMP to provide a basis for confidence that CNFD products will provide their intended safety functions. In order to reach that conclusion, the team evaluated the organization, staffing, training, and qualification of the operators, technicians, and PA staff. At the end of the audit, the team determined that the performance of the CNFD SMP staff, processes, and products in each one of these areas was adequate. The team observed that a strong quality culture existed within CNFD SMP and that good QA practices were routinely performed. The team based this observation on the following items:

(a) The flexibility of the CNFD SMP organization. The organization empowered individuals to focus their expertise on the activities needed to produce a quality product according to approved written procedures.

(b) Because of the W integrated approach, the team could not tell management from labor by the dress or conversation.

(c) Each individual interviewed during this part of the inspection appeared to be well trained and knowledgeable of the related production operations and quality requirements.

On the basis of its evaluations during this portion of the inspection, the team did not identify any weaknesses in or concerns with the CNFD SMP organization or its activities that affect the quality of the Zr tubing manufactured for use in the nuclear power industry.

3.5 <u>CNFD Columbia Plant</u>

The CNFD Columbia Plant (COLA) in Columbia, South Carolina, was a fully integrated fuel fabrication facility performing (a) the conversion of uranium hexafluoride (UF_6) to uranium dioxide (UO_2); (b) fuel pelleting; (c) zirconium diboride integral fuel burnable absorber (IFBA) pelleting; (d) fuel rod fabrication; (e) fuel bundle assembly; and (f) the various verifications, tests, examinations, and special processes. The PWR fuel fabricated by CNFD COLA, which began operation in 1969, is used in reactors originally supplied by <u>W</u> and those supplied by other reactor vendors. CNFD COLA also supplies fuel-related core components, such as top and bottom fuel assembly nozzles, grids, burnable absorbers, and control rods to its customers about the world.

The <u>W</u> fuel assembly mechanical design function was performed by the Product/Process Development & Design (P/PD&D) group of CNFD, located at CNFD COLA, and was fully integrated with the process development and manufacturing activities of CNFD COLA. The team not only evaluated P/PD&D activities during this portion of the inspection, the team also evaluated P/PD&D inputs to and interface with the reload core design and reload safety analysis process during the CNFD/Westinghouse Energy Center portion of the inspection. Therefore, in the interest of both overall readability and convenience, Section 3.3.2 of this report describes the team's evaluation of the P/PD&D fuel assembly mechanical design functions that were performed at both the CNFD/Westinghouse Energy Center and CNFD COLA.

The inspection of CNFD COLA emphasized the manufacturing processes that relate to the fuel rod failure mechanisms (e.g., hydriding, fretting, pellet/cladding mechanical interaction (PCMI), overheating, cladding collapse, bursting, and mechanical fracturing).

3.5.1 Product Assurance

The team evaluated the Product Assurance (PA) organization's independent oversight and the adequacy of the verifications, tests, and examinations performed by PA and other CNFD COLA organizations. The PA Manager reported to the CNFD COLA Plant Manager at the same organizational level as the Manufacturing Manager. The team determined that the CNFD COLA organizational structure provided PA with sufficient independence from cost and schedule concerns to focus on product safety, as required by Section 1, "Organization," of the <u>W</u> QA topical report.

3.5.2 Customer Requirements

Customer purchase orders (POs) for fuel assemblies entered <u>W</u> through the Operating Plant Business Unit (OPBU) organization (vs ESBU), located at the Westinghouse Energy Center. Project Sales Managers and Project Engineers of the Domestic Sales and Customer Projects organization were assigned to interface with individual utilities.

Domestic Sales and Customer Projects initiated Region Orders and/or Job Orders. Region Orders are for an indefinite quantity of an item such as fuel assemblies. Region Orders and Job Orders are issued to CNFD COLA through the Materials, Planning and Services and the Product/Process Development and Design (Design Specification and Drafting) organizations.

The NRC inspection team reviewed several orders and determined that the proper regulatory requirements were followed in the POs to W and in CNFD procurement documents reviewed.

3.5.3 Procurement

In the manufacturing of nuclear fuel assemblies at CNFD COLA, the major components procured externally were tubing for various applications, strip for grid assemblies, bar for end plug fittings, and top and bottom nozzles. The materials (Zr4, ZIRLO™, stainless steel, and Inconel) from which these components were made were defined by materials specifications and drawings that were controlled by CNFD SMP and maintained at CNFD COLA.

The team selected the following fuel assembly component POs to evaluate the CNFD COLA procurement processes:

(a) POs FC-99408-MGM and FD-15704-MGM issued to CNFD SMP for Zr fuel clad tubing per material specifications NFD-31008, "Seamless Improved Zircaloy-4 Tubing," Revision 38, dated July 24, 1992, and NFD-31003, "Seamless Improved Zircaloy-4 Thimble Tubes," Revision 22, dated October 12, 1992. The tubing was supplied by CNFD SMP under the "one roof manufacturing" concept; that is, after being checked for shipping damage it was accepted at CNFD COLA and placed in production.

(b) POs 92777 and 92779-BBR and PO FC-92437 issued to the Vallorbs Jewel Company of Lancaster, Pennsylvania, for Zr top and bottom end plugs per specification NFD-31009, "Cold-Wound Helical Steel Springs," Revision 13, dated April 21, 1993. End plugs were purchased primarily from Vallorbs, where they were machined from bar material supplied by CNFD WZ. As occasion required, some end plugs were machined at CNFD COLA. In either case they were inspected at CNFD COLA in accordance with the requirements of Quality Control Instruction QCI-311202, "Fuel Rod End Plugs and End Pilot - Receiving Inspection," Revision 89, dated January 23, 1995.

(c) PO FD-99731-MGM issued to the Associated Spring Company of Corry, Pennsylvania, for stainless steel alloy fuel rod springs per NFD-31006, "Cold Finished Zircaloy-4 Bar," Revision 18, dated March 4, 1993.

(d) PO FD-12224-MLN issued to CNFD WZ for Zr4 strip for grid straps. Strip for grid strap fabrication was supplied by CNFD WZ, under the "one roof manufacturing" concept. After being checked for shipping damage it was accepted at CNFD COLA and placed in production.

(e) POs FD-16063-MGH and FD-16064-MGH issued to Ulbrich Stainless Steel of Wallingford, Connecticut, for Inconel strip material for grid straps per NFD-31002, "Nickel Alloy 718 Sheet, Strip, and Plate," Revision 19, dated July 22, 1994.

(f) Top and bottom nozzles were primarily purchased from L&S Machine Company (L&S) in Latrobe, Pennsylvania. Some nozzles were produced at CNFD COLA. L&S was a certified supplier of nozzles to the quality requirements of the American Society of Mechanical Engineers (ASME) standard NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications." When receiving nozzles from L&S, CNFD COLA checked them for shipping damage and the presence of the correct information on the paperwork and placed them in production. CNFD COLA performed surveillance on L&S, most recently during the period May 23-27, 1994.

For each of these procurements, the team determined that (a) the requirements of 10 CFR Part 21 and Form 102, "QA Requirements For Purchased Material, Items, and Services," Revision 5, dated August 31, 1993, were imposed per specific drawing and specification requirements; (b) the certified material test reports for each PO contained data showing that the materials met all requirements including those in the CNFD material specifications; and (c) all suppliers were listed on the ESBU Qualified Suppliers List, documented in QSA 95-0187, dated January 23, 1995.

3.5.4 Chemical and Ceramic Operations

The starting material for the chemical conversion process (ammonium diuranate (ADU), or wet conversion process) was uranium hexafluoride (UF_6), received in large cylinders containing up to 1505 kilograms (kg). Each cylinder was sampled and analyzed for enrichment verification before being processed through one of the production lines.

3.5.4.1 Chemical Conversion

The UF_6 cylinders were placed in a steam chest and heated to vaporize the UF_6, which was conveyed through pipes to the hydrolysis column, where, through a highly exothermic reaction, a solution of UO_2F_2 and HF was produced. The UO_2F_2 solution was pumped into the precipitation column, along with a NH_4OH solution. A recirculation pump continuously recycled the resulting slurry (containing particles of ammonium diuranate) to the top of the column. Another pump transferred a portion of the recycling stream to a dewatering centrifuge. A paste-like product from the dewatering centrifuge was pumped into a horizontal dryer. The product from the dryer is conveyed into a gas-heated rotating tube calciner, where H_2 and steam were introduced to the powder flow. The H_2 reduces the U^{+6} to U^{+4} and the steam aided in volatilizing the fluoride (F).

The resulting UO_2 powder was fed directly from the calciner into a hammermill. The hammermilled UO_2 was collected in plastic containers called polypacks in approximately 10-kg quantities. For process control, each polypack was sampled and a composite sample from polypacks were blended and analyzed to determine the stoichiometric oxygen/uranium ratio (O/U), F, BET surface area, Fisher Subsieve Sizer surface area, porosity, bulk density, and H_2O. The team concluded that each critical process parameter was identified and controlled for each process step. The team also concluded that the process and sampling steps in conversion from UF_6 to UO_2 were well defined and controlled and ensured expected-quality UO_2 powder.

3.5.4.2 Powder Blending

The polypacks were stored until analyses are completed. A "picklist" was generated, based on these analyses, for selecting the polypacks that will constitute an approximately 1500 kg UO_2 powder blend. The team verified that the selection process takes into account the various factors that determine pressing behavior and sinterability of the UO_2 pellets (e.g., bulk density, surface area, and O/U).

A blend was assembled by dumping the UO_2 from the selected polypacks into a bulk blender, which is then rotated for a specified number of revolutions at a desired speed, specified in revolutions per minute (RPM). Because enrichment blending was sometimes used to achieve a final specified enrichment, the team reviewed qualification report CD-FB-018-082, which established the adequacy of the blending process by blending natural and depleted materials and checking homogeneity by isotopic analysis. The team concluded that the specified blending process, specified in X-revolutions at Y-RPMs, was adequately supported by the test data documented in CD-FB-018-082.

Qualification report PE-EJS-83-015, which qualified the same blender for enrichment blending by blending depleted UO_2 into enriched virgin UO_2 powder, was also reviewed. The team found the specified processing conditions adequately supported by the test data documented in PE-EJS-83-015.

After blending, the powder was processed through a hammermill to break up agglomerates and tumbled again for X-revolutions at Y-RPMs. It is then sampled at four locations and each sample is analyzed for enrichment, %U, O/U, H_2O, carbon (C), nitrogen (N), F, spectrographic impurities, EBC, sodium (Na), Zr, bulk density, porosity, and surface area (by both the Fisher Subsieve Sizer and BET techniques). The team concluded that the blending and sampling steps were well defined and controlled and ensured expected-quality UO_2 powder.

3.5.4.3 Fuel Pelleting

The 1500 kg UO_2 blends were processed in 18 kg batches for pellet pressing. U_3O_8 powder is added to each batch for density and sinterability control. The mixture was roll compacted and granulated to make the powder suitable for pressing. Before pelletizing, a die lube was added to the polypack and blended by rolling the polypack. Pellets were formed in a rotary press at a specified rate of pellets per minute. The press operator manually checked pellet green density manually at intervals using pellet weight and length data described on a control chart. Sensors on the press tooling measured and controlled the length of each pellet pressed.

An automatic stacker placed the pellets into sintering boats. A follower card was generated for each boat. Pellets were sintered for a minimum specified time at a specified temperature measured at the middle of the high-heat zone of the sintering furnace. Sintering temperatures were recorded on a chart, and the operator manually recorded the temperature once per hour. The temperature was over-checked twice each shift with an optical instrument.

Following sintering, the pellets were processed as lots. Each lot consisted of four sintering boats. A specified number of pellets were taken at random from each lot and ground to the specified diameter and density was measured. Based on the density, the lot was accepted or rejected for grinding. Low-density pellets could be resintered. Accepted pellets were ground to the specified diameter and accumulated into trays for drying. Each tray of pellets was 100% visually inspected for acceptance using visual standards for chipping and cracking. The CNFD COLA inspector's bar-coded badge was read by the computer in order for the inspector to enter the inspection data.

A specified number of pellets were taken at equal intervals throughout the blend sample for H_2 analysis, which included hydrogen contained in absorbed H_2O. Hydrogen content of the pellets was one of the more critical properties of the fuel. Because the precision of the hydrogen analyzer was only ± 50% at the average H_2 ppm content, special instructions were issued in procedure QCI-910210, "UO_2 Pellet Hydrogen Sampling and Release," Revision 69, dated December 19, 1994, designed to ensure that the limit was not exceeded. Pellets were also pulled for a "high block test" per procedure QCI-910219, "UO_2 Pellet Hydrogen High Block Evaluation," Revision 11, dated November 28, 1994. For a pellet lot to be accepted the lot average must be less than or equal to a specified ppm H_2 at the 95% confidence level, and no individual test result can exceed a slightly larger specified ppm H_2. The team examined this issue in great detail and concluded that CNFD COLA's hydrogen control of its fuel pellets was a strength of its pelleting operations.

After the sample pellets were selected for H_2 analysis, a specified number of pellets was selected for SPIDER (System for Pellet Inspection Data Entry and Retrieval). Each of these pellets were measured and weighed for sintered density determination. Perpendicularity, dish depth, and surface roughness were also measured on a limited number of these pellets. After density checking, a specified number of these pellets were selected as archive pellets. Additionally, a specified number of pellets were selected for pellet chemistry measurements. Some of the pellets selected for pellet chemistry were submitted whole for F analysis. The remaining pellets were crushed for the remaining analyses (such as enrichment, %U, O/U, C, N, F, total metallic impurities, and equivalent boron concentration). The team determined that the CNFD COLA UO_2 pellet pressing, sintering, sampling, and analyses were well defined and controlled and ensured quality pellets with low H_2 content.

3.5.4.4 IFBA Pelleting

CNFD reported that, currently, approximately 25% of a reactor reload contained integral fuel burnable absorber (IFBA) rods. UO_2 pellets acceptable for fuel rod loading were brought into the IFBA area, which was segregated from the regular pellet line. Pellets were loaded into a screen cage holder for sputtering the zirconium diboride (ZrB_2) coating onto the cylindrical surfaces of the pellets. Zirconium diboride from the sputtering targets was transferred to the pellet surfaces in a vacuum chamber. Adherence of the coating to the pellets was tested and samples of the coated pellets were analyzed for boron and H_2 content. The pellets were visually examined for oxidation using visual comparison standards (oxidation changes the pellet color).

The team determined that the process methods used for coating the UO_2 pellet surfaces with ZrB_2, the IFBA pellet testing, and the IFBA pellet sampling methods were well defined and controlled to ensure quality IFBA pellets.

3.5.5 Fuel Rod Fabrication and Inspection

The team reviewed the flow of materials (e.g., fuel clad tubing, end plugs, plenum springs, and pellets) from receipt of purchased items to release for fuel rod fabrication.

3.5.5.1 Fuel Rod Fabrication

Fuel rod fabrication began by laser-marking the fuel clad tubing with a unique bar code entered into the Rod Accountability Monitoring System (RAMS) to maintain the identification and to control the processing of each fuel rod as it progressed through the fabrication steps. RAMS is a computer-controlled data processing system that uses the stored data and other information to control the status of each fuel rod through each step of the fuel rod fabrication process. RAMS also contained approved procedures for access by authorized production and quality control (QC) personnel to either enter data or perform activities in accordance with those procedures.

For the fabrication of fuel rods for the Northeast Nuclear Energy Company, Millstone Nuclear Power Station Unit 3 (Millstone 3) Cycle 10 reload, the team observed the top end of Zr fuel clad tubing laser marked with bar codes. Bottom end plugs were then inserted with an interference fit before a welding operator performed the girth welds to fusion-bond the end plug to the fuel clad tube using the gas tungsten-arc welding (GTAW) process. The qualified welding process was computer controlled and subjected to inprocess checks of critical parameters. Instrument calibration, weld current, weld time, post-flow time, helium flow rate, argon flow rate, rotation speed, and electrode-to-seam alignment were some of the parameters that were checked. Every bottom end plug girth weld was ultrasonically (UT) examined to detect porosity, underpenetration, and undercut. An automated visual inspection for weld continuity was made of each weld with a magnified optical image.

Inspected pellets were moved to the rod-loading line after the pellet identification information had been entered into the Item Control System (ICS) by an optical reader. The team observed a rod line operator loading natural and enriched UO_2 pellets into tubes according to procedures displayed on a RAMS monitor at the work station. Plenum length gauges were used by the operator to achieve the required pellet stack length and type per RAMS instructions. The pellet tray and fuel rod bar code were entered into the RAMS through an optical reader.

After springs were inserted, top end plugs were inserted automatically and a welding operator automatically performed the girth weld to fusion-bond the top end plug to the top of the fuel rod. The qualified automatic welding processes were computer controlled and given the same in-process checks and instrument calibration discussed above to weld the bottom end plugs. Every top end plug girth weld was inspected and one rod was pulled from each line at the beginning and the end of the day for the Analytical Services Laboratories tests. Each fuel rod was weighed and the data entered into RAMS.

3.5.5.2 IFBA Rod Fabrication

IFBA fuel rods were fabricated in a controlled area that was separated from the standard fuel rod fabrication. An IFBA fuel rod normally contained about 6 inches of natural UO_2 blanket pellets at the top and bottom; about **[Deleted pursuant to 10 CFR 2.790 - Document describes a specific value]** inches of uncoated enriched UO_2 pellets in the adjacent stack length at the top and the bottom; and enriched UO_2 pellets coated with zirconium diboride (ZrB_2) in the middle of the rod.

The team observed the fabrication of IFBA fuel rods for the Millstone 3 Cycle 10 reload. A dual robotic stacking system collated the coated pellets with uncoated and natural pellets for loading into fuel tubes. The team determined that the IFBA rod fabrication was well defined by procedures and adequately controlled to ensure quality IFBA fuel rods.

3.5.5.3 Rod Inspection

Rods were inspected by either radiographic examination (RT) or UT systems that examined bottom and top girth welds and top seal welds. All RT and UT systems were developed by CNFD COLA Level III's and the operations of those systems were monitored by a CNFD COLA Level II and a PA engineer.

(1) Ultrasonic Examination

The team observed the UT examination of the top end plug welds on fuel rods. The Level II operator verified the UT system calibration by processing standards through the system: (a) a radial drilled for undercut and porosity, (b) a V-notch for underpenetration, (c) an axial drill hole in the face of an end plug for underpenetration in the seal weld, and (d) a tungsten inclusion in a seal weld for detection by X-Ray fluorescence. System verification was performed after a specified number of fuel rods had been inspected.

(2) Radiographic Examination

RT (x-ray) examinations were performed on all fuel rods when the production load permitted. During periods with greater production loads, UT examination was performed instead of X-Ray. The RT acceptance standards included the maximum allowable underpenetration, OD and ID undercut, cracks, laps, seams, pipes, lack of fusion, porosity, and inclusions.

The team observed QC inspectors reading and interpreting radiographic film from the RT examination of fuel rods. The team determined that the RT of fuel rod girth and seal welds were performed according to documented procedures by qualified personnel. This area of the inspection was identified as a strength because of the expertise exercised in the development of the RT examination systems and the training of personnel that control the critical parameters associated with fuel rod weld quality.

(3) Dimensional and Visual

The team observed QC inspectors performing dimensional and visual inspection of fuel rods. This inspection compared visual examination at 5X magnification of all girth and seal welds to visual standards and 100% of fuel rod surfaces for cleanliness, scratches, pits, gouges, arc strikes, and discoloration. All fuel rods were sampled for length. IFBA fuel rods were verified for end plug identification and all fuel rods were verified for laser mark visibility and absence of scatter dwell point, which sometimes occurs during laser marking. This condition was seen as tiny bright spots scattered throughout the bar code.

(4) Helium Leak Testing

Helium leak testing was performed to determine the pressure boundary integrity of fuel and nonfuel rods. Fuel rods were pressurized with helium prior to weld closure and nonfuel rods were not. Two leak detection systems were in operation. The primary system was automated and all fuel rods were processed through it in groups of 25. Groups of 25 rods successfully passing the automated system were not helium leak-tested through the secondary system. The secondary system was used to identify individual leaking rods from groups of 25 that failed the automated system. This system was also used to leak-test the non-fuel rods.

To test their integrity, nonfuel rods were placed in a chamber in which they were pressurized in a helium atmosphere to drive helium into rod defects. They were removed from helium pressurization and placed in the helium leak detector and observed for helium leakage in the same manner as fuel rods. Several qualification reports had been written documenting variables examined to establish the processes and define normal operating parameters. Responsible personnel were qualified to Levels I, II, and III of American Society for Nondestructive Testing (ASNT) standard SNT-TC-1A, December 1988 Edition. Standard leaks were traceable to the National Institute of Standards and Technology (NIST) and were observed to be within date limit. The team observed that activities affecting quality were being performed in accordance with written instructions, as required, and determined that measures had been established to assure that leak detection processes were controlled and accomplished by qualified personnel.

(5) Rod Scanner

The rod scanners contained californium (Cf_{252}) neutron sources, which activated the U_{235} within a fuel rod, causing it to emit gamma radiation. Gamma radiation was detected and evaluated to determine critical parameters used to ensure fuel rod quality, e.g., nonconforming pellets (in diameter and/or enrichment), gaps between pellets, plenum spring presence and its length, and the pellet stack length. Qualification reports had been written documenting variables examined to establish the processes and define normal operating parameters. Standard defective fuel rods were periodically used to calibrate the scanning systems. The team observed that activities affecting quality were being performed in accordance with written instructions, as required, and that measures had been established to assure that rod scanning processes were controlled and accomplished by qualified personnel.

3.5.6 Fuel Bundle Components

The following fuel bundle components were evaluated during the inspection.

3.5.6.1 Top and Bottom Nozzles

Two types of top nozzles were manufactured; one using a precision casting welded to an adapter plate box nozzle and another made of pieces of bar, a top plate, and an adapter plate box nozzle welded together. In comparison, two types of bottom nozzles were manufactured much the same way, one also using a precision casting welded to an adapter plate box nozzle and another made of pieces of bar welded to four castings and an adapter plate box nozzle. Qualification reports for various welding processes and procedures described the parameters investigated to establish welding procedures used for fabricating various types of top and bottom nozzle assemblies.

The team observed that (a) activities affecting quality were being performed in accordance with written instructions, (b) methods had been established to maintain identification of materials, parts, and components and a computer record keeping system was used to track their status through each manufacturing and inspection operation, and (c) suitable tests and verifications had been satisfactorily completed for activities affecting quality.

3.5.6.2 Grid Straps

By observing fabrication and reviewing documents, the team confirmed that grid strap assemblies were constructed in accordance with approved drawings, procedures, and specifications using materials that met design requirements.

(1) Stamping

The team observed operations that automatically stamped Zr grid straps from a coil of feed material. The stamping operation produced both Inconel and Zr straps that meet the drawing requirements for many different parts representing all grid strap designs. During this portion of the inspection, the team observed part 6483E78H03, in-process grid straps, identified with a sheet of notebook paper that stated "Do not use any of these outer straps run on 3-2-95." The team was concerned that this material had been put on hold by a method and form that was not in the described QA system. CNFD COLA PA personnel subsequently revealed that Quality Control Deviation or Notifications (QCDNs) were being written by the QC inspector, who had told the operator to hold the material until the "with material" QCDN was available.

Completed QCDNs 14637 and 14674 subsequently released the material for further processing. The team expressed concern about procedures that did not provide the operator and the QC inspector with clear instructions for holding the material. The team saw as a strength, the QC inspector and the operator's actions in taking the initiative to hold the questionable material. The team determined that procedure QCI-000112, "Quality Control Deviation or Notification (QCDN)," Revision 27, dated July 16, 1993, was confusing and not adequately specific. The team identified this procedural ambiguity as a weakness in the QCDN system. The PA Manager agreed that the employees had taken the correct actions and that the QCDN hold system will be clarified. The team determined that the CNFD COLA actions taken during the inspection satisfied the team's concern.

(2) Vacuum Annealing

The team reviewed the annealing of Zr straps in the vacuum annealing furnace. The annealing produced the properties required after removing the stresses induced by the stamping operation. Annealing temperature is a critical parameter and was achieved through the use of an automatic control system. The team thought that this operation was a strength since it included an annual furnace profile to ensure that the temperatures of annealing loads are being achieved by control thermocouples.

(3) Assembly

The team observed the electronically controlled welding of Zr alloy grids by laser welding operations. The steps of fabricating an Inconel grid were issuing material, assembling straps, spot-welding corners, applying braze, brazing and annealing, recording furnace data, inspecting grids, age-hardening grids, recording data, bead-blasting grids, and inspecting the grids. Inconel grids were normally brazed with the braze paste being installed automatically by robot after the corners of the grid had been spot-welded. The team reviewed the brazing and solution annealing of Inconel grid assemblies. The brazing is done in furnaces that produce sound braze joints and create an initial metallurgical structure, which is then age-hardened to achieve the required mechanical properties. The annealing was done in the required vacuum for the required time-at-temperature, followed by rapid cooling.

Age hardening achieves the required mechanical properties with yield strengths greater than 155,000 pounds per square inch (lbf/in^2). Tensile specimens included with each furnace load were tested in the Metallurgical Laboratory to verify that the time at temperature age-hardening process achieved the required results. The team found this operation to be a strength at the COLA factory since temperatures and time control are the critical parameters in achieving braze quality and strength in the Inconel structural grid assemblies.

(4) Inspection

Zr and Inconel straps were inspected visually by QC inspectors using hand instrumentation, optical comparators, and bright lighting to ensure that the straps met drawing requirements. Straps and sleeves were assembled by hand and then installed in grid fixtures with one type of Inconel grid and all types of Zr alloy grids being laser-welded.

3.5.6.3 Skeleton Assembly

The purpose of the skeleton assembly was to provide the structure into which the fuel rods could be inserted and thimble tubes into which various nonfuel rods could be inserted. The major components comprising the skeleton assembly were the top and bottom nozzles, thimble tubes, and various grid assemblies.

The team observed (a) that activities affecting quality were being performed in accordance with written instructions, as required; (b) that methods had been established to maintain identification of materials, parts, and components and the computerized record-keeping system used to track their status through each manufacturing and inspection operation, appeared to be suitable to prevent the use of incorrect or defective materials, parts, and components, as required; and (c) that suitable tests and verifications had been satisfactorily completed for activities affecting quality, as required.

3.5.7 Bundle Assembly

The team observed the magazine loading operation used to construct bundle assemblies. Fuel rods, standard and IFBA, were loaded into a magazine before being loaded into a skeleton. The appropriate fuel rods were loaded through the required templates until loading was complete. After magazine loading, a bar code scanner was used to scan the rods in sequence. Each fuel rod had been given a unique identification that the computer status system recognized when the rod's bar code was scanned. The system would not accept a fuel-rod type registered in an unexpected location.

The magazine-loading pattern was determined by the operator's selection of the skeleton to be loaded. Only released skeletons could be selected. The loading pattern had been assigned to the skeleton before being released. The team observed the skeleton loading, installation of the bottom nozzle, torquing of the thimble screws, expansion of the thimble screws into the antirotation feature in the bottom nozzle, installation of the top nozzle, strain gage measurement of the force required to remove the top nozzle, installation of the locking tubes, and welding of the instrumentation tube plug.

The team observed bow-and-twist gaging of a final fuel assembly; two stations were available to perform this activity, each by a different method. The wash and double rinse of the fuel assembly was done with an alkaline cleaner and followed by two water rinses. Each fuel assembly was checked for drag using a standard control rod assembly suspended from a force gage, and a final visual inspection was performed. All of the operations and inspections were performed in the sequence specified by a Fuel Assembly Routing Card.

The team observed that activities affecting quality were being performed in accordance with written instructions, as required and that suitable tests and verifications had been satisfactorily completed for activities affecting quality, as required.

3.5.8 Fuel-Related Core Components

The W family of fuel-related core components includes glass burnable absorbers, wet annular burnable absorbers (WABAs), holddown assemblies, thimble plugs, and rod cluster control assemblies (RCCAs).

3.5.8.1 Burnable Absorbers and Thimble Plug Assemblies

Burnable absorbers absorb neutrons in the reactor core during the early stages of a new load of more highly enriched fuel. More highly enriched fuel is designed for longer refueling intervals. During the early portion of longer refueling intervals, the burnable absorbers are necessary to maintain a controllable neutron flux configuration in the reactor core. As the fuel is used, the burnable absorber dissipates, maintaining a relatively constant neutron flux pattern.

The burnable absorber and thimble plug assembly was capable of being assembled in various configurations, depending on the purpose. The major components of the various assemblies were the spring guide/burnable poison welded assembly, the hold-down spring, the hold-down bar, and the various rod types and thimble plugs. Rod types included glass burnable absorber rods and WABA Rods.

The team observed the fabrication of the spring guide/burnable poison assembly and the subsequent fabrication of the hold-down assembly, in which the spring and hold-down bar was added. These activities were performed in the sequence specified by a Hold Down Assembly Routing Card. Final inspection of this part was performed by PA personnel.

The team examined the material specification defining the hold-down bar casting, NFD-31016, "Stainless Steel Castings," Revision 13, dated December 6, 1993. Paragraph 3.4, "Heat Treatment," required, in part, that heat treatment parameters be reported per Section 4.5. Paragraph 4.5.1 required, in part, that the manufacturer furnish a certified test report showing the results of all tests, inspections, retests, and reinspection. A review of the certified test reports showed that the heat treatment parameters had not been reported, even though CNFD COLA had accepted product. However, the CNFD COLA reported that Engineering Change Notice 27066, Revision 0, Change Number YYFF-41003, originated January 15, 1995, partly deleted paragraph 3.4. The Engineering Change Notice was finalized March 6, 1995, and NFD-31016, Revision 14, was issued the same day. Although the previous condition of accepting product not in compliance with the specification was considered a weakness, the team noted that the CNFD COLA corrective action was initiated before the team observed the condition.

Assembly and inspection was performed according to the sequence specified on the Non-Fuel Bearing Routing Card. All manufacturing operations were performed by operating personnel according to written manufacturing procedures. All inspection activities were performed by PA personnel according to written quality instructions.

3.5.8.2 WABAs

WABA pellets were used as the burnable absorber. Assembly and inspection were performed according to the sequence specified on the Non-Fuel Bearing Routing Card. All manufacturing operations were performed by operating personnel according to written manufacturing procedures. All inspection activities were performed by PA personnel according to written quality control instructions.

The team observed that (a) activities affecting quality were being performed in accordance with written instructions, as required; and (b) methods had been established to maintain identification of materials, parts, and components and a computerized record-keeping system used to track them through each manufacturing and inspection operation, appeared suitable to prevent the use of incorrect or defective materials, parts, and components, as required; and (c) suitable tests and verifications had been satisfactorily completed for activities affecting quality, as required.

3.5.8.3 RCCAs

The rod cluster control assembly (RCCA) provides for the insertion of enough negative reactivity into the reactor core to shut down the reactor and maintain it in a safe-shutdown condition. The major components of the RCCA were the body, to which the control rod drive mechanism attaches; the vanes; and the fingers. The number of vanes and fingers depended on the type of fuel assembly to be built. An absorber rod assembly is attached to each finger.

The team observed the fabrication of the spider assembly. These activities were performed in the sequence specified by a Spider Assembly Routing Card. All manufacturing operations were done by operating personnel according to written manufacturing procedures. All inspection activities were performed by PA personnel according to written quality control instructions.

The team examined the process by which control rod assemblies were attached to the spider. Paragraph 3.6, "Chemical Composition," stated, in part, that the analysis may be made either chemically or spectrographically by a method of analysis approved by the purchaser. CNFD COLA did not document the agreement regarding the method of chemical analysis. CNFD COLA personnel immediately contacted the manufacturer of the alloy and requested copies of the analytical methods used to perform the chemical analyses. CNFD COLA reviewed and approved the methods and provided them to the team for review before the close of the inspection.

The team observed that (a) with one minor exception, activities affecting quality were being performed in accordance with written instructions, as required; (b) methods had been established to maintain identification of materials, parts, and components and a computerized record-keeping system used to track their status through each manufacturing and inspection operation appeared suitable to prevent the use of incorrect or defective materials, parts, and components, as required; and (c) suitable tests and verifications had been satisfactorily completed for activities affecting quality.

3.5.9 Calibration

The team reviewed the control and calibration of critical process instrumentation and measuring and test equipment (M&TE). The inspectors noted that CNFD COLA had a very large number of process plant instruments and M&TE in the calibration program. However, not all of these instruments measure or calibrate instruments for determining critical product parameters. The inspectors reviewed a large number of calibration records and found all the instruments to be calibrated within the scheduled frequencies. The inspectors also reviewed traceability of the standards to the National Institute of Standards and Technology (NIST) and found no problems. Given the large population of items in the calibration system, the adherence to calibration schedules was considered a strength.

However, the team observed that the as-found calibration data was not required to be recorded. Generally, no as-found data was recorded unless the instrument was a weighing scale or the calibration was performed by an outside contractor. CNFD COLA stated that a conscience decision was made not to include the as-found data. The team noted that as-found data is useful in determining whether process instrumentation or M&TE exhibit excessive drift and whether an instrument was actually within the calibration tolerances.

The team noted that, with the exception of the M&TE calibrated by the tool and gage group, no evaluations of out-of-tolerance M&TE were performed. For instance, 30-40% of the calibration measurements from 0.0000 to 300.0000 millimeters (mm) were out of tolerance for the coordinate measurement machine (VIEW 1220) which evaluated the critical measurements of all the top and bottom nozzles. The absence of a written evaluation was significant because of the high number of actual measurements that deviated from the nominal, the importance of precise critical nozzle dimensions, and the nozzle drawing dimensions, which are in 10 thousandths of an inch, not in mm. The team reviewed several top and bottom nozzle drawings and determined that the VIEW 1220 out-of-tolerance measurements did not exceed the tolerances set by the nozzle drawings. As a result of this concern, CNFD COLA placed the VIEW 1220 coordinate measurement machine under a procedure requiring evaluations for out-of-tolerance as-found calibration results. The CNFD COLA actions taken during the inspection satisfied the team's concern.

3.5.10 Analytical Services Laboratories

The Analytical Services Laboratories (Metallurgical and Chemical) provided process control and product specification analysis and examinations required for the analysis and certification of fuel and fuel-related items.

3.5.10.1 Metallurgical Laboratory

The CNFD COLA Metallurgical Laboratory processes production samples used to verify that no problems have developed in the rod-welding process and the age hardening of Inconel grids. Weldments from each rod line were sampled daily at the beginning and end of each shift. Each age-hardening run of Inconel grids contained tensile specimens, which were pulled to verify that the heat treatment had achieved the minimum yield strength requirements.

The team determined by microscopic examination that girth and seal weld penetration samples met the requirements specified. Rod line weld samples were corrosion-tested in an autoclave according to procedure QCI-108857, "Autoclave Operating Procedure For Aqueous Corrosion Testing At [specified °C and specified pounds per square inch, gauge (psig) pressure]," Revision 22, dated August 3, 1994. The team determined by visual examination that samples 0532-9-1, 0532-9-2, 0536-3, 0531-7, 0533-8, 0534-2, 0535-1, and 0538 were acceptable by visual standards and in accordance with procedure QCI-108819, "Corrosion Evaluation and Disposition Practices," Revision 46, dated August 12, 1994. The team noted that sample 0537 was not with the other samples. The review of Form 993, per QCI-108857, determined that sample 0537 had been dispositioned acceptable by the a lab technician.

The team was advised by CNFD COLA that the senior lab engineer was holding sample 0537 for engineering review and promptly advised the lab technician that the disposition on Form 993 should have been identified as "Engineering Review." Sample 0537 had passed the corrosion test but was being evaluated for a circumferential gouge which might have been caused during rod manufacturing. QCDN 15527 documented that the gouge was caused when the sample was cut and was not related to the rod manufacturing process. The senior engineer emphasized that Form 993 was not used to release material and that Form 259 was used for that purpose. The team noted that Form 259 in QCI-108819 had not been completed for sample 0537. The senior engineer held a training meeting with all lab technicians on March 6, 1995, to stress the importance of accurately entering all documentation and following current procedures.

Prompt corrective action by CNFD COLA personnel was found to be a strength by the team, and the need for clarification in QCI-108819 relative to QCI-108857 was found to be a weakness. The team observed that the technical competence demonstrated by lab personnel could be supported better by clarifying the documentation requirements. The CNFD COLA actions taken during the inspection satisfied the team's concerns.

3.5.10.2 Chemical Laboratory

The team observed operations in the chemical laboratory, including testing for hydrogen content in the UO_2 pellets. The team identified concerns regarding the calibration of the hydrogen analyzers, which measure the amount of hydrogen in the UO_2 pellets. The team had concerns with the apparently large calibration tolerances set by procedure, the methods used to check calibration, and the implementation of the calibration procedure.

In reviewing calibration records, the team noted that the yearly one-point calibration check of two hydrogen analyzers at 560 amperes found both to exceed the calibration tolerances. One of the hydrogen analyzers was reading low; however, the other read 660 amperes when checked with a current source of 560 amperes. The team was informed that procedures do not require written evaluations for instruments of this type that exceeded calibration tolerances. The team questioned whether the out-of-tolerance analyzer could significantly affect its accuracy and, therefore, erroneously allow acceptance of the UO_2 pellets with hydrogen content above the CNFD COLA action limit specified in ppm. CNFD COLA stated that the hydrogen analyzers were checked daily with a known standard and that, since all calibration checks were within the tolerance band of ±1.6 micrograms, no further action was necessary. According to CNFD COLA, the 560 ampere-hydrogen analyzer reading should correspond to a temperature of 1700 °C. The temperature is an important parameter since at 1700 °C nearly all the hydrogen in the UO_2 pellets comes out of solution and can, therefore, be measured. However, since the hydrogen analyzer was reading high, the temperature was actually less, by about 100 amperes, corresponding to a lower temperature. A lower amperage reading, corresponding to a lower temperature, would make it appear that the UO_2 pellet hydrogen content was lower than it actually was because less hydrogen would be driven from solution at the lower temperature.

The team also had concerns with the hydrogen analyzer calibration tolerances set by procedure I-03, "Determination of Hydrogen in Uranium Oxides, Ceramics, and Metals," Revision 16. The hydrogen is extracted from the sample by heating in an argon atmosphere, and the evolved gases are separated chemically and the amount of hydrogen is read out directly in micrograms. However, during daily calibration checks with a known NIST titanium standard, the procedure allowed a ±36% tolerance or ±1.6 micrograms from a standard of 4.47 micrograms. The accuracy stated in the hydrogen analyzer vendor manual is ±0.10 ppm or 3%, whichever is greater. A review of past calibration records indicated that some of the hydrogen analyzers had not detected the accuracies as stated in the vendor manual or in the titanium standard. The certificate of analysis for the titanium standards stated the amount of hydrogen in ppm as 43 (±3). Although the measurement of hydrogen is in micrograms, the tolerance set by the CNFD COLA procedure would correspond to ±15 ppm. The team noted that during calibration checks, the detected amount of hydrogen is sometimes off by 1.3 micrograms from the standard, or about 13 ppm. Discussions with the hydrogen analyzer vendor indicated that the amount of hydrogen detected should at least be within the ±3 ppm tolerance stated in the titanium certificate of analysis. CNFD stated that the ±1.6 microgram tolerance was acceptable; however, there was no evaluation to on which to base the tolerance.

The team was also concerned with the method used for calibrating the hydrogen analyzer with a blank sample. Each month the hydrogen analyzers are calibrated with a blank tin flux sample that should have no detectable hydrogen. During calibration, however, the amount of hydrogen detected in the tin flux seemed unusually high, in the range of 30-45% of the hydrogen detected in a titanium sample. In addition, the calibration procedure in Section 6.0, "Instrument Calibration," of procedure I-03 required that, in checking the tin flux sample, the digital voltmeter reading should register

0.00 (±0.05) (no scale specified). This reading is expected since there should be no hydrogen in the sample. The procedure further required that steps in Section 6.0 be repeated several times to achieve a 0.00 (±0.05) value. The team noted that these steps were not documented. The procedure further stated that if the digital voltmeter value does not read 0.00 (±0.05), it would be necessary to adjust the blank potentiometer. The inspectors noted that on numerous monthly checks, the blank digital voltmeter readings were above the value of 0.05 and that there was no documented adjustment of the blank potentiometer controls, no written evaluation of the acceptability of this practice, and no statement explaining why the blank sample should be reading so high. The team found that lab technicians added the digital voltmeter reading of the standard sample to the blank sample, although this practice is not mentioned or included in the procedure.

Moreover, the procedure and calculations required that all test samples and test standards be tested on the 7.0 weight compensator gram scale. The mass of the uranium dioxide pellets is generally five grams. The inspectors determined that the weight compensator for three of the hydrogen analyzers were set not at 7.0 grams but at 2.0 grams. The inspectors reviewed the schematic diagram and front view diagram for the model RH-1E hydrogen analyzers and determined that the dial setting for the weight compensator corresponded to 2.0 grams. The CNFD COLA chemical lab personnel stated that the 2.0 gram setting was equivalent to the 7.0 gram setting and that internal circuitry had been modified to achieve this equivalency. The CNFD COLA personnel could not provide documentation of the internal circuity change or whether the dial setting was equivalent to the required 7.0 gram setting.

The team considered the methods used to calibrate the hydrogen analyzers a weakness. The large difference in accuracy between the hydrogen analyzer manufacturer's manual and the CNFD COLA chemical laboratory was significant. In addition, documentation of several steps in the calibration procedure were not clear. Because the team's concerns regarding these calibration practices were not resolved during the course of the inspection, these concerns are unresolved. However, because of the detected hydrogen content in the UO_2 samples (ranging upward from 0.20-0.30 ppm), the apparent inaccuracy in the hydrogen analyzer calibration is not considered a safety concern because; a 30% error would not exceed the CNFD COLA acceptance criteria lower limit of hydrogen ppm for UO_2 pellets.

The team identified CNFD COLA evaluation of the weaknesses identified in the hydrogen analyzer calibration practices as an open item and requested that NRC be notified when CNFD COLA has completed its analysis of the calibration practices. (Open Item 95-01-01)

3.5.11 Conclusions

The team conducted a performance-based audit of CNFD COLA to provide a basis for confidence that CNFD products will perform their intended safety functions. To reach that conclusion, the team evaluated the organization, staffing, training, and qualification of the operators, technicians, and PA staff. At the end of the audit, the team determined that the performance of the CNFD COLA staff, processes, and products in each one of these areas was adequate. The team observed that a strong quality culture existed within CNFD COLA and that good QA practices were routinely performed. The team based this observation on the following:

(a) The training of QC inspectors was very thorough. QC inspectors are retested on a regular basis to maintain their certifications.

(b) All instruments and M&TE used to measure directly or indirectly a critical parameter were calibrated at appropriate intervals.

(c) The heat treating of Inconel 718 and the annealing of Zr4 is controlled automatically by electronic systems that ensure the required properties of critical parameters with low possibility of human error.

(d) 100% RT and UT of fuel rod girth and seal welds were performed automatically by systems that minimize human error and provide exceptional control of weld quality. The Level III and Level IIs exhibited expertise in NDE in their use of these systems for controlling fuel rod weld quality.

(e) The RAMS system for material control tracks individual fuel rods using bar codes for identification and status.

(f) The system of evaluating conditions adverse to quality and corrective actions by the Corrective Action Committee was excellently implemented.

3.6 CNFD Western Zirconium Plant

The CNFD Western Zirconium Plant (WZ) in Ogden, Utah, established in 1978, transforms zircon sand into zirconium (Zr), hafnium, and zircaloy. Zircaloy was fabricated into tubular extrusions, plate, strip, sheet, and bar product forms. CNFD WZ produces Zr2 for use in boiling-water reactors (BWRs); Zr4 for use in PWRs; and ZIRLO™, a W-developed advanced zircaloy that contains niobium for additional corrosion resistance at high temperature.

3.6.1 Product Assurance

The team conducted this inspection, in part, through interfaces with personnel performing specific Zr alloy fabrication operations and the PA organization. The PA organization consists of the following groups: (a) Quality Engineering, (b) Quality Inspection, (c) Laboratory Services, (d) Audit Services, and (e) Records Control.

The team determined that PA personnel were functioning as expected; activities were performed by trained people according to approved written procedures. The team concluded that the PA organization works well and meets QA requirements.

3.6.2 Customer Requirements

The team evaluated the procurement interfaces between CNFD WZ and its sister organizations, CNFD SMP and CNFD COLA, to determine how customer requirements were passed from CNFD SMP and CNFD COLA to CNFD WZ. Customer requirements for various Zr alloys and product forms were expressed in terms of standard CNFD material and product specifications. The team reviewed several procurement documentation packages and determined that the proper regulatory requirements had been invoked. The team also determined that adequate measures were established to assure that applicable regulatory and design basis requirements were suitably included or referenced in the procurement documents.

3.6.3 Zircaloy Fabrication

The team evaluated the CNFD WZ zircaloy fabrication processes performed from the vacuum arc melting of electrodes through the production of finished tube reduced extrusion (TREX) for fuel clad tubing, bar products for fuel rod end plugs, and plate, sheet, and strip products for grid spacers and other fuel-related products.

For its evaluation of the Zr alloy fabrication processes described below, the team reviewed the applicable process procedures, work instructions, and fabrication travelers, which followed process outline descriptions and specified production requirements. On the basis of its evaluation of the Zr alloy fabrication processes and in addition to the specific strengths described below, the team made the following determinations:

(a) Zr alloy fabrication operations affecting quality were well documented in instructions, procedures, and drawings appropriate to the production of Zr alloy products; that the Zr alloy fabrication operations were accomplished in accordance with those instructions, procedures, and drawings; and that those instructions, procedures, and drawings included appropriate acceptance criteria for determining that operations important to quality had been performed satisfactorily.

(b) Measures had been established to assure that induction heating/beta quenching processes were controlled and accomplished by qualified personnel using qualified procedures within the limits established by documented qualification projects.

(c) A program for verifying and inspecting activities affecting quality had been established and was being executed to verify conformance with documented instructions, procedures, and drawings.

3.6.3.1 Zirconium Sponge and Recycle Material

CNFD WZ produces pure Zr sponge through the chemical reduction and extraction of zircon sand, the raw material feed to the Zr alloy fabrication process. The sponge was crushed to facilitate inspection and compaction into briquettes for the construction of production melting electrodes. The team observed removing unacceptable material from crushed sponge. The acceptable sponge was loaded into drums through a process that ensured homogeneity. PA inspectors inspected a sample of the sponge material to be pressed into briquettes and joined together by the electron beam welding (EBW) process to produce a melting electrode. This electrode was vacuum arc melted to produce an evaluation ingot, from which conical samples were milled and chemical analysis was performed. On the basis of this analysis, the sponge lot of drums were released.

Recycle Zr alloy material (forge shears, cut off ends, edge trim, and log butts) that occurred throughout the Zr alloy fabrication process was cleaned and compressed into briquettes and subsequently assembled, through EBW, into a recycle melting electrode. The recycle electrode was vacuum arc melted to produce a recycle ingot, from which conical samples were milled and chemical analysis was performed. The released recycle ingot was forged into a slab, called a spar, used with briquettes of Zr sponge to assembly production melting electrodes.

3.6.3.2 Vacuum Arc Melting

The melting electrode material used to produce a conditioned ingot was constructed by blending the drums of released sponge with calculated and weighed amounts of alloying elements to produce the feed to the compaction press. The team reviewed the technician's use of a computer program to calculate the quantity of alloying elements (tin, iron chromium, nickel, and silicon) to be added to Zr sponge and a recycled spar of previously analyzed chemical composition. The team identified the work of this technician as a strength because the calculations and weighing instructions resulted in the chemical composition of CNFD WZ products to meet the requirements of customer specifications.

Because an off-analysis ingot had been produced when a melting operator inadvertently did not add iron to the compaction press feed material, CNFD WZ initiated a requirement for a second operator to overview the weighing of alloy material by the first operator. The team verified that this overview procedure was followed during the weighing and insertion of alloying elements in the blender and that both operators signed the alloy weighing records for ingot UO3452P.

To ensure that impurities such as tungsten, cobalt, and iron were not trapped in recycled material, spars were RT examined. The feed material was compacted into semicircular briquettes and assembled on either side of a spar and then welded (EBW) to construct the melting electrode for the first melt. The second melt consisted of 2 ingots from the first melt, and the third melt was a final vacuum arc melt of the one ingot from the second melt. The conditioned ingot, produced by the third vacuum arc melt, was conditioned by machining the OD to eliminate visual porosity and UT examined as required by contract.

Chemical analysis samples were taken at specified locations along the ingot length. The results were reported individually for each location; no averaging was performed. Chemical analysis results reported out of limit at any of the five or six sample locations rejected the ingot. Machining the OD or cropping the end in the vicinity of the out of limit sample was performed to eliminate surface effects and additional analysis was performed. This action usually brought the chemical composition within acceptable limits. Ingots identified as acceptable were assigned to customer requirements.

3.6.3.3 Forging

Cast conditioned ingots were forged into logs or slabs for further processing to final products. Ingots for TREXs, as well as bar and wire, were forged to a specified OD and cut into billets of a specified length for extrusion and pilgering or swaging and drawing. Ingots for plate, sheet and strip products was forged into a slabs for rolling.

3.6.3.4 Extrusion and Pilgering

After cutting to length, billets were beta quenched using an induction heating furnace. Billet temperature was monitored using a two-color infrared pyrometer and thermocouples contacting each end of the billet. Proper beta quenching is essential to satisfactory corrosion resistance of fuel clad tubing. To assure proper beta quenching, qualification projects were performed for each combination of induction heating furnace and billet size heated in the furnace. Qualification projects established the working relationships among material properties, processing parameters, and equipment variables. A completed qualification project formed the basis upon which operating variables were selected.

Research[5,6] referenced by CNFD WZ staff established the concept of A-time. A-time is a measure of the accumulated time at elevated temperatures during alpha annealing and stress relieving. Beta quenching sets the A-time clock to zero for subsequent heat treatment time. Different A-time ranges have demonstrated optimum corrosion resistance for Zr2, Zr4 and ZIRLO™. CNFD WZ personnel stated that they used the A-time methodology to govern heat treatment operations.

After beta quenching, a corrosion resistance sample was taken for laboratory testing. Fabrication operations on the billets in a lot continued, but the lot would not be released from final inspection without passing the corrosion resistance test. The billets were rough machined and a centerline hole bored

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

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

in preparation for extruding. The billets were extruded to a specified OD and ID. Surfaces of the extrusions were conditioned in preparation for pilgering and one pilgering pass was performed to produce the TREX.

3.6.3.5 Rolling

Forged slabs were hot rolled to plate product, annealed, and UT inspected. For sheet and strip material, reductions were obtained by cold rolling. After the final cold rolling process, sheet supplied as coil product was slit into strip and final vacuum annealed.

3.6.3.6 Swaging and Drawing

A conditioned ingot was forged into a log using similar procedures to those observed for forged slabs. The logs were cut into billets and beta quenched. Bar products were manufactured by successive swaging reductions followed by either a salt bath anneal or a vacuum anneal.

3.6.4 Calibration

The team evaluated the CNFD WZ calibration laboratories for mechanical, instrumentation, and control devices. CNFD WZ had adequately specified actions to be taken regarding product accepted by calibrated equipment subsequently determined to be out of limit. Calibration records for mechanical, instrumentation, and control devices were maintained in a computer based system that tracked current calibration status and provided advanced notification of required calibration activity.

With one exception, all measuring equipment examined was marked with current calibration stickers. The team observed three stage micrometers in the metallography laboratory that had not been currently calibrated. This situation occurred as the result of a failure to include stage micrometers in the standard calibration system. As a result of the team observation, CNFD WZ proposed to:

(a) Revise the laboratory work instructions to verify that stage micrometers have been calibrated and are in good working condition. Add instructions on the required handling of stage micrometers to prevent damage and deterioration.

(b) Calibrate the existing stage micrometers by a method providing traceability to NIST standards and maintain calibration certifications in the calibration laboratory.

(c) Modify the purchasing program to require all stage micrometers purchased to be delivered to the calibration laboratory prior to being distributed to the end user. Revise calibration instructions to assure that new stage micrometers are calibrated by a method providing traceability to NIST standards.

(d) Assure that the calibration service providing NIST traceability is an ESBU qualified supplier.

The CNFD ac.ions taken during the inspection satisfied the team's concerns.

3.6.5 Laboratory Services

The team reviewed the metallographic examination of 20 samples of extruded Zr4 bar from ingot 403431P. This visual examination traversed mounted specimens at 50x magnification to verify that the tail ends of the first extrusions were free from carbon defects.

The chemical analyses of samples obtained from the evaluation ingot, the recycle ingot, and the conditioned ingot were performed in the North Laboratory which was located for easy access to the vacuum arc furnaces. The team observed a laboratory technician prepare samples, calibrate the ion coupled plasma (ICP) spectrometer using known standards, and analyze evaluation ingot sample 95057. Computer software on the laboratory information management system (LIMS) compared millivolt (mv) readings with a calibration curve of mv and percent of each element in a standard traceable to the NIST. Chemistry reports were automatically printed by LIMS as the analyses were performed. The skill of the personnel and the quality of the analytical equipment and computer software used to perform chemical analyses were observed as strengths by the team.

3.6.6 Training

The team evaluated the indoctrination and training of certain personnel to determine that the training provided was appropriate to the activities performed and that, where appropriate, refresher training was provided. The team observed that in all cases appropriate and sufficient training was provided on the schedule required by the training system. CNFD WZ had recently implemented a computer based training record system which made retrieval of training information easier and also provided timely reminders of required training activity.

The team determined that a program providing for the indoctrination and training of personnel performing activities affectir, quality had been established and maintained. The team also determined that the training program, including the computer based training record system, as a strength of the CNFD WZ operation.

3.6.7 Conclusions

The team conducted a performance-based audit of CNFD WZ to provide a basis for confidence that CNFD products will provide their intended safety functions. In order to reach that conclusion, the team evaluated the organization, staffing, training, and qualification of the operators, technicians, and PA staff. At the end of the inspection, the team determined that the performance of the CNFD WZ staff, processes, and products in each one of these areas was adequate. The team observed that a strong quality culture existed within CNFD WZ and that good QA practices were routinely performed. The team based this observation on the following items:

(a) The personnel skill of a technician's use of a computer program to calculate the quantity of alloying elements added to Zr sponge because the calculations and weighing instructions resulted in the chemical composition of CNFD WZ products to meet the requirements of customer specifications.

(b) The skilled personnel and the laboratory equipment used to perform the chemical analyses that verify the chemical composition of ingots that result from the calculations performed by the above technician.

(c) CNFD WZ implementation of the just-in-time fabrication methodology and the computer based documentation and tracking systems (e.g., LIMS).

(d) CNFD WZ identification, recognition by staff, and control of critical parameters such as chemistry, metallurgical structure, and freedom from defects in Zr alloy products produced by CNFD WZ.

3.7 10 CFR Part 21

During this inspection of CNFD, the team evaluated the ESBU procedure, the CNFD procedure, and CNFD plant specific procedures that address the requirements of 10 CFR Part 21. While the evaluation determined that the ESBU and CNFD procedures met the requirements of 10 CFR Part 21, the evaluation identified minor weaknesses in the conformance of the plant specific procedures with the CNFD procedures such that, when the CNFD plant procedures are taken together as written, they could result in the failure to evaluate deviations.

CNFD responded to this concern by drafting a common procedure that addressed the identified weaknesses. According to CNFD, the common procedure will be integrated into the plant specific administrative procedures and into the EP manual. The CNFD actions taken during the inspection satisfied the team's concerns.

APPENDIX A

PERSONS CONTACTED

The U.S. Nuclear Regulatory Commission staff participating in the inspections of Westinghouse Electric Corporation, Commercial Nuclear Fuel Division (CNFD) facilities at the Westinghouse Energy Center, CNFD Specialty Metals Plant (SMP), CNFD Columbia Plant (COLA), and the CNFD Wester Zirconium Plant (WZ) and the Westinghouse Electric Corporation personnel contacted during these inspections are listed below and designated as follows: a bullet (•) indicates that person attended the entrance meetings; a dagger (†) indicates that person attended the interim exit meetings; and a double dagger (‡) indicates that person attended the interim exit meetings, or the exit meeting, via teleconference or videoconference.

CNFD/Westinghouse Energy Center - February 5-10, 1995

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<u>CNFD Specialty Metals Plant - February 6-10, 1995</u>

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<u>CNFD Columbia Plant – February 27-March 10, 1995</u>

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<u>CNFD Western Zirconium Plant - March 20-24, 1995</u>

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APPENDIX A Continued

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APPENDIX B

GLOSSARY OF ACRONYMS

ADU	Ammonium Diuranate Conversion Process
ALPS	Advanced Loading Pattern Search
ANC	Advanced Nodal Code
ASME	The American Society of Mechanical Engineers
ASNT	American Society for Nondestructive Testing
Beff	Beta-effective
BORDER	Boron Design Requirements
BOM/KS	Bill of Material/Key Sheet
BWR	Boiling-water Reactor
C	Carbon
CAOC	Constant Axial Offset Control
CATD	Contract and Technical Data
CEC	Commonwealth Edison Company
Cf_{252}	Californium
CNFD	Commercial Nuclear Fuel Division
COLA	CNFD Columbia Plant
COLR	Core Operating Limits Report
CRDL	Current Design List
CSR	Contractile Strain Ratio
DEVL	Design Evaluation Verification List
DFBN	Debris-Filter Bottom Nozzle
DNBR	Departure from Nucleate Boiling Ratio
EBW	Electron Beam Welding
EFPT	Error Free Performance Team
EP	Engineering Procedure
ESBU	Energy Systems Business Unit
F	Fluoride
f_n	Hydride Fraction
f_r	Radial Texture Parameter
FPC	Fuel Parameters Checklist
FUDDL	Fuel Design Data List
GDC	General Design Criteria
gpm	Gallons Per Minute
GRF	Grid-To-Rod Fretting
GWD/MTU	Gigawatt-Days Per Metric Tonne of Initial Uranium Metal
H	Hydrogen
HFP	Hot Full-Power
HL&P	Houston Lighting and Power Company
Hf	Hafnium
Hz	Hertz
HZP	Hot Zero-Power
ICP	Ion Coupled Plasma
ID	Inside Diameter
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixer Grids
kg	Kilogram
LIMS	Laboratory Information Management System

APPENDIX B Continued

LOCA	Loss-of-Coolant Accident
LPD	Low-pressure Drop
mv	Millivolt
METCOM	Methods Communication
MTC	Moderator Temperature Coefficient
N	Nitrogen
N_a	Sodium
NDE	Nondestructive Examination
NDR	Nuclear Design Report
NIST	National Institute of Standards and Technology
NSA	Nuclear Safety Analysis
NSD	Nuclear Services Division
NTD	Nuclear Technology Division
OD	Outside Diameter
OPBU	Operating Plant Business Unit
PA	Product Assurance
PCMI	Pellet/Cladding Mechanical Interaction
PDMS	Project and Design Milestone Schedule System
PO	Purchase Order
P/PD&D	Product/Process Development & Design
ppm	Parts-per-million
PSE&GC	Public Service Electric & Gas Company
PWR	Pressurized-water Reactor
QA	Quality Assurance
QCDN	Quality Control Deviation or Notification
RAOC	Relaxation of Constant Axial Offset Control
RAMS	Rod Accountability Monitoring System
RCCA	Rod Cluster Control Assembly
REA	Rod Ejection Accident
RIQ	Reload Initialization Questionnaire
R/PR	Request/Problem Report
RSAC	Reload Safety Analysis Checklist
RSE	Reload Safety Evaluation Report
RSER	Reload Schedule and Energy Requirements
RSLC	Reload Safety and Licensing Checklist
RTN	Removable Top Nozzle
SAS	Safety Analysis Standards
SCE&G	South Carolina Electric and Gas Company
SER	Safety Evaluation Report
SMP	CNFD Specialty Metals Plant
STP	South Texas Project
T/H	Thermal-Hydraulics
TIR	Total Indicator Reading
TSIB	Special Inspection Branch
TREX	Tube Reduced Extrusion
UF_6	Uranium Hexafluoride
UO_2	Uranium Dioxide
UT	Ultrasonic Testing
VIS	Vendor Inspection Section
W	Westinghouse Electric Corporation

WABA	Wet Annular Burnable Absorber
WEPC	Wisconsin Electric Power Company
w/o	Weight Percent
WZ	CNFD Western Zirconium Plant
Zr	Zirconium
ZrB_2	Zirconium Diboride
Xe	Xenon

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Identifier	Title
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Information Notice 95-34	Air Acturator and supply Sir Regulator Problems in Copes-Vulcan Pressurizer Power-Operated Relief Valves

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