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Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
January–March 1992

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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Division of Reactor Inspection and Safeguards
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organization during the period from January 1992 through March 1992.

TABLE OF CONTENTS

PREFACE

A fundamental premise of the 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 total government-industry system for the inspection of commercial nuclear facilities has been designed to provide for multiple levels of inspection and verification. Licensees, contractors, and vendors each participate in a quality verification process in compliance with requirements prescribed by the NRC's rules and regulations (Title 10 Code of Federal Regulations). The NRC performs an overview of the commercial nuclear industry by inspection 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 ongoing quality verification programs.

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

The Vendor Inspection Branch (VIB) 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 licenses (construction permit holders and operating licenses) in vendor-related areas. These inspections are performed to assure that the root causes of reported vendor-related problems are determined and appropriate corrective actions are developed. The inspections also review the vendors' conformance with applicable NRC and industry quality requirements, the adequacy of licensees' oversight of their vendors, and that adequate interfaces exist between licensees and vendors.

The VIB inspection emphasis is placed on 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 identified, NRC assures that affected licensees are informed through vendor reporting or by NRC generic correspondence such as information notices and bulletins.

This periodical (White Book) is published quarterly and 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 to which the results are applicable thereby informing licensees and vendors of potential problems. In addition, the affected Regional Offices are notified of any significant problem areas that may require special attention.

The White Book also contains a list of selected bulletins and information notices involving vendor issues. Copies of other pertinent correspondence involving vendor issues are also included in this White Book issue.

Correspondence with contractors and vendors relative to inspection data contained in the White Book is placed in the USNRC Public Document Room, located in Washington, D.C.

INDEX

INSPECTION REPORTS

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

February 6, 1992

Docket No. 99900296

Mr. Frank Calella, President
Amerace Corporation
530 West Mount Pleasant Avenue
Livingston, New Jersey 07039

Dear Mr. Calella:

SUBJECT: NOTICE OF VIOLATION AND NOTICE OF NONCONFORMANCE
(NRC INSPECTION REPORT NO. 99900296/91-01)

This letter addresses the inspection of your facility at Livingston, New Jersey, led by Mr. J. J. Petrosino of my staff on October 7 through 11, 1991, and the discussions of his findings with Mr. J. F. Gerard, and other members of your staff at the conclusion of the inspection.

The U.S. Nuclear Regulatory Commission (NRC) inspectors conducted the inspection to review a matter identified by your May 20, 1991, 10 CFR Part 21 report about a problem with nuclear safety-related Agastat electrical relays. The report identified a problem with the plating on an internal component, the core stop, in your model E7000 series Agastat relays. The enclosed report discusses the areas examined during this inspection and our findings including an examination of procedures and representative records, interviews with personnel, and observations by the NRC inspection team.

Based on the results of this inspection, certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (NOV), and Notice of Nonconformance. The inspection team found that the implementation of your quality assurance (QA) program failed to meet certain NRC requirements. The inspectors also identified that the Amerace Corporation (Amerace) failed to perform adequate corrective action for previous findings identified in the 1986 NRC inspection at Amerace. During this 1991 NRC inspection, the inspectors identified additional problems in your in-process and receiving inspection area that processes all of Amerace's procured components for its Agastat Model 7000 relays. The findings included inadequate quality control (QC) inspector training, inadequate QC supervisor involvement in receipt inspection, failure to appropriately apply MIL-STD-105D statistical sampling control, and failure to establish adequate procedures to ensure that important characteristics are identified by engineering and inspected by QC for incoming components destined for nuclear power plant Class 1E use. The 1991 findings, your failure to perform adequate corrective action for the 1986 findings, and the problem with the core stop control identified in your May 20, 1991, letter are indicative of significant QA program implementation failures in your receipt inspection and QC inspection areas and warrant your prompt attention and support to adequately correct the problems.

You are required to respond to this letter and should follow the instructions specified in the enclosed NOV 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 NOV, the NRC will determine whether further NRC action is necessary to ensure compliance with NRC regulatory requirements.

You are also requested to provide a written statement in accordance with the instructions specified in the enclosed Notice of Nonconformance.

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

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

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report No. 99900296/91-01

cc w/enclosures: See next page

cc: w/enclosures:

Mr. Joseph F. Gerard
Quality Assurance Director
Amerace Corporation
530 West Mount Pleasant Avenue
Livingston, New Jersey 07039

Mr. Richard Havens
Quality Manager
Amerace Corporation
530 West Mount Pleasant Avenue
Livingston, New Jersey 07039

Enclosure 1

NOTICE OF VIOLATION

Amerace Corporation
Livingston, New Jersey

Docket No. 99900296
Report No. 91-01

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted on October 7 through 11, 1991, the NRC inspection team identified a violation of NRC requirements. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1991), the violation is as follows:

Section 21.21, "Notification of failure to comply or existence of a defect," of Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) requires, in part, that each individual or other entity subject to 10 CFR Part 21 adopt procedures that appropriately provide for evaluating deviations or informing the licensee or purchaser of the deviation in order that the licensee or purchaser may cause the deviation to be evaluated.

Contrary to the above, Amerace Corporation (Amerace) Procedure "10 CFR Part 21 Compliance," incorporated as Section 21 in Amerace's quality assurance manual, was not adequate to ensure that Amerace informs its customers in accordance with 10 CFR Part 21. The inspector found that the Amerace procedure: (1) did not require that a licensee be informed of a deviation if Amerace could not perform an evaluation, and (2) did not require that 10 CFR Part 21 be passed down to subtier vendors. (91-01-01)

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

In accordance with the provisions of 10 CFR 2.201, the Amerace Corporation 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, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation," and should include the following information for the violation:

1. The reason for the violation, or, if contested, the basis for disputing the violation,
2. The corrective steps that have been, or will be taken, and the results achieved,
3. The corrective steps that have been, or will be taken, to avoid further violations, and
4. The date when full compliance will be achieved.

Where good cause is shown, the NRC staff will consider extending the response time allowed.

Dated at Rockville, Maryland
this 6th day of February, 1992.

NOTICE OF NONCONFORMANCE

Amerace Corporation
Livingston, New Jersey

Docket No. 99900296
Report No. 91-01

During a U.S. Nuclear Regulatory Commission (NRC) inspection at the Livingston, New Jersey, facility of the Amerace Corporation (Amerace), on October 7 through 11, 1991, the inspectors identified certain Amerace activities that were not conducted in accordance with NRC requirements. These activities are described below, and have been classified as nonconformances to the requirements 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), imposed on Amerace by contractual agreement with NRC licensees, and self-imposed by the Amerace quality assurance manual (QAM), Revision B, March 7, 1991.

1. Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50 requires, in part, that the quality assurance (QA) program provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. Management shall review the status and adequacy of the QA program.

   Section 2, "Quality Assurance Organization," of Amerace's QAM states, in part, that the quality control (QC) supervisor "shall be responsible for the everyday operation of the inspection department, receiving, in-process and final inspections... indoctrination and training of personnel.... The incoming inspector, reporting to the inspection supervisor, shall be responsible for the quality of incoming materials through proper inspection methods using sampling to MIL-STD 105D and 1.0 AQL Level II, unless otherwise instructed...."

   Contrary to the above, the NRC inspectors identified three experienced QC receiving inspectors who were not suitably trained and indoctrinated in choosing the minimum inspection sample size for incoming shipments of components in accordance with the MIL-STD 105D AQL Level II sample size table requirements. (91-01-02)

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

   Section 2, "Quality Assurance Organization," of Amerace's QA manual states, in part, that "The quality assurance director is responsible for the preparation of instructions, administration, and direction of the quality assurance department... and prepares all quality policies and procedures necessary within the company to assure he end item product shall conform with all applicable requirements by means of: ... 1.3.3 Receiving inspection... All quality-related procedures shall be continually reviewed in order to improve methods, systems and equipment to the state of the art..."

Contrary to the above, Amerace failed to establish adequate procedures or instructions to ensure that its in-process and receipt inspection activities are accomplished in accordance with prescribed inspection requirements. Additionally, Amerace failed to prescribe appropriate quantitative or qualitative acceptance criteria so that its receipt inspection personnel could ensure that important activities are adequately accomplished. Specific examples of QC activities identified by the NRC inspectors that were not adequately prescribed by procedures are: (91-01-03)

- How to determine an inspection sample size for a particular lot of incoming components.
- How the QC inspectors were to apply the Amerace QAM requirement of imposing MIL-STD 105D.
- Acceptance and rejection criteria for the review of certificates of compliance from subtier vendors.
- Determination of component characteristics that need to be inspected.
- Control of non-conforming conditions, other than materials.
- Delineation of how to reduce the inspection sample size based upon product history, as allowed by Amerace's QAM.

3. Criterion VI, "Document Control," of Appendix B to 10 CFR Part 50 requires, in part, that measures be established to control the issuance of documents, that prescribe all activities affecting quality. These measures will assure that the documents are reviewed for adequacy and are distributed to and used at the location where the prescribed activity is performed.

Section 11, "Vendor Quality Survey Reports," of Amerace's QAM states in part that "[t]his procedure establishes the guidelines and methods for conducting, and reporting the quality surveys of vendors, suppliers, facilities, products and services to ascertain their ability to provide the necessary product in accordance with Amerace Corporation requirements.... Vendors shall be resurveyed... based on performance recorded by the receiving inspection department. Those vendors that supply products that are considered to be nuclear qualified shall be resurveyed in accordance with the triennial survey program as defined in the Quality Instruction Manual Document No. 24.0.... This procedure contains those vendors that are identified vendors supplying products that could be used on Class 1E safety related equipment. The quality department will maintain records of vendor quality and shall advise the purchasing department of any deterioration in quality from suppliers...."

Contrary to the above, the Amerace QA department could neither find, nor provide the NRC inspectors with copies of (1) its Quality Instruction Manual Document No. 24.0, "Sub-Vendor Surveys," or, (2) the QC timing inspection procedure for the timing head assembly (Part No. 37508). Therefore, the inspectors concluded that the documents could not be distributed to or be used at the location where the activities are performed. (91-01-04)

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

   Section 6, "In-Process Inspection," of Amerace's QAM states, in part, that "[t]his procedure defines the methods necessary for the...in-process inspection of materials as they proceed through the various stages of manufacture, fabrication, and assembly....Detailed written instructions shall be supplied to all manufacturing inspection areas...to assure conformance of material to all applicable specifications...where detailed...inspection instructions are found to be incorrect, these operations must be changed...."

   Contrary to the above, an in-process QC inspection procedure did not adequately prescribe appropriate QC acceptance criteria. The QC inspection procedure for the terminal block assembly (Part No. 32650) was inadequate. The QC inspection attributes did not correspond with the associated manufacturing shop order assembly sequence and methods; consequently, it did not allow adequate performance of QC in-process inspection of the subassemblies. (91-01-05)

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

   Section 5, "Receiving Inspection," of Amerace's QAM states, in part, that "...[s]ampling inspection shall be accomplished in accordance with MIL-STD 105D, using a "floating" sample size, depending on product history (reduced inspection plan)... 3.3.3 Classification of Characteristics: Critical. Major. Minor... shall be controlled and inspected in accordance with Amerace Corporation GIP-001, GIP-002, and GIP-003.... 3.3.4 Receiving Inspection Record Cards... shall be reviewed and approved by the Quality Control Supervisor for accuracy and shall identify the drawing requirements and those characteristics to be inspected."

a. Contrary to the above, the NRC inspectors' review of several receiving inspection record cards (IRCs) revealed that the QC supervisor did not review or approve any of the cards. (91-01-06)

b. Contrary to the above, the inspectors by their review of several receiving IRCs and their discussions with the QA and QC staff personnel determined that the receiving inspection department had not been using the QAM required GIP-001 procedure, to determine receiving inspection sample size. (91-01-07)

c. Contrary to the above, Amerace failed to ensure that it had executed the minimum inspection requirements for the below listed components. The NRC inspector identified that Amerace does not inspect the minimum number of inspection samples, nor did it inspect for all inspection attributes required: (91-01-08)

- Diaphragm Assembly, Part No. (P/N) 32372-02
- Core Stop Assembly, P/N 32006-00
- Core, P/N 32528-00
- Electrical Coil, P/N 32274-00
- Recycle Helical Compression Spring, P/N 32331-01

6. Criterion XVI, "Corrective Action," of Appendix B to 10 CFR Part 50, requires, in part, that measures will 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.

Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50 requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings, appropriate to the circumstances.

Contrary to the above, Amerace has failed to establish adequate procedures, instructions, or documents to ensure that conditions adverse to quality, other than defective material, equipment, and services are identified and corrected. (91-01-09)

7. Criterion XVIII, "Audits," of Appendix B to 10 CFR Part 50 requires, in part, that planned and periodic audits will be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits will be performed by appropriately trained personnel not having direct responsibilities in the areas being audited.

Section 2, "Quality Assurance Organization," of the Amerace QAM states, in part, that "[i]nternal quality audits shall be accomplished by a certified Lead Auditor to ascertain the conformance to all procedures. These audits, in conjunction with customer audits, shall serve to evaluate the effectiveness of the quality assurance program."

Amerace Specification Sheet No. PS-95, "Schedule For Annual Internal Audit," states, in part, that "[i]nternal audits will be conducted on an annual basis...."

Contrary to the above, the NRC inspection team identified that: (91-01-10)

- Amerace failed to perform its scheduled 1990 annual audit for the Livingston, New Jersey facility.

- Amerace's New Jersey facility annual audits for 1987, 1988, and 1989, which included QA department staff activities, were audited by QA department lead auditors who had direct responsibilities in the QA department staff activities that were audited.

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, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, 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 the item; (2) a description of steps that have been, or will be, taken to prevent recurrence; and (3) the dates when your corrective actions and preventative measures were, or will be, completed.

Dated at Rockville, Maryland
this 6th day of February, 1992.

ORGANIZATION:	Amerace Corporation 530 West Mount Pleasant Avenue Livingston, New Jersey 07039
REPORT NO.:	99900296/91-01
ORGANIZATIONAL CONTACT:	Joseph F. Gerard, QA Director (201) 992-8400
NUCLEAR INDUSTRY ACTIVITY:	The Amerace Corporation manufactures and distributes Agastat timing, magnetic latching and general purpose electrical relays. The Amerace Corporation also manufactures and distributes Buchanan terminal blocks. Either product can be procured as Class 1E qualified, or as commercial grade components.
INSPECTION CONDUCTED:	October 7 through 11, 1991
LEAD INSPECTOR:	Joseph J. Petrosino, Team Leader — 2/5/92 Date Reactive Inspection Section No. 1 (RIS-1) Vendor Inspection Branch (VIB)
OTHER INSPECTORS:	K. Sullivan, Brookhaven National Laboratory T. Tinkel, Brookhaven National Laboratory
APPROVED BY:	Richard P. McIntyre for Uldis Potapovs, Section Chief — 2/5/92 Date RIS-1, VIB, Division of Reactor Inspection and Safeguards
INSPECTION BASES:	10 CFR Part 21 and Appendix B to 10 CFR Part 50
INSPECTION SCOPE:	1. To review the circumstances surrounding a May 20, 1991, Amerace letter to the NRC discussing suspected faulty parts that may have been installed in numerous Agastat Class 1E qualified E-7000 electrical relays. 2. To verify that Amerace has adequately implemented its corrective action to the NRC's 1986 inspection of the Amerace facility. 3. To review the circumstances surrounding an issue where an authorized Amerace distributor changed the serial numbers on commercial grade Agastat relays. The serial numbers were changed into consecutively sequenced numbers.
PLANTS AFFECTED:	Multiple

1 SUMMARY

1.1 Violations

1.1.1 Contrary to Section 21.21, "Notification of failure to comply or existence of a defect," of 10 CFR Part 21, the Amerace Corporation (Amerace) failed to adopt procedures that were appropriate for evaluating deviations or informing the licensee or purchaser of the deviation in order that the licensee or purchaser could have the deviation evaluated. (91-01-01)

1.2 Nonconformances

1.2.1 Contrary to Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50, and Section 2, "Quality Assurance Organization," of Amerace's quality assurance manual (QAM), the U.S. Nuclear Regulatory Commission (NRC) inspectors identified three quality control (QC) receiving inspectors who did not receive suitable training or indoctrination about how to choose a minimum inspection sample size for incoming shipments of components in accordance with the MIL-STD 105D AQL Level II sample size table, as required by Amerace's QAM. (91-01-02)

1.2.2 Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, and Section 5, "Receiving Inspection," and Section 6, "In-Process Inspection," of Amerace's QAM, Amerace failed to ensure that certain of its QC in-process and receipt inspection activities are accomplished in accordance with prescribed inspection requirements. Amerace also failed to prescribe appropriate quantitative or qualitative acceptance and rejection criteria so that its inspection staff could ensure that important activities are accomplished. Specific examples of QC activities that were inadequately prescribed by procedures or instructions are how to: (91-01-03)

- Determine an inspection sample size for a particular lot of incoming components.
- Apply the Amerace imposed MIL-STD 105D requirements by the QC staff.
- Find and use methodology of acceptance or rejection for certificate of compliance reviews.
- Determine characteristics that need to be inspected.
- Control non-conforming materials to prevent inadvertent use and how to control non-conforming conditions other than material and services.
- Reduce the inspection sample size based upon "product history," as allowed by the QAM.

1.2.3 Contrary to Criterion VI, "Document Control," and Criterion XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50, and Section 11, "Vendor Quality Survey Reports," of Amerace's QAM, the Amerace QA staff could not provide and could not find its (1) Quality Instruction Manual Document No. 24.0, which prescribes vendor survey history and requirements, and (2) the QC procedure for inspection of the Agastat Model 7000 timing head assembly (Part No. 37508). (91-01-04)

1.2.4 Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, and Section 6, "In-Process Inspection," of Amerace's QAM, the inspectors identified a QC in-process inspection procedure that did not adequately prescribe appropriate QC acceptance criteria. The QC procedure for inspection of the Agastat Model 7000 terminal block assembly (Part No. 32650) did not correspond to the manufacturing assembly sequence delineated on the internal shop order. (91-01-05)

1.2.5 Contrary to Criterion X, "Inspection," of Appendix B to 10 CFR Part 50, and Section 5, "Receiving Inspection," of Amerace's QAM:

1.2.5.1 The inspectors found that the QC supervisor is not typically reviewing or approving the QC "receiving inspection record cards," (IRCs) as required by Amerace's QAM. (91-01-06)

1.2.5.2 The inspectors identified that the QC department receiving inspection staff were not using Amerace's GIP-001 procedure as required to determine the appropriate receiving inspection sample size. (91-01-07)

1.2.5.3 The inspectors identified that each of the five QC IRCs reviewed for the following components (which represented dozens of individual component lots that Amerace had receipt inspected) indicated that the sample size of each lot inspected was lower than the minimum requirement: (91-01-08)

- Diaphragm Assembly, Part No. (P/N) 32372-02
- Core Stop Assembly, P/N 32006-00
- Core, P/N 32528-00
- Electrical Coil, P/N 32274-00
- Recycle Helical Compression Spring, P/N 32331-01

1.2.6 Contrary to Criterion XVI, "Corrective Action," and Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, Amerace failed to adequately establish procedures, instructions, or other documents to ensure that conditions adverse to quality, other than defective material, are promptly identified and corrected. (91-01-09)

1.2.7 Contrary to Criterion XVIII, "Audits," and Amerace's QA program Specification Sheet No. PS-95, "Schedule For Annual Internal Audits": (91-01-10)

- Amerace failed to perform its scheduled 1990 annual audit for the Livingston, New Jersey, facility.

- Amerace's New Jersey facility annual audits for 1987, 1988, and 1989, which included QA department activity, were audited by QA department lead auditors who had direct responsibilities in the QA department activities that were audited.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 (Closed) Violation 86-01-01 - Report 99900296/86-01

The 1986 NRC inspection report indicated that Amerace failed to impose the provisions of 10 CFR Part 21 on the Control Products Corporation (CPC), Grafton, Wisconsin. CPC manufactures and distributes Class 1E qualified Agastat electrical relay models EGP, ETR, and EML for Amerace. The NRC inspector reviewed several recent Amerace purchase orders (POs) to CPC and verified that Amerace is imposing the requirements of 10 CFR Part 21 to its only subtier vendor who manufactures safety-related components.

2.2 (Closed) Nonconformance 86-01-02 - Report 99900296/86-01

The 1986 NRC inspection report indicated that the QA organization lacked adequate independence from cost and scheduling because the QA manager reported to the production manager. The NRC inspector reviewed Amerace's QA manual and conducted discussions with the QA director regarding the current QA organization. The QA director currently reports to the Amerace President and appears to have adequate independence from cost and schedule considerations.

2.3 (Open) Nonconformance 86-01-03 - Report 99900296/86-01

The 1986 NRC inspection report indicated that the QA staff had not established adequate QA overview to verify the torque values on certain Agastat Model 7000 relay screws and had failed to ensure that inspection criteria were established for a certain critical measurement on a contact strap. However, the NRC inspector did not verify whether Amerace performed its corrective action for this matter.

2.4 (Closed) Nonconformance 86-01-04 - Report 99900296/86-01

The 1986 NRC inspection report indicated that Amerace had failed to establish QC procedures and instructions for the in-process and receipt inspection areas. Report 86-01 discusses the problems that were identified by the NRC inspectors in sections E.4 and E.7. A review of these Sections in conjunction with the findings of this 1991 inspection indicate that the Amerace corrective action regarding 1986 nonconformance 86-01-04 was inadequate.

Although Amerace's corrective action was inadequate, nonconformance 86-01-04 is considered closed because Amerace's corrective action for nonconformances 91-01-02, 03, 05, 06, 07, and 08 should appropriately address the NRC concern.

2.5 (Closed) Nonconformance 86-01-05 - Report 99900296/86-01

The 1986 NRC inspection report indicated that Amerace had failed to effectively control its receipt inspection department. Nonconformance 86-01-05 identified that the QC supervisor was not involved in the material receipt inspection checklists which the QC inspectors were using without benefit of procedures or instructions for guidance. Report 86-01 discusses the NRC findings in section E.5. A review of section E.5 in conjunction with a review of the current findings indicates that Amerace failed to perform adequate corrective action for nonconformance 86-01-05.

However, Nonconformance 86-01-05 is considered closed. Nonconformances 91-01-02, 03, 06,07 and 08 should appropriately address the area of NRC concern.

2.6 (Closed) Nonconformance 86-01-06 - Report 99900296/86-01

The 1986 NRC inspection report indicated that Amerace procured the calibration services of a vendor who was not on its approved vendors list. The vendor was Sheffield Measurement Division (SMD) of the Warner and Swassey Company. After reviewing documents and conducting discussions the NRC inspector determined that Amerace performed adequate corrective action for this matter. Amerace has performed vendor surveys of SMD and currently maintains SMD on its approved vendor listing in accordance with its QAM requirements.

2.7 (Open) Nonconformance 86-01-07 - Report 99900296/86-01

The 1986 NRC inspection report identifies four areas of concern for 86-01-07: (1) that QC inspectors verified only a portion of the stated QC inspection lot population for certain characteristics, (2) that Amerace failed to establish written instructions for the receipt and in-process inspectors, (3) that travelers were not established, and (4) that the final QC inspection practices did not assure that the quality records were completed.

The NRC inspector did not verify whether Amerace performed adequate corrective action for the third and fourth issues which therefore will remain open. The first and second issues were again identified as concerns during the 1991 NRC inspection. Consequently, it would appear that Amerace failed to perform adequate corrective action for the first and second aspects of nonconformance 86-01-07 which will also remain open.

2.8 (Open) Nonconformance 86-01-08 - Report 99900296/86-01

This matter was not reviewed by the NRC inspectors.

2.9 (Closed) Nonconformance 86-01-9 - Report 99900296/86-01

The 1986 NRC inspection report stated that Amerace did not adequately control 68,500 potentially defective contact arm assemblies. The NRC inspector reviewed records and conducted discussions with the Amerace engineering staff about this matter. All of the 68,500 parts were accounted for, and were appropriately disposed. This matter is therefore closed.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

On October 7, 1991, the NRC inspector discussed that scope of the inspection with Amerace QA and engineering staff at Amerace's Livingston, New Jersey, facility. After the entrance meeting, the NRC inspection team was taken on a tour of the Amerace facilities. During the exit meeting at the conclusion of the inspection on October 11, 1991, the NRC team leader summarized the conclusions, findings, and concerns that the team identified during the inspection for the Amerce staff. At this meeting Amerace management representatives committed to the NRC team leader that they would develop a letter and provide their 390 authorized distributors criteria regarding consecutive serial numbers assigned to nonhomegenous lots of commercial grade (CG) Agastat Model 7000 relays by some authorized Amerace distributors. This matter is discussed in section 3.2 and 3.10.

3.2 Background

The NRC received a May 20, 1991, 10 CFR Part 21 report from Amerace regarding a deviation involving a potential problem in its Agastat electropneumatic Model E7000 timing relays. According to Amerace, the potential problem was limited to its E7022 and E7024, A-L series, alternating current (ac) relays. Amerace stated that the relays may not time out within the specified time or may not provide the stated repeat accuracy for the operating life of the relay. This potential problem was caused by one internal part, the "core stop," which is used in the timing head assembly of the E7022 and E7024 ac relays. The Amerace letter indicates that several "lots" of core stops may have had insufficient metallic plating applied, and the poorly plated core stops could have "gotten through receiving inspection, and have been assembled into final product."

The NRC inspectors reviewed this problem and a second area that involved findings identified during the previous NRC inspection performed August 25 through 29, 1986. The inspection was documented in NRC Inspection Report No. 99900296/86-01 (IR 86-01). IR 86-01 identified NRC concerns about the effectiveness of Amerace's QA program. Additionally, NRC Information Notice (IN) 88-35, "Inadequate Licensee Performed Vendor Audits," discussed Amerace's failure to "adequately establish and implement a QA program in several areas." IN 88-35 also discussed the fact that Amerace's QA program problems were not adequately identified during licensee audits of Amerace.

A third area of review concerned information the NRC received indicating that an authorized Amerace distributor may be supplying commercial grade (CG) Agastat Model 7000 relays with labels that have been altered. The information received indicated that the authorized Amerace distributor (AD) supplied CG

Agastat relays with labels that had either been replaced or altered to reflect consecutive serial numbers. The consecutive serial numbers were requested from the vendor and provided part of the basis to use CG components in safety-related application, that is, the same lot/date code as evidenced by consecutive serial numbers.

Therefore, this inspection was performed (1) to verify whether Amerace adequately implemented its corrective actions committed to for the findings in IR 86-01; (2) to verify whether the May 20, 1991 Amerace problem with poorly plated core stops was related to the findings in IR 86-01; and (3) to determine whether Amerace is adequately controlling its ADs who supply CG Agastat Model 7000 relays to NRC licensees for possible use in safety-related Class 1E systems.

3.3 Part 21 of Title 10, Code of Federal Regulations (10 CFR Part 21)

Section 21.21 of 10 CFR Part 21 requires, in part, that individuals, corporations, or other entities subject to 10 CFR Part 21 adopt procedures that appropriately provide for evaluating deviations, or informing the licensee or purchaser of the deviations, in order that the licensee or purchaser may have the deviation evaluated.

The NRC inspector reviewed the Amerace procedure adopted for this purpose. The procedure is titled "10 CFR 21 Compliance," Revision A, dated August 1, 1988, and was incorporated into the Amerace QAM as Section 21. The NRC inspector identified a few inconsistencies in the establishment of the procedure. The procedure was divided into three sections: posting requirements, purchase orders (POs) and notification. The posting section appeared to be adequately established. The PO section required that customer POs received by Amerace, that are nuclear safety related, must state that 10 CFR Part 21 is applicable. The inspector determined by discussion with Amerace that this was meant to implement 10 CFR Section 21.31, "Procurement documents." However, Section 21.31 requires procurement documents issued by the vendor to specify, when applicable, that the provisions of 10 CFR Part 21 apply.

The last section, "Notification," was not adequately established to provide that customers be informed of deviations that Amerace could not evaluate. The NRC inspector also observed that Amerace initiated the development of an interim procedure to address the requirements of 10 CFR Part 21. Violation 91-01-01 was identified in this area.

3.4 Receipt Inspection Area

Section 2, "Quality Assurance Organization," of Amerace's QAM requires, in part, that the incoming inspector, reporting to the inspection supervisor, will be responsible for the quality of incoming materials. The incoming inspector will use proper inspection methods, which include sampling to military standard (MIL-STD) 105D, and associated acceptable quality level (AQL) 1.0 level II sample size table, unless otherwise instructed. The QAM also states that the QC supervisor is responsible for the every day operation of the inspection department and for indoctrinating and training personnel. Section 18, "Indoctrination and Training," of Amerace's QAM requires that inspection personnel who perform activities affecting quality are indoctrinated and trained to assure that suitable proficiency is achieved and maintained. The training program shall include instructions in the use of inspection methods. The 1986 NRC inspection of Amerace at its former Union, New Jersey, facility identified findings in several areas, including receipt inspection, inspection methodology, and inconsistencies in Amerace's receiving inspection record cards (IRCs). These findings are discussed in detail in NRC inspection report No. 99900296/86-01 (IR 86-01). Consequently, the NRC inspector reviewed records, observed activities affecting quality, and conducted discussions with Amerace personnel. The NRC inspector discussed the operation of the E7000 relay with the engineering staff to identify internal components that were important to the functionality of the relay. The inspector chose five parts that are important and requested copies of the current IRCs that were associated with each. The chosen parts are as follows:

- Diaphragm assembly — Part No. (P/N) 32372-02
- Core stop assembly — P/N 32006-00
- Core — P/N 32528-00
- Coil — P/N 32274-00
- Recycle Helical Spring — P/N 32331-01

The IRC documents represent the official inspection record for each Amerace-procured part that is used on the model 7000 relay, both commercial-grade and safety-related. The IRCs for the model 7000 relay parts typically contain the following: the name and P/N of the component, the subtier vendor for each shipment received, the previous shipments, the quantity of each shipment, the inspection sample size inspected, and each inspection characteristic that is to be inspected.

The NRC inspector then conducted discussions with three different experienced QC receiving inspectors. The discussions focused on incoming component shipments that they inspected and documented on the applicable IRC. The NRC inspector requested that the QC inspectors explain how the size of the inspection sample is determined for each incoming shipment, how the inspection characteristics are determined, who determines the characteristics, the involvement and overview of their QC supervisor, and other related aspects. Within these areas, the NRC inspector identified numerous inconsistencies, some of which are very similar, if not identical, to the problems identified at Amerace in 1986. The discussions with the QC inspectors indicate that they are not provided adequate written instructions or procedures, and they do not appear to be appropriately trained or indoctrinated to the QAM requirements, such as how to determine an inspection sample size for an incoming shipment of components. Numerous inconsistencies were noted, and a summary of the majority of the problems the NRC inspectors found follows:

- None of the three Amerace QC receiving inspectors interviewed exhibited adequate proficiency in the method by which an inspection sample size is determined for an incoming component shipment.

- Amerace's QAM requires that the QC staff use MIL-STD 105D's AQL Level II table. However, one inspector stated that the receiving clerk picked out the sample and the sample size and gave it to the QC staff after the shipment was counted.

- None of the IRC's reviewed indicated that the QC supervisor reviewed or approved the documents as required by the QAM.

- The IRC for the helical spring did not require any inspection attribute for spring force or compression. However, the NRC inspector's review of the Amerace design drawing revealed design requirements for spring force/compression. The inspector identified that 27,000 helical springs were receipt-inspected between October 1990 and May 1991 without verifying the spring force/compression listed on the design drawing.

- The QC receipt inspectors stated that they were not using the QAM Section 5 required procedure, GIP-001, which provides guidance in setting the inspection sample size based on the classification of characteristics.

- The IRC for the "coil" indicated that one of the QC inspection attributes to be inspected was "soldering of the coil lead wire splice joint." The IRC indicated that Amerace inspected this soldering for each of the several incoming shipments that was received. The coil, P/N 32274-00 is completely encapsulated in a plastic frame and is covered with potting compound. The coils are not fabricated or manufactured by Amerace. The coil soldered lead wire junction is not exposed, rather it is internal. Therefore, this inspection attribute could not have been inspected by the Amerace QC staff.

- The IRC for the "coil" also showed that the inspection sample size was always the same, regardless of the size of the incoming shipment of coils. Only one coil was inspected. For example, for one shipment of 1080 coils, one coil was inspected; however, the AQL level II table shows that 80 should have been inspected for a lot of 1080 pieces.

- QC inspectors are not provided with written instructions or procedures which prescribe the receipt inspection process other than those discussed above.

- The IRC for the diaphragm assembly showed that on August 8, 1991, Amerace received a shipment of 18,000 components. The NRC inspector reviewed the AQL level II table and identified that 315 is the minimum inspection sample lot for 18,000 pieces. However, the IRC stated that the size of the inspection sample size was 300 pieces. The NRC inspector was told that all 300 pieces received a visual inspection, but that only 8 pieces were inspected for each of the inspection attributes stated on the IRC.

- Every IRC reviewed revealed that the inspection sample size was less than required. On IRCs that indicated that previous lots were rejected, the sample size still remained the same.

In summary, the NRC inspector concludes that Amerace failed to adequately indoctrinate and train its QC receiving inspectors regarding inspection sample size and inspection methodology (Nonconformance 91-01-02); Amerace failed to effectively establish or implement receiving and in-process quality activities, this is also discussed in section 3.5 below (Nonconformance 91-01-03); Amerace failed to adequately establish or implement two QC procedures in its in-process inspection area (Nonconformance 91-01-05); Amerace failed to ensure that its QC supervisor reviewed and approved its receiving IRCs, as required by the QAM (Nonconformance 91-01-06); Amerace failed to ensure that its receiving inspection staff was using the required GIP-001 procedure (Nonconformance 91-01-07); Amerace failed to ensure that its receiving inspection staff was inspecting the minimum number of samples for incoming shipments of Agastat model 7000 and E7000 relays, and Amerace failed to ensure that the receipt inspection sample was being effectively inspected for all applicable characteristics that are important to the functionality of the Agastat E7000 Class 1E qualified relays (Nonconformance 91-01-08).

3.5 In-Process Inspection

Section 5, "Receiving Inspection," and Section 6, "In-Process Inspection," of Amerace's QAM requires that the inspection process be implemented in accordance with the established procedures and instructions. Section 5 requires that sampling inspection shall be accomplished in accordance with MIL-STD 105D, using a floating sample size, depending on product history; all materials and services furnished by a vendor shall be inspected for conformance to the applicable purchase order (PO), drawing, and specification requirements; and that the IRC be reviewed and approved by the QC supervisor for accuracy and to identify the drawing requirements and those characteristics to be inspected. Section 6 requires continuous first-article and in-process inspection of materials as they proceed through the various stages of manufacture; and where detailed manufacturing and inspection instructions are found to be incorrect, these operations must be changed to reflect the current procedure by having the planning department adjust the operational route sheets to conform with the current sequence of operation, and the inspection department shall verify all pertinent data, such as dimensional accuracy to the drawing requirements.

The NRC inspector reviewed documents, observed work activities in progress, and conducted discussions with Amerace personnel. In the Agastat relay assembly area the NRC inspector reviewed the in-process inspection procedure (IP) for the model 7000 and E7000 terminal block assembly (TBA), P/N 32650, and conducted discussions with the QC inspector. The NRC inspector noted that some of the inspection attributes did not correspond with the associated TBA assembly instruction sequence. After comparing the IP and associated manufacturing shop order (SO), the inspector identified two manufacturing assembly work activities that were not listed on the IP. He also identified two other manufacturing work activities that did not correspond with the sequence number on the IP. Specifically, (1) IP sequence number 50 is a weighing and screw tightening step; however, SO sequence number 50 is a terminal block cleaning aid and contact checking step, (2) IP sequence number 60 is a continuity step after cleaning; however, SO sequence number 60 is a weighing step. The NRC inspector requested the IP for the model 7000 and E7000 timing head assembly. However, Amerace was not able to find its IP for the E7000 timing head assembly. Therefore, the inspector compared the model 7000 timing head assembly IP with the SO. The IP contained sequence steps when compared to the SO, similar to those found in the terminal block area.

Based upon the above discussion, the NRC inspector concluded that Amerace has failed to effectively implement its in-process inspection program in the above area. Nonconformances 91-01-03, 04, and 05 were identified in this area. Examples of these inconsistencies are also discussed in section 3.4 above.

3.6 Qualified Vendors

Section 11, "Vendor Quality Survey Reports," of Amerace's QAM requires, in part, that certain activities be performed to ascertain a vendor's ability to provide Amerace the necessary product in accordance with Amerace requirements. These activities include requirements to evaluate and record vendors on a vendor list delineate special capabilities, document vendor surveys, keep the vendor list current, and resurvey vendors who supply products considered to be nuclear qualified in accordance with the triennial survey program, as defined in the Quality Instruction Manual (QIM), Document No. 24.0.

The NRC inspector started to review this area at the beginning of the inspection. At that time the inspector requested a copy of QIM Document No. 24.0. However, Amerace personnel were not able to find or to provide a copy of this document. The NRC inspector could not perform an adequate review of this area. The inspector, therefore, identified a failure to provide and control documents to personnel at the location where the prescribed activity is controlled, and also failed to retrieve QA records. Nonconformance 91-01-04 was identified in this area.

3.7 Conditions Adverse to Quality

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

Section 9, "Control of Non-Conforming Materials," of Amerace's QAM establishes requirements for controlling materials and services found to be nonconforming with drawings, specifications or purchase orders.

The NRC inspector asked the Amerace personnel why they did not write a nonconformance report on a few of the deficiencies discussed in Sections 3.4 and 3.5 above. The NRC inspector found that Amerace had not established measures to control conditions adverse to quality, other than material and component problems and services. For example, if a QC inspector lost an inspection procedure or record, or a drawing was out of control, he did not have any procedure or instructions for guidance on how to disposition the nonconformance, or record the deficiency. Nonconformance 91-01-09 was identified in this area.

3.8 Audits

Section 16, "Quality Audits," of Amerace's QAM requires, in part, that audits of manufacturing, quality procedures, and inspection functions be conducted by an auditor who is selected and approved by the plant manager and QA Director. Amerace Specification Sheet No. PS-95, "Schedule for Annual Internal Audits," states, in part, that internal audits be conducted on an annual basis and that the audits be conducted using the requirements of Section 19 of ANSI/ASME N45.2.

The NRC inspector requested the last four annual audits of the Amerace facility. The Amerace staff informed him that the 1990 audit was not performed. Therefore, he reviewed the 1987, 1988, and 1989 audits. During this review, the NRC inspector identified that each of the audits was performed by only one auditor. The auditor for each audit was an Amerace QA department member. Each of the auditors reviewed areas for which he had direct responsibilities. Both ANSI/ASME N45.2 and Appendix B to 10 CFR Part 50 prohibit a person with direct responsibilities in an area from auditing that area. Therefore the NRC inspector concluded that (1) Amerace failed to perform its regularly scheduled audit for 1990, and (2) Amerace's New Jersey facility annual audits for 1987, 1988, and 1989 were audited by a QA department auditor who had direct responsibilities in certain QA department activities that were included in the audits. Nonconformance 91-01-10 was identified in this area.

3.9 Differences Between E7000 and 7000 Relays

The 7000 ser[illegible] Agastat timing relays are electropneumatic devices. The E prefix is u[illegible] by Amerace to distinguish its nuclear-qualified product line of relays from [illegible] commercial-grade product line (7000 series). Amerace qualified the E700[illegible] for use in Class 1E systems located outside containment. Depending on the specific application and type of relay chosen, the amount of on-delay or off-delay is controlled through the adjustment of a timing dial, located on top of the relay. The setting of this dial basically controls the air bleed-off rate of an internal diaphragm assembly which is mechanically coupled to the relay core assembly. Amerace also provides an electronic control relay line which also has a nuclear qualified series and a commercial grade series. The EGP, EML, and ETR control relays are electrical/electronic devices that are qualified for Class 1E systems outside of containment. The CG line is not prefixed by an E. The E7000 series relays are manufactured only at Amerace's Livingston, New Jersey, facility. The ETR, EML, and EGP are manufactured only at Control Products Corporation, Grafton, Wisconsin, for Amerace. However, the CG line of these relays are manufactured in several different domestic and foreign facilities. NRC IN 87-66, "Inappropriate Application of Commercial-Grade Components," discusses these differences in more detail.

During this inspection the NRC inspectors assessed the differences between the E7000 and the 7000 relays. They also reviewed the initial IEEE 323 and 344 qualification tests for the E7000 relays. Amerace considers many of the documents and specific technical differences as proprietary and company confidential. Therefore, this portion of the NRC inspection report will generalize the areas reviewed and not provide specific details.

Overall, the NRC inspectors noted differences between several components that are used for the E7000 relay and the 7000 relay. Also noted was that the E7000 instruction manual contains some information that is unique for nuclear power plant applications. The NRC inspector performed the actual review and evaluation on Amerace's E7012 and 7012 series relays. The inspector chose the E7012 series because the E7012 is the most widely purchased for use at nuclear facilities according to Amerace staff.

The NRC inspector reviewed Amerace's E7000 design because the NRC inspector had observed in some recent correspondence between Amerace and licensees that Amerace stated its E7000 design was "frozen." The inspector's review showed that Amerace's E7000 design is not frozen. Amerace has made numerous changes since its qualification testing of the E7000 relay. However, Amerace does employ a restrictive design control, which was reviewed. This design control process is controlled by procedure TP-009, "EJR/EJO System Procedure." Procedure TP-009 was not reviewed in detail, instead specific design changes were reviewed to assess the adequacy of Amerace's control. No adverse findings were identified during the NRC inspector's review of this area.

3.10 Authorized Amerace Distributors

The NRC staff received a concern regarding commercial-grade Agastat model 7000 relays, which were purchased from an authorized Amerace distributor for the purpose of dedication (as defined in 10 CFR Part 21) and use at Portland General Electric's (PGE) Trojan nuclear power plant.

In early 1991, Spectrum Technologies (ST), Schenectady, New York, accepted a PGE PO for six Agastat model 7032 PBB relays. The model 7032 is the only 7000 series relay that Amerace does not offer as a qualified Class 1E relay. ST therefore ordered the CG relays from Westinghouse Electric Supply Company (WESCO), Albany, New York, who in turn ordered them from Control Components Supply (CCS), Short Hills, New Jersey. CCS is an authorized Amerace distributor. The ST PO to WESCO, and WESCO's PO to CCS required traceability to Amerace, the same lot and date code, and consecutive serial numbers. ST received the relays from CCS, and the nameplate label (label) serial number (S/N) indicated that the relays were all manufactured in the same week of 1991. The S/Ns were also consecutively numbered. During the dedication testing at ST, however, ST stated that it was having difficulty verifying the calibration of the units. Consequently, ST transported the units to Amerace's Livingston, New Jersey, manufacturing facility for technical assistance. At that time Amerace informed ST that the units' labels had been modified or changed because a date code, which is stamped on each 7000 and E7000 series relay, indicated the units were manufactured in the 39th week of 1989, instead of 1991.

The NRC staff reviewed the circumstances of this matter at the ST, CCS, and Amerace facilities and found that:

1. Authorized Amerace distributors (ADs) can "field" modify the CG 7000 series relays as necessary to comply with customer requirements.

2. The ADs are supposed to install a new label on any relays that are modified to indicate that a "field" change was made. The preprinted labels supplied by Amerace have an "F," for "field" change prefixing the S/N. However, Amerace did not formally express this policy to its ADs.

3. Each electrical coil unit has an "F" prefixed label included in its individual shipping box.

4. Amerace did not contractually require its ADs to use the F prefixed labels on any CGs that were modified. Typical modifications could include different timing disc units (with timing duration dial skirt), contact block assemblies, and different electrical coil units.

5. Before final calibration, test and acceptance of 7000 and E7000 relays, each relay is heat stabilized in an electrical convection oven for 4 hours at a particular temperature under 200 degrees Fahrenheit for two reasons. This heat stabilization is performed to mate the timing disc with the timing ceramic wafer to prevent timing drift and ensure repeat accuracy. Secondly, it is used to stress relieve all non-metallic parts.

6. Amerace states that any timing disc change requires the relay to be re-stabilized. Amerace also states that only a few of its ADs will change timing discs and ceramic wafers on the 7000 series relays. Amerace discourages its ADs from changing anything other than electrical coils and contact block assemblies. However, the NRC inspectors identified that although this may be Amerace's policy, it was not formally transmitted to the ADs.

7. In December of 1991, Amerace stated that it would transmit letters to each of its 390 ADs. A draft of the letter was reviewed by NRC staff.

The letter generally instructed the ADs on Amerace's policy on field changed relays. The draft letter stated, in part:

- Please ensure the label, P/N 38010-01, enclosed with the coil kit is correctly filled out to reflect the new catalog number.
- The serial number <u>must</u> be exactly the same as on the original unit except the number will be prefixed by an "F" indicating field modifications were made.
- Non-adherence of this procedure will void all factory warranties.

8. CCS is one of Amerace's largest distributors for CG 7000 series relays and Buchanan terminal blocks.

9. CCS stated to NRC staff that they typically use their own labels when they modify a CG relay. The CCS labels do not have an F prefix to indicate field modification. They have also [illegible]ically assigned their own consecutive serial numbers to an order o[illegible] [illegible]ays, and they have used the week and year of their modification for the relays S/N (instead of the date of manufacture). For example, CCS used the week and year when they changed the timing disc on ST's order for the six 7032 relays, and CCS also assigned consecutive serial numbers.

10. Amerace will issue certificates of compliance or conformance (CoCs) for its products if contractually required. However, Amerace's CoCs are usually only issued for E7000 series nuclear qualified relays, from Amerace's Livingston, New Jersey facility.

11. Amerace does issue CoCs for some of their CG 7000 series relays in special cases. These CoCs are also issued from the Livingston, New Jersey facility.

12. The NRC inspector reviewed CCS's customer list and associated procurement documents. No NRC licensee safety-related orders were identified. However, the NRC inspector found that approximately 25 NRC-licensed facility owners were listed as customers of CCS. The NRC inspectors also noted that CCS typically supplies relays to other Amerace ADs. Therefore, some NRC licensees may have purchased CG 7000 series relays, which are consecutively numbered but may not be from the same manufacturing lot or date code.

In summary, it appears that the potential exists for NRC licensees to have received CG Agastat 7000 series relays for dedication and use in Class 1E systems that were not from the same manufacturing lot/date code, even though the S/N's were consecutively numbered. CCS stated that they have typically assigned consecutive serial numbers to the 7000 CG relays which they have modified. The CG 7000 series relays are controlled and assembled differently than the E7000 series relays. As discussed above the nuclear qualified relays have some differences from the GS relay parts, are controlled for design and configuration, and were initially tested in accordance with IEEE 323/344.

However, it is important to note that Amerace stated that it does <u>not use</u> its AD network for supply of its E7000 series Class 1E qualified relays. Amerace encourages licensees and vendors to procure its Agastat E7000 relays direct from its Livingston, New Jersey facility. Conversely, Amerace <u>will not</u> supply its commercial-grade products directly to a vendor or licensee. Instead, Amerace uses its established network of approximately 390 ADs to supply Amerace commercial-grade products.

4 PERSONNEL CONTACTED AT AMERACE

<u>Name</u>	<u>Title</u>
*J. F. Gerard	QA Director
*M. R. Bhojwani	Senior Product/Market Manager
*R. F. Havens	Quality Manager
*E. J. Leszczak	Senior Product Engineer
*D. Weisberger	Applications Engineer
*H. Wingerter	Quality Engineer
M. Clark	Quality Inspector

*Attended the exit meeting.

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

FEB 1 7 1992

Docket No. 99900369

Mr. J. Hans Kluge, President
and Chief Executive Officer
Automatic Switch Company
50-56 Hanover Road
Florham Park, New Jersey 07932

Dear Mr. Kluge:

SUBJECT: NOTICE OF VIOLATION AND NOTICE OF NONCONFORMANCE (NRC INSPECTION REPORT NO. 99900369/91-01)

This letter addresses the inspection of your facility at Florham Park, New Jersey, led by Mr. J. J. Petrosino of this office on August 26-30, 1991, and October 21-24, 1991, and the discussions of the team's findings with you and members of your staff at the conclusion of the inspection. The purpose of the inspection was to review the circumstances surrounding five problems recently identified at U.S. nuclear power plant facilities concerning Automatic Switch Company (ASCO) solenoid valves. As a result of problems associated with the ASCO solenoid valves at the power plants, various U.S. Nuclear Regulatory Commission (NRC) licensee designed safety-related systems did not operate as required. The enclosed report discusses the areas examined during this inspection and our findings.

During this inspection, the NRC inspection team examined procedures and representative records, interviewed personnel, and made observations. As a result, certain of your activities appeared to be in violation of NRC requirements, as specified in Enclosure 1, Notice of Violation (Notice), and Notice of Nonconformance (Enclosure 2). The most significant concern identified was that ASCO did not perform any manufacturing verification or quality assurance inspection sampling for minimum air flow rate (Cv) on its 206, 208, 210, or NP series solenoid operated valves (SOVs). ASCO's subsequent testing of several 206-381-5RF SOVs indicated that the majority had actual flow rates that were below the Cv value published in ASCO catalog NP-1. Therefore, ASCO informed its customers of the circumstances surrounding this deviation during the period October 18-21, 1991.

The specific findings and references to the pertinent requirements are identified in the enclosed Notice of Violation and Notice of Nonconformance. Although Section 2.201 to Title 10, of the Code of Federal Regulations, Part 2 (10 CFR 2.201), requires you to submit to this office, within 30 days of your receipt of this Notice, a written statement of explanation, we note that this violation had been corrected and those actions were reviewed during this inspection. Therefore, no response with respect to this matter is required. However, you are requested to provide a written statement in accordance with the instructions specified in the enclosed Notice of Nonconformance.

The response requested by this letter and the enclosed Notice of Nonconformance is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511. In accordance with 10 CFR Part 2.790 (a), a copy of this letter and its enclosures will be placed in the NRC's Public Document Room.

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

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99900369/91-01

cc: w/enclosures
Mr. D. Tompsen, QA Director
Automatic Switch Company
50-56 Hanover Road
Florham Park, New Jersey 07932

ENCLOSURE 1

NOTICE OF VIOLATION

Automatic Switch Company
Florham Park, New Jersey

Docket No. 99900369/91-01

During a Nuclear Regulatory Commission (NRC) inspection conducted on August 26-30, 1991, and October 21-24, 1991, a violation of NRC requirements was identified. In accordance with the Appendix C (1991), "General Statement of Policy and Procedure for NRC Enforcement Actions," to Title 10, of the Code of Federal Regulations, Part 2 (Appendix C to 10 CFR Part 2), the violation is listed below:

Section 21.21, "Notification of failure to comply or existence of a defect," of 10 CFR Part 21 requires in part, that, each corporation, partnership or other entity subject to this regulation, adopt appropriate procedures which will provide for either, evaluating deviations, or informing the licensee or purchaser of the deviation.

Section 21.6, "Posting requirements," of 10 CFR Part 21 states, in part, that, if posting of the procedures adopted pursuant to 10 CFR Part 21 is not practicable, the licensee or firm subject to the regulations in this part may, in addition to posting section 206, post a notice which describes the regulations/procedures, including the name of the individual to whom reports may be made, and states where they may be examined.

Contrary to these requirements, the Automatic Switch Company (ASCO) (1) failed to adopt appropriate procedures which would ensure that all applicable deviations were identified and adequately dispositioned to ensure they were either evaluated or passed on to ASCO customers and (2) failed to include an appropriate description of 10 CFR Part 21 in its posted notice in accordance with 10 CFR 21.6(b). (99900369/91-01-01)

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

The provisions of 10 CFR 2.201, require the Automatic Switch Company to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission. However, we note that ASCO performed the corrective action before the NRC exit meeting, and the NRC inspectors reviewed the corrective action and found it satisfactory during this inspection. Therefore, no response to this Notice of Violation is required.

Dated at Rockville, Maryland
this 18th day of February 1992.

ENCLOSURE 2

NOTICE OF NONCONFORMANCE

Automatic Switch Company
Florham Park, New Jersey

Docket No. 99900369/91-01

During an inspection conducted at the Automatic Switch Company (ASCO) facility in Florham Park, New Jersey, on August 26-30, 1991, and October 21-24, 1991, the inspection team from the U.S. Nuclear Regulatory Commission (NRC) determined that certain activities were not conducted in accordance with NRC requirements. These requirements are contractually imposed upon ASCO by purchase orders (POs) from NRC licensees and their designees. The NRC has classified these items, as set forth below, as nonconformances to the requirements of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix B, "Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants," imposed on ASCO by contract and the supplemental requirements of its nuclear utility customers.

A. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part, that applicable regulatory requirements and the design bases for components to which Appendix B applies are correctly translated into specifications, drawings, procedures, and instructions. This criterion also requires that provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.

Section 24, "Design Control," of ASCO's quality assurance manual (QAM) states, in part, that..."[t]he sales department is responsible for reviewing customer contract requirements and forwarding them to the engineering department for translating into specification...the appropriate quality standards are to be specified and included in the design documents by engineering...Design review and verification of design will not normally be performed ..."

1. Contrary to the above, ASCO failed to ensure that it specified the correct flow coefficient (Cv) values for its 3-way, model 206, 208, and 210, direct current (dc) construction solenoid-operated valves (SOVs) in ASCO's technical data catalog NP-1, "ASCO 3 and 4 Way Solenoid Valves for Pilot Control of Diaphragm and Cylinder Operated Valves Used in Nuclear Power Plants," issued March 1978. (99900369/91-01-02)

2. Contrary to the above, ASCO manufacturing procedures MP-C-027, "NOMEX Interlayer Insulated Class H Coils With Leads," dated August 28, 1973, and MP-C-078, "NOMEX or MICA Interlayer Insulated Class H Coils For Use on Valves For Critical Applications," issued April 4, 1986, contained information which was not correctly translated from the design basis. The procedures required that residual flux remain on the soldered/brazed lead wire junction of

the high temperature electrical solenoid valve coil; however, ASCO intended to require the coil manufacturer to remove all residual flux. (99900369/91-01-03)

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

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

Section 3, "Quality Control/Assurance Organization," of ASCO's QAM states, in part, that "[t]he Director of Quality Assurance is responsible for the establishment and mainterance of an efficient (QA) system for control of all phases of product quality... the QA department shall...provide an efficient quality system and level of quality inspection and material conformance from the receipt of raw material to the shipping of the end product, to assure that all material meets all specified requirements...."

Contrary to the above, (1) ASCO failed to establish adequate manufacturing procedures and instructions to ensure that its model 206, 208, 210 and NP SOVs met the minimum Cv flow value published in ASCO's NP-1 nuclear SOV catalog and (2) ASCO failed to establish QC inspection requirements to verify conformance to the minimum designed air flow requirements by either conducting inspections or monitoring the process. (91-01-04)

C. Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B to 10 CFR Part 50 requires, in part, that measures 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, inspection at the contractor or subcontractor and examination of the products upon delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available prior to installation or use. This documentary evidence shall be retained, and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment.

Contrary to the above, although ASCO stated that it had audited Wabash Magnetics (Wabash) ASCO could not provide adequate objective evidence to ensure that Wabash Magnetics, which in 1982 and 1983 manufactured safety-related high temperature electrical solenoid valve coils: (1) effectively controlled the quality of the coils it manufactured for ASCO, regarding technical specifications and requirements and (2) was an appropriate manufacturer for nuclear grade electrical coils based upon source evaluation and selection. (91-01-05)

D. Criterion XVI "Correction Actions," of Appendix B to 10 CFR Part 50 requires, in part, that the identification of significant conditions adverse to quality, the cause of the conditions, and the corrective action be documented and reported to appropriate levels of management.

Section 11, "Quality Control of Test," of ASCO's QAM states, in part, that "[m]anufacturing shall inform both engineering and QC that difficulties are being encountered, and request corrective action to eliminate the problem. An engineering Investigation Report detailing the difficulty shall be completed by engineering."

Contrary to the above, ASCO manufacturing test personnel failed to initiate an engineering investigation, or other appropriate mechanism to document the problem, when they discovered two model 206-381-5RF SOVs with lower disc stroke settings less than specified. This condition is considered significant because [illegible]stricted flow would have resulted if the SOVs had been supplied with [illegible]wer disc stroke settings less than specified. (91-01-06)

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

Contrary to the above, the ASCO QAM did not prescribe provisions to carry out planned and periodic audits to verify that activities affecting quality performed by ASCO quality control personnel comply with the ASCO quality assurance program. (91-01-07)

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 Mr. Leif J. Norrholm, Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance," and should include for each nonconformance: (1) a description of steps that have been, or will be taken, to correct these items, (2) a description of steps that have been, or will be taken to prevent recurrence, and (3) the dates your corrective actions and preventative measures were or will be completed.

Dated at Rockville, Maryland
this 18th day of February 1992.

Enclosure 3

ORGANIZATION:	AUTOMATIC SWITCH COMPANY FLORHAM PARK, NEW JERSEY
REPORT NO.:	99900369/91-01
CORRESPONDENCE ADDRESS:	Automatic Switch Company 50-56 Hanover Road Florham Park, New Jersey 07932
ORGANIZATIONAL CONTACT:	Mr. David Tompsen, QA Director (201) 966-2350
NUCLEAR INDUSTRY ACTIVITY:	Manufactures and supplies 3 and 4-way solenoid operated valves (SOVs), and pressure and temperature switches.
INSPECTION DATES:	August 26-30, 1991, and October 21-24, 1991
TEAM LEADER:	J. J. Petrosino, Team Leader Reactive Inspection Section No. 1 (RIS-1) Vendor Inspection Branch (VIB) Date: [illegible]
OTHER INSPECTORS:	M. Snodderly, RIS-1:VIB K. Naidu, RIS-2:VIB D. Dempsey, Region I, NRC K. Sullivan, Brookhaven National Laboratory T. Tinkel, Brookhaven National Laboratory
APPROVED BY:	Uldis Potapovs, Section Chief, RIS-1, VIB, Division of Reactor Inspection and Safeguards Date: 2-13-92
INSPECTION BASES:	Appendix B to 10 CFR Part 50, and 10 CFR Part 21
INSPECTION SCOPE:	To review five recently identified problems with the operation of ASCO solenoid valves at various nuclear power plants.
PLANT SITE APPLICABILITY:	Multiple.

1 INSPECTION SUMMARY

1.1 Violation

1.1.1 Contrary to Title 10 of the Code of Federal Regulations, Part 21 (10 CFR Part 21), the Automatic Switch Company (ASCO) failed to include an appropriate description of 10 CFR Part 21 in its posted notice in accordance with 10 CFR 21.6(b), and the procedure was not adequately established to ensure 10 CFR Part 21.21 was implemented regarding the evaluation of deviations. (99900369/91-01-01)

1.2 Nonconformances

1.2.1 Contrary to Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 and Section 24, "Design Control," of ASCO's quality assurance manual (QAM):

- ASCO failed to ensure that it specified correct quality standards for its flow coefficient (Cv) values for its 3-way, model 206, 208, and 210, direct current (dc) construction SOVs in ASCO's technical data catalog NP-1. The values of the "Cv Flow Factor" that ASCO listed on page 5 of catalog NP-1, for its dc construction SOVs are less conservative than actual Cv values measured in these ASCO SOVs. (99900369/91-01-02)

- ASCO incorrectly translated the design data regarding fabrication methodology delineated in its high temperature coil manufacturing procedures: MP-C-027, "NOMEX Interlayer Insulated Class H Coils With Leads," issued August 28, 1973, and MP-C-078, "NOMEX or MICA Interlayer Insulated Class H Coils For Use On Valves For Critical Applications," issued April 4, 1986. In both procedures, ASCO specified that residual flux was to be left on the soldered or brazed junction of the electrical SOV coil lead wire; however, it was ASCO's intent to have the residual flux removed. (99900369/91-01-03)

1.2.2 Contrary to Criterion V, "Instructions, Procedures, and Drawings," and Criterion X, "Inspection," of Appendix B to 10 CFR Part 50, and Section 3, "Quality Control/Assurance Organization," of ASCO's QAM, ASCO failed to establish adequate procedures or instructions to ensure that its model 206, 208, 210, and NP SOVs met the minimum Cv published in ASCO's NP-1 nuclear SOV catalog, and failed to establish an inspection program to verify conformance to the minimum Cv requirements by either conducting inspections or monitoring the manufacturing process. (9990369/91-01-04)

1.2.3 Contrary to Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B to 10 CFR Part 50, ASCO could not provide adequate objective evidence to ensure that its safety-related high temperature electrical SOV coil manufacturer in 1982-1983, Wabash Magnetics: (99900369/91-01-05)

* effectively controlled the quality of the coils manufactured for ASCO regarding the technical specifications and requirements; and
* was an appropriate manufacturer for nuclear grade electrical coils based upon source evaluation or selection.

1.2.4 Contrary to Criterion XVI, "Corrective Actions," of Appendix B to 10 CFR Part 50 and Section II, "Quality Control and Test," of ASCO's QAM, ASCO failed to initiate an engineering investigation, or other appropriate mechanism to document a nonconformance identified in two model 206-381-5RF SOVs that were being tested in ASCO's laboratory. (99900369/91-01-06)

1.2.5 Contrary to Criterion XVIII, "Audits," of Appendix B to 10 CFR Part 50, the ASCO QAM did not prescribe provisions to carry out planned and periodic audits to verify that QC activities affecting quality comply with ASCO's quality assurance program. (99900369/91-01-07)

1.3 Unresolved/Open Items

1.3.1 On June 30, 1991 the Carolina Power and Light Company (CP&L) transmitted a 10 CFR Part 21 letter to the NRC regarding ASCO L206-832 SOVs that did not operate as required at its Brunswick nuclear power plant. This problem prevented two reactor containment primary isolation valves from closing as required. The NRC inspectors reviewed this issue as discussed in Section 3.5.1 below. However, ASCO had not concluded its evaluation of the matter. Pending review of ASCO's final evaluation of this matter, this is identified as open item 9990369/91-01-08.

1.3.2 On April 20, 1991, Arkansas Nuclear One (ANO) discovered that the electrical coil on an ASCO safety-related SOV, model NPL-8321A6V, which was used on ANO's main steam isolation valve (MSIV) 2CV-1060-2 failed. The NRC inspectors reviewed this issue as discussed in section 3.5.2 below. However, ASCO had not concluded its evaluation of the matter. Pending review of ASCO's final evaluation of this matter, this is identified as open item 9990369/91-01-09.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Violations

(CLOSED) Violation 88-01-01 identified ASCO's failure to evaluate numerous deviations for reportability or to inform applicable customers. The NRC inspector reviewed the ASCO corrective action for this violation and as discussed below, found ASCO's actions satisfactory. Therefore, violation 88-01-01 is closed.

(CLOSED) Violation 88-01-02 identified that ASCO failed to establish an adequate procedure as required by 10 CFR Part 21. The NRC inspector reviewed ASCO Procedure IP-1-081, revision D, issued June 23, 1989, "Nuclear Products -

Procedure for Reporting Non-Conformities (Safety Related 10 CFR Part 21)." Although ASCO's corrective action was inadequate, violation 88-01-02 is closed because ASCO's corrective action for violation 91-01-01 should appropriately address the NRC concern. Therefore, violation 88-01-02 is closed.

2.2 Nonconformances

(CLOSED) Nonconformance 88-01-03 identified a potential for a lack of adequate independence from cost and schedule when opposed to nuclear plant safety considerations. ASCO has changed its organizational chart, and in ASCO's November 11, 1988, response letter to Inspection Report 99900369/88-01, ASCO stated that its senior management believes that the QA Director has sufficient independence to achieve results consistent with the intent of Appendix B to 10 CFR Part 50. Therefore, this matter is closed.

(CLOSED) Nonconformance 88-01-04 found that ASCO manufacturing personnel were allowed to modify solenoid valve fabrication procedures and implement the product changes without the requirement of a review of the change for technical adequacy by ASCO engineering staff prior to implementation. The NRC inspector's review of this issue concludes that ASCO now requires manufacturing to receive engineering approval prior to implementation of product modifications. Therefore, this matter is closed.

(CLOSED) Nonconformance 88-01-05 identified a failure to establish adequate measures to control ASCO's product nonconformances to ensure that the issues are properly identified and processed. The inspectors found that corrective action delineated by ASCO in its November 11, 1989, response letter was acceptable. The NRC inspectors reviewed ASCO's method for identifying product nonconformances and found it to be satisfactory. Therefore, this nonconformance is closed.

2.3 Unresolved Items

(OPEN) Unresolved item 88-01-06 states that ASCO may have failed to adequately evaluate a potentially reportable extrusion phenomenon that was observed during its 1982 environmental qualification testing activities for its "Tri-Point" pressure switches. ASCO did not determine the applicability of the phenomenon to other nuclear power plant facilities. This issue was not discussed with ASCO personnel. Therefore, this issue will remain open.

(CLOSED) Unresolved item 88-01-07 stated that ASCO engineering design change engineering report (ER) 86154 which incorporated a strainer in the inlet flow path of an ASCO NP 8320 series does not indicate whether an engineering review for technical adequacy was performed. The inspectors discussed this matter further with ASCO personnel during this inspection and found that ASCO performed SOV qualification testing of several SOVs, including NP 8320. ASCO states that each of the SOVs tested contained the type of strainer as shown on ER 86154. Therefore, this item is closed based on the ASCO test that was reported in AQR-67368, issued March 2, 1982, "Report on Qualification of ASCO NP-1 Solenoid Valves for Safety-Related Application in Nuclear Power Generating Stations."

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

On August 26, 1991, the NRC inspectors discussed the scope of the inspection with Automatic Switch Company representatives from engineering, service, manufacturing, and quality assurance. At the conclusion of this entrance meeting, the NRC inspection team toured the ASCO valve manufacturing facilities. During the interim exit meeting on August 30, 1991, and the exit meeting on October 24, 1991, the NRC team leader summarized to the ASCO staff the team's conclusions, findings, and concerns identified during the inspection.

3.2 10 CFR Part 21 Issues

3.2.1 Section 21.21, "Notification of Failure to Comply or Existence of a Defect," requires that each individual, corporation, partnership, or other entity subject to 10 CFR Part 21 adopt procedures that will appropriately provide for evaluating deviations, or informing the licensee or purchaser of the deviation in order that the licensee or purchaser may cause the deviation to be evaluated. Contrary to the requirement ASCO failed to adopt procedures that would ensure that deviations were either evaluated or passed on to the licensee.

The NRC inspector reviewed the procedure that was adopted by ASCO and concluded that the procedure did not supply sufficient instructions to the employees to ensure that they would report all deviations to their superiors. According to the procedure ASCO employees would be required to determine whether a particular deviation affected the safety functions of a nuclear power plant before they would be required to report a particular deviation. Therefore, the NRC inspector concluded that this statement could lead an employee to not report problems because the statement required the employee to first determine whether the problem "could potentially affect the safety functions of...[a] nuclear power plant." The NRC inspector did not believe that this was an appropriate responsibility for employees, since the corporation should determine the applicability of nonconformances to licensees. Therefore, the inspectors identified this issue as Violation 99900369/91-01-01.

3.2.2 Section 21.6, "Posting requirements," of 10 CFR Part 21 requires that corporations either: post 10 CFR Part 21, Section 206 of the Energy Reorganization Act of 1974 (Section 206), and procedures adopted by the corporation to implement Section 206; or that the corporation, in addition to posting Section 206, also post a notice which describes 10 CFR Part 21 and its adopted procedures. The posted notice also must include the name of the individual to whom reports can be made and must state where the adopted procedures can be reviewed. The NRC inspectors asked to be shown where the 10 CFR Part 21 required postings were displayed, and were escorted to several locations where a copy of the same posting was displayed. The inspectors found that the posted notice addressed the latter method of posting described above and generally met the requirements except for the following statement that summarized the ASCO employee's responsibility: "Any defects or noncompliances which could potentially affect the safety functions of the nuclear power plant should be reported to [the ASCO management representative's name]." The inspectors also observed that Section 206 and Section 21.1, "Purpose," of 10 CFR Part 21, which were expressed on the ASCO notice were almost illegible. Violation 99900369/91-01-01 was identified in this area.

However, ASCO took immediate corrective action to develop a new notice for its bulletin boards that addressed all of the NRC concerns. ASCO's corrective action was accomplished prior to the exit meeting and the NRC inspector reviewed and found the new notice satisfactory.

3.2.3 In accordance with Section 21.21 of 10 CFR Part 21, an entity is required to adopt procedures that ensure that it either performs an evaluation of a deviation to safety-related procurement documents, or that the individual informs the licensee or purchaser of the deviation so that the licensee or purchaser can cause the deviation to be evaluated. The scope of this inspection included the review of the two deviations reported to the NRC staff as discussed in section 1.3 above. The NRC inspectors found the deviations to be currently under review in accordance with 10 CFR Part 21; thus, the two issues were characterized as open items.

However, the NRC inspectors and ASCO staff concluded that ASCO had developed a reasonable amount of data on the SOV Cv issue discussed in Section 3.3 below. This data indicated that ASCO should send an interim notification to their customers. The majority of the data was generated by ASCO as the result of BG&E contacting ASCO about the slow valve stroke times that it was experiencing with safety-related butterfly valves.

3.3 Model 206, 208 and 210 Series SOV's

3.3.1 In the 1970s, ASCO performed testing of their nuclear line of SOVs. ASCO stated that these tests included qualification testing of prototype units. ASCO translated the data from these tests into its SOV design specifications and data, such as ASCO's NP-1 nuclear SOV catalog. On page 5 of NP-1, ASCO states the "Cv Flow Factor" for each different model and orifice size for both "AC construction" and "DC construction" SOVs.

On November 13, 1989, Baltimore Gas & Electric Company (BG&E) (PO 30051-GX) obtained 20 ASCO model 206-381-5RF SOVs for use in safety-related applications. BG&E found that each of these normally closed SOVs exhibited a lower-than-expected flow rate in the de-energized position which resulted in a significant increase in valve stroke times because these SOVs provide instrument air to the valve being stroke tested. BG&E observed flow testing of one of the unsatisfactory SOVs on December 20, 1990. A flow coefficient (Cv) of 0.05 was measured instead of the expected 0.39 as published on page 5 of ASCO's NP-1 Catalog. In a January 22, 1991, facsimile to BG&E, ASCO's supervisor of valve service stated that the restricted flow was caused by a lower disc stroke that was less than specified. Engineering Data Sheet 165, "Bulletin 8300 & 8302 P-37 Upper and Lower Disc Stroke Setting for Resilient Seats," Change Letter A, May 26, 1977, specified the lower stroke to be 0.031-inch (+0.005, -0.000-inch). ASCO also advised BG&E to return the 20 SOVs to be modified.

To replace the questionable SOVs obtained on BG&E PO 30051-GX, BG&E obtained 42 more ASCO model 206-381-5RF SOVs on PO 36852-GX, February 11, 1991. ASCO supplied a certificate of compliance on February 22, 1991, that stated that the work had been completed in accordance with the requirements on the PO and manufactured in accordance with the requirements of revision 5 of their Appendix B to 10 CFR Part 50 QA manual, November 1, 1989. ASCO Catalog NP-1 stated that model 206-381-5RF and all other SOVs with resilient seats and a 1/4-inch orifice on page 5 of the catalog have a Cv of 0.39. However, the NRC inspectors identified that ASCO did not test for minimum flow or verify Cv values before June of 1991 when it instituted a new informal minimum flow test policy. ASCO tests the performance of its SOVs in accordance with Valve Engineering Procedure "TP-3-046-Nuclear," June 29, 1991. The parameters tested included: external leakage, high and low pressure seat leakage, noise, and operation from maximum to minimum rated pressure. However, the test did not include minimum flow or Cv.

These flow problems prompted BG&E to send an auditor to witness Cv verification testing from April 30, 1991, through May 1, 1991, for 7 of the 42 supplied SOVs. Measured Cv values ranged from 0.17 to 0.30. Consequently, ASCO replaced the SOVs' lower stems and the lower disc stroke was reset using the greatest tolerance to allow for maximum flow. The Cv values were remeasured and ranged from 0.24 to 0.38 which is still less than the 0.39 value given in the NP-1 Catalog. The corrective action BG&E has taken for the other 35 SOVs was not identified by the NRC inspectors.

The ASCO Product Engineering Manager stated that the main reason for the discrepancy between the catalog Cv value and the measured Cv value is that the catalog Cv values were determined in the 1970's by verifying the Cv for SOVs of ac construction, while the 206-381 and 208-448 model SOVs are of a dc construction. In general, SOVs of dc construction have lower Cv values than those of ac construction. This was confirmed by Cv verification testing performed by ASCO on June 28, 1991, for two model 206-381-5RF, 1/4-inch orifice, and dc construction. These SOVs were taken from stock and had passed final performance testing and had been accepted by quality control (QC) personnel. These SOVs have an upper stroke requirement of 0.051 (+.005-0.000) inch and a lower stroke of 0.031(+0.005-0.000) inch. ASCO used the SOVs for two separate Cv verification tests; one with the maximum upper and lower stroke (0.056 and 0.036 inch respectively) and the other with the minimum upper and lower stroke (0.051 and 0.031 inch respectively). After the strokes were reset the flow testing showed that Cv values ranged from 0.28 to 0.34 for the maximum stroke setting and 0.27 to 0.30 for the minimum stroke setting for flow through the exhaust path in the de-energized position. Nonconformance 99900369/91-01-02 was identified in this area. ASCO committed to notify its customers of the correct dc construction Cv values and to publish this in the next revision of its NP-1 Catalog.

The inspectors noted from the test data sheets that Cv values were considerably greater for flow through the pressure path than for flow through the exhaust path. ASCO personnel said that separate Cv values would be given for the universal, exhaust, and pressure paths in the next revision of the NP-1 Catalog.

The difference between ac and dc construction accounts for the difference between the 0.39 Cv value published in the NP-1 Catalog and the Cv values measured for BG&E after resetting the lower disc stroke. However, it would not have caused the four of the seven Cv values measured for BG&E, before resetting the lower disc stroke, to be considerably below the minimum expected Cv values measured by ASCO on June 28, 1991. ASCO performed this testing to verify the minimum Cv values for model 206-381-5RF SOVs. The Cv values measured for BG&E increased significantly after the lower disc strokes were reset. The inspectors concluded that the lower disc stroke was less than specified and resulted in restricted flow. The upper and lower disc stroke dimensions for model 206, 208, and 210 SOVs are set by hand. The setting operation involves ASCO personnel grinding the stem to within 0.005 inch of the proper stroke. QC personnel do not oversee this delicate operation frequently. Nor did ASCO verify the minimum flow as part of the final manufacturing performance testing or perform any random or periodic QA sample flow testing for final acceptance.

The inspectors observed that both valve data sheets used during Cv verification testing on June 28, 1991, indicated that before the stroke settings were reset by testing personnel the as-built lower stroke settings were 0.026 and 0.024 inch instead of the required 0.031 inch which was less than Engineering Data Sheet 165 allows. These SOVs were taken from stock and had passed final performance testing and had been accepted by quality control personnel. These as-built settings were set by manufacturing personnel and could have restricted the flow. ASCO testing personnel failed to initiate an engineering investigation, as described in ASCO Procedure No. EDP-13, "Request for Engineering Investigation or Change and Issuing of Investigation Reports," Change Letter T, May 20, 1960, or other appropriate mechanism to evaluate the deviation in accordance with the requirements of Criterion XVI of Appendix B to 10 CFR Part 50. Nonconformance 91-01-06 was identified in this area.

Two separate POs supplied SOVs with restricted flows. The lower disc stroke setting on two model 206-381-5RF SOVs that were drawn from stock were below tolerance. A similar process is used to set disc stroke settings for model 206, 208, and 210 SOVs. These facts led the inspectors to believe that other model 206, 208, and 210 SOVs may have been supplied with restricted flow. Consequently, ASCO committed to inform its customers of this deviation in accordance with 10 CFR Part 21. The inspectors observed that ASCO had not established procedures for this testing, therefore; Nonconformance 91-01-04 was identified in this area.

Licensees that have received model 206, 208, and 210 SOVs prior to June 1991 may not be able to take credit for published Cv values without verifying them based on the tests that ASCO performed which showed that four of the seven SOVs provided under BG&E PO 36852-GX had a Cv value that was lower than expected even after compensating for the Cv discrepancy in the NP-1 Catalog. ASCO committed to notify its customers of re-verified Cv values in the universal, normally opened, and normally closed positions for model 206, 208, and 210 SOVs. ASCO's notification letter was sent out to all licensees in the third week of October, 1991.

3.3.2 The stated purpose of ASCO's manufacturing procedure MP-C-027 is to establish the minimum acceptance construction standards for Class "H," 180°C layer insulated coils. While reviewing this procedure, associated with the discussion in section 3.5.2 below, the NRC inspector identified that item 3B, listed under "Special Instructions, Manufacturing Techniques," stated that extra care must be taken to avoid residual solder or flux. However the inspector found that contrary to this note, item 10, under "Soldering Flux," contained a note which stated "do not remove flux residue..." Both of these statements were also found to be included in ASCO's current manufacturing procedure MP-C-078. The inspector concluded and ASCO staff agreed that the methodology for the design basis would require that the flux be removed to prevent corrosion. However, ASCO incorrectly translated the design requirement to leave the flux on instead of removing it as required. Nonconformance 91-01-03 was identified in this area.

3.4 Cv Flow Measurement

The first issue reviewed as discussed in section 3.3.1 above, identified that ASCO certified that its model 206, 208, and 210 SOVs were in accordance with the procurement documents. However, ASCO did not verify for, or inspect to ensure that it meets the Cv values published on page 5 of the NP-1 catalog, under dc construction. Therefore, Nonconformance 91-01-04 was identified in this area.

3.5 Model L206 Series SOVs

3.5.1 Sticking of ASCO Model L206-832 SOVs was the second issue reviewed by the NRC staff. On June 30, 1991, the Carolina Power and Light Company (CP&L) transmitted a 10 CFR Part 21 report to the NRC regarding ASCO model L206-832-3RVF SOVs at its Brunswick Steam Electric Plant (BSEP). The CP&L letter explained that on June 30, 1991, two ASCO L206-832-3RVF SOVs failed to change state in a manner that resulted in the associated BSEP primary containment isolation valves not closing. BSEP's root cause analysis report 91-0001, July 28, 1991, mainly attributed the failure to gelling of the internal Dow Corning silicone lubricant compound 550 (DC-550), or foreign matter, or both.

One of the supporting documents to the root cause analysis report was Failure Prevention Incorporated (FPI) Report No. 91-163, dated July 6, 1991. In this report CP&L concluded that the root cause of the failure was gelling of DC 550, or foreign matter or both. The report stated: "The temperature for similar solenoid cores operated in similar room temperatures and conditions is about 400°F. Gelling should take place in a year of service." Information provided by Dow Corning entitled, "Information about Silicone Fluids," copyright 1976, stated that the onset of gelling would occur after 14 months at 392°F. Dow Corning determined this by heating a sample in an air-circulating oven. The BSEP report gave the (maximum) normal ambient temperature as 104°F.

The NRC inspector also reviewed and discussed testing performed by ASCO which showed that in an ambient environment of 99°F a maximum temperature of 305°F was seen in the area where sticking occurred. Based on a time to gel Arrhenius plot for DC 550, which is part of the FPI report, at 292°F gelling will occur in 57 years. Testing performed by the Harris E & E Center for BSEP's evaluation measured the temperature of the Top Core Sub Assembly of a similar solenoid valve to be 295°F at 122 Vac at an ambient of 105°F. Although gelling of DC 550 is a major concern at 400°F, a properly maintained solenoid valve should not reach temperature greater than 305°F at 120 Vac and an ambient of 105°F.

Regarding the results of FPI Report 91-163, ASCO stated that a voltage above 120 Vac or any air gap in the magnetic field would increase the coil temperature. Inserting a thermocouple into the solenoid's magnetic field to measure coil temperature could increase the coil temperature. These are some conditions that could cause the difference between the temperature measured by FPI and ASCO. Contaminants such as compressor oil have been known to solidify over time when exposed to temperatures of approximately 300°F. Contaminants at BSEP could come from the Service Air system, which was used to functionally test the subject valves before installation. The Service Air system is filtered for particulate but not for oil. FPI's report did not identify any contaminants that would have been induced from the air.

The NRC inspectors reviewed these reports and could not find an underlying root cause. The inspectors also could not find a reason for the significant difference in the maximum operating coil temperature measured by the different parties. ASCO and licensee representatives were conducting discussions to plan for joint tests to determine the actual coil temperature and to obtain a better root cause analysis for these SOVs.

3.5.2 High-temperature electrical coil failure was the third issue reviewed by the NRC staff. On April 29, 1991, ASCO transmitted a letter to the NRC regarding a possible manufacturing problem with certain coils used on ASCO nuclear qualified valves. The ASCO letter concludes that a good possibility exists that one lot of 99 high-temperature electrical coils (hi-temp coils) manufactured for ASCO in 1982 would likely have problems. Therefore, ASCO committed to inform its customers in accordance with 10 CFR Part 21 and offer to replace the product at no charge.

ASCO issued the letter upon learning that, on April 20, 1991, a high-temperature coil failed at Arkansas Power & Light (AP&L) Arkansas Nuclear One (ANO). This coil was from a lot of 99 produced in 1979, from which other failures had been identified by ANO before April of 1991. On April 20, 1991, ANO-2 discovered the failure of an ASCO model NPL 8321A6V SOV used on Main Steam Isolation Valve (MSIV) 2CV-1060-2. The ASCO SOVs were procured by ANO through a distributor, Carlton-Bates Co., in accordance with ANO PO No. 197532, April 20, 1989, which imposed the requirements of 10 CFR Part 21 and

Appendix B to 10 CFR 50. ANO attributed the SOV failure to a failure of its operating coil and during primary system heatup ANO-2 had experienced a similar failure of another ASCO Model NPL 8321A6V SOV. Initial examination by ANO-2 indicated that both coils appeared to have burn spots on the exterior of the coil where the coil leads attach to the coil. ANO-2 notified ASCO of the failures and replaced the two SOVs which had recently failed and the operating coils of two SOVs which had not yet failed. The licensee sent all four units (two SOVs and two coils) to ASCO for additional analysis. The operating coils of all the units were identified as having ASCO Part No. 220339-001G.

The results of a preliminary ASCO investigation were found to be documented in ASCO interoffice correspondence dated April 22, 1991, from G. C. Laubenstein, to File. This investigation found that the two coils had failed open and that three of the four coils had lead to coil magnet wire solder joints with "what appeared to be heavy corrosion." ASCO found that the three coils exhibiting evidence of corrosion at their soldered junctions were part of a 99 piece lot of coils manufactured by Wabash Magnetics of Huntington, Indiana, during the second quarter of 1982 (Date Code B82). The one coil that did not exhibit corrosion was produced by a different manufacturer, Altron Incorporated. The coils manufactured by Wabash during this period were fabricated according to information contained in ASCO design drawing GV-220-339 and ASCO manufacturing procedure MP-C-027.

ASCO reviewed past records and found that in 1985 Brunswick Steam Electric Plant (BSEP) had experienced similar failures of coils having ASCO part number 220339-1G at BSEP. The ASCO investigation into that incident concluded that the failures were also related to corrosion of the magnet wire. Additionally, these coils were all found to be part of the same 99 piece lot of coils manufactured by Wabash Magnetics during the second quarter of 1982.

On April 29, 1991, ASCO informed NRC of a possible manufacturing problem involving one lot of 99 coils used on ASCO nuclear qualified valves. In its letter ASCO stated that although it is continuing to investigate this problem, it has taken immediate action to identify and notify purchasers of coils, or of products manufactured with coils, from the suspect lot. By letter of April 29, 1991, ASCO had forwarded the three ANO-2 coils to Wabash (the two coils which had failed in service at ANO-2 and the one coil which had not yet failed but also exhibited signs of corrosion at its coil magnet wire to lead wire termination) for additional examination. In this letter ASCO stated that according to its customer (ANO-2) the coils are normally operated at 125 volts dc but occasionally may see 130 volts at an ambient temperature of 120°F (these operating parameters are within the design limits of the coils: 140 volts at an ambient temperature of 140°F). Additionally, ASCO stated that their examination of the two failed coils revealed what appeared to be a burn mark between the coil lead and the final turn of the coil and that "resistance checks made throughout the coil revealed that the failure occurred at this burn mark." ASCO requested Wabash to perform a failure analysis to determine

"the coil cause failure, specifically the reasons for the burning of the coil insulation at the lead wire/coil wire junction."

By letter of May 3, 1991, from T. Hunt (Wabash) to S. Casadevall (ASCO), Wabash supplied ASCO with the results of its investigation. In that investigation Wabash concluded that both coils appeared to have received excessive current causing the coils to heat up, and that since the area of the coil finish wire would have the least heat dissipation, the excess current would cause the finish wire copper to melt. In this evaluation, Wabash also stated that the leads on the three coils were brazed using sil-fos brazing alloy and that the termination on all three coils showed various degrees of oxidation, but no degradation of the termination could be found.

The inspection team reviewed these investigation reports and concluded that the cause of failure identified by the manufacturer (Wabash) does not appear to support the ASCO position that the coil failures were caused by a manufacturing deficiency. Therefore, the actual root cause of the coil failures remains uncertain. ASCO representatives stated that ASCO is continuing its investigation. At the time of the inspection, ASCO representatives maintained that based on their present level of knowledge, the failures were due to a manufacturing deficiency affecting a single lot of 99 coils manufactured by Wabash in the second quarter of 1982, and their recall of these coils should adequately address the concern.

Before September 1988, ASCO Part No. 220-339 Class H coils were used exclusively in nuclear applications. According to ASCO design drawings (GV-220-339) in effect in 1982 when the SOVs procured by ANO-2 and Brunswick were manufactured: "Use of these coils is restricted to NP valves only", with "NP" being the ASCO designator for its nuclear qualified product line of SOVs. Historically, ASCO has procured GV 220-339 coils from several subtier manufacturers including Altron, Wabash Magnetics, Cycle Transformer, and Five Star (formerly Altron). During the inspection a review of ASCO quality control receiving inspection records was performed to determine the number of Class H 220-339 coils that were manufactured by Wabash Magnetics. The inspectors found that approximately 3200 ASCO Part No. 220-339 coils were manufactured by Wabash. The 3200 coils were procured by ASCO in 6 lots of various sizes over a time period of May 1982 to October 1982, with the majority of coils (5 of the 6 lots) procured in a one month time period of May-June 1982. Based on the short time frame in which Wabash actually manufactured this type of coil for ASCO and the ASCO determination that the coil failures experienced by ANO and Brunswick were due to a manufacturing deficiency, the inspector questioned the ASCO basis for limiting its recall to a single lot of 99 coils. In response, ASCO representatives stated that, in their opinion, coil failures were rare and since both of the failed coils were traced to a single lot, it was concluded that the manufacturing deficiency affected only this lot.

ASCO does not maintain a computerized component failure data base. Rather, reported problems are evaluated and dispositioned by engineering via Engineering Investigation Reports (EIR) which are ultimately filed in several volumes of binders. This method does not appear to provide a readily retrievable record of component failure information. Since problems with components such as coils may be reported, evaluated and categorized as either a valve problem or separately as a coil problem, identifying a specific number of component failures that have occurred over a period of time would require a highly labor intensive effort. The EIRs also may not reflect actual failure rates since all component failures are not necessarily reported by a customer. Therefore, the inspector questioned the basis for ASCO determining that a coil failure is a rare occurrence. ASCO representatives stated that they could not provide any objective evidence to support its conclusion, such as a documented review of existing EIRs, and indicated that its determinations of coil failure [illegible] were principally based on the experience and memory of its nuclear valve ser[illegible] personnel who typically discuss reported problems with customers and initiate the EIR's.

The inspectors found that ASCO did not have a strong technical basis for limiting its recall to a single lot of 99 coils. Thus, the team was concerned that the manufacturing deficiencies which ASCO attributed to be the cause of coil failures experienced by ANO-2 and Brunswick may have also affected the quality of approximately 3100 other ASCO Class H 220-339 coils manufactured by Wabash Magnetics during the same period of time. See open item no. 91-01-08 in Section 1.3 above. A further review of the manufacturing process controls ASCO had in place at the time of the coil's fabrication identified the following:

- ASCO representatives stated that all the Class H, 220-339 coils in question were fabricated by Wabash in accordance with ASCO design specification GV 220-339 and manufacturing procedure MP-C-027. However, ASCO could not provide any objective evidence, such as ASCO procurement documents or Wabash certificates of conformance, to ensure that these requirements were, in fact, passed on and adhered to by Wabash. ASCO representatives stated that since the coils were manufactured in 1982, the "seven year statute of limitations" had expired for maintaining such records. The inspection team informed ASCO that while it was not familiar with this statute, it was aware that the Documentation Requirements of the Arkansas Power and Light Company purchase order (PO No. 197532) requires ASCO to "be capable of verifying the validity of any certifications or reports furnished if so requested by the company at a later date." The apparent lack of ASCO procurement documents to Wabash, and the lack of Wabash certifications of conformance to those procurement documents, appears to place the validity of the ASCO certification to the Arkansas Power and Light PO in question.

- ASCO could not provide any objective evidence necessary to verify that it had performed any audits to verify the quality of the manufacturing methods employed by Wabash. Such audits would provide assurance that Wabash was producing the coils in accordance with ASCO design specifications and procedures.

- In its evaluation of the failed coils returned by ANO-2, of May 3, 1991, Wabash stated that the leads on the three coils were brazed using sil-fos brazing alloy. ASCO manufacturing procedure MP-C-027 specified the use of Handy & Harmon Easy Flow #45 solder and did not identify sil-fos as an approved material. During the inspection ASCO representatives contacted Wabash to verify the type of solder material used. The ASCO representatives provided the inspector with information that indicates that Wabash has always used a torch process using sil-fos brazing rod to make this connection. It was noted that MP-C-027 does permit the use of alternative solder material provided it is equivalent to Easy Flow #45. At the time of the inspection, however, ASCO could not provide objective evidence necessary to assure that sil-fos is a suitable brazing material for use in this application.

Therefore, based on the above, nonconformance 91-01-03 and 05 were identified in this area.

3.6 Violation 88-01-01

NRC Inspection Report 99900369/88-01 identified that contrary to 10 CFR Part 21, ASCO was aware of numerous examples of deviations, omissions, and/or potential generic problems but failed to either evaluate them for potential reportability or inform the licensees so they could evaluate them for reportability. The NRC inspectors found in 1988 that ASCO had typically documented these problems on EIRs and did not perform an evaluation pursuant to 10 CFR Part 21. ASCO subsequently informed the NRC as part of its corrective action that it had identified approximately 650 EIRs, and approximately 515 were eliminated from further evaluation after an initial screening of all 650 EIRs. Therefore, the NRC inspectors reviewed the methodology employed by ASCO to screen and disposition the EIRs, and reviewed several to determine acceptability of ASCO's corrective action during this inspection.

The NRC inspectors concluded that the methodology employed by ASCO to review engineering documentation was satisfactory.

3.7 Observation of Dow Corning 550 Application

Associated with the second issue discussed above in Section 3.5, the NRC staff also evaluated some substances observed in the manufacturing area. The NRC is reviewing various reports by NRC licensees that some ASCO solenoid valves have been found with a "sticky" substance on the upper surface of the solenoid core. In a few instances licensees speculated that the sticky substance prevented proper valve operation. The source of the sticky substance has not been positively established. Some licensees contend the substance may be introduced during manufacturing. The reason for this conclusion is that in one case infrared analysis identified the substance as a lubricant with a composition consistent with DOW Corning 550 or one of the Neolube products. ASCO acknowledges that Dow Corning 550 is applied to the end of the core during valve assembly to eliminate core chattering during initial operation. However, ASCO contends that Dow Corning 550 is not sticky nor will it become sticky at normal operating temperatures of the valves. ASCO believes that the sticky substance may be the result of licensee maintenance activities or caused by contamination from licensee systems after installation.

During the tour of the ASCO manufacturing plant, the NRC inspection team observed a gelatinous material on one of the SOV assembly tables. The gelatinous material appeared to be a petroleum based jelly type product.

Therefore, NRC inspectors subsequently observed assembly operations to determine how this substance is used during solenoid valve assembly and whether it might be the source of the sticky substance. In response to this request, the inspectors visited the nuclear valve assembly area to observe the valve assembly process. It was found that valves for nuclear valve order 295011001 were being assembled. This was an order for 15 type 8344A070 ASCO solenoid valves. The ASCO assembler was using ASCO procedure AP-NP8344A-3, Change Letter L. The jelly substance observed during the initial plant tour was determined to be Nyogel. Two types of Nyogel were on the assembly table: Nyogel 775A (clear appearance) and Nyogel 775B (red color). Dow Corning 550 (a clear fluid with a visual appearance similar to water) was also on the table. The assembly procedure calls for certain seals and O-rings to be lubricated with Nyogel 775A and some with Nyogel 775B. None of the lubricated items are located at the upper end of the solenoid core. The lubrication of parts with Nyogel was observed and it was noted that only sparing quantities of the lubricant were applied. The assembly procedure calls for the upper end of the solenoid core to be lubricated with Dow Corning 550. This is accomplished by pouring a small quantity of the Dow Corning 550 fluid onto a sponge material located in a small dish. The core is lubricated with Dow Corning 550 by pressing the end of the core onto the wetted sponge. By analogy, the operation is similar to wetting a postage stamp on a sponge. The amount of Dow Corning 550 applied by this method is small and is similar to the amount of water that would wet the postage stamp in this analogy. The ASCO assembler appeared to be experienced and knowledgeable about the assembly work. She is one of six qualified assemblers. She indicated that she does most of the nuclear assembly work. The possibility of Nyogel contamination of the solenoid core end cannot be positively ruled out solely on the observations made during this inspection. However, no problems were observed during this particular valve assembly job. Further, the inspectors conclude that Nyogel contamination of the solenoid core end is unlikely if the procedures and practices observed during this inspection are employed and ensured by ASCO at other times.

3.8 <u>Review of SOV problems at the Perry Nuclear Power Plant (PNPP)</u>

The fourth issue reviewed concerned slower than normal response time of a PNPP safety-related system. The inspectors conducted this review to examine the procurement documents for the solenoid operated valves (SOVs) procured by the General Electric Company, Nuclear Energy (GENE), San Jose, California, and supplied to PNPP and the relevant requirements of the ASCO QA program, and to determine if ASCO had implemented its QA program adequately to ensure that ASCO SOVs would reliably perform their intended functions.

3.8.1 On December 14, 1990, PNPP informed the NRC pursuant to the requirements in 10 CFR Part 21, that six ASCO Part HV 176-186-1, Model EP-139, 3-way SOVs malfunctioned on July 27, and October 28, 1990. These SOVs were used as scram solenoid pilot valves (SSPVs). These malfunctions caused the associated control rods to insert slower than the technical specification limit. PNPP identified the six SOVs as part of ASCO's production lot F6119A of 100 SOVs. GENE supplied PNPP and the Hope Creek Generating Station with 70 and 30 SOVs, respectively, from ASCO's lot F6119A. The licensee for PNPP had installed 41 of the 70 SOVs when this problem occurred. The licensee replaced all 41 with ASCO SOVs from different production lots.

On October 6, 1991, the licensee observed three control rods insert (scram) slower than normal during the performance of a scheduled surveillance. The licensee determined that the three SSPVs that caused the associated control rods to operate sluggishly were from manufacturing lot 184010001. After placing the plant in hot shutdown on October 6, 1991, the licensee replaced 49 SSPVs from lot 184010001, and resumed operations after successfully testing the affected control rods.

3.8.2 The inspectors reviewed GENE POs 205-90F620, 205-90F737, and 205-90F760 issued to ASCO for the supply of 3-way, dual acting HV-176-816-1 type ASCO SOVs to PNPP. GENE identified the same SOVs by its Drawing 922D138P001. GENE's POs to ASCO required compliance to 10 CFR Part 21 and GENE's quality assurance requirements (QAR) stated in QAR No. 1, Revision 9, August 23, 1990. GENE PO 205-90F760 required ASCO to refurbish 51 SOVs that were supplied by PNPP.

The inspectors reviewed the ASCO test reports documenting the results for the tests conducted on SOVs in accordance with ASCO Test Procedure TP-2-075, "HVA-176-816 (Quick Exhaust Valve For General Electric)," and observed that the valves successfully met all the attributes. The inspectors determined that all the GENE POs required ASCO to comply to GENE's quality requirements stated in QAR No.1, and the reporting requirements in 10 CFR Part 21. The POs required ASCO to provide information on the useful shelf life of the elastomers used in the SOV assemblies.

3.8.3 The inspectors, accompanied by an ASCO QA representative, toured the ASCO facilities and observed core assemblies for SOVs being assembled, SOVs intended for nuclear power plants being tested, and SOVs receiving final QC inspections. The calibration stickers indicated that ASCO had calibrated the instruments and gages being used for inspections and tests. The inspectors observed that ASCO QC personnel performed the final inspections on SOVs designated for safety-related use. The ASCO QA representative stated that the ASCO QA department does not audit the final inspections performed by QC personnel. The inspectors determined that the current ASCO QA manual does not have provisions for performing planned and periodic audits to verify that activities affecting quality performed by ASCO quality control personnel comply with the ASCO QA program.

The inspectors identified this item as nonconformance 91-01-06.

3.8.4 On October 23, 1991, the inspectors, along with representatives from PNPP and GENE, witnessed routine and special tests performed on ASCO SSPVs that were identified to have caused slow control rod motion at PNPP. ASCO followed Test Procedure 2-075, "HVA-176-816 (Quick Exhaust Valve For General Electric)," to test the three SOVs. ASCO personnel also measured the response time of the SOVs and determined that all three SOVs met the response time requirements stated in Paragraph 2.4.6.a of GENE Procedure 23A1443, Revision 2.

3.8.5 On October 23, 1991, the inspectors accompanied by representatives of PNPP and GENE, witnessed the disassembly of the three SOVs. Disassembly revealed particles of what appeared to be sealant (pipe dope) in the threads of all three SOVs. Pipe dope was also observed on the solenoid core assembly and on the Viton seat assembly of SOV 74699D-41. A grey powdery substance was observed on the underside of the lower disc holder in SOV 74699D-41 and traces of pipe dope inside the valve body and in the orifices. In addition a white powdery ring was observed in the inlet port under the diaphragm. The GENE representative stated that GENE would analyze the foreign material deposited in the SOV at its laboratory in San Jose. On disassembling the other two SOVs, sealant material, similar to the material observed in the first SOV, was observed on the threads.

3.9 Review of ASCO Pilot Assembly Sub-Assembly Kit Problems at the Millstone Unit 1, Nuclear Power Plant (Millstone)

The fifth and last issue reviewed by NRC staff concerned GENE supplied ASCO rebuild kits. The inspectors conducted this review to examine the requirements in the procurement documents regarding the GENE purchase of the ASCO rebuild kits supplied to the licensee for the Millstone Nuclear Power Station and to determine if ASCO had implemented its QA program adequately to ensure that the rebuild kits would contain the correct sub-assemblies.

3.9.1 On August 16, 1991, Millstone Unit 1 personnel reported to the NRC that during the control rod scram testing, two control rods failed to meet the minimum time for scram insertion. The licensee for Millstone Unit 1 issued Licensee Event Report 91-025 to document this event. Millstone personnel determined that incorrect core assemblies in the SSPVs caused the control rods to operate slowly. The licensee for Millstone had purchased 150 ASCO rebuild kits from GENE and used several of them to rebuild its SSPVs. ASCO identified the kit as Part 204-139, and GE identified the same as Part 317A6_68P001. ASCO's design requires the diameter of the hole in the core assembly to be 0.156 inches. Millstone personnel measured the diameters of the holes in all the 150 core assemblies and determined that the holes in 28 core assemblies had diameters of 0.177 inches.

3.9.2 The NRC inspectors reviewed the PO documents issued by the Northeast Nuclear Energy Company (NNEC) (the licensee for Millstone Unit 1) and GENE, for the purchase of 150 ASCO pilot sub-assembly kits (kits) 28 of which have been identified to have core assemblies with a larger than specified diameter as discussed in Section 3.10.1. The inspectors also reviewed ASCO's manufacturing and inspection records. NNEC issued PO 907158 dated November 24, 1986, to GENE for the supply of 150 ASCO kits identified by P/N 204-139, GENE PN 317A6168P001. The PO identified the kits to be safety-related items and invoked the requirements of 10 CFR Part 21. GENE issued PO 205-87C179 dated March 13, 1987, for 300 kits, GE Drawing 317A6168P001. The PO classified all items as safety-related items subject to the requirements of 10 CFR Part 21. The GENE PO required ASCO to provide the following information with each shipment (1) cure date for elastomers, (2) date of manufacture and (3) date of assembly. The GENE PO stated that

ASCO could not make a partial or complete shipment unless GENE performed a final source inspection. A GENE quality control representative signed the GENE Product Quality Certificate Form (895B).

The NRC inspectors determined that the POs issued by both NNEC to GENE and by GENE to ASCO contained the appropriate quality requirements for safety-related items including those requirements in 10 CFR Part 21.

3.9.3 ASCO used four shop orders (SOs), 02020T, 98329R, 94711R, and 94708R, for assembling the 300 kits for GENE PO 205-87C179. ASCO selected the items assembled in the kits from the production bins and performed QC inspections on them. The inspectors reviewed the following records for each shop order:

- The preliminary and final inspection requirements according to Procedure MP-I-046. These inspections were to be witnessed or verified by QC personnel.

- The check list of QC contract requirements.

- Pre-kit inspection checklist for GE spare parts kits for nuclear use. This document identified the inspection characteristics, the verification methods, inspection plan, and the total quantity to be inspected. The checklist also required the stamps of the ASCO QC inspectors to indicate the identity of the QC inspector who inspected that attribute and the GENE inspector to indicate GENE's approval to ship the kits.

- A GENE product quality certificate (PQC) that provided the following information.

  Supplier Certification: ASCO certified that the products have been manufactured under a controlled QA program and conform to the procurement quality requirements including applicable codes, standards, and specifications as identified in the PO. The ASCO QA manager signed and dated this portion of the certificate.

  GENE Certification: The GE QA department certified that it had reviewed evidence supporting the ASCO certification statement and had found no product nonconformances from the procurement quality requirements. The GENE QA representative signed this portion of the certificate. The inspectors noted that the PQCs did not list any nonconformances.

3.9.4 Focusing on the ASCO kits, the NRC inspectors reviewed the implementation of ASCO's QA program during the processing of purchase orders for safety-related items and the preparation of the manufacturing shop orders with in-process and final inspection requirements. The inspectors determined the following:

- ASCO sales personnel prepared the shop orders with the in-process and final inspection requirements incorporating the PO requirements in accordance with the requirements of ASCO's procedure MP-I-046 and forwarded them to the quality assurance engineering (QAE) department.

- QAE personnel reviewed the shop orders and documented on them the quality control attributes necessary to meet the GE POs. QAE personnel identified MP-I-086 as the pertinent inspection check list for the kits.

- Engineering personnel compiled "pick lists" using the latest revision of the relevant drawing that they received from the drawing department.

- The production department received the checklists prepared by the engineering department, and manufacturing orders listing the bill of materials (BOM). The BOM identified each part and the pertinent manufacturing specifications and procedures. ASCO provided the QC hold points and the test criteria with acceptance and rejection criteria in the documents.

- ASCO assembled the kits to ASCO shop order 94708R for the Millstone order.

3.9.5 The inspectors selectively reviewed the manufacturing and inspection records for the core assembly that is provided with the kits (part HV 65-177), needle (part GV 39-039-1, Zytel, natural color), springs (part FV 51-550), spring retainer (part 65-720), and Buna-N discs (part 60-452-14). The inspectors reviewed the records to determine if the applicable ASCO procedures were implemented. The records reviewed included those for the material used to manufacture the items, the certification provided by the vendors of the material, ASCO's instructions to the subvendors, and the records documenting the acceptability of the attributes verified during receipt inspections to ensure that subvendors supplied items conforming to ASCO's POs and QC inspection records. ASCO personnel would place acceptable items in bins from which they are selected to be included in kits. The ASCO QC personnel re-inspect each item prior to assembling safety-related kits.

The inspectors determined that ASCO had adequately implemented the established QA program in this area and identified no unacceptable findings.

3.9.6 The NRC inspectors reviewed the inspection records to determine if ASCO inspected the core assemblies. The NRC inspectors determined that ASCO had inspected the pre-kit check list for the entire lot of 300 kits under four ASCO supply orders. Each check list indicated the attributes for every item inspected (some attributes were inspected for only a sample of the assemblies) as evidenced by the ASCO QC inspector's stamp. Page 2 of 5 of the pre-kit check list for the 150 kits supplied to Millstone Unit 1 indicates that ASCO QC inspector No. "A49" inspected all the 150 core assemblies identified as part 65-716-002-A on June 6, 1987, and determined them acceptable. The QC inspection included determining the acceptability of the size of the 0.156-inch inner diameter of the core by attempting to insert a 0.170-inch diameter plug gage. The 0.156-inch diameter hole is correct if the QC inspector can not insert the 0.170-inch gage into the 0.156-inch hole. The QC inspector had affixed his stamp to indicate that all 150 core assemblies had the correct size hole. However, as previously noted, Millstone personnel identified 28 assemblies with incorrect holes.

3.10 Review of Nonconformance Material Reports (NMRs)

The NRC inspectors reviewed the implementation of ASCO's quality assurance program regarding nonconformances observed in materials received from vendors and those identified when ASCO-supplied components are returned from nuclear powers stations.

3.10.1 QC personnel write NMRs on QC 652 Forms to document nonconforming observations during receipt inspections and issue typed NMRs on QC 1111 Forms to vendors with the same information. ASCO requires each recipient of an NMR to respond to the NMR by providing the cause of the nonconformance and the corrective action taken to preclude repetition. ASCO QC and Engineering personnel review the responses provided by the vendors to determine if the corrective action taken by the vendor is acceptable. Engineering personnel forward unacceptable vendor responses to the QC department for further action. All NMRs to vendors and their responses are logged and electronically tracked. QC prepares a monthly status report from the information in the Quality Control Action Log and distributes copies to various ASCO management personnel. The QC Action Log is based on the number of the NMRs sent to vendors and responses received. The NRC inspectors observed that in October 1990, ASCO issued 68 NMRs and received 63 responses.

3.10.2 The inspectors reviewed the NMRs completed by ASCO on cores manufactured by the Precision Screw Machine Company (PSMC) and determined that QC personnel issued NMRs to PSMC when they observed core assemblies that did not meet the PO requirements. The NRC inspectors reviewed the responses from PSMC and determined that the actions outlined by PSMC to correct the adverse findings were adequate.

3.10.3 The ASCO Service Department (SD) receives and evaluates information regarding problems identified in the field. ASCO SD also receives material suspected to be defective returned from customers. ASCO SD evaluates the problems and completes an NMR, when necessary. On October 7, 1991, ASCO SD initiated NMR 335 to document that a GENE representative informed ASCO that approximately 30 percent of the core assemblies in the 150 SSPV replacement kits were incorrect. The NMR also stated that ASCO had requested samples from Millstone and that it could not perform a 10 CFR Part 21 evaluation unless it had examined, tested and evaluated the defective assemblies and identified the cause of the nonconforming condition. ASCO QA personnel evaluated NMR 335 and documented the following findings in an internal memorandum dated October 18, 1991:

- ASCO inspected core assemblies from Millstone Unit 1 returned by GENE and determined them to have incorrect hole sizes.

- ASCO reviewed the QC checklists for the kits and observed that ASCO QC personnel had signed the checklist to confirm that the core assemblies had the correct diameter holes even though they were incorrect.

- ASCO identified the recipients of the kits from this manufacturing lot.

ASCO QC revised the Inspection Operation Sheet for the core assembly on October 17, 1991, to have the QC, inspector record certain attributes to preclude this problem from occurring again. The NRC inspectors reviewed ASCO's corrective action that ASCO performed to preclude recurrence and found it to be satisfactory. Subsequent to the inspection, after evaluating the defective assemblies in the kits, ASCO reported to the NRC in a letter of December 2, 1991, that incorrect assemblies were used in kits. Also, on the same date, ASCO informed GENE of this condition.

4 ASCO PERSONNEL CONTACTED

Name	Title
+* J. Hans Kluge	President & CEO
+* David B. Tompsen	Director, Quality Assurance
* Lynne M. Gordon	Director of Operations - Valve
Phil Chrashewsky	Manager, Computer Systems
+* A. Gregory Byrne	Service Manager
+* George Plechy	Supervisor, Quality Assurance Engineering
* Robert Nosal	Quality Control Technician
+* John Shank	Manager, Nuclear Valve Engineering
Rudi Mozosi	Engineer
Mary Spagnuolo	Valve Assembler
+* Stephen Casadevall	Product Engineer
E. J. Vel nekamp	Manager - Training
* Clark Hale	Manager - Operations
+* Terence Johnson	Director of Engineering
George Laubenstein	Product Manager

*Attended the August 30, 1991 exit meeting
+Attended the October 24, 1991 exit meeting

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

FEB 21 1992

Docket No. 99901159/89-01

Ms. Nancy Eining, President
Basic Controls and Valve Company
16770 South West 72nd Avenue
Portland, Oregon 97224

Dear Ms. Eining:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, conducted by Mr. Randy Moist of this office on April 10-11, 1989, and the discussions of his findings with members of your staff at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers. The NRC concern is discussed in detail in NRC Information Notice (IN) 88-48 and Supplements 1 and 2. Areas examined during the NRC inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's ongoing review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901159/89-01

ORGANIZATION: BASIC CONTROLS & VALVE COMPANY
PORTLAND, OREGON

REPORT NO.: 99901159/89-01	INSPECTION DATE: April 10-11, 1989	INSPECTION ON-SITE HOURS: 5

CORRESPONDENCE ADDRESS: Basic Controls and Valve Company
Ms. Nancy Eining, President
16770 SW 72nd Avenue
Portland, Oregon 97224

ORGANIZATIONAL CONTACT: Nancy Eining
TELEPHONE NUMBER: (503) 620-6060

NUCLEAR INDUSTRY ACTIVITY: Basic Controls and Valve Company (BCVC) supplies spare parts from Hammell Dahl to Trojan Nuclear Power Plant and commercial valves to other suppliers and to the commercial market.

ASSIGNED INSPECTOR: Randolph N. Moist — 1 June 89
R. Moist, Reactive Inspection Section No. 2 (RIS-2) — Date

OTHER INSPECTOR(S):

APPROVED BY: CE Ader for — 2 June 89
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50

B. SCOPE: The purpose of this unannounced inspection was to determine if BCVC had purchased any valves from CMA International Incorporated of Vancouver, Washington and to determine if those valves, if any, were supplied to any commercial nuclear power plants.

PLANT SITE APPLICABILITY: None identified

ORGANIZATION: BASIC CONTROLS & VALVE COMPANY
PORTLAND, OREGON

REPORT NO.: 99901153/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed.

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988 discussed a potential problem concerning Vogt 2-inch valves (Vogt figure No SW-13111), which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington (CMA International, Incorporated). Henry Vogt representatives examined the leaking valves at PG&E's Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) the Vogt name was die-stamped on the side of the valve body instead of being forged onto the side of the body. (2) Vogt valves have round bonnet flanges whereas the valves in question have square bonnet flanges, (3) the subject valves have swing gland bolting which is not used by the Henry Vogt Company, and (4) the end-to-end dimensions of the subject valves are shorter than the Vogt SW-13111.

ORGANIZATION: BASIC CONTROLS & VALVE COMPANY
PORTLAND, OREGON

REPORT NO.: 99901159/89-01	INSPECTION RESULTS:	PAGE 3 of 3

2. Discussions with Basic Controls & Valve Company (BCVC):

At present BCVC does not supply any commercial valves to the nuclear industry. BCVC presently has a quote to Portland General Electric for a butterfly valve which would be supplied by C&C Valve Company for use in the Trojan nuclear power plant.

The inspector reviewed several purchase orders to CMA and invoices from CMA and determined that valves were shipped to other valve distributors such as Familian NW and AMFAC Supply, both of Portland, Oregon. Other valves were apparently shipped to the paper pulp industry.

F. PERSONNEL CONTACTED:

Ms Nancy Eining, President, BCVC

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

FEB 21 1992

Docket No. 99901148/89-01/02

Mr. Clifford Ashley
CMA International, Incorporated
601 NE 117th Street
Vancouver, Washington 98668

Dear Mr. Ashley:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspections of your facility at Vancouver, Washington, led by Mr. J. J. Petrosino, of this office on March 28, 1989 and April 25-26, 1989, and the discussions of the team's findings with you at the conclusion of the inspections.

These inspections were performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers. This NRC concern is discussed in detail in NRC Information Notice (IN) 88-48 and its supplements. Areas examined during the NRC inspections and our findings are discussed in the enclosed reports. These inspections consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors. Release of these reports was delayed during NRC's review of nonconforming and substandard vendor products.

Within the scope of these inspections, we found no instance in which you failed to meet NRC requirements. However, several NRC concerns regarding your activities were identified by the inspectors on April 25-26, 1989. These NRC staff concerns were referred to the U.S. Department of Justice. 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 into the NRC's Public Document Room.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901148/89-01/02

ORGANIZATION: CMA INTERNATIONAL, INC.
VANCOUVER, WASHINGTON

REPORT NO.: 99901148/89-01	INSPECTION DATE: March 28, 1989	INSPECTION ON-SITE HOURS: 10

CORRESPONDENCE ADDRESS: CMA International, Incorporated
2424 East 2nd Street
Vancouver, Washington 98661

ORGANIZATIONAL CONTACT: Mr. Clifford Ashley, President
TELEPHONE NUMBER: (206) 696-0818

NUCLEAR INDUSTRY ACTIVITY: CMA provides surplus and refurbished valves, piping, and piping components to distributors who subsequently could supply the CMA products to nuclear power plants.

ASSIGNED INSPECTOR: Edward T Baker for 4/25/89
J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S): Brooks Griffin, Region IV, NRC

APPROVED BY: Edward T Baker 4/25/89
E. T. Baker, Chief RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21.

B. SCOPE: Review CMA records to determine the original equipment manufacturer of the 70 Vogt valves that were supplied to Diablo Canyon.

PLANT SITE APPLICABILITY: Diablo Canyon, Trojan.

ORGANIZATION: CMA INTERNATIONAL, INC.
VANCOUVER, WASHINGTON

REPORT NO.: 99901148/89-01	INSPECTION RESULTS:	PAGE 2 of 4

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. OPEN/UNRESOLVED ITEMS:

None

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not applicable. This was the first inspection performed at CMA.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Entrance and Exit Meeting

The NRC inspectors informed Mr. Ashley during an introductory meeting held on March 28, 1989 at the CMA facility of the purpose of the inspection. It was explained to Mr. Ashley that the purpose was to determine to what extent CMA personnel were aware of 70 fraudulent Henry Vogt valves that were provided to the Diablo Canyon nuclear plant in 1988. This issue is discussed in NRC Information Notice (IN) 88-48 and IN 88-48 Supplement 1. The inspection findings and concerns were discussed with Mr. Ashley during an afternoon meeting on the same day.

2. Background

In April 1988, Pacific Gas & Electric (PG&E) informed the NRC about a problem concerning Vogt 2-inch valves (Model No. 13111), which exhibited excessive steam leaking around the bonnet and stem. The valves were supplied to PG&E by Western Valve Company as new valves. It was identified by PG&E that the valves were shipped to PG&E by CMA International of Vancouver, Washington. Subsequently Henry Vogt Company representatives examined the Vogt valves at PG&E and determined that they were not manufactured by Vogt.

3. CMA Customers

The CMA President provided the inspectors with a copy of the CMA "customer directory." The CMA customer directory lists 144 customers, and does not necessarily indicate an on-going business relationship. According to Mr. Ashley some of the customers listed on his directory may or may not have conducted business with CMA.

4. Other Comments

The inspectors requested to review the CMA procurement document package regarding the Vogt valves discussed in E.2 above. CMA personnel conducted an approximate 4-hour search of the archived files and were not able to find the procurement document package or any related records.

5. Personnel Discussions

Discussions with CMA personnel revealed that valve modifications have been a typical on-going part of the CMA shop activities. The type of valve modifications included grinding off valve pressure ratings, serial and model numbers, manufacturer's identifications and inserting new rating numbers and identifications. For example, in approximately February 1989, CMA personnel stated that they were instructed to modify a 6-inch, 900 pound valve to be sold as 5-inch, 600 pound valve.

6. Follow-up at Trojan

The inspector identified approximately 12 local suppliers from the CMA customer list who are also sales representatives for different original valve equipment manufacturers (OEMs). The inspector contacted the Trojan NRC Resident Inspector and arranged to visit the facility to review Trojan's records. Representatives from Portland General Electric (PGE) Trojan plant were contacted by the Resident Inspector and provided information to the Vendor Branch inspector during March 29-31, 1989, regarding the sample list of 12 local suppliers. A review was conducted of incoming invoice activity for each of the 12 suppliers over the last five years. The following suppliers from CMA's customer list were found to have supplied valves to Trojan within the past five years: Am-Fac Supply Company (currently Tyler-Dawson), Basic Controls and Valves, Consolidated Supply, Grinnel Corporation, Industrial Valve of Oregon, Keenan Pipe and Supply, and Liberty Equipment and Supply.

During the review, no procurement packages were found to contain CMA invoices. However, several packages were found which did not contain traceability back to the manufacturer of the valve. PGE is currently reviewing the circumstances surrounding the lack of traceability for each of the packages and has committed to keep the NRC informed of its progress.

F. PERSONS CONTACTED:

Name	Title	Company
*C. Ashley	President	CMA
L. Ashley	Vice President	CMA
**D. Nordstron	Licensing Engineer	PGE
**D. Glivinski	QA Manager	PGE
**K. McDonald	QA Engineer	PGE
**R. Barr	Resident Inspector	USNRC

*Attended the CMA exit meeting.
**Attended the NRC/PGE exit meeting on March 29, 1989.

ORGANIZATION: CMA INTERNATIONAL, INCORPORATED
VANCOUVER, WASHINGTON

REPORT NO.: 99901148/89-02	INSPECTION DATE: April 25-26, 1989	INSPECTION ON-SITE HOURS: 30

CORRESPONDENCE ADDRESS: CMA International, Incorporated
2424 East 2nd Street
Vancouver, Washington 98661

ORGANIZATIONAL CONTACT: Mr. Clifford Ashley, President
TELEPHONE NUMBER: (206) 696-0818

NUCLEAR INDUSTRY ACTIVITY: CMA International, Incorporated (CMA) provides new, new surplus, refurbished and modified valves to valve distribution companies who in turn sell directly to nuclear power plants. IMA Valve Refurbisher, a Division of CMA, operates from the same address above. IMA provides valve repairing and refurbishing services. Both companies use the same shop and shop personnel.

ASSIGNED INSPECTOR: Joseph Petrosino June 2, 1989
J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S): Brooks Griffin, NRC, Region IV

APPROVED BY: CE Haller Jr. June 2, 1989
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21.

B. SCOPE: The purpose of this inspection was to review CMA records to determine if CMA has, directly or indirectly, provided valves to any nuclear power plant facilities even though CMA provides commercial grade products.

PLANT SITE APPLICABILITY: Arkansas-1 (50-313), Crystal River-3 (50-302), and Vogtle-1 (50-424). Potentially applicable to all plants.

ORGANIZATION: CMA INTERNATIONAL, INCORPORATED
VANCOUVER, WASHINGTON

REPORT NO.: 99901148/89-02	INSPECTION RESULTS:	PAGE 2 of 6

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. OPEN/UNRESOLVED ITEMS:

None

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

None

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Entrance and Exit Meetings

The NRC inspection team informed Mr. Ashley, during an introductory meeting held on April 25, 1989 at the CMA facility, of the scope of the inspection. It was explained to Mr. Ashley that the inspection team requested access to all of the CMA records that might indicate valve sales to nuclear plant facilities or to distributors that could supply valves to those facilities. Upon completion of the inspector's review of the CMA records on April 26, 1989, Mr Ashley was provided with a receipt for copies of each of the approximately 67 valve order packages which the inspectors noted for further review. Mr. Ashley allowed the inspection team to transport all of the noted 67 packages to the NRC, Region IV offices in Arlington, Texas for further reviews.

2. Background

The NRC inspection team initially visited the CMA facility on March 28, 1989 to obtain information from CMA employees regarding the Henry Vogt 2-inch valve issue which is discussed in NRC Information Notice (IN) 88-48 and to review the documents associated with that order. The March 28, 1989 inspection resulted in obtaining useful information from the CMA employees; however, the CMA office personnel could not locate the Henry Vogt valve associated records so that they could be reviewed by the inspectors. Therefore, a second inspection at the CMA facility was scheduled to perform a record review of all available CMA valve order documents. The inspection scope was to review CMA records from approximately 1980 to the present. However, during the April 25-26, 1989 review, the inspectors were not able to find any records regarding the Vogt 2-inch valves that were shipped to Diablo Canyon by CMA (reference IN 88-48).

3. Record Review

The NRC inspection team reviewed approximately 16 linear feet of CMA records, including valve order packages and bills of lading from the various shipping companies that are used by CMA. IMA Valve Refurbisher, Division of CMA, operates from the same address as CMA. IMA provides valve repairing and refurbishing services. Both companies use the same shop and personnel. The CMA valve order packages were found to contain documents and information such as: (1) IMA and CMA Invoices, (2) IMA and CMA shop orders, (3) IMA and CMA internal notes and directions to shop craftsmen, (4) IMA Valve Refurbisher records, (5) bill of ladings, (6) distributor purchase orders/fascimiles, (7) CMA telephone notes regarding where its valves were procured. (8) packing lists, and (9) other associated documents.

The record review revealed that CMA drop shipped an order of valves to the Arkansas nuclear plant facility and another example indicates that other CMA customers shipped valve orders to the Crystal River and Vogtle facilities. On CMA Invoice 9127, dated July 23, 1986, six Pacific globe valves (5-8" Model 360-1-WE and 1-3" Model 360-1-WE) were sold to Zenith Supply Company in Pittsburgh, Pennsylvania and shipped by CMA to Arkansas Nuclear-1 in Russellville, Arkansas. Another example shows that on CMA Invoice 9152, approximately September 1, 19[illegible], CMA supplied 2-6" 900 psi rated Pacific Check Valves, Model 980 to Midwest Valve and Supply, Detroit, Michigan under Midwest Purchase Order 5798. Midwest supplied the two Pacific Valves to Jay Instrument and Specialty Company, Norcross, Georgia (Jay Reference No. 08-6-2480). Apparently, Jay Instrument supplied the valves to the Vogtle nuclear plant facility who in turn installed them in its reactor condensate and feedwater system. Subsequent licensee/Pacific Valve Company discussions indicated that the valves were not manufactured by Pacific. The last example reveals that Zenith Supply Company, Pittsburgh, Pennsylvania ordered at least one 24" Crane-Chapman check valve from IMA Valve Refurbisher (Division of CMA), under Zenith Purchase Order No. 84D8219 for Florida Power Company (FPC) Purchase Order No. F9022330X, approximately May 1984. A review of FPC records subsequent to this CMA inspection indicate that the

ORGANIZATION: CMA INTERNATIONAL, INCORPORATED
VANCOUVER, WASHINGTON

REPORT NO.: 99901148/89-02	INSPECTION RESULTS:	PAGE 4 of 6

licensee procured the valves commercial-grade and upgraded them to safety-related based, in part, on: an IMA "valve testing and inspection certificate," a "Crane-Valve Service Center" certificate and a FPC QC receipt inspection of the valves. Subsequent discussions with Crane Company representatives by the NRC staff appears to indicate that the Crane certificate was not written or issued by Crane.

During the record review, numerous inconsistencies were observed regarding the validity of the "new" valves that were shipped on the CMA Invoices to its commercial-grade customers. Examples are as follows:

(1) CMA Invoice 9127, dated July 23, 1986, [illegible] sold 5-8" Pacific valves and 1-3" [illegible] Supply and drop shipped the valves [illegible] plant-1 (ANO). The CMA invoice li[illegible]cifics, prices and references the purchase [illegible]cs, but does not state whether the valves are new, used, refurbished, or surplus. However, within the package, a Newmans, Incorporated, Tulsa, Oklahoma Packing List, Number SO-10847, was found which indicates that it sold 5-8" valves to CMA. The Newmans document was dated July 11, 1986. The item description on the Newmans packing slip stated "S2080030 FHCB-1PVB, Petro-valve, 8", 300# WCB-Fldg, (F6-A), HO globe, OS&Y, Bolted Bonnent, surplus-Ship-as-is." This order appears to indicate that CMA procured a surplus valve and provided it as new to Zenith, who sold them to ANO.

(2) CMA valve order package indicates that CMA bought 2-8" used "Kerotest" valves from CANA-WEST, British Columbia, on approximately December 30, 1986, performed reconditioning, machining and drop shipped the valves to Southern California Edison Company as the supplier for Midwest Valve and Fitting Company (Midwest Purchase Order, dated December 31, 1986). Neither the CMA invoice nor the Midwest invoice indicates that the valves are used or reconditioned. The CANA-WEST invoice found in the CMA package specifically states that the valve was a used Kerotest valve.

(3) CMA Invoice 9263, dated February 28, 1986, sells 1-4" Pacific gate valve to Zenith Supply, Pittsburgh, Pennsylvania and drop ships the valve to Wayne Pipe and Supply in Fort Wayne, Indiana. Several CMA notes that are part of the CMA valve

REPORT NO.: 99901148/89-02	INSPECTION RESULTS:	PAGE 5 of 6

order package indicate that the CMA shop personnel were instructed to "grind off CHA MAN and leave ~~CHA-P-MAN~~ [the "P" on the valve body]...find Pacific hardware and modify...." The CMA written shop instructions indicate that CMA took a CHAPMAN valve and modified the Chapman trademark to resemble a Pacific valve (i.e., P=Pacific).

(4) CMA Invoice 8327, dated October 11, 1983, sold 1-6" Lunkenhemier Model 1542 to ITT Grinnel, Portland, Oregon. IMA note[illegible] that were found with the CMA valve order package instruct [illegible]A shop personnel to "...make from Kerotest...may have to [illegible] to make 16 inch "face-to-face." The IMA notes also indi[illegible] the following underlined items to be installed on a [illegible] plate: Size - 6", Figure - 1542, S - 150, F - 500, WO[illegible]. Stem - CR13, Disc - CR13, Seat - NICU, Body - WCB and Serial - 36100.

(5) CMA Invoice 9415, dated March 5, 1987, sold 1-20" Lunkenheimer gate valve, Model 3031, to Grinnell, Portland, Oregon (Grinnell Purchase Order No. 73829AT). CMA notes in the valve order package appear to indicate that CMA bought a surplus 24" Secca gate valve, and then CMA sand blasted, reseated, repacked, painted, installed a gear operator and die-stamped the valve with Lunkenheimers trademark and model number (i.e., "L" 3013).

(6) CMA Invoice 9156, dated September 17, 1986, sold 1-2" Henry Vogt globe valve, Model SW-1023, to Lowe-Parker Corporation in Seattle, Washington. CMA notes in the valve order package indicate that CMA used a new "no-name" CMA stock valve, die-stamped it with "Vogt SW-1023" and added 6[illegible]0 psi rated flanges in order to obtain the required 11 1/2" face-to-face dimension.

(7) CMA Invoice 9623, dated April 5, 1988, stated that it sold 2-6" 150 psi rated Crane gate valves, Model 47XU, 1-3" 150 psi Crane gate valve, 1-4" 150 psi rated Crane gate valve and 1-3" 300 psi rated Crane valve, Model 33XU to Grinnell, Portland, Oregon. The valves were drop shipped to Grinnell (Purchase Order 2171-LT). The CMA shop notes state "refurbish-like new." A CMA certificate of conformance, dated June 6, 1988, signed by the CMA President states to Grinnell, Longview that the subject valves "...were purchased...as new and unused from surplus lots...[CMA]...furnished [subject] valves as new surplus in good faith. Valves were inspected

REPORT NO.: 99901148/89-02	INSPECTION RESULTS:	PAGE 6 of 6

and tested by IMA Valve Refurbishing to ensure certification of compliance." A Crane Company letter, dated May 25, 1988 addressed to the Boise Cascade Paper Group, St. Helens, Oregon [end user] states, in part: "On April 28, 1988, I inspected the following cast steel valves located at your St. Helens facility...

- 1 6" - 47XU [Serial No.] DA 98734
- 1 3" - 33XU [Serial No.] DA 58735
- 1 6" - 47XU [Serial No.] DA 98735
- 1 4" - 47XU [Serial No.] DA 86485
- 1 3" - 47XU [Serial No.] DA 86486

[Crane] Inspection indicated that the above valves were not new Crane valves, but reconditioned or surplus vintage. We were not able to confirm any of the serial numbers as being genuine Crane valve numbers. In the future may we urge that you purchase Crane valves only from authorized distributors, such as Industrial Valve of Oregon. [Signed] W. D. Blakeslee." The Crane letterhead indicates that the Crane Company, Carol Stream, Illinois, facility personnel performed the inspection.

The above examples represent only 10 of the 67 CMA valve order packages that were identified by the inspectors as suspect and that will require additional staff review. In conclusion, it appears from the above, that a CMA supplied suspect valve could go through two or more distributors before reaching the end user facility. A nuclear power plant could buy a commercial-grade valve from a valve distributor (e.g., Zenith, Midwest, Grinnell) and dedicate the valve for safety-related use on the basis of buying from an authorized original valve manufacturer sales representative. However, unless the nuclear plant staff verifies traceability, it could end up receiving one of the above discussed CMA "modified" valves.

F. PERSONNEL CONTACTED:

Clifford Ashley, CMA President

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

January 2, 1992

Docket Nos. 50-269
50-270
50-287

Mr. Hal B. Tucker
Senior Vice President
Nuclear Generation Department
Duke Power Company
Post Office Box 1007
Charlotte, North Carolina 28201-1007

Dear Mr. Tucker:

SUBJECT: ASSESSMENT OF THE PROCUREMENT AND COMMERCIAL-GRADE DEDICATION PROGRAMS AT THE OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, REPORT NOS. 50-269/91-201, 50-270/91-201, AND 50-287/91-201

This letter transmits the report of the assessment conducted July 15 through 19, 1991, at the Charlotte, North Carolina, general office of Duke Power Company (DPC) and at the Oconee Nuclear Station, Units 1, 2, and 3, by R.P. McIntyre, S.D. Alexander, R. Frahm Jr., and L.L. Campbell of the Nuclear Regulatory Commission's (NRC's) Vendor Inspection Branch, and M. Thomas of NRC Region II. At the conclusion of the assessment, we discussed our findings with you, and the members of your staff identified in the appendix to the enclosed report.

The assessment was performed to review DPC's program for the procurement and dedication of commercial-grade items used in safety-related applications at Oconee in accordance with Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) and to determine the extent of the implementation of the Nuclear Management and Resources Council (NUMARC) initiatives in this area.

DPC has made a significant effort to strengthen its commercial-grade dedication program since its inception in 1987. Its overall program description was generally consistent with the dedication approaches described in the Electric Power Research Institute (EPRI) Report NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)," as conditionally endorsed by NRC Generic Letter 89-02, "Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products," March 21, 1989. However, the lack of full implementation of this program was a significant weakness. DPC senior management decided to phase in the new program by March 1993 (revised to December 1991) and to continue to purchase and dedicate commercial-grade items (CGIs) on the basis of documented technical evaluations in existence as of January 1, 1990. Therefore, the purchase and dedication of CGIs previously evaluated and listed on the commercial grade items list (CGIL) as of January 1, 1990, were not based on the requirements of the current program, but only on a review of product and supplier performance history (EPRI Method 4). During a July 26, 1991, conference call between DPC senior management and the NRC, DPC stated that it had decided to accelerate to October 1, 1991, the phase-in of the reevaluation of outstanding evaluations done under the old program. Any remaining CGIs listed on the CGIL without new evaluations completed by October 1 would be placed on hold pending completion of an evaluation using the current program requirements. The assessment team concluded that the dedication methods used for the large majority of CGIs purchased after January 1, 1990, did not meet the DPC programmatic requirements in place and also did not meet the NUMARC initiative on the dedication of commercial grade items. The NUMARC initiative stated that licensee programs would meet the intent of the EPRI NP-5652 guidelines as of January 1, 1990.

The assessment team identified weaknesses both in the overall procurement program and its implementation. DPC's philosphy that allowed selecting only a subset of critical characteristics, instead of requiring verification of all critical characteristics to provide assurance that the item would perform its intended safety functions was a program weakness. The licensee is responsible for identifying the attributes necessary for performance of the item's safety functions, establishing acceptance criteria and providing reasonable assurance of conformance to these criteria. In addition, some critical characteristics specified were not adequately verified for the procurement packages reviewed. With appropriate modifications to address these concerns, the program, if properly implemented, could provide adequate control over the commercial-grade procurement process. Specific strengths and weaknesses are discussed in detail in the enclosed report.

DPC had completed its review and assessment of the comprehensive procurement initiatives suggested in NUMARC 90-13, "Nuclear Procurement Program Improvements," dated October 1990. The initiative called for the licensee to complete its review by July 1, 1991, and to complete implementation by July 1, 1992. The DPC progress in this area should enable you to meet the July 1, 1992, completion date.

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

Although no response to this report is required, we expect you to consider the concerns raised herein and to take appropriate measures. Should you have any questions concerning this assessment, we will be pleased to discuss them with you. Thank you for your cooperation in this assessment process.

Sincerely,

Steven A. Varga

Steven A. Varga, Director
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosure: Assessment Report 50-269/91-201, 50-270/91-201
and 50-287/91-201

cc: See next page

Mr. Hal B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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

Report No.:	50-269/91-201, 50-270/91-201, and 50-287/91-201
Docket No.:	50-269, 50-270, and 50-287
License No.:	DPR-38, DPR-47, and DPR-55
Licensee:	Duke Power Company Nuclear Generation Department Post Office Box 1007 Charlotte, North Carolina 28201-1007
Facility Name:	Oconee Nuclear Station, Units 1, 2, and 3
Assessment at:	Charlotte, North Carolina
Assessment Conducted:	July 15 through 19, 1991

Richard P. McIntyre 12/24/91
Richard P. McIntyre, Team Leader
Vendor Inspection Branch (VIB)
Date

Other Inspectors:
S.D. Alexander, EQ and Test Engineer, VIB
L.L. Campbell, Reactor Engineer, VIB
R.K. Frahm, Jr., QA Engineer, VIB
M. Thomas, Reactor Engineer, RII

Approved by: Leif J. Norrholm 12/24/91
Leif J. Norrholm, Chief
Vendor Inspector Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation
Date

TABLE OF CONTENTS

EXECUTIVE SUMMARY

From July 15 through 19, 1991, the Nuclear Regulatory Commission's (NRC's) Vendor Inspection Branch conducted an assessment of Duke Power Company's (DPC's) activities related to the procurement and dedication of commercial-grade items (CGIs) used in safety-related applications at the Oconee Nuclear Station (ONS), Units 1, 2, and 3. The assessment team reviewed DPC's procurement program to assess its compliance with the quality assurance (QA) requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) and to assess the status of DPC's implementation of the Nuclear Management and Resources Council (NUMARC) initiatives on procurement and commercial-grade dedication.

The NUMARC Board of Directors has approved procurement initiatives as described in NUMARC 90-13, "Nuclear Procurement Program Improvements," dated October 1990, which commit licensees to assess their procurement programs and take specific action to strengthen inadequate programs. The first phase of these initiatives was the NUMARC initiative on the dedication of CGIs (adopted by NUMARC in March 1989) which was scheduled to be implemented by January 1, 1990. Licensees were to meet the intent of the guidance provided in the Electric Power Research Institute (EPRI) Final Report NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)," June 1988. The NRC has conditionally endorsed this guideline in Generic Letter (GL) 89-02, "Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products," March 21, 1989. The second phase of the initiatives provides a comprehensive procurement review and addresses vendor audits, tests and/or inspections, obsolescence, information exchange, and general procurement. Licensees were to review their programs by July 1, 1991, to determine, on the basis of guidance in NUMARC 90-13, if improvements were needed in these areas and to complete such improvements by July 1, 1992.

The NRC performed its assessment to determine the current status of the activities to improve the procurement program related to the industry initiatives discussed above and NRC requirements. The assessment focused on a review of procedures and representative records; interviews with DPC staff, including senior management and ONS site personnel; and observations by the assessment team members. The assessment team also held meetings with DPC's corporate management to discuss relevant aspects of commercial-grade dedication and to identify areas requiring additional information. The assessment team's observations were discussed with DPC's representatives and senior management at the exit meeting held July 19, 1991. The assessment team's specific conclusions are summarized below.

- DPC had made a significant effort to strengthen its commercial-grade dedication program since its inception in 1987 and its overall program description was generally consistent with the dedication philosophy described in EPRI NP-5652.

- The DPC program made the distinction between critical characteristics for design and critical characteristics for acceptance and stipulated that the acceptance critical characteristics are a subset of critical characteristics for design. DPC believed it was not necessary to identify and verify all critical characteristics, but only those critical characteristics for acceptance that provided reasonable assurance that the item received was the item specified. We interpret the "item specified" to encompass attributes necessary for performance of the item's safety functions. The NRC staff's position is that Appendix B requi s the licensee to ve ify all characteristics that are critical to ensure that the item performs its safety functions for its particular plant application.

- In its letter of May 8, 1990, regarding implementation of the NUMARC initiative on the dedication of commercial grade items, DPC decided to continue to purchase CGIs previously listed on the commercial grade items list (CGIL) and dedicate them on the basis of existing evaluations (prepared before January 1, 1990) until a new/revised evaluation was prepared for each CGI to the current program requirements. Therefore, the purchase and dedication of CGIs previously evaluated and listed on the CGIL as of January 1, 1990, were not based on the requirements of the current program, but only on a review of product and supplier performance history (EPRI Method 4). This process of phasing in the completic of reevaluations for items purchased after January 1, 1990, (but under the old program), was to be completed by December 31, 1991. The fact that CGIs procured after January 1, 1990 were being dedicated using previous program evaluations was considered a significant weakness in the DPC program for commercial-grade procurement and dedication. The large majority of CGIs dedicated after January 1, 1990, did not meet the DPC programmatic requirements in place and also did not meet the NUMARC initiative on the dedication of commercial grade , which stated that licensee programs would meet the intent of the EPRI NP-5652 guidelines as of January 1, 1990.

- Quality Assurance Department Procedure QA-606, "Commercial Grade Surveys," required that DPC perform a survey of commercial-grade suppliers at least once every three years and did not require periodic reviews and evaluations of the supplier during this period. However, it may be necessary to perform commercial-grade surveys at a frequency other than on a triennial basis due to changes in the supplier's quality program, procedures, processes, management, or personnel performing the work activities. Commercial-grade surveys should be scheduled at a frequency commensurate with the status, importance, and complexity of the item or process being surveyed.

  Other observations concerning the commercial-grade survey process are discussed in Section 2.4.2 of the report.

- DPC's program did not provide for minimum formal documented training requirements for personnel performing quality-related activities within the commercial-grade procurement and dedication process. However, such training is considered necessary to achieve effective and consistent implementation of the program within design engineering. Therefore, s was considered a weakness.

- DPC initiated interim measures to detect counterfeit and fraudulently marketed products until the completed fraud detection program is implemented as part of the results of the NUMARC comprehensive procurement initiative review. However, DPC was not effectively implementing these measures during the receipt inspection process at ONS and no training had been conducted in these areas.

- DPC has had strong engineering involvement in its commercial-grade dedication program since it was first implemented in January 1987. This involvement consisted mainly of the performance of technical evaluations to support the purchase of CGIs. These evaluations were continually upgraded in scope and content as the program evolved. DPC design engineering, construction, quality assurance, and operations personnel became involved as the dedication program continued to evolve.

- DPC provided management support, input, and sufficient resources to improve its commercial-grade dedication program. However, the NRC staff did not agree with the DPC basis for phasing-in the new program. The DPC staff displayed great interest in the NRC team's assessment effort and management was available for consultation during the assessment.

1 INTRODUCTION

The NRC's Vendor Inspection Branch assessed Duke Power Company's (DPC's) efforts to improve programs for procuring and dedicating commercial-grade items (CGIs) used in safety-related applications. The NRC assessment team reviewed the DPC program to assess its compliance with Appendix B to 10 CFR Part 50 and to assess the status of implementation of the Nuclear Management and Resources Council (NUMARC) procurement initiatives. The assessment was performed between July 15 and 19, 1991, at the DPC general office in Charlotte, North Carolina. The assessment methodology included observations, discussions with licensee managers and corporate and site personnel, and a review of records and procedures associated with the licensee's procurement and commercial-grade dedication program.

This completes the NRC assessments at selected licensees' facilities to review their implementation of improved programs for the dedication of CGIs and to assess the improvements made in the areas covered by the NUMARC comprehensive procurement initiative program. This initiative, approved on June 28, 1990, by the NUMARC Board of Directors, directed licensees to adhere to the guidance provided in the Electric Power Research Institute (EPRI) NP-5652 Final Report, and to review and strengthen their procurement programs in accordance with specific guidance provided in NUMARC 90-13, "Nuclear Procurement Program Improvements," October 1990.

The specific areas reviewed and the team's observations are described in Sections 2 through 4 of this report. The conclusions, strengths and weaknesses are summarized in Section 5 and Section 6 describes the exit meeting. Persons contacted during the assessment are listed in the appendix.

2 COMMERCIAL-GRADE DEDICATION PROGRAM REVIEW

The assessment team reviewed DPC's programs and related commitments associated with the implementation of the NUMARC initiatives, including the program for procurement and dedication of CGIs used in safety-related applications at the ONS. "Dedication" is generally understood to mean the process by which an item, not manufactured and supplied under an approved 10 CFR Part 50, Appendix B quality assurance (QA) program, is verified to be suitable for use in a nuclear safety-related application. A commercial-grade dedication program must be conducted under an Appendix B QA program because it consists of activities affecting quality. Therefore, DPC's commercial-grade dedication program was assessed against Appendix B critiera.

2.1 Procurement Program Overview

Pursuant to the standard assessment plan, the team reviewed procurement program processes and procedures with emphasis on applicability of the dedication process for CGIs intended for safety-related applications including incorporation of dedication approaches described in EPRI NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related

Applications (NCIG-07)," issued in June 1988, as conditionally endorsed by NRC Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," dated March 21, 1989.

The review also included the DPC program and activities for selection and qualification of suppliers, including the use of audits and source surveillances, commercial-grade supplier surveys and source verifications, incorporation of the guidance of GL 89-02 with regard to the use of commercial-grade supplier surveys (Method 2 of EPRI NP-5652) as well as the use of supplier/item history (EPRI Method 4) if applicable, and the use of audits and surveys performed by third parties, such as those conducted by teams representing several utilities sponsored by the Nuclear Procurement Issues Council (NUPIC).

Finally, the team's review of DPC dedication activities included those performed at the ONS after receipt, including receipt inspection and other special tests and inspections under Method 1 of EPRI NP-5652.

2.2 Procedures Review

The procurement process, particularly as it related to CGIs, for ONS (as well as the other DPC nuclear plants, Catawba and McGuire) was described and prescribed by a complicated hairarchy of procedural documentation beginning at the DPC corporate level with the procedures of the Nuclear Production Department (NPD) headquartered in the DPC general office (GO) in Charlotte, North Carolina. The NPD-GO's Administrative Policy Manual (APM), Section 2.4.4.5, was the principal NPD department directive relating to procurement and dedication of CGIs. At the time of the assessment, Section 2.4.4.5 was under revision to make it more general, with more specific guidance being given in the next lower tier procedure, NPD Department Directive 3.3.6(M), "Commercial Grade Program." APM 2.4.4.5 addressed issues primarily related to plant application considerations. It established, for example, in Section 2.4.4.5(c), three categories of CGIs: Commercial Grade Category 1, direct replacement spare part; Category 2, general applications; and Category 3, future applications. However, the wording of (d.1) appeared to contradict the definitions of Categories 1 and 2 by giving engineering evaluation requirements and usage restrictions for Category 1 items further categorized (d.1.2) as those "...which list specific applications or restrictions..." which it distinguished from (d.1.3) "Category 1 direct replacement items which do not list specific applications or restrictions...." According to the wording of the category definitions, (c.1) and (c.2), CGIs approved for unrestricted or generic usage would expectedly be described as Category 2, general applications.

DPC staff explained the apparent contradiction by describing these two situations as being legitimate subcategories of Category 1, distinguished chiefly by seismic and environmental qualification considerations. Subcategory (d.1.2), describes the requirements, including environmental and seismic, of one or a limited number of applications that have been analyzed and the CGI approved only for those specific applications and subcategory (d.1.3), describes those CGIs that have been "preapproved" as direct (i.e., like-for-like) replacements for any plant applications with technical requirements enveloped by the dedication (and qualification) of the CGI to be used. The DPC staff explanation also provided clarification of the type of items for which Category 2, general applications was created; that is items in common use, such as conduit, for which like-for-like replacements would not necessarily be required and which have fewer (or no) seismic or environmental usage restrictions. Nevertheless, none of this was apparent to the reviewer from the text as written and the wording was somewhat ambiguous even when the intent was understood. DPC staff agreed that some clarification was required to make the procedure more conducive to meaningful compliance.

The DPC Design Engineering Department (DED), the architect/engineer for DPC, also was located at the GO in Charlotte and had most of the engineering responsibility relating to procurement and dedication. DED procedures prescribed the mechanics of the processes of (1) safety classification, including upgrading and downgrading; (2) technical evaluation, including direct replacement (like-for-like), acceptable substitute, and design change evaluations; (3) new/alternate application approval for Category 1 direct replacement CGIs, and generic application dedications (including Category 2); and (4) engineering associated with the actual dedication process, including identification of safety functions, failure modes and effects analyses (FMEAs), critical characteristic identification and selection, acceptance method and criteria selection, and establishing CGI technical procurement specifications.

The "Design Engineering Quality Assurance Manual" (DEQAM) and the "Commercial Grade Program Manual" (CGPM) were the two principal DED documents pertaining to procurement in general and procurement and dedication of CGIs in particular. The DEQAM contained several QA procedures pertinent to procurement and dedication of CGIs, including PR-102, "Acceptable Substitutes," PR-103, "Commercial Grade Items," PR-302, "Procurement," and PR-303, "Procurement of Services." The principal implementation procedures for dedication were contained in the CGPM. Chief among these, relevant to CGI procurement and dedication, were Procedure CGP-1.1, "Design Engineering Commercial Grade Technical Evaluation Procedure," and CGP-1.2, "Commercial Grade Program Procurement and Acceptance Manual - Generating and Processing of Documents," or the so-called "CGPA." These documents described the CGI procurement and dedication process (as it involved DED and NPD) for all the DPC plants.

Individual project NPD responsibilities and procedures in this area were largely limited to material requisitioning, performing some special inspections and tests, and conducting and coordinating some post-installation tests. The team briefly reviewed these procedures, with detailed review confined to DED procedures PR-103 and CGP-1.1.

One concern was identified with respect to DED Procedure PR-102, Revision 2, dated January 1, 1990. Section 1.6, which addressed authorized substitute replacement (ASR), stated that an ASR determination was initiated under the following conditions: Paragraph 1.6.1, an identical item with a different part number, paragraph 1.6.2, the original item is no longer available, paragraph 1.6.3, the original equipment manufacturer or the original equipment supplier has a new or improved item that is preferable; and paragraph 1.6.4, a change in the item (form, fit, function, material) results in a change in part number However, if the commercial-grade manufacturer made changes in the design, material or manufacturing process without corresponding changes in model designation, part number, catalog number, or perhaps even the drawing number(s), there is not guidance provided for this situation in the procedure. In fact, such changes may not even be documented and/or controlled as the commercial-grade manufacturer may have no obligation under a quality assurance (QA) program to do so.

This is a significant problem with regard to dedication because a major portion of dedications are often based on a like-for-like determination. However, an acceptable like-for-like determination, such as described in NRC GL 91-05, "Licensee Procurement and Dedication Programs," involves the investigation and determination that the CGI is in fact identical to the original and that there have been no changes in the CGI's design, material, or manufacturing processes. Any such changes would presumably be evaluated (as in paragraph 1.6.4) for their effect on the CGI's ability to perform its safety function under all design basis conditions, but the changes first must be identified, and the conditions given in Section 1.6 of PR-102 for ASRs did not cover this situation.

Overall guidance for the new DPC program for procurement and dedication of CGIs was provided in DED QA Procedure PR-103, Revision 1, dated September 14, 1990. The statement of purpose of the procedure included the assertion that the procedure "met the intent of EPRI NP-5652...." Accordingly, the definition of acceptance was consistent with NP-5652, defining the process as the employment of methods to produce objective evidence to "provide reasonable assurance that the item received is the item specified." However, this definition was not consistent with the staff's position that Appendix B requires the licensee to verify all characteristics that are critical to ensure that item performs its safety functions for its particular plant application. This procedure also related basic components with QA Condition 1 (i.e., nuclear safety-related applications) to be procured as either "approved vendor items" (AVIs) from vendors having approved QA programs and accepting 10 CFR Part 21, or as commercial-grade items to be dedicated for safety service. Strength was added to the procedure by the requirement that QA Condition 1 items not meeting the CGI definition in 10 CFR 21.3(a)(4)(a-1) must be procured as AVIs. The commercial grade items list (CGIL) was defined as a computer database in which approved CGIs and associated, approved applications were listed as well as items evaluated to be AVIs (not meeting CGI definition and non safety-related items.)

In addition to other pertinent definitions, such as the NP-5652 definition of commercial-grade supplier surveys, the procedure also established categories of CGIs similar to APM 2.4.4.5. The description of Category 1 CGI used the term "like-for-like" (defined as identical), but stated that the category could include approved substitutes. It required documentation of the same part number or the approved substitute, same function, same seismic qualification, environmental qualification (EQ) documents (if applicable) or qualified to the same EQ requirements, and meets or exceeds applicable codes, standards, guides, and specifications. Category 2 was described generally as applications controlled by design documents. In the definitions of original equipment manufacturer (OEM) and original equipment supplier (OES), the important distinction was drawn that the OEM of the CGI is not necessarily the OEM or OES of the component, system, or equipment of which the CGI is a part and also that the term OES refers to the original supplier of that parent component, system, or equipment. However, it was not clear if the OES also could include the OEM of the parent component, system, or equipment. Finally, the procedure strengthened the program by defining the term "conditioning" for procurement and dedication purposes as special processes other than routine setup and adjustment or installation, etc., including burn-in, calibration, tuning/adjustment, and "selection testing" [screening]. However, notably absent were definitions of design characteristics or critical characteristics, as was the EPRI NP-6406 concept of critical characterisitcs for design. Only the NP-6406 term critical characteristics for acceptance was defined, and that only as those critical characteristics necessary to provide reasonable assurance that the item received is the item specified.

Procurement/traceability requirements were defined as acceptance requirements including, but not limited to, traceability of the item to the OEM, certificate of compliance from the supplier, or specifying standard procurement notes on the purchase request/purchase order (PR/PO), but no information was provided on the reasons for traceability.

Finally, in addition to good definitions of the acceptance methods of special tests and inspection (and post-installation tests) and source verifications, the procedure introduced the concept of periodic review of technical evaluations to ensure their continued validity. Although the recognition that technical evaluations may become invalid is important, the reasons for such obsolescence are related to specific changes in either the application requirements or scope or changes in the design, materials, or manufacturing processes of the CGI. Therefore, merely conducting periodic review, unless the CGI is being procured continuously, may not capture such events. The team questioned whether this method of validation was adequate, or should the reviews be done for each major procurement of a CGI when significant time has passed since the last review.

Some important concepts were introduced by Section 1.1 in general requirements, including documentation of technical and quality evaluations for CGIs to be used in QA Condition 1 applications to demonstrate that the item qualifies as a CGI, that the supplier is capable of producing a quality product, and that the quality of the CGI can be assured. Section 1.2 formally established periodic technical evaluation review, and Section 1.3 established that DED should evaluate all commercial-grade requests.

Section 3.3 discussed dedication on an emergency basis, but this was described as relating to maintaining station operability and for outage support as opposed to preventing or correcting situations in which safety of the plant, the plant staff, or the public may be jeopardized. Although the verbal approvals were required to be documented, the procedure did not give any specific time limits for completing evaluations or for ensuring that certain requirements are met and confirmed before release of the item for operation. This area should be reviewed in light of the audit finding, identified in QA Departmental Audit SP-90-01 (All), concerning the inadequate dedication under an emergency verbal approval.

Section 3.4 added strength by providing a reasonable discussion of requirements for handling the reclassification of CGIs as AVIs. Section 3.5 provided the same for reclassification of QA Condition 1 applications as non-safety related; however, the reasons for some of the references to other sections in the procedure were unclear.

Section 3.7 required documentation of the identification of critical characteristics for acceptance (CCA) in the technical evaluation, and stated that their selection should be on the basis of complexity, safety function, and performance. However, the actual documentation of the safety functions, and critical characteristics for design (CCD) derived from them were not mentioned. Also, it was not clear why selection of CCA should depend on an item's performance when acceptance of an item depends on its performance or other verification of the CCA, which must consist of all those CCD that are needed to demonstrate performance of the safety function.

Section 3.8 provided a general discussion of acceptance methods for CGIs generally consistent with EPRI NP-5652. Athough, the description of Method 2, commercial-grade supplier surveys, in Section 3.8.2 stated that the survey confirms that the supplier documents its commercial quality controls, the procedure did not state that the survey must confirm that the controls are effectively implemented (as stated in GL 89-02), nor did it explicitly require that the survey confirm that the supplier's quality program actually controls the specified critical characteristics for the specific item being dedicated. In addition, the GL 89-02 guidance was omitted for situations in which there is a distributor as well as a manufacturer involved.

The description of Method 3, source verifications, in Section 3.8.3, was more appropriate to a survey and included activities such as witnessing quality activities at the supplier's facility and verifying that the selected CCA are controlled by the supplier instead of witnessing activities on the actual item being procured and verifying that its CCA have been met.

The description of Method 4, supplier/item history, in Section 3.8.4, did not allow Method 4 to be used by itself. Although the GL 89-02 provisions for applicability to specific critical characteristics and to the application were not explicitly addressed, the phrase "pertinent, industry-wide data," was used. A provision for the control of design, material, and process changes to be confirmed by audit [survey] also was missing.

Section 4.3 provided more specific guidance for dedication documentation. The guidance was fairly comprehensive, and included environmental and seismic qualification, references, surveys, safety classification, CGI determination, and design inputs. However, it did not address the following:

- CGI Categories 1,2, or 3
- parent component's safety functions
- replacement part's safety functions
- CCDs (only CCAs were addressed)
- review of design, material and/or process change history
- like-for-like determination/approved substitutes

Although Section 4.3.14 added significant strength by addressing procurement and traceability requirements, it gave no specific guidance for capturing, reviewing, and filing traceability documentation.

A brief review of the DED procurement procedure, PR-302, Revision 41, dated May 27, 1991, indicated a few discrepancies. Section 6 of PR-302, "Special Procurement Requirements," stated that these requirements would be identified in the CGIL, Appendix CGI to PR-302, or specified in documents referenced in the CGIL; yet, it was not clear why paragraphs I.b and II.b of Appendix CGI stated that Section 6 of PR-302 did not apply. Also, Appendix CGI gave two categories for dedications and their associated acceptance methods: (1) those dedications using methods 1,3, or 4 (or combinations) and (2) those dedications by method 2 alone. Also, not addressed were cases in which it would be appropriate to use method 2 in combination with other methods.

The "Design Engineering Department Commercial Grade Program Manual" (CGPM), Revision 2, dated May 9, 1990, contained CGP-1.1, "Design Engineering Commercial Grade Technical Evaluation Procedure," and CGP-1.2, "Commercial Grade Program Procurement and Acceptance Manual - Generating and Processing of Documents."

The currently effective revision of CGP-1.1, Revision 1, dated October 15, 1990, was to be used in conjunction with PR-103 for conducting and documenting the specific dedication activities involving Method 1 (including QA receipt inspection, special tests and inspections, and post-installation testing). Although, revision 1 of the procedure had incorporated guidance from EPRI NP-6406, "Technical Evaluation of Replacement Items," and had defined in Sections 4.3 and 4.4 respectively, CCA and CCD consistent with the EPRI documents, review of CGP-1.1 identified some concerns. Section 6.0, "Technical Evaluation," was not always consistent or well coordinated with PR-103. For example, it called for considering the safety functions of the parent component, the item, and credible failure modes and effects, but did not say how to document these issues, which were not addressed in the documentation requirements of PR-103.

Section 6.8.2 gave some considerations for selecting CCA as a subset of CCD on the basis of complexity, safety function, and performance. However, it then stated, as did NP-6406, that it was only necessary to verify those critical characteristics that provide reasonable assurance that the item received is the item specified.

Section 6.12 discussed acceptance methods, defining them as means to obtain objective evidence that provides reasonable assurance that (1) the supplier is capable of supplying a quality product, (2) the quality of the item can be assured, and (3) the item received is the item specified. Although this section had strengths, including addressing sampling for destructive tests and requiring documentation of inspection and test results for objective evidence, some concerns were identified. Section 6.12.2 addressed acceptance Method 2, but was not clear on requiring technical as well as QA participation and stated that the survey should be performed and documented in accordance with PR-103, however, PR-103 was weak in this area and inconsistent with the CGP-1.1 survey approval criteria. While the GL 89-02 constraints on Method 2 were included, it was not clear how CCA were to be transmitted to QA for use in surveys. Section 6.12.3 addressed Method 3 and contained similar loose language as PR-103 and stated that source verifications should be performed and documented in accordance with PR-103. This section did not address technical participation, witnessing of operations and tests on an actual item(s) being supplied, hold points, or shipping releases.

DED DEQAM QA Procedure PR-304, "Commercial Grade Items," Revision 2, effective date of May 30, 1988 (original effective date January 1, 1987), was one of the two documents that prescribed the commercial-grade program as it was currently implemented.

Commercial-grade evaluation, (other than identifying item description, application and reference information) in 1988 consisted only of (1) 10 CFR Part 21 criteria, (2) commercial grade (CG) category determination, (3) conditioning requirements, (4) EQ, (5) seismic qualification, (6) FSAR/technical specifications, and (7) testing and performance history. This procedure contained the required documentation for CG evaluations of this type and the means for listing the items on the CGIL. The other principal CG program document was DED Manual Procedure II.4.1, "Nuclear Station Commercial Grade Item Evaluation," originally effective January 2, 1987.

On January 1, 1990, these two procedures were superseded by the new procedures, PR-103 and CGP-1.1/1.2, respectively. The team reviewed the DED Manual Procedure II.4.1, revision dated April 30, 1988, and found that it largely paralleled PR-304, but provided more detailed guidance with respect to methodology. One significant item was that this procedure required that if buying an item as a CGI did not provide any economic or scheduling benefit over the same item as an AVI or if any required conditioning (including functional qualification) was either deemed not cost effective or would adversely impact schedule, then the item was required to be purchased as an AVI.

However, the procedure failed to recognize the actual circumstances under which buying an AVI may be preferable, or at least more practical, to accept certain attributes of an item on the basis of a certificate of conformance (COC), provided adequate supporting information or documentation was provided when required and the validity of all the documentation or information, including the COC was adequately verified before placing the item in service. Allowing the use of unvalidated COCs (as well as other vendor-supplied information with no requirement for verification of validity) for acceptance and use of items in safety-related applications is contrary to the requirements of Criterion VII of 10 CFR Part 50, Appendix B. Use of validated COCs is important to assurance of the suitability of application as required by Criterion III.

DPC staff explained that as of the effective date (January 1, 1990) of the new program, all new evaluations were to be performed according to the new procedures. However, in accordance with DPC's position paper, sent by letter dated May 8, 1990, from the Vice President, Design Engineering Department, regarding the implementation of the NUMARC commercial grade item initiative. DPC decided it would continue to purchase CGIs with existing (prepared before January 1, 1990) evaluations, and dedicate them under those evaluations until a new or revised evaluation was prepared for each item. The process of preparing the phase-in reevaluations for items purchased after January 1, 1990 (but under the old program), was to be completed by December 31, 1991. During a July 26, 1991, conference call between DPC senior management and the NRC staff, DPC stated it had decided to accelerate the phase-in of the reevaluation of outstanding evaluations done under the old program to October 1, 1991, and any remaining CGIs listed on the CGIL without new evaluations completed by October 1 would be placed on hold, pending completion of a new evaluation using current program requirements.

The fact that all the new procedures discussed above were not actually being implemented for the procurement and dedication of CGIs and that newly procured CGIs were being dedicated under the previous program, was considered a significant programmatic and implementation weakness in the DPC program for commercial-grade procurement and dedication.

2.3 <u>Parts Classification System</u>

DED implemented the PCPARTS Program (parts classification) to assist the ONS, as well as all DPC nuclear stations, in determining the QA classification of replacement parts. DED informed the assessment team that to date the PCPARTS Program is limited to the classification of QA-1 valve parts and to selected QA-1 pump parts. The basis for the classification of the valve and pump parts are DED calculations DCP 1205.22-00-0001, Revision 4, dated March 15, 1991, and

DCP 1201.05-00-0001, Revision 4, dated March 25, 1991. The calculations identified design inputs such as the vendor drawings and parts lists and references the FSAR, ASME code cases, and some EPRI documents. Each calculation contained a flow chart to assist in determining if a part performed a safety-related function, however, the answers to the following flow chart decision-block questions were not documented:

(1) Is the part a primary pressure boundary part?
(2) Could failure of the part compromise the pressure boundary?
(3) Could failure of the part cause the valve/pump to be inoperable?
(4) Could failure of the part or interaction compromise the function of a nuclear safety-related system?

Also, questions such as the following were not asked:

(1) What is the function of the parent component?
(2) What is the function of the part?
(3) What are the failure modes of the part?
(4) What are the effects of the failure of the part?

The calculations identified many parts as being non-safety related, but did not provide a documented basis for this determination. Also, it was not clear whether consideration was given to failure of items such as gaskets and O-rings in containment isolation valves and the effects that their failure may have on containment integrity. Further, it was not clear if DED considered the effects that excessive contaminants in items classified as non-safety related could have on the integrity of the reactor coolant system or other safety-related systems.

The team determined that no procedure existed that provided guidance for classifying a part unless the part was being evaluated as part of the commercial-grade dedication process in accordance with Procedure PR-103, "Commercial Grade Items," and Section 6.3, "QA Conditions Determination," of Procedure CGP 1.1, "Design Engineering Commercial Grade Technical Evaluation Procedure." The lack of procedure control and the lack of documentation for the calculations supporting the PCPARTS Program was considered a weakness.

The team discussed the control of contaminants in detail with DED and nuclear maintenance and chemistry personnel. The team expressed a concern that the control of contaminants and the effect that contaminants may have on safety-related components and systems is an area governed by 10 CFR Part 50, Appendix A and Appendix B, and that classifying items such as gaskets, O-rings, and packing as non-safety-related excluded them from the requirements of the DPC QA program and Appendix B of 10 CFR Part 50. Therefore, the control of contaminants that may come in contact with safety-related systems or that may enter safety-related systems is then outside the scope of the DPC QA program. The team also expressed a concern regarding the fact that when an item has been classified as non-safety related (yet the Power Chemistry Materials Guide identifies restrictions on the amount of contaminants that may be present) there exist no in line QA controls or periodic QA checks for contaminants once the item has been approved for use. Also discussed was the fact that constituents of or impurities in materials that are used in contact with safety-related items that exceed the requirements and limits specified in the Power Chemistry Materials Guide could cause deterioration as a result of corrosion processes or other reactions that would adversely affect the integrity of the item, component, or system under normal or accident conditions.

2.4 Commercial-Grade Supplier Selection, Qualification, and Survey

The team reviewed the process for selection, qualification, maintenance, and surveys of commercial-grade suppliers used to support DPC procurements. The team discussed the use of commercial-grade surveys with the QA Vendor Manager and engineers from the Equipment Engineering Section of the Engineering Support Division.

The team also reviewed selected commercial-grade surveys and the following procedures in assessing the use of EPRI NP-5652, Method 2, commercial grade survey of supplier:

- Commercial Grade Program Manual Procedure CGP 1.1, "Design Engineering Commercial Grade Technical Evaluation Procedure," Revision 1, dated October 15, 1990
- Design Engineering Quality Assurance Procedure PR-103, "Commercial Grade Items," Revision 1, dated October 25, 1990
- Quality Assurance Department procedure QA-601, "Vendor Evaluation," Revision 20, dated May 23, 1991
- QA-602, "Vendor Surveillance Procedure," Revision 12, dated April 3, 1990
- QA-606, "Commercial Grade Surveys," Revision 1, dated January 24, 1991
- QA-607, "Vendor Performance Based Audits," Revision 0, dated April 9, 1991

2.4.1 Supplier Selection

DPC typically procured replacement items from the original equipment manufacturer or authorized distributor whether the item is a like-for-like replacement or an authorized substitute item. If the item performed a safety-related function, an attempt was made to purchase the item from a supplier who had a quality assurance program that met the requirements of Appendix B to 10 CFR Part 50 and who accepted 10 CFR Part 21 reportability responsibility. If the supplier did not accept nuclear requirements and the item met the definition of a CGI, the item was purchased commercial-grade and dedicated for safety-related use.

2.4.2 Supplier Qualification and Survey

The QA Vendor group performs commercial-grade surveys to ascertain and verify that a manufacturer or distributor of CGIs adequately controls certain characteristics that DED determined to be critical for satisfactory performance of a designated item. As part of using EPRI Method 2, the assigned DED technical evaluator (TE) reviewed existing commercial-grade surveys by DPC or the Nuclear Procurement Issues Committee (NUPIC) for products included in the survey, critical characteristics covered, and validity of the survey. The TE, when required, requested a survey and provided QA with a list of products and critical characteristics to be verified. The TE met with the assigned QA surveyor before the survey to discuss the conduct of the survey and participated in the survey when required or requested. The QA surveyor arranged and performed the survey, wrote the survey report, resolved discrepancies and comments, and approved the report. The TE also reviewed the survey report to ensure all required information was included. Commercial-grade surveys of suppliers were performed, as a minimum, on a triennial basis with no mandatory requirements for an annual or periodic review of the suppliers program during the 3-year period.

The three commercial-grade surveys reviewed were generally consistent with the guidance provided in EPRI NP-5652 for confirming that a supplier was controlling each characteristic of the item to be purchased. However, the following observations and concerns regarding the conduct and processing of the surveys were discussed with QA Vendor and DED personnel:

(1) The commercial-grade survey for the Sika Corporation for the supply of concrete repair mortars was performed as part of CGPA-1000.00-00-0004, "SikaTop Mortar Repair Kits," Revision 0, dated June 12, 1991. The survey used EPRI Method 2 for verifying the following critical characteristics of the mortar: (1) part number, (2) shelf life, (3) compressive strength, and (4) bond strength. Although numerous statements were made in the survey report about how Sika Corporation controlled characteristics, the team could not determine if some statements were the result of the QA manual and procedure review or if they were the result of direct observation, surveillance, or record review of a given activity. DPC personnel performing the survey informed the team that most statements made in the survey report were the result of either direct observation or record review.

In addition, the Sika Corporation survey report states in part, "This facility does not have a documented QA program; however, there are documented Sika Quality Procedures (SQPs) detailing each of the various tests to be performed." However, QA Procedure PR-103, Section 3.8.2, requires that when Method 2 is used, (1) "all procurement documents shall require a Certificate of Conformance stating that the supplier will furnish the item in accordance with their DPC approved quality program," and (2) "acceptance of the item is completed by performing the QA Receipt Inspection, verifying the accompanying supplier's Certificate of Conformance...." Appendix CGI, Section II.f of PR-302 requires that the "supplier is to certify that the items were supplied under the QA Program approved by DPC and that all other requirements in the purchase order were met." The team's concern is that the supplier had no formal documented QA program; therefore, these DPC procedural requirements could not be met.

(2) The commercial-grade surveys for ITW Ramset/Red Head and distributor, POE Corporation were performed as part of CGPA-1000.00-00-0002, "Procurement Requirements for ITW Ramset/Red Head Wedge and Sleeve Concrete Expansion Anchors," Revision 0, dated December 12, 1990. The surveys were very thorough and included characteristics that DPC categorized as design critical characteristics and critical characteristics for acceptance. However, POE Corporation did not have a documented QA program or procedures for performing work on the expansion anchor; instead it visually examined the anchor and stamped the length code on the end of the anchor.

(3) The commercial-grade survey for Kunkle Industries, Inc., Longeran Valve Division, for the supply of safety relief valves and replacement parts, was very thorough, but the team questioned the report statement, "traceability for commercial-grade items is maintained to storage only. After materials have been receipt inspected and approved and placed in storage, traceability is not maintained." The survey report went on to state that traced the disc (HT #22398) to the purchase order and CMTR. The CMTR was reviewed and approved by QA." The survey report indicated that Longeran's Quality Assurance Manual, Revision 5, dated September 30, 1988, which was written to satisfy the requirements of the ASME Boiler and Pressure Vessel Code, Sections I, IV and VIII, Divisions 1 and 2, was accepted by DPC. The team asked if replacement parts such as the disc and valve body were supplied in accordance with the QA manual, if the valve disc was traceable to a PO and CMTR, and if CGI replacement parts such as O-rings and guide pins, were controlled by Longeran under a QA program other than their ASME QA manual. The team was unable to verify that Longeran supplies all safety relief valve parts in accordance with its ASME QA manual, because no Longeran POs or COCs for these parts were given to the team when requested.

The DPC procedures for conducting EPRI Method 2 activities were reviewed and the team discussed the following observations with DED and QA personnel:

(1) Section 5.2.1.b of QA-606 permits the review and acceptance of a NUPIC member's audit report to serve as the basis for DPC to accept a supplier's program and controls for verifying an item's critical characteristic(s) using EPRI Method 2. Unlike procedure QA-601, which provided detailed requirements for the review and acceptance of a NUPIC audit for Appendix B suppliers, QA-606 provided no guidance for screening NUPIC surveys or audits used for EPRI Method 2 acceptance activities.

(2) Section 5.2.2 of QA-606 provided no guidance for conducting a commercial-grade survey, other than to indicate that DED will provide the survey checklists. Other than Form QA-601A, which is primarily for Appendix B audits, there is no guidance for conducting the surveys, for the methodology used to verify critical characteristics or for what objective evidence must be documented to confirm that critical characteristics are being controlled. In addition, the instructions for using the Form QA-601A only stated, "list below or attach special checklist items or technical requirements that are to be included in the QA Program evaluations."

(3) Procedures PR-103 and CGP 1.1 provided no guidance or requirements for items not specifically reviewed during the commercial-grade survey, but may be considered to be within the scope of the representative groups of CGIs reviewed during the survey. For example, the survey at Longeran Valve included items such as a valve disc and spring, but not a valve guide pin. The survey was later used to support the procurement of a valve guide pin.

(4) Although the QA Vendor group's reaction to adverse findings associated with commercial-grade surveys was not proceduralized and was informal, it appeared satisfactory if performed as indicated. Following an evaluation

of an adverse finding resulting from a DPC survey or from reviewing a NUPIC survey, hardware related findings that may adversely affect the plant would result in a hold tag being placed on the item in the warehouse and a problem investigation report being initiated to evaluate the continued use of items installed in the plant.

(5) The use of a survey of a supplier who has no formal documented QA program, yet may have the necessary quality procedures for controlling the manufacture of the item, seems to be inconsistent with DPC procedural requirements which require the supplier's QA program be documented on both the PO and COC. This DPC procedural weakness was evident during the review of the Sika Corportation survey previously discussed.

The team concluded that the commercial-grade survey reports reviewed generally met the requirements of NP-5652; however, the entire commercial-grade survey process was not addressed procedurally in sufficient detail.

2.4.3 Use of Third-Party Audits

Approximately one-half of DPC's audits were third-party NUPIC member audits. These NUPIC audits are used in support of the DPC Appendix B evaluated suppliers list; however, NUPIC audits and surveys may be used in support of EPRI Method 2, activities. Procedural controls exist for screening NUPIC audits used for Appendix B suppliers, but not for commercial-grade suppliers. The manager of the QA Vendor group indicated that NUPIC audits and surveys would be used in the future as part of the DPC commercial-grade survey program and would be properly screened before their use

2.5 Material Receipt, Documentation and Procedure Control

Receipt inspection of CGIs that were to be dedicated for safety-related applications at the ONS were performed by the QA Technical Support (QATS) group. The QATS group located at ONS performed two major activities; (1) reviewing requisitions and specifying appropriate QA and receipt inspection requirements and (2) performing receipt inspections and document reviews of items received. The team reviewed procedures QA-505, "Processing of Procurement Requisitions," Revision 32, dated June 6, 1991, and QAG-1, "Receipt Inspection and Control of QA Condition Materials, Parts and Components Except Nuclear Fuel," Revision 34, dated June 5, 1991; interviewed QATS personnel; and observed receipt inspections at the ONS warehouse. Additionally, the team discussed the receiving activities performed by materials personnel with the general supervisor, and reviewed Material Manual Section 4.4, "Material Receiving," with a revision date of June 16, 1989.

QATS personnel performed the following activities as part of the receipt inspection process:

* A visual examination of the item and its packaging was performed to determine if any damage occurred during shipping. When requested, a shipping damage inspection on items tied down on a vehicle was performed.

* The PO was reviewed and checked to verify that the information on the PO accurately reflected that on the corresponding approved PR, including any

changes identified as not changing technical requirements. The PO package was placed on hold until discrepancies were resolved. When acceptable, PO and requisitions were filed as quality records.

- A review of vendor supplied quality records was performed for compliance to PO requirements. The review of the records included activities such as (1) checking to ensure that the records were in agreement with procurement documents and that records were legible and not substandard or fraudulent, (2) checking to ensure an identification number was on each record so that it could be traced to the item, (3) checking required physical, chemical, and NDE reports for conformance with applicable specification and code requirements, (4) verifying that ASME Code requirements were met, (5) ensuring that NDE records and radiographic film were reviewed by a DPC Level II examiner, and (6) verifying that special design test reports had been approved by DED (e.g., cable test reports, seismic and EQ reports).

- A visual examination of the item was performed to verify that identification and markings are in accordance with procurement documents and the approved vendor records and that protection covers and seals were satisfactory.

- Performance of any special inspections and testing required by Form QA-505D, "Augmented Receipt Inspection Requirement"; performance of special inspections required by Form QAG-1E, "Receiving Inspection Instruction Sheet"; and any required miscellaneous inspections such as checking coatings, preservatives, inert gas blankets, desiccants, and cleanliness.

Upon satisfactory completion of all the described activities, a QA acceptance number was assigned to the item and the item was tagged or marked, if possible with the QA acceptance number on the item. Also, any QA-505D Forms were signed off for acceptance, and QAG-1E forms that were used were entered on the QAG-1A Form, "Receiving Inspection Report." If post-installation testing was required as part of the dedication process, the CGI was conditionally released for installation and testing, along with a QA hold tag. If testing was satisfactory, then the QA hold tag was cleared and a QA acceptance number issued and entered on the QAG-1A Form.

Following the review of the procedures, discussions with ONS personnel, and observation of warehouse receiving inspection activities, the team concluded that the ONS receiving inspection program generally contained the necessary controls required for the receipt, inspection and testing of commercial-grade items. The team considered receipt inspection actions such as the processing of purchase requisitions and the review of vendor-supplied records to be well defined and personnel performing these activities seemed to be very knowledgeable in these areas. The team also noted that although not formalized, the QATS group forwarded the results of ONS receipt inspection to the QA Vendor group, generally on a monthly basis.

The assessment team identified the following areas of the receipt inspection process that required improvement:

(1) Procedure QA-505 did not address the identification of Form QAG-1E during the requisition review process and Procedure QAG-1 did not acknowledge the use of the form. QAG-1, paragraph 4.5.2.j required any special receipt instructions (Form QAG-1E) be performed and the QAG-1E serial number

recorded on Form QAG-1A. Neither procedure QAG-1 or QA-505 defined when Form QAG-1E was to be used or who had the responsibility to determine when Form QAG-1E was applicable for the receipt inspection of an item. Form QAG-1E was referenced as a requirement only once in a commercial grade procurement and acceptance document (CGPA) that being for the receipt of molded case circuit breakers. The use of the special receiving inspection instruction sheets, Form QAG-1E, was not well defined or controlled at the ONS.

(2) QAG-1 required that vendor-supplied quality records be reviewed for misleading, substandard, or fraudulent information and that during the visual examination, no obvious indications that the items were used, misrepresented or supplied with inadequate or unacceptable documentation be identified. Procedure QAG-1 incorrectly referenced Appendix A of EPRI NP-6629, "Guidelines For the Procurement and Receipt of Items for Nuclear Power Plants, NCIG-15)," instead of Appendix C, "Identifying Substandard/Fraudulent Items." Form QAG-1E was being used to detect fraudulent items and documentation rather than NCIG-15, although recent Appendix B and commercial-grade QA condition 1 receiving inspection reports (Form QAG-1A) did not list fraud detection on Form QAG-1E as a required special instruction. According to QATS personnel, the QAG-1E special receiving inspection instruction sheet for fraud detection would be applicable for all QA Condition 1 items. Discussions with QATS personnel revealed that there was no formal training, other than reading QAG-1, Revision 34, for performing receipt inspection. Section 2.6 of this assessment report discusses in further detail the fraudulent products detection program at ONS.

(3) As written, QAG-1 required signoff and approval of the receiving inspection report before the conditional release of a commercial-grade item for post-installation testing. Section 4.6.2 required when the QA hold tag is cleared, a QA acceptance number was to be issued. This number was then entered on the signed-off and approved Form QAG-1A without any requirement for review and approval of any additional information entered on a previously signed-off and approved quality record.

The team considered this a potential weakness in the receipt inspection process because the one signoff for receipt inspection activities could result in certain activities being inadvertently omitted.

(4) Procedural requirements for the procurement of ASME Section III items as small products and the procurement of structural steel as addressed in procedure QA-505 appeared inadequate. However, the QA vendor group said that Section 5.5, "Procurement of Code Items as Small Products," of QA-505 was incorrect and would be deleted and that Section 5.6, "Structural and Miscellaneous Steel," was being rewritten to address commercial-grade procurement and dedication requirements.

2.6 Fraud Detection

The team reviewed the changes made in the DPC procurement program for the detection and exclusion of fraudulent, counterfeit, and refurbished material in response to NRC Information Notice (IN) 89-70, "Possible Indications of Misrepresented Vendor Products," including Supplement 1, and Generic Letters (GL) 89-02, "Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products," and 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs." DPC had implemented a comprehensive operating experience program within the Nuclear Production Department for the review and distribution of NRC and other industry information. IN 89-70 and its supplement were distributed to the Quality Assurance Department, the Design Engineering Department, and numerous other functional groups to review for awareness only, as opposed to review for accountability and problem avoidance. The generic guidance provided in these documents had not been fully incorporated into applicable receipt inspection procedures or instructions, nor had formal training been conducted to assure that receipt inspectors were aware of and were routinely checking for potentially fraudulent products.

A fraud detection program implementation plan, was developed, as part of the NUMARC comprehensive procurement initiative (CPI) to detect potentially fraudulent products. Guidelines were developed and will be placed in the Nuclear Procurement Engineering Program (NPEP) Manual scheduled to be approved by January 1, 1992. These guidelines summarized the intent of IN 89-70, including Supplement 1 and Appendix C to EPRI NP-6629 "Guidelines for the Procurement and Receipt of Items for Nuclear Power Plants (NCIG-15)." The complete fraud detection and CPI program was planned to be implemented, including detailed training, by March 1, 1992.

DPC was using a standard clause it had developed with each 10 CFR Part 50, Appendix B, and commercial-grade purchase order. The clause stated that only new items shall be supplied and that used or refurbished material is unacceptable. Additionally, a receiving inspection instruction sheet, Form QAG-1E was developed to assist in the fraud detection of molded case circuit breakers (MCCBs) during receipt inspection. Commercial grade procurement and Acceptance requirements (CGPAs) were used to provide receipt inspectors with the requirements for Method 1 acceptance (special tests and inspections) of MCCBs for each of DPC's commercial-grade suppliers. These documents referred the inspector to the QAG-1E for fraud detection characteristics.

Another QAG-1E had been developed to assist receipt inspectors in generic fraud detection, referencing Appendix C to EPRI NP-6629. As previously stated, QAG-1 does not reference the QAG-1E and it incorrectly references Appendix A to EPRI NP-6629 for guidance in detecting fraudulent products. Therefore, there was no procedural requirement or direct connection to consult the QAG-1E or any other document during the inspection process for the detection of potentially fraudulent products, aside from MCCBs as described above. Also, there was no provision for receipt inspection to specifically document the performance of a fraud detection inspection for each item upon receipt, other than listing the QAG-1E on the receiving inspection report. There was no documented, formal training completed for the receipt inspectors on fraud detection.

DPC informed the team that it was considering revising Procedure QAG-1 to include a detailed appendix on fraud detection and modifying the receipt inspection report (Form QAG-1A) to include a check-off block for fraud detection. This procedure change would be followed by formal training.

In conclusion, at the time of the assessment, DPC had initiated measures to detect counterfeit and fraudulently marketed products in response to GL 89-02 and IN 89-70, but were not effectively implementing these measures during the receipt inspection process. This is considered a program weakness. The proposed corrective actions to the present fraud detection program, as noted above, could provide adequate controls to aid in the detection of fraud until the complete fraud detection and CPI program is implemented.

2.7 Procurement Package Review

The assessment team reviewed several procurement and dedication packages for both the electrical and mechanical disciplines to assess the effectiveness of the implementation of the DPC dedication program, including documentation of technical evaluations, identification of safety functions and critical characteristics, and the methods chosen to verify the critical characteristics selected. The team also tried to determine if the necessary procedural controls were in place to ensure that quality characteristics would be correctly translated into procurement documents. The selected individual procurement and dedication packages reviewed are discussed below.

(1) Commercial Grade Item Evaluation (CGD), CGD-1005.00-01-0001, Revision 2, dated January 30, 1991, was prepared for the dedication of carbon steel concrete expansion wedge anchors manufactured by ITW Ramset/Red Head and distributed by the POE Corporation. EPRI Method 2, commercial grade survey of supplier, was used to verify all identified critical characteristics. However, only the part number and the vendor-supplied documentation were reviewed during the standard receipt inspection of the item. There were no Method 1 tests, inspections, or measurements required. The team discussed with DPC that items such as anchors and fasteners purchased from Appendix B suppliers generally are dimensionally inspected on a sample basis during receipt, especially when the manufacturer allows the distributor to affix the marking that indicated the length of the anchor.

DPC purchased the concrete expansion wedge anchors from POE Corporation. The distributor received the anchors from the manufacturer, sorted and marked them, and then shipped them to the buyer. Contrary to the requirements of PR-302, Appendix CGI, PR-103, Section 3.8.2, and CGP 1.1, Section 6.12.2, POE Corporation performed work (marked length identification) and supplied the anchors without having a documented QA program or procedures to control work activities. Also, contrary to these requirements, neither the purchase requisition nor the purchase order required that POE's certificate of conformance (COC) identify POE's commercial quality controls and program governing its work activities.

(2) CGD-2002.08-04-0013, Revision 1, dated January 22, 1991, was prepared for the dedication of a relief valve guide pin supplied by the Longeran Valve Division of Kunkle Industries, Incorporated. EPRI Method 2, was the method identified to verify the critical characteristics, part number, dimensions, and material with no additional Method 1 inspections or tests identified. The team was given a copy of the DPC commercial grade items list, dated July 11, 1991, which listed data for CGD-2002.08-04-0013, guide pin for Longeran relief valve, Model 34-H-204, Size 2XJX2-1/2. There were no procurement or tracebility requirements listed that required a COC stating that the item was manufactured in accordance with Longeran's Quality Assurance Manual, Revision 5, dated September 30, 1988, or any

Longeran QA program. However, the DPC Commercial Grade Items List Data Input Form for CGD 2002.08-04-0013 did identify unique procurement and traceability requirements and a required COC.

There was no documented basis or evaluation supporting the applicability of the Longeran commercial-grade survey for the procurement of the relief valve guide pin. Section 2.4.2 of this report provides additional discussion on the use of the Longeran Valve survey for the relief valve guide pin.

(3) CGD 2012.01-07-0002, Revision 0, dated June 21, 1991, was prepared for the dedication of Texaco Premium RB grease. A concern on use of sampling a CGI's critical characteristics when using EPRI Method 1, inspection and testing was identified. The CGD, as well as the commercial grade procurement and acceptance (CGPA) for CGPA 2000.00-00-0062, Revision 0, dated May 16, 1991, required procurement of grease from a "single lot/batch number and each container should have a minimum 14 ounce product volume." The CGPA continued by permitting the testing of the grease for penetration, dropping point, chlorides, fluorides, and sulfates to be by sampling with MIL-STD-105D as the basis for sample plan. Checking the labels affixed to the tubes received was the sole basis to conclude that lot/batch homgeneity existed. The team concluded that there were insufficient requirements provided in the CGD to ensure that lot/batch homogeneity existed.

(4) CGD-3011.04-04-0001, Revision 1, dated July 19, 1989, was prepared for the procurement of a temperature controller, RTD Input, 4-20 mA Output, supplied by Love Controls Corporation. This CGD, as well as several others reviewed, was prepared by DPC prior to January 1, 1990, but was reviewed to determine the quality of the CGD evaluations that were being used for the dedication of CGIs procured after January 1, 1990.

The CGD documentation package stated: "Justification for Testing/Performance History method: Love Controls Corporation has been incorporated since 1970 and has been producing quality merchandise for industry ever since. McGuire nuclear station has been using the subject item (1) since 1981 with an acceptable work history. An NPRDS report taken on March 21, 198 concluded that there have been no generic problems associated with the subject item (See A7).... Conclusion: Based on the information obtained and documented in this evaluation, the Love Controls Analog Temperature Controller's items (1) & (2) are acceptable for use in QA condition 1 applications when procured as commercial grade. They may be used as a direct replacement (Category 1) part, or, when properly evaluated and documented, used in new (Category 3) applications. For any applications where the subject items are part of the NSSS system, then it must be purchased as indicated in the Westinghouse S.P.I.N.... Attachment 7 consists of the NPRDS failures report run on 3/21/89. In this report, 33 failures were recorded and documented. Note: This report covers any of the temperature controllers in the 54 series. Most of the reported cases were due to maintenance problems, i.e.; loose connections, dust/dirt in assembly, scale calibrations, etc. Oconee, McGuire, and Catawba all have maintenance procedures that regularly check the temperature controllers and should keep these kind of problems minimal. Of the few failures that were reported of the indicators themselves (would not calibrate, blown fuse, alarm relays, etc.), no generic problems were determined. Therefore, from this report it can be concluded that the Love Temperature Controllers 54-838-834-8160-8187-8173-8174 and 54-838-834-8134-8169-8174 show no generic problems and are acceptable for use at Duke Power Company."

The team questioned the validity of many of these statements. Of the 33 failures reported, 13 were the result of the temperature controller being out of calibration, of which 10 were replaced because they could not be recalibrated and another 12 had to be replaced because they could not pass surveillance testing as a result of worn parts. The team also questioned the basis for the traceability to the environmental and seismic qualifications and stated that these traceability issues were not sufficiently addressed in the CGD.

This CGD was a good example of earlier (before January 1, 1990) technical evaluations that formed the basis for the dedication of the CGIs on the CGIL and were used to procure CGIs after January 1, 1990. These evaluations were really only a review of product and supplier performance history (EPRI Method 4) and did not implement the other EPRI NP-5652 dedication methods that are requirements of the current DPC dedication program.

(5) CGD-3014.01-24-0001, Revision 2, dated June 17, 1991, was prepared for the dedication of GE molded case circuit breakers (including auxiliary switches & shunt trips). Review of this file identified several concerns. There were reference problems such as NEMA AB2-76 instead of-84, no reference to AB4-1991, and reference to Section 5.5 (which did not exist) of the GE MCCB application guide, GET2779G. The instantaneous magnetic (IM) function definition and caution in the CGD were contrary to the test method given. The dedication was not application specific, so only some general safety functions were listed. The list of critical characteristics and acceptance methods did not address verification of trip-free operation, interrupting capacity, and insulation resistance. Also, there were no requirements for full-load hold-in capability and no individual pole resistance test. Only a thermal time delay overload test at 300 percent of nominal was specified. Some maximum clearing times were given, not from the time-current curves, and no minimum values were given to be used unless nuisance tripping has occurred. Also, it was not clear how the issue of the full-load rating expectation versus the GE standard rating for 80 percent continuous load (greater than 3 hours) in an enclosure at 25C per UL-489 was addressed.

For the IM trip test, the trip value of 65 percent of the "expected trip point" is an inadequate acceptance criterion because (1) expected value was undefined and (2) the expected result should be no trip at the lowest test value. The origin of the values in the table of trip currents was unclear and tolerances were not evident. Also only the high values were specified which is contrary to the station procedure as well as NEMA standards. For testing at one setting only, it was not clear that the design setting was used or even known. The tolerances given on these values were extremely restrictive (+/-5 or 15 percent) and not likely achieveable. There were no post-installation tests specified for motor starting/running. The explanation of this test was different from the station procedure and did not make sense.

For historical perspective, the team reviewed an earlier version of this MCCB dedication plan that had apparently been used until recently. CGD-3014.01-24-0001, Revision 0, dated October 2, 1989, GE molded case circuit breakers (including auxiliary switches and shunt trips) was based primarily on review of performance history which the team considered inadequate by itself. Additionally, the performance history consisted of a search of the nuclear plant reliability data system (NPRDS) maintained by the Institute for Nuclear Power Operations (INPO) and a somewhat simplistic interpretation of the results. The NPRDS reporting threshold, particularly as interpreted by subscribers, may be too high to capture the majority of relevant MCCB failures and it is also typically very difficult from a scan of an NRPDS record printout to determine in many cases exactly what component actually failed and how, without contacting the reporting party for details and clarification.

DPC had ordered an assortment of GE MCCBs from Mill Power Supply Company of Charlotte under PO A04447-70, dated January 23, 1991. After some initial uncertainity as to whether any MCCBs on this PO had been received, DPC determined that ONS had in fact received them, but DPC was unable to produce any inspection and test records on them during the assessment. The team was not able to determine under which version of the dedication documents described above these MCCBs were dedicated.

In summary, the review of the selected individual procurement packages revealed that (1) safety functions specific to the particular application were not always clearly identified, (2) critical characteristics were not adequately identified as dictated by safety function, (3) all appropriate critical characteristics were not selected for verification, and (4) acceptance testing to verify those characteristics that were selected was not always adequately performed. In addition, as discussed previously, the technical evaluations performed under the previous program requirements that formed the basis for the CGI dedication, were only a review of performance and supplier history.

2.8 Quality Assurance Departmental Audit

Quality Assurance (QA) Departmental Audit SP-90-01 (All) was conducted during November 19, 1990, through January 24, 1991, to evaluate the adequacy and effectiveness of the DPC commercial-grade program for activities performed after January 1, 1990. The audit looked at the QA program to evaluate the adequacy of the procedural guidance and direction, as well as the technical adequacy of numerous commercial grade item evaluations (CGDs) performed by different groups within DPC for various nuclear stations. The audit report was dated April 1, 1991.

The audit appeared to be an extensive, thorough, performance-based audit that documented several pertinent findings and observations in both the commercial-grade dedication program and its implementation. The team verified that all findings had been responded to by the appropriate departments, however, time limitations made it impossible to evaluate the identified corrective actions. If appropriate corrective actions were implemented for all the findings, the audit should be beneficial in upgrading the DPC commercial-grade dedication program.

2.9 Management Involvement

Management had played a significant role in the evolution of the commercial-grade dedication at DPC, as well as overall industry evaluation, through participation on the NUMARC Board of Directors and the NUMARC nuclear plant equipment procurement (NPEP) working group. DPC management also was active in the review, assessment, and implementation of the NUMARC CPI. The resources were made available to put together a task force and a review team from several different departments to make recommendations on how to implement the CPI. As discussed earlier in the Executive Summary, DPC management made the decision to phase-in the reevaluation of commercial-grade evaluations in existence as of December 31, 1989, and continued to use these evaluations to dedicate CGIs procured after January 1, 1990. These earlier evaluations did not meet the requirements of the current dedication program, which was supposed to be consistent with the EPRI NP-5652 guidelines. Although, in retrospect, DPC might reconsider this decision, management documented their basis for phasing-in the new program in a position paper and letter dated May 8, 1990, and later updated that position on July 5, 1991.

3 PROCUREMENT TRAINING REVIEW

The team reviewed the indoctrination and training of personnel involved in the procurement and dedication process at DPC, placing particular emphasis on the Design Engineering Department (DED). Formal training was provided to DED personnel involved in the procurement and dedication process when the revised commercial-grade program became effective on January 1, 1990. The team reviewed the training records for applicable DED personnel and verified that the DED personnel had received training on the applicable procedures used for commercial-grade procurement and evaluations.

However, during further review of DPC's training, the team discovered that there were no minimum formal training requirements for DED personnel performing commercial-grade dedication activities. Individuals who joined those groups that performed the CGI evaluations within DED received training on an individual basis from their immediate supervisors. During discussions with the team, DED personnel stated that although there was no documented program that described the minimum training requirements, it was the responsibility of the individual's immediate supervisor to ensure that adequate training was provided before the individual performed any CGI evaluations. DED personnel further stated that although individuals became familiar with applicable procurement procedures when they joined the various groups, the training was primarily on-the-job training. An individual's knowledge of the commercial-grade procurement process and the applicable procedures was determined once they had become involved in preparing CGI evaluations. However, the team concluded that with the different groups within DED performing CGI evaluations, it would be beneficial if minimum training requirements were specified and documented. This would be one way of ensuring consistency in the training provided to individuals within DED involved in the performance of CGI evaluations. The team considered the lack of minimum training requirements a weakness in the DPC dedication program.

4 NUMARC COMPREHENSIVE PROCUREMENT INITIATIVE IMPLEMENTATION

The assessment team reviewed the status of DPC's implementation of the NUMARC CPI as described in NUMARC 90-13, "Nuclear Program Improvements," approved June 28, 1990, by the NUMARC Board of Directors (DPC's Senior Vice President of the Nuclear Production Department is a member of the Board of Directors). This initiative committed licensees to assess their procurement programs and take specific actions to strengthen inadequate programs. The CPI called for licensees to complete their reviews by July 1, 1991, and to complete implementation by July 1, 1992. These guidelines are summarized in the enclosure to a Commission paper, "NUMARC Initiatives on Procurement," (SECY 90-304), dated August 24, 1990.

DPC established a CPI review team in July 1990, with representatives from the Design Engineering Department, Nuclear Production Department, and Quality Assurance. Representatives were later added from the Purchasing and Construction Maintenance Department. The CPI team developed a position paper, issued on March 4, 1991, which summarized DPC's approach for implementation of the CPI. The results of the review and assessment phase were documented in a licensee internal memorandum dated July 5, 1991. The CPI team agreed to develop a new corporate level manual (Nuclear Procurement Engineering Program) to incorporate procedures required to implement the CPI. The following discussion describes DPC's strategy for implementation of the CPI.

4.1 Vendor Audits

Quality Assurance will perform a review of original equipment manufacturers (OEMs) and authorized suppliers to identify those with a proven performance record. The existing audits or other documentation providing a basis for procurement from those suppliers will continue to be used until the audit in effect until July 1, 1992, expires.

For those OEMs and authorized suppliers not having a proven performance record, and for all other suppliers, a list of equipment and materials procured from each will be screened to determine if a performance-based audit (PBA), special inspection and/or testing should be performed. These improvements will be implemented in order to support any procurements made on or after July 1, 1992. The screening process had not been developed as of this assessment, but DPC personnel stated that a screening process will be developed before July 1, 1992.

DPC personnel stated that, when developed, the screening process will be based on vendor history, item complexity and function, and the extent to which other verification methods such as inspection and/or testing will be performed. PBAs will be used, as appropriate, for vendor audits performed by other utilities and utility-based auditing organizations such as NUPIC. When a PBA is performed, the items that are more complex and perform a function important to plant safety will be targeted. DPC planned to continue to participate in joint audit activities through NUPIC. All audits will be conducted on a triennial basis. Also, DPC will continue with a source inspection program (during production, testing, or final inspection) to supplement the audit program.

4.2 Tests and Inspections

DPC will use a screening process to determine if special inspection and/or testing is appropriate. The determination will be based on such things as supplier/product performance history, item complexity and function, traceability, type of audit performed on the supplier, other testing routinely performed before putting equipment in service, type of test and test equipment required, and the frequency and/or quality of orders. Special inspection/testing will not normally be performed on products from a supplier that was subjected to a PBA unless the PBA indicates a need for special inspection and/or testing.

Guidelines for fraud detection had been developed. The guidelines, based on Appendix C to EPRI NP-6629, were completed by QA in May 1991. The guidelines will be placed in the Nuclear Procurement Engineering Program Manual. The manual is scheduled to be approved by 1992, and training on the fraud detection guidelines is scheduled to be completed by March 1, 1992.

4.3 Obsolete Items

DPC's current practice was to establish traceability to the OEM when items were procured through channels other than the OEM or an authorized distributor (i.e., surplus market). Procedures were being revised to reflect this practice and to require the performance of tests and/or inspections if traceability to the OEM could not be established.

DPC's acceptable substitutes program identified the process for evaluating replacement items. The procedure for this program will be reviewed by DPC to determine if any improvements are needed.

5 CONCLUSIONS

The licensee had made a significant effort to upgrade its commercial-grade dedication program since its inception in January 1987; however, needs for improvement were identified in a number of areas. Of most significance was DPC's failure to fully implement its new program requirements as of January 1, 1990, for CGIs previously evaluated and listed on the commercial grade items list. DPC decided to phase-in the reevaluation of these CGIs using past program requirements, with completion by December 31, 1991. A specific weakness identified in program implementation was not verifying all characteristics identified as critical.

The assessment team found strengths in areas such as engineering involvement in the dedication process, past and present industry involvement, and overall program consistency with the dedication philosophy described in EPRI NP-5652. Also, DPC's achievements in the area of the review of the NUMARC comprehensive procurement initiatives were acceptable and the quality, experience level, attitude, and dedication of its personnel was evident.

6 EXIT MEETING

On July 19, 1991, the assessment team conducted an exit meeting with members of the DPC staff and management at the Charlotte North Carolina general office. Persons contacted during the assessment are listed in the appendix to this report. During the exit meeting, the team summarized the scope of the assessment and the observations. Throughout the assessment, the team met with DPC management and their staff to discuss concerns. The license did not identify any information as proprietary.

APPENDIX

PERSONS CONTACTED

Duke Power Company

* H. Tucker, Senior Vice President, Nuclear Production
* T. McMeekin, Vice President, Design Engineering
* M. Tuckman, Vice President, Nuclear Operations
* G. Grier, Vice President, Quality Assurance
* K. Caraway, Senior Engineering Supervisor, Design Engineering (DE)
* P. McBride, Engineering Supervisor, DE
R. Harris, Engineering Supervisor, DE
* D. DeMart, Engineering Supervisor, DE
* J. Richards, Senior Engineer, DE
R. Oakley, Senior Engineer, DE
* T. Wyke, Chief Engineer, DE
* J. Peele, Division Project Manager, DE
S. Lindsey, Technical Assistant Manager, Nuclear Production (NP)
* J. Temple, Procurement Supervisor, NP
* P. Herran, Nuclear Maintenance Manager, NP
* S. Benesole, Engineering Supervisor, NP
S. Grier, Engineer, NP
A. Haghi, Engineer, NP
J. Sites, Materials Manager, NP/Oconee
B. Millsaps, Maintenance Engineer, NP/Oconee
* L. Davison, Director, Services General Office, Quality Assurance (QA)
* R. Robinson, QA Manager, Vendors
G. Miller, QA Technical Assistant Manager
* R. Smith, General Manager, Purchasing
C. Ballard, Purchasing Agent, Purchasing
S. Schronce, Material Specialist
M. Wells, Material Specialist

Nuclear Regulatory Commission

* G. Zech, Acting Deputy Director, Division of Reactor Inspection and Safeguards, NRR
* J. Johnson, Deputy Director, Division of Reactor Projects, Region II
* U. Potapovs, Section Chief, Vendor Inspection Branch, NRR
* R. McIntyre, Team Leader, NRR
* S. Alexander, EQ and Test Engineer, NRR
* L. Campbell, Reactor Engineer, NRR
* R. Frahm, Quality Assurance Engineer, NRR
* M. Thomas, Reactor Engineer, Region II

NUMARC

* A. Marion, Manager

*Attended Exit Meeting on July 19, 1991

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

FEB 2 1 1992

Docket No. 99901149/89-01

Mr. Jerry Stern, President
Familian Northwest
Post Office Box 17098
Portland, Oregon 97217

Dear Mr. Stern:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, led by Mr. J. J. Petrosino, of this office on April 10, 1989, and the discussions of the team's findings with you at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers. This NRC concern is discussed in detail in NRC Information Notice (IN) 88-48 and its supplements. Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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 into the NRC's Public Document Room.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901149/89-01

ORGANIZATION: FAMILIAN NORTHWEST
PORTLAND, OREGON

REPORT NO.: 99901149/89-01	INSPECTION DATE: April 10, 1989	INSPECTION ON-SITE HOURS: 3

CORRESPONDENCE ADDRESS: Familian Northwest
Attn: Mr. Jerry Stern, President
Post Office Box 17098
Portland Oregon 97217

ORGANIZATIONAL CONTACT: Mr. Jerry Stern
TELEPHONE NUMBER: (503) 2?3-3333

NUCLEAR INDUSTRY ACTIVITY: Familian Northwest (FNW) supplies plumbing supplies to residential and commercial activities. FNW also supplies commercial valves to Trojan Nuclear Power Plant.

ASSIGNED INSPECTOR: [signature] 25 May 1989
D. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S): R. Moist, Reactive Inspection Section No. 2 (RIS-2)

APPROVED BY: [signature] May 26 1989
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50

B. SCOPE: The purpose of this unannounced inspection was to determine whether FNW had purchased any valves from CMA International, Incorporated of Vancouver, Washington, and to determine if those valves, if any, were supplied to any commercial nuclear plant.

PLANT SITE APPLICABILITY: Trojan.

ORGANIZATION: FAMILIAN NORTHWEST
PORTLAND, OREGON

REPORT NO. 99901149/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

NONE

B. NONCONFORMANCES:

NONE

C. UNRESOLVED/OPEN ITEMS:

NONE

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed.

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discussed a potential problem concerning Vogt 2-inch valves (Vogt figure No. SW13111) which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas & Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington (CMA International, Incorporated). Henry Vogt representatives examined the leaking valves at the Diablo Canyon Nuclear Power Plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) the Vogt name was die-stamped instead of being forged onto the side of the body; (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flanges; (3) the subject valves have swing gland bolting which is not used by the Henry Vogt Company; and (4) the end-to-end dimensions of the valves in question are shorter than the Vogt SW-13111.

ORGANIZATION: FAMILIAN NORTHWEST
PORTLAND, OREGON

REPORT NO. 99901149/89-01	INSPECTION RESULTS:	PAGE 3 of 3

2. Discussion at Familian Northwest (FNW):

Mr. Jerry Sterns, President of FNW stated that FNW has done very little business with Trojan in the past few years and very little with CMA. A review of CMA invoices from 1984 thru 1988 verified that point. All valves supplied to FNW were found to have been supplied to commercial end users.

F. PERSONNEL CONTACTED:

Jerry Stern, President
Jack Renner, Accounts Payable Manager

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

FEB 21 1992

Docket No. 99901160/89-01

Mr. Bill Allen, President
Force and Motion Industries
9106 North East Marx Drive
Portland, Oregon 97220

Dear Mr. Allen:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, conducted by Mr. Randy Moist of this office on April 12, 1989, and the discussions of his findings with members of your staff at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers as discussed in NRC Information Notice (IN) 88-48 and its supplements. Areas examined during the NRC inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's ongoing review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 999 60/89-01

ORGANIZATION: FORCE & MOTION INDUSTRIES
PORTLAND, OREGON

REPORT NO.: 99901160/89-01	INSPECTION DATE: April 12, 1989	INSPECTION ON-SITE HOURS: 1

CORRESPONDENCE ADDRESS: Force and Motion Industries
Mr. Bill Allen, President
9106 NE Marx Drive
Portland, Oregon 97220

ORGANIZATIONAL CONTACT: Mr. Bill Al n
TELEPHONE NUMBER: (503) 256-2

NUCLEAR INDUSTRY ACTIVITY: Force & Motion Industries (FMI) represents Continental Hydraulic Valve, Savage, Minneso FMI does not supply any valves to the nuclear industry.

ASSIGNED INSPECTOR: Randolph N. Moist — 1June89
Moist, Reactive Inspection Section No. 2 (RIS-2) — Date

OTHER INSPECTOR(S):

APPROVED BY: C E Ader for — 89
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50

B. SCOPE: The purpose of this unannounced inspection was to determine if FMI had purchased any valves om CMA International Incorporated of Vancouver, Washington and t determine if those valves, if any, were supplied to any commercial nuclear power plant.

PLANT SITE APPLICABILITY: None identified

ORGANIZATION: FORCE & MOTION INDUSTRIES
PORTLAND, OREGON

REPORT NO.: 9990 50/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed.

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 1988, and Supplement 1 to IN 88-48, dated August 24, 1988 discussed a potential problem concerning Vogt 2-inch valves (Vogt figure No. SW-13111), which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas & Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington (CMA International, Inc.). Henry Vogt representatives examined the valves at the Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) the Vogt name was die-stamped instead of being forged onto the side of the body, (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flanges, (3) the valves in question had swing gland bolting which is not used by the Henry Vogt Company, and (4) the end-to-end dimensions of the valves in question are shorter than the Vogt SW-13111.

2. Discussions at Force and Motion Industries (FMI):

It was determined by the inspector after discussions with Bill Allen, President of FMI, that (1) FMI did no business with nuclear power plants, (2) the only valve company that FMI

ORGANIZATION: FORCE & MOTION INDUSTRIES
PORTLAND, OREGON

REPORT NO.: 99901160/89-01	INSPECTION RESUL	PAGE 3 of 3

represented was Continental Hydraulic Valve out of Savage, Minn, and (3) there were no records in FMI's data base for CMA International.

F. PERSONNEL CONTACTED:

Bill Allen, President

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

FEB 2 1 1992

Docket No. 99901150/89-01

Mr. Leigh Porter, Manager
Grinnell Supply Sales Company
3240 North West 29th Avenue
Portland, Oregon 97210

Dear Mr. Porter:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, conducted by Mr. Joseph J. Petrosino, of this office on April 26, 1989, and the discussions of his findings with members of your staff at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers as discussed in NRC Information Notice (IN) 88-48 and supplements. Areas examined during the NRC inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's ongoing review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901150/89-01

ORGANIZATION: GRINNELL SUPPLY SALES COMPANY
PORTLAND, OREGON

REPORT NO.: 99901150/89-01	INSPECTION DATE: April 26, 1989	INSPECTION ON-SITE HOURS: 2

CORRESPONDENCE ADDRESS: Grinnell Supply Sales Company
3240 N. W. 29th Avenue
Portland, Oregon 97210

ORGANIZATIONAL CONTACT: Mr. Leigh Porter, Manager
TELEPHONE NUMBER: (503) 223-7101

NUCLEAR INDUSTRY ACTIVITY: The Grinnell Supply Sales Company (Grinnell) supplies commerical grade valves and piping components to the Trojan nuclear plant and WPPSS nuclear plant which could be dedicated by the licensees for safety-related use.

ASSIGNED INSPECTOR: [signature] June 2, 1989
J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S):

APPROVED BY: [signature] June 2, 1989
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21.

B. SCOPE: This inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers such as Grinnell. This NRC concern is delineated in NRC Information Notice (IN) 88-48 and Supplement 1 to IN 88-48.

PLANT SITE APPLICABILITY: Trojan (5 -344) and WNP-2 (50-397).

ORGANIZATION: GRINNELL SUPPLY SALES COMPANY
PORTLAND, OREGON

REPORT NO.: 99901150/89-01	INSPECTION RESULTS:	PAGE 2 of 4

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

None

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Entrance and Exit Meetings

The NRC inspector explained the scope of his inspection to Grinnell Supply Sales Company (Grinnell) personnel during meetings on April 26, 1989. The inspector explained that the NRC is concerned about commercial grade valves that are supplied by CMA International, Incorporated (CMA) of Vancouver, Washington to Grinnell, who may in turn supply them to nuclear power plants in its sales areas. The inspector explained that commercial grade valves can typically be purchased as commercial grade and used in a nuclear plant's safety-related systems. The scope of the inspection was also stated over the telephone to Mr. Leigh Porter, Office Manager of the Grinnell-Portland facility. Mr. Porter was on travel during the inspection of his facility.

2. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discussed a potential problem concerning Vogt 2-inch valves (Vogt figure SW-13111), which were leaking steam at the bonnent and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington (CMA International, Incorporated). Henry Vogt representatives examined the leaking valves at PG&E's Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

ORGANIZATION: GRINNELL SUPPLY SALES COMPANY
PORTLAND, OREGON

REPORT NO.: 99901150/89-01	INSPECTION RESULTS:	PAGE 3 of 4

The valves appear to be counterfeit based on the following: (1) The Vogt name was die-stamped on the side of the valve body instead of being forged onto the side of the body; (2) Vogt valves have round bonnent flanges whereas the valves in question have square bonnent flanges; (3) the subject valves have swing gland bolting which is not used by the Henry Vogt Company; and (4) the end-to-end dimensions of the subject valves are shorter than the Vogt SW-13111.

3. Inspection Activities

The Grinnel Material Control Manager provided and assisted the inspector in reviewing all of the Grinnell accounts payable procurement packages regarding CMA International for 1983, 1984, 1985, 1986, 1987 and 1988. During the course of the review, three CMA orders were identified that were shipped to local nuclear power plants, as follows: (1) One 3", 180 psi rated Walworth Model 5202W was supplied to Trojan under Grinnell Shop Order (SO) No. 38704 (CMA Invoice 8827); (2) Two 4", 150 psi rated Walworth Model 5341 F swing check valves were supplied to Trojan under Grinnell SO. No. 2130 (CMA Invoice 8276); and (3) One 3", 300 psi rated Lunkenheimer swing check valve was supplied to WNP-2 under Grinnell SO. No. S052427-AT (CMA Invoice 8065, dated April 11, 1983).

During the course of the procurement package review, three CMA orders of commercial grade valves were identified, which were placed in Grinnell stock. One order, CMA Invoice No. 9783 and 9786, (Approximately May 1985), supplied four 125 psi rated Crane Gate valves: 1-14", 1-16" 1-18" and 1-24". It was determined that Grinnell-Portland received these valves and sent them to Grinnell-Kent, Washington under SO. No. 121978. The second shipment of valves came into Grinnell-Portland under CMA Invoices No. 9525 (November 25, 1987) and 9569 (January 8,1988) and went into Grinnell stock. A total of 12 valves were received. They were all Kitz gate valves, and ranged between 3" and 10". The last order was under CMA Invoice No. 9855 (June 22, 1988), ordered by Grinnell-Portland and shipped to Grinnell-Longview, Washington, SO 5330-LT. This order consisted of four 2" Vogt valves, Model SW 28101. Subsequent to this inspection, Trojan and WPPSS procurement QA personnel were notified of suspect valve order specifics discussed above. During the inspector's review at CMA, several CMA invoices were identified which had questionable shop activities noted in the CMA purchase order packages. These invoices supplied valves to Grinnell under the characterization of "new" or "new surplus." Subsequent to this inspection, Trojan and WPPSS procurement QA personnel were notified of suspect valve order specifics discussed above.

ORGANIZATION: GRINNELL SUPPLY SALES COMPANY
PORTLAND, OREGON

REPORT NO.: 99901150/89-01	INSPECTION RESULTS:	PAGE 4 of 4

During the inspector's review at Grinnell, it was determined that none of the CMA valves listed below ended up in any commercial nuclear power plants. The CMA invoice numbers with the end users and Grinnell SO numbers are identified below. The end users are all located in the Portland area.

CMA Invoice	End User	Grinnell SO. No.
8048	Dillingham Ship Yard	923561
8327	Carnation Food	6535
9164	Wellons	78492
9339	Soutnern Oregon Marine	69473
9260	Wellons	63220/63236
9161	Willamette Industries	77898
9132	Fueten's [Spelling unknown]	74719
9032	Williamette Industries	54748

F. PERSONNEL CONTACTED:

Name	Company/Title
Doris Henderson	Grinnell, Material Control Manager
Kay Dasch	Grinnell, Sales Manager
*Leigh Porter	Grinnell, Office Manager

*Contacted Telephonically

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

MAR 1 1 1992

Docket No. 99900056

Mr. Gregory A. Kurkjain, Jr., President
Henry Pratt Company
4(. South Highland Avenue
Aurora, Illinois 60506-5553

Dear Mr. Kurkjain:

SUBJECT: NOTICE OF NONCONFORMANCE
(NRC INSPECTION REPORT No 99900056/92-01)

This letter addresses the inspection of your facilities at Aurora, Illinois and Dixon, Illinois conducted by Mr. L.L. Campbell and Mr. W.C. Gleaves, of this office on February 3-7, 1992 and the discussions of their findings with you and other members of your staff at the conclusion of the inspection. The inspection was conducted as the result of licensee event reports (LERs) submitted to the Nuclear Regulatory Commission (NRC) by the Arizona Public Service Company which identified deficiencies associated with valves supplied by the Henry Pratt Company (HPC) for the Palo Verde Nuclear Generation Station. The performance based inspection was conducted to evaluate the HPC quality assurance program and its implementation in selected areas such as (1) corrective actions associated with 10 CFR Part 21 notifications submitted by either HPC or NRC licensees, (2) engineering activities performed by HPC and their subcontractors, and (3) HPC's commercial grade dedication program.

Areas examined during the NRC inspection and our findings are discussed in the Enclosure 2 inspection report. The inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors.

During this inspection it was found that the implementation of your quality assurance program failed to meet certain NRC requirements. Although HPC has prepared a procedure which addresses the essential elements of the dedication process, HPC's quality assurance manual and implementing procedures do not contain adequate requirements and interfaces to ensure that all items purchased as commercial grade items (CGIs) for use in safety-related applications are adequately dedicated as basic components. As a result of this program deficiency, HPC supplied some CGIs to NRC licensees as basic components without adequately verifying that the material requirements specified in procurement documents had been met. This inspection also identified instances in which HPC failed to implement its requirements for the segregation and storage of nuclear material. The specific findings and references to the pertinent requirements are identified in the enclosures of this letter.

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

The responses requested by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, 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 its enclosures will be placed in the NRC Public Document Room.

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Nonconformance
2. Inspection Report 99900056/92-01

NOTICE OF NONCONFORMANCE

Henry Pratt Company
Aurora, Illinois

Docket No: 99900057/92-01

During an inspection conducted at the Henry Pratt Company (HPC) facilities in Aurora, Illinois and Dixon, Illinois, on February 3-7, 1992, the inspection team from the U.S. Nuclear Regulatory Commission (NRC) determined that certain activities were not conducted in accordance with NRC requirements, which are contractually imposed on HPC by purchase orders from NRC licensees. The NRC has classified these items, as set forth below, as nonconformances to the requirements of Title 10 of the Code of Federal Regulations, Part 50, (10 CFR 50) Appendix B, imposed on HPC by contract and the supplemental requirements of its nuclear utility customers.

A. Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50 requires that activities affecting quality be accomplished in accordance with a quality assurance program which shall be documented by written policies, procedures and instructions and that activities affecting quality shall be accomplished under suitably controlled conditions which include the use of appropriate equipment including identifying the need for special controls, processes, test equipment, tools and skills to attain the required quality, and for verification of quality by inspection and test. In addition, Criterion III, "Design Control," and Criterion VII, "Control of Purchased Material, Equipment, and Services," of 10 CFR Part 50, Appendix B, require that for items intended for use in safety-related applications, the important design, material, and performance characteristics be identified, acceptance criteria be established, and reasonable assurance be provided that the items conform to the acceptance criteria.

Contrary to the above, the HPC Quality Assurance Manual and implementing procedures did not provide sufficient requirements or interfaces to ensure that commercial grade items (CGIs) dedicated as basic components would be adequately verified to be capable of performing their safety functions. As a result of this program deficiency, HPC procured replacement valve spirol pins and squeeze pivot segments as commercial grade items and supplied them as basic components for use in safety related applications and did not perform any activity to ensure that the material met the requirements specified by its nuclear utility customers (92-01-01).

B. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50 requires, in part, that activities affecting quality be accomplished in accordance with instructions, procedures, or drawings.

Section 12.0, "Process Control," of the HPC Quality Assurance Manual, Issue 3, Revision 5, dated October 2, 1990; Section 6.0 of HPC Procedure QAP-3, "Receiving Inspection for Nuclear Projects," Revision H, dated February 9, 1985; and Section 6.0 of HPC Procedure QAP-24, "Control of Nuclear Material Purchased as Stock Items," Revision E, dated January 13, 1984, require, in part, that nuclear material be inspected and appropriate reviews and inspections be conducted such as the review of certified material test reports and other documentation, verification of identification, performance of dimensional checks and other inspections required by the receiving checklist. Accepted nuclear items are then required to be stored in the Nuclear Storeroom, when not released directly to production, in a segregated part number bin with sufficient identification to maintain traceability.

Contrary to the above, one bin of type 302 stainless steel spirol pins contained pins that were type 420 stainless steel material. Additionally, one bin of ASME Section III, Class 2, bolts certified as SA 193, Grade B7, contained bolts that were marked both B7 and A-325. (92-01-02)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, 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 11th day of March, 1992.

ORGANIZATION: Henry Pratt Company Aurora, Illinois

REPORT NO. 99900056/92-01

CORRESPONDENCE ADDRESS: Mr. Gregory A. Kurkjain, Jr., President
Henry Pratt Company
401 South Highland Avenue
Aurora, Illinois 60506-5563

ORGANIZATIONAL CONTACT: Mr. Bruce R. Cummins, Quality Assurance Director

NUCLEAR INDUSTRY ACTIVITY: Manufactures and supplies valves and valve parts for commercial nuclear power plants.

INSPECTION CONDUCTED: February 3-7, 1992

L.L. Campbell, Team Leader — 03/05/92 Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (VIB)

OTHER INSPECTORS: W.C. Gleaves, VIB

APPROVED:

Uldis Potapovs, Chief — 03-09-92 Date
Reactive Inspection Section No. 1
Vendor Inspection Branch

INSPECTION BASES: 10 CFR Part 21 and Part 50, Appendix B

INSPECTION SCOPE: To review and evaluate the Henry Pratt Company (HPC) quality assurance (QA) program and its implementation in selected areas such as (1) corrective actions associated with 10 CFR Part 21 reports; (2) Engineering activities; and (3) HPCs commercial grade dedication program.

PLANT SITE APPLICABILITY: Numerous.

1 INSPECTION SUMMARY

1.1 Nonconformances

1.1.1 Contrary to Criteria II, III and VII of Appendix B to 10 CFR Part 50, the Henry Pratt Company (HPC) quality assurance (QA) manual and implementing procedures did not provide sufficient requirements or interfaces to ensure that commercial grade items (CGIs) dedicated as basic components for use in safety-related applications would be adequately verified to be capable of performing their safety functions. As a result of this program deficiency, HPC procured replacement valve spirol pins and squeeze pivot segments as CGIs and supplied them as basic components without performing any activity to ensure that the material met the requirements specified by its nuclear utility customers (Nonconformance 92-01-01, see Sections 3.4.2 and 3.4.3 of this report).

1.1.2 Contrary to Criterion V of Appendix B to 10 CFR Part 50, the HPC QA Manual, and HPC procedures QAP-3 and QAP-24, one bin in the Nuclear Storeroom contained an accepted batch of type 302 stainless steel spirol pins. Three pins in this bin were type 420 stainless steel material. Additionally, one bin of accepted ASME Section III, Class 2 bolts, certified as SA 193, Grade B7, contained bolts that were marked B7 and A-325 (Nonconformance 92-01-02, see Sections 3.4.4 of this report).

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Nonconformance 99900056/83-02, Item B.5 (Closed)

Nonconformance 83-02, Item B.5, stated that contrary to Criterion V of Appendix B to 10 CFR Part 50 and Paragraph 8.5.4.2 of the Quality Assurance Manual, "Policies and Procedures-Rejected Material Report," No. 1350-902.0 had been implemented and had not been approved by the Quality Assurance Manager or a department manager.

The NRC inspectors determined that HPC processes nonconformances in accordance with Section 5.5.3, "Nonconformities and Corrective Action," of their Quality Assurance (QA) Manual, Revision 4, dated and approved on March 14, 1990. Based on the facts that HPC presently processes nonconformances in accordance with approved QA Manual requirements and that during the inspection the NRC inspectors observed implementation of HPC's program for controlling nonconformances by HPC processing and documenting the nonconformances identified in Section 1.1.2 of this report, the NRC inspectors closed Nonconformance 83-02, Item B.5.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on February 3, 1992, the NRC inspectors discussed the scope of the inspection, outlined areas of concern, and established interfaces with HPC management and staff. In the exit meeting on February 7, 1992, the NRC inspectors discussed their findings and concerns with HPC's management and staff.

3.2 Henry Pratt Company's Actions Relative to Licensee Event Reports (LERs)

The NRC inspectors reviewed HPC's actions relative to three LERs submitted to the NRC by the Arizona Public Service Company (APS) which identified deficiencies associated with valves supplied by HPC for the Palo Verde Nuclear Generation Station (PVNGS). The issues reviewed by the inspectors and the associated actions taken by HPC are discussed below.

3.2.1 LER 89-013-01, "Unqualified Containment Purge Isolation Valves"

LER 89-013-01 was initiated when two PVNGS Unit 1 containment purge isolation valves were determined by APS personnel to not be installed in accordance with their environmental qualification reports. The containment purge isolation valves were found to have handwheels installed on the manual jacking screws and the hand wheels were not believed to have been included in the seismic analysis for the valves.

The NRC inspectors discussed LER 89-013-01 with HPC personnel during the inspection and reviewed the following documents in order to determine if the containment purge valves' seismic qualification included the installed handwheels:

- Purchase Order (PO) #U-0264` HPC to G. H. Bettis Corporation dated November 26, 1984
- Blueprint N521-SR80-M3HW-CW, Rev. A, G. H. Bettis Corporation Actuator issued December 19, 1984
- Seismic Stress Report for 8", NRS w/N521-SR80-M3HW, Nuclear Class 2, Section III of the ASME Boiler and Pressure Vessel Code, Rev. 3, dated October 30, 1985
- Certificate of Compliance from G. H. Bettis Corporation to Henry Pratt Company, for HPC PO #U-02643
- Sales Order #84-9021-0A, G. H. Bettis Corporation
- Price Quotation QD8410-2123, G. H. Bettis Corporation to HPC, dated October 26, 1984
- G.H. Bettis Corporation Quality Assurance Manual, QAM-1276-100, dated March 26, 1979
- HPC Audit of G. H. Bettis Corporation, dated August 21, 1991

The NRC inspectors determined that the HPC revised seismic analysis did include the handwheels. Following a review of the above listed documents, the NRC inspectors determined that, at the time of the issuance of LER 89-013-01 all necessary information supporting the seismic qualification of the replacement actuators with the handwheel installed was available at HPC. The NRC inspectors believe that APS personnel did not perform an adequate background evaluation before they issued the LER as evidenced by the fact that APS did not contact HPC, the original supplier of the actuator in question, prior to initiating the LER. Subsequent to the inspection, HPC informed the NRC inspectors that G. H. Bettis Corporation had forwarded an analysis to them that specifically addressed the seismic qualification of the actuator (in question) with the handwheel installed.

3.2.2 LER 89-018-001, "Henry Pratt Valve Failures"

LER 89-018-001 was initiated by APS as the result of two PVNGS Unit 3 containment purge valves failing to meet the local leak rate test (LLRT) acceptance criteria. The valves were 42 inch butterfly valves manufactured by HPC. A root cause failure analysis performed by APS revealed that the valves had failed their LLRT due to excessive leakage caused by intergranular fracture in stainless steel spirol pins that fixed the position of a thrust bearing collar in the valves. The pins were manufactured from type 420 stainless steel, which is known to be susceptible to hydrogen embrittlement. Subsequent evaluation of the PVNGS valve population revealed that similar pins were installed in other valves in the containment purge system, nuclear cooling water system, and essential cooling water system.

The NRC inspectors discussed LER 89-018-001 with Henry Pratt personnel during the inspection and reviewed the following documents:

- Calculation Sheet, Henry Pratt Co. 42"-1200 Series Thrust Bearing Pin Load, Ref. #D-01184, dated October 25, 1989

- Derivation of Calculation for 42"-1200 Series Thrust Bearing Pin Load, Ref. #D-0118-405, dated October 27, 1989

- Thrust Bearing Pin Analysis, Henry Pratt Co., Ref. #D-0114, Sheets No. 6 and No. 7, dated November 1, 1989

The NRC inspectors evaluated the corrective actions taken by HPC regarding the spirol pin embrittlement in the HPC supplied valves. Corrective actions taken by HPC included the issuance of a letter to all licensees that received HPC valves with type 420 stainless steel spirol pins recommending that they be replaced with type 302 stainless steel spirol pins. HPC also performed an engineering analysis confirming that the type 302 stainless steel spirol pin is an acceptable substitute. The NRC inspectors concluded that all licensees which received HPC valves with type 420 spirol pins have been adequately notified by HPC of the problem, as required by 10 CFR Part 21.

3.2.3 LER 90-005-00, "Spray Pond Cross Connection Valve Failure Due to Material Misapplication by Henry Pratt Company"

LER 90-005-00 was submitted to the NRC by APS as the result of failures of the PVNGS Unit 1 essential spray pond cross connect valves to operate on demand. Further investigation by APS revealed that for all six cross connect valves at PVNGS, the keys that secured the valve stem to the operator torque tube were either partially or completely corroded. The keys, originally supplied by HPC in the valve assembly, were carbon steel and not suitable for the spray pond environment. The original carbon steel keys were replaced with stainless steel keys.

The NRC inspectors discussed corrective actions taken by HPC as a result of this LER and reviewed the following documents:

- Nuclear Engineering Transmittal Sheet, HPC

- Responsible Engineers Checklist for Nuclear Signoffs, HPC

- Bechtel Design Specifications 13-J-086-325, Rev. 0, and 13-J-086-281 Rev. 1, Arizona Nuclear Power Project

- Blueprint D-0118, Sheets No. 4 and No. 5, 8" through 14" 1400 Series Nuclear Valve Cross-Section and Materials List, ASME Section III Class 3, HPC dated April 28, 1980

The NRC inspectors reviewed the Bechtel design specification for the spray pond cross connect valve which lists continuous submergence in spray pond water as its location and duty. The NRC inspectors were informed by HPC personnel that the incorrect selection of a carbon steel shear key for the spray pond cross connect valve was an incident isolated to the HPC supplied cross connect valves for PVNGS. Following the valve failure at PVNGS, APS corrected the problem by substituting stainless steel keys for the corroded or missing carbon steel keys. HPC personnel informed the NRC inspectors that the transmittal sheets and checklists were generated as part of the corrective action for the LER and are now used to review customer design specifications, arrangement and cross-section drawings, and bills of materials before manufacture and shipment. These supplemental checklists are believed by HPC to be adequate to ensure stricter material selection and control in valve applications. The actions taken by HPC were considered by the NRC inspectors to be sufficient to prevent reccurrence.

3.3 10 CFR Part 21

The NRC inspectors determined that HPC has maintained the required 10 CFR Part 21 postings and a procedure, QAP-33, "Reporting of Defects and Noncompliance for Safety Related Basic Components," Revision D, dated January 6, 1978, which implements 10 CFR Part 21 requirements. HPC informed the NRC inspectors that they have copies of NRC Information Notice 91-76, "10 CFR PARTS 21 AND 50.55(e) FINAL RULES," dated November 26, 1991, and the revised 10 CFR Parts 21 and 50 that became effective October 29, 1991. HPC is preparing a revision to QAP-33 to incorporate the reporting requirements presently contained in 10 CFR Part 21.

3.4 HPC Commercial Grade Item Dedication Program

3.4.1 Methodology

HPC presently has two methods for dedicating items procured as commercial grade items (CGIs) as basic components for use in safety-related applications. One dedication method consists of procuring CGIs from a supplier who has not been audited or surveyed, but whose performance is trended, and then performing a standard receipt inspection normally consisting of a review of documentation and the performance of dimensional, marking, and damage checks. HPC informed the NRC inspectors that the performance of these suppliers is trended and rated based on the reject rate during receipt inspection, and not on the actual performance of the dedicated CGIs after being placed in service. All of the dedication packages reviewed by the NRC inspectors used this dedication methodology.

The second method consists of dedicating the CGIs in accordance with a process that, in general, was consistent with the dedication philosophy described in EPRI-NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)," June 1988. This dedication process was formalized in March 1991 and has been used on a very limited basis and only when specified by the customer's PO.

The dedication methodology consisting of a review of documentation from a supplier that has not been surveyed or audited, performing a receipt inspection for damage and dimensions, and trending the results of the receipt inspections is not considered adequate verifications to ensure that the dedicated item will perform its intended safety-related function. The following paragraphs provide additional discussion on this dedication methodology.

3.4.2 Quality Assurance Program Manual and Implementing Procedures

The NRC inspectors reviewed the following documents and discussed their content with the HPC staff in order to evaluate the process used for dedicating items procured as commercial grade items (CGIs) and supplied to HPC's nuclear customers as basic components:

- Quality Assurance Manual, Issue Three, Revision 5, dated October 2, 1990

- QAP-3, "Receiving Inspection for Nuclear Projects," Revision H, dated February 8, 1985

- QAP-24, "Control of Nuclear Materials Purchased as Stock Items," Revision E, dated January 13, 1984

- QAP-40, "Dedication of Safety Related Commercial Grade Replacement Parts for Use in Nuclear Power Plants," Revision 0, dated March 27, 1991

Although procedure QAP-40 addresses essential elements of the dedication process such as defining the item's safety function and failure modes, a listing and discussion of the item's critical characteristics, and specifying the verification methods and acceptance criteria, Section 2.0 of QAP-40 states that the QAP-40 replacement part dedication process is applicable only when specified by the customer. HPC staff interprets the intent of Section 2.0 to mean that, unless a customer's PO specifically requires HPC to perform a dedication, HPC will not use QAP-40 to control the dedication process. Additionally, even if the customer invokes 10 CFR Part 21 and 10 CFR Part 50, Appendix B, requirements on HPC, but does not specify that HPC will dedicate the item, then HPC would not use QAP-40 to control the dedication process.

The NRC inspectors also reviewed HPC's procedures for the procurement and receipt of nuclear grade, non-Code, stock material and determined that there were no requirements or provisions in these procedures for ensuring that CGIs would be properly dedicated when a customer's PO included 10 CFR Part 21 and 10 CFR Part 50, Appendix B, requirements. QAP-24 controls HPC's procurement of nuclear stock items furnished to customers as basic components. QAP-24 has no requirements for ensuring that the critical characteristics of a CGI are identified and properly verified prior to supplying the CGI as a basic component. Additionally, the NRC inspectors determined that HPC's procedure for receipt inspection contains no requirements for ensuring that a CGI's critical characteristics are verified as part of the dedication process.

The NRC inspectors also determined that the HPC QA Manual has no provisions for the dedication process. Section 10.0, "Procurement Control," of the HPC QA Manual does not require audits or surveys of non-Code safety-related items, except for Class IE equipment such as power actuators, limit switches and solenoid valves. Section 15.0, "Commercial Grade or Stock Items Supplied as Spare or Replacement Parts (Non-Code)," of the HPC QA Manual requires that the Application Engineer prepare a spare parts specification sheet and the order form for the stock or CGI used in nuclear applications with approval by the QA Director, but does not identify any specific controls for the procurement and acceptance of these items. Except for certain Class IE items, the HPC QA manual does not provide adequate requirements to ensure that non-Code items purchased by HPC as CGIs are properly dedicated prior to supplying them as basic components. Section 3.4.3 of this report identifies several customer POs which invoked the requirements of 10 CFR Part 21 and 10 CFR Part 50, Appendix B, and identifies items supplied as basic components and certified by HPC as meeting the requirements of the POs that were inadequately dedicated as the result of inadequate program controls.

The NRC inspectors concluded that both the HPC QA Manual and several implementing procedures, as discussed in this report, failed to contain sufficient requirements to ensure CGIs dedicated as basic components would perform their intended safety functions. Also, based on the previous discussions, the NRC inspectors determined that HPC's implementing procedures controlling the procurement and receipt inspection process did not contain sufficient reference to, use of, or interface with HPC's new dedication procedure, QAP-40, and that as written QAP-40 would only be used when a customer specifically required HPC to dedicate CGIs.
(See Nonconformance 99900056/92-01-01).

3.4.3 Review of HPC Dedication Packages

The NRC inspectors reviewed the customer's PO, HPC's PO to their supplier, HPC's receiving inspection reports and documentation received from their suppliers, and the Certificate of Conformance supplied to HPC's customers for each of the following items.

3.4.3.1 The Cleveland Electric Illuminating Company PO S124603, 4 each, 1/2 x 4 inch, 302 stainless spirol steel pin, HPC Part No. 2109573, supplied for the Perry Nuclear Power Plant in October, 1990. This spirol pin as well as the pins addressed in Sections 3.4.3.2 and 3.4.3.3 of this report perform a safety-related function. Valves supplied by HPC, such as the 42 inch containment purge valves, utilize a rubber seating surface on the disc and a relatively hard seating surface on the body. The disc is installed concentrically inside the valve with the final adjustments performed by HPC prior to shipment. The concentricity is axially adjusted utilizing a thrust bearing stud which screws into the lower valve shaft. After the proper gap adjustments are made, the thrust bearing stud is pinned through the valve shaft utilizing a spirol pin. The spirol pins originally specified by HPC for the thrust bearing stud were type 420 stainless steel and now are specified by HPC to be type 302 stainless steel. Failure of these pins resulted in the containment purge valves failing their local leak rate test (see Section 3.2.2 of this report).

3.4.3.2 Tennessee Valley Authority (TVA) PO RD139533, 25 each, 3/8 x 2 inch spirol pin, HPC Part No. 2117000, supplied for the Watts Bar Nuclear Plant in November, 1990.

3.4.3.3 TVA PO RD137870, 10 each, 1/4 x 3 inch thrust collar (spirol) pin, nickel stainless steel type 302, HPC Part No. 2109248, supplied for the Sequoyah Nuclear Plant in March, 1991.

3.4.3.4 Florida Power & Light Company (FP&L) PO G90933/10770, 60 each, segment for 48 inch valve, HPC Part No. 566121, supplied for the St. Lucie Plant, in April, 1991. According to HPC these segments are used to center thrust bearings and are for nuclear valves of an older design and configuration.

Each customer P.O. for the above items contractually invoked 10 CFR Part 21 and 10 CFR Part 50. In each case HPC procured the items as commercial grade from a supplier who had not been audited or surveyed, and dedicated the items by performing a standard receipt inspection. Since these items are non-Code stock material, the documentation of the receipt inspection consists of a signature by an HPC receipt inspector on the copy of the HPC PO for the item. Although Section 6.2.2 of QAP-3 requires that non-Code parts and material be inspected to verify conformance with POs, drawings and other QA requirements, there were no specific receiving inspection instructions identified for these items. The HPC QA Director and the HPC Chief Inspector informed the NRC inspectors that this type of inspection generally consists of performing damage and dimensional checks.

Each of these items were supplied with an HPC certificate of conformance (COC) certifying that the requirements of the customer's PO, including 10 CFR Part 21 and 10 CFR Part 50, Appendix B, had been met. Based on a review of HPC procurement documents and discussions with the HPC QA Director and Chief Inspector, the NRC inspectors determined that receipt inspection of these items consisted of a review for damage, dimensional checks, and a review of any supplied documentation from the HPC supplier. This receipt inspection was the basis for HPC issuing the COC. Also, it was determined that the suppliers of the pins and valve segments, Spirol International and Harris Casting Company, Inc., respectively, had not been audited or surveyed by HPC. The NRC inspectors concluded that HPC procured these items as CGIs and supplied them as basic components for use in safety-related applications without performing any verifications to ensure that the material met the requirements specified by procurement documents (See Nonconformance 92-01-01).

3.4.4 Review of Accepted Material

The NRC inspectors observed several activities at HPC's Dixon, Illinois, facility including receipt inspection, nondestructive examination (NDE), material control and storage, and corrective action. During the examination of the HPC storage area for accepted nuclear items the NRC inspectors determined that the following items were not in conformance with applicable procurement requirements.

3.4.4.1 The NRC inspectors examined an accepted bin of 99 pieces of 1/2 x 4 inch type 302 stainless steel spirol pins, Lot No. 6-5774/#13080, Part No. 2109573, supplied to HPC by Spirol International in 1990 for use in the 36 inch Model 1100 butterfly valves. HPC supplied 4 pins from this bin to the Perry Nuclear Power Plant (PNPP) in October, 1990. The NRC inspectors determined that 3 of the remaining pins were attracted to a magnet and were a darker color than other pins in the box. HPC prepared a nonconformance report to document this condition and to evaluate for 10 CFR Part 21 reportability. During the inspection HPC had the three pins analyzed and determined that they were type 420 stainless steel. Subsequent to the inspection, HPC informed the NRC inspectors that PNPP verified that the material for the four spirol pins received in October, 1990 was type 302 stainless steel (see Nonconformance 92-01-02).

3.4.4.2 The NRC inspectors examined an accepted bin of 3/4 x 3-3/4 Heavy Hex Head Bolts, SA-193, Grade B7, Part No. 1138643R, manufactured by Texas Bolt Company and supplied to HPC by McJunkin in March, 1981. The NRC inspectors determined that one of these bolts was marked "TEXAS BOLT, A-325, JR47," with identification marks of a Grade 5 bolt. All other bolts in this box were marked "TB, B7, JR47." HPC prepared a nonconformance report to document this condition and to evaluate for 10 CFR Part 21 reportability. Subsequent to the inspection, HPC informed the NRC inspectors that they had completed an analysis using SA-325 in lieu of SA-193, Grade B7, bolts and determined that even though the material allowable strength is decreased, the actual shear and combined stresses at twice the weight, acceleration and pressure are well within the material allowable stresses (see Nonconformance 92-01-02).
The NRC inspectors also observed machining activities, weld filler material

control, and the performance of a liquid penetrant (PT) examination of an 8 inch Model 1200 valve body seat ring groove for FP&L's St. Lucie Power Plant. The PT examination was performed in accordance with PT Procedure No. PT-1, "Liquid Penetrant Examination," Revision 4, dated March 17, 1973. The NRC inspectors concluded that these activities were adequately performed.

4 PERSONNEL CONTACTED

* Gregory A. Kurkjain Jr., President

* Bruce R. Cummins, Director of Quality Assurance

* John V. Ballun, Vice President and Manager of Engineering

Jayne Friel, Chief Inspector

William Sweet, Welding and Paint Foreman

Elizabeth Sweet, Quality Assurance Clerk

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* Attended both the entrance and exit meetings of February 3 and 7, 1992.

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

FEB 21 1992

Docket No. 99901151/89-01

Mr. Elmer Stobel, President
Industrial Valve of Oregon, Incorporated
3615 North West Saint Helens Road
Post Office Box 10720
Portland, Oregon 97210

Dear Mr. Stobel:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, conducted by Mr. Joseph J. Petrosino of this office on April 11, 1989, and the discussions of his findings with you at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers. This NRC concern is discussed in detail in NRC Information Notice (IN) 88-48 and its supplements. Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's ongoing review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901151/89-01

ORGANIZATION: INDUSTRIAL VALVE OF OREGON, INCORPORATED
PORTLAND, OREGON

REPORT NO.: 99901151/89-01	INSPECTION DATE: April 11, 1989	INSPECTION ON-SITE HOURS: 5

CORRESPONDENCE ADDRESS: Industrial Valve of Oregon, Incorporated
3615 NW Saint Helens Road
P. O. Box 10720
Portland, Oregon 97210

ORGANIZATIONAL CONTACT: Elmer Stobel, President
TELEPHONE NUMBER: (503) 223-2202

NUCLEAR INDUSTRY ACTIVITY: Industrial Valve of Oregon supplies commercial grade valve and piping components to the commercial nuclear industry as well as to the petro-chemical and paper products industry in the Northwest.

ASSIGNED INSPECTOR: J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) — 24 May 1989 — Date

OTHER INSPECTOR(S):

APPROVED BY: E. T. Baker, Chief, RIS-1, Vendor Inspection Branch — 24 May 1989 — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50.

B. SCOPE: The purpose of this unannounced inspection was to determine if Industrial Valve of Oregon has purchased any valves from CMA International, Incorporated of Vancouver, Washington and to determine if those valves, if any, were supplied to any commercial nuclear plant.

PLANT SITE APPLICABILITY: None identified.

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed

E. Other Comments and Observations:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discussed a potential problem concerning 2-inch valves (Vogt Figure SW-13111), which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington [CMA International, Incorporated (CMA)]. Henry Vogt representatives examined the leaking valves at the Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

The valves appeared to be misrepresented based on the following: (1) The Vogt name was die-stamped instead of being forged onto the side of the body, (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flange (3) the subject valves have swing gland bolting which is not used by the Henry Vogt Company, and (4) the end-to-end dimensions of the subject valves are shorter than the Vogt SW-13111.

2. Entrance/Exit Meeting

The NRC inspector met with Mr. Elmer Stobel, President of Industrial Valve of Oregon (IVO) and discussed the scope of the inspection and discussed the results of the inspection at the conclusion of his review. Mr. Stobel states that he and two other partners started IVO in 19[illegible]. The other two partners are J. Blatner and R. Weinstein. Currently IVO conducts business with Portland General Electric and supplies commercial grade products. He also states that IVO and CMA regularly conducted business with each other prior to 1985-1986 but recently have not.

3. Inspection Results

The inspector reviewed IVO accounts payable records regarding CMA for 1984, 1985, 1986, and 1987. Of these years, 1985 contained the most CMA invoices concerning valve procurements, a total of 12. Each of the end users for the CMA products was identified with no exceptions. The end users identified were companies such as: Louisiana Pacific (IVO sales order [SO] 10705 and 10307), Texaco (IVO-SO's 10912/11005), Weyerhaeuser (IVO-SO12748), Scott Paper (IVO-SO-11943) and Publishers Paper of Oregon City (IVO-SO11943). The invoices indicate that CMA has supplied valves as large as a 16-inch check valve to IVO. The CMA invoices for 1985 that were reviewed are as follows:

CMA Invoice No./Date	Valves Supplied	End User
8838 (3/14/85)	1-2" 300# Globe (Pacific model 366-8)	Texaco
	2-1½" 300# Globe (Pacific model 366-8)	
8903 (5/24/85)	1-6" 150# Globe (Powell model 1139)	NW Maine
8830 (3/1/85)	1-1" 600# Globe (Vogt model 22493)	Louisiana Pacific
8822 (2/8/85)	1-1" 600# Globe (Vogt model 22493)	Louisiana Pacific
8832 (3/6/85)	4-1" 600# Globe (Vogt model 22493)	Louisiana Pacific
8844 (3/22/85)	1-4" 600# Gate (Model Unknown)	Texaco

ORGANIZATION: INDUSTRIAL VALVE OF OREGON, INCORPORATED
PORTLAND, OREGON

REPORT NO.: 99901151/89-01	INSPECTION RESULTS:	PAGE 4 of 4

CMA Invoice No./Date	Valves Supplied	End User
8834 (3/8/85)	2-3" 300# Globe (Crane model 151)	Scott Paper
8908 (5/31/85)	1-16" 125# Swing Check (Walworth model 928F)	Pub. Paper (Oregon City)
8911 (6/4/85)	1-1" 600# Globe (Vogt model 22493)	L-P Somoa
8965 (8/6/85)	2-8" Gate (Crane model 33A)	Weyernaeuser
9007 (9/20/95)	1-4" [Type Unknown] (Crane model 47)	Texaco

During this review no nuclear plant end users were identified by the inspector.

F. PERSONNEL CONTACTED:

Elmer Stobel	President
Don Johnson	Office Manager

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

FEB 21 1992

Docket No. 99901152/89-01

Mr. Gary Pass, Manager
Liberty Equipment and Supply Company
2300 East First Street
Vancouver, Washington 98668

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Vancouver, Washington, conducted by Mr. Joseph J. Petrosino, of this office on April 10, 1989, and the discussions of his findings with members of your staff at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers as discussed in NRC Information Notice (IN) 88-48 and its supplements. Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901152/89-01

ORGANIZATION: LIBERTY EQUIPMENT AND SUPPLY COMPANY
VANCOUVER, WASHINGTON

REPORT NO.: 99901152/89-01	INSPECTION DATE: April 10, 1989	INSPECTION ON-SITE HOURS: 6

CORRESPONDENCE ADDRESS: Liberty Equipment and Supply Company
2300 E~st First Street
Vancouver, Washington 98668

ORGANIZATIONAL CONTACT: Mr. Gary Pass, Branch Manager
TELEPHONE NUMBER: (206) 694-5535

NUCLEAR INDUSTRY ACTIVITY: Liberty Equipment and Supply Company does not currently supply any "safety-related" equipment to the nuclear industry but does supply commercial grade valves, piping and components to at least two nuclear plants.

ASSIGNED INSPECTOR: June 2, 1989
J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S):

APPROVED BY: CG Anderson June 2, 1989
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50

B. SCOPE: The purpose of this unannounced inspection was to determine if Liberty Equipment and Supply has purchased valves from CMA International, Incorporated of Vancouver, Washington and to determine if those valves, if any, were supplied to any commercial nuclear plant.

PLANT SITE APPLICABILITY: None identified during inspection.

ORGANIZATION: LIBERTY EQUIPMENT AND SUPPLY COMPANY
VANCOUVER, WASHINGTON

REPORT NO.: 99901152/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discuss a potential problem concerning Vogt 2-inch valves (Vogt Figure No. SW-13111), which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington [CMA International, Inc. (CMA)]. Henry Vogt representatives examined the valves at the Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) the Vogt name was die-stamped instead of being forged onto the side of the body; (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flanges; (3) the valves in question have swing gland bolting which is not used by the Henry Vogt Company; and (4) the end-to-end dimensions of the valves in question are shorter than the Vogt SW-13111.

2. Inspection Results

The inspector reviewed Liberty Equipment and Supply Company (Liberty) records to determine the extent of business conducted with CMA and to determine the end users of any CMA supplied products. The inspector reviewed Liberty's vendor log books for September 1987 through March 1988 to identify any orders made to CMA. The log books

ORGANIZATION: LIBERTY EQUIPMENT AND SUPPLY COMPANY
VANCOUVER, WASHINGTON

REPORT NO.: 99901152/89-01	INSPECTION RESULTS:	PAGE 3 of 3

represented approximately 29,500 orders for that time period and only one order to CMA was identified. The inspector also reviewed all CMA supplied invoices for 1987, 1988, and 1989. Three additional CMA invoices were identified.

A review of each of the CMA invoices was performed to determine the end user of the CMA product. The record review did not identify any commercial nuclear plants which were supplied the suspect products. The end users of the CMA products were: Alaska Tank Fabricators, Intersox (Brown & Root), Rust Engineering, and American Pacific.

F. PERSONNEL CONTACTED:

Gary Pass, Branch Manager
Fred Reeves, Sales Representative

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

January 24, 1992

Docket No. 99901235

Mr. Gerhard Liesegang, Chairman
Lisega GmbH
Industriegebiet Hochkamp
Postfach 1340
D-2730 Zeven, Germany

Dear Mr. Liesegang:

SUBJECT: NOTICE OF NONCONFORMANCE
(NRC INSPECTION REPORT NO. 999012. -01)

This letter addresses the inspection of your facility at Zeven, Germany, conducted by Mr. Richard P. McIntyre and Mr. Uldis Potapovs of this office on August 20-23, 1991 and the discussions of their findings with you and other members of your staff at the conclusion of the inspection. The purpose of the inspection was to evaluate Lisega's quality assurance program, including the control and audit of subvendors, material certification, material procurement, storage and traceability of nuclear material, the upgrading of stock material per the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and the review of Lisega's 10 CFR Part 21 evaluation process.

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

During this inspection it was found that the implementation of your QA program failed to meet certain NRC requirements which are summarized as follows: (1) improper certification of supplied nuclear components as meeting the requirements of ASME Code, Section III, Subsection NF, and (2) inadequate warehousing and storage of nuclear grade material in order to maintain proper traceability. This inspection also identified two unresolved items concerning the upgrading of stock material per specific ASME Code requirements and the use of certain ASME Code cases.

Please provide us within 30 days from the date of this letter a written statement containing: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have be r will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed. 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 Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC's Public Document Room.

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

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
Notice of Nonconformance
Inspection Report No. 99901235/91-01

ENCLOSURE 1

NOTICE OF NONCONFORMANCE

Lisega GmbH
Zeven, Germany

Docket No. 99901235

During an inspection conducted at the Lisega GmbH facility in Zeven, Germany, on August 20-23, 1991, the inspection team from the United States Nuclear Regulatory Commission (NRC) determined that certain activities were not conducted in accordance with NRC requirements, which are contractually imposed on Lisega GmbH by purchase orders from NRC licensees. The NRC has classified these items as set forth below, as nonconformances to requirements of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50) Appendix B imposed on Lisega GmbH by contract and the supplemental requirements of its nuclear utility customers.

A. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires that measures be established for the selection and review for suitability of materials, parts, equipment, and processes that are essential to the safety-related functions of structures, systems, and components.

The ASME Boiler and Pressure Vessel Code is the basis for demonstrating suitability for application of component supports and hydraulic shock absorbers (snubbers) supplied by Lisega GmbH to the Georgia Power Company for installation in the E.I. Hatch Nuclear Plant.

Contrary to the above, the material and test documentation for several components which were certified by Lisega GmbH as meeting the requirements of ASME Code, Section III, Class 1, Subsection NF did not support this certification or did not correctly represent significant material parameters. Specifically:

1. Lisega Certificate 114 083 for SA-312, type 316 pipe showed product analysis for nickel to be outside the permissible range for this material (10.69% vs 11.0% minimum) and showed no evidence that a flattening test had been performed as required by the material specification.

2. Lisega Certificate 112 720 for ASTM A 500, Grade C pipe did not include manganese in the heat or product analysis. A 500 limits manganese content to 1.4% maximum. Also, there was no evidence that a flattening test had been performed as required by the material specification. Lisega certification 113 197 for the same material did not reference the material heat number or contain heat analysis. Check analysis showed manganese contents as 1.6% which exceeds the ASTM A 500 limit.

3. Lisega Certificate 110 743 showed Charpy-V impact tests as having been performed at - 20°C. The supporting documentation (vendor's certificate), however, indicated that these tests were performed at +10°C.

4. Lisega Certificate 110 025 for SA-182 F6a Class 2 material did not describe heat treatment as required by the applicable material specification and a Lisega check analysis showed the chromium content of this material as 13.8%, which is outside the referenced material specification limits. Additionally, the material supplied under this certification was a 30 millimeter (mm) diameter round bar, whereas material specification SA-182 covers forged or rolled alloy steel pipe, flanges, forged fittings and valves and valve parts. SA-182 references specification A 479/479M or A 739 for bars and products machined from bar stock.

5. Lisega Certificate 111 770 for SA-675 Grade 70 material reported impact test values referencing Charpy-V specimens. The supporting documentation (vendors certificate), however, reported the same impact test values referencing ISO-V specimens.

6. Lisega Certificate 112 178 certifies material to ASME SA-299, "Specification for Pressure Vessel Plates, Carbon Steel, Manganese-Silicon." The material supplied, however, was round bar, 70 mm x 6000 mm. Chemical analyses and physical properties were in compliance with the ASME specification.

7. Lisega Certificate 112 739 certifies the material to ASME SA-696C. This specification covers hot-wrought and cold finished special quality carbon steel bars. The material supplied, however, was a chrome-molybdenum-vanadium alloy steel. The material met the specified SA-696 chemical requirements, but contained significant alloying elements not referenced in the material specification.

B. 10 CFR Part 50, Appendix B, Criterion VIII, "Identification and Control of Materials, Parts, and Components," requires that measures be established to assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.

The Lisega Quality Assurance Manual, Section 8, "Marking and Identification," and Quality Assurance Program Procedure No. 20, "Marking and Identification," describes the procedures for marking and identification of material, components, and products to ensure traceability and identification during material receipt, storage, fabrication, and shipment. These documents state, in part, "If marking is not applied on individual components for operational or fabricational reasons, these parts must be kept on stock separately and identified by appropriate documents (drawing or accompanying card or material slip) ensuring an unobjectionable traceability. The identification established for a certain material lot, in this case must be stamped on these papers.... For material of particular specification and condition, which will be cut into smaller pieces for processing, the marking will be transferred to the individual components by Q.C. personnel."

Contrary to the above, several open bundles of nuclear grade steel material were identified in the front parking lot laydown area that did not contain the original nuclear grade material tagging. The individual material pieces in the bundles had not been stamped by QC to maintain unobjectionable identification and material traceability. Also, several yellow striped nuclear material tags were loose in the laydown area. A nonconformance report was initiated by Lisega when this issue was identified by the inspectors.

Dated at Rockville, Maryland
this 24th day of January, 1992

ENCLOSURE 2

ORGANIZATION:	Lisega GmbH Zeven, Germany
REPORT NO.:	99901235/91-01
CORRESPONDENCE ADDRESS:	Mr. Gerhard Liesegang, Chairman Lisega GmbH Industriegebiet Hochkamp Postfach 1340 D-2730 Zeven, Germany
ORGANIZATIONAL CONTACT:	Mr. Herbert Bardenhagen, Manager of Quality Assurance
NUCLEAR INDUSTRY ACTIVITY:	Nuclear pipe support components including hydraulic shock absorbers (snubbers), constant hangers and other component standard supports.
INSPECTION CONDUCTED:	August 20-23, 1991

Richard P McIntyre — 1/17/92

Richard P. McIntyre, Team Leader — Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (VIB)

Uldis Potapovs — 1-22-92

Uldis Potapovs, Chief — Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (VIB)

INSPECTION BASES:	10 CFR Part 21 and Part 50, Appendix B
INSPECTION SCOPE:	To review Lisega's Quality Assurance program relative to the manufacture and supply of pipe support components including hydraulic shock absorbers.
PLANT SITE APPLICABILITY:	Numerous, including E.I. Hatch, San Onofre, Palo Verde, Nine Mile Point, Catawba, McGuire, St. Lucie, and Millstone.

1.0 INSPECTION SUMMARY

1.1 Nonconformances

1.1.1 Contrary to 10 CFR PArt 50, Appendix B, Criterion III, "Design Control," the material and test documentation for several components that were certified by Lisega GmbH as meeting the requirements of ASME Code Section III, Subsection NF, Class 1, did not support this certification or did not correctly represent the material. Specifically: (90-01-01)

1. Lisega Certificate 114 083 for SA-312, type 316 pipe showed product analysis for nickel to be outside the permissible range for this material (10.69% vs 11.0% minimum) and showed no evidence that a flattening test had been performed as required by the material specification.

2. Lisega Certificate 112 [illegible] for ASTM A 500, Grade C pipe did not include manganese in the heat or product analysis. A 500 limits manganese content to 1.4% maximum. Also, there was no evidence that a flattening test had been performed as required by the material specification. Lisega certification 113 197 for the same material did not reference the material heat number or contain heat analysis. Check analysis showed manganese contents as 1.6% which exceeds the ASTM A 500 limit.

3. Lisega Certificate 110 743 showed Charpy-V impact tests as having been performed at -20°C. The supporting documentation (vendor's certificate), however, indicated that these tests were performed at +10°C.

4. Lisega Certificate 110 025 for SA-182 F6a Class 2 material did not describe heat treatment as required by the applicable material specification and a Lisega check analysis showed the chromium content of this material as 13.8%, which is outside the referenced material specification limits. Additionally, the material supplied under this certification was a 30 millimeter (mm) diameter round bar, whereas material specification SA-182 covers forged or rolled alloy steel pipe, flanges, forged fittings and valves and valve parts. SA-182 references specification A 479/479M or A 739 for bars and products machined from bar stock.

5. Lisega Certificate 111 770 for SA-675 Grade 70 material reported impact test values referencing Charpy-V specimens. The supporting documentation (vendors certificate), however, reported the same impact test values referencing ISO-V specimens.

6. Lisega Certificate 112 178 certifies material to ASME SA-299, "Specification for Pressure Vessel Plates, Carbon Steel, Manganese-Silicon." The material supplied, however, was round bar, 70 mm x 6000 mm. Chemical analyses and physical properties were in compliance with the ASME specification.

7. Lisega Certificate 112 739 certifies the material to ASME SA-696C. This specification covers hot-wrought and cold finished special quality carbon steel bars. The material supplied, however, was a chrome-molybdenum-vanadium alloy steel. The material met the specified SA-696 chemical requirements, but contained significant alloying elements not referenced in the material specification.

1.1.2 Contrary to 10 CFR Part 50, Appendix B, Criterion VIII, "Identification and Control of Material, Parts, and Components," several open bundles of nuclear grade steel material were identified in the front parking lot laydown area that did not contain the original nuclear grade material tagging. The individual material pieces in the bundles had not been stamped by QC to maintain unobjectionable identification and material traceability. Also, several yellow striped nuclear material tags were loose in the laydown area. A nonconformance report was initiated by Lisega when this issue was identified by the inspectors. (91-01-02)

1.2 Unresolved Items

1.2.1 ASME Code Cases N71-15 (for ASTM A 500, Grade C material) and N249-9 (for ASTM A 668, Grade F material) have been used by Lisega for ASME Code material supplied to the Georgia Power Company. Lisega had not determined whether Georgia Power Company had approved the use of these code cases. (91-01-03)

1.2.2 In upgrading stock material, Lisega was not performing a product analysis on each piece of the material as required by ASME NC-3867.4(e). (91-01-4)

2 BACKGROUND

Lisega GmbH is a manufacturer and supplier of pipe support components including hydraulic shock absorbers (snubbers), constant hangers, variable spring hangers, pipe clamps, and other component standard supports. The snubbers are manufactured, assembled, and tested in Bondoufle, France. Lisega (Germany) retests the snubbers upon receipt at the Zeven plant and then certifies them under their American Society of Mechanical Engineers (ASME) Quality System Certificate (QSC) for ASME Code orders. Lisega GmbH also supplies non-ASME code snubbers and other product line components to United States (U.S.) utilities for safety-related nuclear applications under their 10 CFR Part 50, Appendix B quality assurance (QA) program. Involvement with U.S. utilities has been ongoing since 1986-87 and Lisega has been audited and approved by several U.S. utilities. Lisega was issued a QSC as a material supplier (MS) by ASME on September 9, 1990. Lisega has supplied hydraulic shock absorbers (snubbers) and other pipe support components to numerous U.S. utilities as both ASME Code and 10 CFR Part 50, Appendix B items in the last couple of years.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Material Procurement and Certification

Selected material certifications and supporting documentation related to recently completed Georgia Power Company purchase orders were reviewed along with the applicable Lisega procedures to verify the adequacy of the Lisega material procurement, upgrading, and certification process. This review identified several instances in which the material did not fully conform to the referenced specification requirements and resulted in the identification of Nonconformance 91-01-1. Several of the examples identified in Nonconformance 91-01-01 appeared to be related to force-fitting specific materials manufactured to European standards into materials and product forms recognized in the ASME Code. In some cases, this resulted in identifying material purchased to European specifications as bar stock with ASME specification for plate steel and identifying material purchased as alloy steel with ASME specification for carbon steel. Although in all cases reviewed, the mechanical and chemical properties specified for the ASME materials were met or exceeded, the fact that the actual material supplied was of a different product form or contained significant amounts of alloying elements not required or identified in the ASME specification, could cause potential problems when considerations such as weldability are taken into account.

Lisega representatives explained that their equipment design and manufacturing processes were developed using materials purchased to European specifications and that it was sometimes difficult to reconcile these with materials listed in Section II of the ASME Code, while still meeting more stringent impact testing requirements of the European specifications.

It was also noted that in at least two instances material used for Georgia Power Company components was based on the application of specific ASME Code Cases. However, the respective Lisega material certification did not reference the use of a code case nor was there any evidence that Georgia Power Company had been informed of the use of these code cases. This issue was identified as Unresolved Item (91-01-03).

The upgrading of stock material is addressed by Quality Assurance Program Procedure No. 17, "Approval Procedures for Material According to Lisega Specifications," Revision 0. This procedure requires chemical analysis to be performed on a sample from each heat of stock material. Record review indicated that this procedure was generally followed for all code upgraded material supplied to Georgia Power Company. This does not appear to be consistent with paragraph NC 3867.4(e) of the ASME Code which requires a product analysis on each piece of stock material and was identified as Unresolved Item (91-01-04).

3.2 10 CFR Part 21 Evaluation Process

The inspectors reviewed Quality Assurance Program Procedure No. 34, "State of Product Information and Report," Revision 0, dated April 1989. This is the procedure that implements the requirements of 10 CFR Part 21 and "is to be established to ensure report about product failure and insufficiency, as well as quick corrective action."

The inspectors explained to Lisega that the procedure talks about quick corrective action and immediately informing affected customers, but it does not address the actual evaluation of deviations for generic applicability. The whole Lisega review process is handled by writing a nonconformity report that identifies the appropriate corrective action but does not evaluate the problem from a generic standpoint to determine if it affects other customers, past purchase orders, and purchase orders currently being processed.

Lisega stated that the audit findings identified during the June 1991, ASME audit were evaluated for Part 21 reportability as part of the five Nonconformity Reports (Nos. 03060-6-2491 through 03064-6-2491). No Part 21 evaluation was documented to specifically determine the generic implications and if other U.S. customers were affected. The inspectors stated that Procedure No. 34 and the Part 21 evaluation process needed to be clarified to address its purpose, to address the evaluation for possible generic implications, and to address the documentation requirements.

The five nonconformity reports and their corrective actions appeared to adequately address the ASME audit findings from a QA programmatic and technical stand point, but only addressed the Georgia Power Company Purchase Order No. 7005009 for the E.I. Hatch Nuclear Power Plant. A subsequent internal audit of Lisega GmbH's QA program conducted by Lisega USA on July 3-4, 1991, determined that Southern California Edison should also be notified of the audit findings for purchase order 6T051006 for San Onfre Nuclear Generating Station (SONGS). The supposed Part 21 evaluation done as part of the nonconformity report closure did not completely address the generic implications for these specific nonconformity reports, as evidenced by the notifiction to SONGS.

3.3 Identification and Control of Materials, Parts and Components

As part of this inspection, the inspectors conducted a walkdown of the manufacturing facilities, the receipt inspection area, the warehouse, and the material laydown areas. The work stations in the manufacturing plant, the receipt inspection area, and the warehouse were well maintained, very clean, and appeared to be operating very efficiently. Due to a limitation in space, a parking lot in the front of the Lisega building was being used as a material laydown area. During a walkdown of this area the inspectors noticed several yellow striped material identification tags not attached to any material or bundles in the general area. Yellow striped tags identify the material as nuclear grade. Upon further inspection, the inspectors also noticed that several bundles of steel material had been opened and did not have any tagging on the bundle. Also, the individual pieces had not been stamped by QC to maintain unobjectionable identification and traceability as required by Quality Assurance Porgram Procedure No. 20, "Marking and Identification." When this issue was brought to the attention of Lisega, the QA manager stated a nonconformity report would be written to review and correct the problem. Nonconformance (91-01-02) was identified during this part of the inspection.

4 PERSONNEL CONTACTED

G. Liesegang, Chairman
H. Hardtke, President
H. Aberle, Foreign Sales
H. Bardenhagen, QA Manager
W. Wagner, Purchasing Manager
T. Löffler, Manager
G. Lider, Manager
F. Bernert, Manager

All Personnel listed above attended the exit meeting on August 23, 1991.

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

January 15, 1992

Docket No. 99900772

Mr. Ron B. Politi
Vice President and General Manager
NEI Peebles - Electric Products, Inc.
17045 Euclid Avenue
Cleveland, Ohio 44112

Dear Mr. Politi:

SUBJECT: INSPECTION OF A SAFETY-RELATED POWER GENERATOR SUPPLIED TO DIABLO CANYON NUCLEAR POWER PLANT UNIT 2 (NOTICE OF NONCONFORMANCE AND INSPECTION REPORT NO. 99900772/91-01)

We are transmitting the report of the U.S. Nuclear Regulatory Commission (NRC) inspection conducted August 5 through 9, 1991, at the facility of NEI Peebles - Electric Products, Inc. (P-EP) in Cleveland, Ohio. Messrs. Steven M. Matthews, Walter P. Haass, and Michael R. Snodderly of the NRC's Office of Nuclear Reactor Regulation evaluated P-EP's production of a power generator for an NRC licensee, Pacific Gas and Electric Company (PG&E). At the conclusion of this inspection, the NRC's inspection team discussed the inspection findings with you and other members of your staff.

The NRC's team conducted this inspection to assess P-EP's compliance with the NRC requirements imposed in PG&E's purchase order for a power generator. The power generator was for PG&E's new (No. 2-3) emergency diesel generator (EDG) set for the Diablo Canyon Nuclear Power Plant Unit 2 (DCNPP2). P-EP certified that the generator supplied to DCNPP2 was produced in compliance with the NRC's requirements in Appendix B to Title 10, of the Code of Federal Regulations, Part 50 (10 CFR Part 50), and the reporting requirements of 10 CFR Part 21. During the timeframe of this inspection, P-EP was also performing the design and procurement activities for a safety-related power generator for Washington Public Power Supply System's Nuclear Project 2 (WNP2). Although the team focused its inspection activities on the completed power generator for PG&E's DCNPP2, the concerns discussed in this report may have generic implications for WNP2's power generator and any other power generator, or spare and replacement parts, purchased by other licensees. The inspection was conducted to evaluate P-EP's quality program and its implementation in selected areas, such as the control of (1) design processes and interfaces, (2) purchased materials and equipment, and (3) instructions, procedures, and drawings.

As a result of this inspection, a Notice of Nonconformance has been issued to P-EP as Enclosure 1. The inspection report, Enclosure 2, includes a discussion of the areas examined during the inspection and our findings. This inspection consisted of an examination of procedures and representative records, interviews with P-EP's staff, and observations by the team.

The most significant inspection finding was P-EP's failure to demonstrate reasonable assurance (1) that the items specified as critical by PG&E met the quality and reliability requirements of Appendix B to 10 CFR Part 50 and (2) that the critical characteristics of such items have been adequately verified and that the items are capable of performing their design and safety-related functions. Specifically, P-EP failed to demonstrate an adequate verification of the critical characteristics (1) of the items specified as critical that Peebles Electrical Machines (PEM) procured as commercial grade and (2) of the stator coil's resistance temperature detectors, slip rings, adhesives, and the mounting sleeve insulator for the slip rings that P-EP procured as commercial grade.

The team also identified as nonconformances other elements of P-EP's quality program and its implementation that failed to meet NRC requirements. For example, P-EP failed to establish adequate measures for, and to implement adequate control of, its external design interface with its sister organization, PEM of Edinburgh, Scotland.

Please provide a written statement or explanation within 30 days from the date of this letter for the items in the Notice of Nonconformance containing (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. This reply should be clearly marked as a "Reply to Notice of Nonconformance" and submitted to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, Office of Nuclear Reactor Regulation. We will consider extending the response time if you can show good cause.

The response requested by this letter is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511. In accordance with 10 CFR 2.790(a), a copy of this letter and its enclosures will be placed in the NRC's Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you. We thank you for your cooperation during this inspection.

Sincerely,

for Uedi Potapus

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Nonconformance
2. Inspection Report No. 99900772/91-01

cc w/enclosures:
Mr. Peter R. Holroyd, Managing Director
NEI Peebles Limited
Peebles Electrical Machines
East Pilton
Edinburgh, Scotland EH5 2XT
United Kingdom

NOTICE OF NONCONFORMANCE

NEI Peebles - Electric Products, Inc. Docket No. 99900772
Cleveland, Ohio

During an inspection conducted at the facility of NEI Peebles - Electric Products, Inc. (P-EP) in Cleveland, Ohio, August 5 through 9, 1991, the U.S. Nuclear Regulatory Commission (NRC) inspection team determined that certain activities were not conducted in accordance with NRC requirements, which were contractually imposed on P-EP by purchase orders from NRC licensees. The NRC has classified these items, as set forth below, as nonconformances to the requirements in Appendix B to Title 10, of the Code of Federal Regulations, Part 50 (10 CFR Part 50), imposed on P-EP by purchase order contracts with NRC licensees and by P-EP's "Quality Assurance Manual No. 100," (QAM-100), revision dated November 1, 1984.

A. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part, that measures be established to ensure that applicable regulatory requirements and the design bases are correctly translated into specifications, drawings, procedures, and instructions and that design changes shall be subject to design control measures commensurate with those applied to the original design.

American National Standards Institute (ANSI) N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (1974), endorsed in NRC Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (Revision 2, June 1976), and imposed on P-EP by Pacific Gas and Electric Company's (PG&E's) purchase order for the new (No. 2-3) emergency diesel generator (EDG) set for the Diablo Canyon Nuclear Power Plant Unit 2 (DCNPP2) requires, in part, that (1) measures shall be applied to verify the adequacy of the design; (2) design verification shall be performed by competent organizations other than those who performed the original design; (3) the results of the design verification efforts shall be clearly documented and auditable against the verification methods; and (4) where changes to previously verified designs have been made, design verification shall be required for the changes, including evaluation of the effects of those changes on the overall design.

Contrary to these requirements, in Section 3, "Design Control," of QAM-100, P-EP failed to (1) establish adequate measures to control changes in design, materials, and manufacturing processes commensurate with those controls applied to the original design or (2) provide for performing design verifications of the changes in design, materials, and manufacturing processes.

P-EP, in its design-basis reconciliation and verification of changes that affect the design of PG&E's 1990 generator to the design basis for the original 1969 generator, failed to demonstrate (1) that the changes in the design were controlled commensurate with the design controls applied to the original design and (2) that the original design basis had been correctly translated into revised specifications, drawings, procedures, and instructions. PG&E's purchase order required that the new generator for DCNPP2 be identical (i.e., like-for-like) to DCNPP's five existing operating generators (original 1969 design basis) and its 1986 spare generator. However, P-EP's design-basis reconciliation and verification of the design changes for PG&E's generator were documented and verified only to the 1984 timeframe; when the Cleveland manufacturing facility was closed. Thus, P-EP did not perform an adequate design-basis reconciliation or verification of the generator's design changes to ensure the adequacy of the design, and the effects of those changes on the generator's overall design (Nonconformance 99900772/91-01-01).

B. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part, that measures be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of design documents involving design interfaces.

ANSI N45.2.11-1974, imposed on P-EP in PG&E's PO for the generator requires, in part, (1) that the external interfaces between organizations performing work affecting quality of design shall be identified in writing and shall include those organizations providing criteria, designs, specifications, and technical direction; (2) that the responsibilities of organizations shall be defined and documented in sufficient detail to cover the preparation, review, and approval of design documents involving design interfaces; (3) that systematic methods shall be established for communicating needed design information across external design interfaces, including changes to the design information as work progresses; and (4) that design information transmitted from one organization to another shall be documented in specifications, drawings, or other controlled documents that are uniquely identified and issued by authorized persons.

Contrary to these requirements, in Section 3 "Design Control," of QAM-100, P-EP failed to establish adequate measures to control the design interface between it and its sister organization, Peebles Electrical Machines (PEM) of Edinburgh, Scotland, that consisted of the review, approval, release, distribution, and revision of design documents affected by their design interface.

P-EP failed to demonstrate that the results of PEM's design translation activities were equivalent to the design requirements specified by P-EP. P-EP provided its design drawings and specifications to PEM because PEM manufactures P-EP's generators. PEM's engineering organization translated P-EP's design specifications into its own PEM specifications, drawings, procedures, and instructions. The documents produced by PEM were not reviewed or approved by P-EP before use, and PEM-initiated design changes that were not controlled by documented procedure until December 1990, when PEM issued Departmental Procedure No. DP03A004, "Processing of Engineering Change," well after the design activities for PG&E's generator were completed. Although P-EP performed equivalency evaluations of PEM's procedures and material specifications used to fabricate and assemble PG&E's generator, P-EP did not adequately document (1) the critical requirements or acceptance criteria compared during the equivalency evaluation and (2) the results of the equivalency evaluation or other bases to support P-EP's conclusion that PEM's procedures and specifications were equivalent (Nonconformance 99900772/91-01-02).

C. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the component.

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

ANSI N45.2.11-1974, requires, in part, that design inputs (1) shall be identified, documented, and their selection reviewed and approved and (2) shall be specified and provide a consistent basis for making design decisions, accomplishing design verification measures, and evaluating design changes.

Contrary to these requirements, in Section 3, "Design Control," and Section 7, "Control of Purchased Materials, Equipment, and Services," of QAM-100, P-EP failed to establish adequate measures to provide for the selection and review for suitability of the application for materials, parts, and equipment that were procured as commercial grade items and were essential to the generator's ability to perform its intended design and safety-related function.

P-EP failed to adequately verify the properties or attributes of certain materials, parts, and equipment that were utilized in the fabrication and assembly of PG&E's generator and that also directly affect the generator's ability to perf[illegible] its intended design and safety-related function. Spe[illegible]ally, P-EP failed to ensure the suitability (1) [illegible] the stator coil's resistance temperature detectors, slip r[illegible]ngs, adhesives, and mounting sleeve insulator for the slip rings and (2) of the materials, parts, and equi[illegible]nt PEM procured (Nonconformance 99900772/91-0[illegible]03).

D. Criterion V, "Instructions, Procedure , and Drawings," of Appendix B to 10 CFR Part 50, requires, in part, (1) that measures be established to ensure that activities affecting quality be prescribed by documented instructions, procedures, or drawings; (2) that activities affecting quality be accomplished in accordance with these instructions, procedures, and drawings; and (3) that these instructions, procedures, and drawings include quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to these requirements, in Section 5, "Instructions, Procedures, and Drawings," of QAM-100, P-EP failed to establish adequate measures to ensure (1) that all of the activities affecting quality were prescribed by documented instructions, procedures, or drawings and (2) that the instructions, procedures, and drawings include quantitative or qualitative acceptance criteria for determining that important activities were satisfactorily accomplished.

P-EP failed to demonstrate that the activities affecting quality (1) to fit the dovetail rotor pole assemblies to the rotor spider assembly, (2) to perform the brazing required to fabricate the rotor spider assembly, and (3) to perform brazed joint spliced-connections in the field coil winding were documented or accomplished in accordance with instructions, procedures, or drawings that contained quantitative or qualitative acceptance criteria and were equivalent to those specified by P-EP (Nonconformance 99900772/91-01-04).

Dated at Rockville, Maryland,
this 15th day of January 1992.

Enclosure 2

INSPECTION REPORT

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

Report No.:	99900772/91-01
Docket No.:	99900772
Company:	NEI Peebles - Electric Products, Inc. 17045 Euclid Avenue Cleveland, Ohio 44112
Industry Activity:	NEI Peebles - Electric Products, Inc. (P-EP) supplies power generators and static exciter voltage regulators for use in emergency power systems.
Inspection Conducted:	August 5 through 9, 1991
Inspection Team:	Steven M. Matthews, Team Leader, NRR Walter P. Haass, Sr. Reactor Engineer, NRR Michael R. Snodderly, Reactor Engineer, NRR
NRC Consultant:	Kenneth Sullivan, Brookhaven National Laboratory

Prepared by: Steven M. Matthews — 1/14/92
Steven M. Matthews, Team Leader — Date
Reactive Inspection Section 1
Vendor Inspection Branch

Approved by: Uldis Potapovs — 1-14-92
Uldis Potapovs, Chief — Date
Reactive Inspection Section 1
Vendor Inspection Branch

Inspection Bases:	10 CFR Part 21, Appendix B to 10 CFR Part 50, and ANSI N45.2-1971
Inspection Scope:	To assess P-EP's compliance with regulatory requirements and licensees' procurement requirements through a performance-based evaluation of its engineering, procurement, processes, inspections, and tests
Plants Affected:	All licensees with P-EP power generators

CONTENTS

1 INSPECTION SUMMARY

1.1 Nonconformances

1.1.1 Nonconformance 99900772/91-01-01

Contrary to Criterion III, "Design Control," of Appendix B to Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), and American National Standards Institute (ANSI) N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (1974), endorsed in NRC Regulatory Guide (RG) 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (Revision 2, June 1976), in Section 3, "Design Control," of "Quality Assurance Manual No. 100," (QAM-100), revision dated November 1, 1984, NEI Peebles - Electric Products, Inc., (P-EP) failed to (1) establish adequate measures to control changes in design, materials, and manufacturing processes commensurate with those controls applied to the original design, (2) provide for performing design verifications of the changes in design, materials, and manufacturing processes, (3) demonstrate that the changes in the design were controlled commensurate with the design controls applied to the original design, and (4) demonstrate that the original design basis had been correctly translated into revised specifications, drawings, procedures, and instructions (see Section 3.4.1 of this report).

1.1.2 Nonconformance 99900772/91-01-02

Contrary to Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, and ANSI N45.2.11-1974, in Section 3, "Design Control," of QAM-100, P-EP failed to (1) establish adequate measures to control the activities between it and its sister organization, Peebles Electrical Machines (PEM) of Edinburgh, Scotland, that consisted of the review, approval, release, distribution, and revision of documents involving their respective design interface, (2) demonstrate that the results of PEM's design translation activities were equivalent to the design requirements specified by P-EP, (3) adequately document the critical requirements or acceptance criteria compared during the equivalency evaluation, and (4) adequately document the results of the equivalency evaluation or other bases to support P-EP's conclusion that PEM's procedures and specifications were equivalent (see Section 3.4.2 of this report).

1.1.3 Nonconformance 99900772/91-01-03

Contrary to Criterion III, "Design Control," and Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B to 10 CFR Part 50, and ANSI N45.2.11-1974, in Section 3, "Design Control," and Section 7, "Control of Purchased Materials, Equipment, and Service," of QAM-100, P-EP failed to (1) establish adequate measures to provide for the selection and review for suitability of the application for materials, parts, and equipment that were procured as commercial grade items and were essential to the generator's ability to perform its intended design and safety-related function, (2) ensure the suitability of the stator coil's resistance temperature detectors, slip rings, adhesives, and mounting sleeve insulator for the slip rings, and (3) ensure the suitability of the materials, parts, and equipment PEM procured (see Section 3.4.4 of this report).

1.1.4 Nonconformance 99900772/91-01-04

Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, in Section 5, "Instructions, Procedures, and Drawings," of QAM-100, P-EP failed to establish adequate measures to ensure (1) that activities affecting quality were prescribed by documented instructions, procedures, or drawings; (2) that activities affecting quality were accomplished in accordance with these instructions, procedures, or drawings; and (3) that instructions, procedures, or drawings include appropriate quantitative or qualitative acceptance criteria for determining that important activities were satisfactorily accomplished. P-EP also failed to demonstrate that the activities affecting quality (1) to fit the dovetail rotor pole assemblies to the rotor spider assembly, (2) to perform the brazing required to fabricate the rotor spider assembly, and (3) to perform brazed joint spliced-connections in the field coil winding were documented or accomplished in accordance with instructions, procedures, or drawings that contained quantitative or qualitative acceptance criteria and were equivalent to those specified by P-EP (see Section 3.5 of this report).

1.2 Unresolved Item (99900772/91-01-05)

P-EP's original quality assurance manual (QAM-100), in effect during the design, manufacture, and test of PG&E's generator, did not include measures to adequately control all of the activities affecting the quality and safety-related function of components and parts. Although P-EP's second quality assurance manual (QAM-101) superseded QAM-100, it contained several weaknesses that required strengthening before its implementation. Because the team did not evaluate the implementation of QAM-101, this concern will be evaluated in more detail during a future inspection (see Section 3.3 of this report).

2 STATUS OF PREVIOUS INSPECTION FINDING

(CLOSED) Nonconformance 99900772/86-01-01

Contrary to Criterion IV of Appendix B to 10 CFR Part 50, and to the Parson Peebles Electric Products Inc. (EPI) Quality Assurance Manual Procedure EQ 2.5.1, EPI did not indicate the applicable drawings, revisions, specifications, or quality requirements on a purchase order (PO) for a safety-related manual voltage regulator.

P-EP's attempts to correct this problem were inadequate as evidenced by Nonconformance 99900772/91-01-03 of this report. Although a list of clauses for use on POs for nuclear Class 1E materials and components was developed, additional examples of POs for commercial grade materials and parts showed that P-EP continued to procure items that are critical to the generator's ability to perform its safety-related function as commercial grade without specifying the technical and quality requirements. Therefore, the previous inspection finding (Nonconformance 99900772/86-01-01) will be closed and this issue will be tracked under Nonconformance 99900772/91-01-03.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

During the entrance meeting on August 5, 1991, the NRC's inspection team met with P-EP's staff and discussed the scope of the inspection, outlined areas of concern, and established working interfaces. The team observed activities, held discussions with P-EP's staff, and reviewed records and procedures. The specific areas and documentation reviewed and the team's findings are described in Sections 3.3, through 3.5 of this report. The persons who participated in and who were contacted during the inspection are listed in the appendix to this report. During the exit meeting on August 9, 1991, the team summarized the inspection findings, observations, and concerns with P-EP's management.

3.2 Background

P-EP's facility was originally known as Electric Products Incorporated (EPI) and, under various names, supplied over 120 power generators to the nuclear industry. EPI was purchased by Portec, Inc. in 1969, and was known as the Electric Products Division of Portec, Inc. Portec sold the company in 1979 to Parson Peebles, a subsidiary of Northern Engineering Industries Limited (NEI) in England. NEI is a wholly owned subsidiary of the Industrial Power Group of Rolls-Royce. The Cleveland facility was known at that time as Parson Peebles Electric Products, Inc. (also EPI) and was inspected by the NRC on

February 11 through 13, 1986. The results of that inspection were documented in Inspection Report 99900772/86-01. After Parson Peebles' purchase of the Cleveland facility, NEI reorganized its Parson Peebles operations under the name of NEI Peebles Limited and the Cleveland facility became NEI Peebles - Electric Products, Inc. (P-EP). P-EP's manufacturing facility in Cleveland was closed September 1984 and the power generator work was moved to NEI Peebles Limited's Pilton Works facility in Edinburgh, Scotland, known as Peebles Electrical Machines (PEM). The NRC conducted an inspection of PEM October 6 through 8, 1986, and the results of that inspection were documented in Inspection Report 99901065/86-01. The organizational structure of NEI Peebles Limited at the time of this inspection was such that the Vice President and General Manager of P-EP reported directly to the Managing Director of PEM.

According to P-EP, most of its U S. business was spare and replacement parts; about 50 percent of those parts were for nuclear safety-related items. Since 1984, PEM has manufactured the power generators and many of the spare and replacement parts that P-EP has supplied to the nuclear industry. P-EP recently completed a safety-related power generator for PG&E's new sixth (No. 2-3) emergency diesel generator (EDG) set for its Diablo Canyon Nuclear Power Plant Unit 2 (DCNPP2). P-EP also was performing the design and procurement activities for a safety-related power generator for Washington Public Power Supply System's Nuclear Project 2 (WNP2), which will be manufactured by PEM. Although the team focused their inspection activities on the completed power generator for PG&E's DCNPP2, the concerns discussed in this report may have generic implications for WNP2's power generator and any other power generator, or spare and replacement parts, purchased by other licensees.

PG&E's five existing operating power generators (serial 16908022 through 16908026), installed on DCNPP's EDGs (Nos. 1-1, 1-2, 1-3, 2-1, and 2-2), were procured in 1969 from the Electric Products Division of Portec, Inc., and manufactured in the Cleveland facility, described above. In 1986, PG&E procured a spare power generator (serial 38604851) from P-EP on the basis that it be identical (i.e., like-for-like) to DCNPP's five existing operating power generators that were manufactured in 1969 by Portec, Inc. The 1986 spare power generator was manufactured by PEM in its facility in Edinburgh, Scotland.

The power generator for PG&E's No. 2-3 EDG for DCNPP2 was procured by PO ZS-1539-AB-9, Revision 0, dated January 16, 1990. PG&E requested P-EP to supply one 4.16-kV, 2600-kW, 3-phase, 60-Hz, 900-rpm, single-bearing, engine-driven, AC synchronous power generator. The generator was to be supplied as a design Class 1E basic component in accordance with PG&E's Engineers

Material Memo (EMM) DC2-3322-BRH-E, Revision 0, dated January 5, 1990. In the EMM, PG&E required that the generator be like-for-like with DCNPP's 1986 spare generator and DCNPP's five existing operating generators. PG&E's PO to P-EP also invoked the reporting requirements of 10 CFR Part 21.

PG&E's EMM, Attachment 4, "Specification for Supplier's Quality Assurance Program, Spec. No. SP-D-Peebles" (SP-D-Peebles), Revision 3, dated October 11, 1989, required in Section 4.0, "Quality Assurance Program (Cleveland Facility)," that P-EP's quality program for equipment and components comply with British Standard 5750: Part 1 (ISO-9001-1987), Part 2, and Part 3; and that P-EP's quality program for engineering services comply with Appendix B to 10 CFR Part 50 and ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants" (1971), endorsed in NRC RG 1.28, "Quality Assurance Program Requirements (Design and Construction)," June 1972. SP-D-Peebles also imposed the requirements of numerous other ANSI standards. Included among the ANSI standards imposed by SP-D-Peebles was ANSI N45.2.11 (1974), endorsed by NRC RG 1.64, Revision 2, June 1976.

PG&E's generator was completed and tested by PEM during January and February 1991. On March 1, 1991, PEM shipped the completed generator to PG&E's contractor, GEC Alsthom of Toronto, Canada, for the final assembly and skid-mounting of the EDG set, and the combined testing of the diesel engine, the generator, and the EDG's auxiliary systems.

3.3 Compliance With 10 CFR Part 50, Appendix B

P-EP's activities affecting the quality of the generator for DCNPP2 were controlled by its quality program described in QAM-100, originally issued on July 10, 1976, and its latest revision issued on November 1, 1984. QAM-100 was superseded by QAM-101, issued on February 1, 1991, and its Revision 1, issued on July 24, 1991. P-EP's quality programs described in both QAM-100 and QAM-101 committed to implementing the requirements of Appendix B to 10 CFR Part 50, ANSI N45.2-1971, and the Canadian Standards Association's "CAN3-Z299.1-1978 Quality Assurance Program Standards." The QAM-100 program was in effect during the time that PG&E's generator and its component-parts were designed and procured by P-EP and manufactured and tested by PEM. QAM-100 and QAM-101 controlled P-EP's design, procurement, fabrication, assembly, inspection, tests, corrective actions, and commercial grade dedication activities to produce safety-related generators and spare and replacement component-parts. The results of the team's evaluation is described in the following paragraphs.

QAM-100 was originally issued when P-EP's facility manufactured electrical components. The latest revision was intended to address the changes in P-EP's operations that resulted from the closing of the Cleveland manufacturing facility in 1984. However, QAM-100 continued to contain obsolete requirements that were not implemented by P-EP and that were not applicable to P-EP's current operations. The program also failed to establish adequate measures to control several activities affecting the quality of DCNPP2's generator. For example, in the QAM-100 program, P-EP did not describe adequate controls

- for changes in the design, materials, and manufacturing processes commensurate with those controls applied to the original design
- for reconciliation of changes in the design, materials, and manufacturing processes to the original engineering design bases
- for the revision of design documents involving the design interfaces between P-EP and PEM
- for review, approval, and design-bases reconciliation of PEM-initiated changes that affected P-EP's design bases
- for the selection and review for suitability of application of materials, parts, equipment, processes, and services procured from subsuppliers without adequate quality programs
- to ensure that the technical and quality requirements for materials, equipment, and services had been accomplished by source evaluation and selection, objective evidence of quality furnished by the subsupplier, inspection at the subsupplier's source, or the examination of products upon delivery

Although the QAM-101 program was an improvement over the QAM-100 program, it did not adequately address all of the controls over activities affecting the quality of safety-related generators to the extent consistent with their importance to safety. In the QAM-101 program, P-EP failed to establish measures to identify and control the design interface between P-EP and PEM, and to provide procedures for review, approval, release, distribution, and revision of design documents involving the design interfaces between P-EP and PEM.

Although QAM-101 addressed the qualification of commercial grade subsuppliers by establishing documented acceptable subsupplier performance history data, P-EP failed to ensure that the performance data was directly applicable to the item's critical characteristics and its intended safety-related application and to verify by audit that the subsuppliers' measures for the control of design, processes and material changes were adequately implemented.

Although the commercial grade dedication program was described in QAM-101, P-EP failed to establish measures to determine (1) the effect of parts on the component's design function, (2) the part's properties or attributes that are essential for the item to perform its design function, and (3) the effect of the part's credible failure mechanism on the component's design function. In addition, P-EP failed to ensure that special inspections and tests (the only acceptance method prescribed in QAM-101) will adequately verify the critical characteristics for all parts supplied by commercial grade subsuppliers.

Therefore, P-EP's quality program (QAM-100) in effect for PG&E's generator did not establish measures to adequately control all of the activities that affect the quality of the components and parts that are directly applicable to the generator's ability to perform its intended safety function. In addition, several weaknesses were identified in the QAM-101 quality program. This concern is Unresolved Item 99900772/91-01-05 and will be evaluated in more detail during a future inspection.

3.4 Design Control

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, and ANSI N45.2.11 (1974), as endorsed in RG 1.64, require, in part, that measures shall be established to assure that applicable regulatory requirements and the design bases are correctly translated into specifications, drawings, procedures, and instructions and that design changes shall be subject to design control measures commensurate with those applied to the original design. Measures also shall be established for the identification and control of design interfaces and for coordination among participating design organizations and shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution and revision of documents involving design interfaces and for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the component.

P-EP maintains the overall engineering and design control responsibility, in addition to providing sales and service support, for the generators and other power generating equipment procured by the nuclear industry. The team evaluated P-EP's design activities in the four areas described separately below.

3.4.1 Design-Bases Documentation

In its PO, PG&E required that the generator be like-for-like to the 1986 spare generator and DCNPP's five existing operating power generators (original 1969 design basis). The team reviewed P-EP's control of the generator's engineering design basis, which would be necessary to establish the like-for-like relationship of the new generator to the design bases of the generators previously supplied. Specifically, the team reviewed the synergistic effect of the changes that were made to the original engineering design basis, since 1969, to determine what, if any, effect those changes had on PG&E's like-for-like procurement requirement.

P-EP's design-basis reconciliation to the original 1969 design basis consisted of a drawing change review dated June 24, 1991. P-EP's review encompassed the drawings associated with PG&E's generator since 1984, including all revisions. However, the design control measures of Section 3, "Design Control," of P-EP's QAM-100 did not provide for reconciling changes in the design, materials, and manufacturing processes to the original engineering design basis. In addition, P-EP's design-basis reconciliation of design changes for PG&E's generator was documented and verified only to 1984 when the Cleveland manufacturing facility closed. P-EP could not substantiate that the new generator was like-for-like to PG&E's five existing operating power generators.

Therefore, in its design-basis reconciliation, P-EP failed to demonstrate (1) that the established design control measures were commensurate with those applied to the original design and (2) that the original design basis had been correctly translated into revised specifications, drawings, procedures, and instructions. This is Nonconformance 99900772/91-01-01.

3.4.2 Design Interface

A significant design interface existed between P-EP and PEM. Although P-EP maintained the overall responsibility for engineering and design control, PEM's engineering and design organization was completely independent of P-EP's organization and it performed independent design translation activities. P-EP provided its design drawings, procedures, and material specifications to PEM, and PEM's engineering organization translated them into PEM specifications, drawings, procedures, and instructions, including converting dimensions and tolerances from English values to their metric equivalent.

The measures established in Section 3, "Design Control," of P-EP's QAM-100 did not provide for adequate procedures between P-EP and PEM for the review, approval, release, distribution, and revision of documents involving their respective design interface. This deficiency appeared to have resulted from the "sister company" relationship of P-EP and PEM, and the daily interface of their respective staffs. Although PEM issued Departmental Procedure DP03A004, "Processing of Engineering Change," Revision 0, dated December 17, 1990, it did not affect P-EP's control of the design interface activities during most of the fabrication and assembly of PG&E's generator. Moreover, P-EP failed to establish reasonable assurance that PEM's procedure adequately controlled the design interface activities that were P-EP's responsibility.

Equivalency evaluations of PEM's procedures and material specifications used to fabricate and assemble PG&E's generator were completed by a P-EP's engineering staff in July 1991 and reviewed by P-EP's QA manager in August 1991. (The generator was completed by PEM in February 1991.) P-EP performed these evaluations to ensure that PEM correctly translated the design bases into procedures and material specifications. The equivalency evaluations were not auditable because (1) PEM's equivalent procedures or material specifications were not always available for comparison to P-EP's procedures or material specifications and (2) the evaluations consisted of only a brief summary of the procedures or material specifications. P-EP's equivalency evaluations failed to adequately document (1) the critical requirements or acceptance criteria compared during the evaluation and (2) the results of the evaluation or bases to support P-EP's conclusion that the documents were equivalent. This is Nonconformance 99900772/91-01-02.

3.4.3 Selection of Critical Items

PG&E's generator is a complex component, composed of several critical parts that directly affect the ability of the generator to perform its design and safety-related functions (i.e., the credible failure mechanism or long-term degradation of the part could adversely affect the generator's ability to perform its safety-related function). PG&E selected and identified the generator's critical items in its PO.

PG&E's PO (described in Section 3.2 of this report) was modified and issued as Revision 1, February 2, 1990, to add Attachment F, "Critical Items Listing & Dedication Testing," to its EMM. Attachment F listed 14 critical items and their associated critical characteristics and required P-EP to verify the PG&E-identified critical characteristics for each of the 14 critical items by performing tests. PG&E further required that the verification tests to be performed and their respective acceptance criteria be furnished to PG&E for approval before the materials and parts were installed or used.

Revision 2 to PG&E's PO, dated February 22, 1990, addressed specific data that P-EP was to provide to enable PG&E to perform the seismic analysis of the generator.

Revision 3 to PG&E's PO, dated February 6, 1991, included significant revisions to PG&E's Engineers Material Memo (EMM) SP-D-Peebles, Attachment A and the critical items list of Attachment F. Attachment A, Revision 5, dated November 15, 1990, imposed numerous requirements on P-EP that were not previously imposed by SP-D-Peebles Revision 3 included in PG&E's original PO. The most significant additional requirements follow:

- added section 4.2.6(1), the requirements for critical material, parts, or components that were procured as commercial grade items
- added section 4.2.8, the requirements for the identification and control of materials and items
- added section 4.2.9, the requirements for a test program to identify and document all testing required to demonstrate that items will perform satisfactorily in service
- added section 4.2.10, the requirements for the control of measuring and test equipment

The EMM's Attachment F, changed the list of critical items from 14 (shown in Revision 1) to 27 (in Revision 3). Several of the critical characteristics for those items that were to be verified by P-EP also changed. In other changes imposed by the revision, certain sub-assemblies that were previously identified as critical items were divided into individual parts of the sub-assembly and listed separately (e.g., brushes and brush holders was identified as item 7 of Revision 1 and the critical characteristics were identified as (1) size and shape, and (2) final generator test for resistance, material, and contact pressure, however, Revision 3 listed the brushes and the brush holder separately as item 20 and 19, respectively, and listed configuration as the only critical characteristic for both items). Table 1, "Critical Items Procured by P-EP," on page 15 and Table 2, "Critical Items Procured by PEM," on page 17 of this report provide a comparison of the critical items and their critical characteristics as addressed in Revisions 1 and 3 of the PO. These changes were the result of discussions between the staffs of PG&E and NEI-Peebles at QA (quality assurance) audit meetings held in Cleveland during December 1989 and in Edinburgh during October 1990.

P-EP's generic failure modes and effects analysis (FMEA) was applicable to all rotating electrical machinery produced and was part of P-EP's technical documentation that demonstrated a generator's compliance with the requirements of the Institute of

Electrical and Electronics Engineers (IEEE) Standard 323, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Standard 344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The FMEA included the credible failure mode for each individual part of the generator assembly and assigned a criticality level (see definitions below) to the part, on the basis of the effect of the part's credible failure mode on the ability of the generator to perform its safety-related function.

Level 1 - catastrophic failure (i.e., will not operate at all, extensive repair needed)

Level 2 - severely degraded (i.e., operates far off-normal giving warning that a failure will soon occur, extensive repairs needed)

Level 3 - degraded (i.e., operates off-normal but with adequate warning of an impending failure, repairs simple if done promptly)

Level 4 - minor degradation (i.e., operates near-normal but gives a warning of eventual failure, situation very slowly deteriorates; repairs are simple)

Level 5 - no effect (i.e., part does not affect operation, repairs are part of maintenance)

According to P-EP, PG&E's PO did not impose qualification of the generator to the requirements of IEEE Standards 323 or 344, and PG&E did not procure P-EP's FMEA documentation for use in the selection of critical items or their critical characteristics. P-EP also stated that the extent of its involvement in PG&E's selection of critical items or their critical characteristics was limited to only an agreement with PG&E to perform testing necessary to verify the critical characteristics of the critical items identified by PG&E in Revision 1 of its PO.

P-EP reported that it had not been involved in PG&E's selection of the critical items, or their critical characteristics, listed in Revision 3 of PG&E's PO. Furthermore, PG&E's generator was completed when Revision 3 was issued; therefore P-EP did not consider Revision 3 during its design, procurement, and manufacturing activities.

Because of the minimal involvement of P-EP's engineering organization in PG&E's selection of critical items and their critical characteristics, listed in Revision 1, the team was concerned that PG&E's selected list of critical items may not have been sufficiently comprehensive to ensure that all items were included, specifically, those items with a credible failure mode or that, in a degraded condition, could adversely affect the generators ability to perform its design and safety-related function. The team reviewed P-EP's generic FMEA and discussed the technical bases for the critical items and their critical characteristics with P-EP's engineering staff to determine whether PG&E's Revision 1 list of 14 critical items, or its Revision 3 list of 27 critical items included all parts that are critical to the generator's ability to perform its design and safety-related function.

According to P-EP's FMEA, the generator's two major design parameters with regard to the effects of long-term degradation and cyclic fatigue were (1) its operating temperatures and (2) cyclic loading or high vibration forces. On the basis of these design parameters, FMEA criticality levels 1 or 2 were assigned to critical items such as the stator windings, leads and their connections, rotor pole windings, roller bearings, rotor shaft, coil supports, and slip rings. P-EP's generic FMEA documentation indicated that PG&E's lists of critical items did not adequately envelope all of the generator's critical parts having a design or safety-related function. Two examples are discussed below.

(1) Slip-Ring Mounting Sleeve Insulator

The generator was designed with a brush and slip-ring assembly to carry DC excitation voltage to the field coils mounted on the rotor shaft. The slip-ring assembly was concentrically mounted on the rotor shaft. P-EP incorporated a slip-ring mounting sleeve insulator in its design to prevent establishing a current path to ground between the slip-ring assembly and the rotor shaft. The mounting sleeve insulator consisted of a tube of insulating material, with approximately 0.25-inch wall thickness, installed between the shaft and the slip-ring assembly. P-EP's generic FMEA documentation indicated that, if the mounting sleeve insulator between the slip-ring assembly and the rotor shaft failed, DC excitation voltage would be lost and result in catastrophic failure of the generator. In addition, a short-to-ground failure in the insulator could occur from wear or erosion, establishing a current path to ground. However, PG&E did not identify the slip-ring mounting sleeve insulator as a critical item.

(2) Temperature and Vibration Indicating Devices

P-EP provided six resistance temperature detectors (RTDs) to monitor the generator's stator coil operating temperatures and provide a conservative indication of the generator's overall temperature. The operating temperatures of the generator, including localized thermal stresses, affect the stability of the insulation and adhesive materials (e.g., thermal breakdown, aging, fatigue, and wear) which directly affect the fragility of unisotropic structures (e.g., rotor windings) during the installed life of the generator. A limit of 105^{0}C rise over an ambient temperature of 40^{0}C for the maximum generator operating temperature (145^{0}C) was established in P-EP's design basis. Although the RTDs were included in PG&E's Revision 1 list of critical items, they were not included in the Revision 3 list of critical items. RTDs were not provided to monitor the temperature of the shaft's single roller bearing, even though the roller bearing and its operating temperature were identified in P-EP's FMEA as critical items.

In addition to high temperature, fatigue from cyclic loading or high vibration forces on the generator may also directly affect the performance and reliability of the single roller bearing. The roller bearing may be subjected to cyclic loading or high vibration forces caused by an unbalanced rotor shaft, the diesel engine with its crankshaft directly connected to the generator's rotor shaft, and other sources from the skid-mounted EDG assembly.

During the installed life of the generator, subtle damage to the generator may occur from short-to-ground or asynchronous events (e.g., paralleling the generator out-of-phase) that cause significant forces on the stator coils and rotor pole windings. P-EP indicated that PG&E's generator was designed to withstand short-to-ground events that produce magnetic forces on the stator coils, which were mechanically supported by the stator frame's welded structure. The end sections of the stator coils, however, were installed in a cantilevered arrangement with stiffeners to support the coils and prevent or minimize their distortion. An asynchronous event may produce centrifugal forces on the rotor pole windings of such magnitude to cause separation of the windings and an unbalanced rotor shaft. P-EP stated that the generator was not constructed to withstand an asynchronous event. However, PG&E did not identify vibration indicating devices as critical items.

For a complex assembly such as a generator, the selection of critical items and the determination of their critical characteristics would require the involvement of both the licensee's and supplier's engineering staffs. P-EP considered this interface activity to be limited for those critical items identified in Revision 1 of PG&E's PO, and it believed the interface activity was nonexistent for the critical items identified in Revision 3 of the PO. Furthermore, P-EP had completed PG&E's generator when Revision 3 was issued; therefore, Revision 3 was not considered during the design, procurement, and manufacturing activities of the generator.

3.4.4 Selection and Review for Suitability

P-EP or PEM procured each of the critical items identified in Revisions 1 and 3 of PG&E's PO as commercial grade items. The critical items procured by P-EP and PEM are identified in Tables 1 and 2 of this report. P-EP procured 7 of the 14 items listed in Revision 1 of PG&E's PO (or 10 of the 27 items listed in Revision 3), and supplied them to PEM for installation in the generator assembly. P-EP's procurement practice consisted of purchasing items from subsuppliers that were selected on the basis of their performance history, which was determined through the general knowledge and experience of P-EP's staff. The performance history data that was documented and verified during the manufacture of PG&E's generator did not establish an adequate basis for the qualification of the subsuppliers of critical items. Most of P-EP's subsuppliers were not audited to verify that their measures to control design, processes, and material changes were adequately implemented. Therefore, the critical items procured by P-EP for PG&E's generator were procured as commercial grade items. The POs for these items did not impose any quality requirements or the reporting requirements of 10 CFR Part 21 on the subsupplier.

PEM procured the 7 remaining critical items listed in Revision 1 of PG&E's PO (or 17 of the 27 items listed in Revision 3) from its subsuppliers in Europe. PEM was qualified as a subsupplier to P-EP through P-EP's audits of PEM dated September 30, 1985, and August 7 through 9, 1989. P-EP stated that its audits qualified PEM to supply components and parts produced to a quality program equivalent to the requirements of Appendix B to 10 CFR Part 50 and in compliance with the reporting requirements of 10 CFR Part 21. However, P-EP's reports for the 1985 and 1989 audits of PEM did not document adequate objective evidence to substantiate whether PEM's quality program was adequate to perform commercial grade dedication of critical items. P-EP also failed to demonstrate that PEM's dedication activities for critical items that it procured as commercial grade resulted in establishing reasonable assurance that the generator and its critical items will perform their respective design and safety-related functions.

PEM's commercial grade dedication activities will be reviewed by an NRC team in a separate inspection. This report only documents the results of the team's evaluation (1) of P-EP's procurement and commercial grade dedication activities for the items it procured and supplied to PEM and (2) of P-EP's responsibility for establishing reasonable assurance that PEM's commercial grade dedication activities were adequate to ensure that the critical items will perform their design and safety-related function.

Table 1 - Critical Items Procured by P-EP

Critical Items:	PG&E's PO Revision:	Attachment F:	Critical Characteristics
Insulators (5-kV in terminal box)	Rev. No. 1	Item-1	• Dielectric strength • Size and weight
	Rev. No. 3	Item-22	• Dielectric strength • Configuration
Insulating bushings (lead wires through motor case)	Rev. No. 1	Item-3	• Size and shape
	Rev. No. 3	Item-24	• Configuration
Insulating material (sheets, tape, & rings)	Rev. No. 1	Item-5	• Thickness
	Rev. No. 3	Item-26	• Thickness
Bearing seals (felt)	Rev. No. 1	Item-6	• Thickness and shape • Texture
	Rev. No. 3	Item-23	• Configuration • Texture
Brushes and Brush Holders	Rev. No. 1	Item-7	• Size and shape • Final generator test: resistance, material, and contact pressure
Brushes	Rev. No. 3	Item-20	• Configuration
Brush Holder	Rev. No. 3	Item-19	• Configuration

Table 1 continued

Critical Items:	PG&E's PO Revision:	Attachment F:	Critical Characteristics:
Stator resistance temperature detectors (RTDs)	Rev. No. 1	Item-8	• Shape and size • Shop test: continuity, resistance, and insulation
Current transformer and test switch	Rev. No. 1	Item-9	• Size and weight • Dielectric strength • Continuity
Current transformer	Rev. No. 3	Item-21	• Configuration • Mounting • Insulation • Resistance • Continuity
Current transformer test switch	Rev. No. 3	Item-25	• Configuration • Dielectric strength • Continuity
Slip rings	Rev. No. 3	Item-17	• Configuration • Material
Adhesives	Rev. No. 3	Item-27	• Material

Table 2 - Critical Items Procured by PEM

Critical Items:	PG&E's PO Revision:	Attachment F:	Critical Characteristics:
Lead wire	Rev. No. 1	Item-1	• Dielectric strength • Number of strands • Marking on cable • Insulation thickness
	Rev. No. 3	Item-16	• Configuration
Magnet wire	Rev. No. 1	Item-4	• Size and shape • Resistance • Insulation • Dielectric strength
	Rev. No. 3	Item-3	• Material • Insulation • Dielectric strength
Copper bus (in terminal box)	Rev. No. 1	Item-10	• Size • Resistance • Silver plating
Lead to coil termination	Rev. No. 1	Item-11	• Brazing • Weld materials
Roller bearing	Rev. No. 1	Item-12	• Size/type • Visual inspection • Catalog number • Tolerances
	Rev. No. 3	Item-6	• Part number • Configuration

Table 2 continued

Critical Items:	PG&E's PO Revision:	Attachment F:	Critical Characteristics:
Shaft/casting	Rev. No. 1	Item-13	• PEM test
	Rev. No. 3	Item-1	• Material • Configuration • Integrity
Stator and Rotor core	Rev. No. 1	Item-14	• PEM test (losses)
Stator coils	Rev. No. 3	Item-15	• Configuration • Chemical composition • Coating insulation
Stampings	Rev. No. 3	Item-2	• Configuration • Material
Bearing bracket	Rev. No. 3	Item-4	• Configuration • Material
Stud/threaded rod	Rev. No. 3	Item-5	• Dimensions • Material • Welding
Spider end rings	Rev. No. 3	Item-7	• Configuration
Pole end rings	Rev. No. 3	Item-8	• Configuration • Material
Short circuit bars	Rev. No. 3	Item-9	• Configuration • Material
Pole head	Rev. No. 3	Item-10	• Configuration

Table 2 continued

Critical Items:	PG&E's PO Revision:	Attachment F:	Critical Characteristics:
Tapered keys	Rev. No. 3	Item-11	• Configuration • Material Hardness
Rotor wedge	Rev. No. 3	Item-12	• Material
Rivets	Rev. No. 3	Item-13	• Configuration
Insulating washers	Rev. No. 3	Item-14	• Configuration • Material • Dielectric strength
Stator frame	Rev. No. 3	Item-18	• Configuration

P-EP's commercial grade dedication program was governed by Procedure DED-100, implemented on August 2, 1991. The program was not in effect during the procurement and commercial grade dedication of the critical items supplied to PEM for use in PG&E's generator. P-EP considered its standard material receiving activities adequate to dedicate commercial grade items, on the basis of its understanding of commercial grade dedication requirements that existed before P-EP's development and implementation of DED-100. The commercial grade dedication activities performed by P-EP for the items procured and supplied to PEM for PG&E's generator were, therefore, not controlled by documented instructions or procedures.

Although P-EP agreed to perform the testing necessary to verify the critical characteristics of the items identified in Revision 1 of PG&E's PO as critical, P-EP did not (1) identify the items critical to the generator's ability to perform its intended safety-related function or (2) perform a technical evaluation of the items identified in Revision 1 of PG&E's PO to determine the adequacy of PG&E's list of critical characteristics. For the critical characteristics selected by PG&E, P-EP failed to demonstrate their relevance (1) to the properties or attributes of the item necessary to withstand the effects of long-term degradation, (2) to the credible failure mode of the item, and (3) to the ability of the item to perform its safety-related function. P-EP failed to substantiate that PG&E's list of critical items included <u>all</u> parts that are required for the generator to perform its safety-related function and that PG&E's list of critical characteristics were adequate to that the item will perform its safety-related function. Consequently, an evaluation of P-EP's generic FMEA identified additional critical characteristics for certain items that were not identified or verified by P-EP during its commercial grade dedication activities and were not identified by PG&E in its Revision 1 to the PO. These characteristics are described, as applicable, in the summary of P-EP's commercial grade dedication and verification activities for the items it procured and supplied to PEM, as given below.

(1) <u>Insulators</u> (Revisions 1 and 3)

The four 5-kV insulators were installed in the terminal box. PG&E identified their critical characteristics as dielectric strength, size, and weight in Revision 1 of its PO, and dielectric strength and configuration in Revision 3. The acceptance criterion for the insulators dielectric strength was not obtained from their supplier. P-EP verified the insulator's weight and dimensions, including length, outside-diameter (OD), and bolt hole center location. P-EP found the results of its verification activities acceptable, even though it did not obtain the supplier's certification of dielectric strength or perform the test to verify the dielectric strength.

(2) <u>Insulating Bushings</u> (Revisions 1 and 3)

The insulating bushings were installed in the lead wire penetration through the generator housing. PG&E identified the critical characteristics as size and shape in Revision 1 of its PO and configuration in Revision 3. P-EP verified the bushing's dimensions, including the thread OD, length, bushing OD, inside-diameter (ID), and the overall length. P-EP found the results of its verification activities acceptable. However, P-EP did not verify their dielectric strength or concentricity, which were not identified as described above.

(3) Insulating Materials (Revisions 1 and 3)

The insulating materials were installed on the stator's so called "diamond" (shape) coil windings. PG&E identified thickness as the critical characteristic. The insulation materials procured and the amount sampled by P-EP consisted of

- Mica paper tape, 60 rolls. . . . . .6 rolls sampled
- Mica paper tape, 60 rolls. . . . . .6 rolls sampled
- B-Stage mica paper tape, 162 rolls . . . 17 rolls sampled
- B-Stage mica wrapper, 4 rolls. . . . . . . 4 rolls sampled

P-EP verified the thickness of the insulating material on the rolls sampled. P-EP found the results of its verification activities acceptable. However, P-EP did not verify (1) the batch or lot homogeneity of the insulation material to ensure that each batch or lot was sampled and traceable to each batch or lot and (2) the material constituents of the insulating materials or their properties or attributes with regard to the generator's design-basis operating temperature requirements.

(4) Bearing Seal (Revisions 1 and 3)

The bearing seal is a felt disc installed on the rotor shaft. PG&E identified the critical characteristics as thickness, shape, and texture in Revision 1 of its PO and as configuration and texture in Revision 3. P-EP verified the seal's dimensions, including OD, ID, thickness, and shape. According to P-EP, the determination of acceptability of the felt bearing seal's texture was a matter of judgment. P-EP found the results of its verification activities acceptable. However, P-EP did not consider that the different "weights" of oil-seal felt that have different porosity and lubricant holding properties.

(5) Brushes and Brush Holders (Revision 1)

The brushes and brush holders were installed on a brush holder stud and positioned above the slip-rings on the rotor shaft. PG&E identified the critical characteristics as size, shape, final generator test to verify resistance, material, and contact pressure.

Brushes (Revision 3)

The brushes and their wire leads and terminal connections were installed in the brush holders. PG&E identified the critical characteristic as configuration. P-EP verified the brushes' dimensions, including height, length, and width. P-EP found the results of its verification activities acceptable. P-EP did not, however, verify the brushes' (1) material constituents, (2) wire lead size or type, (3) wire lead terminal connections, and (4) electrical resistance, which were not identified as described above.

Brush Holders (Revision 3)

PG&E identified the critical characteristic as configuration. P-EP verified the brush holders' dimensions, including the opening size for the brush and its overall shape. P-EP found the results of its verification activities acceptable. P-EP did not, however, verify (1) the spring tension on the brushes or (2) the technical and quality requirements, or the critical characteristics of the Grade X Spaudite Bakelite cylinder (bushing) that fits over the brush holder stud and functions as the insulator for electrical separation between the brush holder and the generator frame, which were not identified as described above.

(6) Stator Resistance Temperature Detectors (RTDs) (Revision 1)

The stator RTDs were installed in the stator coil assembly. PG&E identified the critical characteristics as shape, size, shop test for continuity, resistance, and insulation. Although continuity and resistance were listed separately as characteristics for several items without further explanation, the team noted that if a quantitative value for resistance was desired then continuity would also be demonstrated without performing a separate test. Specifying continuity in addition to resistance would, therefore, normally be considered redundant. PG&E's critical characteristics, however, did not specify (1) the temperature at which the shop test for resistance was to be conducted or (2) the linearity requirements over the test range. (See the description of temperature and vibration indicating devices, on page 12 of this report.) P-EP failed to demonstrate documented dedication and verification activities for the commercial grade stator RTDs.

(7) Current Transformers and Test Switch (Revision 1)

The current transformers and test switch form a sensing device. PG&E identified the critical characteristics as size, weight, dielectric strength, and continuity.

Current Transformers (Revision 3)

PG&E identified the critical characteristics of current transformers as configuration, mounting, insulation, resistance, and continuity. P-EP verified the current transformers dimensions, including the height, the length measured at the feet, the length of the body, location of mounting holes, weight, insulation resistance, and continuity. The acceptance criterion for the dielectric strength of the current transformers was not obtained from the manufacturer. P-EP found the results of its verification activities acceptable. However, P-EP's verification activities did not consider (1) the electrical loads supplied by the current transformers or whether the current transformers supplied a current to the static exciter voltage regulator or instrumentation and protective circuits and (2) the ratio of the primary to secondary currents, which were not identified as described above.

Test Switch for the Current Transformers (Revision 3)

PG&E identified the critical characteristics of the test switch for the current transformers as configuration, dielectric strength, and continuity. P-EP verified the test switch's dimensions including the cover size, location of mounting holes, weight, dielectric strength, and continuity. P-EP found the results of its verification activities acceptable, even though it did not obtain the supplier's certification of dielectric strength or perform the test to verify the dielectric strength.

(9) Slip-Ring Assembly (Revision 3)

The slip-ring assembly was installed on the rotor shaft. PG&E identified the critical characteristics as configuration and materials. However, P-EP failed to demonstrate documented dedication and verification activities for the commercial grade slip-ring assembly.

(10) Adhesives (Revision 3)

The epoxy adhesive (resin) was applied during the forming of the rotor pole windings. PG&E identified the critical characteristic as material. P-EP Shop Order S-1128 required the use of epoxy resin instead of a polyester resin (polyester resin was used for PG&E's five existing EDGs) because an environmental qualification report showed that the performance characteristics of epoxy resin were acceptable and it was an acceptable substitute for the polyester resin. P-EP, however, did not establish similarity of the commercial grade epoxy resin purchased to the epoxy resin described in the environmental qualification report. P-EP also failed to demonstrate documented dedication and verification activities for the commercial grade epoxy resin.

(11) <u>Slip-Ring Mounting Sleeve Insulator</u> (not identified)

The slip-ring mounting sleeve insulator was installed between the shaft and the slip-ring assembly and provided not only the electrical separation of the slip-ring assembly and the rotor shaft, but also formed the mounting structure for the slip-ring assembly. PG&E did not identify the slip-ring mounting sleeve insulator as a critical item. (See the description of slip-ring mounting sleeve insulator, on page 12 of this report.) P-EP's material routing incoming order review of April 12, 1990, showed that P-EP supplied the slip-ring mounting sleeve insulator to PEM as a commercial grade stock item. Additionally, P-EP Drawing No. A-29412, "Slip Ring Mounting Sleeve Insulator," Revision 3, dated December 20, 1967, showed an obsolete material specification for the sleeve insulator. P-EP stated it would update the drawing. P-EP failed to demonstrate documented dedication and verification activities for the commercial grade slip-ring mounting sleeve insulator.

(12) <u>Vibration Indicating Devices</u> (not identified)

The vibration indicating device would be used to detect high vibration resulting from various sources, including an asynchronous event. (See the description of temperature and vibration indicating devices on page 12 of this report.) However, P-EP failed to include vibration indicating devices in its evaluation of critical items.

P-EP supplied the generator to PG&E as a basic component that complied with the quality requirements of Appendix B to 10 CFR Part 50; therefore, P-EP was responsible for establishing reasonable assurance that the generator and its critical items will perform their respective design and safety-related functions. Although PG&E selected and specified the critical items and their critical characteristics for its generator, P-EP agreed to perform the tests necessary to verify PG&E-specified critical characteristics. P-EP did not demonstrate that the critical characteristics, specified by PG&E, were relevant to the critical item's (1) design characteristics, (2) credible failure modes, (3) ability to perform its safety-related function, and (4) properties or attributes necessary to withstand the effects of long-term degradation and cyclic fatigue. P-EP also failed to demonstrate that PEM's dedication activities, for critical items procured by PEM as commercial grade, resulted in establishing reasonable assurance that the generator and its critical items will perform their respective design and safety-related functions.

P-EP failed to demonstrate reasonable assurance that the technical bases for the critical items and their critical characteristics chosen by PG&E and verified by P-EP during the commercial grade dedication and verification activities adequately (1) ensured that the critical items and the generator will perform their safety-related function and (2) ensured that the critical items have the properties or attributes necessary to withstand the effects of long-term degradation or cyclic fatigue This is Nonconformance 99900772/91-01-03.

3.5 Instructions, Procedures, and Drawings

Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, require, in part, that measures be established to ensure that activities affecting quality be prescribed by documented instructions, procedures, or drawings; that activities affecting quality be accomplished in accordance with these instructions, procedures, and drawings; and that instructions, procedures, and drawings include quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. The most significant concerns identified during this review are discussed below.

P-EP's facsimile transmittal to PEM, dated February 11, 1987, provided the instructions for fitting the dovetail rotor pole assemblies to the rotor spider assembly, even though P-EP did not have a documented procedure that prescribed this activity. PEM incorporated these instructions into Engineering Standard R-6097, "Assembly Procedure for Wound Rotors of Class 1E Generators Having Dovetail Poles." P-EP did not approve PEM's procedure or perform an equivalency evaluation because it did not have a documented procedure to compare to PEM's procedure.

P-EP Drawing C-66827, "Rotor Pole Assembly," Revision 2, specified the use of Brazing Specification EB-4.4 for the fabrication of the rotor pole stampings. P-EP did not perform an evaluation to determine the equivalency of PEM's Brazing Procedure R-6092, "Preparation and Procedure for Brazing Copper/Copper Alloy Rotor Bars to Short Circuiting Rings for Use in Normal Industrial Environments," Issue 1, Revision 0, dated December 7, 1990. In addition, P-EP did not perform an equivalency evaluation of Peebles Power Transformers Procedure 5275, "Process Specification Responsible Department Fabrication," dated March 26, 1987. PEM used this procedure as a broad-based procedure that allowed the user to choose between several welding and brazing processes and joint geometries.

PG&E's generator was designed with eight field coils mounted on the rotor. Each field coil consisted of 404 turns of magnet wire that were wound on a laminated steel, rotor pole core with high permeability. The application of a DC excitation voltage, supplied from the brush and slip-ring assembly, will cause the field coils to generate a magnetic field, and in combination with the rotation of the rotor shaft, this field generates the output voltage. An electrically shorted, or open field coil winding, may result in the failure of the generator to perform its intended design and safety-related function. The field coil windings are also subjected to centrifugal forces from the rotation of the shaft and the resulting mechanical stresses that may affect the integrity of the field coil windings.

PEM's Manufacturing Procedure R-6096, "Manufacturing Procedure for Strip-On-Flat Field Coils Wound Directly Onto Laminated Poles for use In Class 1E Generators," Revision 0, dated January 4, 1991, stated, in part, that spliced joints in the magnet wire were permissible where a continuous length of magnet wire was not available during field coil (rotor pole) fabrication. In the event that the amount of magnet wire available on a single spool was not sufficient to complete the coil winding operation, or where the magnet wire was damaged or broken during the manufacturing process, PEM's Procedure R-6096 permitted making a brazed joint spliced-connection in the field coil winding. If the fabrication and brazing of such a joint was not adequately controlled by procedural guidance and proper quality techniques, the results may be (1) a mechanically weak spliced-connection or (2) a high electrical resistance at the brazed joint, which may not be readily detectable after completing the field coil winding.

P-EP's Production Specification R-6028, dated May 1968, provided the engineering guidance to perform a resistance brazed spliced-connection of magnet wire during the fabrication (i.e., winding) of the field coils. In addition to providing procedural guidance, this specification contained precautionary comments that (1) silfos (a brazing rod material) is brittle and considered to be only half as good as an electrical conductor as that of copper and (2) the phosphorous in the brazing material will bubble, if over heated, resulting in a mechanically weak spliced-connection with high resistance. A high-resistance spliced-connection may cause a thermal "hot spot," leading to an electrical short within the field coil winding. According to P-EP's generic FMEA document, such an occurrence may lead to the generator's failure. The high resistance would be localized in a single winding. Therefore, any change in total field coil resistance may be masked, and not readily detectable, when the field coil was completely assembled. Where the brazing operation resulted in a mechanically weak spliced-connection, mechanical stress, induced by centrifugal forces and/or vibration, may also cause a short or open circuit in the affected field coil winding.

P-EP stated that, to its knowledge, no spliced-connections were made during the fabrication of the field coils and produced a certificate of conformance that indicated that a sufficient quantity of magnet wire per spool was ordered for each field coil assembly. The certificate of conformance, however, did not establish reasonable assurance that PEM had not made spliced-connections as a result of damage to the magnet wire during the winding process, and P-EP did not demonstrate documented verification that PEM did not perform spliced-connections. P-EP did advise the team that spliced-connections may be necessary for the field coil windings for WNP2's generator.

PEM Manufacturing Procedure R-6096 did not produce the guidance or the precautionary statements contained in P-EP Production Specification R-6028. P-EP did not perform an equivalency evaluation of PEM's procedure. PEM's manufacturing procedure did not include quantitative or qualitative acceptance criteria for spliced connections such as resistance measurements and tensile strength tests following the brazed joint splicing operation.

These instances were the result of P-EP's QAM-100 failure to establish adequate measures to ensure (1) that all of the activities affecting quality were prescribed by documented instructions, procedures, or drawings and were accomplished in accordance with these instructions, procedures, and drawings, and (2) that the instructions, procedures, and drawings include quantitative or qualitative acceptance criteria for determining that important activities were satisfactorily accomplished. P-EP also failed to demonstrate that the activities affecting quality (1) to fit the dovetail rotor pole assemblies to the rotor spider assembly, (2) to perform the brazing required to fabricate the rotor spider assembly, and (3) to perform brazed joint spliced-connections in the field coil winding were documented or accomplished in accordance with instructions, procedures, or drawings that contained quantitative or qualitative acceptance criteria and were equivalent to those specified by P-EP. This is Nonconformance 99900772/91-01-04.

APPENDIX A

PERSONS CONTACTED

The U.S. Nuclear Regulatory Commission staff participating in the evaluation of NEI Peebles - Electric Products, Inc. with regard to its design, procurement, commercial grade dedication, and manufacture of a power generator for Pacific Gas and Electric Company's Diablo Canyon Nuclear Power Plant, Unit 2 and the persons contacted during the inspection are listed below.

NEI Peebles - Electric Products, Inc.

•	Clasen, Robert C.	Senior Design Engineer
• *	Marino, Frank D.	Manager, Quality Assurance
*	Mossbrugger, Charles J.	Manager, Engineering
• *	Politi, Ron B.	Vice President and General Manager
*	Rossman, Richard A.	Manager, Materials

U.S. Nuclear Regulator Commission

• *	Haass, Walter P.	Senior Reactor Engineer, Special Projects Section, Vendor Inspection Branch (VIB), Division of Reactor Inspection and Safeguard (DRIS), Office of Nuclear Reactor Regulation (NRR)
• *	Matthews, Steven M.	Team Leader, Reactive Inspection Section 1 (RIS1), VIB/DRIS/NRR
• *	Snodderly, Michael R.	Reactor Engineer, RIS1/VIB/DRIS/NRR
•	Sullivan, Kenneth	NRC Consultant, Brookhaven National Laboratory

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• = Attended the Entrance Meeting
* = Attended the Exit Meeting

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

February 13, 1992

Docket No. 99901065

Mr. Peter R. Holroyd
Manager
NEI Peebles Limited,
Peebles Electrical Machines
East Pilton
Edinburgh, Scotland EH5 2XT
United Kingdom

Dear Mr. Holroyd:

SUBJECT: INSPECTION OF A SAFETY-RELATED POWER GENERATOR SUPPLIED TO DIABLO CANYON NUCLEAR POWER PLANT UNIT 2 (NOTICE OF NONCONFORMANCE AND INSPECTION REPORT NO. 99901065/91-01)

We are transmitting herewith the report of the U.S. Nuclear Regulatory Commission (NRC) inspection, conducted September 23 through 27, 1991, at Peebles Electrical Machines (PEM) located at its Pilton Works in Edinburgh, Scotland. Messrs. Steven M. Matthews, Stephen D. Alexander, and Gregory C. Cwalina of the NRC's Office of Nuclear Reactor Regulation evaluated PEM's activities associated with its manufacture of an emergency ac power generator for PEM's sister company, NEI Peebles - Electric Products, Incorporated (P-EP), of Cleveland, Ohio (both are subsidiaries of NEI Peebles Limited). P-EP procured the generator from PEM for an NRC licensee, Pacific Gas and Electric Company (PG&E). The generator is to be used for the new (no. 2-3) emergency diesel generator set for PG&E's Diablo Canyon Nuclear Power Plant Unit 2 (DCNPP2). At the conclusion of this inspection, the NRC inspection team discussed the inspection findings with you and other members of your staff.

In its acceptance of the purchase order from PG&E for this safety-related (Class 1E) generator for DCNPP2, P-EP accepted the responsibility to assure overall compliance with all the applicable provisions of the quality requirements of Appendix B to Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50) and the reporting requirements of 10 CFR Part 21. P-EP audited PEM's quality program and determined that, although it was not based on Appendix B to 10 CFR Part 50, it nevertheless met those requirements. P-EP believed that it could impose PG&E's requirements on PEM by invoking the PEM quality program. Upon delivery, P-EP provided PG&E with a certificate of conformance that certified that the generator was produced in compliance with Appendix B to 10 CFR Part 50. This certification was based largely on P-EP's audit and determination regarding the equivalence of PEM's quality program to Appendix B to 10 CFR Part 50.

As part of the NRC's independent evaluation of this procurement and of the ultimate acceptability of the new DCNPP2 generator, the NRC team assessed the degree to which PEM's quality program and activities (1) were in compliance with the requirements imposed in P-EP's purchase order to PEM, (2) met the requirements of PG&E's purchase order to P-EP, and (3) ultimately met the applicable NRC requirements. Accordingly, the NRC team evaluated PEM's quality program and its implementation in selected areas such as the (1) control of design processes and interfaces, (2) selection and review for suitability of application of certain parts that were identified in the PG&E purchase order as essential to the generator's ability to perform its safety-related function (critical items), and (3) control of purchased materials, parts, equipment and services, including verification that the critical items met their specifications.

This inspection consisted of an examination of procedures and representative records, interviews with PEM staff, and observations by the NRC team. As a result of the inspection, a notice of nonconformance (Enclosure 1) has been issued to PEM. The inspection report (Enclosure 2) contains a detailed discussion of the areas examined during the inspection and our findings.

The most significant inspection finding was that PEM's documented evidence did not demonstrate reasonable assurance that certain critical items (1) met all of PEM's procurement specifications to its suppliers of commercial grade material, (2) met all of P-EP's procurement specifications to PEM, (3) met all PG&E's requirements imposed on P-EP, and (4) met all the applicable NRC quality and technical requirements. Specifically, there was inadequate documented evidence that all the critical characteristics of such items were identified and adequately verified to ensure the items are capable of performing their safety-related functions. Examples of the critical items that were found not to be adequately dedicated include (1) the rotor pole magnet wire wrapped with varnished insulation tape that was specified to be unvarnished, (2) the Bakelite electrical separation ring that was used as a load-bearing component part of the rotor shaft support assembly without an engineering basis for the design, and (3) certain other commercial grade materials, parts, and equipment described in the report that were accepted on the basis of unvalidated certificates of conformance from PEM's commercial suppliers.

The team also identified other elements of PEM's quality program and its implementation that did not meet NRC requirements. For example, PEM had not established adequate measures for, nor implemented adequate control of, its external design interface with P-EP.

During this inspection, PEM was also fabricating and assembling a safety-related emergency ac power generator for Washington Public Power Supply System's Nuclear Project 2 (WNP2). Although the team focused its inspection activities on the completed generator for PG&E's DCNPP2, the concerns discussed in this report may have generic implications for WNP2's generator and any similar generators, or spare and replacement parts, built by PEM and supplied by P-EP to other licensees.

Please provide a written statement or explanation within 30 days from the date of this letter for the items in the Notice of Nonconformance containing (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. This reply should be clearly marked as a "Reply to Notice of Nonconformance" and submitted to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, Office of Nuclear Reactor Regulation. We will consider extending the response time if you can show good cause.

The response requested by this letter is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511. In accordance with 10 CFR 2.790(a), a copy of this letter and its enclosures will be placed in the NRC's Public Document Room.

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

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Nonconformance
2. Inspection Report 99901065/91-01

cc w/enclosures:
Mr. Ron B. Politi
Vice President and General Manager
NEI Peebles - Electric Products, Inc.
17045 Euclid Avenue
Cleveland, Ohio 44112

Enclosure 1

NOTICE OF NONCONFORMANCE

Peebles Electrical Machines
Edinburgh, Scotland

Docket No. 99901065

During an inspection conducted September 23 through 27, 1991, at Peebles Electrical Machines (PEM) located at its Pilton Works in Edinburgh, Scotland, the U.S. Nuclear Regulatory Commission (NRC) inspection team determined that certain activities associated with PEM's manufacture of an emergency ac power generator for its sister company, NEI Peebles - Electric Products, Incorporated (P-EP), of Cleveland, Ohio, were not conducted in accordance with NRC requirements. The NRC requirements applicable to the safety-related generator P-EP procured from PEM for an NRC licensee, Pacific Gas and Electric Company (PG&E), are contained in Appendix B to 10 CFR Part 50.*

In its acceptance of the purchase order from PG&E, P-EP accepted the responsibility to assure overall compliance with all the applicable provisions of Appendix B to 10 CFR Part 50 and the reporting requirements of 10 CFR Part 21. Pursuant to Criterion IV, "Procurement Document Control," of Appendix B to 10 CFR Part 50, the PG&E procurement documents issued to P-EP for this generator imposed quality assurance requirements on P-EP as follows: P-EP was required to ensure compliance with all codes and standards referenced in the purchase order. These included the American National Standards Institute (ANSI) Standard N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants" (1971); British Standard 5750, Parts 1 through 3; and other standards, including ANSI N45.2.11-1974 on design control.

The PG&E procurement documents specified that this new generator be identical (like-for-like) to DCNPP's five existing emergency ac power generators (built by P-EP's predecessor company in 1969) and also DCNPP's spare generator (built by PEM and supplied by P-EP in 1986), on the basis that the previously supplied generators had already been determined to have met all applicable (including NRC) requirements.

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* Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10, "Energy," of the Code of Federal Regulations (Appendix B to 10 CFR Part 50).

P-EP adapted PG&E's technical and quality procurement specifications into its own procurement specifications, including drawings, bills of material, and material specifications. P-EP then either included or referenced its own documents in its procurement documents to PEM. P-EP audited PEM's quality program and determined that, although it was not based on Appendix B to 10 CFR Part 50, PEM's program nevertheless met the applicable requirements of Appendix B to 10 CFR Part 50. Therefore, P-EP believed that it could impose PG&E's requirements on PEM by invoking PEM's quality program. With the notable exception of 10 CFR Part 21, no other NRC requirements or PG&E requirements were formally imposed on PEM, although PG&E's list of critical items and characteristics was informally transmitted to PEM by P-EP.

As required by PG&E's purchase order, when the DCNPP2's generator was delivered, P-EP provided PG&E with a certificate of conformance that certified that the generator was produced in compliance with Appendix B to 10 CFR Part 50 and the reporting requirements of 10 CFR Part 21. This certification was based largely on P-EP's audit and determination regarding the equivalence of PEM's quality program to 10 CFR Part 50, Appendix B.

The NRC has classified the items set forth as nonconformances to the requirements in Appendix B to 10 CFR Part 50.

A. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part, that measures be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

   American National Standards Institute (ANSI) Standard N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants" (1971), and ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (1974), require, in part, (1) that the external interfaces between organizations performing work affecting quality of design be identified in writing and include those organizations providing criteria, designs, specifications, and technical direction; (2) that the responsibilities of organizations be defined and documented in sufficient detail to cover the preparation, review, and approval of design documents involving design interfaces; (3) that systematic methods be established for communicating needed design information across external design interfaces, including changes to the design information as work progresses; and (4) that design information transmitted from one organization to another be documented in specifications, drawings, or other controlled documents that are uniquely identified and issued by authorized persons. These requirements were imposed on P-EP by PG&E's purchase order and, therefore, applicable to PEM's manufacture of PG&E's new (no. 2-3) emergency diesel generator set for the Diablo Canyon Nuclear Power Plant Unit 2.

Contrary to these requirements, in Section 4, "Design Control," of the "Quality Manual Volume 1" (QMV1), PEM failed to establish adequate measures to control the design interface between it and P-EP. These measures consisted of the review, approval, release, distribution, and revision of design documents affected by this design interface.

PEM failed to demonstrate that the results of its design translation activities were equivalent to the design requirements specified by P-EP. P-EP provided its design drawings and specifications to PEM because PEM manufactures P-EP's generators. PEM's engineering organization translated P-EP's design specifications into its own PEM specifications, drawings, procedures, and instructions. The documents produced by PEM were not reviewed or approved by P-EP before use, and PEM-initiated design changes were not controlled by documented procedure until December 1990 when PEM issued Departmental Procedure 03A00 , "Processing of Engineering Change," well after the design activities for PG&E's generator were completed. Although PEM performed equivalency evaluations of its drawings, procedures, and material specifications used to fabricate and assemble PG&E's generator, PEM did not adequately document (1) the critical requirements or acceptance criteria compared during the equivalency evaluation and (2) the results of the equivalency evaluation or other basis to support PEM's conclusion that its drawings, procedures, and material specifications were equivalent to P-EP's. Therefore, PEM failed to establish adequate measures to control its design interface activities and to demonstrate adequate design equivalency evaluations (Nonconformance 99901065/91-01-01).

B. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part, that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the component.

ANSI N45.2 (1971) and ANSI N45.2.11 (1974) require, in part, (1) that measures be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the function of the component; (2) that the design inputs be identified, documented, and their selection reviewed and approved;

(3) that specified parts, equipment, and processes be suitable for the required application; and (4) that specified materials be compatible with each other and the design environment conditions to which the material will be exposed.

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

Contrary to these requirements, in Section 4, "Design Control," and Section 7, "Purchaser Supplied Product," of the QMV1, PEM failed to establish adequate measures to provide for the selection and review for and verification of suitability of application for materials, parts, and equipment that were procured as commercial grade items and were essential to the generator's ability to perform its intended design and safety-related function (dedication).

PEM failed to adequately verify the properties or attributes of certain materials, parts, and equipment that were used in the fabrication and assembly of PG&E's generator and that also directly affect the generator's ability to perform its intended design and safety-related function. Specifically, PEM failed to ensure the suitability of (1) the rotor pole magnet wire wrapped with varnished insulation tape that was specified to be unvarnished, (2) the Bakelite electrical separation ring that was used as a load-bearing component part of the rotor shaft support assembly without an engineering basis for the design, and (3) certain materials, parts, and equipment that were accepted based on certificates of conformance from PEM's suppliers that were not audited to verify that their measures to control design, processes, and material changes were adequately implemented. Therefore, PEM failed to establish adequate measures for the selection and review for suitability of commercial grade items and to demonstrate an adequate dedication of these items (Nonconformance 99901065/91-01-02).

Dated at Rockville, Maryland,
this 13th day of February 1992.

Enclosure 2

INSPECTION REPORT

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

Report No.:	99901065/91-01
Docket No.:	99901065
Company:	NEI Peebles Limited, Peebles Electrical Machines (PEM) East Pilton Edinburgh, Scotland EH5 2XT
Industry Activity:	PEM manufactures generators and spare and replacement parts for use in emergency ac power systems supplied by its sister company, NEI Peebles - Electric Products, Inc. (P-EP)
Inspection Conducted:	September 23 through 27, 1991
Inspection Team:	Steven M. Matthews, Team Leader, NRR Stephen D. Alexander, Environmental Qualification and Test Engineer, NRR Gregory C. Cwalina, Section Chief, Special Projects Section, NRR

Prepared by: Steven M. Matthews 2/12/92
Steven M. Matthews, Team Leader — Date
Reactive Inspection Section 1
Vendor Inspection Branch

Approved by: Uldis Potapovs 2-12-92
Uldis Potapovs, Chief — Date
Reactive Inspection Section 1
Vendor Inspection Branch

Inspection Bases:	10 CFR Part 21, Appendix B to 10 CFR Part 50
Inspection Scope:	To assess PEM's compliance with regulatory requirements and licensees' procurement requirements through a performance-based evaluation of its engineering, procurement, fabrication, assembly, and tests
Plants Affected:	All licensees with P-EP power generators

CONTENTS

1 INSPECTION SUMMARY

1.1 Nonconformances

1.1.1 Nonconformance 99901065/91-01-01

Peebles Electrical Machines (PEM) failed to meet Criterion III, "Design Control," of Appendix B to 10 CFR Part 50;* and American National Standards Institute (ANSI) Standard N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants" (1971); and ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants," (1974). In Section 4, "Design Control," of "Quality Manual Volume 1," (QMV1), Issue 7, April 14, 1989, PEM failed to (1) establish adequate measures to control the design interface activities between it and its sister company, NEI Peebles - Electrical Products, Inc. (P-EP), of Cleveland, Ohio, (2) demonstrate that the results of PEM's design translation activities were equivalent to the design requirements specified by P-EP, (3) adequately document the critical requirements or acceptance criteria compared during the equivalency evaluation, and (4) adequately document the results of the equivalency evaluation or other bases to support PEM's conclusion that its drawings, procedures, and material specifications were equivalent (see Section 3.5.2 of this report).

1.1.2 Nonconformance 99901065/91-01-02

PEM failed to meet Criterion III, "Design Control," and Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B to 10 CFR Part 50; ANSI N45.2 (1971); and ANSI N45.2.11 (1974). In Section 4, "Design Control," and Section 7, "Purchaser Supplied Product," of the QMV1, PEM failed to (1) establish adequate measures to provide for the selection and review for and verification of suitability of application for materials, parts, equipment, and services that were procured as commercial grade items and were essential to the generator's ability to perform its intended design and safety-related function and (2) ensure the suitability of the rotor pole magnet wire, the Bakelite electrical separation ring, and certain materials, parts, and equipment that were accepted based on certificates of conformance (COC) from PEM's suppliers that were not audited to verify that their measures to control design, processes, and material changes were adequately implemented (see Section 3.6.3 of this report).

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* Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50, "Domestic Licensing of Production and Utilization Facilities," of Title 10, "Energy," of the Code of Federal Regulations (Appendix B to 10 CFR Part 50).

1.2 Unresolved Item (99901065/91-01-01)

P-EP provided the material specification for the rotor pole magnet wire to PEM in purchase order (PO) 16271, which specified that magnet coil wire be provided in accordance with P-EP's material specification MW-25.3, "Magnet Wire - Round, Square, or Rectangular - Unvarnished Fused Polyester Glass Covering, With or Without Enamel Undercoat, Class F (155°C)," dated June 24, 1977. PEM procured the wire from its supplier by PO EM31035 (original), dated April 27, 1990. In its PO, PEM specified that "rotor copper-unvarnished double dacron glass insulated square magnet wire" be used. PEM also listed material specifications that corresponded to those in MW-25.3 and required certification, by a COC, of the chemical composition of copper, the conductor resistivity, and the insulation dielectric "stress" (sic) (strength). The COC, written in French, stated that the material was *Fil de cuivre guipé 2 DAGLAS Imprégné Classe F...* (which means copper wire wrapped with double dacron glass, impregnated, Class F). PEM accepted the wire and used it to wind the rotor poles. However, the team noted that the French word, *imprégné*, means impregnated and that fiber insulation material is commonly impregnated with varnish, indicating that the insulation would not have been unvarnished as specified. Accordingly, the PEM engineers confirmed that the supplied wire had been varnished. Therefore, the wire did not meet the P-EP material specification nor the PEM PO requirement for unvarnished insulation.

PEM immediately informed P-EP of the deviation; whereupon, P-EP reportedly indicated to PEM that P-EP would perform a deviation evaluation (pursuant to 10 CFR Part 21) regarding the varnished insulation, including an evaluation of the compatibility of the varnish with, and its effects on the adhesion properties of, the other materials (such as epoxy adhesive) used in the assembly of the rotor poles. The results of P-EP's and PEM's evaluation of this deviation were not reported to the team before the exit meeting with PEM on September 27, 1991 (see Section 3.6.3 of this report).

2 STATUS OF PREVIOUS INSPECTION FINDING

The NRC's previous inspection, conducted October 6 through 8, 1986, and documented in the U.S. Nuclear Regulatory Commission's (NRC's) Inspection Report 99901065/86-01 did not result in identifying any findings to be addressed during this inspection.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

During the entrance meeting on September 23, 1991, the NRC's inspection team met with PEM's staff and discussed the scope of the inspection, outlined areas of concern, and established working interfaces. The NRC inspection team explained the relationship of NRC requirements to PEM's activities associated with its manufacture of an emergency ac power generator for P-EP. The NRC quality requirements applicable to the safety-related (Class 1E) generator P-EP procured from PEM for an NRC licensee, Pacific Gas and Electric Company (PG&E), are contained in Appendix B to 10 CFR Part 50. This relationship is discussed further in Section 3.2 below.

The team observed activities, held discussions with PEM's staff, and reviewed records and procedures. The specific areas and documentation reviewed, and the team's findings are described in Sections 3.3 through 3.6 of this report. The table, "A Comparison of PG&E's Purchase Order Revisions 1 and 3 for Critical Items and Their Critical Characteristics," located at the end of Section 3.6, provides a comparison of the critical items and their critical characteristics as expressed by PG&E in Revisions 1 and 3 of its PO to P-EP. The Appendix lists the persons who participated in and who were contacted during the inspection. During the exit meeting on September 27, 1991, the team summarized the inspection findings, observations, and concerns with PEM's management.

3.2 Background

The Pilton Works of Peebles Electrical Machines (PEM) in Edinburgh, Scotland, is a sister company of, and the manufacturer for, NEI Peebles - Electric Products, Inc. (P-EP) of Cleveland, Ohio. Both companies are subsidiaries of NEI Peebles Limited. P-EP provided the sales and services office for all of the power generating equipment manufactured by NEI Peebles Limited and sold to U.S. customers. Therefore, the background of P-EP and its relationship to PEM is important to, and an integral part of, the inspection of PEM and the inspection team's use of the NRC's requirements in Appendix B to 10 CFR Part 50 and 10 CFR Part 21 as its inspection criteria.

P-EP's facility in Cleveland, Ohio, was originally known as Electric Products Incorporated (EPI) and, under various names, supplied over 120 generators to the U.S. nuclear industry. EPI was purchased by Portec, Inc., in 1969, and was known as the Electric Products Division of Portec, Inc. Portec sold the company in 1979 to Parson Peebles, a subsidiary of Northern Engineering Industries Limited (NEI) of England. NEI is a wholly owned subsidiary of the Industrial Power Group of Rolls-Royce.

The Cleveland facility was known at that time as Parson Peebles Electric Products, Inc. (also EPI). Subsequent to Parson Peebles' purchase of the Cleveland facility, NEI reorganized its Parson Peebles operations under the name of NEI Peebles Limited and the Cleveland facility became NEI Peebles - Electric Products, Inc. P-EP's Cleveland manufacturing facility was closed in September 1984 and moved to PEM's Pilton Works in Edinburgh, Scotland. The organizational structure of NEI Peebles Limited at the time of this inspection was such that the Vice President and General Manager of P-EP reported directly to the Manager of PEM.

Since 1984, PEM has manufactured the generators and many of the spare and replacement parts that P-EP supplied to the U.S. nuclear industry. PEM recently completed the fabrication, assembly, and testing of a safety-related (Class 1E) emergency ac power generator for PG&E's new sixth (no. 2-3) emergency diesel generator (EDG) set for its Diablo Canyon Nuclear Power Plant Unit 2 (DCNPP2). At the time of this inspection, PEM was fabricating a safety-related emergency ac power generator for Washington Public Power Supply System's Nuclear Project 2 (WNP2). The generator for WNP2 was procured by PO C-30464, dated November 29, 1990 (P-EP shop order no. S-1141, serial no. 260505/1). Although the team focused its inspection activities on the completed generator for PG&E's DCNPP2, the concerns discussed in this report may have generic implications for WNP2's generator and any similar generators, or spare and replacement parts, built by PEM and supplied to P-EP to other licensees.

The NRC quality requirements applicable to PG&E's procurement of this generator for DCNPP2 are contained in Appendix B to 10 CFR Part 50. Other NRC requirements applicable to PG&E's procurement of this generator are contained in 10 CFR Part 21, "Reporting of Defects and Noncompliance," because this procurement constituted procurement of a basic component as defined in 10 CFR Part 21. General NRC technical requirements for this generator to be used as an "alternate ac power source," as defined in 10 CFR 50.2, are contained in 10 CFR 50.2, 10 CFR 50.63 (station blackout), and Criterion 17, "Electric Power Systems," and Criterion 18, "Inspection and Testing of Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. Applicable NRC requirements related to identified licensing and design basis events (DBE), specifically, seismic qualification, are contained in Criterion 4, "Environmental and Dynamic Effects Design Bases," of Appendix A to 10 CFR Part 50.

DCNPP's five existing emergency ac power generators (serial nos. 16908022 through 16908026) installed on EDG nos. 1-1, 1-2, 1-3, 2-1, and 2-2, were procured in 1969 from the Electric Products Division of Portec, Inc., and manufactured in the Cleveland facility, described above. PG&E procured a spare generator (serial no. 38604851) in 1986 from P-EP, specifying that it be identical (i.e., like for like) to DCNPP's five 1969 generators. PEM manufactured the 1986 spare generator in its Pilton Works in Edinburgh, Scotland.

The generator for DCNPP's new 2-3 EDG was procured by PO ZS-1539-AB-9, Revision 0, dated January 16, 1990, in which PG&E requested P-EP to supply one 4.16-kV, 2600-kW, 60-Hz, 3-phase, 8-pole, 900-rpm, single-bearing, engine-driven, ac synchronous generator. The generator was to be supplied as a design Class 1E basic component in accordance with PG&E's Engineers Material Memorandum (EMM) DC2-3322-BRH-E, Revision 0, dated January 5, 1990. In the EMM, PG&E required that the generator be identical to PG&E's 1986 spare generator and DCNPP's five 1969 generators on the basis that the previously supplied generators had already been determined to have met all applicable requirements including the NRC's quality and technical (including seismic DBE) requirements. PG&E's apparent strategy to demonstrate compliance with the requirements for safety-related equipment suitability, including DBE (seismic) and any environmental qualification requirements, was to procure the generator on the basis of a like-for-like comparison with the 1969 generators, which were presumably fully qualified.

In its acceptance of the PO from PG&E, P-EP accepted the responsibility to assure overall compliance with all the applicable provisions of Appendix B to 10 CFR Part 50 and the reporting requirements of 10 CFR Part 21. PG&E's EMM, Attachment A, "Specification for Supplier's Quality Assurance Program," Specification SP-D-Peebles (SP-D-Peebles), Revision 3, dated October 11, 1989, required in Section 1.0, "General," that the supplier's quality assurance (QA) program for supplying equipment and components comply with British Standards Institution's British Standard (BS) 5750, Part 1, "Specification for Design/Development, Production, Installation, and Servicing" (ISO 9001-1987, Quality systems - Model for quality assurance in design/development, production, installation, and servicing), Part 2, and Part 3, and that the supplier's QA program for supplying engineering services comply with Appendix B to 10 CFR Part 50 and ANSI N45.2-1971. In Section 3.0, "Quality Assurance Program (Edinburgh, Scotland)," SP-D-Peebles required that the supplier's QA program detail the procedures and methods used to ensure that all supplier's (PEM) activities satisfy the requirements of BS 5750, Part 1 (ISO 9001-1987), and Parts 2 and 3. In Section 4.0, "Quality Assurance Program (Cleveland Facility)," SP-D-Peebles required that the supplier (P-EP) ensure compliance with the applicable requirements of Appendix B to 10 CFR Part 50, ANSI N45.2-1971, and all other codes or standards referenced in the PO. SP-D-Peebles also imposed the requirements of numerous other ANSI nuclear standards, including ANSI N45.2.11-1974. Additionally, PG&E's PO for this safety-related generator, defined as a basic component in 10 CFR 21.3, invoked the reporting requirements of 10 CFR Part 21.

P-EP adapted PG&E's technical and quality procurement specifications into its own procurement specifications, including drawings, bills of material, and material specifications. P-EP then either included or referenced its own documents in its procurement documents to PEM. P-EP audited PEM's quality program and determined that, although it was not based on Appendix B to 10 CFR Part 50, PEM's program nevertheless met the applicable requirements of Appendix B to 10 CFR Part 50. Therefore, P-EP believed that it could impose PG&E's requirements on PEM by invoking PEM's quality program. With the notable exception of 10 CFR Part 21, no other NRC requirements or PG&E requirements were formally imposed on PEM, although PG&E's list of critical items and characteristics was informally transmitted to PEM by P-EP.

PEM completed and tested PG&E's generator during January and February 1991. PEM issued a COC to PE-P on February 27, 1991, which certified that the generator (serial no. 260274/1) was designed, manufactured, inspected, and tested in accordance with its quality program and the requirements of PE-P's PO 16271. On March 1, 1991, PEM shipped the completed generator to PG&E's contractor, GEC Alsthom of Toronto, Canada, for the final assembly and skid-mounting of the EDG set and the combined testing of the diesel engine, the generator, and the EDG's auxiliary systems. As required by PG&E's PO, when the DCNPP2's generator was delivered, P-EP provided PG&E with a COC that certified that the generator was produced in compliance with Appendix B to 10 CFR Part 50 and the reporting requirements of 10 CFR Part 21. This certification was based largely on P-EP's audit and determination regarding the equivalence of PEM's quality program to Appendix B of 10 CFR Part 50. In its COC to PG&E dated March 27, 1991, P-EP certified that the generator complied with the provisions of PG&E's PO 2S-1539-AB-9 and added that the generator was the same in form, fit, and function, as the original generators supplied in 1969 (serial nos. 16908022 through 16908026).

The last NRC inspection of PEM was conducted on October 6 through 8, 1986; P-EP, however, was last inspected by the NRC on August 5 through 9, 1991. The inspection of P-EP was conducted primarily to evaluate P-EP's QA program and its implementation as it was applied to the safety-related generator supplied to PG&E. For the purposes of clarity and understanding, this report of the inspection of PEM contains references to certain activities that were performed by P-EP or to certain concerns that were identified during the inspection of P-EP. In either case, the NRC report of the inspection of P-EP, Inspection Report 99900772/91-01, describes all references to P-EP contained herein.

3.3 P-EP's Procurement Documents Issued to PEM

P-EP issued PO 16271 (shop order no. S-1128) to PEM on January 29, 1990, for PG&E's generator. The PO specified that the generator be identical to the generator previously ordered by P-EP's PO 14673, dated February 25, 1986 (shop order no. S-1076, and job no. 259132), with some exceptions. The most significant exceptions were (1) the phase rotation was changed per Drawing C-08391U, (2) the pole insulation specification was changed from polyester resin to epoxy resin MV-20.9 per Specification EI-1.5.1, and (3) the rotor pole assembly was changed per Drawing A-66843-7, Revision 2. P-EP's PO also imposed the reporting requirements of 10 CFR Part 21 on PEM. P-EP required that NEI Peebles Limited's QA program comply with Attachment A (SP-D-Peebles) of PG&E's EMM, and provide the generator's specifications for (1) the tests to be witnessed, (2) the applicable material specifications, (3) the applicable manufacturing specifications, and (4) the documentation requirements. P-EP's PO further required PEM to provide certification that PEM's manufacturing process complied with P-EP's and PEM's drawings and PEM's QA program, Issue 5, dated December 18, 1986, which was imposed because it was applicable to PG&E's 1986 spare generator. P-EP stated that PEM's QA program was equivalent to the requirements in Appendix B to 10 CFR Part 50, as discussed above and in Section 3.4 of this report.

The original issue of P-EP's PO did not identify the generator's critical items. Although P-EP issued several change orders to its PO during the fabrication, assembly, and test of PG&E's generator, it still failed to identify the items of the generator specified as critical by PG&E. This issue is discussed further in Section 3.6.1 of this report.

3.4 PEM's Quality Assurance Program

NEI Peebles Limited's Quality Manual Volume 1, Issue 7, dated April 14, 1989 (known in this report as PEM's QMV1), delineated the QA program applicable to the overall operations of PEM and Peebles Power Transformers. The QMV1 was developed by NEI Peebles Limited to comply with the requirements of BS 5750, "Quality Systems," Part 1 (ISO 9001-1987). However, Attachment A (SP-D-Peebles) of PG&E's EMM required that the QA program for equipment and components comply with BS 5750, Part 1 (ISO-9001-1987), Parts 2 and 3.

P-EP's audits of PEM, dated September 30, 1985, and August 7 through 9, 1989, were conducted to qualify PEM as a supplier of safety-related components and parts. According to P-EP, these audits qualified PEM to supply components and parts to P-EP in accordance with PEM's QMV1, which met the applicable requirements of Appendix B to 10 CFR Part 50 as well as the reporting requirements of 10 CFR Part 21. P-EP developed an equivalency evaluation of PEM's QMV1 and concluded that the QMV1 met the requirements of Appendix B to 10 CFR Part 50. Since the requirements of Appendix B to 10 CFR Part 50 are the basis for acceptance of safety-related components supplied to the U.S. nuclear industry, the team's evaluation of PEM's QA program and its implementation was based on those requirements.

However, P-EP's reports of the 1985 and 1989 audits did not document objective evidence to substantiate that PEM's QMV1 established adequate measures to provide control over certain activities affecting the quality of safety-related components. Specifically, P-EP failed to show that PEM had measures (1) for the control of design interface activities with P-EP; (2) for the selection and review for suitability of application of material, parts, equipment, and processes; and (3) for the commercial grade dedication of items essential to the safety-related function of the generator. P-EP also failed to demonstrate that PEM's dedication activities, for critical parts procured by PEM as commercial grade, resulted in establishing reasonable assurance that the parts and the completed generator will perform their respective design and safety-related functions. This concern is discussed further in Section 3.6.3 of this report.

3.5 Design Control

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, and ANSI N45.2.11-1974, require that measures be established to ensure that applicable regulatory requirements and the design bases are correctly translated into specifications, drawings, procedures, and instructions and that design changes be subject to design control measures commensurate with those applied to the original design. Measures also shall be established for the identification and control of design interfaces and for coordination among participating design organizations including procedures for the review, approval, release, distribution, and revision of documents involving design interfaces and for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the component.

P-EP maintained the overall engineering and design control responsibility, in addition to providing sales and services support, for the generators and other power generating equipment procured by the U.S. nuclear industry. However, PEM's engineering and design organization performed independent design activities. The team evaluated PEM's design activities in the areas described separately below.

3.5.1 Design Basis Documentation

In its PO, PG&E required that the generator be like-for-like to its 1986 spare generator and DCNPP's five 1969 generators. The team reviewed P-EP's and PEM's control of the generator's engineering design basis that would be necessary to establish the like-for-like relationship of the new generator to the design basis of the generators previously supplied. Specifically, the team reviewed the synergistic effect of the changes that were made to the original engineering design bases since 1969 to determine what, if any, effect those changes had on PG&E's like-for-like procurement requirement.

P-EP's design basis reconciliation to the original 1969 design consisted of a drawing change review dated June 24, 1991. P-EP's review encompassed the drawings associated with PG&E's generator since 1984, including all revisions. However, P-EP's reconciliation of design changes for the generator was documented and verified only to 1984 when the manufacturing facility closed in Cleveland, Ohio. Therefore, neither P-EP nor PEM could substantiate that the new generator was like-for-like to PG&E's five existing 1969 generators.

3.5.2 Design Interface

A significant design interface existed between P-EP and PEM. Although P-EP maintained the overall responsibility for the generator's engineering and design control, PEM's engineering and design organization functioned completely independent of P-EP's organization and it performed certain independent design activities. P-EP provided its design drawings, procedures, and material specifications to PEM, and PEM's engineering organization translated them into PEM specifications, drawings, procedures, and instructions to fabricate and assemble PG&E's generator. This process also included converting dimensions and tolerances from English values to their metric equivalents.

PEM-produced documents were not reviewed or approved by P-EP before use, and PEM-initiated engineering changes were not controlled by documented procedures until December 1990. The measures established in Section 4, "Design Control," of PEM's QMV1 did not provide for adequate procedures between PEM and P-EP for the review, approval, release, distribution, and revision of documents involving their respective design interface. This deficiency appeared to have resulted from the "sister company" relationship of PEM and P-EP and the daily interface of their respective staffs. Although PEM issued

Departmental Procedure 03A004, "Processing of Engineering Change," Revision 0, dated December 17, 1990, it did not affect PEM's design interface activities during most of the fabrication and assembly of PG&E's generator.

PEM performed equivalency evaluations of its drawings, procedures, and material specifications to P-EP's drawings, procedures, and material specifications and initiated design changes, as required. The equivalency evaluations were not auditable because (1) P-EP's drawings, procedures, or material specifications were not always available for comparison to PEM's documents and (2) the documentation of the evaluations consisted of only a brief summary of the drawing, procedure, or material specifications. In its equivalency evaluations, PEM failed to adequately document (1) the critical requirements or acceptance criteria compared during the evaluation and (2) the results of the evaluation or basis that supported PEM's conclusion that the documents were equivalent to P-EP's.

Therefore, PEM failed to establish adequate measures to control its design interface activities and to demonstrate adequate design equivalency evaluations. This is Nonconformance 99901065/91-01-01.

3.6 Dedication Process

Dedication is the selection and review for and verification of suitability of application to ensure the adequacy of critical parameters (characteristics) of commercial grade items that are to be used in safety-related applications. PG&E's generator is a complex component composed of several critical parts that directly affect the ability of the generator to perform its design and safety-related functions. The credible failure mechanism or long-term degradation of the part could adversely affect the generator's ability to perform its safety-related function. PG&E was aware that its generator was actually to be manufactured by P-EP's sister company, PEM, and became involved in the dedication of certain commercial grade parts by selecting the critical parts of the generator and specifying their critical characteristics.

3.6.1 Selection of Critical Items

PG&E's PO ZS-1539-AB-9 (described in Section 3.2 of this report) was modified by Revision 1, February 2, 1990, to add Attachment F, "Critical Items Listing & Dedication Testing," to its EMM. Attachment F listed 14 critical items and their associated critical characteristics and required P-EP to verify the PG&E-identified critical characteristics for each of the 14 critical items by performing tests. PG&E further required that P-EP's verification tests and their respective acceptance criteria be furnished to PG&E for approval before the materials and parts were installed or used. P-EP subsequently passed to PEM the responsibility for procuring seven of the items and verifying their critical characteristics. However, P-EP did this indirectly by identifying only those items it would procure and supply to PEM as safety-related items. P-EP transmitted PG&E's list of items and their critical characteristics to PEM without making it a part of or referencing it in P-EP's PO.

In its PO to PEM, P-EP identified the material specifications applicable to certain parts of the generator and required PEM to supply certificates of analysis, test reports, or certificates of conformance for those materials and parts. The material specifications specified such items as materials, identification, ordering information, approved suppliers, and storage requirements. In many cases, the material specification contained an approved suppliers list that included specific products, listed by trade name, that P-EP had approved as meeting the material specification.

The team immediately identified three concerns with these actions that were distinct from other procurement and technical issues discussed in Section 3.6.3 of this report. First, PG&E's selected critical items were not made a formal part of P-EP's PO for procurement of the generator from PEM. Second, the listed critical items (including their critical characteristics) did not correspond to P-EP's material specifications and other requirements specified in the PO. Third, P-EP did not amend its PO to PEM to address the revisions to PG&E's PO.

Revision 2 to PG&E's PO, dated February 22, 1990, addressed specific data that P-EP was to provide to enable PG&E to perform the seismic analysis of the generator.

Revision 3 to PG&E's PO, dated February 6, 1991, included significant revisions to EMM Attachment A (SP-D-Peebles), and the critical items list of Attachment F. In Attachment A, Revision 5, dated November 15, 1990, PG&E imposed numerous requirements on P-EP that were not previously imposed in Revision 3, which was included in PG&E's original PO. The most significant additions are listed below.

- Section 4.2.6(1), requirements for critical material, parts, or components procured as commercial grade items

- Section 4.2.8, requirements for the identification and control of materials and items

- Section 4.2.9, requirements for a test program to identify and document all testing required to demonstrate that items will perform satisfactorily in service

* Section 4.2.10, requirements for the control of measuring and test equipment

In EMM Attachment F, PG&E changed the list of critical items from 14 (shown in Revision 1) to 27 (in Revision 3). Several of the critical characteristics for those items that were to be verified also changed. In addition, certain subassemblies that were previously identified as critical items were divided into individual parts of the subassembly and listed separately. For example, the brushes and brush holder were listed as item 7 in Revision 1 and the critical characteristics were identified as size and shape and final generator test for resistance, material, and contact pressure. However, Revision 3 listed the brushes and the brush holder separately as items 20 and 19, respectively, and identified configuration as the only critical characteristic for both items. A comparison of the critical items and their critical characteristics, as expressed by PG&E in Revisions 1 and 3 of its PO, is provided in the table located at the end of Section 3.6.3.

P-EP's generic failure modes and effects analysis (FMEA) was applicable to all rotating electrical machinery produced and was part of P-EP's technical documentation that demonstrated a generator's compliance with the requirements of the Institute of Electrical and Electronics Engineers (IEEE) Standard 323, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Standard 344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The FMEA included the credible failure mode for each individual part of the generator assembly and a criticality level (see definitions below) was assigned to the part on the basis of the effect of the part's credible failure mode on the ability of the generator to perform its safety-related function.

<u>Level 1</u> - catastrophic failure (i.e., will not operate at all, extensive repair needed)

<u>Level 2</u> - severely degraded (i.e., operates far off-normal giving warning that a failure will soon occur, extensive repairs needed)

<u>Level 3</u> - degraded (i.e., operates off-normal but with adequate warning of an impending failure, repairs simple if done promptly)

<u>Level 4</u> - minor degradation (i.e., operat . near-normal but gives a warning of eventual failure, situation deteriorates very slowly; repairs are simple)

<u>Level 5</u> - no effect (i.e., part does not affect operation, repairs are part of maintenance)

According to P-EP, PG&E's PO did not impose qualification of the generator to the requirements of IEEE Standards 323 or 344 and PG&E did not procure P-EP's FMEA documentation for use in the selection of critical items or their critical characteristics. P-EP also stated that the extent of its involvement in PG&E's selection of critical items and their critical characteristics was limited to only an agreement with PG&E to perform testing necessary to verify the critical characteristics of the critical items identified by PG&E in Attachment F of Revision 1 to its PO.

Both P-EP and PEM reported that they had not been involved in PG&E's selection of the critical items or their critical characteristics listed in Revision 3 of PG&E's PO. Furthermore, PG&E's generator was completed when Revision 3 was issued; therefore, neither P-EP nor PEM considered Revision 3 during its design, procurement, and manufacturing activities.

Because of the minimal involvement of P-EP's engineering organization in PG&E's selection of critical items and their critical characteristics listed in Revision 1, the team was concerned that PG&E's selected list of critical items may not have been sufficiently comprehensive to ensure that all items were included, specifically, those items with a credible failure mode or that, in a degraded condition, could adversely affect the generator's ability to perform its design and safety-related function. The team reviewed P-EP's generic FMEA and discussed the technical bases for the critical items and their critical characteristics with the engineering staffs of both P-EP and PEM to determine whether PG&E's Revision 1 list of 14 critical items, or its Revision 3 list of 27 critical items, included all parts that are critical to the generator's ability to perform its design and safety-related function.

According to P-EP's FMEA, the generator's two major design parameters with regard to the effects of long-term degradation and cyclic fatigue were its operating temperatures and cyclic loading or high vibration forces. On the basis of these design parameters, criticality levels 1 or 2 were assigned in the FMEA to critical items such as the stator windings, leads and their connections, rotor pole windings, roller bearings, rotor shaft, coil supports, and slip rings. From its review of P-EP's generic FMEA documentation, the team determined that PG&E's lists of critical items did not adequately envelope all of the generator's critical parts having a design or safety-related function (i.e., the slip-ring mounting sleeve insulator and the temperature and vibration indicating devices, as discussed in Section 3.5.3 of NRC's Inspection Report 99900772/91-01).

For a complex assembly such as a generator, the selection of critical items and the determination of their critical characteristics would require the involvement of both the licensee's and supplier's engineering staffs. Although in

Revision 3 of its PO, PG&E revised the introductory statement of Attachment F, in part, to state that this listing was based on discussions between the staffs of PG&E and NEI-Peebles at QA audit meetings held in Cleveland, Ohio, during December 1989 and in Edinburgh, Scotland, during October 1990, PEM and P-EP considered this interface activity to be limited to those critical items identified in Revision 1 to PG&E's PO, and they believed the interface activity was nonexistent for the critical items identified in Revision 3 of the PO. Furthermore, PEM and P-EP had completed PG&E's generator when Revision 3 was issued; therefore, Revision 3 was not considered during the design, procurement, and manufacturing activities of the generator.

Although P-EP agreed to perform the testing necessary to verify the critical characteristics of the items identified in Revision 1 of PG&E's PO as critical, P-EP did not (1) identify the items critical to the generator's ability to perform its intended safety-related function or (2) perform a technical evaluation of the items identified in Revision 1 of PG&E's PO to determine the adequacy of PG&E's list of critical characteristics. For the critical characteristics selected by PG&E, P-EP failed to demonstrate their relevance (1) to the properties or attributes of the item necessary to withstand the effects of long-term degradation, (2) to the credible failure mode of the item, and (3) to the ability of the item to perform its safety-related function. P-EP failed to substantiate that the PG&E-identified critical items included <u>all</u> parts that were required for the generator to perform its safety-related function and that the PG&E-identified critical characteristics were adequate to ensure that the part will perform its safety-related function. Consequently, an evaluation of P-EP's generic FMEA identified additional critical characteristics for certain items that were not identified or verified by PEM during its commercial grade dedication activities and were not identified by PG&E in its Revision 1 to the PO.

3.6.2 Review for Suitability

PEM and P-EP procured the critical items identified in Attachment F of Revisions 1 and 3 of PG&E's PO as commercial grade items. The critical items procured by PEM and P-EP are identified in the table at the end of Section 3.6.3. P-EP procured 7 of the 14 items listed in Revision 1 of PG&E's PO (or 10 of the 27 items listed in Revision 3) and supplied them to PEM for installation in the generator assembly. The 7 remaining critical items listed in Revision 1 of PG&E's PO (or 17 of the 27 items listed in Revision 3) were procured by PEM from its suppliers in Europe.

PEM procurement practice consisted of purchasing items from suppliers that were selected on the basis of their performance history, which was determined through the general knowledge and experience of PEM's staff. Although this procurement practice, or custom, is commonplace for European manufacturers, the NRC placed conditions on its acceptance of this method to dedicate commercial grade items. In its Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," dated March 21, 1989, the NRC stated that supplier/item performance history was an acceptable method to dedicate commercial grade items provided (1) the established historical record is based on industry-wide performance data that is directly applicable to the item's critical characteristics and its intended safety-related application and (2) the supplier's measures to control changes in design, materials, and manufacturing processes have been adequately implemented as verified by audit.

Most of PEM's suppliers, however, were not audited to verify that their measures to control design, processes, and material changes were adequately implemented. The performance history data that were documented and verified did not establish performance data that were directly applicable to the item's critical characteristics or its intended safety-related application. For the most part, the POs to the suppliers of these items did not impose any quality and technical requirements and none imposed the reporting requirements of 10 CFR Part 21. Therefore, the critical items for PG&E's generator were procured by PEM as commercial grade from suppliers whose ability to adequately control changes in design, materials, and manufacturing processes had not been substantiated, as necessary to support the use of acceptable supplier/item performance history as an acceptable portion of PEM's commercial grade dedication activity.

3.6.3 Verification of Suitability

The team reviewed the drawings, procedures, and material specifications for the generator and examined similar components in fabrication for a comparable generator PEM was building for WNP2. In discussions with PEM staff, the team identified what appeared to be the most likely components corresponding to the PG&E list of critical items. The team reviewed the procurement documentation for the critical items procured by PEM and evaluated PEM's methods for meeting P-EP's procurement requirements. The team also evaluated the extent to which the PG&E-listed critical characteristics (as well as others) were ultimately verified by PEM. A summary of PEM's commercial grade dedication activities for a sample of the critical items specified by PG&E in Attachment F to Revision 1 and, where applicable, Revision 3 of its PO to P-EP is given below.

(1) Lead Wire (Revisions 1 and 3)

In Revision 1 of its PO to P-EP, PG&E identified lead wire (Attachment F, item 1) as a critical item and specified the critical characteristics as (1) dielectric strength, (2) number of strands, (3) the markings on the cable, and (4) the insulation thickness. However, in Revision 3 of PG&E's PO only configuration was specified as the critical characteristic for lead wire (Attachment F, item 16). PEM had specified the lead wire to be used for dc field leads (the segment from the brush-rigging to the external terminal box) without guidance from P-EP. In all the pertinent documentation provided by P-EP, the team could not identify any wire suitable for this application. The only document that may have referred to this wire specified wire of insufficient ampacity for this application. Therefore, PEM chose what appeared to be a suitable type of wire and procured it in a similar manner to other lead wire used for this generator. However, the wire was procured without apparent knowledge or consent of P-EP, and PEM did not verify the critical characteristics specified by PG&E

(2) Magnet Wire (Revisions 1 and 3)

In Revision 1 of its PO to P-EP, PG&E identified magnet wire (Attachment F, item 4) as a critical item. This insulated copper wire is wound in a coil of turns or windings (approximately 450 for this machine) around each of eight (for this 60-Hz, 900-rpm machine) laminated steel rotor poles. Each rotor pole creates a constant magnetic field from the direct current flowing in its windings, which induces alternating current in the stator windings (coils) as each pole passes the stator windings. A prime mover (in this case the diesel engine) turns the rotor shaft, which causes relative motion between the magnetic field of the rotor poles and the stator windings, inducing generator voltage and current. The generator is synchronous because the frequency of the output voltage and current is directly proportional to the speed of rotation of the rotor.

P-EP provided the material specification for the rotor pole magnet wire to PEM in PO 16271. The P-EP PO specified that magnet coil wire be provided in accordance with P-EP Material Specification MW-25.3, "Magnet Wire - Round, Square, or Rectangular - Unvarnished Fused Polyester Glass Covering, With or Without Enamel Undercoat, Class F (155°C)," dated June 24, 1977. This version of MW-25.3 provided detailed specifications and the codes and standards to be met for the wire and its insulating system, including enamel undercoat and fibrous (dacron and fiberglass tape) covering.

PEM procured the wire from its supplier, Insulation Systems & Machines, Ltd. (ISM), by PO EM31035 (original) dated April 27, 1990. In its PO, PEM specified, "rotor copper - unvarnished double dacron glass insulated square magnet wire," and listed material specifications that corresponded to those in MW-25.3. PEM required ISM to provide test certificates for the chemical composition of copper, the conductor resistivity, and the insulation dielectric "stress" (sic) (strength). ISM subsequently ordered the material from its Italian subsupplier, UDD-FIM, by PO P-00-86-48 (original), dated April 30, 1990. UDD-FIM supplied the material to ISM with a Quality Inspection Report (test certificate) and a COC. The COC, written in French, stated that the material was *Fil de cuivre guipé 2 DAGLAS Imprégné Classe F...* (copper wire wrapped with double dacron glass, impregnated, Class F). ISM provided the wire and documentation to PEM with a COC that certified the material met the requirements of PEM'S PO. PEM accepted the wire and used it to wind the rotor poles. However, the team noted that the French word *imprégné* means impregnated and that fiber insulation material is commonly impregnated with varnish; therefore, the insulation would not have been unvarnished as specified. PEM engineers contacted ISM who confirmed that the supplied wire had been varnished.

Therefore, the wire did not meet the P-EP material specification or the PEM PO requirement for unvarnished insulation. In addition, PEM had no documented analysis addressing the use of varnished insulation tape in this application and no information from P-EP regarding the basis for the specification of unvarnished insulation. Accordingly, PEM immediately informed P-EP of the deviation. P-EP agreed to perform a deviation evaluation (pursuant to 10 CFR Part 21) regarding the varnished insulation, including an evaluation of the compatibility of the varnish with, and its effects on the adhesion properties of, the other materials (such as epoxy adhesive) used in the assembly of the rotor poles. This is Unresolved Item 99901065/91-01-01.

Although not clearly documented, PEM was assumed to be responsible for dedication of the magnet wire for the rotor (presumably because it procured the wire). PEM's documented responsibility was to verify that the wire met the material specifications cited in P-EP PO 16271, and P-EP expected PEM would verify the PG&E-identified critical characteristics as well in the course of meeting the material specification and carrying out the specified testing.

In Revision 1 of its PO to P-EP, PG&E identified the critical characteristics of the magnet wire as size and shape, resistance, and insulation dielectric strength. Although these characteristics were critical, PG&E omitted other pertinent material properties of the magnet wire, such as mechanical strength and allowable bend radius, as well as characteristics of the insulation system, such as thermal capability. These characteristics were not merely manufacturing considerations because they could affect generator reliability given the stresses involved during normal operation of the generator (let alone the additional stresses from asynchronous events, adverse extremes of the normal service environment, or a design-basis event (DBE) such as seismic excitation). Although some of these characteristics may ultimately have been addressed by P-EP's material specification and final testing, PG&E had not identified them as critical.

ISM supplied a COC attesting that the wire met the required specifications and also supplied the COC and quality inspection report (test certificate) from UDD-FIM as required by PEM PO EM31035. However, although both the COC and the test report certify that the material met all specifications, there was no basis for acceptance of the COC. PEM did not survey either supplier and did not conduct independent testing to verify the accuracy of the COC or the test report. As a result, PEM accepted and used nonconforming material. This is one of several examples of PEM accepting a COC at face value with no audits, surveys, or verification testing to verify the validity of the COC.

The revision of June 24, 1977, of MW-25.3 listed approved suppliers and the trade names of their products. The approved magnet wire was listed as being available from two approved U.S. manufacturers and described as "Armored Polythermaleze + Dacron - Glass" (as manufactured by Belden Mfg. Co.) and also as "Polythermaleze 2000 + Dacron-Glass" (as manufactured by Phelps Dodge). Although PEM used one of these approved types of magnet wire, it obtained the wire through its regular supplier, ISM. ISM, in turn, procured the wire from a company in Italy called UDD-FIM who manufactured it under license from Phelps Dodge. However, PEM did not specify the material by trade name in its PO to ISM, which may have contributed to receiving the wrong material.

PEM prepared an engineering change note (ECN) to obtain P-EP approval to obtain the material specified in MW-25.3 from an alternate supplier to ensure conformance with QA requirements. However, the ECN was not prepared until November 15, 1990, nearly 7 months after the order had been placed with ISM and well after the wire had been received by

PEM. Nonetheless, P-EP replied that no addition to the material specification was required because the trade name was specifically identified on the material specification. Although PEM considered this response an approval, P-EP's reply was an inappropriate response because P-EP effectively abdicated its design control responsibility in granting what was tantamount to blanket supplier selection authority on the sole basis of the product's trade name.

(3) Lead to Coil Terminations (Revision 1)

In Revision 1 of its PO to P-EP, PG&E identified the lead to coil terminations (Attachment F, item 11) as critical items and specified the critical characteristics as brazing and weld materials. Revision 3 of PG&E's PO did not include the lead to coil terminations as critical items, although PEM's engineering staff agreed with the team that the lead to coil terminations were critical. Moreover, PEM pointed out that all connection and termination joints were critical to the generator's ability to perform its design and safety-related function.

The completed generator assembly contains several connections and terminations that can be classified into one of the following three types:

- brazed, high-temperature silver-solder joints that connect the magnet wires of the rotor poles to cable leads

- overlapped compression joints that connect copper conductors to copper conductors (e.g., the stator coil windings to other stator coil windings and the stator coil windings to the copper conductors of the parallel rings) or copper conductors to cable leads (e.g., the copper conductors of the parallel rings to the cable leads that run to the generator's main terminal box)

- crimped joints that connect cable leads to lugs (e.g., ring-tongue terminals used for bolted terminations)

PG&E identified the lead to coil terminations as critical items with critical characteristics listed as brazing and weld materials, even though weld materials are not used to perform brazing operations. PEM used brazed connections only to connect the magnet wires of the rotor poles to cable leads that run along the surface of the rotor shaft to the slip-ring assembly. However, PG&E did not identify the generator's other connections and terminations as critical items, even though PEM considered them to be critical.

PEM did not establish a documented procedure to control the high-temperature silver-solder brazing operation. PEM, however, d.d h. e skilled craft wich several years of experience to ma, the brazed joints. PEM failed to document (1) qualification of the brazing materials and methods used, (2) inspection of the brazed joints, or (3) verification that the joints were adequate and met expected quality and technical requirements.

To control the overlapped compression joints in the stator assembly, PEM developed Procedure R 6081, "Compression Jointing of Copper Conductors Within a Stator Winding Using AMP Products," dated November 20, 1990. PEM prepared trial joints for the overlapped compression joints that connect the stator coil windings to each other and the stator coil windings to the parallel ring to establish the fabrication parameters for the same type of compression joints to be performed during the manufacturing of the generator. However, PEM failed to document the results of the test and inspection of the qualifying trial joints. PEM also failed to document objective evidence of any inspection or verification to ensure that the joints made during fabrication were adequate and met expected quality and technical requirements.

PEM did not establish a documented procedure to control the crimped joints that connect the cable leads to ring-tongue terminal lugs that form bolted connections (1) at the terminal box for the cable leads that run from the stator's parallel rings, (2) at the slip-ring assembly for the cable leads that run along the rotor shaft from the rotor poles, and (3) at the brush-rigging assembly and the field terminal box for the cable leads that connect those two items. In addition, PEM failed to document objective evidence of its inspection or verification of the crimped joints to ensure that the joints were adequate and met expected quality and technical requirements.

(4) Roller Bearing (Revisions 1 and 3)

In Revision 1 of its PO to P-EP, PG&E identified the roller bearing (Attachment F, item 12) as a critical item and specified the critical characteristics as size and type, visual inspection (the team noted that this PG&E-identified characteristic is not a valid critical characteristic of the roller bearing), catalog number, and tolerances. However, in Revision 3 of its PO, PG&E specified the roller bearing's (Attachment F, item 6) critical characteristics as part number and configuration.

PEM issued a PO to its supplier, FAG (UK) Limited, for the roller bearing and specified, "spherical roller bearing, cat. no. 22226-C3, SKF or equiv." FAG issued a COC, dated September 14, 1990, to PEM for the roller bearing certifying that the roller bearing supplied (catalog no. 22226EAS-M-C3) was equivalent to the SKF-22226-C3 ordered. The difference in design between the two bearings was that the bearing ordered had a steel cage and the bearing supplied and installed had a forged cage. PEM evaluated the difference and determined that the roller bearings were equivalent.

Even though the spherical roller bearing was procured as a commercial grade item from a supplier that had not been audited, PEM accepted the COC for the bearing, as was its custom, and performed a receipt inspection. The results of the receipt inspection documented acceptance of the bearing after verification of the catalog no. and visual inspection for damage.

(5) Rotor Shaft (Revisions 1 and 3)

In Revision 1 of its PO to P-EP, PG&E identified the rotor shaft (Attachment F, item 13) as a critical item and specified the critical characteristics as "require dedication by factory test" without specifying what should be included in the test. However, in Revision 3 of its PO, PG&E specified the rotor shaft's (Attachment F, item 1) critical characteristics as material, configuration, and integrity.

In its PO to PEM, P-EP required that the rotor shaft forging comply with Material Specification MS-70.42, "Shaft Forging, Carbon Steel (Not Recommended for Welded Lands) Used for All Flanged Shafts and All Shafts Over 10-Inch Diameter," dated November 10, 1972. MS-70.42 specified the shaft material comply with American Society for Testing and Materials (ASTM) A-470, Class 1, "Vacuum-Treated Carbon and Alloy Steel Forgings for Turbine Rotors and Shafts." However, P-EP's Drawing C-67400-1, "Shaft, Single Bearing, Forged, Flanged for Alco Engine," Revision 7, dated November 19, 1990, specified that the shaft material comply with ASTM A-292, Class 1. The team determined that ASTM A-292 was superseded by ASTM A-469, "Vacuum-Treated Steel Forgings for Generator Rotors," and that P-EP Drawing C-67400-1 had not been revised to reflect ASTM A-469 for generator rotor shafts instead of the obsolete A-292 specification. The issue of concern is that PEM did not document a reconciliation of the apparent conflict between the material specified in the drawing and the material specified in

MS-70.42. Neither PEM nor P-EP documented the basis or rationale for ordering the generator's rotor shaft to a material specification intended for turbine rotors and shafts (ASTM A-470) as opposed to the material specification for generator rotors (ASTM A-469).

ASTM A-469 required a permeability test of the rotor shaft be performed in accordance with ASTM A-341, "Test Method for DC Magnetic Properties of Materials Using DC Permeameters and the Ballistic Test Methods," or ASTM A-773, "Test Method for DC Magnetic Properties of Materials Using Ring and Permeameter Procedures with DC Electronic Hysteresigraphs." ASTM A-470 did not require a permeability test of the rotor because the specification was intended for turbine rotors. Moreover, a permeability test was not performed or documented in PEM's inspection records for the rotor shaft. Neither P-EP nor PEM evaluated the necessity to determine the rotor shafts permeability; therefore, the proper material and its characteristics were not adequately verified by PEM.

PEM ordered the rotor shaft from La Forgia di Bollate s.p.a. of Milan, Italy. PEM's PO specified "shaft forging to Drawing B-67405-1, to be rough turned condition, material spec: ASTM A-470-77, Class 1, also BS-970 080 M40," even though PEM did not document an equivalency evaluation between ASTM A-470-77, Class 1, and BS-970 080 M40. La Forgia di Bollate issued its COC, dated December 6, 1990, to PEM and certified that the rotor shaft complied with PEM's Drawing B-67405-1 and Material Specification BS-970 080 M40. The COC also certified the shaft was nondestructively examined (NDE) according to the requirements of ASTM A-418, "Ultrasonic Inspection of Turbine and Generator Steel Rotor Forgings," and reported that "no noteworthy defect was found, positive results." The shaft was shipped to Weir Engineering Services, Alloa Works, located in Alloa, Scotland, where PEM procured the final shaft machining in accordance with Drawing C-67400-1. Weir Engineering Services issued a COC to PEM that certified that the shaft had been inspected and conformed to Drawing C-67400-1. PEM performed a dimensional verification of the shaft to Drawing C-67400-1 during receipt inspection to ensure the configuration characteristic of the rotor shaft.

The only NDE performed on the rotor shaft was an ultrasonic (UT), straight beam, examination, which may not detect shallow internal discontinuities (i.e., cracks or tears and bursts that occur during the processing of ingots or billets) immediately below the surface of the rotor shaft. Although PG&E identified integrity as a critical characteristic of the rotor shaft, PEM did not perform a magnetic particle (MT) examination, which would detect these discontinuities, even though certain conditions peculiar to forgings require the use of more than one NDE method to provide reasonable assurance of the integrity of the rotor shaft forging.

(6) Stator and Rotor Core (Revision 1)

In Revision 1 of its PO to P-EP, PG&E identified the stator and rotor core as a critical item (Attachment F, item 14) and specified their critical characteristic as factory testing (electrical losses). However, in Revision 3 of PG&E's PO the stator core and rotor pole were omitted as a critical item and stampings was identified (Attachment F, item 2) with the critical characteristics of configuration and material. The stator core and rotor pole stampings are addressed separately below.

- Stampings (Stator Core) (Revision 3)

  In PO 16271 to PEM, P-EP specified that stator core stampings (electrical steel) be provided in accordance with P-EP Material Specification MS-70.77, "Steel-Electrical Sheet - Fully Processed." The February 14, 1991, revision of MS-70.77 allowed core steel material for machines built by PEM to be purchased according to PEM Specification R 8046, "Electrical Core Steel For Rotating Machines, Coated On Both Sides With An Insulating Resin Or Varnish," and stated that "Grade 310-50-A5... is universally acceptable under MS-70.77."

  PEM procured the material from Joron Steel by PO EM31024 (original estimated date February 1990). PEM's PO specified "stator core steel to purchase standard R 8046, Grade 310-50-A5" and required test certificates for the chemical composition of steel and insulation resistivity.

  Joron procured the steel from EBG in Germany. EBG provided a test report indicating the steel core loss, but not the chemical composition or insulation resistivity. Joron subsequently provided the test report to PEM with some additions (coils numbers, contract number, and purchase order number).

  Although PEM specified testing for both chemical composition and insulation resistivity in its PO to Joron, it accepted the material without either of those tests being performed. This is another example of PEM accepting material from a supplier who has not met the PO requirements without generating a discrepancy report. In addition, although Revision 1 of the PG&E PO required factory testing for electrical losses, PEM did not pass this on to its supplier. Even though EBG provided the results of the factory test for electrical losses to PEM through Joron, there is no basis for accepting the EBG test report because PEM did not audit its suppliers.

- Stampings (Rotor Pole) (Revision 3)

  In its PO to PEM, P-EP specified that rotor pole stampings (pole iron) be provided in accordance with P-EP Material Specification MS-70.38, "Steel - Hot Rolled Pole Steel." The February 14, 1991, revision of MS-70.38 allowed rotor pole steel material for machines built by PEM to be Tensiloy 250.

  PEM issued PO EM31042 to British Steel Corporation requesting Tensiloy 250 steel. The PO required test certificates for chemical composition, mechanical properties (tensile, yield, percent-elongation), and dc permeability.

  Although Revision 1 of PG&E's PO identified "losses" (presumably referring to ac hysteresis) as a critical characteristic, PEM recognized that to be inappropriate for dc rotor pole stampings, even though it did not notify P-EP, because the critical characteristic of rotor pole stampings are mechanical and dc permeability. Thus, even though PEM did not pass on the "losses" requirement to its supplier, PEM did specify the correct critical characteristics. PEM's supplier, British Steel Corporation, did supply a certificate of magnetic testing (dc permeability) that identified the product as Tensiloy 250 and provided results of mechanical and dc permeability testing. Chemical composition of the steel was not provided. Again, PEM accepted the test certificate without an adequate basis since no audits of British Steel Corporation had been performed.

(7) Stator Resistance Temperature Detectors (Revision 1)

In Revision 1 of its PO to P-EP, PG&E identified the stator resistance temperature detectors (RTDs) as critical items, but P-EP did not invoke or provide a material specification for the RTDs. However, P-EP PO 16271 to PEM included, in the description of the generator, "6 embedded 10-ohm detectors," which indicated that P-EP supplied the RTDs to PEM for PG&E's generator. However, PEM issued PO JA30241 (original) (date not discernible on copies) to Carel Components Ltd. for "8 ea stator winding resistance temp detectors 10-ohms at 25°C, 3 wire 6-inch lg x 11/32-inch wide x 0.50-inch thk," which showed that PEM had procured the RTDs that were actually installed in the generator. Carel subsequently procured the RTDs from its subsupplier, Minco Products, Inc. Although the original PO from PEM did not specify the insulation material, PEM modified its PO in a telex to Carel, dated March 28, 1990, which Carel acknowledged by letter dated March 29, 1990. The modification specified the Minco part number in accordance with the catalog description. The Minco part number identified the model number (including element type, insulation class and thickness, and lead wire size), length, lead wire insulation, width, number of lead wires, and lead wire length. PEM did not require a COC from Carel in its original or revised (by telex) PO.

Revision 1 of the PG&E PO inadequately identified the critical characteristics of the stator RTDs as only size and shape; Revision 3 did not identify the stator RTDs as critical items at all. Although Revision 1 of the PG&E PO did require a shop test for RTD continuity, resistance (but no associated temperature), and insulation, PEM identified none of these characteristics to Carel in PO JA30241. The RTDs were shipped by Minco on May 4, 1990, and were received by PEM on May 15, 1990. According to the PEM record of a telephone conversation of September 14, 1990, to Carel, PEM requested a COC for the RTDs. Minco issued a COC (undated) to Carel, which was then provided to PEM certifying that the RTDs met the specifications as defined by the PO (i.e., part number). PEM performed its standard receipt inspection, verifying dimensions and shop testing for insulation resistance. In addition, PEM stated that its standard practice was to test RTDs during stator winding and also during testing of the completed generator. However, PEM test records did not indicate the expected values and tolerance for the RTD resistance with regard to temperature and the temperature at which the RTD resistance was measured was not recorded. Therefore, it was difficult to determine if the measured value was within the expected range.

PEM receipt inspectors did not always have all applicable documents available. PEM receipt inspectors were supposed to verify that incoming materials met the PO specifications by checking the delivered material against a copy of the PO. In this case, the PO was changed by telex to specify a part number and the receipt inspector was not provided a copy of the change notification. Therefore, the receipt inspector was not able to verify that the correct part number was received. Checking against the PO could have led to accepting incorrect material because Minco provides 2 different classes of RTDs that are identical except for the body material and the PO did not specify body material.

In addition, even though PEM had completed PG&E's generator before Revision 3 was issued and reported that Revision 3 was not considered during the design, procurement, and manufacturing activities of the generator, PEM acknowledged that certain items specified in Attachment F of Revision 3, although not listed in Attachment F of Revision 1, had been considered critical to the generator's ability to perform its intended design and safety-related function and, therefore, included in PEM's commercial grade dedication and verification activities. The team's review of a sample of these critical items is given below.

(1) Stator Coils (Revision 3)

Although Revision 1 of PG&E's PO inappropriately omitted the stator coils as critical items, Revision 3 did identify stator coils (Attachment F, item 15) as critical items with critical characteristics of configuration, chemical composition, and coating insulation. Nevertheless, in PO 16271 to PEM, P-EP invoked material specification MW-25.5 for the stator coil magnet wire. The MW-25.5 revision dated May 10, 1982, "Magnet Wire - Round, Square, or Rectangular Class H (180°C)," provided detailed specifications, including codes and standards to be met for the copper wire, enamel first insulation coating, and packaging. ANSI Standard C7.9 (for square or rectangular soft or annealed copper wire) and ASTM B-3 (for soft or annealed copper wire) were among the standards called for. In addition, MW-25.5 listed approved suppliers and the trade names of their products to meet the material specification. One approved magnet wire of the type available to PEM was listed in MW-25.5 as "Polythermaleze 2000," manufactured by Phelps Dodge.

PEM procured the stator magnet wire from its supplier, ISM, by PO EM31003. In its PO, PEM appropriately specified the material by trade name as well as by description (stator copper 0.256-inch-wide x 0.102-inch-thick insulated with polythermaleze 2000 enamel). The PO listed material specifications corresponding to those specified in MW-25.5 with the exception of ASTM B-3, which was not contained in any of the other specifications listed.

PEM (PO EM31003) required (1) a test certificate for chemical composition of copper, electrical resistivity, and insulation dielectric strength and (2) a COC attesting to conformance with the National Electrical Manufacturers Association, Standard Publication MW1000, "Thermal Classification and Insulation Voltage Withstand Level for the Type of Wire Specified." ISM subsequently supplied the material to PEM with a test certificate from ISM's subsupplier, SAFI-CONEL, and an ISM COC. However, PEM could produce no documentation that could connect the SAFI-CONEL test certificate to the PEM purchase order.

Although Revision 3 to PG&E's PO was issued less than 1 month before the generator was shipped, P-EP passed it on to PEM, and PEM tried to dedicate the stator coil wire in accordance with the new revision. However, PG&E inadequately listed the critical characteristics of the stator coils as configuration, chemical composition, without specifying particulars for the latter two. PEM's dedication methodology, apart from final testing, consisted of invoking P-EP's material specifications through PO requirements for its supplier, but the material and/or documentation received did not always meet these requirements.

PEM PO EM31003 to ISM required a test certificate indicating the chemical composition of the copper, electrical resistivity, and insulation dielectric strength. ISM supplied a COC attesting that the wire met the required specifications and also supplied a test certificate from SAFI-CONEL, but the test certificate addressed only the insulation dielectric strength. PEM apparently had not received any test certificates indicating the chemical composition of the copper or the insulation resistivity, and there was no documented basis for acceptance of the COC. PEM had not surveyed ISM or SAFI-CONEL and did not provide independent testing to verify the accuracy of the COC or the test report.

PEM maintained that it should not be held responsible for inadequate dedication of an item after the fact. The team determined that, although PEM accepted and used the stator coil wire without an adequate COC and test report, this did not constitute a deviation from the P-EP PO or the PG&E PO because Revision 1 to the PG&E PO did not specify the stator coil wire as a critical item and Revision 3 was issued well after the generator had been assembled.

However, of greater concern to the team were the issues of controlling and surveying suppliers, identifying nonconforming material, and holding suppliers accountable for nonconformances. At the time of the inspection, PEM was not in the practice of auditing or surveying its suppliers; therefore, its basis for accepting COCs from its suppliers was inadequate. In addition, PEM accepted and used material for which the COC certified that PO requirements had been met when, in fact, the requirements had not been met. In the stator coil procurement, the material supplier certified that PO specifications were met but did not furnish test certificates as required by the PO. PEM neither held the supplier (ISM) accountable nor documented this as a supplier noncompliance for future reference.

The team's tour of the material receiving area, review of documents, and interviews with PEM personnel generally supported PEM's claim that it inspected all incoming material for compliance with PO requirements. Nonconforming material was quarantined until the engineering staff determined disposition. If PEM's engineering staff determined the material to be unacceptable, it would be rejected (returned to the supplier) and a discrepancy report would be prepared. Discrepancy reports were to be reviewed on a routine basis to evaluate supplier performance. If, however, the material were to be evaluated by PEM's engineering staff as acceptable as is, no discrepancy report would be issued, even if the material (or the documentation) did not meet all the PO requirements. However, this practice, with regard to discrepancy reports, would not identify and track the performance of vendors who may occasionally, or even routinely, provide marginally acceptable materials or incomplete or inadequate documentation.

(2) <u>Bearing Bracket</u> (Revision 3)

In its PO to P-EP, PG&E identified the bearing bracket (Attachment F, item 4) as a critical item and specified its critical characteristics as configuration and material. PG&E's generator was a single bearing design. One end of the generator's rotor shaft was supported by a spherical roller bearing and bearing bracket assembly while the other end of the rotor shaft was flanged for mounting to the diesel engine.

PEM Drawing RA-14896, "Non-Drive End Roller Bearing Bracket Kit," Revision 0, dated February 16, 1990, was the design drawing for the bearing bracket assembly. The assembly consisted of (1) a spherical roller bearing, (2) the bearing bracket hub, (3) the bearing seal, (4) the bearing cover, and (5) the insulation ring.

The bearing bracket hub (part no. 30767-0274, Drawing B-66863-1) was a welded assembly of two concentric machined rings. The inside diameter (ID) of the inner ring of the bearing bracket hub abutted the outside diameter (OD) of the roller bearing and held the roller bearing in place, laterally, on the rotor shaft. This ring was machined with ports to lubricate (grease) the bearing. Welded to the OD of the inner ring was a mounting ring, with a smaller

L-shaped cross section attached to the inner ring by a continuous 3/8-inch fillet weld on both sides. The mounting ring was drilled to accommodate eight bolt holes, equally spaced circumferentially.

PEM procured this fabricated assembly from its supplier as a commercial grade item. PEM Material Specification MS-70.14 specified that the material for both rings comply with BS-4360, Grade 43A. However, the supplier did not provide PEM with a COC for the material or the fabrication. Although PEM's receipt inspection appeared to consist of a visual inspection for workmanship, the results of the inspection were not documented. In addition, PEM failed to specify any NDE of the continuous fillet welds that form critical load-bearing members of the support assembly of the bearing-end of the rotor shaft.

The insulation ring (Drawing A-64934-A) provided the electrical separation between the bearing bracket assembly and the generator frame. The ID of the 0.437-inch-thick (± 0.010-inch) insulation ring was fitted over a portion of the L-shaped mounting ring on the bearing bracket hub. The OD of the insulation ring appeared to be larger than the OD of the mounting ring and, therefore, the insulation ring stood proud of (extended beyond) the mounting ring. This configuration required the insulation ring to abut directly to the generator frame in such a way that it appeared to constitute a load-bearing component part of the support assembly for the bearing end of the rotor shaft. PEM's Material Specification MI-5.3, specified the material for the insulation ring as C.B. Bakelite. The insulation ring also was drilled to accommodate eight bolt holes, equally spaced circumferentially, that aligned with the bolt holes in the mounting ring. The eight bolts (5/8-inch hex-head) placed through the holes in the mounting ring and the insulation ring were attached to the generator frame and formed the supporting attachments for the bearing end of the generator.

PEM procured the fabricated (ID and OD cut to size and the bolt holes drilled) insulation ring from its supplier as a commercial grade item. However, the supplier did not provide PEM with a COC for the material or the fabrication. Although PEM's receipt inspection appeared to consist of a visual inspection for workmanship, the results of the inspection were not documented. Neither P-EP nor PEM demonstrated an engineering basis for the design of the insulation ring in combination with the mounting ring of the bearing bracket hub, which used the insulation ring as a load-bearing component part of the support assembly of the bearing end of the rotor shaft.

Therefore, PEM's inspection or verification of the commercial grade bearing bracket hub and insulation ring failed to demonstrate reasonable assurance that the parts were adequate and met expected quality and technical requirements.

Although not specifically a component part of the bearing bracket assembly, the brush-rigging was attached to the bearing bracket assembly by using a threaded stud. To form the electrical separation between the brush-rigging and the bearing bracket assembly (and, therefore, the rotor shaft), the stud was installed inside a mounting tube insulator. The material for the mounting tube insulator was specified in Drawing A-18405 as Grade X Spaudite Bakelite. PEM agreed that the tube insulator was a critical item, even though no critical characteristics were identified by either PG&E or P-EP and PEM did not perform any dedication activities to ensure that the tube insulator met expected quality and technical requirements.

(3) Stud/Threaded Rod (Revision 3)

In its PO to P-EP, PG&E identified the threaded rod studs (Attachment F, item 5) as critical items and specified their critical characteristics as dimensions, material, and welding. The generator's rotor spider assembly was formed by steel stampings that were laminated together and fitted concentric over the rotor shaft. The rotor spider assembly was designed with eight, equally spaced, dovetail grooves, which were used to mount the eight rotor pole assemblies. The rotor spider stampings were produced with penetrations to accommodate eight threaded rod studs. The studs were placed through the laminated stamping penetrations and extended the entire axial length of the rotor spider assembly. The exposed threaded ends of the studs were fitted with nuts, which were torqued to compress the rotor spider lamination and hold the assembled stampings together. When the proper compression of the rotor spider lamination was achieved, the nuts were tack welded to the studs to prevent loosening.

The threaded studs (Drawing A-66668-G 354, Revision 3, dated June 11, 1980) were 7/8-inch-diameter x 35-1/2 inches long, and 3-inches of each end were threaded with UNC-2A threads. The material for the studs was specified as ASTM A-108 and the minimum yield strength was required to be 72000 psi, even though the material actually used by PEM was BS-970, Grade 605 M36, Condition T. Although PEM's supplier furnished a COC that the stud material complies with BS-970, Grade 605 M36, Condition T, the COC did not provide the yield or tensile strength values for the material. PEM, in conjunction with P-EP, performed an equivalency evaluation of the material specified, compared the material actually used, and determined the material was acceptable, even though the technical basis to support that determination was not adequately documented.

Although the threaded studs were procured as commercial grade items from a supplier that had not been audited, PEM accepted the COC and performed a receipt inspection. The material for the nuts (7/8-inch x UNC-2A thread) was not specified and a COC for the commercial grade nuts was not included. PEM's receipt inspection appeared to consist of a visual inspection for workmanship; however, the results of the inspection were not documented. PEM also failed to document objective evidence of its inspection or verification of (1) the torque pressure applied to the nuts to compress the spider stamping assembly and (2) the tack welds that joined the nuts to the threaded studs.

(4) Spider End Rings (Revision 3)

In its PO to P-EP, PG&E identified the spider end rings (Attachment F, item 7) as critical items and specified their critical characteristic as configuration. The generator's spider end rings (one on each end of the rotor spider assembly) consisted of a head ring with eight mounting-lug ribs welded in an equally spaced configuration that extended radially from the axis of the head ring.

PEM Drawing B-66865, "#408 Pole Rotor Spider Head," Revision 4, dated February 6, 1970, prescribed the assembly of the head ring and the eight mounting-lug ribs. The ID of the head ring was concentrically fitted over the rotor shaft and abutted the spider stamping assembly. The OD of the head ring was smaller than the circumference formed by the eight threaded studs that held the spider stampings in a compressed assembly. Each head ring was produced with eight penetrations, equally spaced circumferentially to accommodate the eight rivets that extended through the spider stamping assembly and were welded to the head rings on each end. Eight mounting-lug ribs were attached to each head ring (1/4-inch fillet welds on each side of the mounting-lug ribs) in an equally spaced arrangement so that the ribs extended radially from the rotor's axis. The mounting-lug ribs were drilled and tapped to accommodate the bolted attachments of the rotor end ring and the generator's fan assembly.

PEM procured the spider end rings from its supplier as commercial grade fabricated assemblies. Although PEM's supplier provided a COC for the spider end rings, the COC failed to address NDE or visual inspection of the mounting-lug attachment welds, which form the critical load-bearing members of the support assembly for the generator's fan assembly. PEM's receiving inspection appeared to consist of a visual inspection for workmanship; however, the results of the inspection were not documented.

(5) Short Circuit Bars (Revision 3)

In its PO to PEM, P-EP specified that damper bars (short circuit bars or rotor bars) of hard oxygen free copper be provided in accordance with P-EP Material Specification MC-80.6, "Copper - Hard Drawn Oxygen Free or Deoxidized - Bar Rods and Shapes." However, the MC-80.6 revision of February 14, 1991, allows damper bars to meet BS-1433, Grade 103C.

Therefore, PEM issued PO JA30274 to Thomas Bolton & Johnson Ltd. for, "copper rods 1/2-inch dia. X 34-inch lg to conform to ASTM B-187 high conductivity round bar to BS-1433, 1970, hard drawn, designation C103." The PO required test certificates for chemical composition, tensile strength, percent elongation, and conductivity, hardness, and embrittlement tests.

Revision 3 of PG&E's PO identified the short circuit bars (damper bars) as critical items with critical characteristics of configuration and material. Bolton provided the material to PEM with a test certificate specifying all applicable requirements. Once again, PEM accepted the COC from Bolton without an adequate basis.

(6) Rivets (Revision 3)

In its PO to P-EP, PG&E identified the rivets (Attachment F, item 13) as critical items and specified their critical characteristic as configuration. The eight rivets were placed through the rotor spider assembly and extended its entire axial length. The ends of the rivets penetrated the head ring of the spider end ring assembly and were chamfered to facilitate performing a groove weld that joined the rivet to the head ring of the spider end ring assembly.

PEM Drawing RE-1734, dated November 15, 1990, prescribed the details for the 7/8-inch-diameter x 35-5/8 inches long rivets made from material complying with BS-970, PT1 (1983), Grade 605 M36, Condition T. PEM, in conjunction with P-EP, performed an equivalency evaluation of the material specified, compared the material actually used, and determined that the material used was acceptable, even though the technical basis to support that determination was not adequately documented. PEM's receiving inspection appeared to consist of a visual inspection for workmanship; however, the results of the inspection were not documented. PEM failed to specify any NDE examination of the groove welds that attach the rivets to the head ring of the spider end ring assemblies, which form load-bearing members of the support assembly for the generator's fan assembly.

(7) Stator Frame (Revision 3)

In its PO to P-EP, PG&E identified the stator frame (Attachment F, item 18) as a critical item and specified the critical characteristic as configuration. The stator frame formed the structural support for the stator and the completed generator assembly.

P-EP Drawing D-56825-1, Revision 3, dated November 17, 1970, described the construction details of the stator frame. Although P-EP's stator frame drawing was furnished to PEM, PEM's engineering staff found the drawing to be unacceptable for construction use. Specifically, PEM found that portions of the stator frame drawing were too difficult to read and properly interpret and noted that the drawing did not specify certain critical fabrication details, such as the length and pitch of the increments of intermittent fillet welds that join structural members.

P-EP's drawing, which was originally prepared by the Electric Products Division of Portec, Inc., specified the structural details of the stator frames in PG&E's five existing 1969 generators, which were qualified with respect to DCNPP's seismic requirements. PG&E required the new no. 2-3 generator to be identical to PG&E's 1986 spare generator and DCNPP's five 1969 generators in an apparent attempt to demonstrate compliance with the requirements for safety-related equipment suitability, including seismic and any environmental qualification requirements. However, PEM's new drawing consisted of some design changes from the original drawing in areas where the original was not clear or the details were not specified and, therefore, constituted changes to the original design.

PEM's new drawing for the frame was not reviewed and approved by P-EP and no evaluation was performed or documented to establish that the new drawing of the frame design was identical to the frame design of the previous frames supplied to PG&E. Fabrication of the stator frame to PEM's new drawing did not ensure that the stator frame was identical to the original seismically qualified 1969 stator frames.

PG&E selected and specified the critical items and their critical characteristics for its generator, and P-EP agreed to perform the tests necessary to verify the specified characteristics. However, P-EP supplied the generator to PG&E as a basic component that complied with the quality requirements of Appendix B to 10 CFR Part 50 and, therefore, was responsible for establishing reasonable assurance that the generator and its critical items will perform their respective design and safety-related functions. Therefore, P-EP failed to identify all of the design critical characteristics of the critical items (i.e., the properties or attributes that are essential for the item's form, fit, and functional performance) and P-EP did not demonstrate its bases for determining that the critical characteristics specified by PG&E were relevant to the critical item's (1) credible failure modes, (2) ability to perform its safety-related function, and (3) properties or attributes necessary to withstand the effects of long-term degradation and cyclic fatigue.

PEM failed to demonstrate that its dedication activities for critical items procured as commercial grade resulted in establishing reasonable assurance that the generator and its critical items will perform their respective design and safety-related functions. PEM procured the critical items for PG&E's generator as commercial grade from suppliers whose ability to adequately control changes in design, materials, and manufacturing processes had not been substantiated, as necessary to support the use of acceptable supplier/item performance history as an acceptable portion of its commercial grade dedication activity. Apart from final testing, PEM's dedication methodology consisted largely of imposing the material specification requirements on its suppliers and then verifying conformance of the material to those material specifications. Verification methods included basic receipt inspection and review of the suppliers' documentation, which was typically accepted without verification of its validity through audits or surveys of the supplier, as is common practice among European businesses. However, there were instances in which PEM accepted material through engineering resolution without the supplier having supplied all the documentation specified in the PO.

PEM failed to demonstrate reasonable assurance that the critical items and their critical characteristics chosen by PG&E were adequately verified during the commercial grade dedication and verification activities to ensure (1) that the critical items and the generator will perform their safety-related function and (2) that the critical items have the properties or attributes necessary to withstand the effects of long-term degradation or cyclic fatigue. This is Nonconformance 99901065/91-01-02.

TABLE

A Comparison of PG&E's Purchase Order Revision 1 and 3 for Critical Items and Their Critical Characteristics

ITEMS PROCURED BY PEM:

Critical Item	PO Revision	Attachment F	Critical Characteristics
Lead wire	1	Item 1	• Dielectric strength • Number of strands • Marking on cable • Insulation thickness
	3	Item 16	• Configuration
Magnet wire	1	Item 4	• Size and shape • Resistance • Insulation • Dielectric strength
	3	Item 3	• Material • Insulation • Dielectric strength
Copper bus (in terminal box)	1	Item 10	• Size • Resistance • Silver plating
Lead to coil terminations	1	Item 11	• Brazing • Weld materials
Roller bearing	1	Item 12	• Size/type • Visual inspection • Catalog number • Tolerances
	3	Item 6	• Part number • Configuration
Shaft/casting	1	Item 13	• PEM test
	3	Item 1	• Material • Configuration • Integrity

Critical Item	PO Revision	Attachment F	Critical Characteristics
Stator and Rotor core	1	Item 14	PEM test (losses)
Stampings	3	Item 2	• Configuration • Material
Stator coils	3	Item 15	• Configuration • Chemical composition • Coating insulation
Bearing bracket	3	Item 4	• Configuration • Material
Stud/threaded rod	3	Item 5	• Dimensions • Material • Welding
Spider end rings	3	Item 7	• Configuration
Pole end rings	3	Item 8	• Configuration • Material
Short circuit bars (damper bars)	3	Item 9	• Configuration • Material
Pole head	3	Item 10	• Configuration
Tapered keys	3	Item 11	• Configuration • Material • Hardness
Rotor wedge	3	Item 12	• Material
Rivets	3	Item 13	• Configuration

Critical Item	PO Revision	Attach-ment F	Critical Characteristics
Insulating washers	3	Item 14	• Configuration • Material • Dielectric strength
Stator frame	3	Item 18	• Configuration

ITEMS PROCURED BY P-EP:

Insulators (5-kV in terminal box)	1	Item 1	• Dielectric strength • Size and weight
	3	Item 22	• Dielectric strength • Configuration
Insulating bushings (lead wires through motor case)	1	Item 3	• Size and shape
	3	Item 24	• Configuration
Insulating material (sheets, tape, & rings)	1	Item 5	• Thickness
	3	Item 26	• Thickness
Bearing seals (felt)	1	Item 6	• Thickness and shape • Texture
	3	Item 23	• Configuration • Texture
Brushes and Brush Holders	1	Item 7	• Size and shape • Final generator test: resistance, material, and contact pressure

Critical Item	PO Revision	Attachment F	Critical Characteristics
Brushes	3	Item 20	• Configuration
Brush Holder	3	Item 19	• Configuration
Stator resistance temperature detectors (RTDs)	1	Item 8	• Shape and size • Shop test: continuity, resistance, and insulation
Current transformer and test switch	1	Item 9	• Size and weight • Dielectric strength • Continuity
Current transformer	3	Item 21	• Configuration • Mounting • Insulation • Resistance • Continuity
Current transformer test switch	3	Item 25	• Configuration • Dielectric strength • Continuity
Slip-rings	3	Item 17	• Configuration • Material
Adhesives	3	Item 27	• Material

APPENDIX

PERSONS CONTACTED

The U.S. Nuclear Regulatory Commission staff participating in the evaluation of Peebles Electrical Machines's design interface activities, procurement, commercial grade dedication, and manufacture of an emergency ac power generator for Pacific Gas and Electric Company's Diablo Canyon Nuclear Power Plant Unit 2 and the persons contacted during the inspection are listed below.

<u>NEI Peebles Limited, Peebles Electrical Machines</u>:

•	*	Brunton, David	Insulation and Development Engineer
	*	Francis, Les	Drawing Office Manager
•	*	Holroyd, Peter R.	Manager
•	*	Mac Naughton, Harry	Calibration Engineer
•	*	Miller, John	Quality Assurance Engineer
•	*	Nicoll, Harold W.	Quality Manager
•	*	Smith, Robert B.	Engineering Manager
•	*	Taylor, James	Chief Inspector
		Tweedale, Les	Chief Mechanical Designer

<u>U.S. Nuclear Regulator Commission</u>:

•	*	Alexander, Stephen D.	Environmental Qualification and Test Engineer, Reactive Inspection Section 2 (RIS2) Vendor Inspection Branch (VIB), Division of Reactor Inspection and Safeguards (DRIS), Office of Nuclear Reactor Regulation (NRR)
•	*	Cwalina, Gregory C.	Section Chief, Special Projects Section, VIB/DRIS/NRR
•	*	Matthews, Steven M.	Team Leader, RIS1, VIB/DRIS/NRR

• Attended the entrance meeting.
* Attended the exit meeting.

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

FEB 2 1 1990

Docket No. 99901155/89-01

Mr. Norman Kaplon, Vice President
Newman Incorporated
3850 South East International Way
Milwaukie, Oregon 97222

Dear Mr. Kaplon:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Milwaukie, Oregon, conducted by Mr. Randy Moist of this office on April 11, 1989, and the discussions of his findings with members of your staff at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers as discussed in NRC Information Notice (IN) 88-48 and supplements. Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's ongoing review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901155/89-01

ORGANIZATION: NEWMANS INCORPORATION
MILWAUKIE, OREGON

REPORT NO.: 99901155/89-01	INSPECTION DATE: April 11, 1989	INSPECTION ON-SITE HOURS: 3

CORRESPONDENCE ADDRESS: Newmans Incorporated
3850 SE International Way
Milwaukie, Oregon 97222

ORGANIZATIONAL CONTACT: Norman Kaplon, Vice President
TELEPHONE NUMBER: (503) 653-0210

NUCLEAR INDUSTRY ACTIVITY: Newmans Incorporated (NI) supplies valves to the commercial industry.

ASSIGNED INSPECTOR: CE Ader for June 2, 1989
R. Moist, Reactive Inspection Section No. 2 (RIS-2) Date

OTHER INSPECTOR(S):

APPROVED BY: CE Ader for June 2, 1989
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50.

B. SCOPE: The purpose of this unannounced inspection was to determine if NI has purchased any valves from CMA International, Incorporated of Vancouver, Washington and determine if those valves, if any, were supplied to any commercial nuclear part.

PLANT SITE APPLICABILITY: None identified.

ORGANIZATION: NEWMANS INCORPORATION
MILWAUKIE, OREGON

REPORT NO.: 99901155/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discussed a potential problem concerning Vogt 2-inch valves (Vogt Figure No. SW13111), which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington [CMA International, Inc. (CMA)]. Henry Vogt representatives examined the valves at the Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) The Vogt name was die-stamped instead of being forged onto the side of the body, (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flanges, (3) the subject valves have swing gland bolting which is not used by the Henry Vogt Company and (4) the end-to-end dimensions of the subject valves are shorter than the Vogt SW-13111.

2. Discussions At Newmans Incorporated (NI)

The inspector conducted discussions with Mr. Norman Kaplon, Vice President of NI relating to business activities with CMA. Mr. Kaplon stated the only business conducted with CMA in recent years was sending NI valves to CMA for refurbishment.

ORGANIZATION: NEWMANS INCORPORATION
MILWAUKIE, OREGON

REPORT NO.: 99901155/89-01	INSPECTION RESULTS:	PAGE 3 of 3

The valves were then sold to other distributors and subsequently shipped to commercial end users by NI. The inspector reviewed several purchase orders to CMA from NI and determined that all the NI valves sent to CMA were refurbished and sold to other distributors and shipped to commercial end users by NI. Mr. Kaplon stated that NI does not sell valves directly to nuclear power plants. NI represents commercial grade NEWCO valves which are supplied to commercial end users. Mr. Kaplon was very cooperative and helpful with the NRC inspector during this inspection.

3. PERSONNEL CONTACTED

Mr. Norman Kaplon	Vice President and Manager, NI Oregon Division

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

FEB 21 1992

Docket No. 99901156/89-01

Mr. Kenneth W. Grothe, President
Paramount Supply Company
816 South East Ash Street
Portland, Oregon 97214

Dear Mr. Grothe:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, conducted by Mr. Joseph J. Petrosino, of this office on April 10, 1989, and the discussions of his findings with members of your staff at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers as discussed in NRC Information Notice (IN) 88-48 and Supplements 1 and 2. Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's ongoing review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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.

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901156/89-01

ORGANIZATION: PARAMOUNT SUPPLY COMPANY
PORTLAND, OREGON

REPORT NO.: 99901156/89-01	INSPECTION DATE: April 10, 1989	INSPECTION ON-SITE HOURS: 2

CORRESPONDENCE ADDRESS: Paramount Supply Company
816 S. E. Ash Street
Portland, Oregon 97214

ORGANIZATIONAL CONTACT: Mr. Kenneth W. Grothe, President
TELEPHONE NUMBER: (503) 232-4137

NUCLEAR INDUSTRY ACTIVITY: Currently Paramount Supply Company supplies only commercial grade products to the nuclear industry in the Oregon-Washington area.

ASSIGNED INSPECTOR: J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) — 2 June 1989 — Date

OTHER INSPECTOR(S):

APPROVED BY: CE Abbott for — E. T. Baker, Chief, RIS-1, Vendor Inspection Branch — 2 June 1989 — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50.

B. SCOPE: The purpose of this unannounced inspection was to determine if Paramount Supply Company has purchased any valves from CMA International, Incorporated of Vancouver, Washington to determine if those valves, if any were supplied to any commercial nuclear plant.

PLANT SITE APPLICABILITY: None identified during inspection.

ORGANIZATION: PARAMOUNT SUPPLY COMPANY
PORTLAND, OREGON

REPORT NO.: 99901156/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discussed a potential problem concerning Vogt 2-inch valves (Vogt Figure No. SW-13111), which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington [CMA International, Inc. (CMA)]. Henry Vogt representatives examined the leaking valves at the Diablo Canyon nuclear power plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) The Vogt name was die-stamped instead of being forged onto the side of the body; (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flanges; (3) the valves in question have swing gland bolting which is not used by the Henry Vogt Company; and (4) the end-to-end dimensions of the subject valves are shorter than the Vogt SW-13111.

2. Inspection Activities

Discussions were conducted with Mr. Ken Grothe, President of Paramount Supply. Mr. Grothe stated that Paramount has conducted very little business with CMA and all of the CMA products that Paramount procures go to the marine industry.

REPORT NO.: 99901156/89-01	INSPECTION RESULTS:	PAGE 3 of 3

He also stated that he is sensitive to nuclear plant business and the requirements that are usually imposed. He also stated that his Kennewick, Washington office has supplied nuclear safety-related valves in the past but is not currently supplying safety-related components. No records were reviewed at the Paramount supply facility based on Mr. Grothe's statements that no CMA supplied equipment has ever been provided to nuclear power plants.

F. PERSONNEL CONTACTED

K. Grothe, Jr., President, Paramount Supply

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

JAN 09 1992

Docket No. 99901213

Mr. R. H. Slavin, President
Potter & Brumfield, Incorporated
200 S. Richland Creek Drive
Princeton, Indiana 47671-0001

Dear Mr. Slavin:

SUBJECT: NOTICE OF VIOLATION
(NRC INSPECTION REPORT NO. 99901213/91-01)

This letter addresses the inspection of your facility in Princeton, Indiana, conducted by Messrs K. R. Naidu and R. A. Spence of this office on November 12-14, 1991, and the discussions of their findings with Mr. L. O. Hume and other members of your staff at the conclusion of the inspection.

The purpose of this inspection was to review a matter discussed in your letter of September 6, 1991, and the circumstances surrounding the failures involving MDR rotary relays manufactured by Potter & Brumfield (P&B) experienced at various nuclear power stations. The enclosed inspection report describes the areas examined during the NRC inspection and our findings. The inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by members of the inspection team. The inspection team observed that P&B failed to develop an adequate procedure to implement the requirements of Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) during the period 1978 to 1988. You are required to respond to this letter and should follow the instructions specified in the enclosed Notice of Violation when preparing your response.

Your staff informed our inspection team that, based on the recommendations of your legal staff, you discontinued to accept the reporting requirements of 10 CFR Part 21. Even though you discontinued to accept the requirements of 10 CFR Part 21 from 1988, you still retain reporting responsibility for the relays and components previously supplied in accordance with that regulation.

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

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

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
Notice of Violation
Inspection Report No. 99901213/91-01

NOTICE OF VIOLATION

Potter & Brumfield, Incorporated
Princeton, Indiana

Docket No. 99901213
Report No. 91-01

During an inspection conducted at the Potter & Brumfield, Incorporated (P&B) facilities in Princeton, Indiana, on November 12-14, 1991, a violation of the U.S. Nuclear Regulatory Commission (NRC) requirements was identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1991), the violation is listed as follows:

Section 21.21, "Notification of failure to comply or existence of a defect," of 10 CFR Part 21 requires, in part, that "Each individual, corporation, partnership or other entity subject to the regulations in this part shall adopt appropriate procedures to (1) Provide for: (i) Evaluating deviations or (ii) informing the licensee or purchaser of the deviations in order that the licensee or purchaser may cause the deviation to be evaluated..."

Contrary to the above, Quality Control Procedure QC 16.01, issued October 2, 1978, "Deviation Monitoring Procedure For MDR Relays," was inadequate in that the procedure did not provide for (1) evaluating deviations or (2) informing the licensee or purchaser of the deviations in order that the licensee or purchaser may cause the deviation to be evaluated.

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

Pursuant to the provisions of 10 CFR 2.201, Potter & Brumfield is hereby required to submit a written statement of explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that have been 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 deadline.

Dated at Rockville, Maryland
this 9th day of January 1992

ORGANIZATION:	Potter & Brumfield, Incorporated Princeton, Indiana 47671-0001
REPORT NO.:	99901213/91-01
CORRESPONDENCE ADDRESS:	Mr. R. H. Slavin, President Potter & Brumfield, Incorporated 200 S. Richland Creek Drive Princeton, Indiana 47671-0001
ORGANIZATIONAL CONTACT:	K. McGrew, Manager Quality Assurance Planning
INSPECTION CONDUCTED:	November 12-14, 1991
SIGNED:	K Naidu Kamalakar R. Naidu, Special Projects Section Vendor Inspection Branch 1/6/92 Date
OTHER INSPECTORS:	Robert A. Spence, Office for Analysis and Evaluation of Operational Data (AEOD)
APPROVED:	Gregory C. Cwalina Gregory C. Cwalina, Chief Special Projects Section Vendor Inspection Branch 1/7/92 Date
INSPECTION BASES:	10 CFR Part 21, and 10 CFR Part 50, Appendix B
INSPECTION SCOPE:	To review Potter & Brumfield's (P&B) compliance with 10 CFR Part 21, and the circumstances surrounding the failures involving P&B's MDR relays at various nuclear power plants.
PLANT SITE APPLICABILITY:	Numerous

1.0 INSPECTION SUMMARY

1.1 Violation

Contrary to Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21), the Quality Control Procedure "Deviation Monitoring Procedure For MDR Relays," established by Potter & Brumfield, Incorporated (P&B) on October 2, 1978, was inadequate in that the procedure did not provide for (1) evaluating deviations or (2) informing the licensee or purchaser of the deviation in order that the licensee or purchaser may cause the deviation to be evaluated. (91-01-01)

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS:

This was the first NRC inspection of P&B.

3.0 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on November 12, 1991, the NRC inspectors discussed the scope of the inspection, outlined areas of concern, and established contacts with P&B management and staff. In the exit meeting on November 14, 1991, the inspectors discussed their findings with P&B's management and staff.

3.2 Inspection Scope

P&B manufactures and supplies various electrically-operated relays either directly or through distribution companies to nuclear power plants and to commercial customers.

The inspectors reviewed P&B's evaluations of MDR relay failures experienced at various nuclear power plants since 1984 and the actions taken by P&B to correct the problems.

The inspectors examined the circumstances which led P&B to inform the U.S. Nuclear Regulatory Commission (NRC) in a letter of September 6, 1991, that P&B failed to notify the Nuclear Energy Division of the General Electric Company (GENE) in 1988 that P&B only supplies commercial grade relays.

The inspectors examined P&B's design enhancements and changes in manufacturing practices, purchase orders issued by various nuclear customers to P&B, and audits performed by nuclear customers of P&B's activities.

3.3 P&B's 10 CFR Part 21 Notification to the NRC

On September 6, 1991, P&B informed the NRC that, during a review of specifications for products purchased by GENE, P&B found it had inadvertently failed to inform GENE that it had discontinued compliance with the reporting requirements of 10 CFR Part 21. P&B engineers stated that after receiving reports indicating that the Palo Verde Nuclear Generating Station (PVNGS) experienced MDR relay problems in 1988, its attorneys advised that P&B should stop complying with the requirements of 10 CFR Part 21 and supply only commercial grade relays. Accordingly, starting in 1989, P&B, in individual quotations and in purchase order acknowledgements, took exception to compliance with the requirements of 10 CFR Part 21. However, GENE used a different method to purchase MDR relays from P&B. GENE's purchase orders to P&B referenced the relevant drawing number of the MDR relay which contained all the technical requirements and included a statement that the relay was a Class 1E component. The P&B general sales manager stated that he did not inform GENE that P&B had ceased to comply with 10 CFR Part 21 since the GENE purchase order did not mention that compliance with 10 CFR Part 21 was required. He also stated that P&B did not inform all of its customers that it no longer complied with 10 CFR Part 21. The inspectors reviewed the following correspondence between selected nuclear power stations and P&B and determined that P&B had adequately informed them that the requirements of 10 CFR Part 21 would no longer be accepted:

a. The Southern California Edison Company's quotation request of March 13, 1989, for various quality Class I and II, safety-related, MDR relays for its San Onofre Nuclear Generating Station (SONGS) enclosed terms and conditions which included compliance with 10 CFR Part 21. P&B's quotation of March 27, 1989, in response to this request took exception to 10 CFR Part 21. P&B took similar exceptions to compliance with 10 CFR Part 21 in subsequent requests for quotations for MDR relays intended for SONGS.

b. An acknowledgement from Pennsylvania Power & Light Company regarding purchase order 9-13733-1 of February 27, 1990, indicates that P&B supplied commercial grade MDR relays.

c. P&B quotation 1189125 of November 28, 1989, to Electro-Mechanics, Incorporated, of New Britain, Connecticut, for several types of MDR relays states, "10 CFR Part 21 does not apply. Products being manufactured for Electro-Mechanics are commercially available to anyone in any industry who may wish to purchase them, subject to applicable law." The P&B general sales manager informed the inspectors that Electro-Mechanics purchases equipment for Combustion Engineering, Incorporated, one of the manufacturers of pressurized water reactors.

3.4 Review of 10 CFR Part 21 Notification by the Carolina Power and Light Company (CPL)

On May 24, 1990, CPL reported in licensee event report (LER) 90-15 a 10 CFR Part 21 condition related to a deficiency in the design of an emergency load sequencer (ELS) at the Shearon Harris Nuclear Power Plant, Unit 1 (Shearon Harris). During certain emergency scenarios, P&B MDR relays would be subjected to excessive inductive loads. LER 90-15 references LER 89-16 in which a similar relay malfunction is discussed. LER 89-16 stated that, on September 11, 1989, CPL operators were performing a periodic test of emergency safeguards sequencer system 1B-SB ELS at Shearon Harris. The CPL operators pushed the "Test Stop" button and observed that the 1B-SB ELS did not properly reset and caused the inadvertent start of the emergency service water pump (ESWP) because an MDR 137-8 type P&B relay malfunctioned. The control room operators observed the inadvertent start and secured the ESWP. CPL investigated the failure and determined that, in the test circuit, an MDR 137-8 type P&B relay failed to reset at the proper time and continued to supply power to the equipment actuation relays longer than designed causing the ESWP to start inadvertently. On September 12, 1989, when CPL personnel removed and inspected the failed P&B MDR relay, they observed that the relay contacts were burned and the leaf spring contact had melted into the plastic armature. CPL documented identical failures (the same type of relay in the same location in the sequencer) in LERs 88-29 and 88-08.

CPL conducted qualification tests on the relay contacts and determined that the relays operated acceptably when two contacts of the P&B MDR relay are connected in series because this arrangement reduces the inductive load on each individual contact. CPL reconnected the MDR relay contacts in the ELS system in series and has not reported any similar failures since that time.

The inspectors discussed this problem with P&B engineers to ascertain if P&B was aware of the problem. A P&B engineering representative stated that they were not aware of the problems experienced at Shearon Harris. P&B recommended that the users should conduct tests to determine the inductive loads that the contacts of the MDR relay are expected to switch and design the circuit accordingly. A P&B engineer provided the following comments partially from The Engineers' Relay Handbook published by The National Association of Relay Manufacturers (NARM):

> D.C. [direct current] loads are more difficult to turn off than A.C. [alternating current] loads because the DC voltage never passes through zero. As the contacts open, an arc is struck and may be sustained by the applied voltage until the distance between opening

contacts becomes too great for the arc to sustain itself. The arc energy can seriously erode away the contacts. Frequently, arc extinguishing capabilities for D.C. inductive loads can be enhanced by connecting two contacts in series. This provides a larger total contact gap and a faster rate of contact separation, thereby providing improved performance.

Paralleling sets of relay contacts to switch loads greater than a single set can handle is often unsuccessful. Lack of absolute simultaneity of contact opening results in one contact taking all the load causing early failure.

The inspectors concur with CPL's conclusion documented in LER 90-15 that a deficiency in the design of the ELS caused the P&B MDR 137-8 relays to fail.

3.5 Review of P&B Procedure For 10 CFR Part 21

The inspectors reviewed P&B Quality Control Procedure QC 16.01, "Deviation Monitoring Procedure for MDR Relays," issued on October 2, 1978, to implement the requirements of 10 CFR Part 21. The stated purpose of the QC procedure was, "To detail the deviation monitoring procedure for MDR relays which require reporting of defects and non-compliance in accordance with Title 10 of Code of Federal Regulations Part 21 (10 CFR 21)." The inspectors informed the P&B quality assurance representative that this procedure did not contain provisions for (1) evaluating the deviations or (2) informing the licensee or purchaser of the deviation in order that the licensee or purchaser may cause the deviation to be evaluated as required in paragraph 21.21(a)(1) of 10 CFR Part 21.

The inspectors identified violation 99901213/91-01-01 in this area.

3.6 MDR Relay Failures at the LaSalle County Station

On January 14, 1986, September 17, 1987, and December 8, 1987, an emergency diesel generator (EDG) failed operability surveillance tests at the LaSalle County Station, Units 1 and 2 (LaSalle). In each case, when the Commonwealth Edison Company (CECO), the licensee, attempted to synchronize the EDG to its bus, the EDG output breaker would not close. CECO determined that the three events at LaSalle resulted from the failure of a P&B MDR-137-8 or MDR-138-8, 125 Vdc normally energized relay contacts to close. CECO performed diagnostic testing after the earlier events but could not repeat the failure. This lack of repeatability appears to be typical of MDR intermittent failures. CECO replaced all P&B MDR relays in the output breaker closing circuits with General Electric HFA type relays. The NRC staff is not aware of any other MDR relay failures since at LaSalle. The inspectors discussed this subject with a representative from GENE. He stated that the MDR relay contacts were also used in applications such as equipment status display lights, computer inputs, and annunciators. The inspectors discussed the use of P&B MDR relays in such applications with P&B engineers during this inspection. P&B furnished the following comments to the NRC and stated that they were based on guidance in The Engineer's Relay Handbook on using MDR relay contacts to switch low level loads:

> A relay contact rating does not necessarily apply for all loads from zero up to the magnitude specified. The fact that a relay contact can reliably switch 10 amperes does not necessarily mean it can reliably switch 10 milliamperes. The MDR contact structure is designed for 10 amp 115 V AC at 50% PF [power factor], 3 amp 28 V DC resistive and 0.8 amp 125 V DC resistive load switching. It does not have the contact structure design configuration necessary for low level switching applications that inhibit contact resistance build up.

Even though P&B engineers were not aware of the exact applications, they recommended that users who experience intermittent contact failures should review the design, and determine if the problem is due to low level switching.

3.7 MDR Relay Failures at the Palo Verde Nuclear Generating Station Units 1, 2, and 3

On October 10, 1988, Arizona Public Service Company (APSC), the licensee for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, (PVNGS) submitted a report in accordance with 10 CFR Part 21. This report documented 18 instances (over a 2-year period) in which P&B MDR relays failed to change position. APSC found that three of the P&B MDR relay rotors at PVNGS would not move more than 12 degrees of the complete 30-degree arc. The failed relays, located in cabinets without forced ventilation, were in an ambient temperature of 95 to 104^{0}F (the design limit is 149^{0}F) and had an external surface temperature of 157^{0}F. APSC detected no relay failures in cabinets with forced ventilation which provided an ambient temperature of 81^{0}F or less. Such relays had a temperature of 112^{0}F on their external surfaces. APSC determined that it had inadvertently applied up to 39.8 Vdc to the 28 Vdc MDR relay coils. APSC tested 7 of the 18 failed relays at an 18 month frequency and 10 at a 62 day frequency. APSC had the relay failures analyzed and determined that varnish on the relay coils outgassed, condensed, and deposited material between the rotor and the end-bell bearings, binding the rotor and the bearings together. The outgassing was due to excessive coil temperatures that occurred when the coils were continuously energized at voltages above their nominal ratings. The heat may also have caused the release of chlorine from (1) the plastic coating on the fiberglass tubing covering the solder joint between the magnet wire and the Teflon coated lead wire, (2) the neoprene rubber grommet through which the coil lead wires penetrate the base of the relay. The chlorine may have corroded brass parts inside the relay. P&B and APSC concluded that long intervals between deenergizing of the relays may have also contributed to the corrosion problem.

In May 1989, APSC installed replacement P&B relays at PVNGS that were manufactured with coils coated with epoxy instead of varnish. APSC conducted tests on the replacement relays and determined that 5 of the 42 relays tested would not rotate to their de-energized position and that 5 other relays operated slowly. At APSC's request, two independent laboratories investigated the failures and observed that (1) P&B's epoxy had not been properly cured, (2) other uncured epoxy contaminated the rotor, and (3) P&B did not deaerate the epoxy prior to use, contrary to the manufacturer's recommendations. This caused the rotor and stator surfaces to bond together, preventing the rotor from rotating freely. P&B engineers stated that they replaced all the replacement MDR relays supplied to PVNGS and also improved the manufacturing techniques to apply epoxy to the relay coils.

3.8 MDR Relay Failures at The Waterford Steam Electric Station

On June 15, 1990, Entergy Operations, Incorporated (EOI), the licensee for the Waterford Steam Electric Station (WSES), informed the NRC Senior Resident Inspector that P&B models MDR 66-4, MDR-4076, and MDR-5061 rotary latching relays had reached a high failure rate (see NRC Region IV Inspection Report 50-382/90-15). EOI performed a root cause analysis and determined that the design of the electrical system used the MDR latching relay contacts to deenergize its closing and reset coils after use. However, minor variations in the timing of these contacts prevented the relays from repositioning and from resetting the contacts for the next operation of the relays. Consequently, the relays failed in the intermediate position.

The inspectors discussed this anomaly with P&B engineers and determined that P&B was unaware that MDR relays were used in this application. The P&B engineers stated that it is possible to incorporate a special feature in the MDR latching relay to accomplish this. P&B engineers informed the NRC that, with prior knowledge of the application of the relay, P&B could have designed and manufactured an MDR rotary latching relay to reliably accomplish the intended function. For such special applications, P&B stated that it would assign a unique drawing number for the design and manufacture of the MDR rotary latching relay and would expect the recipient to reference that P&B drawing number in subsequent purchase orders to ensure that P&B furnished an identical replacement relay.

3.9 MDR Relay Failures at The River Bend Station

On July 19, 1991, according to LER 50-458/91-14, engineered safety feature (ESF) actuations occurred at the River Bend Station because a P&B MDR relay malfunctioned. The Gulf States Utilities Company (GSU), the licensee, determined that a high resistance on one set of contacts on a P&B 24 Vdc, MDR-5111-1 rotary relay, which should have been closed, caused a voltage drop to the downstream relays which opened their contacts and resulted in the ESF actuation. GSU later performed bench testing of the failed relay and verified that the relay actuated properly and that all contacts changed state properly and exhibited proper continuity. The coil was meggered and found to be acceptable. The contacts all appeared to be clean and shiny, with no evidence of pitting or residue. GSU did not observe foreign material in the relay or on the rotor shaft and found nothing that may have contributed to the high resistance across the contacts.

On July 23, 1991, according to Revision 1 to LER 50-458/91-14, GSU investigated another MDR relay failure at River Bend and found two MDR-5111-1 relay contacts open that should have been closed when the coil was in its energized position. GSU also found that the contacts operated intermittently with some contacts closing several minutes after the coil was energized or sometimes not closing at all.

Both failed relays at River Bend had been in service within tightly-regulated design voltage and temperature conditions and were mounted inside stainless steel isolation cans for divisional separation. GSU measured the temperature inside the isolation can at 113^{0}F, while the ambient cabinet temperature was 92^{0}F. In each case, the failed relay had been recently cycled because of a short term loss of power to the coil that had occurred a few days before the relay failure was discovered. It appears that not all contacts engaged properly when power was restored.

The NRC inspectors discussed these problems with P&B engineers, witnessed tests conducted on relays returned from River Bend (discussed in paragraph 3.12), and concluded that the River Bend failures demonstrated that even MDR relays that operated within design specifications are susceptible to unpredictable failure mechanisms if they had been manufactured before approximately 1990. As discussed in paragraph 3.11, P&B has made significant design changes to the MDR relays.

3.10 Discussion of The Failure Mechanisms of MDR Relays

The primary failure mechanism of the P&B model MDR rotary relay is a mechanical binding of the rotor caused by an outgassing and deposition of contaminants and corrosion particles on the relay rotor shaft. The contaminants accumulate in the end-bell bearings and sleeves and cause the rotor shaft to bond or stick to the bearing, preventing the rotor shaft from fully rotating when the relay coils are energized or deenergized. The principal contaminant is outgassed material emitted from the brown enamel varnish (used before 1986) to coat the relay coils. This contamination may not be apparent to the naked eye. Gulf States and P&B disassembled six operable and two failed relays that had been in service since December 1984. The thickness and color of the deposits on the rotor, sleeve, and end-bell bearings of the relay varied widely among the eight relays, indicating varnish outgassing. Corrosion of the contacts may result from chlorine released from the rubber grommets and the polyvinyl chloride sleeves.

A secondary failure mechanism is intermittent continuity of the electrical contacts. High resistance and intermittent continuity may result from chemical reactions on the fixed and movable silver contacts. P&B tested an MDR-5112-1, 125 Vdc relay that had been in service at River Bend and found intermittent continuity on a set of clean, unused contacts.

A number of variables contribute to these failure mechanisms and reduce the length of the operating life of the complex P&B MDR rotary relays. These variables include coil wattage, applied ac or dc voltage, normally energized or de-energized coil, manufacturing tolerances, ambient and coil temperatures, varnish thickness, mounting configurations and enclosures, cabinet ventilation, relay breathing, testing frequency, operational cycling, the number of contact decks, and the amperage and voltage of the contact load. These contributory factors cause an apparent random failure history. While each of these MDR relays failed between 1 month to 13 years after it was placed in service, most failed within 2 to 5 years.

3.11 Enhancements to the MDR Relays

P&B engineers informed the NRC inspectors that P&B made the following design changes to the MDR models during the past several years in an effort to produce a reliable product:

- Changed the movable contacts from silver to silver-cadmium-oxide in October 1985.

- Changed the coil coating from varnish to epoxy resin in February 1986 to reduce the coil outgassing rate. Dolphon, the manufacturer of Dolphon CC-1090 type epoxy, recommends deaeration of the epoxy before use. However, P&B informed the NRC that P&B does not deaerate the epoxy prior to use and does not intend to do so in the future. Furthermore, P&B informed the NRC that the epoxy manufacturer plans to cease production of the currently used and tested epoxy. The NRC does not know when P&B will change to a new epoxy. Based on the problems caused by outgassing, and by improper curing and application of the epoxy, P&B should assure that current and future epoxies are properly applied and cured in accordance with the manufacturer's recommendations.

- Replaced the brass switch studs in MDR relays with stainless steel studs in November 1986.

- Began lubricating end-bell bearings in July 1988.

- Changed materials containing chloride to chloride-free materials in June 1989.

3.12 Observation of Disassembly of MDR Relays Returned From River Bend

The NRC inspectors witnessed P&B perform tests on the following MDR relays that had been in service at River Bend:

- MDR-5111-1, 24 Vdc relay, Serial No. 8121S-18, which had failed at River Bend. At GSU's request, P&B performed an analysis of the failed relay. P&B reassembled and tested the relay and determined it acceptable. P&B measured the number of turns and resistances of the two coils separately and determined them to be within P&B specification limits.

- MDR-5111-1, 24 Vdc relay, Serial No. 8121S-19, which had failed at River Bend was returned to P&B and at GSU's request had been previously analyzed by P&B. During the test, the inspectors observed that there was no continuity between its H and J contacts on deck 1. However, the measured values of the number of coil turns and the resistance of the coil were within P&B specification limits.

- MDR-5111-1, 24 Vdc relay, Serial No. 8121S-26, which had operated properly for about 6 years at River Bend. The inspectors observed that the measured values of the number of turns and resistance of the coil were within P&B specification limits. The relay successfully withstood high potential and functional tests.

- MDR-5112-1, 125 Vdc relay, Serial No. 81165-43, which had operated properly for about 6 years at River Bend. P&B personnel's efforts to use the MDR automated MDR relay tester to test this relay were unsuccessful because there was no continuity between a set of its unused contacts, H and J on deck 1. A P&B engineer manually tested this relay and found intermittent continuity across this set of contacts. When the relay was disassembled, the inspectors observed varnish and corrosion products on the rotor shaft, sleeve and end-bells. The NRC inspectors observed the contact surfaces to be clean and shiny without any foreign particles. The NRC inspectors examined the contacts through a microscope and observed no corrosion products. P&B did not perform chemical tests on the contact surface. The measured values of the number of coil turns and resistances were within specification limits. P&B could not determine the reason that the relay operated intermittently.

3.13 Other Notifications Of Potential P&B Relay Problems

On November 22, 1988, the Institute of Nuclear Power Operations (INPO) issued Significant Event Report (SER) 33-88 "Failure of Relays Operated At Greater Than Rated Voltage," to inform its members of MDR failures experienced at the PVNGS on July 29, 1988, and the failures of P&B, KHS-type relays at the Prairie Island Nuclear Generating Station on December 11, 1987 and March 3, 1988. The inspectors discussed this problem with P&B engineers as detailed in paragraph 3.7.

On September 10, 1990, GENE issued Rapid Information Communication Services Information Letter 053 to address P&B MDR relay failures reported at two GE boiling water reactors. P&B believed that chlorine released from rubber grommets and polyvinyl chloride sleeves caused corrosion and that varnish on the coils outgassed while the relay was continuously energized. Both chlorine-corrosion products and varnish accumulated in the bottom end-bell bearing and caused the rotor shaft to bond to the bearing. P&B suspected that the failed relays were exposed to high ambient temperatures and could have been exposed to high coil voltages or could have been rarely cycled.

P&B informed the inspectors that the relays may have failed because of exposure to high ambient temperatures and high coil voltages or exceptionally infrequent deenergization activity or because of misapplication such as switching low level loads. However, P&B stated that, without knowing the actual applications and the intended design, it could not explain the cause of failure.

On November 2, 1990, GENE issued Potentially Reportable Condition 90-11 in which it stated that both 24 Vdc and 120 Vdc coils had lower coil powers than the 125 Vdc relays and were therefore not vulnerable to this failure mode. GENE concluded that no substantial safety hazard existed. However, upon investigating the failed MDR relays at River Bend as discussed in paragraphs 3.9 and 3.12, the NRC obtained results that contradict these conclusions.

3.14 Review of Audits Performed on P&B

The inspectors reviewed the audits performed by representatives of various nuclear power plants and other equipment suppliers and determined that P&B took appropriate actions to correct the adverse findings identified during the audits. The inspectors reviewed the following audits:

- GENE quality assurance audit report of October 14, 1988, conducted on September 20 and 21, 1988. The audit identified one finding and five observations. The results of the audit indicated that GENE considered P&B's status satisfactory, and P&B remained an approved supplier for safety-related MDR relays and commercial grade KH relays.

- Westinghouse Electric Corporation, Nuclear Services Integration Division (WNSID), conducted an audit on October 24-25,1984, and identified three findings: (1) P&B Internal Sales Orders were not always being reviewed and signed off by quality assurance personnel with the benefit of the customer's original contract, (2) discrepant material was not being tagged with "Hold For Disposition" tags, (3) during final assembly, defective MDR relays were not being identified with a "Defective" stamp. On November 27, 1984, P&B responded to WNSID's audit findings. WNSID audited P&B during October 23-25, 1990, and identified two adverse findings to which P&B responded satisfactorily on November 21, 1990.

- The Southern California Edison Company (SCEC) audited P&B during March 27-29 and May 11-13, 1988, for compliance with 10 CFR Part 50, Appendix B, and conducted a commercial grade survey in 1990 to assess P&B's control of critical characteristics during the manufacture of MDR relays. The results of the survey indicated that SCEC retained P&B on its Evaluated Suppliers List.

The inspectors did not identify any adverse findings in this area.

4 PERSONNEL CONTACTED

*L. O. Hume, Director, Governmental Affairs and Internal Audit
*K. D. Lueneburger, Director of Engineering
*W. Lamb, Manager, Product Engineering
*T. Loyd, Director of Manufacturing
*D. E. Patton, General Sales Manager
*R. Market, Manager, Product Engineering
*K. McGrew, Manager, Quality Assurance Planning
B. Mosier, District Sales Manager
M. Scully, Director of Quality Assurance

* Denotes the P&B personnel who attended the exit meeting on November 14, 1991

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

FEB 21 1992

Docket No. 99901158/89-01

Mr. Tim Swenson, Manager
Shop Services Incorporated
19390 South West Shaw Street
Aloha, Oregon 97007

Dear Mr. Swenson:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the attempted inspection of your facility at Aloha, Oregon, conducted by Messrs. J. Petrosino and R. Moist, of this office on April 11-12, 1989. The inspection was planned to follow-up on an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers. The NRC concern is discussed in detail in NRC Information Notice (IN) 88-48 and its supplements. The circumstances regarding the attempted inspection and subsequent discussions are discussed in the enclosed report. Release of this report was delayed during NRC's review of nonconforming and substandard vendor products.

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.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901158/89-01

ORGANIZATION: SHOP SERVICES INCORPORATED
ALOHA, OREGON

REPORT NO.: 99901158/89-01	INSPECTION DATE: April 11-12, 1989	INSPECTION ON-SITE HOURS: 2

CORRESPONDENCE ADDRESS: Tim Swenson, Manager
Shop Services Incorporated
19390 SW Shaw Street
Aloha, Oregon 97007

ORGANIZATIONAL CONTACT: Susan Swenson
TELEPHONE NUMBER: (503) 642-7874

NUCLEAR INDUSTRY ACTIVITY: Local sales representative for Crosby Valve.

ASSIGNED INSPECTOR: [signature] June 2, 1989
J. J. Petrosino, Reactive Inspection Section No. 1 Date
(RIS No.1)

OTHER INSPECTOR(S): R. Moist, RIS No. 2

APPROVED BY: CE Ader for June 2, 1989
E. T. Baker, Chief, RIS No.1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50.

B. SCOPE: The purpose of this unannounced inspection was to determine if Shop Services had purchased any valves from CMA International, Incorporated of Vancouver, Washington and to determine if those valves, if any, were supplied to any commercial nuclear power plants.

PLANT SITE APPLICABILITY: None Indicated.

REPORT NO.: 99901158/89-01	INSPECTION RESULTS:	PAGE 2 of 4

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed.

E. OTHER COMMENTS AND OBSERVATIONS:

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 to IN 88-48, dated August 24, 1988, discussed a potential problem concerning Vogt 2-inch valves (Vogt figure no. SW-13111) which were leaking steam at the bonnet and packing. The valves were purchased by Pacific Gas & Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually shipped from a valve salvage and refurbishment company in Vancouver, Washington (CMA International, Inc.). Henry Vogt representatives examined the subject valves at the Diablo Canyon Nuclear Power Plant and determined that they had not manufactured the valves.

The valves appear to be counterfeit based on the following: (1) the Vogt name was stamped on the side of the valve body instead of being forged on the body; (2) Vogt valves have round bonnet flanges whereas the valves in question have square bonnet flanges; (3) the valves in question have swing gland bolting which is not used by the Henry Vogt Company; and (4) the end-to-end dimensions of the valves in question are shorter than the Vogt SW-13111.

ORGANIZATION: SHOP SERVICES INCORPORATED
ALOHA, OREGON

REPORT NO.: 99901158/89-01	INSPECTION RESULTS:	PAGE 3 of 4

2. Discussions at Shop Services Incorporated on April 11, 1989

Initial discussions with Ms. Susan Swenson, Assistant Manager, indicated that Shop Services Incorporated (SSI) represented Crosby Valve. However the inspector was unable to obtain information regarding SSI's nuclear customers or if SSI did any business with CMA. The inspector left his phone number with Ms. Swenson and asked that the Manager of SSI call the NRC inspector the following week in order to explain the concerns of the NRC to the manager.

Subsequent to the inspector's visit, Mr. Tim Swenson called Mr. Bill Brach, Chief, Vendor Inspection Branch. After their discussion, Mr. Swenson indicated to Mr. Brach, that SSI would cooperate with the NRC.

3. Discussions at Shop Services Incorporated on April 12, 1989

Both NRC inspectors revisited SSI to reiterate the purpose of the visit and to seek SSI's cooperation. IN 88-48 and Supplement 1 to IN 88-48 were explained to the Assistant Manager. The Assistant Manager informed the inspectors that the requested information would have to be obtained from the SSI corporate office in Salt Lake City, Utah since SSI, Aloha, Oregon does not retain any purchase orders or invoices. However, the Assistant Manager did show the inspectors a recent purchase order from Washington Public Power Supply System addressed to Crosby Valve in care of SSI, Aloha, Oregon for purchase of Crosby Valves.

4. Subsequent Telephone Discussions with Shop Services Incorporated

Additional discussions were conducted the following week between the SSI Manager and NRC/VIB personnel regarding this matter. The manager explained that a misunderstanding had occured and, as stated before, would fully cooperate with the NRC inspection team. Mr. Swenson requested that the NRC contact him prior to traveling to his facility so that he could assure that he would be present to assist them (Mr. Swenson also stated that to the best of his knowledge SSI has not bought any valves from CMA or its valve repair facility, IMA Valve Repairing).

ORGANIZATION: SHOP SERVICES INCORPORATED
ALOHA, OREGON

REPORT NO.: 99901158/89-01	INSPECTION RESULTS:	PAGE 4 of 4

F. PERSONNEL CONTACTED:

Susan Swenson, Assistant Manager
*Tim Swenson, Manager

*By telephone

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

MAR 0 9 1992

Docket No. 99900081

Mr. Carl Volmer
Quality Assurance Manager
Siemens Nuclear Power Corporation
Engineering and Manufacturing Facility
2101 Horn Rapids Road
P.O. Box 130
Richland, Washington 99352-0130

Dear Mr. Volmer:

SUBJECT: NOTICE OF NONCONFORMANCE
(NRC INSPECTION REPORT NO. 99900081/92-01)

This letter addresses the inspection of your facility at Richland, Washington, conducted by Mr. S. L. Magruder, Mr. R. N. Moist, Mr. K. R. Naidu and Mr. J. J. Petrosino of this office on February 10-13, 1992, and the discussions of their findings with you and your staff at the conclusion of the inspection. The purpose of the inspection was to review Siemens Nuclear Power Corporation's (SNP) Engineering and Manufacturing Facility operations and quality assurance (QA) program. In addition, the inspectors reviewed SNP's corrective actions taken in response to a material control problem that led to the shipment of nonconforming fuel to Pennsylvania Power & Light Company's Susquehanna plant.

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

The team noted several strengths during the inspection, especially SNP's policy of over-checking material purchased from suppliers of safety-related material. SNP's responsiveness to customer and NRC concerns, and the level of knowledge and experience of the technicians and operators interviewed during the inspection were also considered strengths.

During this inspection it was found that the implementation of your QA program failed to meet certain NRC requirements regarding the development and implementation of appropriate QA procedures. These are summarized as follows: (1) a lot card was not being maintained in accordance with procedures; (2) a lot card was being used to document cleaning instead of a component cleaning record as required by procedure; (3) issue slips are not being used to reissue short zircaloy bar stock in the machine shop as required by procedure;

(4) nonconforming end caps were found on the same cart with acceptable end caps with no hold tags, red stickers or other identifying documentation as required by procedure; (5) certain operations involved in the welding of end plugs were not being controlled by a documented procedure; and (6) portions of a test performed to determine fuel rod fill gas purity and rod pressure were not being controlled by a documented procedure. The specific findings and references to the pertinent requirements for the above nonconformances are identified in the enclosures to this letter.

In addition to the nonconformances noted above, the team raised some concerns with SNP management during the inspection. The primary concern was with SNP's policy guide on the subject of 10 CFR Part 21. The team felt that there was a possibility that the policy guide may be misinterpreted, thereby causing some employees to not raise potentially reportable conditions to management's attention. The team felt that some additional guidance may be needed from SNP to clarify this issue.

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 its enclosures will be placed in the NRC Public Document Room.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Nonconformance
2. Inspection Report No. 99900081/92-01

NOTICE OF NONCONFORMANCE

Siemens Nuclear Power Corporation
Engineering and Manufacturing Facility
Richland, Washington
Docket No.: 99900081

Based on the results of an NRC inspection conducted on February 10-13, 1992, it appears that certain activities were not conducted in accordance with NRC requirements. The NRC has classified these items, as set forth below, as nonconformances to the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), imposed on Siemens Nuclear Power Corporation (SNP) by contract, and SNP's internal policies and procedures.

A. Criterion V of Appendix B to 10 CFR Part 50 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 and or drawings."

Contrary to the above, in the four examples that follow, SNP personnel failed to follow documented procedures (92-01-01).

1. Paragraph 2.4 of Quality Control Procedure Number ANF-P69600, "Guidelines for Completion of Follower Cards and Other Data Forms," Revision 3, dated January 30, 1991, states, "[i]f you make an erroneous entry on a data sheet do not erase or attempt to obliterate the entry. Simply strike out the false entry with a single line, record the correct entry, and initial and date the correction nearby. Violation of the practice may be construed to be falsification of documentation."

   Contrary to the above, Lot Card, ANF Lot No. 9132-38811, for zircaloy bar stock, had several blocks where previous entries, such as, the job number, release number and quantity, were completely obliterated.

2. Paragraph 2.0 of Quality Control Standard (QCS) P68146, "Cleaning and Passivation of Tubing, Small Components, and Hardware," Revision 7, dated October 12, 1989, states, "[v]erify that cleaning and/or passivation has been acceptably performed by attaching a filled out component cleaning record to the cleaned components container."

Contrary to the above, Lot Card, ANF Lot No. 5575-2302-36050, for lower end caps was used, instead of a component cleaning record, to annotate that the cleaning inspection was performed.

3. Paragraph 2.1 of Purchasing and Logistics Procedure 9.12.8, "Materials Moves and Issues," Revision 0, dated June 30, 1986, states, in part, that "[i]ssue slips will be prepared for all materials and hardware transfers."

   Contrary to the above, residual zircaloy-2 and zircaloy-4 bar stock left over from the Star screw machine used for machining upper and lower end caps were not being issued back to material control or reissued back to the machine shop by an issue slip.

4. Section 3.1.2, "Identification," of SNP's QA procedure P00,039, "Control of Nonconforming Items," Rev. 12, dated August 30, 1991, states, "[n]onconforming items shall be appropriately identified to prevent inadvertent mixing with acceptable material. A red hold tag or red sticker shall be used unless identification and control are specifically covered by an approved implementing procedure; e.g., hard scrap to be recycled. The person identifying the nonconforming item shall complete the pertinent information required on the tag or sticker and shall affix it to the nonconforming item, place it with the lot card or attach it to the manufacturing order/follower card which identifies the nonconforming items."

   Contrary to the above, nonconforming end caps from five different SNP jobs (numbers 36919, 37236, 36903, 37263, and 36730) were found on a cart, near the end cap laser printing machine, alongside two boxes of acceptable end caps (job numbers 38311 and 38312). The NRC inspectors observed that the nonconforming items were not identified with hold tags, red stickers, or other identifying documentation, and that there was no means to control inadvertent mixing of the nonconforming material with the acceptable material.

B. Criterion V of Appendix B to 10 CFR Part 50 states: "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 and or drawings. Instructions, procedures, or drawings, shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished."

Contrary to the above, in the two examples that follow, SNP failed to ensure that activities affecting quality were adequately prescribed in its QA program documents (92-01-02).

1. Section 3.2 of QCS P68537, "Rod Inspection," Rev. 24, dated January 8, 1992, states, in part, "[w]eld traces for voltage, amperage, RPM, and pressure will be reviewed at the time the rods [end caps to rods] are welded to assure there are no electrode touches or other anomalies. Quality control shall overcheck approximately 10% of weld [strip chart] traces per welder for anomalies."

   Contrary to the above, SNP failed to establish or provide quantitative or qualitative acceptance and rejection criteria for QC inspectors to determine whether or not the end cap to zircaloy tube weld traces indicated anomalies that would cause a weld to be rejected.

2. Section 5, "Instructions, Procedures, and Drawings," of SNP's topical report, ANF-1A, "Quality Assurance Program For Nuclear Fuels," Rev. 24, dated January 21, 1991, states, in part, that "quality-related design, procurement, fabrication, inspection, handling, and shipping activities are prescribed by documented instructions, procedures, and drawings, as appropriate, to assure adequate definition of the inspections for satisfactory completion of activities."

   Contrary to the above, SNP failed to establish or provide steps in Analytical Procedure P69346, "Determination of Fill Gas Purity and/or Pressure in Fuel Rods," Rev. 6, dated July 10, 1986, to direct the lab technician to perform a check of the test setup at pressure as is the standard SNP lab practice.

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, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, 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 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 9th day of March, 1992

ENCLOSURE 2

ORGANIZATION: SIEMENS NUCLEAR POWER CORPORATION
ENGINEERING AND MANUFACTURING FACILITY
RICHLAND, WASHINGTON

REPORT NO.: 99900081/92-01

CORRESPONDENCE ADDRESS:

Mr. Carl Volmer
Quality Assurance Manager
Siemens Nuclear Power Corporation
Engineering and Manufacturing Facility
2101 Horn Rapids Road
P.O. Box 130
Richland, Washington 99352-0130

ORGANIZATIONAL CONTACT:

Mr. Carl Volmer
Quality Assurance Manager

NUCLEAR INDUSTRY ACTIVITY: Nuclear fuel pellet and assembly supplier.

INSPECTION CONDUCTED: February 10-13, 1992

Stewart L. Magruder 3/5/92

Stewart L. Magruder, Team Leader — Date
Special Projects Section
Vendor Inspection Branch (VIB)

Randolph N. Moist, VIB
Kamalakar R. Naidu, VIB
Joseph J. Petrosino, VIB

Gregory C. Cwalina 3/5/92

Gregory C. Cwalina, Chief — Date
Special Projects Section
Vendor Inspection Branch

INSPECTION BASES: 10 CFR Part 21 and 10 CFR Part 50, Appendix B

INSPECTION SCOPE: Observe the fabrication and testing of fuel assemblies and review Siemens Nuclear Power Corporation's (SNP) implementation of its quality assurance program. In addition, review SNP's corrective actions taken in response to a recent material control problem there.

PLANT SITE APPLICABILITY: Numerous Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) sites.

1 INSPECTION SUMMARY

1.1 Nonconformances:

1.1.1 Nonconformance 99900081/92-01-01

Contrary to Criterion V of Appendix B to 10 CFR Part 50, SNP personnel did not follow documented procedures in four instances:

1. Contrary to Section 2.4 of SNP Quality Control (QC) Procedure P69600, "Guidelines for Completion of Follower Cards and Other Data Forms," Revision 3, dated January 30, 1991, Lot Card, ANF Lot No. 9132-38811, for zircaloy bar stock, had several blocks where previous entries, such as, the job number, ANF release number and quantity, were completely obliterated.

2. Contrary to Section 2.0 of SNP QC Standard ANF-P68146, "Cleaning and Passivation of Tubing, Small Components, and Hardware," Revision 7, dated October 12, 1989, Lot Card, ANF Lot No. 5575-2302-36050, for lower end caps was used, instead of a component cleaning record, to annotate that the cleaning inspection was performed .

3. Contrary to Section 2.1 of Purchasing and Logistics Procedure No. 9.12.8, "Material Moves and Issues," Revision 0, dated June 30, 1986, residual zircaloy-2 and zircaloy-4 bar stock left over from the Star screw machine used for machining upper and lower end caps were not being issued back to material control or reissued back to the machine shop by an issue slip.

4. Contrary to Section 3.1.2 of Quality Assurance (QA) Procedure P00,039, "Control of Nonconforming Items," Rev. 12, dated August 30, 1991, SNP failed to identify and control five bags of nonconforming end caps.

1.1.2 Nonconformance 99900081/92-01-02

Contrary to Criterion V of Appendix B to 10 CFR Part 50, SNP did not ensure that activities affecting quality were adequately prescribed in QA program documents in two instances:

1. Contrary to Section 3.2 of Quality Control Standard (QCS) P68537, "Rod Inspection," Rev. 24, dated January 8, 1992, SNP failed to establish welder strip chart (trace) acceptance and rejection criteria for end cap to zircaloy tube welding.

2. Contrary to Section 5 of SNP's topical report, ANF-1A, "Quality Assurance Program For Nuclear Fuels," Rev. 24, dated January 21, 1991, SNP failed to establish or provide steps to direct lab technicians to perform checks of a test setup at pressure as is the standard SNP lab practice.

2 STATUS OF PREVIOUS INSPECTION FINDINGS:

2.1 (Closed) Nonconformance (86-01-01)

Contrary to Section 3.0, "Design Control," of the Exxon Nuclear Company, Incorporated Quality Assurance Program Topical Report for Nuclear Fuel Design and Fabrication, XN-NF-1A, Revision 7, dated January 1985, and Section 1.2.12 of Quality Assurance (QA) Procedure XN-NF P00,002, the verification calculations for the addition of the Moody critical flow model in the RELAP/MOD5 computer code (version UOCT85) did not adequately test the implementation of the new modelling with the theoretical basis.

SNP issued a report titled "Additional Verification of the Moody Model," dated April 9, 1987, which expanded the original computer code calculations to consider different values for steam quality. The NRC inspectors reviewed this report and consider this issue resolved.

2.2 (Closed) Nonconformance (86-01-02):

Contrary to Section 9, "Control of Special Processes," of XN-NF-1A, Revision 7, an operator used an unapproved procedure to operate the x-ray machine while performing fluoroscopy on fuel rods in the manual mode.

The NRC inspectors reviewed XN-NF-1000, "Through Rod X-Ray Fuel Inspection Program Operation Reference Manual," dated November 1986, which provides instructions for the operator and procedure P66,789, "Radiographic Procedure for Through Rod X-Ray," Rev. 3, dated November 27, 1990, which incorporates the instructions into SNP's QA program. The NRC inspectors consider these procedures to be adequate to resolve this issue.

2.3 (Closed) Nonconformance (86-01-03)

Contrary to Section 9 of XN-NF-1A, Revision 7, an operator used an unapproved coordinates chart to read the values on a current ultrasonic testing (UT) trace being used for the zero settings on the Gould ES 1000 recorder.

The NRC inspectors reviewed procedure P69027, "3-D Gage," Rev. 14, dated September 19, 1990. Section 6.6 of this procedure has been updated to include a reference to the coordinates chart. The NRC inspectors consider this adequate to resolve this issue.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

The NRC inspectors informed SNP staff of the scope of the inspection, outlined areas of concern, and established working interfaces during the entrance meeting on February 10, 1992. On February 13, 1992, the NRC inspectors summarized the results of the inspection for SNP management during the exit meeting.

3.2 Background

SNP produces fuel assemblies for General Electric design BWRs and Westinghouse and Combustion Engineering design PWRs. They also provide fuel pellets for Babcock & Wilcox's Commercial Nuclear Fuel Plant. Operations performed at SNP include: converting UF_6 gas into UO_2 fuel pellets, manufacturing gadolinium pellets loading the pellets into zircaloy rods, manufacturing end caps for the rods, welding the end caps, leak testing and scanning the rods, manufacturing spacer grids and tie plates, building the assembly, and testing the products at various stages of the processes. This inspection was intended to provide the NRC inspectors with an overview of the operations at the facility and an opportunity to assess the effectiveness of the QA program. It also provided an opportunity to review the corrective actions being taken by SNP in response to a material control problem that resulted in nonconforming fuel being shipped to Pennsylvania Power & Light's Susquehanna plant.

3.3 Observation of Activities in Progress

3.3.1 Fuel Pellet Fabrication

The NRC inspectors observed several operations in SNP's pellet fabrication area. The overall area is divided into three major activities: conversion of UF_6 into UO_2; processing UO_2 into pellets; and pellet processing and acceptance. The NRC inspectors noted that the manufacturing and quality control activities in this area are specified in two SNP QCSs; P68150, "UO_2 Powder Processing," Rev. 8, dated September 15, 1989, and P68152, "UO_2 Powder and Pellet Processing and Certification," Rev. 55, dated September 30, 1991.

At the beginning of the process, SNP receives canisters of UF_6 that are usually provided by the applicable licensee. SNP converts the UF_6 gas into UO_2 powder and then blends, mills, and granulates it to a certain consistency. During this process the uranium enrichment for each batch-lot is determined, and sample pellets from each UO_2 batch are chosen for an isotopic analysis. Following the batch lot sampling, the UO_2 is pressed into "green" pellets, and sintered to achieve a ceramic-type density. After the pellets are sintered, they are ground to meet dimensional requirements, weighed, sampled, and verified by manufacturing. The pellets are then over-inspected by QC. At the completion of this process, the pellets are transferred to the next area for loading into zircaloy tubing.

The inspectors observed a QC inspector pick a sample of pellets from the production lot and subject them to various tests as delineated in the "Uranium Pellet Physical Properties Certification" check list. The check list provided information on the characteristics to be verified, the location where the requirement for the characteristic can be found, the inspection method to be used to verify the requirement, the sample size, and acceptance/rejection criteria.

3.3.2 Fuel Rod Assembly

The NRC inspectors observed several in-process activities in the fuel rod assembly area. Numerous activities are performed in the fuel rod assembly area including: upper and lower end cap welding to the zircaloy tubing, fuel pellet loading into the tubes, laser marking of end caps, rod plenum length verification, helium leak checks, weld radiography, nuclear assay for determining the enrichment of the pellet column and gadolinium content, and final rod inspection. Several of these areas were inspected to determine whether the activities were being performed in accordance with written instructions, procedures, and drawings.

The NRC inspectors noted that many of the manufacturing and QC activities being performed in the area were delineated in QCS P68537, "Rod Inspection," Rev. 24, dated January 8, 1992. Other SNP procedures which address particular activities in the area were also identified and reviewed, such as, procedure P66,787, "Helium Leak Check, Horizontal Test Station UO_2 Building," Rev. 3, dated July 17, 1990. The NRC inspectors observed and questioned manufacturing and QC personnel in the area regarding their specific responsibilities, training, and whether they had procedures and instructions to guide them in their work.

The NRC inspectors reviewed a Manufacturing Order Follower (MOF) at the lower end cap weld station. The MOF furnished the manufacturing order number and detailed the various operations to be performed to complete the product. A typical MOF lists all the operations to complete a batch of fifty rods. The origin of the material used for the production of that batch of rods can be traced by an assigned QC release number. At this operation, the lower end cap was being welded to the clad using Weld Procedure Specification (WPS) EMF-PQ-572, Revision 1, of October 7, 1991, "Lower End Weld,0.425" O.D. [outside diameter] 0.028" Wall, K Station." The inspection and test plan specified the relevant drawing and the QC standard to inspect the completed weld.

The NRC inspectors observed the loading of UO_2 fuel pellets with various enrichments into the clad. The MOF reviewed identified the various operations required to be performed to complete the loading, the sequence of the operations, the number of pellets rejected, and the date when the operation was performed. The QC inspection requirements and the standard to which the inspection was to be performed were also detailed. A "Rod Characteristic Sheet" provided the information to set up the work order for the rod loader. The loading operation, which was assisted by a computer input, was performed on a batch of fifty fuel rods. Each batch of different fuel enrichment pellets was weighed and verified before being loaded into the clad to ensure compliance with the applicable design drawing.

At the end closure weld station, the inspectors observed the upper end cap being welded to the clad. The welding was performed after inserting the spring, outgassing the pellets in the clad, and filling it with helium. WPS EMF-PQ-545 R5, Revision 5, of October 28, 1991, "Upper End Weld, 0.425" O.D., 0.028" Wall, H&M Station," was used for this welding operation.

The NRC inspectors concluded that the SNP activities in the fuel rod assembly area were generally well controlled, and the personnel exhibited proficiency in the performance of their duties. However, two problems were identified by the inspection team. The first problem concerned a failure of SNP to establish QC inspector acceptance and rejection criteria to comply with the intent of Section 3.2, "Welding," of QCS P68537, "Rod Inspection," Rev. 24, dated January 8, 1992. The second problem was the failure of SNP to maintain control over nonconforming end caps in accordance with Section 3.1.2, "Identification," of QAP P00,039, "Control of Nonconforming Items," Rev. 12, dated August 30, 1991.

The first problem was identified during discussions with QC staff regarding the method by which QC performed the 10 percent over-check of manufacturing's verification of the automated end cap welding process parameters. The QC inspector explained to the NRC inspectors that he checked the weld machine strip chart "weld traces" for electrode touches or other anomalies. When the NRC inspectors asked how the QC inspector determined what was acceptable or rejectable, the QC inspector revealed that SNP had not established any criteria for QC inspectors to use. The NRC inspectors determined that before December 13, 1991, the QC staff was not required to perform any over-inspections of the manufacturing process or examine weld traces for anomalies. SNP established this new QC oversight requirement in Rev. 23 of the rod inspection procedure, P68537, but apparently failed to establish or provide its QC inspectors with acceptance/rejection criteria. Consequently, the QC inspectors obtained their own weld trace inspection methodology and criteria from the manufacturing personnel who performed the welds and from QC engineering personnel. The NRC inspection team, therefore identified this as a failure to establish acceptance and rejection criteria. (See Nonconformance 92-01-02)

The second problem was identified by the inspection team near the end cap laser print room. The NRC inspectors observed several boxes containing small plastic bags with different quantities of end caps in each bag. The boxes had been placed on the top shelf of a small cart used for moving parts around on the shop floor. Upon closer examination of the boxes, the inspectors discovered that: (1) two of the boxes contained used plastic bags (trash); (2) two other boxes contained end caps identified with SNP job numbers 38311 and 38312, both of which contained a green SNP job ticket indicating acceptable material; and (3) the last box contained five plastic bags of end caps that appeared to be nonconforming because the numbers on them were X'd out. The five bags were marked with the following SNP job numbers: 36730, 36903, 36919, 37236, and 37263.

Consequently, the NRC inspectors noted two concerns, (1) neither the box nor the five plastic bags of nonconforming end caps were identified with a red hold tag, red sticker, or other nonconforming material document; and (2) the box of nonconforming end caps, and the two boxes of acceptable end caps were found alongside each other on the shop cart. (See Nonconformance 92-01-0[illegible])

3.3.3 Fuel Bundle Assembly and Final Inspection

The inspectors observed the assembly of fuel bundles identified as ANE-359 and ANE-362 for manufacturing order GGA 7893 ANF-1.5. The inspectors determined that the operators were following the written instructions and waiting for QC inspectors to perform in-process inspections at hold points specified in the work order.

The NRC inspectors observed QC personnel perform final inspections on bundle assembly ANE-359 to verify the following attributes while it was in the horizontal position: (1) the orientation of the lower tie plate; (2) the orientation of the spacers; (3) the presence and legibility of the engraved fabricator/reactor code (UD26) and the fuel assembly serial number; and (4) the orientation of the upper tie plate.

The fuel bundle was raised to a vertical position and the NRC inspector accompanied the QC inspector in a crane to observe him perform inspections to verify that the following attributes met the design drawing: (1) length measured between the lower and upper tie plate; (2) assembly envelope and channeling; (3) the perpendicularity of the assembly; (4) all fuel rods are seated in the lower tie plate; (5) the spacing between rods are within tolerances; and (6) all compression springs are in-place and properly seated.

3.3.4 Neutron Absorber Fuel Rod Assembly

SNP mixes UO_2 and gadolinium to form pellets for its neutron absorber fuel (NAF). The pellet forming and assembly of the NAF rod assemblies is done in a separate building to avoid mixing the NAF pellets with regular fuel pellets. The inspectors limited their observations in this building to the assembly of the NAF rods. During the inspection, completed NAF rods were being subjected to passive nuclear assay to verify the uniformity of the uranium enrichment over the entire length of the rod. A review of the MOF for the completed rods indicated that the various operations including inspections to verify: the integrity of the lower and upper end cap welds; NAF pellet loading; outgassing of the pellets; and the lengths of the plenum and column had been successfully completed.

3.4 Susquehanna Lower End Cap Issue

3.4.1 Background

On January 16, 1992, SNP notified Pennsylvania Power & Light (PP&L) Company that 57 fuel rods fabricated for its Susquehanna Steam Electric Station (Susquehanna) Unit 1/Cycle 7 reload contained lower end caps made from zircaloy-4 instead of the required zircaloy-2 material. The 57 fuel rods were contained in 8 fuel bundles at Susquehanna Unit 1 and in 4 fuel bundles at SNP. On January 27, 1992, SNP notified PP&L that an additional 22 fuel rods were identified having lower end caps made from zircaloy-4 material. These additional 22 fuel rods were contained in 5 fuel bundles at Susquehanna Unit 1.

SNP replaced the affected fuel rods in the 4 bundles that still were at SNP. SNP plans to replace all affected fuel rods at Susquehanna Unit 1, prior to the affected bundles being loaded in the reactor.

3.4.2 SNP Response

As a result of the incident, SNP convened an Incident Review Board (IRB) to focus on the following areas: (1) review the circumstances surrounding the improper use of the wrong alloy for end caps used on Susquehanna fuel rods; (2) evaluate the disposition of the affected product; (3) consider generic implications of the problem; (4) determine root causes; (5) establish solutions to preclude recurrence; and (6) establish corrective actions and a means to follow up on implementation.

The IRB was chaired by the QA manager and was composed of representatives from the following groups: QA engineering, fuel design engineering, process engineering, materials and scheduling, and quality control.

Prior to the NRC inspection, SNP had held 5 IRB meetings to investigate and evaluate the facts to determine the root cause of the problem. Although the IRB has not concluded its investigation, it did identify six elements that may have caused or contributed to SNP producing end caps with the wrong alloy. The elements include: (1) inadequate zircaloy bar stock material control in the machine shop at the Mori Seiki lathe (primary root cause); (2) short zircaloy rod bar ends left over from the Star screw machine were allowed to accumulate instead of being returned to material control in a timely manner; (3) operators at the Mori Seiki lathe identified bar stock based on diameter instead of alloy; (4) the QC receiving inspector released the initial bar stock after over-checks to verify alloy, and subsequent releases did not specifically verify alloy; (5) SNP standard operating procedures did not require a zircaloy cleanout between bar stock lots on the Mori Seiki lathe; and (6) lot traceability of end caps was lost when the next lot of end caps was mixed in at the laser marking station.

The following corrective actions have been proposed by the IRB: (1) standard operating procedures in the machine shop should be revised to include zirconium cleanouts of the Mori Seiki lathe at each lot change to assure that material lots are not mixed; (2) operators should be trained on changes to standard operating procedures and the importance of alloy composition and traceability; (3) material control should pick up all bar ends from the Star screw machine in a timely manner; (4) alloy composition should be added to the information on a lot card; (5) consideration should be given to having vendors stamp the zircaloy alloy type on the ends of the rod bar; (6) the standard operating procedure at the laser marking machine should be revised to include only one method of loading laser marker magazines and how to maintain lot control; and (7) QC should provide over-checks of component alloys by part number control.

SNP evaluated this issue and determined that the condition was not reportable under 10 CFR Part 21 based on the following: (1) zircaloy-4 alloy has mechanical properties and microstructure that are close to those of zircaloy-2 alloy; (2) the material difference between the two different alloys is not sufficient to significantly affect the metallurgical stability or the weld properties; (3) the hydriding and corrosion tendencies differ somewhat between the two alloys but would not have a significant effect on fuel performance; (4) welding of the two different alloys has been a standard practice in the design and manufacturing of BWR fuel assemblies since spacer capture rod sleeves made of zircaloy-4 sheet are spot welded to Zircaloy-2 spacer rod tubes; and (5) the alloy mix up was an isolated incident.

3.4.3 NRC Review

The SNP QA manager described for the NRC inspection team the preliminary corrective actions that SNP has proposed as a result of the Susquehanna incident. The NRC inspectors reviewed SNP's current program for the fabrication of end caps to verify the corrective actions that had been implemented. This review included discussions with personnel involved in purchasing and receiving the material, as well as, observations of work in the machine shop. In addition, the NRC inspectors reviewed documentation associated with the Susquehanna incident.

The NRC inspectors reviewed the contract between PP&L and SNP, dated April 19, 1989, for fuel bundle fabrication and related services and verified that Article 14 of the contract, "Quality Assurance and Control," imposed 10 CFR Part 50, Appendix B, Quality Assurance Program and 10 CFR Part 21. In addition, the NRC inspectors verified that Appendix A, "Fuel Bundle Design," of the contract specified Zircaloy-2 as the end cap material.

The NRC inspectors also reviewed SNP Purchase Order (PO) R-071999, dated February 25, 1992, to CEZUS, C/O Pechiney World Trade (USA) of Secaucus, New Jersey, for the purchase of zircaloy-2 rod stock. The NRC inspectors verified that Appendix B to 10 CFR Part 50, 10 CFR Part 21, and Design Specification ANF-S35007, "Zircaloy Bar and Rod Stock," was imposed in the PO. The NRC inspectors verified that CEZUS was on SNP's Approved Vendors List, ANF-595, Revision 12.

The NRC inspectors were interested in knowing whether CEZUS' manufacturing plants in Rugles and Montreuil, France, received the same PO requirements that were sent originally to Pechiney World Trade (USA), in New Jersey. The SNP Purchasing Manager stated that the acknowledgment copy of the PO is approved and accepted by the manager of export sales in France and that the return address on the envelope is from France.

During this inspection the SNP Purchasing Manager asked a member of his staff to call Pechiney World Trade (USA) for clarification on how they passed on purchasing requirements to France. Pechiney World Trade (USA) sent a letter to SNP during this inspection confirming that they send the entire original PO documents to Paris, France, where there is a second, in-depth, review of documents by commercial and production engineering departments, and quality control personnel. When everyone is in agreement, the acknowledgment copy of the PO is signed and sent directly to SNP, constituting CEZUS' official acceptance of the order. In addition, SNP performs vendor audits at the manufacturing plants in Rugles and Montreuil, France.

The NRC inspectors selected Component/Material release number 37233 for zircaloy-2 rod stock that was purchased on the above PO to verify that all activities affecting quality were performed in accordance with SNP procedures. When the above material was received at SNP, a Material Lot Card was generated by material control, which showed the lot number, drawing and revision number, part name, part number, job (project) number, PO number, vendor number, vendor lot number, and the current quantity of the part. The material control technician affixed a lot card label to the Lot Card to show the QC status of the material. In this step a yellow label is affixed to the Lot Card denoting that the received material was not inspected. The material is then moved to the receiving inspection area.

The NRC inspectors reviewed the incoming inspection documentation for release number 37233 such as: SNP's Analytical Laboratory Report (over-checks to verify chemical composition); SNP's Test Reports (Stringer and Porosity Examination and dimensions); CEZUS' Analysis and Inspection Certification from Mill; CEZUS' Final Product Inspection Report showing gas analysis, mechanical properties at room and elevated temperatures, corrosion test, macrographic, metallographic, and ultrasonic examinations, visual and dimensional inspection; and Archive Samples Storage Form. The NRC inspectors determined that the PO requirements were met.

The NRC inspectors also observed work in progress in the machine shop. Specific operations observed included the machining of end caps on the Star screw machine for fuel rods, the control of rod stock to and from the machine shop, and the control and inspection of the machined end cap. The NRC inspectors reviewed a Lot Card, ANF Lot No. 9132-38811, that was attached to the Star screw machine for the current job being processed. The NRC inspectors noted that the Lot Card had several blocks where previous entries, such as, the Job number, ANF release number and quantity were completely obliterated and new information added. The material control technician showed the NRC inspectors documentation that indicated that the material had been conditionally released and subsequently accepted. In addition, the NRC inspector verified that a green label denoting "acceptable" covered over the blue label denoting "conditional release" which had previously been applied to the front of the Lot Card. The NRC inspector noted that SNP did not follow procedures which state that any changes to entries require a strike out with a single line with the individuals initials and date. (See Nonconformance 92-01-01)

The NRC inspectors asked the Manager of Materials and Purchasing how zircaloy rod stock material was being controlled at the Star screw machine and Mori Seiki lathe. The Manager stated that the rod stock is issued to the machine shop for use at the Star screw machine from material control on an issue slip and also returned to material control on an issue slip. In addition, the NRC inspector identified that the operators at the Star screw machine and Mori Seiki lathe have an MOF which shows the part number of the rod bar stock used and release number to perform their operation. The Lot Card for the rod stock also shows the part number and release number. This information allows the operator to determine if the right alloy material was used to machine end caps.

The NRC inspectors asked the material control technician how the residual rod bar stock (approximately 12" pieces) that was left over from the Star screw machine was controlled. He stated that short rod bar stock is placed in a box (which has the release number annotated on it) along with the Lot Card by the operator of the Star screw machine. The senior technician, who oversees operations at both the Star screw machine and Mori Seiki lathe, stated that he takes the box to material control every morning when there is material to take.

The NRC inspectors asked the material control technician if issue slips are used for the residual short bar stock being transferred back to material control and reissued back to the machine shop for use on the Mori Seiki lathe which is capable of handling the short bar stock to make end caps. The material control technician stated that once the short bar stock is received from the machine shop, he logs the material by pieces instead of length at his computer terminal and places the material in stock. The NRC inspectors went to the stockroom and observed that the short bar stock was located in a box along with the Lot Card and that the box had the release number printed on it. The material control technician stated that when the machine shop requests short bar stock, material control takes the box containing the bar stock and Lot Card to the operator at the Mori Seiki lathe at the machine shop for continued fabrication of end caps.

The material control technician stated that issue slips had never been used and are not currently used for controlling the residual short bar stock left over from the Star screw machine. The NRC inspectors identified this concern as a nonconformance since SNP was not following their purchasing and logistics procedure which states that issue slips will be prepared for all materials and hardware transfers. (See Nonconformance 92-01-01)

Although no work was being performed at the laser marking area, the NRC inspectors asked questions concerning why lots were mixed when marking end caps. The general supervisor of the rod and bundle area stated that two methods were being used to segregate lots of end caps depending on the shift and operator. One method was to leave a blank space in the end cap carrier tray where normally an end cap would be placed. This would signal the automatic laser marking machine to stop when it passed over the blank space in the end cap carrier. However, this method caused mixing of end cap lots because the first shift operator thought that the third shift operator failed to place an end cap in the missing space in the end cap carrier. Another method was to place an empty end cap carrier between end cap lots which would signal the laser marking machine to stop. The general supervisor stated that currently the operators on any shift at the laser marking machine only process one end cap lot at a time and that procedures are being updated to reflect this methodology.

The NRC inspectors reviewed MOFs and inspection and test plans for the machined lower end caps. Inspection characteristics were selected from the drawing and specification requirements and incorporated in the inspection and test plan. No discrepancies were identified by the NRC inspectors during this review.

While in an area just outside of the laser marking machine, the NRC inspectors observed a bag of end caps on a desk with a Lot Card, ANF Lot No. 5575-2302-36050, attached to the bag. The Lot Card had a red QC stamp imprinted on it annotating that QC had inspected the cleaning of the end caps. The NRC inspectors identified this concern as a nonconformance since SNP procedures state that cleaning inspection of end caps should be annotated on a component cleaning record. (See Nonconformance 92-01-01)

Finally, the NRC inspectors reviewed the archive files, which show what materials were used on a particular project, including rod bar stock, for eight nuclear power plants. This review did not identify any other instances where the wrong zircaloy material was used. This same over-check of the archive file was what led the material control technician to discover the Susquehanna incident.

The NRC inspection team determined that since SNP was still investigating and implementing their proposed corrective actions for the mix-up of the zircaloy material, the results of SNP's investigations and corrective actions would be reviewed during a future NRC inspection.

3.5 Laboratory Operations

The NRC inspectors observed operations in both the UO_2 and the Metallurgical Laboratories during the inspection. The technicians observed during the inspection were very knowledgeable about the tests they were performing and were well aware of the importance of their work on the quality of the fuel.

Processes observed in the Metallurgical Lab included burst tests of zircaloy tube samples, a tensile test of a zircaloy tube sample, and metallagraphic analysis of end plug weld samples. The NRC inspectors also reviewed the procedures for the burst and tensile tests: P69514, "Burst Testing of Tubing," Rev. 2, dated June 26, 1991; and P69517, "Tensile Testing of Tubing, Sheet Stock, Rod and Bar Stock," Rev. 4, also dated June 26, 1991. The procedures were followed closely and provided adequate guidance for the technician. The NRC inspectors also briefly observed a technician reading radiographic film of end plug welds.

Processes observed in the UO_2 Lab included tests on a completed fuel rod for fill gas purity and pressure, a test to determine the levels of fluorine and chlorine in a fuel pellet, and a gravimetric test to determine the percent uranium and oxygen to uranium ratio in a fuel pellet. The procedures for the latter two tests, P69256, "Fluoride and Chloride by Pyrohydrolysis," Rev. 10, dated December 1, 1989, and P69221, "Gravimetric Percent Uranium and Oxygen to Uranium Ratio," Rev. 10, dated October 1, 1989, were reviewed and found to be detailed and easy to follow. The technician performing these two tests followed them both closely.

The procedure for the fuel rod fill gas purity and pressure test, P69346, "Determination of Fill Gas Purity and/or Pressure in Fuel Rods," Rev. 6, dated July 10, 1986, was also reviewed by the NRC inspectors. This procedure was not well organized and was difficult to follow. When the actual test was observed, the technician performing it confirmed that the procedure was too difficult to follow and that he did not use it when he performed the test. The NRC inspectors observed that there were many pen and ink changes to the procedure in the lab that were not included in the latest revision obtained from the document control center. They also noted that the technician performed several operations to check the test set up at pressure that were not specified in the procedure. (See Nonconformance 92-01-02)

3.6 10 CFR Part 21 Program

The NRC inspectors reviewed SNP's procedures written to implement the requirements of 10 CFR Part 21 (Part 21) and evaluated the effectiveness of the program during the inspection. SNP has established Policy Guide (PG) 10.2, "Nuclear Safety Hazards Reporting," dated December 17, 1991 to implement its Part 21 program. The NRC inspectors reviewed PG 10.2 and verified that it and the other required documents were posted in accordance with Section 21.6, "Posting Requirements," of Part 21. The inspection team also noted that PG 10.2 appeared to contain all the necessary requirements of the July 31, 1991, revision of Part 21.

The NRC inspectors were concerned, however, about two sections of PG 10.2 that may be subject to misinterpretation by SNP employees or consultants. Part IV, "Responsibilities," Section C and Part V, "Procedures," Section B, both contain specific language that could be interpreted as requiring the SNP employees to perform an initial evaluation of a deviation to determine whether it could result in a substantial safety hazard. Part IV, Section C reads as follows:

> All employees are responsible for reporting to their management any failure to comply or deviation or defect which could result in a substantial safety hazard or a significant impairment of a basic component of a licensed facility operated by SNP or by its customer.

Part V, Section B reads as follows:

> All employees shall provide to their immediate supervisor any information reasonably indicating that a facility, activity, service, or basic component supplied to a facility: [1] Fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order or license of the Commission relating to substantial safety hazards, or [2] contains errors, deviations or defects which could create a substantial safety hazard.

The NRC inspectors were concerned that the language in the PG may be difficult for some employees to understand and may reduce the reporting of problems to management. Specifically, the NRC inspectors were concerned that an SNP employee may interpret the sections quoted above to mean that the individual employee is responsible for evaluating the safety significance of the deviation and that only those deviations that could create a substantial safety hazard, in their opinion, would have to be reported to management.

This concern was expressed and discussed with SNP management during the inspection.

3.7 Internal Audits

The NRC inspectors reviewed SNP's internal audit program during the inspection. The program was found to be well organized and well run. The following recently completed audits were reviewed:

AUDIT NO.	10 CFR PART 50 APPENDIX B CRITERIA COVERED
91:13	1 & 2
91:33	6, 7 & 10
91:67	8 & 9
91:71	11
91:87	16 & 18

The audits were generally conducted by one member of the SNP QA staff for a full week and appeared to be thoroughly done. It was apparent from the reports that the auditors were given the independence and authority to look at anything they needed and to be candid in their report. Each audit reviewed produced several findings which were recorded on Corrective Action Reports (CARs). Some of the audits also produced comments and suggestions for areas that could be improved. The CARs and comments were answered in a manner that indicated that the problem had been well researched and adequately resolved.

3.8 Review of Quality Assurance Records

The NRC inspectors reviewed records to determine whether SNP personnel performed required inspections, had qualified the weld procedure specifications used in welding the upper and lower end caps, and had ensured that its vendors supplied materials that met applicable PO requirements.

3.8.1 Review of Fuel Rod Records

The NRC inspectors reviewed the records for a recently completed manufacturing order for the Entergy Operations Inc. Grand Gulf Plant, order no. GGA 7893, and determined that all the necessary inspections had been performed. The records indicated that several NAF rods had to be re-scanned because an error had been made when the active scanning machine was operated. SNP determined that the operators should have activated two switches. One switch sets up the machine to verify the enrichment in the gadolinium segments and the other to measure the total fissile content of the rod. Instead, the operator had only activated the latter switch. SNP personnel discovered the error while reviewing the records, after questions were raised during an audit by Grand Gulf personnel, and subsequently notified Grand Gulf of the problem. Grand Gulf has informed SNP that it intends to write a CAR on this issue and SNP has already begun to investigate the problem. All the rods have been re-examined and determined to be within specifications.

3.8.2 Review of Cladding Records

The NRC inspectors reviewed PO R-072239 of August 1, 1991, to Sandvik Special Metals (SSM), Kennewick, Washington, for 5,000 pieces of 0.425-inch outside diameter (OD), 0.364-inch inside diameter (ID), Zircaloy-2 cladding. The cladding was to meet the requirements of specification ANF-S35055, Revision 6, and drawing AN-305,002. SSM manufactured the cladding from ingots identified with heat number 232180Q, supplied by Teledyne Wah Chang (TWC) of Albany, Oregon. TWC's information on the ingots included chemical analysis, product chemistry, Brinell hardness, grain size, ultrasonic tests, and eccentricity. SSM provided certified material test reports (CMTRs) on the cladding produced from the ingots provided by TWC. The NRC inspectors verified that the test results met the acceptance criteria of SNP Specification ANF-S35055.

3.8.3 Review of Records for Retaining Springs

The NRC inspectors reviewed the QA records for an order of retaining springs which are part of a bundle assembly. SNP issued PO R-71830 dated July 23, 1991, to Northwest Spring & Manufacturing (NSM), Lake Oswego, Oregon, to supply 1,480 retaining springs, Part Number 133699, to Drawing Number AN 305516, Revision 13. NSM purchased Inconel X-750 alloy wire with a copper coated finish, identified by heat No. 4198XK from National Standard Company which provided certifications indicating that all of the PO requirements had been met. Koon-Hall Company, an independent testing laboratory, provided the chemical analysis and physical properties of the Inconel alloy wire. Heat Treaters, Incorporated, of Portland, Oregon, provided certification that the wire was heat treated to meet NSM requirements. While conducting receipt inspections, SNP performed tensile tests on a retaining spring specimen and accepted the entire lot.

3.8.4 Review of Records for Tie Rod Adjusting Nuts

The NRC inspectors reviewed the QA records for an order of tie rod adjusting nuts which are used to assemble fuel bundles. SNP issued PO R-072282, dated September 18, 1991, to Wilson Tool & Manufacturing (Wilson), Spokane, to supply 2,450 tie rod adjusting nuts identified as Part Number 133620, to meet SNP's drawing AN 305945, Revision 1, and Specification S-35052. Wilson manufactured the tie rod adjusting nuts from 3/8-inch diameter stainless steel American Society for Testing and Materials (ASTM) A-276 rod and provided a copy of the Materials Suppliers Certification, the physical properties of the material, and test certificates that the material meets SNP's Specifications S-35011, Revision 5, for the stainless steel rod and S-35052, Revision 1, for the tie rod adjusting nuts.

The NRC inspectors determined that these records were acceptable. However, they did experience difficulties establishing the traceability of heat numbers provided by the ingot supplier to other subvendors who provided services, such as, independent testing or heat treating. Instead of the heat number, the subvendor documents referenced the SNP PO in the documents it issued to certify the operation it performed. The responsible SNP QC person assured the

NRC inspectors that he would remind both the SNP QC personnel, who review quality assurance records, and the subvendors to maintain the traceability of the material through its ingot heat number instead of through the SNP PO.

3.8.5 Review of Welder and Weld Procedure Qualifications

The NRC inspectors reviewed weld procedure qualifications and the methodology to verify that SNP permits only qualified welders to perform welding operations. Weld Procedure Qualification (WPQ) document EMF-PQ-572, Revision 1, of October 7, 1991, "Lower End Weld, 0.425" O.D., 0.028" Wall, K Station," which was being used to weld the lower end cap to the cladding, was the first procedure reviewed. This procedure provided the parameters, such as the preflow, the initial and final current slopes, and the initial weld, taper, and final currents to be used during the welding process. It also specified the size and type of electrode, the vertex-angle of the electrode tip, and the polarity of the electrode. It included a sketch of the joint design and electrode gap. SNP nondestructive examination personnel subjected the weld coupons, made by using this weld procedure, to visual, radiographic, and metallographic examinations, and determined the welds to be acceptable. The tensile strengths and weld build-up of the welded coupons were also determined to be acceptable.

The NRC inspectors also reviewed the WPQ qualification records for WPQ EMF-PQ-545, Revision 5, of October 28, 1991, "Upper End Weld, 0.425" O.D., 0.028" Wall, H&M Station," which was being used to weld the upper end cap to the fuel rods. The NRC inspectors determined that the qualification process was the same as for WPQ EMF-PQ-572 and that the records were complete and acceptable.

The NRC inspectors reviewed the methodology used to keep the qualification of the welders current and to prevent operators with expired qualifications from performing welding operations. SNP maintains a computerized list of qualified welders with a mechanism to void the qualification of individuals if they do not weld over a specific extended period of time. During this period, the welding machines will not acknowledge them as an authorized operator when they attempt to log in to weld. The NRC inspectors randomly verified the operability of the system and determined it to be acceptable.

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+ Attended Entrance Meeting on February 10, 1992
* Attended Exit Meeting on February 13, 1992

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

March 24, 1992

Docket Nos. 50-206, 50-361, and 50-362

Mr. Harold B. Ray
Senior Vice President
Southern California Edison Company
23 Parker Street
Irvine, California 92718

Dear Mr. Ray:

SUBJECT: INSPECTION OF THE PROCUREMENT AND COMMERCIAL GRADE DEDICATION PROGRAMS AT THE SAN ONOFRE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3, (REPORT NOS. 50-206/91-201, 50-361/91-201, AND 50-362/91-201)

This letter transmits the report of the inspection conducted December 9 through 13, 1991, at the San Onofre Nuclear Generating Station (SONGS), Units 1, 2, and 3, by R. P. McIntyre, S. D. Alexander, L. L. Campbell, and B. H. Rogers of the Nuclear Regulatory Commission's (NRC's) Vendor Inspection Branch (VIB) and W. J. Wagner of NRC Region V. The inspection was related to activities at the plant site authorized by NRC licenses DPR-13, NPF-10, and NPF-15. At the conclusion of the inspection, we discussed our findings with Mr. H. E. Morgan, Vice President and Site Manager, and the members of your staff identified in Section 5 of the enclosed inspection report.

The inspection was conducted to review the implementation of the Southern California Edison (SCE) programs for the procurement and dedication of commercial grade items used in safety-related applications at SONGS. The results of the inspection indicate that SONGS failed to properly dedicate certain commercial grade items (CGIs) procured for use in safety-related applications. Consequently, numerous CGI of indeterminate quality were installed or available for installation in safety-related plant systems. The specific deficiencies contributing to this condition included failure to identify safety functions of the CGI based upon appropriate design criteria and failure modes; failure to identify critical characteristics relating to the specific safety functions of the CGI; failure to adequately verify the critical characteristics that were identified on the "Procurement Engineering Package for Procurement Level V" (PEP5); and failure to identify and provide verification methods for the seismic qualification of most CGIs.

We recognize that SCE identified many of the program and implementation deficiencies in 1991, as evidenced by the scheduled implementation of revised procurement and dedication procedures in early 1992, and consider this self-identification a positive action. However, there is an apparent 2 year delay in upgrading your program to be in accordance with the Nuclear Management and Resources Council's (NUMARC's) first initiative on the dedication of commercial grade items. NUMARC's initiative stated that utility programs should meet the intent of the guidance provided in the Electric Power Research Institute (EPRI) NP-5652 Final Report, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)," by January 1, 1990.

The inspection findings presented to your representatives during the exit meeting at SONGS on December 13, 1991, and discussed in this letter and the enclosed report are considered deficiencies in your commercial grade procurement and dedication activities and will be referred to the NRC Region V office for any appropriate enforcement action. However, in view of the large number of items of indeterminate quality which were identified during this inspection, you should make a prompt assessment of potential safety implications of these deficiencies and take appropriate corrective actions based on your review of the information contained in this report. In this regard you may wish to review Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," with respect to NRC expectations for review of non-conforming items.

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

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

Sincerely,

Bruce A. Boger, Director
Division of Reactor Projects III, IV, V
Office of Nuclear Reactor Regulation

Enclosure: Inspection Report 50-206/91-201, 50-361/91-201
and 50-362/91-201

cc: See next page

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

Report No.:	50-206/91-201, 50-361/91-201, and 50-362/91-201
Docket No.:	50-206, 50-361, and 50-362
License No.:	DPR-13, NPF-10, and NPF-15
Licensee:	Southern California Edison Company 23 Parker Street Irvine, California 92718
Facility Name:	San Onofre Nuclear Generating Station, Units 1, 2, and 3
Inspection at:	San Clemente, California
Inspection Conducted:	December 9 through 13, 1991

Richard P McIntyre 03/10/92
Richard P. McIntyre, Team Leader Date
Vendor Inspection Branch (VIB)

Other Inspectors: S.D. Alexander, EQ and Test Engineer, VIB
L.L. Campbell, Reactor Engineer, VIB
B.H. Rogers, Reactor Engineer, VIB
W.J. Wagner, Reactor Engineer, RV

Approved by: Leif J. Norrholm 3/16/92
Leif J. Norrholm, Chief Date
Vendor Inspector Branch
Division of Reactor Inspection
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Office of Nuclear Reactor Regulation

TABLE OF CONTENTS

EXECUTIVE SUMMARY

From December 9 through December 13, 1991, representatives of the Nuclear Regulatory Commission's (NRC's) Vendor Inspection Branch and Region V inspected Southern California Edison Company's (SCE's) activities related to the procurement and dedication of commercial grade items (CGIs) used in safety-related applications at the San Onofre Nuclear Generating Station (SONGS), Units 1, 2, and 3. The inspection team reviewed SCE's procurement and dedication program to assess its compliance with the quality assurance (QA) requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50).

On August 24, 1990, the NRC staff forwarded to the Commission SECY-90-304, "NUMARC Intitiatives on Procurement," in which the staff reported the status of Nuclear Management and Resources Council's (NUMARC's) initiatives on general procurement practices. Procurement initiatives as described in NUMARC 90-13, "Nuclear Procurement Program Improvements," dated October 1990, committed licensees to assess their procurement programs and take specific action to strengthen inadequate programs. The initiative on the dedication of CGIs, which was supposed to be accomplished by January 1, 1990, stated that licensee programs should meet the intent of the guidance provided in the Electric Power Research Institute (EPRI) Final Report NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)," dated June 1988. The staff also stated in SECY-90-304 that it would conduct assessments at selected sites to review the licensees' implementation of improved procurement and commercial grade dedication programs, assess improvements made in the areas covered by the NUMARC initiatives, and report the results of those assessments to the Commission. From February to July 1991, the NRC's Vendor Inspection Branch conducted eight assessments of selected licensees to determine the current status of activities to improve the procurement programs related to industry initiatives and NRC requirements. On September 16, 1991, the NRC staff forwarded to the Commission SECY-91-291, "Status of NRC's Procurement Assessments and Resumption of Programmatic Inspection Activity," in which the staff reported on the results of its assessments and noted that it was resuming inspection and enforcement activities.

NRC conducted this inspection, the first since completing the eight assessments, to review SCE's procurement and dedication program and its implementation since January 1, 1990, the effective date of the NUMARC initiative on dedication of CGIs. The inspection focused on a review of procedures and representative records (including approximately 40 procurement and dedication packages for mechanical and electrical CGIs); interviews with SCE staff, including senior management and SONGS site personnel; and observations by the inspection team members. The inspection team also held meetings with SCE's management to discuss relevant aspects of commercial grade dedication and to identify areas requiring additional information. The inspection team's findings were discussed with SCE's representatives and senior management at the exit meeting held December 13, 1991. The inspection team's findings are identified as two deficiencies and are summarized below.

Deficiency 91-201-01

The inspection team identified numerous examples in which SCE either installed CGIs in safety-related plant applications or had identified them as available for installation in safety-related applications at SONGS without adequately reviewing the suitability of application of those materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components. These examples are discussed in detail in Section 3 of this inspection report. SCE failed to adequately dedicate and adequately conduct an oversight of vendors, which resulted in the use or warehousing of safety-related CGIs of indeterminate quality as indicated in the following representative examples:

(1) Mechanical Dedication Package (MDP)-04 for Procurement Engineering Package (PEP) K448, for safety injection pump shafts procured on purchase order (PO) 6Q041040 dated November 23, 1991. The item's safety function was not identified and the verification method for the critical characteristics identified was a source inspection. No guidance for verifying materials during the source inspection was given on the PEP and, consequently, material was not verified.

(2) MDP-08 for PEP Q849, impeller for a spent fuel pool pump was procured on PO 6N051028 dated May 30, 1991. The safety function of the item was not identified. Material was listed as one of the critical characteristics and was supposed to be verified during a source inspection. Material was not tested during receipt inspection or verified during the source inspection. When the impeller was recently sent out for testing to verify it was 304 stainless steel (SS), the preliminary results indicated it was out of specification for 304 SS.

(3) MDP-10 for PEP Q894, stem and plug assembly for the auxiliary feedwater (AFW) pump turbine throttle valve was procured on PO 6Q081048 dated August 22, 1991. Two critical characteristics identified were material and dimensions. Safety function and seismic qualification were not addressed. Material and dimensions were to be verified during a source inspection at the supplier. Four certified material test reports (CMTRs) were received from the supplier as evidence of material traceability. The source inspection did not review the QA controls at the supplier for maintaining material and its traceability, therefore did not establish the validity of the four CMTRs. The verification of the dimensions during the source inspection was poorly documented and did not indicate which dimensions were actually measured and with what tolerances.

(4) MDP-23 for PEP Z587/V168 (superseded Z587), for ASCO "Red Hat," (nonnuclear line) solenoid-operated valves (SOVs), parts, and rebuild kits for control operating air supply to loop C seal water return isolation valve PCV-1115C was procured on PO 6A120019 dated December 24, 1990. No safety functions were identified and critical characteristics were limited to part number, configuration, and operability. Acceptance was by standard receipt inspection/visual examination and operability

test. The operability test, as described in the specified verification test procedure, SO123-I-1.75, Attachment 1, Step 1.3.9, was: "Check solenoid operating temperature." There was no verification of operability in terms of ability to change and maintain state with no leaks at minimum and maximum voltage, minimum and maximum pressure/differential pressure, and at maximum operating temperature. The standard plant post-installation test (PIT), SO123-I-8.61, required only cycling several times and checking for leaks and correct function at nominal pressure and rated voltage.

(5) MDP-24 for Commodity List Item Evaluation (CLIE) 81-36, for oil and lubricants, was procured on PO 6D030003 dated March 6, 1990. The CLIE accepted the oil based on the label identification on the cans and drums and often on the supplier's certificate of conformance. Important characteristics such as viscosity, flash point, and additives were not identified or verified to meet safety function. No audits, surveys or overchecks were performed to verify this information. PEP V178 was written at a later date to include "oil, liquid mineral and synthetic base, lubricating and hydraulic, including fire resistant and petroleum based types." Revision 3 of PEP V178 was a significant improvement over the previous CLIE but still did not address critical characteristics such as flash point and additives and did not verify viscosity as a critical characteristic until March 1991. Oils and lubricants are used and have been used in many safety significant plant applications at SONGS.

(6) Electrical Dedication Package (EDP)-18 for PEP Q667 for 10 Sigma relays from Magnecraft Electric Company which were procured on PO 6N031003 dated March 21, 1991. The PEP failed to document consideration of the relay's safety function and failed to evaluate significant attributes such as insulation resistance, coil and contact voltage ratings, coil and contact current ratings, contact timing, and seismic capability as critical characteristics. Several of the relays have been installed in the plant for demineralizer inlet temperature monitoring.

(7) EDP-1 through EDP-9 were for General Electric (GE) molded-case circuit breakers (MCCBs) dedicated under PEP 82, Revision 0, for stock replenishment for various safety-related plant applications at SONGS, Units 1, 2, and 3. They were procured on PO 6A020002 dated December 24, 1990. EDP-10 (PEP 88) and EDP-11 (PEP 96) were for Simens-ITE (ITE) and Klixon MCCBs, respectively.

The PEPs did not consider all safety functions for the MCCBs and failed to identify appropriate critical characteristics for several of the safety functions which were identified. The verification methods and acceptance criteria for several identified critical characteristics were not adequately specified.

Deficiency 91-201-02

The inspection team identified several generic procurement program and implementation weaknesses which contributed to the specific examples of deficient CGI dedication described in Deficiency 91-201-01.

The most significant weakness concerned the failure of the program to require the identification of the safety functions of the CGIs being dedicated on the basis of the appropriate design criteria and failure modes. The dedication packages reviewed by the team failed to identify critical characteristics that related to the items' safety functions. The critical characteristics identified on the Procurement Engineering Package for Procurement Level V (PEP5) document were those "identifiable and measurable attributes/variables of a CGI, which once selected to be verified, provide reasonable assurance that the item received is the item specified." However, the NRC interprets the "item specified" to encompass attributes necessary for performance of the item's safety functions. The NRC staff's position is that Appendix B to 10 CFR Part 50 requires the licensee to demonstrate suitability of service for its particular plant application.

Another weakness was the failure to adequately verify those critical characteristics that were identified on the PEP5, including materials of construction for mechanical items such as a safety injection pump shaft, an impeller for a spent fuel pool pump, and packing and adjusting rings for the charging pumps. Also, important characteristics for oils and lubricants and several electrical characteristics for relays and MCCBs are examples of other critical characteristics that were not adequately verified.

Generic weaknesses within the dedication process included the failure to verify that the original seismic qualification for replacement electrical and mechanical items was still valid. This weakness is a direct result of reliance on the Supplier Deviation Request (SDR) process. If deviations from the ordered configuration (form, fit, function, and materials) were not identified by the vendor/supplier on the SDR (supplied by SONGS with the purchase order), then SONGS assumed that the item was identical to what may have been supplied previously and, therefore, the original seismic qualification was presumed to be maintained. SONGS had not performed commercial grade surveys at most of these vendors to verify that they, in fact, had the necessary controls in place to handle changes to design, manufacturing process, and materials. Also, many purchase orders were issued to distributors who would not be aware of design, manufacturing, or material changes made by the manufacturer.

Another generic weakness concerned specifying PITs as part of the verification for critical characteristics without ensuring that the PIT actually verified the identified critical characteristics. Most of these PITs are the routine tests to verify normal function of the item.

The last generic weakness concerned the inappropriate application of Mil-Std-105D for sampling during receipt inspection testing. Sampling during receipt inspection was implemented on an inconsistent basis without knowledge of lot/batch traceability or homogeneity.

Material Support Procedure SO123-XI-2, "Procurement Document Control," governing P/L III procurement ("Verification Method"), was stated to be applicable to items "governed by nuclear-unique codes, standards or programs," but for which the supplier does not have an SCE evaluated QA program in effect. The items or services in this category were to be accepted on the basis of SCE's own QA controls which were to ensure that "critical characteristics specified are verified to be present" through various methods including surveillances, inspections and tests. However, a weakness was noted here in that the procedure did not specify (or even reference) the pertinent 10 CFR Part 21 requirements, yet the description of P/L III clearly included basic components and, by its definition, excluded CGIs as defined in 10 CFR 21.3(a)(4)(a-1). In consulting associated procedure, SO123-XI-2.6, "Procurement Level III Evaluations," the team also found no reference to 10 CFR Part 21 requirements.

The team found that as a result of this programmatic weakness, several items which did not fully meet the definition of CGIs were procured under P/L III without specifying in the procurement documents that 10 CFR Part 21 applied.

1 INTRODUCTION

During this inspection, the NRC team reviewed the Southern California Edison (SCE) program and its implementation for the procurement of commercial grade items (CGIs) used in safety-related applications at the San Onofre Nuclear Generating Station (SONGS), Units 1, 2, and 3 (The SCE SONGS unit abbreviation convention: SO1, SO2, SO3, or SO12, SO23, SO13, or SO123 will also be used). The team also reviewed the SCE program and its implementation for determination or verification of suitability of those CGIs for their intended or approved safety-related applications, a process referred to as dedication.

Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) defines dedication as the point at which an item or service becomes a "basic component," that is, essentially an item (or service) with safety-related functions. However, the 10 CFR Part 21 definition of CGI (Section 21.3(a)(4)(a-1)), distinguishes them from items procured as basic components. The regulation, then, allows the procurement of items that are to become basic components, that meet the definition of CGIs without invoking 10 CFR Part 21 in the procurement documents.

When CGIs are procured for safety-related service, their procurement and dedication constitute activities affecting quality, and, therefore, these activities must be controlled in accordance with the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50). In particular, Criterion III, "Design Control," and Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B are most pertinent to procurement and dedication of CGIs; therefore, the team reviewed the SCE program governing these activities and the implementation of that program for compliance with these and other applicable Appendix B criteria as well as the requirements of 10 CFR Part 21.

Additionally, the NRC has provided further guidance on the requirements of Appendix B as they pertain to the procurement and dedication of CGIs in NRC Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," dated March 21, 1989, and GL 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs," dated April 9, 1991. Therefore, the SCE CGI procurement and dedication program and its implementation were also evaluated for consistency with the guidance and NRC staff positions promulgated in these GLs.

Finally, with respect to procurement in general, including procurement and dedication of CGIs, SCE committed to various industry standards and other publications (as endorsed or conditionally endorsed by NRC Regulatory Guides (RGs), NUREGs, and GLs), as stated in the SCE QA topical report, SCE-QA-1, "Quality Assurance Program Description," referenced in Section 17, "Quality Assurance," of the SCE Updated Safety Analysis Report (USAR) for SONGS, and as expressed for the industry by the Nuclear Management and Resources Council (NUMARC) in the NUMARC initiative on the dedication of CGIs as part of NUMARC 0-13, "Nuclear Procurement Program Improvements." In particular, SCE committed to have established a program for procurement and dedication of CGIs consistent with Electric Power Research Institute (EPRI) Final Report NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety

Related Applications (NCIG-07)," on or before January 1, 1990. The acceptance methods described in NP-5652 were conditionally endorsed by the NRC in GL 89-02 and the NRC staff positions on several dedication issues were later clarified in GL 91-05. Therefore, the team assessed the degree to which the SCE CGI procurement and dedication program, in effect since January 1990, and its implementation were consistent with the pertinent SCE commitments.

2 COMMERCIAL GRADE DEDICATION PROGRAM REVIEW

2.1 Procedures Review

The SCE program for procurement and dedication of CGIs for safety-related applications in SONGS is described and prescribed in a hierarchy of procedural documentation beginning at the SCE corporate level with the Nuclear Engineering, Safety & Licensing Department (NES&L) procedures. Nuclear Organization Units 1, 2, and 3 Material Support Procedure SO123-XI-2, "Procurement Document Control," now incorporates the SCE general guidance for virtually all SONGS procurement activities, including procurement of CGIs, and briefly addresses dedication which is covered in more detail in other site procedures. The team reviewed the currently effective revision of SO123-XI-2, Revision 3, with Temporary Change Notice (TCN) 3-4, dated July 3, 1991, and observed the following.

The team questioned the appropriateness of characterizing (under section 2.0 "References") 10 CFR Part 21, 10 CFR 50.49, and 10 CFR Part 50, Appendix B, (References 2.1.3-2.1.5) as "NRC Commitments" (Section 2.1) as opposed to NRC requirements. The "Topical Quality Assurance Manual" (TQAM), Chapter 3-A, "Procurement Document Development," was cited as Reference 2.1.6, yet also the procedure listed American National Standards Institute (ANSI) Standard N45.2.2 on packaging, shipping, receiving, storage, and handling, but did not list ANSI N45.2.13 on procurement or ANSI N45.2.11 on design control. Additionally, perinent regulatory guides, NUMARC documents (e.g., NUMARC 90-13), and EPRI reports (e.g., NP-5652 (NCIG-07)) were not listed as references; although, SCE's SONGS QA Topical Report, SCE-1-A, was listed in Attachment 2 to the procedure, "Developmental Resources," under "NRC Commitments."

Under Section 6.1, "Purchase Order Requisitions," (PORs) the procedure established the Procurement Engineering (PE) group as the authority to determine the quality class (or the so-called "quality-affecting program designation") of an item to be procured, and hence, whether the procurement will be done under QA program controls. For SO1, the quality classes are simply safety-related (SR) and non-safety-related (NSR), with quality-affecting subclasses NSR-RFP (fire protection-related) and NSR-ATWS (related to anticipated transients without scrams). For SO23, the safety-related quality classes (QC) are QC I (primary coolant pressure boundary) and QC II (other safety-related). QC III and IV include SO23 nonsafety, but quality-affecting, classes including fire protection and ATWS. Other SO123 categories of NSR, but quality-affecting items, included American Society of Mechanical Engineers (ASME) Code Section III and ASME Section XI Code Classified items and services, NSR, but environmentally qualified (EQ) items [as described in

10 CFR 50.49(b)(2) and (b)(3)], ASME Code Section VIII items, radioactive sources and standards, and radioactive material shipping containers. The SONGS unit Q-List contained the safety or QC designations for individual unit system components and most other quality-affecting equipment. In some cases, components and subcomponents and parts of SR/QC I and II systems and components have been reclassified/downgraded to NSR status through Component Classification Evaluation Documents (CCEDs). This process is discussed in more detail in Section 2.3 of this report.

According to the procedure, PE is to review safety-related/quality-affecting PORs for technical and quality requirements and prepare Procurement Engineering Packages (PEPs), which were to be used to document the QA and technical requirements for procuring and accepting quality-affecting items and services as well as for the critical characteristics evaluation and pertinent special instructions. As used by SONGS, PEPs were found essentially to be generic dedication documents. PEPs for some items were found to contain special receiving inspection instructions.

The team noted a strength in the program with respect to detection of fraudulent equipment in the requirements in Section 6.1.5 for screening all manual PORs (as opposed to pre-approved, computer-generated PORs) against the Control of Problem Equipment (COPE) List for disqualified or conditionally qualified items, ensuring that appropriate restrictions are incorporated for conditionally qualified items and that disqualified items are not used. PE also was required to screen Substitute Equivalency Evaluations (SEEs) against the COPE List.

The procedure required PE to consult various references in determining QA program application to procurement including the Q-List, EQ Master List, various design documents, previous procurement documents, the TQAM, and the Plant and Equipment Data Management System (PEDMS), and CCEDs. Parts level CCEDs were to be prepared as required if the quality classification of an item to be procured was to be different from that of its parent component. A significant weakness in the program was identified here in that part safety functions and failure modes and effects were not required to be documented when the item was classified as safety-related or quality-affecting. Therefore, this important information was not available for use in determining critical characteristics for the dedication of safety-related or quality-affecting plant items to be procured as CGIs.

Section 6.3 prescribed the process by which PE was to assign the Procurement Level (P/L) with its associated program controls according to the items' quality classification and procurement type or basis. A procedural interface strength noted here was the explicit invoking of the particular implementing procedure for the P/L assigned. During this inspection, the program in effect included three P/Ls, P/L II, III, and V, for quality-affecting procurement, to be processed in accordance with Material Support Procedures SO123-XI-1.9, -2.6, or -2.8, respectively.

P/L II (Section 6.3.1), the "Evaluated Supplier Method," was to be used for items and services (specifically including ASME Code Section III) to be procured from suppliers with an SCE evaluated QA program in effect, as listed in the SCE Evaluated Suppliers List (ESL). However, unlike the following sections on P/L III and V, this section did not specifically designate the quality classes to which it was applicable. Also, a weakness was noted here in that no reference to specifying the applicability of 10 CFR Part 21 in P/L II procurement documents (clearly intended for basic components) was found among the P/L II requirements. An additional concern was identified regarding P/L II procurement from other utilities/licensees (Paragraph 6.3.1.3.1.1). The item was required to have been originally purchased by the selling utility from a supplier "which had been evaluated by SCE Quality Assurance." However, there was no requirement to ensure that the supplier was fully qualified at the time of the item's manufacture or the original purchase, nor was there any requirement to evaluate the selling utility's QA controls while the item was in its possession.

Section 6.3.2, governing P/L III procurement ("Verification Method"), was stated to be applicable to items "governed by nuclear-unique codes, standards or programs," but for which the supplier does not have an SCE evaluated QA program in effect. The items or services in this category were to be accepted on the basis of SCE's own QA controls which were to ensure that "critical characteristics specified are verified to be present" through various methods including surveillances, inspections and tests. The procedure stated that ASME Code Section III and XI material may not be procured under P/L III, but that Code Classified items and Section XI services may be under certain circumstances. However, a weakness was noted here in that the procedure did not specify (or even reference) the pertinent 10 CFR Part 21 requirements, yet the description of P/L III clearly included basic components and, by its definition, excluded CGIs as defined in 10 CFR 21.3(a)(4)(a-1). In consulting the associated implementing procedure, SO123-XI-2.6, for P/L III, the team also found no reference to 10 CFR Part 21 requirements.

The team found that as a result of this programmatic weakness, several items which did not fully meet the definition of CGIs were procured under P/L III without specifying in the procurement documents that 10 CFR Part 21 applied.

Section 6.3.3, "Procurement Level V - Commercial Grade Item Method," deals with the area that was the major focus of this inspection. The procedure stated that P/L V was to be used for both safety-related CGIs as well as for NSR, quality-affecting items. However, SCE stated that they intended to separate these two categories, and place the NSR quality-affecting items in a new category, P/L VI, to be established in the near future. ASME Code Section III, XI, and Code Classified items were excluded from procurement through P/L V. The procedure called for an evaluation for CGIs that was to document the following: item description to preclude unauthorized substitution, conformance to the 10 CFR Part 21 CGI definition, critical characteristics, acceptance methods, engineering and QA approvals, and restrictions on use, manufacturer, and supplier, as necessary. However, there was no requirement to document the application specific requirements such as safety functions, failure m[illegible] effects or other suitability requirements, which must be identifi[illegible] order to adequately derive the critical characteristics.

While the P/L V specification requirements (Section 6.3.3.4) called for complete part/item description and citation of applicable standards, military specifications, etc., and Section 6.3.3.5 stated that a commercial survey of the supplier may be used as a basis for acceptance, there was no requirement for the suppliers to have a documented, effectively implemented commercial quality program, as established through a survey, or for verification of the distributor's controls, as well as those of the manufacturer, when a distributor is involved. Nor were the provisions of EPRI NP-5652 to invoke the supplier's approved commercial quality program and request a certificate of conformance (COC) to it required in the procedure. The acceptance methods given for P/L V were consistent with EPRI NP-5652 Methods 1, 2 and 3, that is, tests and inspections (standard receiving inspection and special tests and inspections were listed), commercial grade surveys, and source verification. EPRI Method 4, supplier product performance history, was not listed as an approved acceptance method for CGIs.

Section 6.6.1 dealt with the technical qualification of a supplier, which it stated was required for P/L II if the item was not a replacement in kind [presumably a substitute equivalency evaluation (SEE) or plant modification would be processed in this case]. However, it was not clear why a technical qualification was not required for P/L III (unless requested by the requisitioner) or for P/L V at all. Without an approved QA program the supplier providing a basic component under P/L III gets SCE QA coverage and presumably, critical characteristics (as used by the SCE procedure) are verified, but it would seem all the more important to ensure that the supplier is technically capable of controlling those critical characteristics. Why a technical qualification is not required for P/L V was also unclear, unless the commercial grade survey is intended to perform this function.

Section 6.7, "Critical Characteristics Evaluation," defined critical charac-terteristics for commercial grade dedication, P/L V, as "measurable attributes/variables of a commercial grade item, which once selected to be verified, provide reasonable assurance that the item received is the item specified." This definition is consistent with EPRI NP-5652, but is not conducive to compliance with the requirements in 10 CFR Part 50, Appendix B, particularly Criterion III and Criterion VII, to ensure that the item is suitable for its intended safety-related plant application. The NRC staff explained its position on this issue in GL 91-05.

Section 6.8 dealt with configuration reviews and item substitutions. However, configuration reviews were only required by the procedure for P/L II - to be done by the supplier, if the item was from the original equipment manufacturer or supplier (OEM or OES). This review was to be done by PE when the item is not from the OEM or OES. Configuration reviews and certifications of no changes in design, materials, or manufacturing processes from P/L III or P/L V suppliers was not addressed, yet those suppliers, by definition, have no approved QA program and may not have commercial design controls in effect. In response to this concern, SCE stated that they relied on the supplier to take exception to the no configuration change requirement in the PO. However, this supplier deviation request (SDR) process only addressed deviations from the description of the items as identified in the PO and did not specifically address changes to the items' design, material, and process that may have occurred. Also, the procedure did not explain how changes to form, fit, function, or material were to be evaluated. For instance, it did not mention the impact on seismic qualification; although, EQ was addressed in Paragraph 6.8.5.2.

SONGS Material Control Procedure SO123-XI-2.8, "Procurement Level V Evaluations," was the implementing procedure for the procurement and acceptance of CGIs for safety-related applications at SONGS. The team reviewed the currently effective revision of SO123-XI-2.8, Revision 0, with TCN 0-2, dated July 26, 1990, with the following observations:

Similar to SO123-XI-2, the implementing procedure did not address the identification, documentation, or use of safety function or failure modes and effects information in the derivation of critical characteristics, nor did it require that all critical characteristics be verified. The guidance on critical characteristic determination was limited to the same EPRI NP-5652 definition previously identified, with the addition of referring to the list in SO123-XI-2.6, "Procurement Level III Evaluations," of potential critical characteristics. In addition, there was no guidance in this procedure on establishing and maintaining documented verifiable traceability to the OEM in instances in which OEM information, such as destructive type testing, is relied on to support the dedication.

Attachment 1 to SO123-XI-2.8 did contain some strong guidance for verification of critical characteristics, once selected, to provide reasonable assurance that the item received is the item specified (PEP5 Keypoints, Section C, Item 14). Nevertheless, the lack of guidance for derivation of critical characteristics from safety functions, failure mode information, or other essential safety-related application suitability requirements no requirement to verify all critical characteristics once properly identified, contributed to numerous examples of CGIs that were inadequately dedicated.

In summary, the weaknesses identified in the SCE procurement and commercial grade dedication program and it's implementing procedures are cited as Deficiency 91-201-02 in the Executive Summary of this report.

2.2 Commercial Grade Supplier Surveys

SCE Quality Assurance Procedure (QAP) N18.16, "Commercial Survey," prescribed the methods of planning, conducting, and documenting commercial grade supplier surveys conducted under the auspices of the Supplier Quality Assessment Section (SQAS) of the Nuclear Oversight Division (NOD) of the SCE/SONGS QA organization for use in dedication of CGIs at SONGS. The team reviewed the currently effective revision of QAP N18.16, Revision 2, dated February 4, 1991, with the following observations:

SCE's procedure for commercial grade surveys was generally acceptable, and its survey team makeup requirements were a strength. It was ostensibly item and critical characteristic specific and factored requirements for supplier and product performance history data into survey planning. However, the team identified the following concerns: Section 3.0, "Responsibilities," required surveys to be conducted every 3 years for "active suppliers." This frequency may not be adequate depending on, but not necessarily limited to: (1) the complexity of the CGIs in question, (2) the frequency and size of purchases,

(3) the critical characteristics to be verified by survey and the extent to which those are relied upon to support dedication, (4) the strength of the supplier's controls on design, materials, manufacturing processes, and subsuppliers of parts and services, and (5) the strength of the supplier's commitment or obligation to either not make changes in certain products, or at least to inform the customer of any changes made.

The procedure called for identifying critical characteristics for the CGIs to be supplied, but Paragraph 4.1.1 stated that when many items are available from the same supplier, a representative set of items with critical characteristics that "envelope" the items the supplier is capable of supplying is acceptable. This would imply that the survey of such a supplier need only verify generically that the supplier has controls for all the critical characteristics associated with the CGIs in the supplier's product line of interest to SCE. However, while this provision may theoretically ensure that the supplier has controls for a given critical characteristic for some CGI it can produce, it does not necessarily ensure that particular critical characteristic is controlled for the CGI being procured and dedicated by SCE. Hence, it does not ensure verification of control of every critical characteristic (selected for verification by survey) of each CGI to be dedicated by SCE. The team was concerned that such a survey might verify that the supplier controls a given critical characteristic, but not necessarily for the CGI of interest. Although the procedure referenced NRC GL 89-02, it did not include provisions for surveys of distributors as well as manufacturers, where applicable, as discussed in GL 89-02.

2.2.1 <u>Third-Party Commercial Grade Supplier Surveys</u>

Section 4.1.2 of QAP N18.16 called for scheduling surveys when a previous survey by a third party is not acceptable. Paragraph 4.1.2(2) required that third-party surveys be evaluated in accordance with QAP N18.14, "Coordination of Audits/Surveys Performed by Contractors, Consultants, Utilities/Licensees, or Other Organizations Such as NUPIC or WUSAC." This would imply that SCE surveys, conducted under QAP N18.16, are only done when other adequate surveys are not available, and the provision would allow SCE to heavily rely on such surveys. However, this procedure did not specify any limits on how long such a survey could be considered valid before the intended procurement and did not contain any other guidance or acceptance criteria for such surveys. However, in the currently effective revision of N18.14, Revision 6, dated February 4, 1991, the team found the evaluation criteria to be general in nature and largely slanted toward broad-based Appendix B type QA audits of basic component suppliers. Although Section 4.1.2 of N18.14 did specify survey applicability to the same or similar items being procured by SCE, and from the same supplier surveyed, there were no requirements for (1) the third party survey to have verified that the supplier had documented, effectively implemented commercial quality controls, (2) that the specific critical characteristics selected by SCE for verification by survey were in fact verified and documented in the third party survey, and (3) that both distributor and manufacturer controls were verified where applicable. Neither did N18.16 nor N18.14 specifically require that third party surveys, to be acceptable, must as a minimum have met the requirements for an acceptable commercial grade supplier survey in accordance with N18.16.

To assess the effectiveness of the implementation of SCE's commercial grade survey program in support of dedication, the team reviewed a number of completed survey reports associated with some of the individual dedication packages reviewed. Surveys thus evaluated are discussed in Section 3 of this report in conjunction with the discussion of the associated dedication.

2.2.2 Source Verifications

SCE's CGI dedication procedures provided for acceptance of CGIs through source verifications (EPRI method 3). Accordingly, the team reviewed the SCE procedure governing this method, QAP N10.01, "Source Inspections." The team reviewed the currently effective revision of N10.01, Revision 18, dated September 9, 1991. This procedure provided acceptable guidance for the performance of source verifications for P/L III as well as P/L V procurement, and specified inclusion of critical characteristics given in the PEP in the inspection plan data report (IPDR). The only weakness was that the detailed instructions for the inspection report, Attachment 2, called for a "narrative summary of inspection activities," but did not specifically require that the particular critical characteristics be listed and their method of control and verification and results be documented to provide documented objective evidence of verification of those critical characteristics.

2.3 Parts Classification

The inspection team reviewed SCE Procedure NES&L 37-7-11, "Quality Classification of Components and Piece Parts," Revision 1, October 4, 1990, and discussed the methodology for parts classification with the PE Supervisor of Technical Evaluations. The methodology and criteria used to determine safety classification of parts includes identifying and documenting information such as the following in a component classification evaluation document (CCED):

- Parent component data such as system, P&ID number, technical description, applicable Updated Final Safety Analysis Report (UFSAR) sections, system descriptions and bill of materials,
- Component functional description, component safety function and classification, and
- Parts classification and safety function, if any, and the basis for that classification.

The inspection team performed a limited review of CCED No. 70068, "Units 2/3 Chill Water System 8" Containment Isolation Valves," Revision 1, November 12, 1991. These valves are located outside containment, are normally open, fail closed, and automatically close on a containment isolation actuation signal. The team reviewed 15 parts of the 8 inch Fischer Controls butterfly valve and determined that they were properly classified as safety-related. Included in this sample were parts critical for the valve to perform its safety-related function for containment isolation of the common inlet and outlet headers for the containment normal cooling units. Also included were those required for maintaining the integrity of the pressure boundary, integrity of the moving assembly that controls flow, seat tightness, integrity of the 1E power supply and deenergization to vent air from the top of valve diaphragm, and parts that provide force to automatically close the valve. The team also reviewed 5 parts classified as non-safety-related and found the classification of all 20 parts acceptable.

The inspection team discussed an observation with PE that Procedure NES&L 37-7-11, Revision 1, does not require failure modes and effects analysis to be included as part of the CCED process and the SCE commercial grade dedication program does not presently require performing and documenting this analysis.

Additional discussions with PE and review of SCE procedures revealed that items such as gaskets, O-rings, lubricants, and valve packing are not generically classified for use at SONGS, but are evaluated as they are used. However, these items may be procured as consumables and are subject to technical and quality requirements applicable for their intended end use.

2.4 Substitution Equivalency Evaluations

The inspection team reviewed SCE Procedure NES&L 37-26-18, "Substitution Equivalency Evaluations (SEE)," Revision 0, February 7, 1990, and discussed with the PE Supervisor of Technical Evaluations the methodology for determining if replacement parts and components not conforming to original or existing configurations are equivalent. The methodology and criteria used to determine if a substitute item is equivalent includes identifying and documenting information such as the following in a SEE package:

- Component reference data such as part name and identification, manufacturer, specification and codes for the item presently used, and the proposed substitute item,

- Applicable requirements such as changes in physical and performance characteristics, changes in environmental and seismic capabilities, and changes in codes/standards,

- An evaluation of the differences, and

- An equivalency justification, and as appropriate updates to documents authorizing the use of the substitute item.

The inspection team performed a review of the following SEE packages:

(1) SEE No. 91-0031, Revision 0, May 25, 1991, evaluated and authorized use of ASTM B-584 GR C93200, a leaded-bronze bearing alloy, as a substitute material for the Unit 1 charging pump seal assembly packing adjustment ring that is presently ASTM B-148 GR C95200, an aluminum-bronze alloy. According to the pump designer and manufacturer, APV Gaulin, the use of the leaded-bronze bearing alloy meets the original intent for this application according to APV Gaulin Manual VPL S023-928-18, "Inspection Manual for Gaulin Model NP 18-3TPS, Reciprocating Charging Pump."

Additionally PE identified that both materials have comparable wear and corrosion properties and that, although aluminum bronze has superior tensile strength to the leaded bronze, both materials are adequate for this nonstructural application. Based on the input received from the manufacturer of the charging pumps and the PE documented evaluation, the use of a leaded-bronze alloy for the adjustment ring for the charging pump appeared to be a reasonable substitute.

The inspection team selected SEE No. 91-0031, Revision 0, for review against its implementing PEP K597. Section 3 of this inspection report discusses PEP K597 in detail. PEP K597 did not identify any material requirements for the substitute charging pump adjustment ring and only required that the retainer ring be verified as a copper type alloy, which was accomplished by a visual color check. Based on a review of samples in the SCE test lab at SONGS, the inspection team found that the ASTM B-584, GR C93200 sample was not copper in color, but was a gold colored material. The SEE No. 91-0031 requirement that the adjustment ring material be ASTM B-584 GR C93200, a leaded-bronze bearing alloy, was not correctly translated into PEP K597 and may have resulted in indeterminate material being used for the charging pump adjustment rings.

(2) SEE 91-0042, Revision 0, June 3, 1991, evaluated and authorized the use of a knurled pin/A479 Type 316 for the grooved lock pin used in the inner valve stem assembly that keeps the valve stem and inner assembly from unscrewing. This pin was a grooved pin/A15I 300 Series. These valves are used in several systems at SONGS. Supplier Deviation Request (SDR) No. 157E is included as part of the SEE package and indicates that since PE has imposed 10 CFR Part 21 and requires a CMTR, the pin will have to be a knurled pin of A479 Type 316 material. Based on input received from the valve manufacturers, Anchor/Darling Valve Co., and the PE documented evaluation, the use of the knurled pin appeared to be a reasonable substitute.

2.5 Quality Class Upgrades

The inspection team reviewed SCE Material Control Procedure SO123-XI-1.4, "Upgrading an Item's Quality Class," Revision 1, with Temporary Change Notice 1-1, April 29, 1991, and discussed the upgrade process with PE. The methodology and criteria used to determine if an item can be upgraded to a safety-related quality class from non-safety-related includes identifying information such as the system, quality class, design specifications, Codes, safety-related function, critical characteristics, and acceptance basis, including the verification methods. This information is documented in a stock upgrade requirements evaluation (SURE) package. Six SURE packages have been prepared since January 1990.

The inspection team reviewed SURE No. 91B02, February, 1991, which upgraded a silicon rectifier, Type SK 3051, originally purchased as a general purpose item by station maintenance. The silicon rectifier was upgraded for use in a safety-related application that provides positive 125 V dc to relays 194 and 194-1 in SONGS Unit 1. Critical characteristics for the silicon rectifier included the manufacturer's part number, configuration (general size and shape), and electrical characteristics such as measurement of forward voltage, reverse current leakage, break-down voltage, and continuous dc current. The SURE required that only one of the four silicon rectifiers be tested to verify that the critical characteristics were present. However, the SURE contained no basis for sampling the four silicon rectifiers, and Section V of the SURE required that the "test component shall be rejected and destroyed after the completion of the test." With the exception of break-down voltage, the testing performed should not destroy the silicon rectifier because the other tests only confirm that the rectifier will function within the performance parameters specified by the manufacturer. The inspection team concluded that with the exception of having no knowledge of homogeneity as a basis for sampling, the critical characteristics and verification methods identified, if performed on each silicon rectifier (with the exception of breakdown-voltage), should provide an adequate basis for upgrading.

2.6 Trending of Suppliers

The tracking and trending of deficiencies related to supplier performance is a requirement of Quality Assurance Procedure (QAP) Number N2.07, entitled "Reporting of Quality Trends." Section 2.4 of this QAP assigns this responsibility to the Supplier Quality Assessment Supervisor (SQAS). No formal procedures at this time described how the SQAS is complying with QAP No. N2.07. Through discussions with the SQAS and QA personnel, the team was able to review and evaluate the effectiveness of the trending program.

Initially, the trended items only included problems identified in audit reports but has since evolved to include pertinent information from the following documents for potential procurement related deficiencies: Corrective Action Requests (CARs), Licensee Event Reports (LERs), surveillances, Problem Reports (PRs), Nonconformance Reports (NCRs), Supplier Deviation Requests (SDRs), and Notices of Violation. Trend codes are assigned to supplier related deficiencies which, when warranted, have resulted in the issuance of a CAR or a meeting with the supplier to discuss their decline in performance. The trending program appears to have been a successful effort which is evident by the reduction of open warehouse NCRs.

3 DEDICATION PACKAGE REVIEW

To facilitate the NRC review of individual dedications, SCE prepared a number of dedication record review packages, compiled from diverse records, but each pertaining to one PEP, as selected by the team from a review of various Level V PEP tracking reports. SCE called these files Electrical Dedication Packages (EDPs) (also covering instrumentation and control equipment) and Mechanical Dedication Packages (EDPs) (also covering materials procurement). The following examples are items that were purchased commercial grade and either installed or available for installation in safety-related plant applications without performance of an adequate review for suitability for service.

(1) EDP-1 through EDP-9 were for General Electric (GE) molded-case circuit breakers (MCCBs) dedicated under PEP 82, Revision 0, dated November 29,

1989, and purchased from GE Supply Company (GESCO), El Monte, California, on SCE PO 6A020009, dated February 24, 1990. SCE purchased these MCCBs for stock replenishment (primarily as a result of NRC Bulletin 88-10 replacements) for various safety-related plant applications at SONGS, Units 1, 2, and 3. EDP-10 (PEP 88) and EDP-11 (PEP 96) were for Siemens-ITE (ITE), and Klixon MCCBs respectively. None of these dedicated commercial grade MCCBs had been installed in safety-related applications at the time of the inspection. The team reviewed the above captioned packages with the following observations:

The safety functions for the MCCBs in their plant applications were not all documented. Safety functions for these MCCBs that were not documented included performing the following (under all operational and design basis conditions): (1) reliably providing and maintaining power to Class 1E loads absent designed power interruption conditions, (2) performing circuit protection functions, i.e., tripping, under designed power interruption conditions, (3) tripping in response to remote actuation (undervoltage or shunt trip devices), and (4) not allowing manual shutting under trip conditions (the so-called "trip free" action). Also, no analysis was documented for credible failure modes detrimental to safety.

The critical characteristics listed in these PEPs did not cover all the listed safety functions. Missing were (1) interrupting capacity for available short circuit current at the highest expected service voltage, (2) trip-free mechanical action, (3) insulation resistance/dielectric strength (this was supposed to be measured by Megger according to the post installation test (PIT) procedure, but was not called for in the PEP), and (4) seismic capability. The critical characteristics not listed that would be necessary to ensure satisfactory performance of the safety functions not listed would include (1) start-up operability (this may be verified by PIT or operational testing), (2) response to momentary voltage and load fluctuations after equilibrium, full-load operation (PIT), (3) full-load hold-in capability, (4) not tripping below specified voltages/currents/times, (5) undervoltage and shunt trip device operability under all expected variations of voltage and other design conditions, and (6) individual pole resistance.

In some cases the verification of the listed critical characteristics was left up to the judgement of plant electricians who do not routinely perform these tests. In particular, the procedure for the instantaneous-magnetic (I-M) trip function testing was ambiguous with respect to acceptance criteria. The PIT procedure, SO123-I-4.7, had a note that stated that trip point data (other than manufacturer's time-current trip curves) "may be provided by engineering," but the procedure did not explain how to obtain the data or how to use it. Adjustable I-M trip setting data would be needed to properly test this function on MCCBs (except those with motor loads for which the procedure provided specific guidance). In addition, the instructions for measuring insulation resistance did not specify what Megger voltage to use, and did not specify a high potential (HIPOT) or dielectric withstand test as an alternative.

The team noted that a new draft PEP for MCCBs required that seismic qualification was to be verified by special receiving inspection instructions, which were unclear. SCE explained that MCCBs are assumed to be seismically qualified if they are the same model number as a previously qualified MCCB. How the original qualification was established, and how similarity or traceability to the qualification documents was to be established, was not clear.

In summary, not all safety functions for the MCCB dedications reviewed were identified. For those safety functions that were identified, not all critical characteristics were identified or selected for verification. And finally, for those critical characteristics selected for verification, not all were verified adequately.

(2) EDP-14 for PEP K433, dated February 11, 1991, was for a 15-Vdc power supply for excore safety channels procured under PO 6Q041002 to Sorrento Electronics and received on Receiving Inspection Data Report (RIDR) RSO-3064-91, dated February 4, 1991. Dedication consisted of an operational check per PIT procedure S023-II-5.2 at nominal voltage. Not verified was the ability of the power supply to deliver required voltage/current under (1) all design load steady state and transient conditions, (2) all expected input voltage and frequency variations, (3) worst case temperature and/or (4) design seismic conditions, and (5) quality of output power in terms of noise and/or ripple.

(3) EDP-24 for PEP V188, dated June 13, 1991, was for coils from the Aktomatic Valve Company. The dedication did not identify safety functions. Missing critical characteristics included verification of temperature ratings/capability, insulation resistance, and winding resistance. There was no test of magnetic force developed by the coil in a valve under all design/operational conditions. Manufacturer's data was attached, but there were no engineering specifications on its use and no objective evidence of traceability.

(4) EDP-15 for PEP K598, dated June 05, 1991, was a generic PEP for key interlock switches from Microswitch Inc. The PEP listed no safety functions. Critical characteristics listed were only that (1) the switch be key operated, (2) the number of positions be specified, and (3) physical inspection per VTP-E-061 with no defects. The acceptance basis was standard receipt inspection for (1), and special receipt inspection for (2) and (3). The RIDR/VTP-E-061 listed safety function and some critical characteristics, however, there were no electrical tests for critical characteristic 6.2 and no seismic consideration.

(5) EDP-27 for PEP W271, dated June 14, 1991, was for a Foxboro contact unit, type 70, M0129FM. Purchased on PO 6J061027, received on RIDR RSO-1642-91, installed on MO 91040794001 for auxiliary cooling system-spent fuel pit temperature unit TAG No. S1-SFP-TIC-615. No safety functions were identified and the critical characteristics consisted of part number, dimensions, and operability, which was to be verified per system calibration by PIT S0123-II-9.123. There were no seismic

considerations, no test for insulation resistance, and no contact resistance/signal continuity tests specified.

(6) EDP-29 for PEP X561, dated January 1, 1991, was for American Insulated Wire (AIW) Co. 4/C-8 AWG, 2000-VAC, 90 degree C power cable and tinned copper, 3/C-12 AWG, 600-VAC, control cable for the manipulator crane festoon. The cable was purchased on PO 6G011003, dated January 14, 1991, from United Constructors & Engineers, received on RIDR RS0-0121-91, and installed per MO 90080864003. The PEP had not identified safety functions. Critical characteristics not specified included continuity /conductor resistivity, insulation material, physical properties (pull test), and dielectric strength. The PIT only included Meggering, continuity and resistance checks. It was not clear how the other critical characteristics were verified.

(7) EDP-16 for PEP K629, dated September 7, 1991, was for a Yokogawa "180-type" 55-Hz to 65-Hz, frequency meter (180-degree panel meter). The dedication documented no safety functions nor was there an analysis for failure modes detrimental to safety. Critical characteristics included were; part number (standard receipt inspection), range scale as specified (special receipt inspection instructions), and satisfactory calibration (PIT). The critical characteristics not listed/verified included seismic qualification, and the characteristics of any attached accessories, for example, transducers, etc.

(8) MDP-04 for PEP K498, dated February 28, 1991, was reviewed. SONGS PO 6Q041040 with Change Order 01, dated November 23, 1991, was issued to Dresser Industries, Inc. for the purchase of SONGS Unit 1 safety injection pump shafts. The shafts were Worthington Pump parts manufactured in Harrison, NJ. The safety function of the shaft was not identified in the PEP. The critical characteristics listed on the PEP for the shaft were: (1) part number as specified in the PO, (2) configuration (as a minimum the overall length, diameter(s) and end thread connections), and (3) materials of construction. The verification methods for accepting each critical characteristic were listed on the PEP as: (1) above is verified as part of standard receiving inspection, (2) and (3) above shall be verified by the SCE source inspector using manufacturer's drawing(s) and material records provided at the supplier's Harrison, NJ facility. The Witness/Hold Points (Section 5K) of the PEP identified that information concerning the source inspection would be provided to the manufacturer of the shaft. The inspection team identified the following deficiencies in the procurement and dedication of the shaft:

- There was no audit or survey of the manufacturer to support acceptance of the verification of the shaft's material of construction by reviewing the manufacturer's drawings and material records.

- The initial PO for the pump shaft, dated April 23, 1991, listed a technical description of the shaft as "Shaft, Pump Type, 5-1/2" DIA x 69" LG Size, Monel Material, Both Ends Thd Construction For Model

12 LNS-38 Pump." Change Order 01 to the PO revised the material description for the shaft to read "Shaft, Pump Type, Monel Material, Both Ends Thd Construction For Model 12 LNS-38 Pump." PE informed the inspection team that the 5-1/2" DIA x 69" LG dimensions were determined to be incorrect based on a review of the manufacturer's drawing, which required the shaft to be 4.999" DIA x 68-9/16" LG. The inspection team could not determine the technical basis for the initial identification of the 5-1/2" DIA x 69" LG shaft dimensions, the technical basis for deleting these dimensions from procurement documents, or the basis for accepting the manufacturer's dimensions of 4.999" DIA x 68-9/16" LG.

- PE informed the inspection team that probably several item descriptions in the data base used to procure items may be incorrect because this information was gathered and entered into the data base by non-quality and technical personnel and that this information collected and entered has never been validated as being correct by engineering or quality assurance. The material data base is used to identify technical requirements in procurement documents and contains invalidated data. PE informed the inspection team that personnel preparing procurement documents are aware of the potential use of invalidated information in the material data base, however, there were no procedural requirements identified to the inspection team that address this condition.

(9) MDP-05 for PEP K596 and PEP K597, both dated June 5, 1991, was reviewed. SONGS PO 6E01104 with Change Order 1 and Change Order 2, dated June 13, 1991 was issued to APV Gaulin, Inc. for the purchase of packing adjustment rings and packing for the Unit 1 charging pumps, Model 2640-NP-18-3TPS. The safety functions of the packing adjustment ring (PEP K596) and packing (PEP K597) were not identified in the PEPs. The critical characteristics listed on the PEP for the packing adjustment ring were: (1) part/catalog number as specified, (2) dimensions (ID, OD, height) and (3) material is a non-ferrous copper alloy and for the packing were (1) Part/Catalog number as specified, (2) dimension (Nominal ID and OD) and (3) material is a nylon type.

The verification methods for accepting each of the critical characteristics for the packing adjustment ring were listed on the PEP as: (1) is verified as part of standard receiving inspection and (2) & (3) above are verified as special receiving inspections. These special receiving inspection requirements included "material is a non-ferrous copper type alloy." Neither the PO, PEP or RIDR No. RSO-1861-91 identified the actual material required for the packing adjustment ring. Further review of engineering documents such as manufacturer correspondence and SEE 91-0031 (See Section 2.4 of this inspection report) revealed that the packing adjustment ring was ASTM B584, GR C93200, a leaded-bronze bearing alloy. SEE 91-0031 addressed the wear and corrosion properties of ASTM B584, GR C93200 as being acceptable for use in the Unit 1 charging pumps. Based on a review of technical documents supporting the purchase and dedication of the packing adjustment ring and discussions with PE and Receiving Quality Control (RQC) receiving inspection personnel, the inspection team determined that the packing adjustment ring was accepted based on a visual examination. The inspection team identified the following deficiencies in the procurement and dedication of the packing adjustment ring:

- The specific material type and analysis of acceptability identified in engineering documents were not translated into requirement on the PEP, PO, or RIDR for the materials of construction for the packing adjustment ring.

- The PEP and RIDR did not provide direction to RQC on how to verify the material of construction.

- The practice of performing a visual inspection to determine material type when no specified material tests are listed on the RIDR is not sufficient in most instances to verify materials of construction. Copper alloy materials include material that may be gold, silver, copper or other colors. As implemented and documented by RQC, the visual inspection does not provide objective evidence as required by Section 7.3.2, Receiving Inspection, of ANSI N45.2.13 to substantiate that the material is ASTM B584, GR93200.

The verification methods for accepting each critical characteristic listed on the PEP for the packing were essentially the same as for the packing adjustment ring except that the special receiving inspection requirements on RIDR No. RSO-1862-91 required the material to be a nylon type. Research of documents by the inspection team revealed that the pump manufacturer identified the specific material for the packing as "Nylon, C'Duct, Nitrile and TFE." The inspection team identified the following deficiencies in the dedication of the packing:

- The specific material (type of nylon) was not identified on the PEP, PO or RIDR for the materials of construction for the packing. The PEP and RIDR did not provide direction to RQC on how to verify material of construction.

- As implemented and documented by RQC, the visual inspection did not provide objective evidence that the packing was the required material of construction.

(10) MDP-08 for PEP Q849, dated December 12, 1990, was reviewed. PO No. 6N051028, dated May 30, 1991, was issued to Pump Engineering Company for the purchase of an 11 $^5/_8$ inch diameter SS impeller for a Worthington centrifugal spent fuel pit pump. The safety function of the impeller was not identified. The critical characteristics listed on the PEP were dimensions, material, and balance. Acceptance basis for these characteristics was to be accomplished during a source inspection at Post-Precision Casting Inc., a subtier supplier to Worthington-Dresser. A source inspection done at Pump Engineering, not Post-Precision Casting verified dimensions and balancing, but noted on the Inspection Planning

Data Report (IPDR) No. Pump-S1-91 that material was to be verified at SONGS receipt inspection by spectro test. RIDR No. RSO-1804-91, dated August 22, 1991, only verified that materials conform to the manufacturer's drawing. The spectro test was never conducted during receipt inspection to verify material composition. The material was later sent out for testing and the preliminary results indicated it was out of specification for 304 stainless steel. Little guidance was given on the PEP for material verification outside the statement, "In lieu of testing, material verification may be performed through review of supplier's documentation for control of materials." This type of activity would normally be performed during a commercial grade survey and not a source inspection. No survey was performed at either Pump Engineering or Post-Precision Casting. Finally, the impeller material was never verified, but was available for plant installation in the nonconforming condition.

(11) MDP-10 for PEP Q894, dated November 20, 1990, was reviewed. PO No. 6Q081048 dated August 22, 1991, was issued to Masoneillan-Dresser Industries (M-D) for the purchase of a stem and plug assembly for the Unit 1 AFW pump turbine control valve, as well as other spare parts for the Worthington single stage pump. Critical characteristics identified for the stem and plug assembly included dimensions and materials. Safety function and validation of the seismic qualification of the component were not identified or verified. Material and dimensions were supposed to be verified during a source inspection at the supplier. The source inspection, as documented on IPDR MADI-S1-91, did not review the QA controls in place at the supplier, M-D, for maintaining materials and their traceability. Therefore, the validity of the certified material test reports (CMTRs) was not established. Also, the verification of the various dimensions during the source inspection was poorly documented and did not indicate which dimension or tolerance was actually measured. The stem and plug assembly material was never adequately verified, but the assembly was warehoused and available for installation.

(12) MPD-19 for PEP Y500, Revision 1, dated October 2, 1989, and Revision 2, dated April 11, 1990, was reviewed. SONGS PO 6D050022 with Change Order 01 dated, November 15, 1990, was issued to Lamons Metal Gasket Company for the purchase of flange fitting type gaskets (6" pipe size, 2500 pound rating and 1-1/2" pipe size, 150 pound rating) used in various applications at SONGS. The safety function of the gaskets was not identified in the PEP. The critical characteristics listed on Revision 2 of PEP Y500 were: (1) gasket inside and outside diameters, (2) workmanship such as spot welding, scratches, tears, and plating, and (3) markings, "gaskets must be properly marked to indicate size (or size range) pressure class (or pressure class range), API-601, Manufacturer's Mark and color coding (see page 3)." The verification methods for accepting each critical characteristic listed on PEP Y500, Revision 2, referred to the RIDR on page 5 which required RQC to verify the following by sampling per Mil-Std-105D, Table 2, AQL 2.5%, (1) inside and outside diameter listed in a table on page 4 of the PEP, (2) workmanship, such as: "Spot welding of the inner and outer windings; Scratches or tears across the surfaces of the windings; Plating quality of carbon steel gauge ring (if utilized), and (3) markings, "gaskets must be properly

marked to indicate size (or range) pressure class (or pressure class range), API 601, Manufacturer Mark and color coding on the outer gauge ring (see page 3)." The inspection team identified the following deficiencies in the procurement and dedication of the gaskets:

- Materials of Construction was not listed as a critical characteristic. Verification of part number identification and color coding by the supplier does not verify that the gaskets were the proper material. Discussion with PE revealed that there was no documented audit or survey of Lamons Metal Gasket Company supporting acceptance of the part number identification and color coding as the basis for assuring that the required gasket material was present.

- The use of sampling for all three verifications listed on the RIDR is not supported by a survey or audit to support homogeneity of the lots received.

  The gaskets were received and accepted by RQC receiving inspection in February 1991, using Revision 1 of PEP Y500. During the review of the PEP and subsequent discussion with PE, the inspection team questioned the use of Revision 1 of PEP Y500 by RQC. PE informed the inspection team that Revision 2 of PEP Y500 should have been used to accept the gaskets. The major difference between the two PEP revisions was that Revision 2 permitted the use of sampling. There was no objective evidence entered on RIDR RSO-0357-91 that RQC had measured and accepted the outside and inside diameters of the gaskets.

(13) MDP-21 for PEP Z316, dated November 16, 1990, was reviewed. SONGS PO 6Q011018 with Change Order 01, dated February 12, 1991, was issued to San Diego Seal Company for the purchase of valve packing used in various applications at SONGS, including the letdown isolation valve. The safety function of the packing was not listed on the PEP. The critical characteristics listed on the PEP for the packing were part number and configuration (Arrangement) and the method of acceptance for the packing was identified as a certificate of conformance from the supplier based on a survey of A.W. Chesterton Co. the manufacturer of the packing. PE provided the inspection team a survey of the supplier, San Diego Seal Company, and the manufacturer, A.W. Chesterton Co. The inspection team identified the following deficiencies in the procurement and dedication of the packing:

- Materials of construction, including the control of water leachable chlorides, was not listed as a critical characteristic. Specification SO23-408-1, "Quality Class I, II, and III Specification for Nuclear Service Valves for San Onofre Nuclear Generating Station, Units 2 & 3," dated July 27, 1973, was provided to the inspection team in order to review the requirements for packing installed in nuclear valves at SONGS. Section 4.08.5, Packing, of Specification SO23-408-1 requires that "Packing shall contain a corrosion inhibitor to prevent stem pitting. Low chloride (200 PPM max) packing shall be used for stainless steel valves." Neither the PEP, PO, or the RIDR addressed this packing requirement.

However, a COC from A.W. Chesterton Co. was included in the RIDR documentation package certifying that the amount of water leachable chlorides for the supplied packing was less than 200 parts per million.

- Problem Review Report (PRR) No. SO-109-91, dated October 17, 1991, identified 11 POs issued to A.W. Chesterton Company without the supplier's qualified location, QA program title, revisions, and date incorporated into the PO. PO 6Q011018 procured packing manufactured by A.W. Chesterton Company and supplied by San Diego Seal Company, but did not identify either company's QA program title, revision, or date. PO 6Q011018 was not listed on PRR SO-109-91.

(14) MDP-23 for PEP V168, dated February 21, 1991, (superceded PEP Z587) for ASCO "Red Hat," (non NP-1) SOVs, parts, and rebuild kits purchased under PO 6A120019, dated December 24, 1990, and received on RIDR RSO-0910-91, Revision 1. One SOV was installed on MO 91043008(001) for control of operating air supply to loop C seal water return isolation valve (PCV-1115C). The dedication listed no safety functions. Critical characteristics consisted of part number, configuration, and operability. Acceptance was by standard receipt inspection/visual examination and operability test. The operability test, as described in the specified verification test procedure, S0123-I-1.75, Attachment 1, Step 1.3.9, was "Check solenoid operating temperature." Critical characteristics not separately listed, but supposedly implied by part number would be coil voltage rating, temperature rating, insulation class/temperature rating of coil magnet wire and lead insulation. Other missing critical characteristics would include coil resistance, coil/lead insulation resistance, dimensions, elastomer material, durometer of elastomers, spring free length and constant. No verification of operability in terms of ability to change and maintain state with no leaks at minimum and maximum voltage, minimum and maximum pressure/differential pressure, and at maximum operating temperature were evident. The standard plant PIT (not called out in the PEP Z587 or RIDR, but was later used in PEP V168), S0123-I-8.61, Step 6.8.1.1, after installation, required cycling several times and checking only for leaks and correct function at nominal pressure and voltage.

(15) MDP-24, Commodity List Item Evaluation (CLIE) 81-36, Revision 5, dated November 9, 1988, was reviewed. SONGS PO 6D030003, dated March 6, 1990, was issued to Chevron USA, Inc. for the purchase of motor oil, Delo 6170, in 55 gallon drums, supplied in sealed containers with a manufacturer's identification label SAE 40 for use in the diesel engines. The critical characteristics listed on the CLIE for the oil were "must be a high alkaline blend of paraffinic and naphthionic base oils. Contains additives to provide high detergency and dispersancy." The CLIE also listed critical defects as "Any change in viscosity, contamination by foreign matter and impurities, and any infringement of container seal to the extent that contamination of the lubricant could be possible." The verification methods for accepting the oil listed on the CLIE included a standard receiving inspection and "for cans and drums confirm that the manufacturer's label identification on the containers agrees with purchase order documents. Examine and verify that container seal

integrity has not been breached. For bulk shipments, supplier shall provide certification that the load is Delo 6170." The inspection team identified the following deficiencies in the procurement and dedication of the oil:

- Viscosity was not identified as a critical characteristic for acceptance, however, viscosity is the single most important characteristic of oil in determining its lubricating properties.

- Flash point was not identified as a critical characteristic, however, flash point is significant in assuring the oil's useful temperature range. The additive package could also be critical to the ability of the oil to perform under operating conditions. Due to insufficient information contained in the CLIE, such as failure modes and the safety function of the oil, the inspection team determined that insufficient information was available to properly identify all the oil's critical characteristics.

- The CLIE accepted the oil based on the label identification on the cans and drums agreeing with that required by the PO, and under certain conditions a supplier's certification that the oil load was Delo 6170. There were no audits, surveys, or overchecks performed by SCE to support the use of these acceptance methods.

PE presented PEP V178, Revision 3, dated November 5, 1991, to the inspection team for review. PEP V178 scope includes "oil, liquid mineral and synthetic base, lubricating and hydraulic. Includes fire resistant and petroleum based types." Grease, dry lubricant, and special colloidal mixtures were excluded. Revision 3 of PEP V178 was considered a significant improvement over CLIE 81-36, Revision 5, however the inspection team identified the following deficiencies in PEP V178:

- PEP V178 dedicates oil that is used for lubricating and hydraulic applications. These applications each have a unique safety function. Hydraulic fluid (oil) is classified as safety related because it performs the safety-related function of transmitting force to modulate an item. Failure modes, functions, and critical characteristics for lubricants and hydraulic fluids vary depending on application. PE informed the inspection team that they were considering developing a PEP specifically for hydraulic fluid.

- Before March 1991, the viscosity of oil was not verified. General safety-related components, such as the motor-driven auxiliary feedwater pump have calculations to support operability under accident conditions that specifically take credit for the viscosity properties of oil to prevent pump bearing heatup. The use of lubricants with specific properties is essential to ensure equipment operability. Identification and verification of critical characteristics such as flash point and additives were not addressed in the PEP.

- Becauses this item (oil and lubricants) could impact environmental qualification (EQ) in applications, EQ should have been addressed.

* PEP V178, Revision 3, permits sampling, however, there are no requirements for an audit or survey of the manufacturer to support homogeneity of lots/batches received.

(16) The team reviewed the SCE RQC test lab activities. During this review, cap screws (5/8" diameter x 1-3/4" long, socket head, STL Material, 11 UNC CL 3A threads, ASTM A574) were being inspected and dedicated for use in various safety-related applications at SONGS. RQC was performing inspections in accordance with the requirements of PEP V133, Revision 3, dated April 26, 1991, and PO 6Q81072 with Change Order 01, dated October 12, 1991. The safety function of the cap screws was not listed on the PEP. The critical characteristics listed on the PEP for the cap screws were: (1) dimensions (thread, pitch diameter, body diameter, length, width across flats and head height, as applicable, (2) material, and (3) configuration (physical arrangement as depicted in ANSI B18.3). The verification methods for accepting each critical characteristic for the cap screws were listed on the PEP as, "(1), (2) & (3) shall be verified equal to purchase order and ANSI B18.3 requirements. (1) Tolerances provided in ANSI B18.3 apply. (2) Material shall be verified by Test No. 1 on PG.6. When Test No. 1 is not Feasible Test No. 6 shall be used. When plated material is evaluated the plating shall be removed for verification of base metal. (1), (2) & (3) shall be verified on 25 percent of samples required by Mil-Std-105E, Level S-4, AQL per the table on Page 5, from each line item lot." The inspection team identified the following deficiencies in the procurement and dedication of the cap screws:

* Material Test No. 1 of PEP V133 is a spectro-analysis, and Material Test No. 6 is an alloy separator test. These tests alone will not verify that the material of the cap screws is ASTM 574. For example, A-574 screws are required to be heat treated with a 37 to 45 HRC for 0.625 inches and larger. Also A-574 identifies proof loads and tensile requirements for the screws as well as chemical requirements. The two tests specified do not confirm that the material is A-574.

* The use of sampling to verify all critical characteristics is not supported by a survey or audit to support homogeneity of the cap screw configuration or material of the lots received. The sample size of 25 percent less than that required by Mil-Std-105E is not justified.

(17) Six PEPs for relays purchased commercial grade and intended for or installed in safety-related applications, K371 (EDP-12), K541 (EDP-13), Q667 (EDP-18), K407, K525, and Z717, were reviewed. From the review, the inspectors determined that none of the PEPs had identified the relays' applicable safety functions. A typical safety function of a relay would have been the ability to change and maintain state as required under all design conditions. These abilities could have been verified by critical characteristics such as contact configuration (the contacts were positioned correctly for each state and would maintain the correct state),

pick-up and drop-out voltages (the relay would not drop out on voltage transients within the design limits, the relay would energize on the minimum voltage within the design limits, and would remain energized without damage on the maximum voltage within the design limits), contact resistance (the resistance was not so large as to cause overheating or an excessive voltage drop), coil resistance (the coil resistance was as designed, indicating that the correct coil was installed and that the coil was not shorted or having too few turns), insulation resistance (no unwanted conduction paths to ground, from coil to contact, or contact to contact), coil and contact voltage and current ratings (the relay was able to operate within the voltage and current design limits), contact timing (the relay did not operate sluggishly), and seismic capability (the relay would operate with acceptable chatter during a design basis event).

Discussion with SONGS personnel indicated that the component's safety function was not necessarily utilized in determining critical characteristics and that they were more likely to be based on the manufacturer's listed specifications to identify the item received was the item purchased. When a PEP was developed without defining the safety function of the component, the critical characteristics listed must envelop the potential safety functions of all applications in which the component might have been utilized. With this consideration, PEPs for very similar components, for which the safety function was not defined, would be expected to have listed very similar, if not identical, critical characteristics.

PEP K371, dated February 21, 1991, was performed for the purchase and dedication of three relays from Potter and Brumfield and identified the following critical characteristics: part number, serial number, insulation resistance, contact configuration, coil resistance, and pick-up and drop-out voltages. The PEP failed to evaluate coil and contact voltage ratings, coil and contact current ratings, contact resistance, contact timing, and seismic capability as critical characteristics.

PEP K541, dated April 5, 1991, was performed for the purchase and dedication of 15 Ross-Midtex relays from Newark Electronics Company and identified the following critical characteristics: part number, coil voltage rating, mounting configuration, and pick-up and drop-out voltages. The PEP failed to evaluate contact configuration, insulation resistance, contact voltage rating, contact and coil current ratings, coil and contact resistances, contact timing, and seismic capability as critical characteristics.

PEP Q667, dated March 21, 1991, was performed for the purchase and dedication of 10 Sigma relays from Magnecraft Electric Company and identified the following critical characteristics: mounting configuration, contact configuration, coil resistance, and pick-up and drop-out voltages. The PEP failed to evaluate part number, insulation resistance, coil and contact voltage ratings, coil and contact current ratings, contact timing, and seismic capability as critical characteristics.

PEP Z717, dated September 11, 1991, was performed for the generic dedication of relays and identified the following critical characteristics: part number, contact configuration, coil voltage and current rating, mounting configuration, and pick-up and drop-out voltages. The PEP failed to evaluate insulation resistance, contact voltage ratings, contact current ratings, coil and contact resistance, contact timing, and seismic capability as critical characteristics.

PEP K407, dated January 2, 1991, was performed for the dedication of general purpose Potter and Brumfield relays and identified the following critical characteristics: part number, coil voltage rating, contact configuration, pick-up and drop-out voltages, and enclosure type. The PEP failed to evaluate insulation resistance, coil current rating, coil and contact voltage ratings, coil and contact resistance, contact timing, and seismic capability as critical characteristics.

PEP K525, dated April 4, 1991, was performed for the dedication of Allen-Bradley relays and identified the following critical characteristics: part number, coil voltage rating, contact configuration, enclosure type, mounting configuration, and pick-up and drop-out voltages. The PEP failed to evaluate insulation resistance, contact voltage rating, coil and contact current ratings, coil and contact resistance, contact timing, and seismic capability as critical characteristics.

Review of the above six PEPs showed that 10 different critical characteristics were identified among the PEPs, with 5 identified, per PEP, on average. Only 1 critical characteristic, pick-up and drop-out voltages, was common to all 6 PEPs. The 2 next most common critical characteristics, part number and contact configuration, were listed on 5 of the 6 PEPs. While some of the items listed as critical characteristics such as enclosure type and mounting configuration could be application specific, others such as insulation resistance, contact configuration, coil and contact voltage ratings, coil and contact current ratings, and seismic capability would be applicable to most safety functions in most applications. The PEPs failed to consistently evaluate those critical characteristics required to verify a typical safety function of a relay. There was no indication that a seismic screening or evaluation had been performed on the PEPs reviewed. SONGS personnel indicated that they assumed if the same part number was procured, it constituted a like-for-like replacement and that no seismic evaluation was required. This reasoning failed to consider the instability or lack of commercial grade manufacturer's quality control programs and that material, configuration, and design could be routinely changed without resulting in a new part number.

(18) MDP-11 for PEP V141, dated November 26, 1990, was reviewed. SCE PO No. 6Q091004 was issued September 5, 1991, to Coast Engineering and Manufacturing Company, for a shaft coupling assembly for the containment polar crane. PEP V141 specified material as one of the critical characteristics to be verified by receipt inspection. The acceptance basis was to verify "steel," "mild steel" and "carbon steel" materials by checking magnetism. Magnetic check was performed as documented on RIDR No. RSO-2663-91. The item's safety function was not specified. Material

verification by magnetic check is not a valid method for determining the specific type of steel received.

(19) MDP-16 for PEP Q932 dated February 8, 1991, was reviewed. SCE PO No. 6L021008 was issued to Familian Pipe and Supply Company for various brass pipe fittings for the safety-related portion of the air supply system for the pressurizer power relief valve actuator. PEP Q932 specified the critical characteristics to be identified as dimensions and material. RIDR-0359-91 specified using a Weight Alloy Comparator to verify that material is red brass. However, since RQC inspection did not have a calibration standard for red brass, they used a leaded red brass standard which has the same copper composition requirement, 85 percent, as red brass.

SCE failed to provide justification for accepting the material as being red brass based upon test results utilizing the leaded red brass standard. If the remaining material composition is inconsequential, it should have been addressed because red brass, leaded red brass, high leaded tin bronze, and aluminum bronze all contain 85 percent copper. The PEP also did not specify the item's safety function.

(20) MDP-25 and 26 for commodity list item evaluations were reviewed. SCE PO No. 6D070028 was issued July 29, 1990, to J. Arthur Moore CA., Incorporated, for pipe nipples for the Unit 3 diesel generator circulating pump. These items were dedicated and documented on CLIE 82-89, dated December 7, 1982. The critical characteristics were not specified and receipt inspection was given as the standard requirements for quantity, damage, and dimensions. No safety function was identified. Material was not specified as a critical characteristic and, therefore, was not verified. The critical characteristics and verification acceptance methods were not in accordance with procedure S0123-XI-2.8 in effect at the time of these dedications.

(21) MDP-28 was for CLIE 84-125. SCE PO No. 6D100012 was issued October 16, 1990, to San Diego Valve and Fitting Company for an A 316 stainless steel (SS) globe valve for the reactor coolant pump barrier delta-p seal flow controller. The dedication of this globe valve was based on an engineering evaluation documented on CLIE 84-125, Revision 4. The critical characteristics to be verified were identified as cleanliness and dimensions. Receipt inspection to verify cleanliness and damage was performed on November 12, 1990, and documented on RIDR No. RSO-3243-90. This dedication was inadequate because SCE failed to identify and then verify material as a critical characteristic.

(22) EDP-25 for PEP V198, dated October 22, 1991, was reviewed. SONGS PO No. 6Q101008 dated October 15, 1991, was issued to Wallace and Tieram Incorporated for the purchase of relays. PEP V198 identified the safety function of the general purpose, ac or dc relay, as being able to change state as required in order to energize or deenergize the appropriate downstream components and permit the next desired operation in the control sequence. The safety function, as indicated, fails to recognize the requirement that relays be able to maintain the required state under

all design conditions. This led to a failure to list critical characteristics, such as pick-up and drop-out voltages (the relay would not drop out on voltage transients within the design limits, the relay would energize on the minimum voltage within the design limits, and would energize without damage on the maximum voltage within the design limits), contact insulation resistance (verify no unwanted conduction paths from contact to ground or contact to contact), coil and contact voltage and current ratings (the relay was able to operate within the voltage and current design limits), contact resistance (the resistance was not so large as to cause overheating or an excessive voltage drop), contact timing (the relay did not operate sluggishly), and seismic capability (the relay would operate with acceptable chatter during a design basis event).

PEP V198 identified the following critical characteristics: (1) part number, (2) mounting configuration, (3) contact configuration, (4) dimensions, (5) coil resistance and insulation integrity, and (6) contact operation. The PEP failed to evaluate pick-up and drop-out voltages, contact insulation resistance, coil and contact voltage and current ratings, contact resistance, contact timing, and seismic capability as critical characteristics to verify that the relay would be able to perform its required safety function under all design conditions.

Critical characteristics (1), (2), (3), and (4) were verified during receiving inspection by performing a visual inspection of the component and comparing it to the manufacturers literature. Critical characteristic (5) was verified by a pre-installation test.

The inadequate dedications of the CGIs discussed above, some of which were installed, constituted a failure to perform and document an adequate review for suitability of application, and in some cases, adequate design verification (seismic/EQ), for items intended for safety service, that is contrary to the requirements of Criterion III of 10 CFR Part 50, Appendix B. These inadequate dedications also constituted a failure to verify that the items received met the specifications for their safety-related applications, that is contrary to the requirements of Criterion VII of 10 CFR Part 50, Appendix B. Representative examples of the inadequate dedications listed in Section 3 are cited as Deficiency 91-201-01 in the Executive Summary of this report.

4 PROCUREMENT AND DEDICATION TRAINING

The inspector reviewed the SONGS Training Program Description T-04, "Technical Training for Nuclear Procurement Engineering," Revision 0, dated May 13, 1991. The program was developed by performing an analysis of the procurement engineer's job, dividing the job into basic tasks, and developing courses to cover all the tasks. The courses that an employee would take were determined by whether the employee performed a task covered by a course.

The program consisted of two primary sections, Phase I and Phase II. Phase I, Initial Training, was scheduled to begin January 1, 1992, and all PE employees were to be trained, exempted (employee already possessed course knowledge), or waived (course knowledge not required for the position) in the 15 courses listed by July 1992. The inspector reviewed a draft of the Phase I course T1P001, "Procurement Process," which was directly applicable to the dedication process and found it to be programmatic. It failed to be specific in the methods and thought processes needed to perform such tasks as determining critical characteristics and the appropriate verification methods.

Phase II, Initial Training, was scheduled to begin by July 1992 and all PE employees were to be trained, exempted, or waived, in the 14 courses by January 1994. Phase II offered two courses directly applicable to the dedication process; D1P002, "Procurement Specifications," and D3PE01, "Procurement Requirement Documents."

The PE training curriculum, T-04, appeared to provide a framework for development of the knowledge and skills required by a procurement engineer but was so narrow in focus that it failed to provide training on the entire dedication process and excluded such areas as parts classification, failure analysis, and determination of safety function. Though these tasks may not be specifically performed by procurement engineers, the knowledge is important to the overall process.

5 EXIT MEETING

On December 13, 1991, the inspection team conducted an exit meeting with members of the SCE staff and management at the SONGS site. During the exit meeting the team summarized the inspection findings and observations. The following individuals were present.

<u>Southern California Edison</u>

H. Morgan, Vice President and Site Manager
J. Reilly, Manager of Nuclear Engineering and Construction
H. Newton, Manager, Site Support Services
B. Katz, Manager, Nuclear Oversight
R. Rosenblum, Manager of Nuclear Regulatory Affairs
L. Rice, Manager, Material Support
R. McWey, Supervisor, Supplier Quality Assessment
D. Stonecipher, Supervisor, Site Quality Control
F. Holts, Manager, Nuclear Procurement
T. Herring, Supervisor, Procurement Engineering (PE)
W. Frick, Supervisor, Assessment Engineering
J. Wimberly, Supervisor, Material Inspection Services
L. Hadley, Supervisor, PE Mechanical/Civil Group
R. Clift, Supervisor, PE Electrical/Controls Group
T. Mackey, Supervisor, PE Technical Evaluations
S. Kraus, Supervisor, Nuclear Procurement
G. Gibson, Supervisor, Generic Licensing Section
S. Brown, Supervisor, Contract Development
C. Anderson, Procurement Engineering
J. Walderhaug, Licensing Engineer
L. Campoy, Licensing Engineer
[illegible]. Hammons, Supplier Audits

M. Marzec, Material Support
K. Beagle, Material Coordinator
E. Rinard, Supervisor, Warehouse Operations

<u>Nuclear Regulatory Commission</u>
D. Matthews, Acting Deputy Director, Division of Reactor Inspection and Safeguards (DRIS), NRR
L. Norrholm, Chief, Vendor Inspection Branch (VIB), NRR
U. Potapovs, Section Chief, VIB, NRR
D. Kirsch, Chief, Reactor Safety Branch, Region V
R. McIntyre, Team Leader, VIB, NRR
S. Alexander, EQ and Test Engineer, VIB
L. Campbell, Reactor Engineer, VIB
B. Rogers, Reactor Engineer, VIB
W. Wagner, Reactor Engineer, Region V

<u>Other Personnel</u>
B. Bradley, Senior Project Manager, NUMARC
D. Douglass, QA, Arizona Public Service
G. Wooley, Manager, Procurement QA, Washington Public Power Supply System (WPPSS)
B. Van Erem, Supervisor, Procurement Engineering, WPPSS

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

March 19, 1992

Docket No. 99901119

Mr. Brij M. Bharteey, President
Spectrum Technologies USA, Incorporated
133 Wall Street
Schenectady, New York 12305

Dear Mr. Bharteey:

SUBJECT: COMMERCIAL-GRADE DEDICATION PROGRAM INSPECTION AT SPECTRUM TECHNOLOGIES (NRC INSPECTION REPORT 99901119/92-01)

We are forwarding the report of a commercial-grade (CG) component dedication inspection performed from March 2 through 5, 1992, at Spectrum Technologies USA (Spectrum). Schenectady, New York, involving activities authorized by 10 CFR 21. The Nuclear Regulatory Commission (NRC) staff from the Vendor Inspection Branch of the Office of Nuclear Reactor Regulation conducted the inspection. An exit meeting was held on March 5, 1992, during which we discussed the team's findings with you and members of your staff.

Areas examined during the inspection are discussed in the enclosed copy of our inspection report. The inspection team assessed the adequacy of Spectrum's dedication program for qualifying CG equipment for use in safety-related applications in nuclear power plants. The inspection consisted of a selective review of relevant procedures, representative records, CG equipment being dedicated for safety-related applications, and interviews with engineering and technical support staff.

The team considered Spectrum's dedication program for qualifying CG molded case circuit breakers (MCCBs) and Agastat relays to be generally acceptable. Certain strengths were identified in the areas of responding to recent traceability problems, organization of dedication documents, and staff interface on quality assurance issues. However, findings were identified with regard to the implementation of the Spectrum quality assurance program. For example, Spectrum failed to address the premature tripping of the instantaneous magnetic trip function of MCCB's in their acceptance test procedures. Specific findings and references to the pertinent requirements are identified in the enclosed Notice of Violation and Notice of Nonconformance.

Pursuant to the provisions of 10 CFR Part 2.201, you are required to submit to this office within 30 days of your receipt of this Notice of Violation, a written statement of explanation for each violation. However, during the inspection the team noted that action had been taken by you to correct the identified violations and to prevent recurrence. Consequently, no reply to the violations is required. However, you are requested to provide a written statement in accordance with the instructions specified for the enclosed Notice of Nonconformance.

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

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

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99901119/92-01

ENCLOSURE 1

NOTICE OF VIOLATION

Spectrum Technologies USA, Incorporated
Schenectady, New York 12305

Docket No. 99901119
Report No. 92-01

During the Nuclear Regulatory Commission (NRC) commercial-grade component dedication inspection conducted from March 2 through 5, 1992, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1992), the violations are listed below:

A. Section 21.21, "Notification of failure to comply or existence of a defect and its evaluation," of 10 CFR Part 21, states, in part, that each individual, corporation, partnership, or other entity subject to the regulations in this part must adopt appropriate procedures to evaluate deviations and failures to comply in all cases within 60 days of discovery.

   Contrary to the above, Spectrum Technologies USA, Incorporated (Spectrum), failed to incorporate into its procedures the time limit for notification and other new requirements that are specified in the current revision of 10 CFR Part 21, dated July 31, 1991 (92-01-01).

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

B. Section 21.6, "Posting requirements," of 10 CFR Part 21 requires, in part, that individuals and corporations post current copies of 10 CFR Part 21, Section 206 of the Energy Reorganization Act of 1974, and its procedures adopted to implement 10 CFR Part 21.

   Contrary to the above, Spectrum failed to post a current copy of 10 CFR Part 21, dated July 31, 1991. (92-01-02).

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

Pursuant to the provisions of 10 CFR Part 2.201, you are required to submit to this office within 30 days of your receipt of this Notice, a written statement of explanation for each violation. However, during the inspection the team noted that action had been taken by you to correct the identified violations and to prevent recurrence. Consequently, no reply to these violations is required.

Dated at Rockville, Maryland
this 19th day of March 1992.

NOTICE OF NONCONFORMANCE

Spectrum Technologies USA, Incorporated
Schenectady, New York

Docket No.: 99901119
Report No.: 92-01

During a Nuclear Regulatory Commission (NRC) commercial-grade (CG) component dedication inspection conducted from March 2 through 5, 1992, at Spectrum Technologies USA, Incorporated (Spectrum), Schenectady, New York, the inspection team determined that certain activities were not conducted in accordance with NRC requirements. These requirements were contractually imposed by licensees' purchase orders to Spectrum. The NRC has classified these items as nonconformances to the requirements of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix B.

A. Criterion V, "Procedures, Instructions, and Drawings," of Appendix B to 10 CFR Part 50 requires, in part, that activities affecting quality be prescribed by documented instructions or procedures of a type appropriate to the circumstances.

1. Spectrum Acceptance Test Procedure, AP9200120/1 commits to National Electrical Manufacturers Association (NEMA) Standards Publication/No. AB4-1991, "Guidelines for Inspection and Preventive Maintenance of Molded Case Circuit Breakers Used in Commercial and Industrial Applications." NEMA AB4-1991 requires a current to be applied up to 5% below the breaker's lower tolerance limit of the manufacturer's published instantaneous trip range to verify that the breaker does not trip at a current lower than the instantaneous range.

   Contrary to the above, Spectrum Acceptance Test Procedure, AP9200120/1 dated February 20, 1992, for the dedication of CG molded case circuit breakers (MCCBs), did not address the premature tripping of the instantaneous magnetic trip function for these breakers. This procedure was being used to dedicate CG MCCBs for use in safety-related applications at the Kewaunee Nuclear Station (92-01-04).

2. Paragraph 13.4 of Section 13, "Handling, Storage and Shipping," of Spectrum's Quality Assurance Manual (QAM), Revision 3, dated March 31, 1988, states in part, that storage of items shall be conducted in accordance with established instructions or procedures specified by Spectrum Technologies.

   Section 2.2 of Spectrum's QAM commits to the requirements of American National Standard Institute (ANSI) N45.2, "Quality Assurance Requirements for Nuclear Power Plants." ANSI N45.2.2-1972 Section 6.1.2, "Levels of Storage," requires, in part, that level B storage items shall be provided with uniform heating and temperature control or its equivalent, to prevent condensation and corrosion. Level B items include switchgear components such as breakers.

Contrary to the above, Spectrum Quality Assurance Procedure 13/001, "Packaging, Handling, Storage, and Shipping," Revision 0, dated January 22, 1988, did not address heating and temperature control requirements for Spectrum's Level B storage (92-01-03).

Please provide a written statement or explanation to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, 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 19th day of March 1992.

ENCLOSURE 3

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

ORGANIZATION:	SPECTRUM TECHNOLOGIES USA, INCORPORATED SCHENECTADY, NEW YORK
REPORT NO.:	99901119/92-01
CORRESPONDENCE ADDRESS:	Mr. Brij M. Bharteey, President Spectrum Technologies USA, Incorporated 133 Wall Street Schenectady, New York 12305
ORGANIZATIONAL CONTACT:	Mr. Brij M. Bharteey, President (518) 382-0056
NUCLEAR INDUSTRY ACTIVITY:	Dedication of commercial-grade (CG) components for safety-related applications.
INSPECTION CONDUCTED:	March 2-5, 1992
SIGNED:	Joseph J. Petrosino For 3/18/92 Randolph N. Moist, Team Leader — Date Reactive Inspection Section No. 2 Vendor Inspection Branch (VIB)
OTHER INSPECTORS:	Joseph J. Petrosino, VIB
APPROVED:	Anil S. Gautam 3/18/92 Anil S. Gautam, Acting Chief — Date Reactive Inspection Section No. 2 Vendor Inspection Branch
INSPECTION BASES:	10 CFR Part 21 and 10 CFR Part 50, Appendix B
INSPECTION SCOPE:	To assess the adequacy of Spectrum Technologies USA, Incorporated (Spectrum), dedication program for qualifying CG electrical equipment for use in safety-related applications in nuclear power plants; observe testing of molded case circuit breakers (MCCBs) performed by Spectrum; and review Spectrum's corrective action for previous inspection findings.
PLANT SITE APPLICABILITY:	Numerous

1.0 INSPECTION SUMMARY

1.1 Violations

1.1.1 Contrary to Section 21.21, "Notification of failure to comply or existence of a defect and its evaluation," of 10 CFR Part 21, Spectrum failed to incorporate into its procedure titled, "Part 21 Reporting," Revision 3, dated March 31, 1988, the time limit for notification and other new requirements specified in the current revision of 10 CFR Part 21, dated July 31, 1991 (Violation 92-01-01, see section 3.4 of this report).

1.1.2 Contrary to Section 21.6, "Posting requirements," Spectrum failed to post a current copy of 10 CFR Part 21, dated July 31, 1991 (Violation 92-01-02, see section 3.4 of this report).

1.2 Nonconformances

1.2.1 Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, Spectrum's Quality Assurance (QA) Procedure 13/001, "Packaging, Handling, Storage, and Shipping," Revision 0, dated January 22, 1988, did not address heating and temperature control requirements for Spectrum's Level B storage (Nonconformance 92-01-03, see section 3.3 of this report).

1.2.2 Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, Spectrum's Acceptance Test Procedure, APS200120/1 for Job Number JN9200120, Revision 0, dated February 20, 1992, for the dedication of commercial-grade (CG) MCCBs, did not address the premature tripping of the instantaneous magnetic trip function for these breakers. This procedure was being used to dedicate CG MCCBs for use in safety-related applications at the Kewaunee Nuclear Station (Nonconformance 92-01-04, see section 3.2.1 of this report).

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Nonconformance 99901119/88-01-01 (Closed)

Nonconformance 88-01-01 stated that contrary to Criterion VII, "Control of Purchased Material, Equipment, and Services," Spectrum did not verify the validity of certificates of conformance received from Westinghouse Electric Supply (WESCO), of Albany, New York, for 250 circuit breakers sold by Spectrum to the Peach Bottom Nuclear Power Plant (Peach Bottom). Also, Spectrum did not verify if the breakers sold to Peach Bottom were new, as required by the licensee's purchase order to Spectrum and by Spectrum's purchase order to WESCO.

During this inspection, the team reviewed selective dedication packages and confirmed that implementation of Spectrum's test procedures would detect refurbished equipment. In addition, the team confirmed that each package reviewed had adequate documentation to support traceability of the circuit breakers to the original equipment manufacturer (OEM). No further concerns were identified.

3.0 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings.

During the entrance meeting on March 2, 1992, the NRC inspectors discussed the scope of the inspection, outlined areas of concern, and established interfaces with Spectrum's management and staff. At the conclusion of the inspection on March 5, 1992, the inspectors summarized their findings and concerns, and Spectrum's management and staff acknowledged this information.

3.2 Dedication Program for CG Components.

The team assessed the adequacy of Spectrum's dedication program for qualifying CG electrical equipment for use in safety-related applications in nuclear power plants. The NRC inspectors reviewed selected elements of the dedication program and the implementation of this program at the Spectrum facility.

The team reviewed the dedication of CG MCCBs and Agastat relays, relevant test procedures and records for MCCBs, storage of equipment, and training of personnel.

The team reviewed a sample of Spectrum's CG component dedication packages, and verified conformance with NRC regulations and industry practices. Good organization of documents in the dedication packages was noted.

The team reviewed markings, dimensions and configurations of the MCCBs and relays. The team confirmed that Spectrum had adequate documentation to provide proof of traceability of selected MCCBs to the OEM, and valid certificates of conformance from the OEM. The team reviewed receipt inspection performed by Spectrum for selected breakers and relays and determined that the receipt inspection was acceptable. The team also noted that Spectrum had responded to traceability problems with the Agastat 7000 series relays and had informed the NRC of a potentially generic issue. The team considered this a strength.

The team reviewed a selection of Spectrum's indoctrination records and conducted discussions with Spectrum staff regarding the frequency of indoctrination and the methodology employed. The team determined that Spectrum employees had received an initial QA program indoctrination. The team also noted that good interface existed between Spectrum's management and support staff. For example, weekly meetings were held among staff to discuss quality assurance issues.

In general, the inspectors found the Spectrum CG dedication program to be acceptable. However, certain findings were identified and are described in Sections 3.2.1 through 3.4 of this report.

3.2.1 Dedication Testing of CG MCCBs.

The team observed the dedication testing of selected MCCB.. Spectrum stated that it prepared a new test procedure for each purchase order of MCCBs so as to customize the test procedure for the type of breaker being supplied. The test procedures included criteria for the testing of thermal trip, insulation resistance, operating and instantaneous trip ratings. All tests were required to be performed in accordance with National Electrical Manufacturers Association (NEMA) AB4-1991.

The NRC team selected three samples of Westinghouse 480 volt commercial-grade MCCBs that were scheduled for testing by Spectrum. The samples included:

(2) - 1 pole, Type EHD1015, 45 ampere MCCBs
(1) - 3 pole, Type EHD3100, 100 ampere MCCB

These breakers were being supplied by Spectrum to Wisconsin Public Service for use in safety-related applications at the Kewaunee Nuclear Station.

During their review, the team noted that the Spectrum Acceptance Test Procedure AP9200120, Revision 0, dated February 20, 1992, did not address the premature tripping of breakers. This was contrary to NEMA AB4-1991 which required a current to be applied up to 5% below the breaker's lower tolerance limit of the manufacturer's published instantaneous trip range to verify that the breaker did not trip at a current lower than the instantaneous range. The inspectors were concerned that the potential tripping of a breaker outside the lower limit of its trip range could cause a loss of coordination in accident mitigating circuits at the Kewaunee plant. On further evaluation, the team determined that Spectrum had performed additional tests which applied appropriate currents in the instantaneous range, and that these tests demonstrated that premature instantaneous tripping would not occur for the sample of breakers tested.

The team noted two additional acceptance test procedures (for Texas Utilities Electric Company and Northern States Power Company) which also did not address the premature tripping of breakers. The team also noted that none of the test procedures addressed the current application method (run up or pulse) used during the testing of the breakers. To mitigate the team's concerns, Spectrum took immediate corrective action to modify its program and address premature tripping. Spectrum changed its acceptance test procedures to address instantaneous trip tests for both adjustable and non-adjustable thermal-magnetic and magnetic breakers, and noted in its procedures that the pulse method was to be used when conducting the instantaneous trip test (Nonconformance 92-01-04).

The team also observed that Spectrum's Acceptance Test Procedure AP9200120, Revision 0, required the application of 6000 amperes to the Westinghouse Type EH3015 MCCBs during the instantaneous trip test. This current was much higher than the MCCB's upper tolerance instantaneous trip rating of 1050 amperes. The inspectors noted that even though the MCCBs had an interrupting capacity of 14000 amperes, the application of 6000 amperes could unnecessarily stress the breakers. After discussions with the OEM, Spectrum confirmed that the breakers would not be damaged if 6000 amperes were applied. However, Spectrum agreed to revise the acceptance test procedure to reflect the application of a lower current within the tolerances of the MCCBs. No further concerns were identified.

3.2.2 Agastat 7000 Series Relays.

The NRC reviewed the dedication of CG Agastat Model 7000 electrical relays. Agastat Model 7000 relays are manufactured by the Amerace Corporation (Amerace), Livingston, New Jersey. In 1990 Spectrum ordered six Agastat Model 7032 PBB CG relays from a WESCO office, in Albany, New York. WESCO ordered the relays from an authorized Amerace distributor, Control Components Supply (CCS), Short Hills, New Jersey. Spectrum planned to dedicate the CG relays for safety-related applications.

Spectrum noted that the relays supplied to them did not have proper traceability to the OEM. Based on discussions with the OEM, Spectrum subsequently identified that modifications had been made to the relays and that these modifications were not reflected on the nameplate labels.

Spectrum informed the NRC staff of this problem. Consequently, the NRC staff conducted several meetings with Amerace and CCS representatives during the period of October 1991 through January 1992. Amerace stated that it allowed its authorized distributors (ADs) to make field modifications to CG relays to meet their customer requirements. Modifications included changing the electrical coil module for different voltage level applications, adding or changing the electrical contact assembly module, and changing the time duration disc and wafer. Amerace stated that, if an AD modified a CG Agastat model 7000 relay, the AD was supposed to install a new nameplate label with the serial number (S/N) prefixed by an F. The NRC noted that the six subject relays supplied to Spectrum had been disassembled, modified, and reassembled by CCS but that the new labels affixed by CCS did not contain the required F prefix. However, the NRC staff also noted that prior to January 1992, Amerace had not formally required its ADs to use the F prefixed labels.

Amerace stated that before final calibration, test, and acceptance of the 7000 series relays, Amerace heat stabilized each relay in an electrical convection oven for 4 hours. This heat stabilization was performed to make the timing disc with the ceramic timing wafer to prevent timing drift, ensure repeat accuracy, and to stress relieve all non-metallic parts. The NRC noted that the six subject relays supplied to Spectrum had their timing discs changed by CCS; however, there was no evidence of the relays being re-stabilized by the AD. An information notice is being issued by the NRC titled, "Distributor Modification To Certain Commercial Grade Agastat Electrical Relays," to address these concerns.

During their current review, the team examined three CG Agastat relay test specimens to determine if any speciman had been modified by an AD and if proper traceability existed. One specimen was a CG Agastat Series 7000 relay Model 7024 PD, S/N 88411041. This relay was procured for Virginia Electric and Power (VEP) Company, Purchase Order (PO) No. CNT 309853, dated October 7, 1988, for use in safety-related applications at the Surry Power Station. The team's review of Spectrum's 1988 dedication package revealed that Spectrum had failed to ensure traceability to the OEM. The S/N on the label indicated that the relay was manufactured during the 41st week of 1988, and that it was the 1041st relay that was manufactured during that week. Upon comparison of the stamped Amerace date codes on the relay, the team determined that the relay was actually manufactured during the 28th week of 1988, and that it had been modified by an AD after leaving the Amerace facility. However, this relay did not exhibit any evidence regarding the type and extent of modifications, nor did the relay have a F prefix on its nameplate label.

The team also noted that Spectrum's Qualification Test Report (QTR) No. 8800150, dated October 14, 1988, for the seismic qualification of the relays, identified problems with the accuracy of the tested relays. The report stated that one of each model of Agastat relay seismically tested at the Ontario Hydro Laboratory, Canada, had operated "at less than the pre-set value of 5.0 seconds at the change of state during the seismic event." Spectrum had identified this as a testing anomaly.

Based on the discrepancies of date codes stamped on the relays and the seismic test anomaly, the NRC inspectors were concerned that the 35 relays supplied by Spectrum to Surry as Class 1E relays had the potential of not performing their intended safety functions. Spectrum discussed this matter with VEP and informed the team that VEP had written a nonconformance report (NCR) No. 88-102, dated November 10, 1988, due to the test anomalies identified in Spectrum's QTR 8800150, and that VEP had concluded in the NCR that these relays were to be used only in non-Class 1E applications. No further concerns were identified.

3.3 Storage of Spare Parts.

During a tour of the Spectrum facility, the team observed a room designated for storage of Spectrum's qualification test specimens, spare parts and equipment being dedicated for safety-related applications. This room was located in the basement of the facility. The team was concerned because this room did not contain any heating, cooling, or humidity control.

Spectrum provided the team a copy of their Quality Assurance Procedure (QAP) 13/001, "Packaging, Handling, Storage, and Shipping," Revision 0, dated January 22, 1988. The inspectors noted that Spectrum's QAM was committed to ANSI N45.2, "Quality Assurance Requirements for Nuclear Power Plants." Section 6.1.2, "Levels of Storage," of ANSI N45.2.2-1972, states, in part, that environmental conditions for items classified as Levels A, B, C, and D, shall meet requirements as described: Level B items shall be provided with uniform heating and temperature control, or its equivalent, to prevent condensation and corrosion. Minimum temperature shall be 40 degrees Fahrenheit (F) and maximum temperature shall be 140 degrees F.

The team noted that Level B items included switchgear components such as breakers and relays. The team concluded that Spectrum's QAP 13/001 failed to include the heating and temperature control requirements of this ANSI standard, for Level B items (Nonconformance 92-01-03).

3.4 10 CFR Part 21 Posting and Proce ure.

The team observed that Spectrum had not posted the current, July 31, 1991, revision of 10 CFR Part 21 in its facility, and had also failed to ensure that Spectrum's 10 CFR Part 21 procedure encompassed all of the requirements of the July 31, 1991, revision of 10 CFR Part 21.

Spectrum stated that it was not aware that a new revision to 10 CFR Part 21 had been issued but that it had received a copy of NRC information notice (IN) 91-39, "Reporting of Defects and Noncompliances," and had posted a copy of the 10 CFR Part 21 regulation that was enclosed in this information notice. The team noted that IN 91-39 was issued on June 17, 1991 by the NRC's Office of Nuclear Material Safety and Safeguards (NMSS) and that this notice was issued for and addressed to NRC material licensees regulated by NMSS. Spectrum had incorrectly assumed that IN 91-39 included a new revision of 10 CFR Part 21.

Spectrum agreed to take immediate corrective action. They provided the team with a revised 10 CFR 21 posting and procedure. The NRC team found the corrective action appropriate (Violations 91-01-01 and 91-01-02).

3.5 Johnson Yokogawa Corporation Controllers.

Spectrum notified the NRC by a December 23, 1991, letter of a "deviation" identified to them by Johnson Yokogawa Corporation (JYC). JYC is the OEM for Yokogawa programmable indicating controllers (PICs). The deviation concerned CG PICs procured and dedicated by Spectrum for safety-related applications in the Sequoyah nuclear power plant. Spectrum's letter to the NRC had stated that the problem was limited to Yokogawa's YS-80 series Model SLPC 281*E/MTS/NPR/HTB.

The deviation identified by JYC concerned the potential of a loose electrical connection between an electronic chip and its socket. The PIC receives input from a read only memory (ROM) imbedded in a chip that is plugged into a ROM socket. The ROM socket is located on one side of the PIC. The ROM chip socket is designed to capture the pins of the chip with spring pressure applied by the socket's wear plate blades. JYC stated that the material used for the wear plate blades did not have sufficient spring characteristics to adequately secure the chip in the ROM socket on the identified Yokogawa PIC models. As a result, intermittent electrical contact between the ROM chip and ROM socket could occur. Symptoms of this condition are as follows:

1. The "fail" lamp lights and check code numbers "10," "00," and/or "04" will be displayed.

2. An instrument may operate as if it is receiving input, even though the digital input is not, or is being intermittently, received.

3. The trouble phenomenon could differ depending on the program in the ROM.

Spectrum told the team that JYC had not made an assessment of any generic concern. To address NRC concerns in this area, Spectrum telephoned JYC during the inspection and was informed by JYC that the problem affected other PIC models including SLPC*E, SLMC*E, SLMS*E, and SPLR*E. JYC also told Spectrum that the SLPC*E models were the only PICs being used in Class 1E applications.

The NRC staff was also informed of this problem by Nutherm International Incorporated (NII) on February 3, 1992. NII's letter contained information similar to that provided by Spectrum, with the exception that NII's letter identified an additional PIC model, SLPC-271*E, as having the same problem. The NRC will continue to review this deviation with JYC.

4.0 PERSONNEL CONTACTED

Spectrum Technologies

+*B. Bhartey, President
+*B. Willis, Quality Assurance Manager
A. Sadaghiany, Test Technician
C. Hicks, Acting Manager of Inspection & Test
D. Kelleman, Consultant
P. Woomer, Qualification Engineer
W. Allen, United Controls (Telephonically)
P. VanDenheuvel, Wisconsin Public Service Corporation (Source Inspector)

Nuclear Regulatory Commission

* A. Gautam, Acting Section Chief, VIB, NRR, NRC

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* Attended the exit meeting
\+ Attended the entrance meeting

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

March 26, 1992

Docket No. 99901226

Mr. Rubin Feldman, President
Thermal Science, Incorporated
2200 Cassens Drive
St. Louis, Missouri 63026

SUBJECT: NRC INSPECTION REPORT 99901226/91-01

Dear Mr. Feldman:

This letter addresses the December 16-20, 1991, inspection of Thermal Science, Incorporated (TSI) in St. Louis, Missouri. The inspection was conducted by Messrs. R. C. Wilson and R. N. Moist of this office. The inspection findings were discussed at the conclusion of the inspection with you and the members of your staff identified in the enclosed report. The purpose of the inspection was to review TSI's program for supplying Thermo-Lag fire barrier material for use in commercial nuclear power plants.

Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and records, interviews with personnel, and observations by the inspectors. The inspection identified that the implementation of your QA program failed to meet certain U.S. Nuclear Regulatory Commission (NRC) requirements. Specifically, your quality assurance procedures failed to specify a requirement for measuring the minimum thickness and maximum weight of prefabricated panels and conduit sections of Thermo-Lag fire barrier material. These measurements are important to safety because thin sections may not provide assured fire barrier capability, and overweight sections could exceed cable tray and conduit support capabilities. Although in only one case was a maximum thickness specified in a purchase order, evidence was not available to show conformance with this requirement. Maximum thickness may be an important consideration in licensee ampacity derating calculations.

In addition, although the inspection did not concentrate on qualification tests, we found that your procedures did not adequately specify controls over such tests, particularly regarding incomplete definition of test specimen construction and the role of Industrial Testing Laboratories, Inc. (ITL) in observing tests performed by TSI. These concerns challenge the validity of these tests and their use by NRC licensees in verfiying conformance with NRC requirements.

The specific findings and references to the pertinent requirements for the above nonconformances are identified in the enclosed Notice of Nonconformance.

We were also concerned by the installation support provided by TSI to your customers. Although TSI trained installers and provided an installation guide, some licensees have reported installation deficiencies with Thermo-Lag material in commercial nuclear power plants. These deficiencies resulted in inadequate fire barriers and possible loss of redundancy in engineered safety feature systems. Based on actual nuclear plant experience, the TSI position that customer installation procedures supplemented by general customer training should be sufficient to ensure adequate installation of Thermo-Lag may not be correct.

The response requested by the enclosed Notice of Nonconformance is 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 inspection was restricted to documents and personnel at TSI, and the inspectors did not review any site documents or attempt to close any ongoing NRC reviews.

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,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Nonconformance
2. Inspection Report 99901226/91-01

ENCLOSURE 1

NOTICE OF NONCONFORMANCE

Thermal Science, Incorporated
St. Louis, Missouri 63026

Docket No.: 99901226/91-01

During a U.S. Nuclear Regulatory Commission (NRC) inspection conducted at Thermal Science, Incorporated (TSI) in St. Louis, Missouri on December 16-20, 1991, the NRC inspection team determined that certain activities were not conducted in accordance with NRC requirements that were contractually imposed on TSI by purchase orders from an NRC licensee. The NRC has classified these items as nonconformances to the requirements of Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix B, Quality Assurance Program.

A. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50 requires in part that activities affecting quality be prescribed by documented instructions or procedures.

Section 6.1 of TSI's Nuclear Quality Assurance (QA) Program Manual, Revision X, dated January 12, 1987, states that documented instructions and procedures are provided to prescribe all TSI activities affecting QA.

Contrary to the above, TSI's documented instructions and procedures used for purchase orders invoking 10 CFR Part 50, Appendix B, did not require verification of the maximum weight and minimum thickness of prefabricated panels and conduit sections during final inspection. (91-01-01)

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

Several instances were observed of the failure of TSI's qualification testing to conform to Appendix B to 10 CFR Part 50, including the following.

1. Section 6.1 of TSI's Nuclear QA Program Manual, Revision X, dated January 12, 1987, states that documented instructions and procedures are provided to prescribe all TSI activities affecting QA.

Contrary to the above, TSI's fire endurance qualification test plans did not provide complete instructions for fabricating the test specimens. Several dimensions were not specified and instructions for filling joints were not specific. (Test records provided as-built data for some but not all of this information.)

2. Section 13.4 of TSI's Nuclear QA Program Manual, Revision X, dated January 12, 1987, states that written calibration procedures shall specify the method to be used and the time interval between calibrations for test equipment.

   Contrary to the above, no documentation was found specifying calibration of furnace thermocouples used for qualification testing of fire barrier specimens for use in commercial nuclear power plants.

3. Section 5.3 of TSI's Nuclear QA Program Manual, Revision X, dated January 12, 1987, states in part that the requisitioning of services for use in nuclear safety-related activities shall use the purchase requisition form, which shows information such as the applicable regulations, codes, specifications and standards.

   Section 5.6 of TSI's Nuclear QA Program Manual, Revision X, dated January 12, 1987, states that the purchase order shall contain all pertinent and applicable requirements listed on the purchase requisition.

   Contrary to the above, TSI had no written contract with Industrial Testing Laboratories, Inc. (ITL), which served as an independent observer for qualification tests actually conducted by TSI. The TSI president stated that only an oral agreement existed, which specified rates but not scope of work.

4. Section 19.4 of TSI's Nuclear Quality Assurance Program Manual, Revision X, dated January 12, 1987, states in part that suppliers to TSI of services for nuclear safety-related activities shall be audited as required by the Manager of Quality Assurance. The frequency of such audits normally will be determined by the purchased item's potential to adversely affect quality, the complexity of the purchased item, the quantities involved and the past performance of the vendor.

   Contrary to the above, there was no record of audit of Industrial Testing Laboratories Inc. to support their role in qualification testing of fire barrier material and ITL was listed on TSI's Approved Vendor List based only on experience. (91-01-02)

Please provide a written statement or explanation to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Vendor Inspection

Branch, Division of Reactor Inspection and Safeguards, 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 26th day of March 1992

ORGANIZATION:	THERMAL SCIENCE, INCORPORATED ST. LOUIS, MISSOURI
REPORT NO.:	99901226/91-01
CORRESPONDENCE ADDRESS:	Mr. Rubin Feldman, President Thermal Science, Incorporated 2200 Cassens Drive St. Louis, Missouri 63026
ORGANIZATIONAL CONTACT:	Mr. Rubin Feldman, President (314) 349-1233
NUCLEAR INDUSTRY ACTIVITY:	Thermo-Lag fire barrier materials and related installation training services
INSPECTION CONDUCTED:	December 16-20, 1991
SIGNED:	RCWilson 3/6/92 Richard C. Wilson, Team Leader — Date Reactive Inspection Section No. 2 Vendor Inspection Branch (VIB)
OTHER INSPECTORS:	Randolph N. Moist, VIB
APPROVED:	for Gregory C Cwalina 3/9/92 Chris A. VanDenburgh, Chief — Date Reactive Inspection Section No. 2 Vendor Inspection Branch
INSPECTION BASES:	10 CFR Part 21, 10 CFR Part 50, Appendix B and 10 CFR Part 50.48
INSPECTION SCOPE:	To review Thermal Science, Inc.'s program for supplying Thermo-Lag fire barrier materials and related services for fire protection applications in nuclear power plants
PLANT SITE APPLICABILITY:	Numerous.

1 INSPECTION SUMMARY

1.1 Nonconformances

1.1.1 Nonconformance 91-01-01 (Open)

Contrary to Criterion V, "Instructions, Procedures, and Drawings," of 10 CFR Part 50, Appendix B, Thermal Science, Inc.'s (TSI's) documented instructions and procedures used for NRC licensee purchase orders invoking 10 CFR Part 50, Appendix B, did not require maximum weight and minimum thickness measurements of prefabricated panels and conduit sections during final inspection (Nonconformance 91-01-01. See Section 3.3 of this report).

1.1.2 Nonconformance 91-01-02 (Open)

Contrary to Criterion V, "Instructions, Procedures, and Drawings," of 10 CFR Part 50, Appendix B, TSI failed to comply with its documented instructions and procedures when conducting tests intended to qualify fire barriers for commercial nuclear power plants. (Nonconformance 91-01-02. See sections 3.4, 3.5, 3.7, and 3.8 of this report.)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

The NRC had not previously inspected TSI.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on December 16, 1991, the NRC inspectors discussed the scope of the inspection, outlined areas of concern, and established interfaces with TSI's management and staff. In the exit meeting on December 20, 1991, the inspectors discussed their findings and concerns with TSI's management and staff.

3.2 Inspection Scope

TSI manufactures Thermo-Lag patented heat blocking and fire retardant materials. Major applications include aerospace, oil drilling, commercial nuclear reactors, and tank cars. TSI employs between 50 and 100 personnel in a 60,000 square foot building. Commercial nuclear power plant sales grew to about half of TSI's business in the mid-1980s, and have declined to a very low current level. Only the Thermo-Lag 330 product line is supplied for commercial nuclear plants, usually in the form of panels or pre-cast conduit sleeves and trowelable mastic. TSI performs on-site training and certification of installation personnel provided by the licensees. TSI also supplies fire endurance qualification and ampacity derating test reports, and installation procedures manuals.

The NRC inspectors reviewed TSI's program for supplying Thermo-Lag 330 materials and related services both generically and against the requirements of numerous licensee purchase orders. The inspection was restricted to documents and personnel at TSI, and the inspectors did not review any site documents.

3.3 Manufacturing Process

TSI mixes Thermo-Lag 330 material in batches of 20,000 pounds maximum, with 10,000 pounds typical. Material is mixed for specific orders, rather than to maintain an inventory. Tests performed on each batch of material include a drop test and a mandrel bend test which verifies that a thin sample is essentially cured within 72 hours at 77°F and 50 percent humidity. The bulk material is loaded into drums or five gallon pails labeled with batch tickets that are coded to show constituent materials. TSI either ships the containers of material to a plant site, or uses them to fabricate flat panels or preshaped conduit sections.

The panels are cured in a large oven at 120 to 180°F for 15 to 30 days, based on in-process moisture measurements. The measurements are performed on a sample of panels using TSI Test Procedure A-29, Revision 0. A moisture content of less than ten percent is required. Although the procedure's purpose states that it applies to panel coatings, TSI's QC manager stated that it is used for Thermo-Lag 330 panels. Numerous thickness measurements are made after drying and before final QA acceptance testing. High and low spots are corrected.

Minimum thickness limits for panels and conduit sections are 0.500 inch for a one hour fire rated panel and 1.000 inch for a three hour fire rated panel. These thicknesses are intended to provide the minimum mass of material necessary to ensure the fire rating of the panel. Maximum thickness is not usually specified in Purchase Orders (POs) and is not usually certified, even though an overly thick section could affect ampacity deratings. TSI provides customers a weight sheet dated June 7, 1986, with guaranteed maximum weights for prefabricated conduit and panel sections that can be used by the customer for seismic calculations (such as cable tray hanger load). The maximum weights for flat panels are 3.5 lb/ft^2 for a one hour panel and 7.0 lb/ft^2 for a three hour panel. Minimum weights are not guaranteed.

Thickness is verified using TSI Test Procedure A-33, Revision 0, which specifies 18 measurements per panel. Weight is verified using an unnumbered TSI test procedure titled "Panel Weight Determination." Even though TSI performed thickness and weight measurements to TSI test procedures, the NRC inspectors found no procedure requiring performance of the measurements. SI's president and QC manager stated that they were not aware of any TSI procedure that required that thickness and weight measurements be performed. These values are important to safety because thin sections may not provide assured fire barrier capability, and overweight sections could exceed cable tray and conduit support capabilities. Criterion V of 10 CFR Part 50, Appendix B requires that activities affecting quality be prescribed by documented instructions or procedures. For safety-related procurements, TSI's failure to specify a requirement for performing thickness and weight measurements is designated as Nonconformance 91-01-01.

TSI's inspector signs off on the maximum weight and minimum thickness verifications on a form titled, "Thermo Lag Prefabricated Panel Q C Form." The material batch number and stress skin lot number are written on the panels and on tags attached to the panel stress skins.

The NRC inspectors reviewed shipping invoice No. 18802 under Texas Utilities (TU) Generating Co. Purchase Order (PO) No. 665-71871, Supplement 10, dated December 7, 1989, for Thermo-Lag prefabricated panels without the normal stiffener ribs. TSI personnel stated that panels without the ribs are intended for use only when attached to steel structural supports in the plant, where the stiffening capability of the ribs is not needed. No records of other shipments of panels without ribs were observed by the inspectors.

The NRC inspectors asked about a "cure accelerator." The QA manager advised that an accelerator is available which promotes early mechanical setup and is useful in cold weather. The accelerator actually does not affect drying or curing. Like the Thermo-Lag 330 materials, it is water-based. TSI does not use the accelerator in poured panels, but it can be used in spray or trowel applications and has been provided to customers. TSI's QA manager stated that an Underwriters Laboratories Inc. (UL) fire test showed that the accelerator has no adverse effects. TSI stated that UL fire tests also showed no problems with the topcoat material that TSI provides for weather resistance. The NRC inspectors did not review the UL test reports or form any conclusions regarding the use or effects of the accelerator.

The NRC inspectors asked how the six month shelf life is established for bulk Thermo-Lag 330 material in containers. TSI's QC manager stated that the bulk material's shelf life starts on the day the material is shipped to the customer. The policy is to not manufacture any material with shelf life limitations until a customer order is received. TSI can perform thermogravimetric analysis on samples returned by customers to determine if the material is still usable, because the subliming material has a relatively low volatility temperature. TSI's Bills of Lading specify that bulk material must be stored above 32°F and below 100°F at all times, and shipments are accompanied by a pail containing a temperature recorder.

The NRC inspectors showed TSI's QA manager paragraph 6.6.6 of TU's Comanche Peak nuclear plant procedure ECC 10.07, Revision 3, dated March 5, 1989, regarding the plant's criteria for repair of surface cracks or pinholes in prefabricated panels. The only criterion listed was for the width of the defect, with no repair required for less than 0.050 inch. Surface patching was specified for larger cracks or holes. There were no depth or length criteria. TSI's QA manager could not provide a basis for this procedure. He indicated that the paragraph needed more context to be meaningful, including the definitions for surface cracks and pinholes. The inspectors did not pursue this matter further.

3.4 Quality Assurance Program

TSI'S Nuclear Quality Assurance (QA) Program Manual, Revision X, dated January 12, 1987, governed its 10 CFR Part 50, Appendix B, quality assurance program. TSI Quality Control Operating Procedures Manual, Revision X, dated September 22, 1986, implemented and supported the Nuclear Quality Assurance Program Manual. The implementing procedures controlled activities affecting quality during raw materials receiving inspection and the manufacture of the Thermo-Lag 330 materials.

TSI has applied its Nuclear QA program to all Thermo-Lag 330 materials shipped to commercial nuclear power plants, regardless of what QA requirements were specified in the PO or whether the procurement was by the licensee or by another party. TSI personnel stated that the principal improvements related to the nuclear QA program are care of manufacture, records, traceability, and material purity. Although TSI's procedures make provision for procuring raw materials in accordance with 10 CFR Part 50, Appendix B, TSI personnel stated that all of their procurements have been commercial grade.

The NRC inspectors verified the implementation of TSI's QA program by reviewing selected criteria from 10 CFR Part 50, Appendix B, including nonconforming materials, identification and control of materials, handling, storage and shipping of materials, control of measuring and test equipment, and control of purchased materials. TSI did not manufacture any Thermo-Lag 330 materials during this inspection.

To verify traceability, the NRC inspectors selected batch numbers from TSI Certificates of Conformance (COCs) for selected materials (Thermo-Lag bulk material, prefabricated panels and conduit sections) that were shipped to commercial nuclear power plants. The NRC inspectors traced the batch numbers back to the batch mixes, including the lot numbers of the raw materials used. The NRC inspectors concluded that TSI had adequate quality control records and procedures for demonstrating the traceability of raw materials purchased from suppliers used in manufacturing Thermo-Lag 330 material.

The NRC inspectors selected measuring and test equipment that TSI used to verify the adequacy of the purchased raw materials, batch samples, and finished prefabricated panels (fire endurance test instruments were not reviewed, except as noted in the next paragraph). The inspectors concluded that TSI's calibration program, QC records, and procedures were adequate to perform and document the testing. In addition, the NRC inspectors verified that the calibration of measuring and test equipment was traceable to the National Institute of Standards and Technology.

The NRC inspectors briefly addressed the calibration of thermocouples used in American Society for Testing and Materials (ASTM) Standard E 119 fire endurance type qualification tests. The thermocouples that monitor specimen temperature are replaced with each specimen, and new units are obtained with current supplier calibrations. However, the thermocouples that monitor furnace temperatures are never calibrated after installation and TSI has no procedure specifying calibration. Since these chromelalumel thermocouples are exposed to flames reaching about 2000°F and remain in the furnaces for years, their ability to maintain calibration is questionable. Criterion V of 10 CFR Part 50, Appendix B requires that activities affecting quality be prescribed by documented instructions or procedures. TSI's failure to maintain calibration of the furnace thermocouples forms a portion of Nonconformance 91-01-02.

The NRC inspectors asked how TSI controls the calibration of its test and measuring equipment at nuclear power stations. The QC manager indicated that TSI has no inspection function or acceptance function at any site; therefore, any TSI test and measuring equipment at a site is not under TSI calibration control.

The NRC inspectors verified that TSI had a nonconformance program in place. In addition, the NRC inspectors reviewed several nonconformance notices and verified that TSI closed the notices on a timely basis and took adequate corrective actions.

The NRC inspectors verified that TSI had 10 CFR Part 21 procedures in place and met the posting requirements of 10 CFR Part 21. No notifications had been submitted to TSI's clients. Within the scope reviewed the inspectors did not identify any concerns with TSI's program for satisfying 10 CFR Part 21.

TSI's QA manager stated that about one dozen licensees had audited TSI's QA program. The NRC inspectors reviewed records of audits that TU performed at TSI between 1982 and 1989. TU's audits did not identify any major concerns with TSI's QA program.

TSI had not audited its material suppliers. TSI obtains commercial COCs and performs infrared spectroscopic analyses on all lots of material purchased for Thermo-Lag 330 use. The NRC inspectors verified that TSI had receiving records, QC reports, and COCs for the lot numbers selected for subliming powder and stress skin procurements. In addition, the NRC inspectors verified that a certified material test report from the mill was in the data package for the lot number selected for the stress skin.

Based on the observations reported above and the file review of POs for six commercial nuclear power plant sites, the NRC inspectors concluded that TSI's QA program for supplying Thermo-Lag 330 material was adequate with the exception of the two nonconformances cited in this inspection report.

3.5 <u>Customer Purchase Order (PO) Requirements</u>

This section of the inspection report addresses PO contractual requirements on TSI as observed by the NRC inspectors, with the exception of the on-site support requirements discussed in the next section. The content of TSI's Certificates of Conformance is also addressed.

The NRC inspectors reviewed records for all of the POs in TSI's files for Thermo-Lag 330 material for the following six commercial nuclear power plant sites:

Callaway Nuclear Power Generating Plant
Comanche Peak Steam Electric Station
Perry Nuclear Power Plant
River Bend Station
Susquehanna Steam Electric Station
Washington Nuclear Project, Unit-2 (WNP-2)

Site selection was based primarily on Thermo-Lag site problems reported in NRC Inspection Reports, NRC Information Notices and Licensee Event Reports. The inspectors were also interested in whether different PO QA criteria affected what TSI supplied, and had asked TSI to prepare a list of plants that specified various criteria including 10 CFR Part 50, Appendix B. TSI was unable to complete the list by the end of the inspection, partly because a typical plant file included either numerous POs or numerous PO change orders.

3.5.1 Commercial Grade PO Requirements

Procurements for the listed plants began between 1981 and 1984. For four plants (all except Comanche Peak and WNP-2) the initial procurements were by the architect-engineer or another contractor to the licensee. By the mid-to-late 1980s all six licensees were procuring directly from TSI. All of the procurements were commercial grade except for Comanche Peak, where all of the POs reviewed (except those for on-site services) invoked 10 CFR Part 50, Appendix B.

The typical PO covered both bulk material and prefabricated panels and conduit sections. Certification that the materials meet specified criteria, including TSI's QA/QC program, was often required. Material certifications are of limited value because the qualification type tests covered fabricated installation designs, not generic materials or the prefabricated panels and conduit sections supplied by TSI. Other criteria that some POs specified are identified below in the COC discussion.

The Callaway nuclear plant provided an example of a requirement for material certification. Daniel PO No. 7186-NS-87593, dated February 7, 1984, invoked Bechtel Specification No. 10466-E-097, "Technical Specification for Furnishing and Installation of Fire Barrier Materials for the Standardized Nuclear Unit Power Plant System (SNUPPS)," Revision 0, dated October 11, 1983. Section 4.1.b of the specification required the following: "Manufacturer's certification showing material has been tested and is qualified for use as 1-hour and 3-hour rated barriers by the applicable standards or codes."

The NRC inspectors also obtained a copy of a February 7, 1984, letter to Daniel from TSI's national sales manager which stated: "This will advise you that TSI's THERMO-LAG 330 Fire Barrier Materials Systems meets [sic] all the prerequisites delineated in the reference specification." The NRC inspectors also noted that the PO invoked no QA requirements on TSI (except repetition of the cited requirement to submit material certification), and that TSI's COC merely certified that the materials "meet TSI's manufacturing and written quality control specifications."

The inspectors reviewed Stone & Webster Engineering Corp. (S&W) PO No. 12210-30454, dated September 24, 1984, for the River Bend Station. The technical and QA requirements were specified per S&W Nonengineered Item Data Sheet 211.161, which described the materials and specified thickness ranges for prefabricated panels. One hour panels and shapes were to be 1/2 inch -0.00, +0.125 inch and three hour to be 1 inch -0.00, +0.250 inch. The NRC inspectors observed a TSI COC dated March 14, 1985, which certified only a 1.00 inch minimum thickness for a three hour panel.

3.5.2 Comanche Peak 10 CFR Part 50, Appendix B PO Requirements

The NRC inspectors found that POs for TU (the licensee for Comanche Peak) appeared to impose two types of additional requirements on TSI beyond the scope of the typical PO. First, TU's POs invoked the safety-related QA requirements of 10 CFR Part 50, Appendix B, on TSI's scope. Second, TU's POs imposed a specification which appeared to impact TSI's responsibilities for the applicability of qualification test reports and installation procedures to the plant installations of Thermo-Lag material.

The NRC inspectors reviewed TU PO No. CPF 1557-S, dated April 19, 1982. The PO and its supplements specified materials and technical assistance services for a Thermo-Lag 330 subliming coating envelope system for the Comanche Peak nuclear power plant. The PO specified that all materials and services must be in strict compliance with TU Specification 2323-MS-38H, "Cable Raceway Fire Barriers," Revision 1, dated April 2, 1982, (prepared by Gibbs and Hill, Inc.) and any subsequent revisions. Although the specification is labeled "Non-Nuclear Safety Related QA Program Applicable," the PO specified that "work performed herein shall be performed as applicable in compliance with T.S.I. Inc.'s nuclear quality assurance program manual" as qualified by the licensee. The PO also specified that "services shall be accomplished in accordance with T.S.I. Inc.'s written quality assurance program conforming to the requirements of ANSI [American National Standards Institute Standard] N45.2 [and] 10CFR50, Appendix B ... as applicable, subject to verification by [TU's] quality assurance department." The PO stated that the provisions of 10 CFR Part 21 may apply.

Specification 2323-MS-38H placed broad requirements on the vendor (and, in some cases, the "vendor/applicator"). Section 3.1.1 defined the vendor/applicator scope to include "the design, furnishing, quality assurance/quality control, and performance testing of all materials and components required for the cable raceway fire barriers." Section 3.3.1 required the vendor to "guarantee the satisfactory material performance, and installation instructions and procedures of all cable raceway fire barrier materials furnished." Section 3.4.1 invoked (without distinguishing between vendor and vendor/applicator) NRC Branch Technical Position APCSP 9.5.1, which included criteria for the design and qualification of fire barriers.

Section 3.7.1.1 of specification 2323-MS-38H required the vendor to "supply documented tests of product performance referencing the materials used, the type of installation and the method of application as a basis for meeting the requirements specified herein." Section 3.10.4 requires submittal for approval of "Certified test results which demonstrate that all fire barrier arrangements have been tested in accordance with the requirements of" the specification. These requirements contribute to the basis for Nonconformance 91-01-02 as defined elsewhere in this inspection report.

TU exercised its contractual right to approve documents, as evidenced by a TU letter to TSI dated June 22, 1989, subject:

"Notification of Document Status" for PO No. 665-71871, which showed general approval of six Industrial Testing Laboratories, Inc. (ITL) test reports; another test report; two TSI Technical Notes regarding thermal and dynamic loads and ampacity rating; and documents titled, "Determination of Chloride, Fluoride, Sodium and Silicate concentrations in Thermo-Lag 330-1 Subliming Coating," and "Summary of Ampacity Derating Tests." The NRC inspectors noted, however, that TU's letter did not address installation procedures or drawings.

By reviewing TU source inspection reports, the NRC inspectors verified that TU exercised its contractual right to perform source inspections prior to shipment, although TU sometimes waived that right. TU's source inspections included verification of thickness and weight measurements.

The NRC inspectors reviewed a November 10, 1989, TSI internal memorandum for PO No. 665-71871 to all quality control and production personnel. TSI's QC and production managers issued the memorandum to implement an agreement between TU and TSI to add additional steps to TSI's inspection program. Specifically, in addition to the normal 18-point thickness inspection of prefabricated panels, the memorandum specified additional thickness checks to be made along the panel edges to identify undesirable compressions. The weight of each prefabricated panel would also be recorded by the QC inspector on his acceptance tag (this was normally a go/no go signoff).

The NRC inspectors found another example of TU invoking Specification 2323-MS-38H. TU's PO No. 8 0029731, dated October 30, 1991, procured safety-related replacement parts from TSI. The PO invoked Pre-Engineered Item Data Sheet # NE30011, which stated in Section 1.2 that "products listed in the purchase order are identical to those products previously tested and supplied in accordance with TU Electric Specification 2323-MS-38H Revision 1."

The NRC inspectors noted that the Comanche Peak site used a Thermo-Lag installation procedure designated as "TU Electric - Generating Division, Engineering and Construction, Construction Department Procedure ECC 10.07, Application of Fire Protection Materials (for example, Revision 3 dated May 5, 1989)." This procedure did not reference any TSI documents, but did reference licensee drawings for Thermo-Lag installation details. Thus, despite the wording of Specification 2323-MS-38H, the NRC inspectors saw no evidence that TU relied upon TSI to guarantee the completeness of TU installation procedures. However, the inspectors did not review site records that might clarify this issue.

3.5.3 Certificates of Conformance (COCs)

The typical COC stated "this will certify that the materials listed above [or below] under purchase order number ______ meet

TSI's manufacturing and written quality control specifications." The COC also listed the materials shipped, showing product type, quantity, and batch or lot number; date; bill of lading number; and truckline. Each COC was signed by TSI's manager of quality control. Many COCs named TSI's QA manual and cited a specific controlled copy that had been issued to the customer. For Comanche Peak only, the COCs generally stated that 10 CFR Part 50, Appendix B and ANSI N45.2 applied.

The NRC inspectors observed numerous variations of the typical COC format. Often the <u>materials</u> were certified as being identical to those that had been qualification-tested (although the tests qualified only specific configurations). Some COCs named specific criterion documents, such as ASTM Standard E 119 and American Nuclear Insurers (ANI) Bulletin 5-79, with words such as, "when used in approved configurations." Additional standards addressed in this manner were ASTM E 84, "Surface Burning Characteristics of Building Materials," ANSI A2-1, and NRC Regulatory Guide (RG) 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Some COCs stated that the requirements of the PO were met. Some stated, under "product description," a 1.00 inch minimum thickness for three hour panels.

TSI also provided some Certificates of Analysis. Those observed covered density, pH, and sometimes leachable chloride content for material batches. TSI's QC manager told the NRC inspectors that TSI discontinued chloride analysis of Thermo-Lag material on November 20, 1989, because the leachable chloride limit never approached the 200 ppm limit specified in RG 1.36. Since that date TSI's COCs and COAs have not specified individual batch chloride tests, and TSI now recommends that customers desiring the analysis obtain it from another source.

3.6 On-Site Responsibilities

3.6.1 Discussions with TSI personnel

TSI usually contracts to perform on-site training of installation and quality control personnel provided by the licensee. TSI informed the NRC inspectors that it does not perform, inspect, or approve installation work. Occasionally, as at the WNP-2 and Comanche Peak plants, TSI personnel have been on-site for cumulative periods of more than a year. TSI's QA manager noted that such extended residence was sometimes the result of a licensee ensuring that a TSI representative would be available for training several groups of craftspersons, and that the representative might perform additional duties such as inventory monitoring. In this regard, the NRC inspector noted in the WNP-2 file an inventory list signed by the representative whose living expenses were billed to the licensee over an extended period.

TSI's QA procedures provide for the position of Manager of Field Service Operations, whose responsibility includes "exercising technical control over product application activities at the client nuclear plant site" (procedure NQAP 3-1, section 3.3.3). TSI's QA manager stated that TSI has never had a field service manager.

TSI regards training as a best-effort activity. Although trainees must pass a test, TSI stated that trainee retention is beyond TSI's capability. TSI stated that personnel to be trained are normally experienced in heating, ventilating, and air conditioning (HVAC) installations. Often on newer plants they are the personnel who installed the plant HVAC, penetration seals, and pipe wraps. Although TSI stated that many were journeymen and master craftsmen, TSI does not select the personnel or specify selection criteria.

The documentation of TSI's on-site training is poor. Prior to the inspection TSI provided to the NRC a two-page training outline that contained no installation information, but merely named various applications (such as "prefabricated panel design for junction boxes - installation of one hour fire barrier design"). During the inspection, the TSI QA manager provided a new informal "Applicator Training Program Lesson Plan." In addition to simply naming the applications covered, the new plan also named aspects of each installation (such as "spacing of tie wire, banding and fasteners" and "joint filling and sealing"). TSI still provides no written training documentation covering concerns such as those noted in the following paragraphs. The TSI position is that the customer's installation procedures, supplemented by hands-on training of customer-selected personnel in the general nature of Thermo-Lag 330 installations and the customer's QC inspection of the plant installations, should be sufficient to ensure adequate installation.

TSI routinely supplies customers with TSI Technical Note 20684, "Thermo-Lag 330 Fire Barrier System Installation Procedures Manual - Power Generating Plant Applications." The latest version is Revision V, November 1985. This document, and its predecessors, were approved for insurance purposes by ANI. TSI stated that the document has not been revised since ANI suspended its approval activities. However, as a result of discussions with the NRC a new revision is scheduled for issue by January 31, 1992. Examples of planned additions cited by TSI were specifying curing time, redefining how to seal joints and cut the stress skin, and adding a note to wear goggles.

TSI personnel characterized Technical Note 20684 as a generic document, and frequently referred to it as an application guide. TSI stated that architect-engineers or licensees provided the plant-specific installation manuals. TSI might be asked to comment on a plant-specific manual, and would comment on whether a configuration had been tested. TSI stressed that this would be an opinion, not a responsibility; even if a similar configuration had been tested, analysis would be required. TSI considers Technical Note 20684 to be accurate, and as complete as necessary when supplemented by training of competent crafts personnel.

The NRC had previously informed TSI that Technical Note 20684 did not cover certain important installation characteristics, such as which side of a panel should be scored or V-grooved for bending, when pre-buttering would be necessary for joints, and the maximum allowable thickness of material. TSI responded that these matters were all covered in hands-on training. During this inspection the inspectors noted a deficiency in Technical Note 20684. The second and third paragraphs of Section 1.0, page II-2, specifies that scored corners and joints of Thermo-Lag panel sections are to be filled with trowel grade material <u>after</u> the panel sections are tied or banded around a cable tray. However, at that stage it would be impossible to fill the seams with trowel-grade material. These types of deficiencies allow plant installation configurations that may not be represented by qualification type test specimens.

3.6.2 PO Requirements for On-Site Responsibilities

The NRC inspectors' review of files for the six plant sites generally supported the position presented by TSI personnel. POs were non-safety related and contained no QA or QC requirements for on-site work; often the PO specified that site procedures would govern. Certain POs for Comanche Peak were particularly limiting, containing statements such as "neither TSI nor the TSI loaned employees were providing engineering services in connection with the work of the loaned employees, and TSI had no responsibility or liability for the installation or design of Thermo-Lag material." Some POs specified additional requirements for on-site assistance by TSI, as described below.

For Comanche Peak, TU PO No. CPF 1557-S, dated April 19, 1982, and its supplements specified both materials and technical assistance. The PO specified compliance with Gibbs and Hill Co. Specification 2323-MS-38H, "Cable Raceway Fire Barriers," Non-Nuclear Safety Related, Revision 1, dated April 2, 1982, and any subsequent revisions. Paragraph 3.3.1 required the vendor to guarantee satisfactory material performance and installation instructions and procedures for all cable raceway fire barrier materials. Paragraph 3.10.4 required the vendor to submit drawings, documents, and procedures with its proposal, for approval.

For WNP-2, PO No. 37115 dated July 28, 1982, specified training services. It also required that the TSI technical service representatives "shall assure the raceways coated with Thermo-Lag meet the requirements as previously tested (sample articles) by TSI Inc." It also specified TSI support of the owner's commitments to ANI with respect to the use of Thermo-Lag materials, and that daily working direction would be provided by the owner's construction manager. There were no QA or QC requirements.

Also for WNP-2, Contract No. C20610, as proposed to TSI in 1986, required TSI "corporate approval of specific configurations of Thermo-Lag application to steel penetrating the fire barrier to assure compliance with tested configurations" and to "perform regular inspections of installation and provide Certificates of Conformance to 'three-hour' fire protection requirements at the completion of installation." TSI's June 10, 1986, letter to WPPSS took the following exceptions: "TSI is not an approving authority for Nuclear Power Generating Plants. TSI will provide, however, a Certificate of Conformance, when required, with regard to compliance of the installed configurations with those previously tested" and "Regular inspections of the installation can be provided by our field service engineer while onsite at WPPSS. A Certificate of Conformance can also be provided to the test configurations following procedures delineated in TSI's Quality Assurance/Quality Control Operating Procedures Manual. After the completion of the installation, additional inspections can also be arranged in accordance with a mutually agreeable schedule and at our standard Field Service Engineering rates." WPPSS's letter to TSI dated June 13, 1986, transmitted an executed original of the contract, and stated that the TSI exceptions were acceptable and TSI's letter would be retained in the contract file along with the unmodified contract. These WNP-2 provisions, if implemented, appear to comprise limited exceptions to TSI's general policy limiting on-site support.

For Susquehanna, Contract No. 8856-F-56718, dated October 15, 1981, specified that a TSI field service representative would be required on-site for approximately 12 weeks. Schedule A to Technical Services Agreement 8856-FTSA-22, dated November 12, 1981, specified that TSI must "provide all necessary technical and professional services required to support and document the installation of" TSI's Thermo-Lag 330 subliming coating system on electrical raceways in accordance with Bechtel Technical Specification 8856-E-E61, Revision 1, dated November 12, 1981. Schedule A also required TSI to furnish "all personnel and test equipment necessary to document and monitor the application of T.S.I., Inc.'s QA/QC program and application procedures." The NRC inspectors noted that Section D.1.(b) of Schedule A identified TSI's QA program manual as the "application procedures." The only QA requirements were for TSI's program.

TSI's QA manager stated that TSI did not supervise or perform any quality control functions or installation at Susquehanna. The NRC inspectors found only one invoice, Number FS-104 dated November 16, 1981, for field services; the span was 12 days. Although the invoice did not indicate what services were provided, TSI's QA manager stated that the service was limited to training on setting up spray equipment and on the proper method of spraying Thermo-Lag on stress skin. The contract also stated under the warranty clause that the buyer assumed all responsibility and risks for proper application, safety, and use of the material. Based on this information, the NRC inspectors concluded that TSI's role at the Susquehanna site appeared to be limited to non-safety related training services.

For Callaway, PO No. 7186-NS-87593, dated February 7, 1984, from Daniel International Corp. specified field services, with no QA or QC requirements. Daniel was the construction contractor, although documents indicated that Thermo-Lag installation was actually performed by Owen-Corning Fiberglas Corp., Power and Process Contracting Services. TSI furnished an installation procedure, TSI Technical Note 11266 titled "Installation Procedures for the "Ready Access Designs" of the Thermo-Lag 330-1 Subliming Fire Barrier Systems" to Union Electric Co. (the licensee) as a guide for use in installing Thermo-Lag materials at the Callaway plant. Bechtel (the architect-engineer) personnel changed the TSI Technical Note number from 11266 to [illegible]-1001 and made numerous pen and ink changes in the procedure. Daniel Field Change Request (FCR) No. 2FC-3247-E, incorporated a marked copy of the technical note which had been reviewed and signed by TSI's QA manager on March 19, 1984. Bechtel indicated their review and approval on March 20, 1984, by initialing the changes in the application guide and the approval block of the FCR. TSI's QA manager stated that TSI's role in producing this plant-specific installation manual remained advisory, and TSI did not assume responsibility for the manual's application, as described above.

Based on the file reviews and discussions with TSI personnel reported above, the NRC inspectors concluded that TSI appeared to satisfy its contractual requirements for on-site support at the commercial nuclear power plants reviewed during the inspection. However, the support actually provided, as described by TSI, essentially placed full installation responsibility on the licensee and its contractors. TSI clearly resisted customer attempts to increase TSI's role.

TSI's installation guide lacked considerable detail necessary for installation; TSI stated that it accepted only an advisory role in applying qualification tests to plant installations; the content of training provided by TSI was not documented; TSI had no prerequisites for the selection of installation or site inspection personnel; and TSI did not appear to be involved in determining if the inspection personnel received any training. Thus, TSI did not appear to exercise control over installed Thermo-Lag 330 fire protection systems except for the material itself.

3.7 Qualification Type Testing

ASTM E 119 fire endurance qualification type tests have been performed on several Thermo-Lag 330 installation designs at TSI and elsewhere. This inspection only addressed testing at TSI, which is performed under the observation of Industrial Testing Laboratories, Inc. (ITL) as addressed in Section 3.8 of this inspection report. The NRC inspectors did not witness any qualification testing. TSI personnel described test preparations as follows.

Either the customer (licensee or architect-engineer) or TSI prepares the test plan. TSI and the customer also determine the general design of the test specimen and the location of thermocouples. The test plan does not give full details of the test specimen construction; as-built information may be sketched in the daily work sheets for the test. TSI personnel stated that prior to 1986 ANI approved the test plans, witnessed the test specimen construction and installation, witnessed performance of the tests, and approved the test report for insurance purposes. Customers have also witnessed testing.

The test specimen is assembled by a TSI crew of manufacturing personnel assigned to the test, using materials selected from the QA-approved inventory (which normally is quite small, since materials are basically mixed and fabricated to order). No attempt is made to select worst-case or other specific characteristics. TSI builds the test specimens in a small area near the test furnace. TSI maintains current calibrations of data logging instruments, as described in the QA program section of this inspection report (section 3.4). TSI has two furnaces. Usually the larger and better-instrumented furnace is used for nuclear tests.

Section 3.8 of this inspection report describes the NRC inspector's review of two qualification test reports, dated 1987 and 1990. Neither test plan fully described the design of the test specimen. For example, only a few dimensions were specified, and filling of joints was not described in detail. Some, but not all, of the omitted information was provided in as-built specimen descriptions in the daily record sheets appended to the test report. Criterion V of 10 CFR Part 50, Appendix B requires that activities affecting quality be prescribed by documented instructions or procedures. For safety-related procurements, TSI's failure to adequately specify specimen construction in the qualification test plans forms a portion of Nonconformance 91-01-02.

TSI also has performed ampacity derating tests. The customers designed the tests and supplied the cable samples. TSI has not performed ampacity derating calculations, but under a present contract from Gulf States Utilities is arranging for a local university to perform them.

TSI maintains a complete set of qualification type test reports, both ITL and others, arranged chronologically in a file cabinet.

3.8 Industrial Testing Laboratory Role

TSI has stated that several ASTM E 119 type qualification tests of Thermo-Lag installation design specimens have been conducted under the independent auspices of Industrial Testing Laboratories, Inc. (ITL) of St. Louis. For example, a TSI document titled "Synopsis on the Thermo-Lag 330 Fire Barrier System for Power Generating Plant Applications, 10 February 1987," summarizes and references various tests. It makes the following statement regarding fire endurance tests on page two: "The above tests were performed under the supervision and total control of an ANI accepted third party, independent testing laboratory, Industrial Testing Laboratories, Inc., who also published the test results."

In order to assess the scope of ITL's efforts, the NRC inspector interviewed an ITL representative (a professional engineer) together with TSI's president. Although it has not performed fire barrier endurance tests, ITL has conducted numerous tests, including flame tests, for a wide variety of customers. ITL first tested Thermo-Lag material for aerospace applications in the late 1950s. ITL is listed on TSI's Approved Vendor List based on performance history, with no record of an audit. Criterion V of 10 CFR Part 50, Appendix B requires that activities affecting quality be prescribed by documented instructions or procedures. For safety-related procurements, TSI's failure to audit ITL forms a portion of Nonconformance 91-01-02.

The TSI president stated that TSI has an oral agreement with ITL that specifies rates but not work scope. Criterion V of 10 CFR Part 50, Appendix B requires that activities affecting quality be prescribed by documented instructions or procedures. For safety-related procurements, TSI's failure to contractually specify ITL's role in fire endurance qualification tests forms a portion of Nonconformance 91-01-02.

ITL does not participate in preparation or approval of the test plan, the design of the test specimen, or the location of thermocouples. ITL does not witness the construction of the test specimens, and at TSI's option may or may not witness installation of the specimen into the furnace. The ITL representative stated that he does not compare the test specimen dimensions with the test plan or daily work sheets. ITL also does not review calibration records for the test instrumentation.

ITL's role is observing the actual performance of the test. The ITL representative stated that he reviews the criteria documents including the test plan, discusses the text with the test supervisor to ensure understanding, witnesses performance of the test, signs the daily work sheets, and collects and issues the raw data to ITL, TSI, and TSI's customer. The ITL representative stated that his role in the test ended with issuing the raw data; his function was to witness the test and verify that it was conducted as it was supposed to be, according to the test plan and other criteria documents. He was never involved in issuing a test report. TSI's president stated that TSI writes the test report text, types the report including the raw data, and obtains its customer's approval. The report is then given to ITL for what was described as a minimal review, and issued by ITL.

The NRC inspector questioned the ITL representative and TSI's president concerning a 1990 fire endurance test that had been observed by the ITL representative interviewed. The inspector noted that the raw data package highlighted an out-of-limit temperature that was not correspondingly emphasized in the draft test report (the actual number was included in the typed data, but its significance was not noted there). The ITL representative stated that his activities would not include such a comparison. TSI's president stated that the discrepancy would be identified in TSI's review of the draft report and corrected before issue.

In reviewing a typical fire endurance test report, ITL Report No. 87-5-76 dated June 1987, the NRC inspector commented that the report's appearance suggested that ITL's role may have been greater than it really was. For example, the cover sheet bears ITL's name and logo, but not TSI's. The title page is similar, except that it does identify TSI by name and address as the "test location." It also bears an ITL disclaimer concerning the use of the report, and the only approval signature is that of ITL's director. A reader would not know that the report had actually been written and typed by TSI, or that ITL's role in the test was essentially limited to witnessing data acquisition. The ITL representative and TSI president did not dispute these comments.

The inspectors found only one requirement for test laboratory independence in the files reviewed during the inspection. TU PO No. CPF 1557-S invoked Gibbs & Hill Specification 2323-MS-38H, Revision 1, which stated in section 3.7.2.1 that "fire and hose stream tests shall be performed and documented by a recognized independent testing laboratory." The specification in section 3.4.1.4(b) also invoked NRC Branch Technical Position APCSP 9.5.1, which defines a fire barrier rating in hours as established by a nationally recognized testing laboratory. The NRC inspectors were unable to determine an NRC requirement was actually violated in this regard. However, the inspectors believe that the appearance of the test reports and the representation of them as ITL reports could be misunderstood by users.

3.9 Conclusions

Section 3.3 of this report cites Nonconformance 91-01-01 concerning TSI's failure to procedurally require minimum thickness and maximum weight measurements for prefabricated, safety-related panels and conduit sections. Sections 3.5, 3.7, and 3.8 provide a basis for Nonconformance 91-01-02 involving TSI's failure to adequately control qualification testing for NRC licensees such as Texas Utilities, as identified in section 3.5.2.

Based on the file reviews and discussions with TSI personnel reported above, the inspectors found no other violations of NRC requirements for supplying materials and qualification documentation to commercial nuclear power plants. However, the inspectors were also concerned by the limited scope of installation support that TSI provides to its customers, as discussed in Section 3.6.

4 PERSONNEL CONTACTED

Thermal Science, Inc.:

* + R. Feldman, President
* + R. A. Lohman, Manager, Quality Assurance
* + B. E. Evans, Manager, Quality Control
* + M. G. Murphy, Administrator

Industrial Testing Laboratories, Inc.:

D. Wylan, Staff Consultant

US NRC:

\+ C. A. VanDenburgh, Section Chief
\+ L. R. Plisco, Section Chief
\+ K. S. West, Senior Project Manager

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* Attended the entrance meeting on December 16, 1991
\+ Attended the exit meeting on December 20, 1991

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

FEB 21 1992

Docket No. 99901161/89-01

Mr. Gary Shroyer
Tyler-Dawson Supply Company
Post Office Box 3067
Portland, Oregon 97208

Dear Mr. Shroyer:

SUBJECT: RELEASE OF NRC INSPECTION REPORT

This letter addresses the inspection of your facility at Portland, Oregon, conducted by Mr. J. Petrosino of this office on April 11, 1989, and the discussions of his findings with you at the conclusion of the inspection.

The inspection was performed as a follow-up to an NRC concern regarding potentially substandard valves that may have been supplied to nuclear power plants through valve material suppliers. This NRC concern is discussed in detail in NRC Information Notice (IN) 88-48 and its supplements. Areas examined during the NRC inspection and our findings are discussed in the enclosed report. This inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector. Release of this report was delayed during NRC's review of nonconforming and substandard vendor products.

Within the scope of this inspection, we found no instance in which you failed to meet NRC requirements. 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 into the NRC's Public Document Room.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report No. 99901161/89-01

ORGANIZATION: TYLER-DAWSON SUPPLY COMPANY
PORTLAND, OREGON

REPORT NO.: 99901161/89-01	INSPECTION DATE: April 11, 1989	INSPECTION ON-SITE HOURS: 3

CORRESPONDENCE ADDRESS: Tyler-Dawson Supply Company
Post Office Box 3067 (97208)
5051 North Lagoon
Portland, Oregon 97217

ORGANIZATIONAL CONTACT: Mr. Gary Shroyer
TELEPHONE NUMBER: (503) 289-9145

NUCLEAR INDUSTRY ACTIVITY: Currently Tyler-Dawson infrequently supplies commercial grade products for use at the Trojan Nuclear Plant.

ASSIGNED INSPECTOR: [signature] 6/12/89
J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S):

APPROVED BY: C E Anderson 6/12/89
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21 and Appendix B to 10 CFR Part 50.

B. SCOPE: The purpose of this unannounced inspection was to determine whether Tyler-Dawson has purchased any valves from CMA International, Incorporated of Vancover, Washington and to determine if those valves, if any, were supplied to any commercial nuclear power plant.

PLANT SITE APPLICABILITY: None identified during inspection.

ORGANIZATION: TYLER-DAWSON SUPPLY COMPANY
PORTLAND, OREGON

REPORT NO.: 99901161/89-01	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED/OPEN ITEMS:

None

D. PREVIOUS INSPECTION FINDINGS:

No previous inspections have been performed.

E. OTHER COMMENTS AND OBSERVATIONS

1. Background

NRC Information Notice (IN) 88-48, dated July 12, 1988, and Supplement 1 of IN 88-48, dated August 24, 1988, discussed a potential problem concerning Vogt 2-inch valves (Vogt Figure No. SW-13111), which were leaking steam around the bonnet and packing. The valves were purchased by Pacific Gas and Electric (PG&E) from Western Valve Supply Company in California. Although supplied as new, the valves were actually drop shipped from a valve salvage and refurbishment company in Vancouver, Washington [CMA International, Inc. (CMA)]. A Henry Vogt Company representative examined the valves at Diablo Canyon nuclear power plant and determined that they had not manufactured the subject valves.

The valves appear to be counterfeit based on the following: (1) the Vogt name was die-stamped instead of being forged onto the side of the valve body; (2) Vogt valves have round bonnet flanges whereas the subject valves have square bonnet flanges; (3) the subject valves have swing gland bolting which is not used by the Henry Vogt company and; (4) the end-to-end dimensions of the valves in question are shorter than the Vogt SW-13111.

ORGANIZATION: TYLER-DAWSON SUPPLY COMPANY
PORTLAND, OREGON

REPORT NO.: 99901161/89-01	INSPECTION RESULTS:	PAGE 3 of 3

2. Tyler-Dawson/Am-Fac Company

The inspector conducted discussions with Mr. Gary Shroyer of the Tyler-Dawson (TD) Company regarding business activities of the Am-Fac Company with CMA. Mr. Shroyer stated that TD took over the Am-Fac business and facility on July 18, 1988. At that time TD terminated the Am-Fac Branch Manager, Mr. J. Dunlap, and appointed Mr. T. McMullen, former M-Co Company President, as the TD Office Manager. Mr. Shroyer started with Am-Fac in approximately 1979 as an outside sales representative who usually deals with the nuclear plants and has continued employment with TD in the same capacity. In addition to Mr. Shroyer staying on at the former Am-Fac facility, Messrs. T. Brynelson, J. Wroe, and P. Williamson also stayed on and are employed by Tyler-Dawson. Sometime just prior to TD taking control of its new facility, Am-Fac collected and sent all customer records and documents to its corporate offices in Honolulu, Hawaii or Folson, California.

3. Inspection Activities

A review of the vendor lists was conducted. A February 1, 1988 Am-Fac vendor list was reviewed and found to list CMA as an authorized vendor. TD personnel stated that Am-Fac was typically conducting business with CMA. However, the current TD electronic vendor listing does not contain CMA as an authorized vendor. TD personnel stated that they would not typically nor do they remember buying any components from CMA since July 1988.

Therefore, no determination could be reached regarding whether or not suspect CMA parts were supplied to any nuclear facility by the Am-Fac Company.

F. PERSONNEL CONTACTED:

G. Shroyer, Sales Representative, Tyler-Dawson
T. Brynelson, Sales Representative, Tyler-Dawson

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

February 6, 1992

Docket No. 99900005

Mr. Ronald H. Koga, Manager
Columbia Plant
Westinghouse Electric Corporation
Commercial Nuclear Fuel Division
Drawer R
Columbia, South Carolina 29250

Dear Mr. Koga:

SUBJECT: NOTICE OF NONCONFORMANCE
(NRC INSPECTION REPORT NO. 99900005/92-01)

This letter addresses the inspection of your facility at Columbia, South Carolina conducted by Mr. S. L. Magruder, Mr. K. R. Naidu and Mr. R. K. Frahm, Jr. of this office on January 13-17, 1992, and the discussions of their findings with you and your staff at the conclusion of the inspection. The purpose of the inspection was to review Westinghouse's Commercial Nuclear Fuel Division (WCNFD) plant operations and quality assurance program.

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

The team noted several strengths during the inspection, especially WCNFD's internal audit program. The WCNFD Audit Commitment Tracking System and the level of knowledge of the technicians and operators interviewed during the inspection were also considered strengths.

During this inspection one nonconformance was identified. In particular, it was found that an operator may have performed work under a superseded procedure since several weeks elapsed before the operator acknowledged the revision to the procedure. The specific finding and reference to the pertinent requirement 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 its enclosures will be placed in the NRC Public Document Room.

Sincerely,

Leif J. Norrholm

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosures:
1. Notice of Nonconformance
2. Inspection Report No. 99900005/92-01

ENCLOSURE 1

NOTICE OF NONCONFORMANCE

Westinghouse Electric Corporation
Commercial Nuclear Fuel Division
Columbia, South Carolina
Docket No.: 99900005

Based on the results of an NRC inspection conducted on January 13-17, 1992, it appears that a certain activity was not conducted in accordance with NRC requirements. The NRC has classified this item, as set forth below, as a nonconformance to the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50), imposed on Westinghouse's Commercial Nuclear Fuel Division (WCNFD) by contract, and WCNFD's internal policies and procedures.

Criterion V of Appendix B to 10 CFR Part 50 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 and or drawings."

Section 7.1.1.C of WCNFD procedure CA-006, "Columbia Plant Training Policy," Revision 3, dated October 12, 1990, states, in part, that "the Section Manager ensure and document that training related to a new procedure or procedure change is accomplished in a timely manner ("timely" will be interpreted as routinely within five working days of issuance, but required prior to performing the new/changed task)."

Contrary to the above, an operator performed a procedurally controlled manufacturing operation for several weeks before acknowledging a revision to the procedure. Specifically, an operator performed the preplug/preweld operation in accordance with Rev. 3 of Manufacturing Operations Procedure (MOP) 750605, "Automatic Welding of Fuel Tubes on Preplug/Preweld Line," from July 3, 1991, to July 25, 1991, before acknowledging Rev. 4 to the procedure which had been issued on July 3, 1991. (92-01-01)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Safeguards, 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 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 6th day of February, 1992

ENCLOSURE 2

ORGANIZATION:	WESTINGHOUSE ELECTRIC CORPORATION COMMERCIAL NUCLEAR FUEL DIVISION COLUMBIA, SOUTH CAROLINA
REPORT NO.:	99900005/92-01
CORRESPONDENCE ADDRESS:	Mr. Ronald H. Koga, Manager Columbia Plant Westinghouse Electric Corporation Commercial Nuclear Fuel Division Drawer R Columbia, South Carolina 29250
ORGANIZATIONAL CONTACT:	Mr. Richard W. Pensak Quality Assurance Manager
NUCLEAR INDUSTRY ACTIVITY:	Nuclear fuel assembly supplier.
INSPECTION CONDUCTED:	January 13-17, 1992

Stewart L. Magruder — 2/5/92

Stewart L. Magruder, Team Leader — Date
Special Projects Section
Vendor Inspection Branch (VIB)

Kamalakar R. Naidu, VIB
Ronald K. Frahm, Jr., VIB

Gregory C. Cwalina — 2/6/92

Gregory C. Cwalina, Chief — Date
Special Projects Section
Vendor Inspection Branch

INSPECTION BASES:	10 CFR Part 21 and 10 CFR Part 50, Appendix B
INSPECTION SCOPE:	Review plant operations and Westinghouse's Commercial Nuclear Fuel Division (WCNFD) quality assurance program. Also, review verification and testing of WCNFD products and actions taken in response to recent fuel failures.
PLANT SITE APPLICABILITY:	Numerous Pressurized Water Reactor (PWR) sites.

1 INSPECTION SUMMARY

1.1 Nonconformance 99900005/92-01-01

Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 7.1.1.C of WCNFD procedure CA-005, "Columbia Plant Training Policy," Revision 3, dated October 12, 1990, an operator performed the preplug/preweld operation for several weeks in accordance with an outdated revision to the governing procedure before acknowledging the correct revision to the procedure.

2 STATUS OF PREVIOUS INSPECTION FINDINGS:

There were no open findings to address.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

The NRC inspectors informed WCNFD staff of the scope of the inspection, outlined areas of concern, and established working interfaces during the entrance meeting on January 13, 1992. On January 17, 1992, the NRC inspectors summarized the results of the inspection for WCNFD management during the exit meeting.

3.2 Background

WCNFD produces fuel for the majority of Pressurized Water Reactors (PWRs) in the United States. The Columbia Plant of the WCNFD receives raw materials such as UF_6 gas and zircalloy bar and sheet stock and converts them into fuel pellets and subcomponents of the fuel assembly. Final products such as complete fuel assemblies and control rods are produced at the Columbia Plant by combining the products manufactured at the plant with finished products purchased from outside vendors. Zircalloy tubing used for fuel cladding is manufactured at another WCNFD plant in Blairsville, Pennsylvania. This inspection was intended to provide the NRC inspectors with an overview of the operations at the facility and an opportunity to assess the effectiveness of the quality assurance program. It also provided an opportunity to investigate some specific concerns related to the performance of WCNFD fuel in operating reactors.

3.3 Internal Audits

The NRC inspectors reviewed the WCNFD internal audit program during the inspection. The program was found to be well organized and well run. The following recently completed audits were reviewed:

AUDIT NO.	10 CFR PART 50 APPENDIX B CRITERIA COVERED
91-005	3, 4, 5, 8, 15, 16 & 17
91-009	8, 10, 13 & 14
91-011	5, 6, 8, 9, 10, 12 & 14

The audits were conducted by a three member team for a full week and were very thorough and in depth. It was apparent from the audit reports that the auditors were given the independence and authority to look at anything they needed and to be candid in their report. One of the audits reviewed was led by a WCNFD employee who does not work at the Columbia plant. Each audit produced several findings which were tracked by WCNFD's Audit Commitment Tracking System. The NRC inspectors reviewed WCNFD's responses to the findings and determined that, in general, appropriate corrective actions had been specified. The internal audit program was considered to be a strength by the NRC inspectors.

3.4 Nonconformance/Corrective Action Program

The NRC inspectors verified the implementation of procedures to identify nonconforming items, the actions taken to correct the nonconformances, and the corrective action taken to preclude repetition. The NRC inspectors' verification was limited to selectively reviewing Deviation Notice - Disposition Requests (DNDRs), and Quality Control Deviations or Notifications (QCDNs) issued in 1991.

3.4.1 Review of DNDRs

The NRC inspectors reviewed procedure TA-203, "Deviation Notice - Disposition Request, Transfer Request," Rev. 5, dated September 27, 1991. This procedure establishes the requirements to identify parts or materials which do not conform to product specifications, and disposition them appropriately. It was found by the NRC inspectors to be adequately detailed.

In accordance with the procedure, DNDRs can be initiated only on parts and materials (1) which do not adversely affect subsequent manufacturing or assembly operations and (2) which are considered usable by manufacturing and cannot be made to specification. Parts and materials which obviously do not meet the above criteria for further use are scrapped. The inspector reviewed 10 DNDRs and determined that the preparation and disposition complied with the following attributes in procedure TA-203:

- The originator of the DNDR, and the shop order related to the deviation, were identified.
- The relevant specification, and the quality control check identified in the specification against which the deviation was observed, were stated.
- The deviation was adequately described.
- The action taken to correct the deviation was stated.

The NRC inspectors discussed the 10 deviations with the manufacturing or product assurance engineers responsible for initiating the DNDRs to determine if appropriate corrective action had been taken. DNDRs D91-014-01 of February 18, 1991, D91-016-01 of April 8, 1991, and D91-019-01 of

April 16, 1991, identified deviations in either the size or the axial location of thimble tube bulges. Even though the deviation and the actions taken to correct the deviation were stated, the actions taken to preclude repetition were not identified on the DNDR.

Additional information on this issue was obtained through discussions with the responsible engineer. Tube inserts placed in the grids are deformed by introducing a bulge with a special tool after inserting the thimble tube. Bulges are formed on either side of the grid. The key parameters that affect the joint strength of the insert, the thimble tube, and the grid are bulge diameter and bulge position relative to the end of the insert. The engineer explained that DNDR D91-014-01 was issued to correct previously produced undersized bulges. However, after the adjustments were made, the diameters of the samples produced were not measured to determine whether the adjustments were adequate to ensure that the size of the bulge was acceptable. The applicable Manufacturing Operating Procedure (MOP), MOP-731109, was subsequently revised to require the size and location of the bulge sample to be measured and verified after each adjustment. In addition, Tool Room Internal Procedures (TRIPs) 418 and 419 were revised to require the tool maker to verify the tightness of the pull rod and tool shaft locking nuts every day at the beginning of the operation and to require the verification of probe block slot dimensions.

The NRC inspectors observed activities related to the bulging operations and determined that the established procedures were being adhered to and the diameters and axial locations of the bulges met the applicable specification requirements. The NRC inspectors also discussed the deviations identified in the other seven DNDRs and determined that adequate corrective action had been taken to preclude repetition of the nonconforming conditions identified in the DNDRs.

The NRC inspectors were concerned that the corrective action taken to preclude repetition of the deviation was not adequately described on the DNDRs reviewed. This concern was raised, as an observation, with plant management.

3.4.2 Review of Quality Control Deviation or Notifications (QCDNs)

The NRC inspectors reviewed procedure QA-617 "Quality Control Deviation or Notification," Rev. 0, dated June 21, 1991. This procedure establishes the requirements for using a QCDN to document (1) deviations to quality control instructions (QCIs) or drawings, (2) notification of concerns or, (3) unusual events. It was considered by the NRC inspectors to provide adequate guidance for WCNFD personnel.

The NRC inspectors reviewed 10 QCDNs to determine whether the procedure was being properly implemented. QCDN 24485, dated December 4, 1991, identified that absorber tubes with different traceability numbers were found in one box. The cause was identified to be that the relevant MOP was not followed. Corrective action taken was to instruct the operator to follow the MOP more closely. In another example, QCDN 10734, dated January 22, 1991, identified that the length and width of zircalloy grid straps exceeded the values stated in the relevant drawing. This deviation was reviewed by the Engineering

Department. Engineering Change Notice (ECN) 26228 was subsequently issued to accept the straps. The NRC inspectors concluded that the QCDN program was being effectively implemented. The same concern about the adequacy of comments in the corrective action block of the QCDN form as was raised regarding DNDR forms was also noted.

The NRC inspectors also reviewed the following Quality Control Instructions (QCIs) used to evaluate and disposition the ten QCDNs and determined that the instructions and references contained in the QCIs provided adequate technical guidance to perform thorough inspections.

QCI 928025 "Helium Leak Test For Batch Systems," Rev. 7, dated November 18, 1991. Provides instructions to operate and calibrate the Helium leak tester and criteria to test fuel and non-fuel bearing rods.

QCI 108819 "Corrosion Evaluation and Disposition Practices," Rev. 27, dated October 14, 1991. Provides guidelines for proper processing, evaluation, disposition, and documentation of weld samples which have been subjected to corrosion tests.

QCI 929101 "MAP [Manufacturing Automated Process] Verification of Standard Run and Line 9 UT [ultrasonic test] Process Control," Rev. 37, dated January 14, 1991. Provides procedures for accessing the MAP database to verify that the UT machine standards were run at the proper time and that the UT machines read these standards properly during the startup period of the MAP line to UT the fuel rod welds made in line 9.

QCI 108820 "Penetration Disposition Practices," Rev. 19, dated October 14, 1991. Provides guidelines for proper processing, disposition, and documentation of the results of examining the adequacy of penetration in weld samples collected from seal and girth welds made on fuel rods and non-fuel bearing rods manufactured from either stainless steel or zircalloy.

QCI 927103 "Non-fuel Rod Manufacturing In-process Inspection," Rev. 76, dated November 16, 1991. Provides instructions for quality control inspectors to perform in-process inspections on secondary source seal welds produced in the operating lines where non-fuel rods are manufactured and to check the test equipment used to examine the welds.

QCI 933017 "Zirc Grid Strap - Final Inspection," Rev. 53, dated May 13, 1991. Provides instructions for quality control inspectors to verify various attributes during final inspections on inner and outer zircalloy grid straps.

QCI 980212 "Boron Coated Pellets," Rev. 8, dated August 21, 1991. Provides the acceptance criteria for inspection of the coater run and pellet salvage operations in the production of Integrated Fuel Burnable Absorber (IFBA) pellets.

3.5 Chemistry/Metallurgical Laboratory Operations

The NRC inspectors observed operations in both the Chemistry (Chem) and Metallurgical (Met) Laboratories during the inspection. Processes observed in the Met Lab included tests for end plug weld penetration and autoclave corrosion tests. The NRC inspectors also reviewed two procedures; QCI 108819, "Corrosion Evaluation and Disposition Practices," Rev. 27, dated October 14, 1991, and QCI 108857, "Autoclave Operating Procedure for Aqueous Corrosion Testing," Rev. 16, dated February 6, 1990. The procedures were well written and appeared to be closely followed by the lab technicians.

Processes observed in the Chem Lab included calibration checks on hydrogen detection equipment, and analysis of impurities in fuel pellets. The NRC inspectors also reviewed several Columbia Operations Chemistry Lab (COCL) procedures including: A-01, "Determination of Metallic Impurities in Uranium Compounds," Rev. 10, dated October 13, 1989; U-05, "Determination of Total Uranium in UF_6," Rev. 6, dated December 15, 1988; and I-03, "Determination of H_2 in Uranium Oxides, Ceramics, and Metals," Rev. 14, dated August 16, 1988. The procedures were adequately detailed and the lab technicians observed by the NRC inspectors followed them very closely. The operations observed in both labs were well controlled and the technicians were very knowledgeable.

The NRC inspectors did discuss an observation regarding the control of procedures in the Chem Lab with the plant management. A review of the index of COCL procedures revealed that the annual review of the procedures had not been completed for 1991. Discussions with the lab manager indicated that the review should have been completed in December 1991, but was not due to a backlog of work. WCNFD procedures do not specifically require an annual review, however, it has been their practice to do so. Plant management was aware, prior to the inspection, that the review was delinquent, and the review was completed prior to the end of the inspection.

3.6 Recent Fuel Failures

The NRC inspectors discussed recent fuel failures reported at Wolf Creek (reference Event Notification Number 21986, dated October 11, 1991, and Region IV Morning Report dated October 17, 1991), Comanche Peak (reference Region IV Morning Report, dated October 28, 1991), and Zion (reference Event Notification Number 18413, dated May 8, 1990, and Region III Morning Report, dated May 10, 1990) with WCNFD personnel. They were very knowledgeable about the events and provided the NRC inspectors with the latest information they had on them. They also detailed the procedures that are followed when a failure is reported from the field, including, a root cause failure analysis, and a review of the applicable manufacturing processes.

3.6.1 Wolf Creek Failures

During the fifth refueling outage, Wolf Creek discovered 40 leaking rods in three fuel assemblies located near the center of the core. These rods had been exposed for two cycles and did not leak after their first cycle. The cause of failure is believed to be grid-to-rod fretting due to one of the following: (1) loose spacer springs caused by problems in grid fabrication, (2) handling damage, or (3) abnormal flow effects. Although no final conclusion about the root cause of failure has been made yet, WCNFD has made several changes in the spacer grid manufacturing process that are designed to improve its performance.

3.6.2 Comanche Peak Failures

Comanche Peak discovered two leaking rods during the first refueling outage. One of the failures involved an upper end cap becoming disengaged from the rod and lodging in the top nozzle of the fuel assembly. The licensee chose not to analyze the other failure. WCNFD research indicated that most of the fuel, including the failed rods, was fabricated in 1981. Although the initial reaction to the end cap failure was a defective weld, the root cause of the failure is still not known. Welding records from 1981 indicate that an acceptable weld was made, although detailed weld parameter information was not available at that time. Grain boundary separation is also considered a possible failure mode.

WCNFD personnel emphasized that, as a result of their goal of constantly improving the quality of their fuel, many improvements have been made in the welding process since the Comanche Peak fuel was fabricated in 1981. Some of the improvements discussed with the NRC inspectors include: 100% of lower end plug welds are UT inspected before loading fuel pellets, end plug and tube tolerances have been tightened, joint location tolerances have been tightened to reduce electrode drift, gripping of end plugs has been improved to prevent plug movement, 100% of end faces are automatically cleaned before plugging and welding, tolerances on welding amperage have been tightened, and the weld process has been changed to a two pass (low/high amperage) operation.

3.6.3 Zion Failures

While performing a core reload, fuel handlers at Zion encountered problems inserting a fuel assembly into the core. The assembly had been in the core for one previous cycle. As it was being lowered, difficulty in seating the assembly, apparently caused by bowing, was encountered. As the assembly was being pulled from its core location the handlers felt some resistance. After the assembly had been completely removed they noted that a piece of the bottom grid strap had been torn off.

WCNFD personnel noted that bowing of fuel assemblies is a well known phenomenon caused by irradiation growth of the assembly and that it is accounted for in the design of the assembly. They also noted that several design improvements have been made recently to both zircalloy and inconel spacer grids to minimize the potential for snagging them in the reactor core.

3.7 <u>Data Packages</u>

The NRC inspectors reviewed seven data packages for fuel assemblies supplied to the United States nuclear industry in 1991. The records were completed properly and the packages were all inclusive per the folder checklist. The routing sequences were properly stamped, signed, and dated and each operation was performed to the proper revision.

3.8 Observation of Work in Process

The NRC inspectors observed work in process at several points throughout the manufacturing operation in both the Ammonia Diuranate (ADU) and the IFBA areas. Specific operations observed in the ADU area included: pellet pressing, pellet loading into fuel rods, fuel rod welding (girth welds at rod connection to end plugs and seal welds at top end plug), fuel rod handling, skeleton assembly (fuel assembly less fuel rods), and final assembly. Specific inspections observed in the ADU area included: green (unsintered) pellet density checks, sintered density checks, pellet inspection, ultrasonic testing (UT) of welds, leak testing of fuel rods, and dimensional and visual checks of fuel rods and skeleton assemblies. Operations observed in the IFBA area included the preparation and coating of pellets with boron. The operators and QC inspectors in all areas were competent and aware of their responsibilities. The NRC inspectors also checked several tools and gages and found them all to be within calibration due dates.

All operations and inspection procedures in the plant are maintained and controlled by an electronic procedure system (EPS). There are many terminals distributed throughout the manufacturing floor, and no hard copies of procedures are kept by the operators or QC inspectors. The EPS is set up such that releases of, or revisions to, procedures can be made available to all plant personnel at the same time. Each operator or inspector is required to review and acknowledge revisions to procedures applicable to the scope of their qualified work in the EPS.

The EPS terminals allow operators to view all the procedure numbers and titles required for their work profile, the applicable revisions and issue dates, and the date the individual acknowledged reviewing the procedure release or revision. In several cases the NRC inspectors noted that there was a significant gap between the date of issue and the date of acknowledgement of a given procedure for a given individual. WCNFD personnel noted that it was possible that the individuals may not have been assigned to the work areas at the time of the revision, however, this information was not readily available.

The NRC inspectors discovered one instance in which Rev. 4 of MOP-750605, "Auto Welding of Fuel Tubes on Preplug/Preweld Line," was issued on July 3, 1991, but was not acknowledged by an individual operator until July 25, 1991. After review of the Rod Accountability and Monitoring System (RAMS) data, it was determined that the individual did perform the preplug/preweld operation for several weeks, and therefore colud have been working to the wrong procedure revision (See Nonconformance 92-01-01).

It should be noted that WCNFD personnel did have a similar finding in their internal audit report 91-011, dated October 28, 1991. In the instance noted in their report, an operator failed to acknowledge a revision to a procedure within five days. The operator did not, however, perform the operation governed by the procedure before acknowledging the revision. The corrective action portion of the response to the audit finding focused mainly on revising the administrative procedure, CA-006, to remove the requirement to review revisions within five days of issue. The NRC inspectors believe that the mainproblem is with operator's familiarity with the EPS and their training regarding revisions to procedures as discussed below.

The NRC inspectors also made two observations to plant management regarding the manufacturing operations observed. The first observation was related to the EPS. Several operators and QC inspectors were asked by the NRC inspectors to access the procedure to which they were presently working. Most individuals proved competent in accessing their procedures, but a few were not familiar enough with the system to access them without help, and many encountered minor hardware problems (i.e. keyboard depression and terminals locking up) at their local terminal and were forced to use a terminal further from their primary work area.

The second observation was related to the final pellet inspection operation. Section G.1 of QCI 910101, "Pellet Inspection - Procedural Outline," Rev. 92, dated November 18, 1991, requires the operator to "return the tray to Manufacturing for detailing if excessive (e.g. 10 or more) scrap pellets are found." Contrary to this, an inspector, observed inspecting three trays, was routinely scrapping in excess of 20 pellets per tray and then passing the trays on to the next manufacturing step. The inspector's supervisor stated that this was the normal procedure for "good" pellets and that Manufacturing was only informed when there were significant problems with the pellets. WCNFD management agreed that the procedure was not clear enough and committed to change it. The NRC inspectors were satisfied that the quality of the pellets being passed on was being adequately checked.

4 PERSONNEL CONTACTED

+	*	T. Bartman, Manager, Product/Process Development
		L. Bell, Production Engineer, Traffic
		L. Boykin, Rod Operator
		J. Brackett, Rod Operator
		M. Branham, Operator, Vapor Deposition, IFBA
		B. Brashier, QC Inspector, Non-fuel Bearing Rods
		D. Brown, Met Lab Tech A
+	*	J. Bush, Manager, Product Assurance
		J. Clay, Fuel Assembler
	*	S. Deller, Manager, Human Resources
		C. Dingle, NDT Inspector
+	*	J. Fici, Manager, Materials, Planning and Control
		M. Field, Product Assurance Engineer, Grid Area
		B. Goodwin, NDT Inspector
+	*	W. Goodwin, Manager, Regulatory Affairs
		B. Greenwaldt, Chem Lab Tech B
		G. Grier, Manufacturing Engineer, Machine Shop
		J. Higginbottom, Mgr, Product Assurance Surveillance
		C. Hightower, Manufacturing Engineer, Skeletons
	*	D. Hodge, QC Inspector
		P. Hyman, Mfg Supervisor, Skeleton & Final Assembly

* C. Joyner, Regulatory Affairs Engineer
W. Joyner, Quality Control Supervisor, Grid Area
\+ * E. Keelen, Manager, Manufacturing
A. Knotts, Assembler
\+ * R. Koga, Manager, Columbia Plant
F. Kramer, Manager, Engineering and Mfg Technology
E. Locklear, Final Assembly Inspector
C. Lott, Operator, Non-fuel Bearing Rods
\+ * S. McDonald, Manager, Technical Services
* H. Menke, Manager, Product Design
C. Miller, Supervising Engineer, Metallurgical Labs
\+ * R. Pensak, Manager, Quality Assurance
* C. Perkins, Manager, Mfg and Industrial Engineering
* R. Pollard, Manager, Product Assurance Engineering
M. Reid, Rod Operator
\+ * G. Rice, QA Engineer
J. Richards, Manufacturing Engineer, Grid Area
E. Roberts, Mgr, Materials & Mechanical Process Development
J. Roland, Tool and Gauge Inspector
\+ * E. Schwartz, QA Engineer
C. Sharpe, NDT Inspector
J. Sowers, Rod Operator
* W. Ward, Manager, Pellet and Rod Manufacturing
C. Wessinger, Met Lab Tech A
M. Wessinger, Chem Lab Tech B
\+ * D. Williams, Customer Support Engineer
D. Workman, Manager, Analytical Services Lab

\+ Attended Entrance Meeting on January 13, 1992
* Attended Exit Meeting on January 17, 1992

Selected Bulletins and Information Notices
Concerning Adequacy of Vendor Audits
and Quality of Vendor Products

	ISSUED	TITLE
1.	Information Notice 92-03	Remote Trip Function Failures in General Electric F-Frame Molded-Case Circuit Breakers
2.	Information Notice 92-04	Potter & Brumfield Model MDR Rotary Relay Failures
3.	Information Notice 92-05	Potential Coil Insulation Breakdown in ABB RXMH2 Relays
4.	Information Notice 92-19	Misapplication of Potter & Brumfield MDR Rotary Relays
5.	Information Notice 92-22	Criminal Prosecution and Conviction of Wrongdoing Committed by a Commercial-Grade Valve Supplier
6.	Information Notice 92-24	Distributor Modification to Certain Commercial-Grade Agastat Electrical Relays

CORRESPONDENCE RELATED TO VENDOR ISSUES

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

FEB 2 6 1992

Docket No. 99901239

Mr. Stephen W. Glaser
Glaser & Associates
5635-B San Diego
El Cerrito, California 94530

Dear Mr. Glaser:

SUBJECT: NRC INFORMATION NOTICE 89-59

On September 19, 1991, you wrote a letter to Mr. Gregory Cwalina of my staff regarding your company's inclusion in U.S. Nuclear Regulatory Commission (NRC) Information Notice (IN) 89-59, "Suppliers of Potentially Misrepresented Fasteners." The NRC staff issued IN 89-59 to provide nuclear power plant licensees with the names and addresses, if known, of companies supplying the nuclear industry with fasteners that were potentially misrepresented, that is, marked as having a material content and composition different from the actual content. The NRC was concerned that supplying fasteners that did not meet specifications could adversely affect plant safety. The NRC, in cooperation with other Federal agencies, has been providing information on such fasteners to the nuclear and other industries. The NRC issued IN 89-59 as a part of its efforts in this area.

The staff developed IN 89-59 after reviewing data that licensees provided in response to NRC Bulletin 87-02, "Fastener Testing to Determine Conformance With Applicable Material Specifications." Those fasteners not meeting the licensee-identified material specifications which were mismarked or unmarked were identified by the staff as potentially counterfeit or misrepresented and their suppliers were listed in IN 89-59. The NRC recognizes that the fasteners supplied by Glaser were supplied as nonsafety-related items and were not mismarked and may not have been intentionally misrepresented.

It is the NRC's position that all material in a nuclear power plant must have applied to it quality considerations commensurate with its significance to plant safety, even though it may not meet the narrower definition of safety-related. The NRC determined that information concerning nonsafety-related fasteners should be made known to all nuclear facilities for their use, as appropriate.

In your letter to the NRC (Enclosure 1), which took exception to the inclusion of Glaser in IN 89-59, you stated, "We are not denying that we may have shipped incorrect material but it was not purposely done...." The NRC staff did not state in the information notice that any of the suppliers, including Glaser, intentionally furnished fasteners that did not meet specifications.

In accordance with Section 2.790 of Title 10 of the Code of Federal Regulations, a copy of this letter and its enclosure will be placed in the NRC's Public Document Room. In addition, a copy will be included in the next edition of NUREG-0040, "Licensee Contractor and Vendor Inspection Status Report." That NUREG is published quarterly and distributed to all nuclear power facilities.

Sincerely,

for Gregory C Cwalina

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
As stated

GLASER & ASSOCIATES

5635-B San Diego
El Cerrito
California 94530
527-1705 (415)

SEPTEMBER 19, 1991

NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

ATTENTION: GREGORY C. CWALINA, NRR

RE: NRC INFORMATION NOTICE NO. 89-59 DATED 8/16/89

MR. CWALINA

SINCE RECEIVING NOTIFICATION OF OUR APPARENT TRANSGRESSION OF SUPPLYING "SUSPECTED COUNTERFEIT FASTENERS TO THE NUCLEAR INDUSTRY", I HAVE CONTACTED PACIFIC GAS & ELECTRIC CO. TO GET TO THE BOTTOM OF IT. WHAT I HAVE FOUND IS, IN MAY OF 1986 WE SUPPLIED P.G.& E. WITH 50 PC. 1/2 X 3 GR 5 HEX HD CAP SCREWS. NINETEEN MONTHS LATER (JANUARY 1988) A TEST WAS RUN ON THESE BOLTS AND IT WAS FOUND THAT THEY DID NOT COMPLY WITH THE SPECIFICATION. THE FASTENER TESTING DATA SHEET FOR THESE BOLTS SHOWS THAT THEY HAD NO MATERIAL OR MANUFACTURER MARKINGS ON THE HEAD. WE COULD THEREFORE CONCLUDE WITHOUT TESTING THAT THE BOLTS WERE NOT WHAT WAS ORIGINALLY ORDERED AND WERE IN FACT SOMETHING OTHER THAN GR.5 MATERIAL. IF THESE BOLTS WERE NOT STAMPED WITH HEAD MARKINGS THEY SHOULD HAVE BEEN BROUGHT TO OUR ATTENTION UPON RECEIPT OF MATERIAL- NOT 19 MONTHS LATER DURING A TEST FOR THE N.R.C.. IF YOU ARE TRYING TO FIND MANUFACTURERS OR DISTRIBUTORS OF MISMARKED BOLTS - FINE, I AM ALL FOR IT, BUT YOU HAVE LABLED GLASER & ASSOCIATES, INC. A PROVIDER OF SUSPECTED COUNTERFEIT BOLTS AND THAT IS NOT TRUE. WE ARE NOT DENYING THAT WE MAY HAVE SHIPPED INCORRECT MATERIAL BUT IT WAS NOT PURPOSELY DONE AND I DONT LIKE YOUR INSINUATION THAT IT WAS. THE VERY LEAST YOU COULD HAVE DONE, WAS TO NOTIFY US THAT WE WERE BEING SUBJECTED TO THIS INCREDIBLE LETTER.

I AM REQUESTING A RETRACTION. I AM NOT SURE OF THE DAMAGE THAT YOU AND THE N.R.C. HAVE DONE TO EITHER MY COMPANIES OR MY OWN REPUTATION. I AM CERTAIN THAT YOU HAVEN'T DONE THEM ANY GOOD.

BE ASSURED MR. CWALINA, THIS MATTER WILL NOT FADE AWAY. I'LL BE WAITING FOR YOUR REPLY.

S W Glaser

STEPHEN W. GLASER

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

FEB 2 6 1992

Mr. Carlo Caso, Vice President,
Energy Systems Business Unit
Westinghouse Electric Corporation
Nuclear and Advanced Technology Division
PO Box 355
Pittsburgh, Pennsylvania 15230 - 0355

Dear Mr. Caso:

SUBJECT: 10 CFR PART 21 REPORT REGARDING ASL DRY TYPE TRANSFORMERS

Your letter of November 14, 1991 (ET-NRC-91-3638), addressed to the Nuclear Regulatory Commission (NRC) discussed a potential deficiency regarding cracked insulators observed in ASL Power Center Dry Type transformers at the Washington Public Power Supply System's WNP-3 plant. The attachment to that letter, which was to be sent to all your customers states

> Westinghouse sold the transformer product lines, including ASL Dry Type Power Center Transformers, to ABB Power Transmission and Distribution Company Inc. Consequently, Westinghouse does not have the necessary technical information or expertise available to identify all of the potentially affected plants, the cause, nor the ultimate solution.

ASEA Brown Boveri (ABB), in a letter to the NRC of November 21, 1991, stated that the above information is false and misleading. ABB also sent the NRC a copy of the November 21, 1991 letter to Westinghouse which denies any product liability.

The NRC is unaware of the details of the agreement but does have a potential concern regarding Westinghouse's implementation of the evaluation and reporting requirements of Part 21 of Title 10 of the Code of Federal Regulations (Part 21). Part 21 applies to the firm "supplying the components" to nuclear facilities. Part 21 does not allow companies which discontinue supplying basic components to relieve themselves of Part 21 responsibilities for items which were previously supplied under Part 21 provisions. Therefore, unless specifically addressed in the agreement, Westinghouse retains the responsibility for implementing Part 21 evaluation and reporting requirements for all equipment supplied pursuant to the provisions of Part 21. Unless Westinghouse can document by means of the agreement between itself and ABB, or, otherwise, that Westinghouse transferred to ABB the information necessary for an evaluation thus putting ABB in a better position to perform an evaluation, NRC requests that Westinghouse review its files to determine if that information is available and an evaluation in accordance with 10 CFR Part 21 can be performed.

This letter does not conclude that Westinghouse does, indeed, have product liability. Rather, the NRC is concerned with our regulatory responsibility to assure the health and safety of the public. It is imperative that the NRC is informed od all defects which could cause a significant safety hazard so that the NRC can be assured that the information is distributed to the nuclear industry.

If Westinghouse supplied the basic component, in accordance with the requirements of 10 CFR 21.21(a)(1), the NRC expects Westinghouse to evaluate deviations or in accordance with 10 CFR 21.21(b) to inform the purchaser or affected licensees so that they can cause the deviation to be evaluated. Please provide us with a list of all the nuclear power plants which received similarly designed ASL Dry Type transformers and a list of those customers who have been notified. If you have any questions, please contact Mr. Greg Cwalina at (301) 504-3221.

Sincerely,

for Gregory C Cwalina

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

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

March 31, 1992

Docket No. 99901138

Mr. Bobby Corte
Cor-Val, Incorporated
Post Office Box 9076
Houma, Louisiana 70361

Dear Mr. Corte:

SUBJECT: NRC INFORMATION NOTICE 88-97, SUPPLEMENT 1

Thank you for your October 31, 1991 letter to the U.S. Nuclear Regulatory Commission (NRC) staff, enclosed, regarding the inclusion of Cor-Val, Incorporated (Cor-Val) in Information Notice (IN) 88-97, "Potentially Substandard Valve Replacement Parts." The NRC staff issued IN 88-97 and Supplements to provide nuclear power plant licensees with information regarding potentially substandard valve replacement parts. The information notice identified that the valve manufacturer's authorized distributor procured the parts from both the original equipment manufacturer and secondary sources as commercial-grade components without adequately verifying that the parts would fulfill their function.

Your letter requested the NRC staff to retract statements that it made in regard to the involvement of Cor-Val in supplying Masoneilan-Dresser valve parts that were dimensionally and, in some cases, metallurgically incorrect. As a result of your query, NRC staff performed a review of the circumstances that prompted the issuance of IN 88-97 and concluded that Supplement 1 to IN 88-97, dated April 28, 1989, is an accurate account of the circumstances in this matter and we do not intend to issue an amendment to the information notice. Amendments are reserved for cases in which the original Information Notice contained erroneous information.

My staff contacted representatives from Sample-Webtrol Controls, Incorporated (SW Controls), Masoneilan-Dresser Valve Company (MD), and the Consumers Power Company (CPCO) Palisades nuclear power plant. Dimensionally incorrect steam bypass valve replacement parts at CPCO's Palisades plant prompted a CPCO investigation and subsequent 10 CFR Part 21 report. Records show that the dimensionally incorrect steam bypass valve parts were procured from Cor-Val by SW Controls in 1986. These parts were supplied to the CPCO facility.

Your October 31, 1991 letter stated that "the NRC has done an injustice to Cor-Val by portraying us as dealing in unethical business practices." The subject IN does not express any view as to the type of business practices being conducted by any of the four vendors discussed in the IN. Instead, the subject IN informs NRC's licensees that CPCO identified several MD valve replacement parts which deviated from the MD technical specifications.

The NRC recognizes, and made clear in the IN, that the subject parts were procured by the manufacturer's authorized distributor as commercial-grade and some of the parts were improperly dedicated to safety-related use by others. Therefore, the purpose of the IN was to alert NRC licensees to the possibility that valve replacement parts purchased by them directly as commercial-grade components might not meet the original manufacturer's equipment specifications.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," the NRC will place a copy of this letter and its enclosure in the NRC's Public Document Room. In addition, a copy will be included in the next edition of NUREG-0040, "Licensee Contractor and Vendor Inspection Status Report." That NUREG is published quarterly and distributed to all nuclear power facilities.

Sincerely,

Leif J. Norrholm, Chief
Vendor Inspection Branch
Division of Reactor Inspection
and Safeguards
Office of Nuclear Reactor Regulation

Enclosure:
As stated

P.O BOX 9076
HOUMA, LA 70361

COR-VAL, INC.

(504) 851-3160
FAX: (504) 851-5102

October 31, 1991

Mr. Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation
Washington, DC 20555

Dear Mr. Rossi:

I feel that the N.R.C. has done an injustice to Cor-Val, Inc. by portraying us as dealing in unethical business practices. As per the N.R.C. information notice 88-97, Supplement 1.

Cor-Val did not and has never represented its products to be anything other than our own. In the incident that the N.R.C. investigated, Cor-Val had no direct involvement. It is true, we have acted as a contractor for S.W. Controls. But we had no knowledge of their intentions for resale to the nuclear industry. In fact, it is important to note that none of the parts examined in the investigation were identified as having been manufactured by Cor-Val, Inc. However, some were found to be manufacture by CVS. And others were sold by M-D who purchased them from an apparent unqualified source.

Cor-Val has manufactured and sold valve replacement parts for the past 11 years. Our formal Quality assurance program has been through numerous and often stringent audits by outside parties. We are A.P.I. certified and a D.O.D. Level I Subsafe contractor for the Navy's nuclear division.

The N.R.C. has done damage to my companies reputation and effectively limited my ability to solicit work from the Nuclear Power Industry. Therefore, I respectively request that a retraction in writing be made by your office and mailed to all parties who would have received the ori[illegible]nal notices.

Mr. Rossi, I would appreciate your assistance in this matter. I believe you would agree that this is a reasonable request.

Thank you for your time and assistance.

Sincerely,

Bobby Corte

Bobby Corte

SPEC 6A

COR-VAL, INC.

PRODUCTS & SERVICES
P. O. BOX 9076
HOUMA, LOUISIANA 70361
(504) 851-3160
NATIONAL WATS 1-800-624-7324
FAX (504) 851-5102

NAVR

QUALITY ASSURANCE

Cor-Val's Q.A. Program conforms to A.P.I. Spec. Q1 and MIL-1-45208A Inspection System Requirements. Additionally this program meets the specifications of 10CFR50 Appendix B, 10CFR21 Section 206 requirements for the nuclear industry and DOD Level/1 Subsafe Nuclear Requirements.

VALVES, ACTUATORS and INSTRUMENTATION REPAIRS

Complete repair/remanufacturing of all makes of A.P.I. and ANSI GATE, GLOBE, BALL, CHECK, BUTTERFLY, SAFETY, RELIEF and CONTROL VALVES. Precision machining, code welding, heat treating and pressure testing (hydrostatic and gas) services are available. REPAIR SERVICES FOR ALL VALVE RELATED INSTRUMENTATION used with process control valves. REPAIR OF ALL MAKES OF ACTUATORS pneumatic, hydraulic and electric. COMPLETE LINE OF NEW and REMANUFACTURED VALVES, ACTUATORS and INSTRUMENTATION AVAILABLE.

FIELD SERVICE - INDUSTRIAL PROCESS PLANTS

Field service personnel available to repair all types of control valves, actuators and instrumentation on site.

FIELD SERVICE INLAND and OFFSHORE

Valves repaired both on and offshore. Lubricators to set all types of back pressure valves, C.I.W., GRAY, NATIONAL and McEVOY 5,000# to 20,000#.

CONTROL VALVE TRIM

Cor-Val trim available for FISHER, MASONEILAN, C.E. NATCO/INVALCO, NORRISEAL and many others. Trim repair service available.

CHOKES and REPAIR PARTS

Cor-Val choke parts available for BEST, CAMERON IRON WORKS, GULFCO, GRAY TOOL, McEVOY, NATIONAL OCT/FMC, THORNHILL CRAVER and many others. COMPLETE ASSEMBLY NEW and REMANUFACTURED. REPAIR/REMANUFACTURING SERVICE AVAILABLE.

PUMPS

Complete repair facility for glycol, centrifugal, chemical injection and plunger type pumps. Exchange program offered.

FLANGES and CONNECTOR HUBS

A.P.I., ANSI, MSS and LENS RING FLANGES/RING GASKETS. CONNECTOR HUBS and SEAL RINGS. Orifice, blind and other special seal rings available. All sizes and materials.

ORIFICE PLATES

Universal and paddle type. All sizes and materials.

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U.S. NUCLEAR REGULATORY COMMISSION

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This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from January through March 1992.

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