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Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
April–June 1993

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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**Division of Reactor Inspection and Licensee Performance
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001**

ABSTRACT

This periodical covers the results of inspections performed by the NRC's Vendor Inspection Branch that have been distributed to the inspected organizations during the period from April 1993 through June 1993.

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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.

INSPECTION REPORTS

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

June 8, 1993

Docket No. 99901256

Mr. Wendell E. Jones, Jr.,
Quality Assurance Manager
ABB Power Distribution
Circuit Breaker Division
I-95 and Mechanicsville Hwy.
P.O. Box 100524
Florence, South Carolina 29501-0524

Dear Mr. Jones:

SUBJECT: NOTICE OF VIOLATION AND NOTICE OF NONCONFORMANCE
NRC INSPECTION REPORT NO. 99901256/93-01)

This refers to the inspection conducted by Messrs. B. Rogers, K. Naidu, R. Frahm, Jr., and J. Winton of this office on April 13 through 16, 1993. The inspection included a review of activities authorized for your ABB Power Distribution, Circuit Breaker Division (ABB) facility in Florence, South Carolina. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report. The NRC inspectors had additional questions related to the findings subsequent to the completion of the inspection. You provided a response to these questions by telephone on May 24, 1993.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

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 (Notice). The violation is of concern because it potentially impacts your ability to evaluate and report defects in basic components in accordance with the requirements of 10 CFR Part 21.

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

In addition, during this inspection it was found that the implementation of your QA program failed to meet certain NRC requirements. Numerous occurrences were identified where documented ABB Quality Assurance Procedures were not followed. 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.

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.

The responses directed by this letter and the enclosed Notices are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96.511.

Sincerely,

Gary G. Zech

for Charles E. Rossi, Director
Division of Reactor Inspection
and Licensee Performance
Office of Nuclear Reactor Regulation

Enclosures:
Notice of Violation
Notice of Nonconformance
Inspection Report 99901256/93-01

Enclosure 1

NOTICE OF VIOLATION

ABB Power Distribution
Circuit Breaker Division
Florence, South Carolina

Docket No.: 99901256/93-01

During an NRC inspection conducted on April 13 through 16, 1993, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1993), the violation is listed below:

Section 21.21, "Notification of failure to comply or existence of a defect and its evaluation," of 10 CFR requires, in part, that each corporation subject to the regulations adopt appropriate procedures for either evaluating deviations and failures to comply, or informing the licensee or purchaser of the deviation or failure to comply. In addition, Section 21.6 requires that a current copy of 10 CFR Part 21 be posted.

Contrary to the above requirements, ABB had not revised its procedures to address the substantive revisions to 10 CFR Part 21 that became effective on October 29, 1991, and had not posted a current copy of 10 CFR Part 21.

This is a Severity Level V violation (Supplement VII). (99901256/93-01-01)

Pursuant to the provisions of 10 CFR 2.201, ABB Power Distribution, Circuit Breaker Division, 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 Licensee Performance, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 8th day of June 1993

Enclosure 2

NOTICE OF NONCONFORMANCE

ABB Power Distribution
Docket No.: 99901256/93-01

Based on the results of an NRC inspection conducted on April 13 through 16, 1993, it appears that certain of your activities were not conducted in accordance with NRC requirements.

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 procedures and be accomplished in accordance with these procedures. The inspectors determined that ABB was not properly implementing their prescribed quality assurance program as evidenced by the following examples: (99901256/93-01-02)

- Paragraph 3.5 of Quality Assurance Procedure (QAP) 2.5, "Dedication Program for the Utilization of Switchgear Systems Equipment and Spare or Replacement Parts in Nuclear Safety-Related Applications," Revision 1, dated October 29, 1992, stated that the process for acceptance of nuclear safety related (NSR) items was based on an annual audit of all NSR Suppliers and validation of the NSR suppliers' Certificates of Conformance.

  Contrary to the above, two NSR suppliers had not been audited since October of 1991.

- Paragraph 4.4.9 of QAP 4.3, "Procurement Documentation Control System - General," Revision 1, dated October 29, 1992, stated that items used in circuit breaker assembly were only to be purchased from approved vendors.

  Contrary to the above, ABB had purchased items from two vendors which were not listed on the Approved Vendors List (AVL) and used the items in assembling NSR circuit breakers.

- Paragraph 3.1 of QAP 4.3 stated that vendors and their products were evaluated to assure that their quality system and product performance are such that they satisfy the specification requirements and meet applicable industry standards.

  Contrary to the above, ABB had not documented an evaluation to support the basis for inclusion of all vendors on the AVL.

- Paragraphs 3.1.2 and 3.2.1 of QAP 7.1, "Receiving Inspection - Components," Revision 1, dated October 29, 1992, referenced "QAP 6.5," and paragraph 3.2.4.2 referenced "QAP 16.2."

  Contrary to the above, "QAP 6.5" and "QAP 16.2" did not exist and therefore could not be followed.

- Paragraph 6.2 of QAP 2.4, "Inspection and Test Personnel Qualification," Revision 1, dated October 29, 1992, stated that a logbook shall be maintained which includes job descriptions and certificates of qualification of personnel.

  Contrary to the above, a logbook including job descriptions and certificates of qualification of personnel did not exist.

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

Dated at Rockville, Maryland
this day of 1993.

ORGANIZATION:	ABB Power Distribution Circuit Breaker Division
REPORT NO.:	99901256/93-01
CORRESPONDENCE ADDRESS:	Mr. Wendell Jones, Quality Assurance Manager ABB Power Distribution Circuit Breaker Division I-95 and Mechanicsville Hwy. P.O. Box 100524 Florence, South Carolina 29501-0524
NUCLEAR INDUSTRY ACTIVITIES:	Provides safety-related products for commercial nuclear power plants.
INSPECTION CONDUCTED:	April 13 through 16, 1993
ASSIGNED INSPECTOR:	Bill Rogers ___ 6/1/93 Bill Rogers, Lead Inspector — Date Reactive Inspection Section No. 2 Vendor Inspection Branch
APPROVED:	___ Gregory Cwalina, Chief — Date Reactive Inspection Section No. 2 Vendor Inspection Branch
OTHER INSPECTORS:	Kamalakar Naidu Ronald Frahm, Jr. Jeffrey Winton
INSPECTION BASES:	10 CFR Part 21 and Appendix B to 10 CFR Part 50
INSPECTION SCOPE:	To evaluate selected portions of the ABB Power Distribution, Circuit Breaker Division, 10 CFR Part 21 program and quality assurance program and its implementation in providing items for safety-related use in accordance with the requirements of Appendix B to 10 CFR Part 50.
PLANT SITE APPLICABILITY:	Numerous

1 INSPECTION SUMMARY

1.1 Violations

Contrary to the requirements of 10 CFR Part 21, ABB procedures did not address the evaluation and reporting requirements of the current revision of 10 CFR Part 21 which first became effective October 29, 1991, and ABB had not posted a current copy of 10 CFR Part 21 (Violation 99901256/93-01-01).

1.2 Nonconformances

Contrary to Criterion V of Appendix B to 10 CFR Part 50, ABB did not follow documented procedures as illustrated by the following examples: (Nonconformance 99901256/93-01-02)

- Two nuclear safety-related suppliers had not been audited annually as required by procedure QAP 2.5, "Dedication Program for the Utilization of Switchgear Systems Equipment and Spare or Replacement Parts in Nuclear Safety-Related Applications," Revision 1, dated October 29, 1992.
- ABB had purchased items from two vendors which were not listed on the Approved Vendors List (AVL) contrary to procedure QAP 4.3, "Procurement Documentation Control System - General," Revision 1, dated October 29, 1992.
- ABB had not documented an evaluation to support the basis for inclusion of vendors on the AVL as required by procedure QAP 4.3.
- QAP 7.1 "Receiving Inspection - Components," Revision 1, dated October 29, 1992, referenced "QAP 6.5" and "QAP 16.2" which did not exist and therefore could not be followed.
- A logbook including job descriptions and certificates of qualification of personnel did not exist as required by procedure QAP 2.4, "Inspection and Test Personnel Qualification," Revision 1, dated October 29, 1992.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

The previous inspection performed by the Nuclear Regulatory Commission, Vendor Inspection Branch, and referred to in this report, was of BBC Brown Boveri, Inc. In 1988, BBC Brown Boveri, Inc. merged with ASEA, Inc. to form ASEA Brown Boveri and the portion of the company previously known as the Switchgear Products Group was changed to ABB Power Distribution, Inc. In 1992 ABB Power Distribution, Inc. merged with ABB Power T & D Company, Inc. and is now known as ABB Power T & D Company, Inc., which includes ABB Power Distribution, Circuit Breaker Division, Florence, South Carolina and ABB Power Distribution Switchgear Division, Sanford, Florida.

2.1 Nonconformance 99900835/86-01-01 (Closed) The nonconformance identified that contrary to Criterion VII of Appendix B to 10 CFR Part 50, BBC (now ABB) had not established controls necessary to ensure charging motors purchased from subvendors conformed to BBC drawings. Additionally, three charging motor vendors were identified as not being on the BBC approved vendors list. At the time of the 1986 inspection ABB did not maintain an AVL but listed approved vendors on the drawing itself. During the current inspection, the inspectors determined that charging motors were procured from only one vendor who was documented on the AVL and that ABB had adequate procurement and receipt inspection procedures to assure that charging motors and other components conform to ABB drawings. The nonconformance was closed, however, additional concerns with the AVL were identified during this inspection as identified in nonconformance 93-01-02.

2.2 Nonconformance 99900835/86-01-02 (Closed) The nonconformance identified that contrary to Criterion III of Appendix B to 10 CFR Part 50, BBC had not established controls necessary to ensure environmental qualification when similar parts are substituted for those originally installed in qualified circuit breakers. Additionally, BBC had not established controls necessary to ensure design changes made by charging motor subvendors were reviewed for their effect on equipment qualification. During the current inspection, the NRC inspectors noted that the engineering change control procedures were revised in January, 1987, to include a formal signoff for the engineering review for effects on equipment qualification. The inspectors concluded that ABB currently had adequate engineering and procurement procedures to assure that substitute parts were properly tested to ensure environmental qualification. The nonconformance was closed.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

During the entrance meeting on April 13, 1993, the NRC inspectors discussed the scope of the inspection and the areas to be reviewed and established the persons to contact within ABB's management and staff. During the exit meeting on April 16, 1993, the NRC inspectors discussed their findings and concerns with ABB's management and staff.

3.2 Background

ABB Power Distribution, Circuit Breaker Division, manufactures circuit breakers for safety-related applications under a 10 CFR 50 Appendix B quality assurance program and provides service on safety-related circuit breakers at their facility. The Circuit Breaker Division sells safety-related circuit breakers only to ABB Power Distribution, Switchgear Systems Division. The Switchgear Systems Division assembles the circuit breakers into switchgear panels which are sold as safety-related components to utilities or other customers. Field service is provided by ABB Service Company from service centers in numerous locations throughout the country. ABB Service Company is a separate business entity from ABB Power Distribution. For the remainder of this report "ABB" refers only to ABB Power Distribution, Circuit Breaker Division; other entity names will be fully specified.

3.3 Inspection and Scope

3.3.1 10 CFR Part 21 Program - Procedures

The NRC inspectors reviewed procedure QAP 15.5, "Reporting of Product Defects," Revision 1, dated October 29, 1992, which implemented the requirements of 10 CFR Part 21. The procedure had not been revised to incorporate the substantive changes which became effective on October 29, 1991. Discussion with the ABB QA manager indicated that ABB had been unaware of the revision to 10 CFR Part 21. In addition, the inspectors reviewed ABB's posting of 10 CFR Part 21, as required by 10 CFR 21.6, and determined that the version posted was not the current revision. The ABB QA manager stated that ABB intended to revise QAP 15.5 to reflect the current regulations and to post the current version of 10 CFR Part 21. This discrepancy constituted a violation of 10 CFR Part 21, Sections 21.21 and 21.6.
(Violation 99901256/93-01-01)

The NRC inspectors also noted that QAP 15.5 used the terms "deviation" and "defect" interchangeably, not as defined in 10 CFR 21.3. The ABB QA manager indicated that QAP 15.5 would be revised to correct the use of the terms deviation and defect.

3.3.2 10 CFR Part 21 Program - Deviation Evaluations

3.3.2.1 Issues Discussed and Evaluations Reviewed

The inspectors discussed various issues concerning ABB equipment and reviewed files containing information related to evaluations that ABB had performed in accordance with the requirements of 10 CFR Part 21. The issues discussed and files reviewed included:

- The Shearon Harris Nuclear Power Plant had several instances in the 1980's where the trip coils burned out in LK-series circuit breakers when they failed to open on demand. The trip coils were designed to withstand only momentary energization. To correct this problem, ABB changed the design of an internal component, the bridge blade, and added booster springs to the existing opening springs. In spite of these design modifications, selected circuit breakers subjected to cycling duty, such as the pressurizer backup heaters continued to remain closed. ABB further determined that circuit breaker bearings were losing lubrication during cleaning procedures and subsequently provided Shearon Harris with alternate bearings and cleaning and lubricating instructions, and in addition, redesigned the arcing contact springs of the circuit breakers to reduce contact pinch force. Shearon Harris is the only nuclear plant at which LK-type breakers are installed in safety-related applications.

- On May 6, 1988, ABB notified the NRC of a deficiency in the high instantaneous circuitry solid-state trip devices (SSTD), having an extended instantaneous pick-up trip setting of 24 multiples of per unit current, that had been installed on its K-line circuit breakers. ABB had supplied circuit breakers with these types of SSTDs only to Perry Nuclear Power Plant (PNPP) and Salem Generating Station, (Salem). Specifically, the SSTD instantaneous element did not operate or was slow to operate in applications using a single-pole low-voltage primary-current test device at primary-current levels starting at approximately 10 times-per-unit current rating (225 or 600A). The SSTD problem was identified and corrected by ABB personnel at Allentown where the SSTDs were manufactured. In addition, ABB provided a recommended procedure for removal and replacement of the SSTDs and follow-up testing to PNPP and Salem.

- On April 9, 1991, ABB reported to the NRC that current transformers manufactured of epoxy-anhydride formulations (EP1 epoxy) during the 1972 - 1973 time period had exhibited cracking on the stem area. The EP1 material was no longer used for safety-related current transformers following January 1990. ABB had tested a current transformer with cracks and had determined that there was no failure expected due to this problem. ABB had notified all customers and recommended that the current transformers be inspected periodically for cracks and replaced with a new polyurethane based version as required.

- On May 24, 1991, Carolina Power and Light Company (CPL) reported to the NRC that when a 5HK350, 1200 ampere circuit breaker, being used as the diesel generator output circuit breaker at the Brunswick Steam Electric Plant, was racked into its test position and an attempt was made to close it, the charging springs failed to charge. The charging circuit had activated and energized the charging springs motor, but the springs did not charge because three charging pawls had been installed instead of the required combination of one charging pawl and two holding pawls. The incorrect assembly caused the holding pawls to misalign and mesh with the face of only about 50 percent of the ratchet plate teeth. As a result, the ratchet teeth were effectively experiencing double their normal load which caused one tooth to break, thereby preventing the holding pawl from engaging. ABB subsequently repaired the circuit breaker and has changed production methods to minimize the occurrence of incorrect or improperly installed components. (See section 3.3.6 of this report.)

- On April 7, 1993, ABB reported to the NRC that K-4000 circuit breakers had failed to meet certain rating requirements, based on ANSI C37.16. The arc may not be extinguished as the circuit breaker is opening in applications where rated maximum voltage may appear across a single circuit breaker pole. ABB had determined three specific electrical configurations were this could occur (not typical) and notified the customers with recommendations for corrective actions.

The inspectors concluded that ABB had performed adequate technical evaluations of the deviations and had met the reporting requirements of 10 CFR Part 21. The documentation of these evaluations is discussed in section 3.3.2.2.

3.3.2.2 Procedural Requirements for Documentation of Evaluations and Implementation

QAP 15.5, "Reporting of Product Defects," Revision 1, dated October 29, 1992, specified in Section 12.0 that the Check List for Record of Evaluation would be used to record 10 CFR Part 21 and 10 CFR 50.55 evaluations. This checklist required eleven items to be documented which included:

- a description of the deviation, defect and circumstances
- the location and date
- an analysis of safety implication
- identification of the steps to be considered in the evaluation and personnel assigned for the analysis, calculations, test, trips to jobsites, and factory and field inspections

- documents or drawings requiring changes
- reference documents
- corrective actions with dates
- actions to prevent recurrence with dates
- conclusions
- identification of the preparer
- identification of the reviewer

The NRC inspectors determined that the most recent evaluation, concerning the K-4000 circuit breakers and ANSI requirements, which was performed after October 29, 1992, when Revision 1 of QAP 15.5 became effective, met the requirements of QAP 15.5.

The remaining evaluation files that the inspectors reviewed were performed prior to October 29, 1992, in accordance with an earlier revision of QAP 15.5 which did not require the checklist to be used to record evaluations. These files typically contained only the letter to the NRC documenting the Part 21 determination and did not address items such as identification of the steps to be considered in the evaluation and personnel assigned for the analysis, calculations, tests, trips to jobsites, and factory and field inspections, documents or drawings requiring changes or reference documents. In addition, the files were not organized in any fashion, contained little if any supporting documentation for the conclusions that were documented, and ABB was unable to provide additional supporting documentation in a reasonable manner. The inspectors concluded that the checklist for the record of evaluation and the subsequently performed evaluation are an improvement in ABB's Part 21 program.

3.3.3 Vendor Approval and Control

The NRC inspectors reviewed the procurement procedures and their implementation. Components which were to be installed by ABB in nuclear safety-related circuit breakers were procured as either nuclear safety-related (NSR) or nuclear safety qualified (NSQ). NSR components were basic components procured in accordance with Appendix B to 10 CFR Part 50 and 10 CFR Part 21. Components procured as NSQ were commercial grade items that were dedicated for nuclear safety-related applications by ABB during production or testing.

ABB maintained an Approved Vendor List (AVL) for both the NSR and NSQ suppliers. The approved NSR vendors listed on the Approved Vendors List were ABB Allentown for solid state trip units and

Kema Powertest for circuit breaker testing. The NRC inspectors reviewed sales orders with each of these vendors and determined that the orders stated that the components or services were nuclear safety-related and that 10 CFR Part 21 and 10 CFR Part 50 Appendix B applied. Paragraph 3.5 of QAP 2.5 "Dedication Program for the Utilization of Switchgear Systems Equipment and Spare or Replacement Parts in Nuclear Safety Related Applications," Revision 1, dated October 29, 1992, stated that the process for acceptance of NSR items was based on an annual audit of all NSR Suppliers and validation of NSR suppliers' Certificates of Conformance (COC). The NRC inspectors determined that COCs had been received for three recent NSR orders but that audits had not been performed on either NSR vendor since October 1991. This discrepancy was an example of a failure to follow procedures and constituted a Nonconformance to Criterion V, "Procedures," of Appendix B to 10 CFR Part 50. (Nonconformance 99901256/93-01-02)

Paragraph 3.3 of QAP 4.5 "Purchase Order Control - Nuclear Safety Qualified (NSQ)," Revision 1, dated October 29, 1992, stated that ABB would verify that the vendor selected for procured NSQ items appeared on the AVL. In addition, Paragraph 4.4.9 of QAP 4.3 "Procurement Documentation Control System - General," Revision 1, dated October 29, 1992, stated that items used in circuit breaker assemblies were only to be purchased from approved vendors listed on the AVL. The NRC inspectors reviewed the bill of material for a K-800S circuit breaker, part number KLS8E90141, to determine whether NSQ components had been procured from qualified vendors listed on the AVL. The NRC inspectors noted components procured from two vendors which were not listed on the AVL. These vendors were AFI, a primary supplier of common hardware, and Florence Vocational Rehabilitation Center, a company under consignment from ABB to perform minor assembly functions. The QA manager stated that these vendors were unintentionally left off the AVL and would be added. This discrepancy was another example of a failure to follow procedures and constituted a Nonconformance to Criterion V, "Procedures," of Appendix B to 10 CFR Part 50. (Nonconformance 99901256/93-01-02)

Paragraph 3.1 of QAP 4.3 stated that NSQ vendors and their products would be evaluated to assure that their quality system and their product performance were such that they satisfied the specification requirements and met applicable industry standards. The QA manager stated that some NSQ vendors had been approved based on performance history and review of receipt inspection records, but official evaluations were never documented to support the basis for inclusion on the AVL. The QA manager indicated that evaluations would be performed and documented for each of the NSQ vendors on the AVL. This discrepancy was another example of a failure to follow procedures and constituted a Nonconformance to Criterion V, "Procedures," of Appendix B to 10 CFR Part 50. (Nonconformance 99901256/93-01-02)

3.3.4 Receipt Inspection

The NRC inspectors reviewed the process and procedure used to inspect components upon receipt from vendors. The governing procedure for receipt inspection was QAP 7.1 "Receiving Inspection - Components," Revision 1, dated October 29, 1992. The NRC inspectors verified that the receipt inspectors followed the inspection plan and recorded results on the QC record card as required by QAP 7.1 for several components including insulated connectors and current transformers. The receipt inspectors verified dimensions and visually checked for cracks and breaks. The NRC inspectors also verified that COCs had been received for three recent NSR orders, and that all test and measuring equipment located in the receipt inspection area was within the calibration cycle. ABB's receipt inspection and testing process appeared to be effective to minimize the likelihood of fraudulent materials being used in the manufacture of safety-related circuit breakers. The NRC inspectors noted that paragraphs 3.1.2 and 3.2.1 of QAP 7.1 referred the inspector to "QAP 6.5" for use of the QC record card, and paragraph 3.2.4.2 referred the inspector to "QAP 16.2" for more information on corrective actions. QAPs 6.5 and 16.2 did not exist and therefore could not be followed. This discrepancy was another example of a failure to follow procedures and constituted a Nonconformance to Criterion V, "Procedures," of Appendix B to 10 CFR Part 50. (Nonconformance 99901256/93-01-02)

3.3.5 Training and Qualifications

The NRC inspectors reviewed the training and qualification process and procedures and their implementation. QAP 2.4 "Inspection and Test Personnel Qualification," Revision 1 dated October 29, 1992, defined ABB's method for certification of inspection and test personnel. Criterion II, "Quality Assurance Program," of Appendix B to 10 CFR Part 50 requires, in part, that the quality assurance program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. The NRC inspectors noted that the scope of QAP 2.4 was limited to personnel performing inspection, test, and auditing functions, and did not include all personnel performing activities affecting quality. The inspectors noted that ABB required that welders be certified and all welders were verified to have current certifications.

QAP 2.4 further stated that the primary means of certification was on the job training with emphasis on actual supervised performance of inspections and test procedures followed by a capability demonstration, however ABB did not formally document this process. In general, records were not present to document the training requirements and subsequent qualifications for personnel performing activities affecting quality. Paragraph 6.2

of QAP 2.4 stated that a logbook shall be maintained which includes job descriptions and certificates of qualification of personnel. The NRC inspectors determined that a logbook including job descriptions and certificates of qualification of personnel did not exist. The human resources manager and the QA manager indicated that the training program was being revised and improved to include more diversified training and more complete documentation. This discrepancy was another example of a failure to follow procedures and constituted a Nonconformance to Criterion V, "Procedures," of Appendix B to 10 CFR Part 50. (Nonconformance 99901256/93-01-02)

3.3.6 Manufacturing and Production Testing

The inspectors observed work performed on several circuit breakers at various stages of assembly. At the time of the inspection, the circuit breakers being assembled were for commercial grade order or stock, no safety-related orders were being filled. However, ABB indicated that all circuit breaker work, such as assembly and testing, was performed under their Appendix B quality assurance program. The inspectors were particularly interested in this aspect of the ABB program since the NRC had received a report from a licensee of a circuit breaker with incorrect components installed (see section 3.3.2.1 of this report). Discussion with ABB management indicated that ABB had taken steps to reduce the likelihood of these types of problems, such as developing key personnel to track mistakes and determine corrective action. In addition, ABB indicated that they were in the process of implementing a "work cell" method of production. This method made a particular worker responsible for fabricating a particular subassembly or performing a specified portion of the total process of constructing a circuit breaker. Prior to use of the work cell method, personnel would essentially construct an entire circuit breaker with piece parts and sub-assemblies. Since this entire process could not necessarily be carried out in a single work shift, the possibility of errors would be introduced with the interruption in work flow due to the end of a shift. In addition, during periods of increased production, other personnel might have continued the assembly process, which could also have lead to the possibility of introducing errors. ABB indicated that the work cell method increased productivity and decreased the opportunity for error.

The inspectors observed the production tests typically performed on K-Line safety-related circuit breakers. The tests observed were a demonstration performed by ABB for the inspectors' benefit since no safety-related testing was required by production during the period of the inspection. Control circuitry tests were performed on a K-800 circuit breaker which used 125 VDC control voltage. These tests included verifying closing and opening at three points within the voltage specification range (low, medium, and high) and demonstration of the ability of the control circuitry to withstand high potential voltage. Load tests were performed on a manually operated K-800 circuit breaker. These included a test of the ground fault detection circuitry that open the circuit breaker when a ground fault exists, and a primary injection test, where various low voltage currents are injected into the line side of the circuit breaker to verify the circuit breaker would open in accordance with the prescribed points on the test curve. The inspectors noted that the personnel performing the tests followed a specified test procedure, applicable to the circuit breaker and test being performed, and that the results were documented.

The inspectors concluded that ABB had taken significant steps to reduce the likelihood of missing or incorrectly installing components and that the production and testing processes exhibited adequate implementation of the ABB quality assurance program.

4 PERSONNEL CONTACTED

W. Gibson, Vice President/General Manager
*+ W. Jones, Quality Assurance Manager
* B. Johnson, Operations Manager, Switchgear Systems Division
+ T. Jablonsky, Operations Manager
W. Book, Engineering Manager
+ G. Snyder, Marketing Manager
C. Porter, Purchasing Manager
+ G. Marler, Human Resources Manager
R. Cope, Engineering Administrator
J. Heiden, Human Resources Administrator
G. Grote, Low Voltage Engineer
* C. Blasio, Quality Assurance Engineer
H. Woodberry, Associate Quality Assurance Engineer
D. Ringley, Tester/Technician

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* Attended the entrance meeting on April 13, 1993
\+ Attended the exit meeting on April 16, 1993

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

JUN 28 1993

Docket No. 99901264

Mr. Malcolm M. McQueen, President
Fluid Components, Incorporated
1755 La Costa Meadows Drive
San Marcos, California 92069

Dear Mr. McQueen:

SUBJECT: NOTICE OF VIOLATION AND NOTICE OF NONCONFORMANCE (NRC INSPECTION REPORT 99901264/93-01)

This letter addresses the inspection of Fluid Components, Incorporated (FCI) at San Marcos, California, conducted by Messrs. R.C. Wilson and S.D. Alexander of this office on April 13-15, 1993, and the discussion of their findings with you and members of your staff on April 15, 1993. The purpose of the inspection was to review FCI activities conducted under your 10 CFR Part 50, Appendix B, quality assurance program and 10 CFR Part 21 reporting program, with emphasis on flowmeter calibration.

Areas examined during the Nuclear Regulatory Commission (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 inspectors found that the implementation of your quality assurance program failed to meet certain NRC requirements. Specifically, FCI did not actually determine or document the accuracy of safety-related air flowmeters that were certified to an accuracy of ±3% of full scale, and did not appear to have identified or quantified all of the applicable error sources. In addition, procedures were lacking for significant calibration activities, the procedure that did cover production unit calibration was not properly controlled and was not always followed in practice, and vendors providing calibration services were not audited to verify their technical and quality programs. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter.

The inspectors also identified that certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (Enclosure 1). Specifically, FCI failed to initiate timely evaluations of two potential deviations identified by discrepancy reports dated October 4, 1992, and February 15, 1993, which reported errors of as much as 7 to 16% of reading in transfer standards used to calibrate flow switches supplied for safety-related service with certified accuracy of ±3% of full scale. Even though your subsequent evaluation reportedly showed that these concerns did not necessarily violate the certified accuracy specification of delivered flow switches, that information was not developed within the reporting period specified by 10 CFR 21.21.

Further, your procedure for implementing 10 CFR Part 21 that was in effect at the close of the inspection on April 15, 1993, did not reflect revisions which had instituted substantial changes in evaluation and reporting requirements; it could preclude reporting of deviations by employees; it restricted the scope of failures to comply that would be reported; and it did not ensure that all affected licensees or purchasers are informed of deviations that FCI determines it cannot evaluate. In addition, the issue of 10 CFR Part 21 effective October 29, 1991 was posted rather than a current copy.

Flowmeters and flow switches supplied by FCI are used in numerous commercial nuclear power plants licensed by the NRC. Failures of these instruments could significantly impact plant safety. We expect that you will develop a plan to identify and correct deficiencies in your calibration program, including underlying causes. I trust that as you carry out that plan, you will bear in mind your responsibilities under 10 CFR Part 21 to report any safety-related deviations that you discover that could affect previously-shipped instruments, as well as current production. After you have completed your upgraded program to meet the quality assurance requirements of Appendix B to 10 CFR Part 50, we will perform a followup inspection of your facility that will also cover the aspects of your safety-related quality assurance program that were not included in the present inspection.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice of Violation 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 this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

Further, please provide us within 30 days from the date of this letter a written statement in accordance with the instructions specified in the enclosed Notice of Nonconformance.

In accordance with 10 CFR 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.

The responses directed by this letter and the enclosed notices 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.

Sincerely,

Leif J. Norrholm

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

Enclosures:
1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99901264/93-01

Enclosure 1

NOTICE OF VIOLATION

Fluid Components, Incorporated Docket No.: 99901264/93-01
San Marcos, California

During an NRC inspection conducted on April 13 through 15, 1993, 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. 10 CFR 21.21, "Notification of Failures to Comply or Existence of a Defect and Its Evaluation," requires, in part, that each corporation subject to the regulations adopt appropriate procedures to ensure the evaluation and proper reporting of deviations and failures to comply, and to submit an interim report to the Commission if the evaluation of the deviation or failure to comply cannot be completed within 60 days. Section 21.21 further requires that if a deviation or failure to comply is discovered by a supplier of basic components or services associated with basic components, and the supplier determines it does not have the capability to perform the evaluation to determine if a defect exists, the supplier must inform the purchasers or affected licensees within five working days of this determination so that the purchasers or affected licensees may evaluate the deviation or failure to comply.

   Contrary to the above, Fluid Components, Incorporated (FCI) was informed in two discrepancy reports of significant errors in transfer standards used to calibrate delivered basic components--i.e., flow switches--and failed to evaluate the impact of the errors on the accuracy of the basic components. Specifically, Discrepancy Reports No. 02726 dated October 4, 1992, and No. 02914 dated February 15, 1993, reported that two transfer standard turbine flowmeters differed by as much as 7 to 16% of reading from the sonic nozzle traveling standards six months after their previous calibrations. These errors could potentially cause the basic components to deviate from their technical procurement specifications for accuracy. Thus, FCI failed to evaluate a possible deviation to determine if it could create a substantial safety hazard within the 60 days prescribed by 10 CFR Part 21, nor was an interim report made to the Commission as required when the evaluation was not completed within the allotted time, nor were all affected licensees or purchasers informed of the deviation. Although your subsequent evaluation of the second instance determined that the specific error reported did not apply to purchase orders for basic components, that information was not developed within the reporting period specified by 10 CFR 21.21.

This is a Severity Level IV violation (Supplement VII). (99901264/93-01-01)

B. 10 CFR 21.21, "Notification of Failure to Comply or Existence of a Defect and Its Evaluation," requires, in part, that each corporation subject to the regulations adopt appropriate procedures for either evaluating deviations and failures to comply, or informing the licensee or purchaser of the deviation or failure to comply. In addition, 10 CFR 21.6, "Posting Requirements," requires posting a current copy of 10 CFR Part 21.

Contrary to the above requirements, at the time of the inspection FCI had not revised Quality Assurance Procedure 704011, "10CFR21 Reporting of Defects and Nonconformances," Revision B, October 31, 1988, to address the substantive revisions to 10 CFR Part 21 that became effective on October 29, 1991, and November 24, 1992. In addition, the procedure could preclude reporting of deviations by employees, it unduly restricted the scope of failures to comply that should be reported, and it lacked provisions to ensure notification to affected licensees or purchasers of deviations or failures to comply when FCI was unable to determine if a defect existed. Further, FCI had posted the issue of 10 CFR Part 21 effective October 29, 1991, rather than a current copy.

This is a Severity Level V violation (Supplement VII). (99901264/93-01-02)

Pursuant to the provisions of 10 CFR 2.201, Fluid Components, Incorporated, 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 Licensee Performance, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 28th day of June 1993

Enclosure 2

NOTICE OF NONCONFORMANCE

Fluid Components, Incorporated
San Marcos, California

Docket No.: 99901264/93-01

Based on the results of an inspection conducted on April 13 through 15, 1993, it appears that certain of your activities were not conducted in accordance with NRC requirements.

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

Section 2, "Quality Assurance Program," of Fluid Components, Incorporated's (FCI's) Quality Assurance Manual No. 8003, Revision K, dated January 10, 1991, states that the quality assurance program set forth in the manual complies with the quality system requirements of Appendix B to 10 CFR 50.

Contrary to the above, the NRC inspectors found numerous examples of failure to adequately prescribe or accomplish calibration activities necessary to support accuracy certifications for safety-related type LT81A flowmeters. Specific examples include the following for flowmeters shipped to the Virginia Electric Power Company on September 30, 1991, and March 11, 1992:

1. The accuracy of flowmeters certified to ±3% of full scale was not actually determined or documented, and not all of the error sources affecting accuracy were identified.

2. The procedure for calibration activities was not properly controlled, was not always followed in practice, and did not address all calibration activities. (99901264/93-01-03)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Vendor Inspection Branch, Division of Reactor Inspection and Licensee Performance, 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 28th day of June 1993.

ORGANIZATION:	FLUID COMPONENTS, INCORPORATED 1755 LA COSTA MEADOWS DRIVE SAN MARCOS, CALIFORNIA 92069
REPORT NO.:	99901264/93-01
CORRESPONDENCE ADDRESS:	Malcolm M. McQueen, President Fluid Components, Inc. 1755 La Costa Meadows Drive San Marcos, California 92069
ORGANIZATIONAL CONTACT:	Stephen R. Mitchell, Quality Assurance Manager (619) 744-6950
NUCLEAR INDUSTRY ACTIVITY:	Flow and level instruments for all types of commercial nuclear power plants
INSPECTION CONDUCTED:	April 13-15, 1993
TEAM LEADER:	RCWilson 6/17/93 Richard C. Wilson, Senior Engineer — Date Reactive Inspection Section 2 (RIS2) Vendor Inspection Branch (VIB)
OTHER INSPECTOR:	Stephen D. Alexander, Equipment Qualification and Test Engineer, RIS2, VIB
APPROVED:	Gregory C. Cwalina, Chief — Date Reactive Inspection Section 2 Vendor Inspection Branch
INSPECTION BASES:	10 CFR Part 21 and 10 CFR Part 50, Appendix B
INSPECTION SCOPE:	To selectively review the implementation of FCI's quality assurance program for supplying nuclear safety-related equipment, with emphasis on the calibration of flowmeters.
PLANT SITE APPLICABILITY:	Numerous

1 INSPECTION SUMMARY

1.1 Violation 99901264/93-01-01 (Open)

Contrary to the requirements of 10 CFR Part 21, Fluid Components, Incorporated (FCI) did not begin timely evaluations of discrepancy reports describing potential deviations involving flow switch accuracy (see Section 3.6 of this inspection report).

1.2 Violation 99901264/93-01-02 (Open)

Contrary to the requirements of 10 CFR Part 21, FCI procedures reflected an obsolete revision of 10 CFR Part 21 and did not properly address reporting and evaluation requirements, and FCI had not posted a current copy of 10 CFR Part 21 (see Section 3.7 of this inspection report).

1.3 Nonconformance 99901264/93-01-03 (Open)

Contrary to several criteria of Appendix B to 10 CFR Part 50, which was invoked on FCI by licensee purchase orders, FCI certified the accuracy of flowmeters as ±3% of full scale without an adequate basis because of numerous flaws in the calibration process (see Sections 3.4 and 3.5 of this inspection report).

2 STATUS OF PREVIOUS INSPECTION FINDINGS

There was no previous NRC inspection of this facility.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on April 13, 1993, the NRC inspectors discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with FCI management and staff. In the exit meeting on April 15, 1993, the inspectors discussed their findings and concerns with FCI management and staff.

3.2 Inspection Scope

FCI designs and manufactures fluid flow and liquid level instruments. The company was established in 1964, has about 140 employees, and occupies about 50,000 square feet. Commercial nuclear power plant business peaked at about 40% of the total, and recently has been in the 10-15% range.

The inspection concentrated on the calibration of type LT81A and similar mass flowmeters, which are frequently supplied for air flow measurements in commercial nuclear power plants. Other areas addressed were the handling of discrepancy reports and the program for meeting the reporting requirements of 10 CFR Part 21. The FCI facilities were observed, again with emphasis on calibration. FCI personnel stated that materials were procured commercial grade, certified material test reports were obtained, and material from every vendor was sampled every six months or upon purchase.

The FCI Type LT81A mass flowmeter uses a patented thermal dispersion principle. A fluid flows across two resistance temperature detectors (RTDs), one of which is preferentially heated by a heating element. The temperature difference between the RTDs varies with fluid flow, and is greatest at zero flow. Electronic circuitry converts the difference in RTD resistances to an output signal that is essentially linear with flow. The signal from the unheated RTD is also used for process fluid temperature compensation.

The NRC inspectors selected seven safety-related purchase orders (POs) for flowmeters from a list of about 150 nuclear plant POs provided by FCI. One of the seven was in-process and had not reached the calibration stage, so the calibration was not reviewed. Another PO covered steam flowmeters, which are calibrated differently. Their calibration was not reviewed, but the inspector noted that the drawing specified "Accuracy ±3%."

The customer requirements in each PO were reviewed. The POs covered original and replacement equipment. They specifically invoked 10 CFR Part 21 and Appendix B to 10 CFR Part 50, as well as the technical requirements of applicable earlier POs. In some cases calibration traceable to the National Institute of Standards and Technology (NIST) was required.

FCI prepared an assembly drawing for each original equipment PO, and the original equipment drawings were used for the replacement equipment POs. In each case the drawing specified the "linearizable flow range" and stated "Accuracy: ±3% of full scale." The only exception reviewed by the inspectors was drawing no. 706146 sheet 3, Revision R, for Georgia Power Company (GPC), which specified repeatability as ±1% of range and accuracy as ±5% of range. This drawing applied to six flowmeters on Sales/Shop Orders (S.O.s) 17586 and 17596, GPC POs P-50658 and P-50659, all dated in February 1989 with calibrations performed on Stand CL on March 9 and 16, 1989. The GPC orders did not clearly invoke Appendix B to 10 CFR Part 50, but did invoke industry standards for environmentally qualified equipment, referenced the FCI qualification test report, and stated that 10 CFR Part 21 applies to nuclear safety items.

All of the flowmeters reviewed had turndown ratios (ranges) of 10:1 or less. The catalog provided by FCI during the inspection specified an accuracy of ±1% of full scale or ±3% of reading, whichever is better, for turndown ratios of 10:1 or less in air, and repeatability of ±1% of full scale. The FCI Certificates of Conformance stated that the instruments were certified to have been manufactured, tested, and inspected to the requirements of the PO.

3.3 Description of Flowmeter Calibration

The calibration of an LT81A air mass flowmeter involves the following major steps (excluding the display and totalizer calibrations, which the inspectors did not review), using "Document # 008072, LT81 Calibration Procedure Board #0017 Rev. B," Revision B, February 7, 1989, for step (1), with no documented procedure for the remainder.

(1) After temperature compensation, the LT81A is installed in a calibration test stand containing three turbine flowmeters used as transfer standards. The nonlinearized voltage from the sensor head ("Pin 6 voltage") is recorded at ten different flow rates covering the range of the instrument. The computer-processed reading of one of the turbine meters is taken as the actual air flow through the LT81A at each flow rate. Potentiometers in the circuitry are adjusted to provide an output signal (usually 4-20 mA) that is linear with flow. A five-point final check of the linearized output signal against the Pin 6 voltage is then made in the test stand.

The calibration procedure states that the allowable tolerance on the output is 1% of full scale, and specifies that if any signal does not fall within tolerance "some slight adjustments to the calibration will be required," as directed by an experienced technician during the final check. (As noted above, FCI specifies and certifies ±3% of full scale accuracy; the 1% value is an in-process criterion. The "4 to 20 mA calibration table" sheet, used to record the final five-point check of linearized output vs. flow that is specified in the calibration procedure, states at the bottom: "max. deviation = ± 0.16 mA @ 1.0 %." This is another in-process criterion, amounting to 1% of span in this case.)

The calibration procedure also states "look for Pin 6 repeatability from the original Pin 6 data," but does not specify any further action with regard to the Pin 6 data. The 4-20 mA sheets observed by the inspectors showed that, on the final check, if the Pin 6 voltages did not agree with the values corresponding to the desired flow for a given output within well under 1%, the flow was adjusted until close agreement was reached.

With some exceptions, manufacturer's records and data for the standard flowmeters were not provided to the inspectors. At least three manufacturers were represented, one of the meters was described as custom, and the manufacturer of one was not evident by observation.

Two open loop test stands, A and B, have normally been used for safety-related flowmeters. The inspectors noted that a 1989 safety-related PO used Stand CL. It contained an anemometer as a transfer standard, as did Stand A for a 1988 safety-related calibration, in lieu of the presently-used turbine flowmeters.

(2) The transfer standard turbine flowmeters are individually calibrated at multiple flow values against sonic nozzle traveling standards (low flows) or against traveling standard 4" turbine flowmeter FM-46 (insufficient air pressure is available to drive sonic nozzles in 4" pipes). A computer then fits a curve to the specific calibration data points for each transfer standard. When calibrating delivered flowmeters per step (1) above, it is the values from the curve that are used as "actual" flow values. FCI did not consider possible changes with time (drift) of the turbine flowmeters in the calibration process; this concern is addressed in Section 3.5(3) of this inspection report.

(3) For low flow rates, four sonic nozzles (FM-81, -82, -83, and -84) served as traveling standards used to calibrate the smaller turbine flowmeters installed in the test stands. They were calibrated against other sonic nozzles at Flow Dynamics, Inc. in Scottsdale, Arizona, in October 1991, with an estimated error of ±0.25 percent against the NIST standard. The curves fit the data points within 0.17% of reading or better, whereas the inspectors observed errors of as much as 0.4% of reading in the turbine flowmeter curve fits. The NRC inspectors did not observe any data reflecting drift of the sonic nozzles with time, and FCI did not include drift error in accuracy determinations. The inspectors did not investigate calibration of the transfer standards used for low flows prior to use of the sonic nozzles.

(4) For higher flow rates, 4" turbine flowmeter FM-46 served as a traveling standard for the turbine flowmeters installed in the test stands. It was calibrated annually, beginning in 1989, at the Colorado Engineering Experimental Station, Inc. in Nunn, Colorado, with an error estimated by the laboratory each time at ±0.5% of reading vs. the NIST standard. As for the other standard flowmeters, FCI did not include drift errors in determining accuracy.

3.4 Calibration Error Sources

The NRC inspectors had numerous concerns with the calibration of FCI type LT81A air flowmeters for nuclear safety-related service. These concerns generally related to the failure to issue, and follow, adequate calibration procedures. Specific concerns identified by the inspectors during the inspection, in reviewing certain calibration records after the inspection, and in telephone discussions with the FCI QA manager from May 10-26, 1993, are addressed in this section (calibration error sources) and the next section (additional concerns) of this inspection report.

FCI personnel stated that licensee inspectors generally did not look beyond the simple calibration of a delivered flowmeter against the transfer standard. The NRC inspectors also observed that a recent audit of FCI by an industry group team did not raise the concerns addressed in this inspection report.

The NRC inspectors found no evidence that FCI had identified or combined all of the various calibration errors to determine the absolute accuracy of delivered flowmeters. The failure of FCI to adequately consider all error sources in calibrating safety-related flowmeters constitutes a portion of Nonconformance 99901264/93-01-01.

The NRC inspectors tabulated the identifiable quantified error sources involved in calibrating two flowmeters on Stand B on March 10, 1992, under FCI S.O. 32263. These flowmeters were type LT81A, drawing 88-138561, Revision A. The customer was Virginia Electric Power Company (VEPCO) under PO SSY-368340, dated December 3, 1991, and the FCI sales representative was United Control Co. of Richmond, Va. The invoice stated that the shipping date was March 11, 1992. For flowmeter serial number 3680-1 (1-10 ft/sec range) the calibration path to NIST with the largest identified errors was through turbine meter FM-78 and sonic nozzle FM-84. The sum of the absolute values of the quantifiable errors in this path was about 1.4% of full scale. For serial 3681-1 (8.61-51.66 ft/sec) the calibration path to NIST with the largest identified errors was through turbine meters FM-63 and FM-46. The absolute sum of the errors for this path was about 2.0% of full scale.

The NRC inspectors also estimated the quantified errors in calibrating four type LT81A flowmeters on Stand A on 9/27/91 for VEPCO under FCI S.O. 29907, shipped on September 30, 1991. Two flowmeters were 1-10 ft/sec units (serial numbers 3491-1 and 3492-1), and two were 8.61-51.66 ft/sec units (serial numbers 3493-1 and 3494-1). The absolute sum of the errors estimated for each of the four totalled about 2.0 percent of full scale.

The inspectors considered the following error sources in these tabulations:

- output voltage and Pin 6 voltage deviations during the final check
- calibration, curve fit, and drift errors for turbine meter standards
- calibration and curve fit errors for sonic nozzle standards

These are the only error sources for which the inspectors could obtain values. The following additional error sources could not be quantified.

(1) The practice of making circuit adjustments against Pin 6 voltages rather than standard turbine meter readings, during the final check of the LT81A flowmeter, makes its calibration subject to the repeatability error of the sensing head, which is in effect used as a transfer standard for the final check. The final check then becomes only a check of the circuit adjustments, rather than of the entire flowmeter, and that check is influenced by the repeatability of the sensing head. The inspectors had no basis for estimating the drift error, although the FCI catalog and one of the assembly drawings stated a repeatability of ±1% of full scale for the entire flowmeter.

(2) The inspectors found no data for estimating the drift of the sonic nozzle standards.

(3) The inspectors observed range gaps in the calibration of flowmeters against standards for the specific cases discussed above. [This matter is discussed from the procedural standpoint in Section 3.5(2) of this inspection report.] Specifically, in Stand B, there was a gap in the calibration of standard turbine meter FM-78 against sonic nozzles FM-83 and FM-84, from 21.4 to 24 ft/sec, and a gap in the calibration of standard turbine meter FM-63 vs. sonic nozzle FM-84 and turbine meter FM-46, from 114 to 144 ft/sec. In Stand A, there was a gap in the calibration of standard turbine meter FM-47 from 4.35 to 5.65 ft/sec against sonic nozzles FM-56 and FM-57, and a gap in calibrating the delivered flowmeters serial numbers 3493-1 and 3494-1 against turbine meters FM-47 and FM-59 from 17.2 to 26.7 ft/sec (more than 20% of their calibrated range). For stand B, in each case one of the ten calibration points, but none of the final check points, fell in the gap. For Stand A, in each case two calibration points and one final check point fell in the gap.

The inspectors did not attempt to estimate the errors resulting when calibration or final check points fell within uncalibrated gaps. Where standards overlapped, the inspectors included the larger of the two errors in the estimates given above.

(4) Data sheet entries discussed in Section 3.5(4) of this inspection report suggest that the air flow used in open loop test stands was not pressure-corrected, so that necessary density corrections were not made and additional error was introduced into the calibration.

(5) For flowmeter serial 3680-1, neither the top of range or bottom of range point was included in the final check, as discussed in Section 3.5(9) of this inspection report. The effect of this omission was not estimated, but it could be

significant. Slightly less serious was the omission of the bottom of range points for serial numbers 3491-1, 3492-1, and 3493-1.

3.5 Additional Inspector Concerns Regarding Calibration

In addition to the sources of experimental error discussed in Section 3.4 above, the inspectors noted the following additional anomalies and inconsistencies in the flowmeter calibration process. The failure of FCI to prevent or correct these deficiencies in calibrating safety-related flowmeters constitutes a portion of Nonconformance 99901264/93-01-03.

(1) The "In-House Certificate of Calibration" data sheets used to document calibration of the transfer standard flowmeters did not provide a space for identifying the calibration test stand, although "A Stand" or "B Stand" was sometimes written on the sheet in the title block area. The "Test Department Calibration Data: Actual Test Conditions" form used to document calibration of delivered flowmeters did provide a space for identifying the stand, and provided for identifying the decade resistance boxes used for calibration, but did not identify the standard turbine flowmeters. The sheet titled "4 to 20 mA Calibration Table," which contains the results of the final five-point calibration check, identified neither stand nor standard.

In a few instances, in response to questioning, FCI produced a "Final Acceptance Test Procedure" form that sometimes gave additional information. This form, although listed on the Shop Order front sheet, is not listed on the "Pre-Flight" sheet for inclusion in the flowmeter calibration files, and was, in fact, not included in any of the calibration files reviewed by the NRC inspectors.

The NRC inspectors noted that the transfer standards (but not the test stands) used for calibrating nonsafety-related flowmeters were identified on "Certificates of Calibration," where traceability to NIST was specified; a commercial grade procurement by a licensee was so documented. However, the "Certificates of Calibration" observed by the inspectors for safety-related POs did not identify the transfer standards.

FCI personnel stated that the transfer standard flowmeters installed in the test stands were not changed. However, the seven POs reviewed by the NRC inspectors contained four instances of transfer standard identification discrepancies that involved use of different transfer standards, including use of a different type of standard and an unidentifiable standard.

- The 4" standard turbine meter apparently used in Stand A for the example discussed in Section 3.4 above was FM-59. According to its records, FM-59 was previously calibrated on

October 11, 1990; April 16, 1991; and September 11, 1991. The April 16, 1991, "In-house Certificate of Calibration" sheet indicated that FM-59 was in Stand B, while the other two calibration sheets designate Stand A. FCI stated that the discrepancy was due to a clerical error in entering information in the title block area of the calibration sheet, but could offer no evidence in support of this assertion.

- The "Equipment Used" block of the "In-House Certificate of Calibration Sheet" for standard turbine meter FM-47 dated September 10, 1991, showed sonic nozzles FM-55 and FM-56 as the traveling standards used to calibrate it, while the data tables on the same sheet showed FM-56 and FM-57.

- Stand CL was used to calibrate six flowmeters under S.O.s 17586 and 17596, Georgia Power Co. POs P-50658 and P-50659, on March 9 and 16, 1989. For each of these flowmeters the calibration data sheet states that the transfer standard was a Davis anemometer, FCI tag no. EL010. However, FCI's records show that EL010 is a digital voltmeter. FCI personnel were unable to identify the transfer standard during the inspection.

- On August 29, 1988, FCI calibrated two type LT81A flowmeters on S.O. 13856, shipped on August 30, 1988, under VEPCO PO SSY-178975 dated March 28, 1988, which imposed the requirements of 10 CFR Part 50, Appendix B, and invoked 10 CFR Part 21. However, the August 29, 1988, calibration date for these delivered instruments preceded the earliest calibration records for the turbine meter transfer standards examined by the inspectors. Upon questioning, FCI personnel produced a "Final Acceptance Test Procedure" sheet for the S.O. that listed the standard as "EL-74." FCI's commercial grade calibration records also showed that flowmeter EL-74 was used in Stand A around August 1988.

  The calibration records for EL-74 identified it as a 0-80 ft/sec Davis anemometer. It was calibrated on August 9, 1988, using four "rotometers" [sic], numbers FM-139, -145, -146, and -147. The required accuracy was listed as "± .75 FPS (1% of range)." However, EL-74 was only calibrated up to 25.04 ft/sec, whereas it was supposedly used to calibrate LT81A serial number 2791-1 up to 51.66 ft/sec. Other discrepancies on the calibration data sheet were that it was not signed by the QA representative as required, nor were the as-received data, procedure number, and serial number recorded. A previous calibration of EL-74 (February 8, 1988) only extended up to 36.51 ft/sec, and letters from the manufacturer indicated calibration only up to 34 ft/sec.

When questioned by the NRC inspector, FCI justified extension of EL-74's use beyond its calibrated range on the basis of "engineering R&D" testing performed on November 1, 1988 (two months after hardware shipment), when EL-74 was compared to sonic nozzles (of unspecified identity and accuracy) up to 70 ft/sec and its K-factor (analogous to that of a turbine meter) varied from 152.5 to 155 cycles per cubic foot.

These uncertainties as to which test stands and standard flowmeters were used for calibrating previously delivered flowmeters raise questions concerning the accuracy of the calibrations. Further, the NRC inspectors did not attempt to identify or evaluate the accuracy of standard flowmeters other than for the specific samples selected.

(2) As addressed in Section 3.4(3) of this inspection report, FCI's calibration of a flowmeter often requires the use of multiple standards, whose ranges may or may not overlap, each covering part of the range of the meter being calibrated. The NRC inspectors found no evidence that FCI had evaluated whether the multiple transfer standards used to cover the range of a delivered flowmeter covered the full range, with overlap and with acceptable accuracy.

The inspectors also found no documentation addressing setup or use of a test stand that addressed the gaps or overlaps of the standard flowmeter ranges. Therefore, the origin of the data points for the so-called "actual flow" on the Test Department Calibration Data sheets in the gap regions was not clear. The inspectors further noted that the gaps changed--and could appear and disappear--with the semi-annual or annual recalibrations of the standards, depending on the end points of the calibrations.

For calibration of transfer standard turbine flowmeters against traveling standards (sonic nozzles and turbines), the "In-house Certificate of Calibration" form shows which standard was used as the reference for each flow point. For calibration of delivered flowmeters, neither the "Test Department Calibration Data" sheet used to record calibration data, nor the "4 to 20 mA Calibration Table" sheet used to record the final five-point check, nor any other sheet in the file, identifies which transfer standard meter was used for which flow point where the standards overlap.

The inspectors noted that the upper boundary of each gap was often formed by the low-flow end of a turbine flowmeter's calibration, where errors are usually largest, the calibration curves nonlinear, and extrapolation questionable. The lower boundary was formed by the high flow end of the calibration of either a turbine flowmeter--where overspeed is a concern--or a sonic nozzle--where an adequate supply of high-pressure air is necessary.

(3) The NRC inspectors found no evidence that FCI had accounted for the effect of transfer standard drift on the accuracy of delivered LT81A flowmeters, between calibrations of the standards. A transfer standard calibration was usually used for approximately six months (one year for the traveling standard), or until a calibration shift was suspected. If the next calibration differed significantly, bearing replacements and possibly other repairs were made, and new calibrations performed, without consideration for the LT81A flowmeters calibrated and shipped during that interval.

For the traveling transfer standard FM-46, the 1990 and 1991 calibration curves agreed very closely, but were consistently about ½% higher than the 1989 curve and ½% lower than the 1992 curve (except at very low flows, where the changes were larger). All of the calibrations were performed at CEESI. Upon review of the December 11, 1992, data FCI decided to replace the turbine bearings and have the meter recalibrated. That effort was in progress during the inspection. FCI personnel suggested that, since FM-46 was used less than the transfer standard turbine meters installed in the test stands, its bearings may have seated and worn more slowly.

Transfer standard flowmeter FM-47 was used in Stand A during calibration of the delivered flowmeters discussed in Section 3.4 above, based on its calibration on September 10, 1991. The next recorded calibration of a 1.5" flowmeter in Stand A was on March 25, 1992, for FM-87. On the next FM-47 calibration sheet dated October 26, 1992, the technician noted that it had been modified and out of service. The inspectors could not determine at what point--before or after its presumed use for calibrating the LT81As--FM-47 developed a non-correctable error or other condition that caused it to be replaced and modified. Also, there was no indication in the records that FCI evaluated the impact of FM-47's presumed failure on the accuracy of the delivered flowmeters that it was used to calibrate. It is therefore possible that significant error could have been introduced into LT81A calibrations.

The failure of FCI to address discrepancy reports concerning transfer standard flowmeter drift in a timely manner is identified as a violation of 10 CFR Part 21 in Section 3.6 of this inspection report.

(4) On the sheet titled "Virginia Power LT81 Equivalent Air Calibration Data 9 March 92" for flowmeter 3680-1, the loop test stand calibration pressure is stated as 0.00 psig. For the "Test Department Calibration Data" sheets for flowmeters 3680-1 and 3681-1, "amb" is entered in the block titled "Pressure". No pressure is recorded for the final check. The purpose of the pressure measurement is to permit density-correcting the measured air flow rate to standard cubic feet per minute. Gauge pressure

cannot be used for this purpose, since it represents only the difference between the test loop pressure and the ambient atmospheric pressure at the time, neither of which was recorded. This suggests that calibration corrections were not made for pressure. Even if corrections were made, the data sheets do not record the basis for the corrections.

(5) The inspectors noted that the calibration process did not address response to variations in ambient conditions such as temperature, humidity, voltage, and frequency that should be included in accuracy determinations. In this context the temperature concern is with error introduced by ambient air effects, most likely on the circuitry, rather than the process air temperature variation for which the LT81A flowmeters are temperature-compensated. The inspectors did not investigate the treatment of harsh environment effects related to environmental qualification, which would not be related to normal environment calibration.

(6) FCI used Document # 008072, "LT81 Calibration Procedure, Board #0017 Rev. B," Revision B, issued February 7, 1989, to calibrate delivered flowmeters. This procedure was not signed or approved, it was not under Appendix B control, and there was no requirement in the Appendix B document hierarchy to use it or any other calibration procedure. Furthermore, no procedures for calibration of the transfer standard flowmeters were found.

(7) The calibration accuracy of delivered flowmeters was never recorded in the files reviewed by the inspectors, nor were accuracy calculations included.

(8) The "4 to 20 mA Calibration Table" sheets used to record the final calibration check data for delivered flowmeters lack information such as technician identity (signature, stamp), date, identification of test stand and standard flowmeters.

(9) The inspectors noted two significant discrepancies in the documentation for flowmeter serial number 3680-1 for VEPCO:

- In the final calibration check on the "4 to 20 mA Calibration Table" sheet, the technician did not check the 4 mA output point (bottom of range) as typed on the sheet.

- Near the title block of the "Test Department Calibration Data" sheet, "10.70" was entered as the range high, with the unsigned, undated notation "wrong, should be 10.07." A similar notation was made near the 10.70 ft/sec entry in the calibration data table, and both the original calibration and final check Pin 6 voltages are crossed out. A new and slightly lower Pin 6 voltage was entered next to the crossed out values (presumably the voltage measured for 10.07 ft/sec at the time of the notations), but there was no original calibration voltage recorded to compare with the check value. The original calibration evidently contained an error, which apparently was discovered during the final check, at which time the notations apparently were added.

The effect of these two errors is that two of the five points specified in the calibration procedure for final check, including both endpoints, were not checked for flowmeter 3680-1. The 120% over-range point at 12.09 ft/sec was also checked, although it was not covered by procedure or noted on the final check sheet and no deviation was recorded. The only in-range points checked were at about 3.3, 5.4, and 7.8 ft/sec for the 1-10 ft/sec instrument. (NOTE: the difference between the 1-10 ft/sec range specified by the licensee and the 1.007-10.07 ft/sec calibration range reflects the difference between the licensee's specified 74°F normal operating temperature and FCI's 70°F standard temperature.)

(10) The practice of making circuit adjustments against Pin 6 voltages rather than standard turbine meter readings during the final check of the LT81A flowmeter is not spelled out in the calibration procedure. Further, the technician's entries appeared to deviate from the intended format of the "4-20 mA sheet" for the sheets observed by the inspectors. The first column, headed "Indicated Flow," was left blank. The second and third columns, headed "Actual Flow" and "Signal Output," contained typed entries of the form "8.673 f/sec = 4.000 mA," with lines for entering "Indicated" and "Deviation" in the third column; it is in these spaces that output voltages were recorded. The last two columns share the heading "Pin 6 Volts," but entries were made in only the first of these columns; those values were also entered in an unlabeled column on the "Test Department Calibration Data" sheet containing the original calibration data for the product being calibrated.

Also on the "4-20 mA" sheets, the technician used "signal output" values of about ¼ of those typed on the sheet (e.g., 4.999 instead of 20 mA) without explanation. The calibration procedure specifies "flow the unit at the 5 points indicated on the 4-20 sheet." FCI personnel explained that the recorded values are voltages measured across a 250 ohm resistor, but the "4-20 sheet" continues to show mA. The inspectors did not investigate the tolerance of the resistor or the error resulting from its use.

(11) There was no documentation that either calibration laboratory used by FCI--Colorado Engineering Experimental Station, Inc. or Flow Dynamics, Inc.--was capable of performing safety-grade calibrations traceable to NIST; e.g., no evidence of a technical or QA audit by FCI or indication of an Appendix B program at either place.

(12) The inspectors noted that three different units of flow were involved in calibrations, which complicates the overall error analysis. Calibrations of the traveling standard turbine meter FM-46, and of the transfer standard turbine meters against any of the traveling standards, were expressed in units of K-factor (cycles per cubic foot) vs. output frequency in Hertz; the readings from the turbine meters during calibration of the delivered flowmeters were expressed in standard cubic feet per minute (scfm); and the LT81A meters were calibrated in feet per second (ft/sec). The calibration procedure (Document # 008072) did include formulas for computing flow rates and tables of conversion factors.

3.6 Handling of Discrepancy Reports

The NRC inspectors reviewed FCI's Discrepancy Report (DR) logs for 1992 and 1993 and selected two DRs affecting the calibration of turbine flowmeters installed in the A and B stands used to calibrate nuclear safety-related flowmeters: DR 02726, dated October 4, 1992, and DR 02914, dated February 15, 1993. These DRs reported that two of the transfer standard turbine flowmeters differed by as much as 7 to 16% of reading from the sonic nozzle traveling standards six months after their previous calibration. FCI had not dispositioned either DR at the time of the inspection. For three other DRs issued during the same time period that did not involve nuclear safety-related equipment, the inspectors found that one had been closed after notifying the customer, one had not been acted on, and for the third the customer had been notified and the DR not yet closed.

The NRC inspectors noted that Section 5 of FCI QA Procedure 704029, "Evaluation of Measuring and Test Equipment," Revision D, dated June 26, 1987, requires that if a piece of calibrating equipment is found to have been received by the calibrator in an out of tolerance condition, records shall be checked to determine if the equipment was used to perform any final acceptance tests; if so, a DR should be submitted to the Material Review Board for disposition. QA Procedure 704004, "Discrepancy Report," Revision F, dated June 17, 1991, discusses the processing of DRs, but no required time frame is identified. The NRC inspectors found no evidence of review or evaluation of DRs 02726 and 02914.

On May 10, 1993, the NRC inspector asked FCI by telephone for the results of evaluations of the two nuclear-related DRs. The FCI QA manager stated that neither had been dispositioned, and initiated evaluations in response to the telephone call.

DR 02726 addressed turbine flowmeter FM-87, installed in calibration Stand A. In a May 13, 1993, telephone call FCI stated that Stand A was used for three nuclear safety-related POs in the six-month period prior to issue of the DR (i.e., the interval when drift occurred after the previous semi-annual calibration was performed). Each PO involved type FR72-4 flow switches, rather than analog flowmeters. The affected S.O.s are:

- S.O. 35607, Tennessee Valley Authority, two units calibrated September 29, 1992
- S.O. 34622, Tennessee Valley Authority, five units calibrated July 23, 1992
- S.O. 35682, Northeast Utilities Corporation, five units calibrated October 5, 1992

In a May 26, 1993, telephone call the FCI QA manager stated that all of the subject flow switches were certified to ±3% of full scale accuracy, and full scale was at least 50 ft/sec. The maximum observed drift in FM-87 as reported in DR 02726 was 0.22 ft/sec, which is 7.6% of reading at about 3 ft/sec, but only about 0.44% of full scale or less for the delivered flow switches. Thus, the DR focused on a bottom-of-range error that represented a large percentage of the actual reading, but only a small error in terms of the full scale accuracy certified for the delivered flow switches.

DR 02914 addressed turbine flowmeter FM-78, installed in Stand B. FCI reported in the May 13, 1993, telephone call that no nuclear POs used Stand B in the six months prior to the date of the DR.

DR 02726 was written on October 4, 1992, and DR 02914 was written on February 15, 1993. Each reported a potential deviation for basic components that FCI should have evaluated for reportability. 10 CFR 21.21 requires evaluating deviations within at most 60 days of discovery, or providing an interim report. FCI did not begin evaluation until telephoned by the NRC inspector on May 10, 1993. Even though subsequent evaluation by FCI reportedly showed that the concerns raised in these DRs did not necessarily violate the certified accuracy specification of delivered flow switches, the failure to evaluate DRs 02726 and 02914 in a timely manner constitutes Violation 99901264/93-01-01.

3.7 10 CFR Part 21 Program

The NRC inspectors reviewed FCI QA Procedure 704011, "10CFR21 Reporting of Defects and Non-Conformances," Revision B, dated October 31, 1988. This was FCI's current procedure for reporting defects and noncompliances pursuant to 10 CFR Part 21. The procedure did not contain the time limits for notification or the requirements for an interim report that have been added to 10 CFR Part 21 since 1988.

The inspector pointed out to FCI that the term "Non-Conformances" used in the title of Procedure 704011 has no meaning in the context of the language of Part 21. The title of 10 CFR Part 21 is "Reporting of Defects and Noncompliance," and the term "noncompliance" is defined in the regulation.

The inspector also found that the reporting requirements of both the earlier and the current versions of 10 CFR 21.21 were improperly addressed in Procedure 704011. Paragraph 3.1 stated the responsibility of an FCI employee to notify the QA manager if the employee learned that a basic component supplied by FCI "contains a defect or fails to comply with 10CFR21." This requirement confused deviations from technical specifications, which an employee may discover, with defects, which are deviations that could create substantial safety hazards. The procedure effectively established such a high threshold for employee reporting that it precluded any reporting by employees. Conversely, employees could identify deviations and are required to report them by 10 CFR Part 21, but not by the FCI procedure.

Paragraph 3.1 of QA Procedure 704011 also improperly restricted the ongoing notification and evaluation requirements of 10 CFR 21.21(a)(3)(i), which addresses failures to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the Commission relating to a substantial safety hazard, and not just with Part 21 as stated in the FCI procedure.

Paragraph 4.1.c of QA Procedure 704011 required employees to submit a written report to the QA manager to define the nature of the defect or failure to comply, and the safety hazard which was or could be created. The QA manager agreed that FCI employees would seldom if ever be able to make such determinations. This requirement indicated further lack of understanding of NRC requirements relating to defects and safety hazards.

Paragraph 4.2 of QA Procedure 704011 stated that the QA manager would determine if the defect or failure to comply is a "reportable incident." In fact, 10 CFR 21.21(c) requires notifying the NRC in case of a failure to comply or defect affecting a basic component. The procedure gave no further definition of what constitutes a so-called reportable incident.

Finally, QA Procedure 704011 did not reflect the provisions required by 10 CFR 21.21 to ensure that all affected licensees or purchasers are informed of deviations that FCI determines it cannot evaluate. The procedure did not address this situation, which the FCI QA manager conceded is the most likely case for a deviation.

Based on these deficiencies, the NRC inspectors concluded that QA Procedure 704011 did not ensure that deviations will be evaluated, that defects or failures to comply will be reported to the responsible officer, or that all affected purchasers or licensees will be informed of deviations when FCI cannot perform the evaluation. These deficiencies, together with the failure to incorporate new requirements of Part 21 and to post the current revision of Part 21--the issue effective October 29, 1991, was posted--constitute Violation 99901264/93-01-02.

4 PERSONNEL CONTACTED

<u>FCI</u>:

+ * M.M. McQueen, President
+ R.A. Deane, Treasurer
+ * R.E. Ogle, Director of Administration
+ * A.D. Johnson, Director of Engineering
+ * S.R. Mitchell, QA Manager
+ * M. Bess, Test Engineering Manager
+ R. Thorpe, Contracts Manager
+ * W. Franz, Consulting Engineer

+ Attended the entrance meeting on April 13, 1993
* Attended the exit meeting on April 15, 1993

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

JUN 28 1993

Docket No. 99901266

Debra A. Sullivan
Q.C. Manager
Huron Industries, Incorporated
2301 16th Street
P.O. Box 610104
Port Huron, Michigan 48060

Dear Ms. Sullivan:

SUBJECT: TRANSMITTAL OF NRC INSPECTION REPORT (REPORT NO. 99901266/93-01)

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection conducted by Messrs. K.R. Naidu and J.J. Petrosino of this office on June 2-3, 1993, of the Huron Industries, Incorporated (Huron) facilities in Port Huron, Michigan and the discussions of our conclusions with you at the end of the inspection.

The specific areas examined during the inspection and our findings are discussed in the enclosed report. The inspection team noted that you have established and implemented a quality assurance (QA) program to comply with the requirements of Military Specification Instruction (MIL-I) 45208A, "Inspection Systems Requirements," to control the manufacture and supply of pipe-thread lubricants and sealants for use at commercial nuclear reactor power plants. The team observed that the products you manufacture and supply are commercial-grade items for which the provisions of Part 21 of Title 10 of the Code of Federal Regulations, are not applicable. Within these areas, the inspection consisted of an examination of licensee purchase orders, associated records, interviews with personnel and discussions regarding your activities.

Based on the results of this inspection, the team determined that you manufacture and supply commercial-grade sealants and lubricants and such activities are not required to be controlled in accordance with NRC requirements. Therefore, the NRC inspection team did not evaluate the implementation of your QA program that was adopted to meet the requirements of MIL-I-45208A. The specific areas reviewed and discussed are identified in the enclosure to this letter.

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

Sincerely,

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

Enclosure:
Inspection Report 99901266/93-01

ENCLOSURE

ORGANIZATION:	Huron Industries, Incorporated
REPORT NO.:	99901266/93-01
CORRESPONDENCE ADDRESS:	2301 16th Street P.O. Box 610104 Port Huron, Michigan 48060
ORGANIZATIONAL CONTACT:	Ms. Debra A. Sullivan, Q.C. Manager (313) 984-4213
NUCLEAR INDUSTRY ACTIVITY:	Manufactures and supplies commercial-grade pipe-thread lubricant and sealant
INSPECTION CONDUCTED:	June 2-3, 1993
TEAM LEADER:	K.R. Naidu, Team Leader Reactive Inspection Section 2 (RIS-2) Vendor Inspection Branch (VIB) Date: 6/21/93
OTHER INSPECTOR:	Joseph J. Petrosino, RIS-2: VIB
APPROVAL:	Gregory C. Cwalina, Chief RIS-2, VIB Division of Reactor Inspection and Licensee Performance Date: 6/21/93
INSPECTION BASES:	10 CFR Part 21 and 10 CFR Part 50, Appendix B
INSPECTION SCOPE:	To review licensee purchase orders, related records, and associated activities of Huron Industries, Inc.
PLANT SITE APPLICABILITY:	Numerous

1 INSPECTION SUMMARY

Huron Industries, Inc. (Huron) supplies only commercial grade items to nuclear power plants for which compliance to 10 CFR Part 21 and 10 CFR 50, Appendix B is not required.

2 STATUS OF PREVIOUS INSPECTION FINDINGS.

This is the first NRC inspection at this facility.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

On June 2, 1993, during the entrance meeting, the NRC inspectors discussed the scope of the inspection and the areas to be reviewed with the Quality Control (QC) Manager. On June 3, 1993, during the exit meeting, the inspectors discussed their conclusions with the QC Manager.

3.2 Background

Huron supplies a line of commercial-grade pipe-thread lubricants and sealants to military, nuclear steam supply system suppliers and nuclear power plants that includes the following:

- Pipe-Lubricant Neolube No. 1 (DAG-156). Acheson Colloidals (Acheson) manufactures this compound and supplies it to Huron in 55-gallon drums. Huron repackages it in 2 and 8 ounce plastic containers. Before dispatching the drum(s), Acheson collects samples of the compound and sends them to an independent testing laboratory to verify that they meet the requirements of Military Specification (MIL)-L-24131B, "Lubricant, Colloidal Graphite In Isopropanol." Huron does not recommend the use of Neolube 1 on fittings in areas where the operating temperatures are greater than 400° Fahrenheit.

- Dryfilm Pipe-Thread Lubricant Neolube No. 2 (DAG-154). Acheson manufactures and supplies a concentrated mixture of Neolube No. 2 in 55-gallon drums. Huron dilutes this with certified alcohol meeting Federal Specification TT-I-735A using a proprietary formula and repacks the mixture in 2, 16, 32 and 128 ounce containers. Neolube No. 2 is extensively used as an anti-seize lubricant in a variety of applications in commercial nuclear power plants and military applications. On request, Huron sends samples of this product to an independent testing laboratory to verify conformance to MIL-L-24131B.

- Neolube Pipe-Thread Sealant No. 100. Loctite Corporation (Loctite) manufactures and packages one of their products for Huron in plastic tubes as Neolube 100. This product is a light paste sealant which seals threaded pipe, plugs and fittings. In the absence of oxygen, and at above room temperature it cures efficiently. The manufacturer guarantees each batch of this product for a shelf life of one year when stored at the specified temperature. When a customer requests an extension, the manufacturer retests the sealant and extends the shelf life if the specimen is found to have retained its original properties. At the bottom of the tube, the manufacturer imprints a code number from which the year and month of manufacture, the manufacturing plant and the batch number can be deciphered. For example, 3GN468 denotes the products was made in 1993, G the month during which it was manufactured (in July), N the plant where it was manufactured and 468 denotes the batch.

- Neolube Pipe-Thread Lubricant No. 650. Union Carbide manufactures this product in 55-gallon drums and ships it to a packaging company where it is packaged for Huron in small containers and is sold as a high temperature anti-seize paste. It is composed of graphite and petroleum based carrier. Huron recommends Neolube No. 650 for use as a lubricant in close fitting threaded joints of two inches and smaller diameter pipe size in service applications where high temperature and high pressure or both are experienced.

- Neolube Pipe-Thread Sealant No. 1260. Union Carbide also manufactures this product in 55-gallon drums and ships it to a packaging company where it is packaged for Huron in small containers and is sold as a high temperature anti-seize paste. This product is a pipe-thread sealant and is composed of graphite and petroleum based carrier. Huron recommends Neolube No. 1260 for use in close fitting threaded joints of two inches and smaller, and in service applications where high temperatures and high pressures or both are experienced.

3.3 Review of Huron's Quality Assurance Program

Huron's QA Manual, Revision 3 of October 27, 1988, has been developed to comply with MIL-I-45208A, "Inspection System Requirements." The inspection team reviewed the QA program requirements at Huron that were delineated in its QA Manual. Some of the Sections that were reviewed included the following:

- Design control appears to be based on the manufacturer's specifications which in turn have to meet Military Standards.

- The manufacturer collects and sends samples of the material manufactured to an independent testing laboratory for confirmation that the product meets the relevant Military Specifications (MIL-Specs). After the test results are determined acceptable, the material is shipped to Huron. In addition to these tests, if a purchase order (PO) specifies certification by an independent testing laboratory, Huron collects and sends samples to a test laboratory designated by the customer. The team determined that Huron utilizes four different laboratories to independently test samples of lubricants and sealants routinely, and when specifically requested.

- Receipt inspections. Huron indicated that it inspects incoming material to ascertain if the materials meet the specifications in the PO as documented in the certificate of conformance. Huron stated that the results of the inspection are documented in receipt inspection reports which identify the item inspected, the number of the PO, batch number, lot size, sample size, quantity, condition, certification and leakage, if present. Huron personnel also apply stickers to the containers of the accepted material for identification and traceability.

- The team reviewed the Deficiency Reports in which Huron documented conditions adverse to quality. The inspectors observed that the actions taken to correct the adverse conditions were adequate to prevent recurrence.

3.4 Review of Procurement Document Control

The inspectors selectively reviewed procurement documents which included licensees' POs to Huron, Huron's internal invoices to specify the respective quality requirements and the required certificates of compliance. The inspectors determined that Huron has developed a computer program with controls to ensure compliance with the quality requirements in the POs. For example, an interlock in the accounting computer program prevents further processing of a purchase order which specified special test and verification requirements. In such cases, the program reminds the operator of the special requirements. The operator then enters the customer's special requirements in Huron's shop order.

3.5 Control of Measuring Equipment

During the visit to Huron's facility in Jeddo, Michigan, where it packages sealants and lubricants, the team observed that Huron uses an Ohoius Triple Beam Balance to weigh the bottles it fills with sealant or lubricant. Huron has established internal controls to verify the accuracy of the balance and stated that it has the unit calibrated annually by an outside calibration agency.

3.6 Review of Corrective Action

The inspectors reviewed six Deficiency Reports that had been generated between 1989 and 1992 to document conditions adverse to quality and noted that Huron had taken actions to correct and disposition them.

3.7 Review of Audits Performed

The inspectors reviewed the audits performed by Wolf Creek Nuclear Operating Corporation, Carolina Power & Light Company, Gulf States Utilities Company, Niagara Mohawk Power Corporation and Entergy Operations, Inc. to qualify Huron as an acceptable commercial-grade item supplier for their respective nuclear power plants. The audits indicate that Huron's quality assurance program was evaluated and determined to be acceptable. Therefore it was qualified as a commercial-grade supplier for its lubricants and sealants. Other utilities qualified Huron based on written answers to surveys.

3.8 Stated Shelf Life

The NRC inspectors conducted discussions with the QC Manager regarding product shelf life and reviewed the Huron technical data sheets for each of their products and the associated certificates of conformance (CoCs). During this discussion and review, the team observed that the stated shelf life in the CoC was different from the manufacturer's recommendations in the technical data sheets for three Huron products, specifically: Neolube Thread Sealant No. 100; Neolube Pipe Thread Lubricant No. 650; and Neolube Pipe Thread Sealant No. 1260.

The technical data sheet for Neolube Thread Sealant No. 100 stated that the shelf life at 72 degrees Fahrenheit is 1-year minimum. However, Huron's CoC stated that the shelf life was 1-year, and that recertification after expiration date is available upon request.

The technical data sheet for Neolube Pipe Thread Sealant No. 650 stated that the shelf life was a minimum of two years from date of first use. However, Huron's CoC stated that the shelf life was two years, the material may be satisfactory if soft in tube, and that recertification after the expiration date is available upon request.

The technical data sheet for Neolube Pipe Thread Sealant No. 1260 stated that the shelf life was a minimum of two years from date of first use. However, Huron's CoC stated that the shelf life was two years, that the material may be satisfactory if soft in tube, and that the shelf life may be extended after the expiration date by submitting a sample tube for retesting.

The team was mainly concerned with the shelf life that was stated on the technical data sheets because an NRC licensee may file the product CoCs in their QA record filing area and transmit the technical data sheets for use by installing organization personnel without the benefit of the CoCs. Consequently, the difference in the shelf life recommendations that Huron stated in its documents may not be recognized and the installing organization may rely on the more relaxed, less conservative, shelf life that was stated on the Huron technical data sheets.

The team discussed this difference in the characterization of the shelf life with the QC Manager and its scenario of possible misuse by licensee staff as a result of the differences in stated shelf life. As a result of the discussion Huron committed to the NRC team to research the manufacturer's minimum and maximum shelf life recommendations, correct the technical data sheets as required and to send any corrected technical data sheets to all of its affected customers. The team concluded that Huron's commitment to resolve this matter satisfactorily resolves the NRC inspection team concern.

4 PERSONNEL CONTACTED

D. A. Sullivan, QC Manager
L. Meddaugh, Quality Technician
K. Sexton, Administrative Assistant

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

Docket No. 99900100

JUN 28 1993

Dr. Ivan E. Wilkinson, P. E.
Vice President, Engineering
Limitorque Corporation
5114 Woodall Road
Lynchburg, Virginia 24506

Dear Dr. Wilkinson:

SUBJECT: NRC INSPECTION REPORT NO. 99900100/93-01

This refers to the inspection conducted by an inspection team, led by Mr. Jeffrey Jacobson of this office, on May 10-14, 1993. The inspection was conducted to review activities associated with Limitorque's supply of valve actuators to the nuclear industry. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the enclosed inspection report (Enclosure 2). The inspection was conducted to evaluate Limitorque's: manufacturing and engineering programs for safety-related items including design control, configuration control, and control of sub-vendors; corrective actions associated with 10 CFR Part 21 reports; commercial grade dedication activities; and corrective actions regarding unresolved items and concerns from previous inspection reports.

Based on the results of this inspection, two violations were identified. The first violation concerns Limitorque's failure to complete its evaluation and report the effects that relaxation of the actuator spring pack may have on the operability of motor operated valves.

The second violation concerns Limitorque's failure to implement procedures as required by 10 CFR 21.21, to address certain substantive revisions of 10 CFR Part 21 that became effective October 29, 1991. This failure to update internal procedures in a timely manner perpetuated Limitorque's misinterpretation of 10 CFR Part 21.21 reporting requirements and resulted in the failure to submit several completed evaluations or interim reports to the NRC within 60 days. Corrective actions and preventive measures that were presented during the inspection, as documented in Enclosure 2, NRC Inspection Report 99900100/93-01, were satisfactory to close this violation, therefore, no response to this violation is required.

The results of the inspection indicate you have implemented significant improvements in your procedures for controlling commercial grade dedication, sampling, and material testing. However, your actions with respect to evaluating and resolving technical issues arising from valve operating experience and in reporting the results of these evaluations were found to be inadequate. Many of these issues have resulted from inadequacies in the original design of the motor operators and valves. Your timeliness in evaluating technical issues and in reporting the results of your evaluations is necessary to ensure reliable performance of safety-related motor operated valves in nuclear power plants, and required by NRC regulations. We intend to closely monitor your future actions in this area.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. 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.

The responses requested by this letter and the enclosed Notice of Violation 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.

Sincerely,

Leif J. Norrholm

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

Enclosures:
1. Notice of Violation
2. Inspection Report 99900100/93-01

ENCLOSURE 1

NOTICE OF VIOLATION

Limitorque Corporation
Lynchburg, Virginia

Docket No. 99900100/93-01

During an NRC inspection conducted May 10 through May 14, 1993, 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 (1993), the violations are listed below:

Section 21.21, "Notification of Failure to Comply or Existence of a Defect and Its Evaluation," of 10 CFR requires, in part, that each corporation subject to the regulations adopt appropriate procedures for either evaluating and reporting deviations and failures to comply, or informing the licensee or purchaser of the deviation or failure to comply.

A. Contrary to the above requirements, Limitorque Corporation (Limitorque) failed to complete its evaluation and hence did not report a condition associated with relaxation of motor actuator spring packs. Limitorque had closed this issue out on June 1, 1992, without a documented basis. Test data which Limitorque had collected prior to June 1, 1992, indicated that spring pack relaxation can occur and that the magnitude of the relaxation could be significant. Limitorque failed to evaluate the effect the relaxation could have on actuator torque output capability and failed to notify the NRC or its users of this potential safety issue.(99900100/93-01-01).

This is a Severity Level IV Violation.

B. Contrary to the above requirements, Limitorque had not revised its procedure, required by 10 CFR 21.21, to address certain substantive revisions to 10 CFR Part 21 that became effective on October 29, 1991. This procedure inadequacy contributed to Limitorque's misinterpretation of 10 CFR Part 21.21 reporting requirements and resulted in the failure to complete several evaluations within 60 days or to submit interim reports to the NRC (99900100/93-01-02).

This is a Severity Level IV violation.

Corrective actions and preventive measures that were presented during the inspection, as documented in Enclosure 2, NRC Inspection Report 99900100/93-01, were satisfactory to close this violation.

Pursuant to the provisions of 10 CFR 2.201, Limitorque 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

Licensee Performance, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for the first violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieve. Where good cause is shown, consideration will be given to extending the response time. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Date at Rockville, Maryland
this 28th day of June, 1993.

ORGANIZATION:	Limitorque Corporation Lynchburg, Virginia
REPORT NO.:	99900100/93-01
CORRESPONDENCE ADDRESS:	Dr. Ivan E. Wilkinson, P. E. Vice President, Engineering Limitorque Corporation 5114 Woodall Road Lynchburg, Virginia 24506
ORGANIZATIONAL CONTACT:	Rory D. Segen, Quality Assurance Manager 804/528-4400
NUCLEAR INDUSTRY ACTIVITY:	Motorized valve operators, their replacement parts, and services
INSPECTION CONDUCTED:	May 10-14, 1993

J.B. Jacobson, Team Leader — 5/27/93 Date
Team Inspection Section B
Special Inspection Branch

OTHER INSPECTORS:	T. Scarbrough, Mechanical Engineering Branch, Office of Nuclear Reactor Regulation (NRR) L. Campbell, Vendor Inspection Branch, NRR D. Brewer, Vendor Inspection Branch, NRR

Uldis Potapovs, Chief — 6-9-93 Date
Reactive Inspection Section No. 1
Vendor Inspection Branch

INSPECTION BASES:	10 CFR Part 21 and Part 50, Appendix B
INSPECTION SCOPE:	Evaluate (1) Limitorque's manufacturing and engineering programs for safety-related items including design control, configuration control, and control of sub-vendors, (2) corrective actions associated with 10CFR Part 21 reports, (3) Limitorque's commercial grade dedication activities, (4) corrective actions regarding unresolved items and concerns from previous inspection reports.
PLANT SITE APPLICABILITY:	Numerous

1 INSPECTION SUMMARY

1.1 Violations

1.1.1 Contrary to the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 21.21, Limitorque Corporation (Limitorque) failed to complete its evaluation of the effects that the relaxation of the actuator spring may have on the operability of motor operated valves (MOVs) (99900100/93-01-01, see Section 3.2 of this report).

1.1.2 Contrary to the requirements of Section 21.21, Limitorque had not revised its procedures to address certain substantive revisions to 10 CFR Part 21 that became effective on October 29, 1991 (99900100/93-01-02, see Section 3.2 of this report).

1.2 Unresolved Items

1.2.1 Unresolved Item 99900100/93-01-03 (Open)

On July 30, 1992, Washington Public Power Supply System (WPPSS) provided preliminary notification of finding Society of Automotive Engineers (SAE) Grade 1 or 2 screws in an application where SAE Grade 5 screws were specified. The screws were used to attach the housing cap to the motor actuator housing on Limitorque's Model SMB-000 motor actuator. The motor actuators were installed in a safety-related application at Washington Nuclear Plant-Unit 2. Formal notification, "WNP-2, OPERATING LICENSE NPF-21, 10 CFR PART 21 REPORT, MOTOR-OPERATOR CAP SCREW," was dated August 13, 1992.

WPPSS determined that SAE Grade 1 or 2 screws, at minimum published yield strength, could produce a service failure if the motor actuator was set at greater than 94% of the rated thrust.

Limitorque was not aware of this situation until two days before the inspection and had not determined the cause, extent of the condition, or preventive action. The NRC will follow the progress of Limitorque's evaluation of this situation as Unresolved Item 99900100/93-01-03.

1.2.2 Unresolved Item 99900100/93-01-04 (Open)

The NRC will follow the progress of the formal update of Limitorque's inspection plans to address its current practice for sampling (see Section 3.4.2 of this report).

1.2.3 Unresolved Item 99900100/93-01-05 (Open)

The NRC will follow the progress of Limitorque's commitment to alert licensees to errors in the specified values for run and stall efficiencies for certain SMB-3 actuators (see Section 3.8.1 of this report).

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Nonconformance 99900100/91-01-02 (Closed)

Limitorque's practice for accepting bar stock, tubing, and plate was reviewed during the inspection and was identified as a significant improvement over its acceptance practice reviewed during the 1991 NRC inspection. Limitorque's past practice of testing one piece of material from each supplier once a year has changed to testing one piece of certain types of material received from a supplier quarterly or monthly and in certain instances testing each heat/lot of material stock used for critical parts such as motor pinion keys. Limitorque is developing inspection plans to formally implement this practice. The NRC will follow the progress of the formal update of Limitorque's inspection plans to address its current practice as Unresolved Item 99900100/93-01-04 (see Section 3.4.2 of this report).

2.2 Nonconformance 99900100/91-01-04 (Closed)

Limitorque's practice of dedicating motors supplied by Peerless-Winsmith was reviewed and found to adequately identify dedication activities performed by each vendor. Although this item was closed during the last inspection, the basis for activities performed by each dedication party was not clearly defined. The inspectors reviewed the basis for dedication activities performed and found them satisfactory to close this nonconformance (see Section 3.4.3 of this report).

2.3 Unresolved Item 99900100/91-01-05 (Closed)

The motor actuator characterization software system is no longer being used to obtain output torque data for actuators. Therefore, this unresolved item is closed.

2.4 Unresolved Item 999000100/91-01-06 (Closed)

Limitorque committed to notify certain NRC licensees of a possible defect concerning the required tension of Reliance motor end bolts. This notification was included in Limitorque Corporation Maintenance Update 92-2. Therefore, this unresolved item is closed.

2.5 Unresolved Item 99900100/91-01-07 (Open)

Limitorque committed to notify certain NRC licensees of a possible defect concerning improper machining of actuator limit stop housings for HBC-1 actuators. The notification was to have been made by way of Limitorque's maintenance bulletin. The inspectors found that Limitorque had failed to perform this action. This item remains open as Unresolved Item 99900100/91-01-07 until the maintenance bulletin is issued.

2.6 Unresolved Item 99900100/91-01-03 (Closed)

Limitorque failed to complete its evaluation of the effects that the relaxation of the actuator spring pack may have on the operability of MOVs and incorrectly identified the evaluation as completed. This unresolved item is now being tracked as Violation 99900100/93-01-01 (see Section 3.2 of this report).

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on May 10, 1993, the NRC inspectors discussed the scope of the inspection and established interfaces with Limitorque management. During the exit meeting on May 14, 1993, the NRC inspectors discussed their findings and concerns with Limitorque's management and staff.

3.2 Implementation of 10 CFR Part 21 Procedures

The NRC inspectors determined that Limitorque has maintained the required 10 CFR Part 21 posting and a procedure for implementing 10 CFR Part 21 requirements, Quality Assurance Procedure (QAP) 13.2, "Reporting of Defects for Safety Related Equipment", Revision 8, dated October 21, 1992. Limitorque informed the NRC inspectors that they were aware of some of the changes to 10 CFR Part 21 requirements such as the requirement to file an interim report if an evaluation has not been completed within 60 days, and had attempted to incorporate these changes, as they understood them, into QAP 13.2. Limitorque, however, failed to revise QAP 13.2 to address all aspects of the revision to 10 CFR Part 21, such as defining what is considered objective evidence which demonstrates that the NRC has been adequately informed of a defect or failure to comply. Limitorque informed the NRC inspectors that they believed neither an evaluation nor an interim report was necessary if the NRC had been made aware of the issue during discussions at nuclear industry meetings, during undocumented casual conversations with the NRC, or by the issuance of a Limitorque technical bulletin update. The inspectors informed Limitorque that their interpretation of 10 CFR Part 21 was incorrect and that 10 CFR Part 21, Section 21.21, Paragraph (c)(2) requires written notification to the NRC. This notification should include all details as delineated in 10 CFR Part 21, Section 21.21, Paragraph (c)(4).

Limitorque's misinterpretation of 10 CFR Part 21 reporting requirements apparently resulted in the failure to complete several evaluations within 60 days or to submit interim reports to the NRC. The NRC inspectors reviewed Limitorque's 10 CFR Part 21 Log and determined that Limitorque had failed either to complete its evaluations within 60 days or to submit interim reports to the NRC for the following items (Violation 99900100/93-01-02). Limitorque's evaluation as to reportability is in parentheses following each example:

1. Limitorque Log No. 08, Spring Pack Relaxation (Not Reportable)

2. Limitorque Log No. 20, AC Motor Ambient Temperature Effects (Reportable)

3. Limitorque Log No. 28, Spring Pack Curve Data (Not Reportable)

4. Limitorque Log No. 37, Worm Gear Failures (Reportable)

5. Limitorque Log No. 38, HBC Over-Rated When Mounted to SMB (Not Reportable)

6. Limitorque Log No. 39, Breather/Drain Plug (Not Reportable)

7. Limitorque Log No. 40, Declutch Lever Seismic Test Reassurance (Not Reportable)

8. Limitorque Log No. 41, SBD Bolt Thread Engagement (Reportable).

During the conduct of the inspection, Limitorque processed a written notification to the NRC of its failure to comply with the requirements of 10 CFR Part 21 as discussed above, and also issued QAP 13.2, Revision 9, dated May 14, 1993, that incorporated current 10 CFR Part 21 reporting requirements. The revision to QAP 13.2 identifies acceptable notification procedures and incorporates other current requirements of 10 CFR Part 21 that were missing or incorrectly stated in Revision 8 to QAP 13.2. Additionally, the NRC inspectors verified that all outstanding Part 21 evaluations have been completed except for the issue concerning spring pack relaxation which is detailed below. These corrective actions and preventive measures that were presented during the inspection were satisfactory.

The inspectors determined that the condition associated with actuator spring pack relaxation (Log No. 08) had been incorrectly identified as closed and that the evaluation report for this condition had not been completed. Spring pack relaxation can occur on stressed spring packs, such as those on normally closed motor operated valves. The result is that for a given torque switch setting, less output torque will be delivered by the motor actuator. Limitorque's Part 21 Log indicated that the evaluation for this condition was started on August 16, 1988. The Part 21 file for this condition contained test data which indicated that for the samples tested, spring pack relaxation of between 3 and 10 percent could occur over a two year period. Further review of the data in the evaluation file and discussions with Limitorque revealed that Limitorque had failed to complete its evaluation of the effect that the relaxation of the actuator spring pack could have on actuator torque output. The Part 21 evaluation form had been dated as complete on June 1, 1992, but no evaluation for reportability had been documented, nor was the evaluation form signed. Based on a review of the technical data which existed in the evaluation file, the NRC inspectors considered that sufficient data existed in June 1992 to determine that this was a reportable condition and hence constituted a violation of 10CFR Part 21 reporting requirements (Violation 99900100/93-01-01). Limitorque committed to complete its evaluation of this issue and make suitable notification under 10 CFR Part 21 by June 15, 1993.

3.3 Limitorque's Actions Relative to Licensee Event Reports (LERs), Part 21 Reports, and Other Reports

3.3.1 LER 275/91-021-00

LER 275/91-021-00, "Failure of Motor Pinion Keys in Limitorque SMB-3-80 Motor Operators Due to Inadequate Design of Material," dated August 28, 1992, was initiated by Pacific Gas and Electric (PG&E) Company as a result of the motor pinion key shearing in a Limitorque SMB-3-80 motor operator (3380 RPM). The MOV was installed on the residual heat removal heat exchanger outlet on Diablo Canyon Unit 1. The date of the event was September 16, 1991. On April 29, 1992, PG&E's chemical analysis of the key material identified it as a low carbon, resulfurized, and leaded steel, such as ASTM A-29, Grade 12L13.

The Limitorque-specified key materials have been AISI 1018 for actuator Models SMB-000, SMB-00, SMB-0, SMB-1, & SMB-2, and AISI 4140 for Models SMB-3, SMB-4, & SMB-5. Over a period of years, sheared motor pinion keys have been reported in various SMB actuators and the cause has generally been identified with the use of low carbon, resulfurized, and leaded steel or a material other than the one specified for the application.

During an NRC inspection at Limitorque in May 1988, inspectors found, at the time of manufacture of an unspecified number of motor actuators, keys were purchased from a commercial source without certificates of conformance. In addition, no testing was performed by Limitorque to verify material requirements. On August 30, 1988, Limitorque's inspection procedures were revised to assure receipt of the correct key materials and shipment of keys made of the correct materials. Existing stocks of keys and key materials were scrapped. Limitorque's current practice for dedicating motor pinion keys is discussed in Section 3.4 of this report.

The NRC has issued three Information Notices regarding sheared motor pinion keys since such failures were first noted: IN 81-08, "Repetitive Failures of Limitorque Operator SMB-4 Motor-to-Shaft Key;" IN 88-84, "Defective Motor Shaft Keys in Limitorque Motor Actuators;" and IN 90-37, "Sheared Pinion Gear-to-Shaft Keys in Limitorque Motor Actuators."

Limitorque Maintenance Update 92-2, Section 1., Motor Pinion Keys, states, "The material for motor pinion keys for SMB-000 through SMB-2 actuators has recently been changed from an American Iron and Steel Institute (AISI) 1018 steel to AISI 4140 steel." Requests for replacement keys will now be filled with AISI 4140 regardless of operator model. AISI 4140 steel keys and the current dedication methodology used by Limitorque are expected to minimize sheared motor pinion keys. This issue is closed.

3.3.2 Limitorque's Part 21 Report, Incorrect Screw Length

On October 5, 1992, Limitorque reported insufficient thread engagement for screws securing the cover on the SMB to SBD-1 actuator conversion. The design requires the joint to withstand a stall thrust of 2.5 times the standard thrust rating of 45,000 lbs. Actual thread engagement produced a joint theoretically capable of withstanding 2.24 times the standard thrust rating.

In a letter dated October 5, 1992, Limitorque notified its customers that if the subject actuator in a given application has the capability of developing more than 2.24 times the standard thrust, corrective action would be necessary. Corrective action would require the replacement of existing 6.5" screws with 7" screws and the placement of a specified shim under the head of the 7" screws to prevent bottoming.

Limitorque drawing No. 60-021-0050-3, "Disk Spring Housing," for the SMB to SBD-1 conversion, was originally issued December 8, 1977. The condition described above has existed since then. Preventive action was documented in Revision D to the drawing dated July 29, 1992, when the depth of the counter-bore was changed to provide for the proper thread engagement. This issue is closed.

3.3.3 Limitorque's Part 21 Report, Seismic Effect on Declutch Mechanism

On December 7, 1992, Limitorque notified the NRC of a potential defect in the declutch mechanism for its SMB/SB-00 and SMB/SB/SBD-00 actuators. Limitorque reported that, when the actuator is vibrated in the vertical axis with sufficient amplitude near the natural frequency of the declutch system, oscillations of the declutch system can cause the motor to become disengaged. This would result in the actuator stopping its movement of the valve disc during the seismic event. When the seismic motion ceased, the actuator would return to normal operation. Limitorque has designed a new declutch lever to eliminate the potential problem for future orders. In its December 7, 1992, notification, Limitorque recommended that each licensee evaluate the potential for actuator malfunction that would lengthen the valve stroke time during seismic events at the applicable frequencies. Limitorque indicated that the declutch shaft might not oscillate if the actuator was operating under sufficient load. Limitorque stated that licensees could purchase new declutch levers to correct the potential problem. Limitorque recommended that, in the interim, declutch levers be secured to prevent potential movement. The December 7, 1992, notification contains an attachment with a list of the recipients. The inspectors considered the December 7, 1992, notification sufficient to alert licensees to this potential problem and to provide adequate corrective action.

3.3.4 Texas Utilities Electric Company's Part 21 Report, Motor Deficiency

In letters on December 16, 1991, and May 15, 1992, Texas Utilities Electric Company (TU) discussed a deficiency involving the potential failure of Limitorque 80 ft-lb motors to meet rated capacity during maximum expected differential pressure valve operation. During an NRC inspection at Comanche Peak in August and September 1992 (NRC Inspection Report 50-445 and

446/92-34), the inspectors found that the licensee could not determine the exact cause of the motor failures, but that the licensee believed it was related to the testing methodology in that manual control of the test loading and data recording phase extended the length of time that the motor was exposed to high amperages. The inspectors considered that the licensee had taken appropriate corrective action. The staff considers this issue to be plant specific.

3.3.5 Limitorque Technical Update 92-1

In Technical Update 92-1, Limitorque states that the results of a study by Kalsi Engineering allowed the SMB-000 to 1 actuators to undergo 140% of their thrust rating for 2000 cycles. Limitorque also sent a letter to certain licensees with access to the details of the proprietary Kalsi report allowing the actuators to undergo 162% of their thrust rating for 2000 cycles. In the technical update, Limitorque indicated that the actuator bolts must be torqued to a specific value to allow the increased thrust levels. During the inspection, Limitorque personnel stated that the bolt torque provision remained applicable. Limitorque stated that Kalsi is evaluating the provision on bolt torque and that it will consider modifying the technical update when Kalsi's review is complete. Limitorque stated that it was not currently planning to modify the actuator thrust ratings because of the desire to retain margin in the thrust capability of the actuators and to provide a foundation for exceeding 2000 cycles of actuator operation. Limitorque stated that new actuators are sized with the original thrust ratings. Limitorque stated that it did not object to establishing a limited number of cycles beyond the Kalsi allowable thrust.

3.3.6 Limitorque Technical Update 92-2

On October 9, 1992, Limitorque released Technical Update 92-2, "Recommended Spring Pack Replacement Procedures for Limitorque SMB Actuators," for the use of its customers. The staff had previously reviewed the technical update as Part 21 Log No. 92-205. The staff closed its review of the technical update as not a true Part 21 issue (NO21) because the update provided guidance for licensees in ordering new spring packs when determined to be needed. During the inspection at Limitorque, the staff inspectors confirmed that Technical Update 92-2 was not reportable under 10 CFR Part 21.

3.3.7 Limitorque Maintenance Update 92-1

The inspectors discussed Maintenance Update 92-1 with Limitorque personnel. In a section with the title of "Allowable Overloads of Limitorque SMB Actuator," Limitorque recommended inspection of the actuator if it experienced two and one-half times the thrust rating (or two times the torque rating) or if the "overload" occurred more than once. During the inspection, Limitorque stated that it defined multiple "overload" as greater than 140% (or 162% where the Kalsi report is being used directly) of the thrust rating of SMB-000 to 1 actuators, or greater than 120% of the thrust rating of other sized actuators. Limitorque also stated that "overload" would include 120% of the torque rating of its actuators. Later in that section, Limitorque recommended visual inspection of the gearing and worm if "excessive torque" was applied to the actuator. During the inspection, Limitorque indicated that "excessive torque" was 120% of the torque rating of the actuator. The inspectors did not consider the issues addressed in the maintenance update to be reportable under 10 CFR Part 21.

3.3.8 Limitorque Maintenance Update 92-2

The inspectors discussed Maintenance Update 92-2 with Limitorque personnel. Limitorque had evaluated the issues discussed in the maintenance update for reporting under 10 CFR Part 21. The inspectors agreed with the decision by Limitorque that the issues discussed in Maintenance Update 92-2 were not reportable under 10 CFR Part 21.

3.4 Commercial Grade Dedication

With the exception of certain motors and electrical wire, Limitorque commercially purchases products used for manufacturing actuators supplied to nuclear power plants. Limitorque's Quality Management System Manual (QMSM) Revision 2, dated January 29, 1993, requires that all product lines identified for use in safety-related applications be subject to additional inspections and tests. Critical characteristics for items to be used in safety-related applications are termed attributes and are identified and verified through inspection or test as specified in an applicable inspection plan or procedure. Satisfactory completion of these inspections and tests form the basis for the dedication of commercially procured products as basic components. At Limitorque, dedication occurs following final acceptance for an item designated for safety-related application at the time the item is certified (Certification of Compliance issuance) by Quality Control.

The NRC inspectors selected the following dedication activities for review to determine if Limitorque's commercial grade dedication activities were being effectively implemented and as a follow up to the 1991 NRC inspection report.

3.4.1 Motor Pinion Keys

Limitorque presently uses only AISI 4140 material to manufacturer safety-related motor pinion keys. Limitorque's Inspection Plan (IP) No. 023, "Bar Stock, Tubing Plate," Revision 6, dated October 2, 1991, is the current IP applicable for the receipt inspection of AISI 4140 bar stock used for the motor pinion keys (keys). IP No. 023 does not reflect Limitorque's current practice for accepting AISI 4140 bar stock. The NRC inspectors were informed that the current practice for accepting AISI 4140 bar stock used for keys will be incorporated in the near future into IP No. 023, and will require that a piece of material from each heat of AISI 4140 bar stock received be subjected to physical testing and chemical analysis. Once the AISI 4140 material heat is acceptable, it is placed in stock. The NRC inspectors reviewed recent documentation and confirmed that these tests and analyses were being performed and were acceptable, even though they had yet to be incorporated into the applicable procedures.

AISI 4140 material is withdrawn and sent to a machine shop where the motor pinion keys are machined in accordance with applicable Limitorque drawing requirements. IP No. 076, "Keys - Clutch, Motor Pinion, Intermediate," Revision 1, dated March 28, 1991, is the current IP applicable for the receipt inspection of the machined keys. IP No. 076 does not reflect Limitorque's current practice for accepting the machined keys. The NRC inspectors were informed that IP No. 015 will be issued in the near future and will require that each lot of keys machined be subjected to chemical analysis and hardness testing on a sample basis. Once the machined keys are acceptable they are placed in stock. The NRC inspectors reviewed recent documentation and confirmed that the chemical analysis and hardness tests were being performed and were acceptable.

When keys are withdrawn from stock and are to be supplied for safety-related applications, the keys are processed in accordance with Quality Control Procedure No. 10.5, "Inspection of Safety Related Parts and Orders," Revision 2, dated September 4, 1992. In addition to standard visual inspections, hardness measurements are performed on each key. The NRC inspectors reviewed recent documentation and confirmed that visual inspections and hardness tests had been performed on the keys with results acceptable.

Limitorque does not audit or survey the supplier of the stock material used for the keys. However, the tests and chemical analysis performed on each material heat and on the machined key lot, along with the additional inspections and hardness tests performed on keys designated for safety-related applications, provide reasonable assurance that the keys will perform their intended safety function.

3.4.2 Sampling Bar Stock, Tubing and Plate

The 1991 NRC inspection identified that Limitorque's practice for accepting bar stock, tubing and plate based on testing one piece of material from each supplier once a year as an unacceptable sampling frequency. Limitorque informed the NRC inspectors that, for certain critical actuator parts, each material type received from a supplier may be tested quarterly or monthly, and in certain instances, such as motor pinion keys, testing is performed on each heat or lot received. On June 10, 1991, Limitorque issued an internal directive concerning sampling for destructive testing. Also, in Limitorque's response to Nonconformance 99900100/91-01-02 on February 25, 1992, Limitorque indicated that it would develop and implement a sampling plan for use in selecting material sampling frequencies by May 29, 1992. Limitorque informed the NRC inspectors that draft revisions for applicable IPs were still being developed and would be finalized in the near future to formally implement its current sampling practices as discussed and being implemented during the inspection. The NRC will follow the progress of the formal update of Limitorque's IPs to address its current sampling practices as Unresolved Item 99900100/93-01-04.

3.4.3 Peerless-Winsmith Motors

As a result of the issuance of Nonconformance 99900100/91-01-04 identifying inadequacies during the dedication of Peerless-Winsmith electric motors, Limitorque implemented IP No. 111, "Peerless-Winsmith Critical Material Testing," Revision 0, dated July 15, 1991. During the inspection, the NRC inspectors determined that the inspection requirements of IP No. 111, Revision 0, had not been followed. However, the requirements of IP No. 111, Revision Level 1, dated March 25, 1993, were being followed. Revision 0 required Limitorque to perform testing of metallic and non-metallic motor parts, whereas Revision Level 1 only required Limitorque to inspect and test non-metallic motor parts. The NRC inspectors questioned the basis for Limitorque not inspecting or testing metallic parts.

Limitorque informed the NRC inspectors that Peerless-Winsmith had resolved past audit findings affecting the acceptance and implementation of its quality assurance program and had provided a detailed basis description for acceptance of metallic parts used in its motors. The NRC inspectors reviewed this basis document (transmittal dated May 7, 1993, from Peerless-Winsmith) and IP No. 111, Revision Level 1, and determined if properly implemented, acceptance methods used by Limitorque and Peerless-Winsmith should provide reasonable assurance that the motors will perform their safety functions.

3.5 Hardness Testing

To verify the adequacy of hardness testing procedures and equipment, two motor pinion keys were selected; part number 60-563-0268-1, for Model SMB-1, and part number 60-563-0154-1, for Models SMB-3 & 4. These keys were AISI 4140 with a hardness requirement in the range of HB 250-345 (Brinell).

The NRC inspectors requested and received a copy of Limitorque's hardness testing procedure, QCI-10.4, "General Instructions for Using Rockwell Hardness Testers," Revision 0, dated December 3, 1992. The procedure was determined to be in accordance with American Society for Testing and Materials (ASTM) E-18, the standard for Rockwell hardness testing.

The NRC inspectors witnessed a Limitorque QC inspector performing the hardness tests. The above procedure was followed in preparing to perform the Rockwell C hardness test. One hardness indentation was made on each of the selected motor pinion keys. Measured Rockwell C values were converted to Brinell values by the use of a standard conversion chart. The Brinell hardness values so obtained were within the specified range.

The NRC inspectors requested and received a copy of the calibration record for the hardness tester used to perform the above tests. Examination of the record showed the machine to be within the latest calibration period. Calibration and routine maintenance had been performed by a contractor from the Wilson Instruments Company, the manufacturer of the hardness testing equipment. The calibration record showed the tester was considered to be in good condition.

Limitorque's hardness testing procedures and equipment met the requirements of their procedures and ASTM E-18.

3.6 Design Control

The inspectors reviewed Limitorque's procedures for performing design changes as delineated in QAP 4, "Design Control Procedure," and QAP 5, "Engineering

Drawings and Standards-Issuance and Revision Procedure." The review consisted of both changes made by Limitorque which affect the Limitorque manufacturing process as well as changes made by sub-vendors to actuator components. QAP 5 requires that a review be performed for the effect of the proposed change on the equipment's nuclear equipment qualification. The proposed change is documented on form L147, "Change/General Release Notice." This form contains a section which addresses equipment qualification and contains blocks designated as "None," "Not Applicable," and "Qualification Affected." The inspectors noted that the blocks were not filled out uniformly for similar design changes and that QAP 5 was not clear as to the difference between the "Not Applicable" and the "None" blocks. In no case, however, did the inspectors find where a change that could have affected qualification receive an inadequate review. Limitorque agreed to provide additional guidance as necessary to more uniformly implement the intent of QAP 5.

Limitorque's policy in regard to sub-vendor design changes is to discourage any changes which could affect the qualification of equipment. As such, very few changes are made to parts contained in nuclear supplied actuators. The inspectors reviewed one change made concerning SMB-00 torque switches. This change was made as a result of failures experienced with the torque switch roll pins. The change involved upgrading the roll pins to a high alloy steel and increasing the roll pin diameter from 3/32 inch to 1/8 inch. The inspectors found the modification process with regard to this change to be well controlled.

3.7 <u>Bases for Motor Ratings and Motor Data</u>

The inspectors reviewed Limitorque's basis for motor data typically supplied by Limitorque to utilities. The data could include motor speed-torque curves, motor power factors, locked rotor torque values, and locked rotor current values. The inspectors determined that the motor curves currently supplied by Limitorque are derived from testing done on one motor of the specific design. Actual motor performance could vary significantly. For new motors, Limitorque requires the motor manufacturer to perform a locked rotor stall test on each motor. This test information is maintained by Limitorque but is not typically supplied with the motors. To try to quantify the uncertainty which exists with the motor curves, Limitorque reviewed test data taken from 81 recently supplied 10 foot-pound motors of an identical design. The test data indicated that the stall torque for the motors ranged from 10.9 to 12.3 foot-pounds with the majority of the motors exhibiting stall torques of between 11.2 to 11.6 foot-pounds.

Motor power factors for new motors can be calculated using the locked rotor stall test data and the following equation:

$$PowerFactor = \frac{Power}{Volts \times Amps \times \sqrt{3}}$$

For older motors Limitorque has to retrieve specific motor power factors from the motor manufacturer.

3.7.2 Motor Ambient Temperature Effects

The inspectors reviewed Limitorque's actions with respect to the potential effects that high ambient temperature can have on ac motor performance. Limitorque has derated its dc motors for high temperature service but had not evaluated potential ac motor torque reductions due to high temperature conditions. This issue had been discussed previously with Limitorque and has been identified as an open issue by the NRC in our review of utility programs in response to Generic Letter 89-10. Limitorque had identified the discovery date of this issue as July 27, 1989.

In October of 1992, Limitorque completed testing of five ac motors which indicated that ac motor stall torque output decreased significantly with increasing ambient temperatures. The torque reduction measured ranged from 14 to 25 percent at 336°F. Consequently, Limitorque contracted with the motor manufacturer, Reliance Motor Co., to calculate the expected torque reduction for all Limitorque-supplied ac motor designs. The Reliance data indicated the expected motor torque reductions to vary widely with the worst case being a decrease of torque of 30.8 percent at 180°C (353°F).

During the inspection, as a result of the above testing and calculations, Limitorque filed a Part 21 report dated May 13, 1993. The Part 21 report provides the expected motor torque reduction for the numerous Reliance ac motor designs. The inspectors found the report to be technically accurate and comprehensive. However, the inspectors expressed their concern regarding the long period of time taken by Limitorque to resolve this issue.

3.8 Actuator Sizing and Torque Switch Settings

3.8.1 Basis for Parameters in Actuator Sizing Equation

Limitorque has a standard practice for sizing motor actuators described in its "SEL" documents. Limitorque predicts the torque output of a motor actuator as follows:

Actuator torque = MT x Eff x AF x OAR x DVF

where
- MT = nominal motor starting torque
- Eff = gear efficiency of the actuator
- AF = application factor
- OAR = overall actuator ratio
- DVF = degraded voltage factor

Motor Torque

In its sizing of motor actuators, Limitorque uses the nominal motor starting torque to determine the appropriate size for the motor. In a letter to Cleveland Electric Company on September 17, 1992, discussing a particular containment isolation MOV, Limitorque stated that there is a high probability that at least 110% of the start torque rating would be generally available in a Reliance ac motor as a by-product of the design requirement for high motor speed in revolutions per minute (RPM) at its starting torque. Limitorque further stated that, in a scenario in which the torque switch is bypassed allowing the motor to reach stall, it believed that using 110% for motor starting torque would model actuator capability with a reasonable degree of probability.

Gear Efficiency

In its actuator selection data, Limitorque provides a table of pullout, run, and stall efficiencies for various actuators and their overall actuator ratios. In its Technical Update 92-2, Limitorque states that pullout efficiency is representative of the actuator overcoming inertial loads at start-up. Limitorque recommends that pullout efficiency be used in the sizing of actuators. In establishing a value for pullout efficiency, Limitorque assumes that the motor is at half speed and estimates the efficiency of the worm/worm gear interface from guidance in the American Gear Manufacturers Association (AGMA) standard 440.04 (October 1971), "AGMA Standard Practice for Single and Double-Reduction Cylindrical-Worm and Helical-Worm Speed Reducers." Limitorque then uses engineering judgement to reduce the worm efficiency predicted by the AGMA standard to estimate a pullout efficiency for the actuator.

When establishing run efficiency, Limitorque assumes that the motor is at full speed and determines the AGMA-predicted worm efficiency. Limitorque then reduces the worm efficiency by engineering judgement to estimate run efficiency. In its letter dated September 17, 1992, to Cleveland Electric, Limitorque stated that, for the containment isolation MOV in that case, the licensee could substitute the run efficiency for pullout efficiency because the application involved a close safety function with no potential for the actuator stopping at any point during the close stroke.

In Maintenance Update 92-1, Limitorque indicates that the stall efficiency is not a true efficiency and is to be used only in overload analysis. Limitorque estimates an approximate stall efficiency by increasing the run efficiency based on engineering judgement to account for inertia.

During discussions with Limitorque personnel, the inspectors questioned the relative values for run and stall efficiencies as stated in the Limitorque efficiency table for certain SMB-3 actuators. Limitorque subsequently determined that these efficiencies were reversed in the table. Limitorque stated that a notice under 10 CFR Part 21 would be submitted to alert licensees to the errors in the table. (Unresolved Item 99900100/93-01-05).

Limitorque has not performed testing to determine the accuracy of its estimates of gear efficiency. The inspectors noted that significant uncertainties exist in the use of worm efficiency and engineering judgement to predict overall gear efficiency. In addition, Limitorque does not address the AGMA standard recommendation to establish a thermal power rating limit by testing when determining worm/worm gear efficiency under high ambient temperature applications.

Application Factor

In Technical Update 92-2, Limitorque stated that the application factor takes into account variances of the motor starting torque and the pullout efficiency at varying voltage levels (down to 90% of nominal) and various actuator speeds and conditions. In its sizing criteria, Limitorque provides an application factor of 0.9 for most cases with reduced application factors for more complex conditions. The inspectors found that the application factor is not based on testing, but on the engineering judgement of Limitorque. During the inspection, Limitorque reaffirmed the basis for the application factor and stated that it does not recommend removing the application factor.

Degraded Voltage Factor

The degraded voltage factor is set to one where the minimum voltage at the motor terminals is equal to or greater than 90% of the rated motor voltage. If the minimum voltage is less than 90% but greater than 70% of the rated voltage, Limitorque states that the degraded voltage factor is equal to the ratio of the minimum voltage to the rated voltage for dc motors and to this ratio squared for ac motors. Limitorque does not have a specific relationship below 70% of the rated voltage. During the inspection, Limitorque stated that it based the relationship down to 70% of rated voltage on specifications for MOVs and motor vendor statements, and not on specific testing. With regard to determining the minimum motor terminal voltage, Limitorque stated that a study of power factor was being conducted and should be complete by the end of 1993.

In summary, the inspectors found that the values for individual parameters assumed in the Limitorque criteria for sizing motor actuators are not determined by testing, but are based almost entirely on engineering judgement. Limitorque has confidence in the sizing criteria as a result of actuators delivering the torque predicted by the sizing criteria during testing at Limitorque and nuclear power plants. Limitorque stated that it continues to recommend the use of nominal motor starting torque, pullout efficiency, and the application factor in reviewing the capability of its motor actuators. The NRC staff has confidence in the prediction of output torque by the Limitorque sizing criteria based on the success of the Limitorque criteria for many years in sizing actuators. However, the staff does not believe that confidence exists in the relative values for individual parameters in the sizing criteria. In other words, one value assumed for a particular parameter may be low and the value for another parameter may be high, but the overall output torque predicted by the sizing criteria may be adequate. Limitorque stated that requests for a Limitorque position on actuator output and other technical issues may be obtained only from the Limitorque Nuclear Support Group and must be confirmed in writing.

3.8.2 Use of Spring Pack Curves in Setting Actuator Torque Switches

In the past, Limitorque prepared curves that provided an estimate of actuator output torque based on torque switch settings or the displacement of the spring pack in the actuator. Limitorque stated that the upper limit of the curve was typically based on 75 to 80% compression of the spring pack while the lower limit represented the pre-load torque of the spring pack. Limitorque stated that these old spring pack curves were developed by a combination of test and analysis. Limitorque did not have any information on the tolerance bands for the curves, but stated that they might be significant.

Limitorque is developing new curves to reflect actuator torque as a function of spring pack compression with tolerance bands. Limitorque also will attempt to develop curves of actuator torque versus torque switch setting although significant uncertainties exist regarding torque switch differences and repeatability. Limitorque stated that the data should be collected by mid-summer 1993. Limitorque stated that it is determining a more accurate method to set the pre-load based on spring pack force or deflection using a spring pack tester. This method will require the spring pack tester to be extremely stiff to allow an accurate pre-load setting. Limitorque has not determined the length of time that the calibration of a particular spring pack will remain accurate. Limitorque has not determined whether the new curves will be applicable to spring packs currently installed in actuators.

4 PERSONNEL CONTACTED

*	+	Ivan Wilkinson, Vice president, Engineering
*	+	Rory Segen, Quality Assurance Manager
*	+	William Miluszusky, Quality Control Manager
*	+	Frank Napoli, Quality Assurance Engineer
	+	John Franklin, Vice President Finance
*	+	Pat McQuillan, Nuclear Project Manager
		Gregory Pence, Chief Engineer
		Curtis Eshleman, Special Projects Engineer
		Jesse Puryear, Quality Control Inspector
		David Page, Quality Control Inspector
		Hubert Riley, Gage Laboratory Technician

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* Attended the entrance meeting on May 10, 1993
\+ Attended the exit meeting on May 14, 1993

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

APR 19 1993

Docket No. 99900081

Mr. Carl Volmer, Director
Quality Assurance
Siemens Power Corporation
Nuclear Division
Engineering and Manufacturing Facility
2101 Horn Rapids Road
P.O. Box 130
Richland, Washington 99352-0130

Dear Mr. Volmer:

SUBJECT: NRC INSPECTION REPORT NO. 99900081/92-02

This letter addresses the inspection of your facility at Richland, Washington conducted by Mr. S. L. Magruder, Mr. D. F. Kirsch, and Mr. E. D. Kendrick of the Nuclear Regulatory Commission (NRC) and Mr. C. E. Beyer and Ms. J. M. Cuta of Pacific Northwest Laboratory (PNL) on December 14-17, 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 evaluate the effectiveness of the engineering department at Siemens Power Corporation (SPC), Nuclear Division-Engineering and Manufacturing Facility, with regard to providing adequate core design information to reactor licensees.

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 the overall talent and experience level of SPC's engineering department. SPC's internal audit program was also considered to be a strength.

No violations, nonconformances, or unresolved items were identified during this inspection, however, the inspectors noted weaknesses in some program areas which are summarized as follows: (1) the lack of a formal program to track technical documents received from outside sources; (2) the lack of a defined program, approved at the corporate level, for technical training; (3) guideline procedures used in the design analysis process not being maintained up to date; and (4) calculation notebooks not being maintained as QA records between the time they are reviewed and microfilmed. No response to this letter is required.

The relationship between SPC and licensees, Washington Public Power Supply System (WPPSS) in particular, was also examined by the inspectors. A concern was raised that the definition of responsibility for core design, and the related analysis, is not always clear between SPC and the licensee. SPC should continue with the actions it has been taking to ensure that licensees understand the responsibilities they are assuming and that the best possible analysis is done.

The inspection team discussed the likely informational content of the inspection report with regard to propietary documents reviewed by the team during the inspection. It is our understanding that nothing in this report is considered to be proprietary. Unless informed otherwise, within 10 days of the date of this letter, a copy of this letter and its enclosure will be placed in the NRC Public Document Room in accordance with 10 CFR 2.790 of the NRC's "Rules of Practice."

Sincerely,

Leif J. Norrholm

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

Enclosure:
Inspection Report No. 99900081/92-02

ORGANIZATION:	SIEMENS POWER CORPORATION NUCLEAR DIVISION ENGINEERING AND MANUFACTURING FACILITY RICHLAND, WASHINGTON
REPORT NO.:	99900081/92-02
CORRESPONDENCE ADDRESS:	Siemens Power Corporation (SPC) Nuclear Division Engineering and Manufacturing Facility 2101 Horn Rapids Road P.O. Box 130 Richland, Washington 99352-0130
ORGANIZATIONAL CONTACT:	Mr. Carl Volmer, Director Quality Assurance
NUCLEAR INDUSTRY ACTIVITY:	Nuclear fuel pellet and assembly supplier.
INSPECTION CONDUCTED:	December 14-17, 1992

Stewart L. Magruder — 4/5/93

Stewart L. Magruder, Team Leader — Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (VIB)

Edward D. Kendrick, NRC,
Reactor Systems Branch
Dennis F. Kirsch, NRC, Region V
Judith M. Cuta, Pacific Northwest Laboratory
Carl E. Beyer, Pacific Northwest Laboratory

Uldis Potapovs — 4-5-93

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

INSPECTION BASES:	10 CFR Part 21 and 10 CFR Part 50, Appendix B
INSPECTION SCOPE:	Evaluate the effectiveness of the SPC engineering department with regard to providing adequate core design information to reactor licensees.
PLANT SITE APPLICABILITY:	Numerous Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) sites.

1 INSPECTION SUMMARY

Siemens Power Corporation (SPC) supplies nuclear fuel assemblies for both Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) design reactors. The purpose of this inspection was to evaluate the effectiveness of the engineering department at SPC with regard to providing adequate core design information to reactor licensees. The inspection was prompted by an NRC Augmented Inspection Team (AIT) inspection conducted at Washington Nuclear Power, Unit 2 (WNP-2) in August 1992. (Inspection Report No. 50-397/92-30)

No violations, nonconformances, or open items were discovered during the inspection. Weaknesses in the areas of: interface with external organizations, a defined training program, the design process, and records handling were noted. The overall knowledge and experience level of SPC's engineering department and SPC's internal audit program were considered to be strengths. The inspectors determined that the problems with power oscillations that led to the AIT at WNP-2 were not the result of any SPC errors or deficiencies. The inspectors concurred with SPC's analysis of the event which concluded that the primary cause was a highly skewed power distribution caused by the startup rod pattern used by the WNP-2 operators.

2 STATUS OF PREVIOUS INSPECTION FINDINGS:

The scope of the inspection did not include a review of the status of previous inspection findings.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

The inspectors informed SPC staff of the scope of the inspection, outlined areas of concern, and established working interfaces during the entrance meeting on December 14, 1992. On December 17, 1992, the inspectors summarized the results of the inspection for SPC management during the exit meeting. Uldis Potapovs, Section Chief, Reactive Inspection Section 1, Vendor Inspection Branch, and John B. Martin, Administrator, Region V, also participated in the exit meeting.

3.2 Inspection Scope

SPC produces fuel assemblies for General Electric design BWRs and Westinghouse and Combustion Engineering design PWRs. SPC reload fuel has been supplied to 21 currently operating U.S. light water reactors: 9 BWR and 12 PWR plants. SPC also produces fuel pellets for Babcock & Wilcox's Commercial Nuclear Fuel Plant.

SPC's engineering staff performs analyses of the reactor core design, as specified by the utilities, including neutronic, thermal-mechanical, and thermal-hydraulic calculations.

This inspection was a performance-based engineering inspection at the SPC Engineering and Manufacturing Facility. The inspection evaluated the effectiveness of the engineering department at SPC with regard to providing adequate core design information to reactor licensees. The inspectors examined the engineering design process from a programmatic point of view and also conducted detailed reviews of selected design calculations from recent reloads at Grand Gulf Nuclear, Unit 1 (GGN-1) and WNP-2. Specifically, the inspectors covered the following areas: engineering department organization; training and qualification; design process; configuration control; internal audit program; interface with NRC and industry; core analysis codes and procedures; and the recent core instability event at WNP-2.

3.3 Engineering Department Organization

The inspectors reviewed the organization and staffing of the engineering department. The department is made up of four major groups: corporate information services; product mechanical engineering; BWR nuclear engineering (BWR-NE); and PWR nuclear engineering (PWR-NE). Product licensing is also included in the department's responsibilities. The BWR-NE and PWR-NE groups are responsible for the analyses that the inspectors were interested in, therefore, the inspectors focused mainly on personnel in these groups. The scope of these groups' responsibilities include: (1) ensuring that technical specifications on fuel operating limits for events listed in Chapter 15 of the NRC Standard Review Plan are met; and (2) nuclear analyses such as enrichment and burnable poison, core design, fuel cycle design, and startup and operations reports.

The inspectors determined that jobs and responsibilities were well defined and understood in the engineering department and that the workload did not appear to be excessive. SPC uses a project team concept that allows junior engineers to work with senior engineers in a mentoring environment. This concept seems to be an effective way of integrating new personnel into the department while maintaining a high quality of work.

3.4 Training and Qualification

Training of the staff consists of both QA indoctrination (new employee training) and technical training performed by the engineering organizations. QA indoctrination is programmatically mandated by procedure ANF-P00,019, "Indoctrination and Training," Rev. 7, April 1990. The inspectors examined this procedure and found that the procedure:

- covers QA indoctrination and training, and training records

- makes supervisors/managers responsible for defining, implementing and documenting job related technical training

- doesn't mandate any technical training program, but suggests on-the-job training, in-house training programs or specialized training by organizations external to the company

- requires employee signature for reading and understanding engineering design related activity program/policy procedures.

The inspectors reviewed the technical training program that was being implemented by the BWR-NE group. The inspectors found an excellent, well documented, program that involved a weekly lecture series consisting of approximately 50 different technical topics. These lectures were assigned to the various experts in the group and included topics such as: methodology and inputs for the applicable computer codes; transient and accident analysis; axial enrichment study; fuel assembly design; neutronic overview; stability; core design; and reactor systems. Training records were maintained for each individual in the group to indicate what lectures they had attended.

The inspectors also examined PWR-NE training records for four engineers and one team leader. These appeared to be similar to BWR-NE training records and were in accordance with policies. The PWR-NE seminar/lecture programs had recently (November 1992) been established as a bi-weekly lecture program. The lecture topic coverage is much less extensive than that of the BWR-NE training program. The PWR-NE organization had not established a formally documented program for staff technical training. The inspectors concluded that PWR-NE training is much less developed than the BWR-NE technical training program.

The inspectors examined two internal audits of training. The first, performed during November-December 1991 found that a training program compliant with procedure ANF-P00,019, Rev.7, had not been developed and identified necessary actions to correct the discrepancies. The second audit was performed during July 23-31, 1992. Four Corrective Action Reports (CARs) were issued to correct the deficiencies noted in the audit report, primarily in the area of training record documentation. The audits were quality efforts with good findings; however, it appears that the thoroughness and adequacy of technical training was not fully evaluated.

The inspectors concluded that the lack of a defined program, at the corporate level, represented a weakness in the training area. It was clear that the BWR-NE group had done a good job of specifying technical training for engineers; however, the PWR-NE group was less sophisticated.

3.5 Design Process

The inspectors examined the following procedures specifying the controls over the design process:

- EMF-954, "Preparation of Calculation Notebooks," Rev. 6, July 1992

- EMF-1040(P), "Preparation of Design Calculations," Rev. 2, March 1992

- ANF-868, "Procedures for BWR Safety Analysis Calculations"

- ANF-1238, "Procedures for PWR Safety Analysis Calculations"

- EMF-P00052, "Procedure for Fuel Design Analysis Review," Rev 0, October 1992

- ANF-P00045, "Internal and External Interfaces for Reload Fuel Design Parameters," Rev. 4, March 1991

- QAP-3, "Design Control for Nuclear Fuel," Rev. 21, May 1991

- QAP-6, "Document Identification and Control System for Nuclear Fuels," Rev. 25, July 1992

The inspectors considered that the design controls specified by the above program documents were adequate and appropriate for detailing the necessary design program controls. In particular, the process specified for performing, documenting, and reviewing original and revised design calculations appeared adequate.

The following attributes were specifically considered in reviewing the procedures:

- preparation of design calculations

- documentation of design calculations and assumptions

- interfaces between engineering, the customer, and other SPC organizations

- design change control

- independent design review of original and changed designs

- definition of individual responsibilities

Interviews with two BWR-NE design engineers revealed that guideline procedures ANF-868 and ANF-1238, which are used to ensure consistency in design analysis, have not been maintained up to date and are not as useful as they should be. Some of the codes that are referenced in the procedures are no longer in use and the engineers stated that some of the procedures are not followed at all.

The inspectors raised this concern to SPC management at the exit meeting. It was noted that this problem had been identified by SPC in internal audit report 92:56 and the inspectors did not find any examples of calculations that had been done incorrectly as a result. SPC is planning to revise both guideline procedures.

The inspectors concluded that, with the above exception, the SPC procedures provided an adequate and appropriate system for defining necessary design considerations and controlling the design process.

The inspectors sampled two design changes to assess whether the design changes were processed in accordance with procedural specifications and concluded that the changes were done as specified.

The staffing levels on two contracts and the overall adequacy of staffing were discussed during interviews with several engineers. The inspectors concluded that staffing levels were adequate and that project resources did not appear to be stressed.

3.6 Quality Records

Quality records such as calculation notebooks and design calculations are retained by the responsible engineer until about six months after contract completion and then sent to the document control facility for microfilming and storage. The inspectors expressed the concern that this practice did not adequately protect these quality documents from loss, damage, or deterioration between the time of their generation and approval until their transfer to the document control center.

The inspectors examined the SPC records storage facility and observed that:

- the storage facility was enclosed by concrete walls, floor and ceiling
- the environment was temperature and humidity controlled
- the door to the facility was at least a two-hour fire rated door

- access was monitored and restricted
- a small CO_2 wall mounted fire extinguisher was located in the facility
- electrical wiring was routed in conduits
- microfiche and hard copy records were stored in standard metal cabinets

ANSI N45.2.9 requires a method of fire protection in such a storage facility. The SPC facility has a single extinguisher and a smoke detector that alarms at the guard's desk. The inspectors determined that this was adequate protection since all the wiring was routed in conduits and almost all the equipment in the facility is turned off at night.

3.7 Internal Audit Program

The inspectors reviewed four recent internal audits of the SPC engineering department. The inspectors found that the audits were very well done. All of the reports indicated that the auditors had done a thorough job and had raised thoughtful and important questions about the way things were being done in the department. 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.

Audit 90:90, "Design Control," was conducted by a team of QA auditors from October 10, 1990, to November 2, 1990. The scope of the audit was defined as reviewing design records for H.B. Robinson ANF-11 and Washington Public Power Supply System (WPPSS) ANF 9x9-9X and ANF 9x9-IX lead fuel projects in accordance with Criterion III of 10 CFR Part 50, Appendix B. The audit resulted in four findings, one concern, and six comments. The findings and concern were issued as Corrective Action Reports (CARs) which were promptly and adequately answered by engineering.

Audit 91:85, "Design Control," was conducted by a QA auditor December 12 - 14, 16 & 20, 1991. The scope of the audit was defined as reviewing design documentation applicable to Dresden 3 ANF-6 and Millstone 3 ANF-2 fuel. The documentation reviewed included design criteria, calculation notebooks and design calculations. Two CARs were issued and promptly answered.

Audit 92:29, "Computer Code Model Development," was conducted by a multi-disciplined team of QA auditors and engineers from March 3-6, 1992. This audit was conducted at the request of the Vice President of Engineering as a result of a disgruntled engineer leaving the company. The audit scope included two pieces of software that are still being developed; EXEM BWR ECCS and REALISTIC LOCA. Calculation notebooks and software development records were reviewed by the audit team. Four CARs were issued from this audit and were promptly answered by engineering.

Audit 92:56, "Engineering," was also conducted by a multi-disciplined team of QA auditors and engineers from November 16-20, 1992. The scope of the audit was very broad and included reviewing the engineering department's implementation of the requirements of Criteria I, II, III, V, VI, XI, XVI, XVII, and XVIII of Appendix B to 10 CFR Part 50. The audit report, which was still in draft form, concluded that most of the SPC QA program and sub-tier procedures were being effectively implemented. It did note that certain areas of calculation guidelines/procedures and product testing procedures require serious review. Seven CARs were issued as a result of the audit.

The internal audit program was considered to be a strength by the inspectors. In particular, the audits were found to be thorough and were conducted by knowledgeable personnel. The audits generated good findings that prompted thoughtful corrective actions and the inspectors found that the use of multi-disciplined audit teams was a very effective approach.

3.8 Interface with NRC and Industry

The inspectors reviewed SPC's interfaces with the NRC and various industry groups with a particular interest in determining whether SPC was receiving, and acting on, technical documents from these sources. The review was prompted by the discovery by the AIT that SPC personnel were not aware of a March 18, 1992, BWR Owner's Group (BWROG) advisory letter, "Implementation Guidance for Stability Interim Corrective Actions."

The inspectors found that SPC does not have a formally documented method for receiving and tracking documents from outside sources. Currently, incoming documents are routed to the responsible individuals, however, no system exists for ensuring that there is a record that the document was received and that any required actions were taken.

The inspectors did note that this problem had been identified by SPC in internal audit report 92:56 and that the Product Licensing Manager had been recently assigned the responsibility of developing an effective program.

Discussions with SPC personnel revealed that SPC would like to become more involved in the BWROG, but inherent obstacles periodically arise having to do with involved parties revealing proprietary information. The inspectors urged the SPC personnel to try to stay informed of BWROG activities through their utility contacts.

3.9 Core Analysis Codes and Procedures

The inspectors reviewed both the core analysis computer codes and the core analysis procedures currently used by SPC in the neutronic, thermal-mechanical and thermal-hydraulic design and licensing process for fuel reload design. The computer codes examined during this review included MICBURN-3, CASMO-3G, MICROBURN-B, RODEX2, RODEX2A, COTRAN, COTRANSA, XCOBRA, HUXY, and XCOBRA-T. The procedures for the application of these codes were reviewed by inspecting the fuel reload design process for two specific BWR reloads: Cycle 6 of GGN-1, and Cycle 8 of WNP-2. Due to time constraints, all calculational files could not be reviewed; however, a representative set of similar calculations for each of the two specific reloads were examined in detail to check for consistency between the application of codes and procedures.

As an extension of this core design process review, the inspectors also reviewed the design responsibility interface between SPC and its customers to determine the process used by SPC to determine the thermal-hydraulic compatibility of its fuel with other vendor fuel designs.

3.9.1 Review of Computer Codes

The review of the computer codes included verification that only approved versions of the codes are being used and that any limitations or restrictions imposed in the NRC Safety Evaluation Reports (SERs) are being followed. Coding changes that have been implemented were also reviewed to ensure that they are well documented, following the approved SPC QA procedures, and that no changes to the approved methodologies were inadvertently made; in particular, when individual components of the core design package are "upgraded". SPC assigns a code custodian to each of the computer codes. The custodian is responsible for making all changes to the code and ensuring that these changes only correct unintentional errors and do not alter the code or modeling procedures approved by the NRC. A user or code developer for an NRC-approved code provides a written request to the code custodian giving the reason for the change and the proposed alteration to the code. If the change is found acceptable by the custodian, it is implemented by the custodian and the coding changes are independently reviewed by a staff member knowledgeable in the development and application of this code. Each of the computer codes reviewed is discussed briefly below.

3.9.1.1 MICBURN-3/CASMO-3G

The MICBURN-3/CASMO-3G code package is described in Supplement 3 to Volume 1 of the SPC topical report XN-NF-80-19(P) (A), "Advanced Nuclear Fuels Methodology for Boiling Water Reactors, Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Rev. 0, February 1989. The NRC SER accepted this model for BWR fuel assembly two-dimensional lattice neutronics applications in 1990. For advanced fuel designs, this model has replaced the previously approved XFYRE (HRG/THERMOS) methodology. Specific limitations/restrictions placed on the use of the new methodology were that any application to fuel designs that differ significantly from those in the Supplement 3 data base should be supported by additional code validation to ensure that the approved methodology and uncertainties are applicable.

3.9.1.2 MICROBURN-B

MICROBURN-B was also described in SPC topical report XN-NF-80-19(P) (A) and was approved by the NRC for use as a three-dimensional core simulator, supplementing the previously approved XTGBWR CMDT model. Restrictions on the application of this methodology were that the previously approved Traversing Incore Probe (TIP) asymmetry uncertainty value of 6.0 percent should continue to be used in determining the radial bundle power uncertainty.

Coding changes that were made to the approved version were reviewed by tracing the Software Development Records (SDRs) for the last four revisions: UOCT89, UDEC89, UDEC91 and UFEB92. It was confirmed that the changes involved only minor corrections, output edit expansions, or conversions to run on new computer hardware and that no changes to the approved methodology were made. Also, all changes were made in accordance with ANF-608, "Engineering Computer Code Control Requirements," Rev. 6, December 1988.

3.9.1.3 RODEX2

The NRC SER for the RODEX2 fuel performance code was issued in an NRC memorandum (proprietary) dated October 13, 1983 and the approved code was documented by SPC in topical report XN-NF-81-58 (P) (A), "RODEX2 - Fuel Rod Thermal Mechanical Response Evaluation Model," Supplements 1 and 2, Rev. 2, March 1984.

Since the issuance of the proprietary SER, SPC has made some changes to the RODEX2 code. The current SPC code custodian was questioned about a recent change to the fuel swelling model in RODEX2 that was recommended by the code developers. These changes were found to be acceptable and were implemented. The code developers then checked the changes implemented by the code custodian. These changes were evaluated by the inspection team and found to be consistent with the total fuel swelling prediction of the original model and, therefore, acceptable.

From the RODEX2 documentation of changes to the code, it was noted that a change was recommended by a SPC engineer in 1984 that appeared to change the calculation of gap conductance. The code documentation suggested that the change was not implemented for the licensing version of RODEX2. This issue was further discussed with the current RODEX2 code custodian who confirmed that this code change was not implemented in the more conservative licensing version. Further discussions with SPC BWR-NE staff, who utilize the code for licensing calculations for fuel reloads, confirmed that the original approved RODEX2 model for gap conductance is used for loss-of-coolant accident (LOCA) and minimum critical power ratio (MCPR) calculations.

A third change to the RODEX2 code, in the model for BWR cladding creep, was also recommended in the RODEX2 documentation, but it was not apparent from the documentation if the change was implemented. The inspectors discussed the issue with the SPC Product Licensing Manager and determined that the changes to cladding creep were not implemented in the licensing version of RODEX2. It was noted that the RODEX2 documentation of code changes did not make it clear that this change had not been included in the NRC approved licensing version of RODEX2. The SPC code users did, however, appear to be aware that certain changes are not to be used for licensing calculations.

The inspectors concluded that RODEX2 was maintained in a satisfactory manner and the code and models within are the same as originally approved by the NRC. In addition, those SPC staff that utilize this code appear to be knowledgeable about which changes to the code are acceptable for licensing applications and those that are not acceptable. The latter changes are not applied for licensing calculations. However, RODEX2 code documentation of changes do not make it clear which changes to the code should not be used for licensing calculations in support of reload analyses. The inspectors suggested that this could be made clearer in the documentation of changes to the code.

3.9.1.4 RODEX2A

The RODEX2A code, XN-NF-85-74(P) (A), "RODEX2A (BWR) - Fuel Rod Thermal Mechanical Evaluation Model," Rev. 0, August 1986, is used for thermal-mechanical design analyses that include calculations for internal fuel rod pressures, steady-state cladding strain, corrosion, initial conditions for calculating cladding collapse, and steady-state fuel temperatures for fuel melting for normal operation. Calculations of rod internal pressure must be performed to at least a minimum peak pellet burnup of 50 megawatt-days per kilogram (MWd/kgM), using approved power history methodology, to establish that the design limit is satisfied. Application of RODEX2A was found acceptable for licensing applications for steady state operation only for the specific analyses cited in the NRC SER.

3.9.1.5 COTRAN

COTRAN, XN-NF-691(P) (A) and Supplement 1, "Stability Evaluation of Boiling Water Reactor Cores, Sensitivity Analyses and Benchmark Analyses," Rev 0, August 1984, is a two-dimensional (R-Z) reactor kinetics computer program which solves the space and time dependent one-group neutron diffusion equation, using one prompt and six delayed neutron groups, with fuel temperature (Doppler) and void reactivity feedback. The reactivity feedbacks are determined from a solution of the equations of mass, energy and momentum for the coolant coupled with a fuel conduction model which calculates the axial and radial temperature distribution for a fuel rod. The COTRAN model uses input from the CASMO and MICROBURN (formerly XFYRE and XTGBWR) codes including cross sections, rod worths, initial flux and power shapes, peaking factors and other initial condition parameters. RODEX2 provides initial values of some physical parameters, such as gap conductance and rod geometry. The code uses forcing functions as a function of time for several system parameters to allow COTRAN to model the reactor while including the total system feedback. COTRAN is approved both for reactor core (reactivity) stability and for channel hydrodynamic (flow oscillations) stability analysis. Limitations on the application of the model include:

- COTRAN is to be applied to the analysis of BWR channels with low speed flow and significant surface heat transfer,

- COTRAN is acceptable only for the analysis of the transients described in XN-NF-80-19(P), "Exxon Nuclear Methodology for Boiling Water Reactors: THERMEX Thermal Limits Methodology, Summary Description," Vol. 3, Rev. 1, April 1981.

The application of the stability methodology for use in licensing reload fuel was also approved under either of the following conditions:

- the calculated decay ratio for the proposed cycle is less than or equal to 0.75 and acceptable Technical Specification (TS) restrictions are placed on natural circulation operation; or

- the calculated decay ratio for the proposed cycle is less than or equal to 0.90 and acceptable TS requirements are placed on natural circulation and single loop operation including proper surveillance of both Local Power Range Monitors and Average Power Range Monitors.

3.9.1.6 COTRANSA

COTRANSA2, XN-NF-84-67 (P), Supplement 2, "Stability Analysis Methodology for BWR Cores," Rev. 0, July 1984 and XN-NF-86-113(P), "COTRANSA Updated Hot Channel Model," Rev. 0, August 1986, is a primary system analysis code that calculates the flow around the primary loop using a one-dimensional modeling approach. Structures such as piping, pumps, and the reactor vessel are treated as essentially one-dimensional components, and models for specific structures such as jet pumps are included as semi-empirical models or correlations. The core is treated as an average one-dimensional channel, with power input determined by appropriate collapsing and averaging of calculations from the physics code, MICROBURN.

3.9.1.7 XCOBRA/HUXY

This core analysis code set, XN-NF-80-19(P), "Exxon Nuclear Methodology for Boiling Water Reactors: THERMEX Thermal Limits Methodology, Summary Description," Vol. 3, Rev. 1, April 1981 and XN-CC-33 (A), "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Revision 1, November 1975 was derived from the COBRA-IV code, developed at Pacific Northwest Laboratories for the NRC. It uses a mixture model of two-phase flow, in which the conservation equations for mass, energy, and momentum are solved for a single homogeneous fluid, using a subchannel formulation. Phase slip is modeled with constitutive correlations for two-phase pressure drop and void/quality relations, with subcooled boiling models to account for void formation at the heated wall that can occur when the bulk fluid is subcooled. Since the BWR core is modeled by SPC as an array of parallel one-dimensional channels that see a uniform pressure drop boundary condition, the subchannel analysis capability is not used in SPC applications.

3.9.1.8 XCOBRA-T

This code, XN-NF-84-105(P), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Supplements 1 and 2, Rev. 2, May 1985, is essentially the same as XCOBRA, and was designed to automate calculations formerly done with the XCOBRA and HUXY codes. It includes a simple leakage flow model to calculate core bypass flow, and has a fuel rod model that can account for transient thermal effects in the fuel.

Because of the time limits of the review, the thermal-mechanical and thermal-hydraulic codes (RODEX2A, COTRAN, COTRANSA, XCOBRA/HUXY, and XCOBRA-T) were not reviewed by the inspectors to verify that they have not been changed from those originally approved by the NRC. The applications of these codes to reload licensing applications were reviewed and are discussed in the following subsection.

3.9.2 Application of Computer Codes and Analytical Models to Reload Design

The application of the computer codes and analytical models for reload licensing and design analyses for the GGN-1, Cycle 6 and for the WNP-2, Cycle 8 core reloads were reviewed. This review was guided by EMF-1040(P), "Procedure for Preparation of Design Calculations," Rev. 2, March 1992 and EMF-954, "Procedure for Preparation of Calculation Notebooks," Rev. 6, July 1992. The applications are discussed below for each code/model reviewed.

3.9.2.1 Neutronics codes (MICBURN-3/CASMO-3G/MICROBURN-B)

Calculation notebooks were reviewed by inspection and by interviewing the analysts and reviewer for both of the subject reload designs. The design reports that are supplied to the customer to support licensing activities were also reviewed and cross-compared. A special effort was made to identify calculation files which had undergone changes and/or corrections during the QA review process. It was noted that a reload design is developed over a span of two to four years from the initial contract to cycle startup, although the intensive design phase covers approximately one year. All calculations, assumptions and applications that were reviewed were determined to be in accordance with the approved methodologies.

3.9.2.2 Thermal-mechanical fuel performance codes

3.9.2.2.1 RODEX2

The RODEX2 code is used primarily for calculating initial stored energy, rod pressures, and gap conductance values for input to COTRAN, COTRANSA, and XCOBRA-T for transients and accidents [e.g., LOCA, anticipated operational occurrences (AOOs), transient strain, and minimum critical power ratios (MCPRs)]. The application of RODEX2 for stored energy and gap conductance input for the LOCA analysis was examined by the inspection team for both the GGN-1 Cycle 6 and WNP-2 Cycle 8 reloads. Both of these analytical applications were found to be conservative and consistent with the original NRC approval of RODEX2. However, in examining the 9x9-5 design input for the GGN-1 reload analysis, it was noted that the fuel-to-cladding gap size was different from that originally approved by NRC for the 9x9-5 fuel design. SPC was questioned on this change in gap size. The Product Licensing Manager stated that they notified NRC of this change by a letter dated March 5, 1991, and that NRC approval of the change in gap size for the 9x9-5 fuel design was received in a letter dated November 6, 1991.

The application of RODEX2 for determining gap conductance values for input to MCPR analyses for both of the BWR applications was also examined. SPC was questioned on whether the cycle specific power history and axial power shape changes during the cycle were considered in the RODEX2 calculations of gap conductance. SPC BWR-NE staff indicated that the cycle specific power histories and axial power shapes are used in both the RODEX2 calculations of gap conductance and in determining the MCPR limits for each cycle. In addition, SPC has indicated that they follow the axial power shape change during the transient for determining the MCPR limits. This analysis approach is consistent with the NRC approval of this code. However, based on current technology for determining gap conductance values for MCPR, the RODEX2 calculated values may lead to potential non-conservatisms for this analysis. At the time RODEX2 was originally developed (late 1970s to early 1980s), LOCA analysis was generally the limiting condition for plant operation and, therefore, SPC (and NRC) primarily concentrated on making the RODEX2 code conservative in relation to LOCA stored energy and gap conductance. Low gap conductance values for LOCA are considered to be conservative; however, the reverse is true (high gap conductance values are conservative) for MCPR analyses. In addition, because RODEX2 is considered to be a conservative code for LOCA, the reverse is true for MCPR analyses. Many current BWR and PWR fuel operating limits are based on MCPR and departure from nucleate boiling (DNB) analyses, respectively, rather than LOCA limits. SPC relies on their engineering expertise to ensure that the proper code is used. In addition, they currently have a "best estimate" code (RODEX3), submitted to NRC for review, that should alleviate the potential non-conservatism in MCPR analyses.

3.9.2.2.2 RODEX2A

The RODEX2A analyses examined in this inspection of the GGN-1 Cycle 6 and the WNP-2 Cycle 8 reloads were internal fuel rod pressures, initial conditions for calculating cladding collapse, and steady-state fuel temperatures for fuel melting. From examination of the RODEX2A input and calculational results for the analyses described above for GGN-1 and WNP-2 reloads, it was concluded that these RODEX2A applications were conservative and consistent with the original NRC approval of RODEX2A.

3.9.2.3 Other Thermal-Mechanical Methods - Axial Rod and Assembly Growth

The other analytical models evaluated in the inspection of the GGN-1 and WNP-2 reloads were the axial growth models for the fuel rods and fuel assemblies for 9x9 designs, ANF-88-152(P) (A), "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel," Rev. 0, November 1988, and ANF-89-014(P) (A), Supplements 1 and 2, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," Rev 0, November 1991.

These two analytical models are used to determine that the clearances between (1) the end cap shanks of the fuel rod and the upper and bottom tie plates; and (2) the assembly tie plates and core internals are adequate to prevent disengagement or an interference fit at end-of-life. These analytical models were chosen because the NRC SERs of both the 9x9-5 and 9x9-IX designs recommended that SPC collect additional axial growth data up to the maximum burnup level for these designs. At the time the SERs were written, SPC had axial growth data only to within 80 to 85% of the maximum burnup levels requested. In addition, indications of unexpected differential axial growth were observed for some 8x8 fuel bundles at GGN during the last reload outage fuel inspection.

SPC was questioned on the additional axial growth data up to the maximum burnup levels, and the clearances between the end cap shanks of the fuel rods and the assembly tie plates, and between the assembly tie plates and core internals. SPC produced their measurements of 9x9 fuel rod and assembly growth that were very near the maximum burnup levels for the 9x9-5 and 9x9-IX designs. SPC also stated that the differential growth observed for an 8x8 design at GGN-1 had been traced to channel and end fitting binding. They also produced the calculational results, and associated analytical methodology, of clearances for the GGN-1 and WNP-2 reloads that demonstrated that end-of-life clearances were satisfactory. From these results it was concluded that SPC has complied with the recommendations of the earlier SERs; that the application of the fuel rod and assembly axial growth models are consistent with these SERs; and that there are adequate clearances in these applications to prevent disengagement or an interference fit at end-of-life for these fuel assemblies.

3.9.2.4 Thermal-hydraulic codes (COTRAN)

COTRAN is the SPC program for kinetics analysis of BWR cores. The application of COTRAN to core stability analyses was investigated in depth by the AIT after the WNP-2 event and was not pursued in detail during this inspection other than to assure that the approved methodology was used with the restrictions stated in the SER.

The thermal-hydraulic codes COTRANSA2, XCOBRA, and XCOBRA-T are interlinked for many of SPC's licensing analyses. The use of these codes by SPC is discussed in this section along with a description of how it is linked to the other thermal-hydraulic codes. Following the description of each individual code and its application, the inspection team evaluation of the application of these SPC thermal-hydraulic codes to the GGN-1 and WNP-2 fuel reloads is provided.

COTRANSA and COTRANSA2 provide the core average pressure drop and flows which are used to provide system boundary conditions for XCOBRA and XCOBRA-T calculations. The main SPC application of XCOBRA is in core thermal margin calculations for steady state conditions. The code evaluates the thermal-hydraulic conditions in the core, including the hot assembly, and determines the critical power ratio using an appropriate critical power ratio correlation. The core is not modeled in detail; the SPC model consists of only a half-dozen or so channels representing the hot assemblies of the various fuel types in the core, and a one-dimensional representation of the bypass. A water tube channel model has been added to the code, for newer fuel designs with extremely large water rods, but in other fuel design applications, the fuel assemblies are treated as one-dimensional flow channels. SPC has stated that they informed the NRC of this modeling change and that it did not change the overall results in terms of critical power ratio analyses. Power distribution data from calculations with the neutronics codes provide input for these calculations, and the pressure drop or flow boundary condition comes from a systems code calculation. XCOBRA-T is used to evaluate thermal margin limits in operational transients. In general, a single hot bundle is modeled for these analyses, with the initial steady-state conditions determined from a core model calculation using the XCOBRA code.

3.9.3 Evaluation of Application of Thermal-Hydraulic Codes

A sampling of calculations performed with the COTRAN, COTRANSA, and XCOBRA codes were reviewed by the inspection team for the GGN-1 and WNP-2 nuclear plants. Extensive discussions were also held with the SPC staff responsible for the calculations, and the inspection team asked specific questions on code structure, applications procedures, and modeling assumptions used in the analyses.

General observations and conclusions regarding the analytical capabilities at SPC in the area of core thermal-hydraulics were as follows:

- The SPC code users had a good technical grasp of the capabilities and limitations of the codes they were using and knew how to use them correctly.
- The SPC code users were also capable of adding new models to the thermal-hydraulic codes (for example, a water tube channel model in XCOBRA) to meet new analytical requirements for advanced designs.
- There was a marked improvement over time (from about 1978 to the present) in the completeness of the details of the calculations and procedures followed, as recorded in the calculation notebooks. This indicates that there has been

an evolution in the QA procedures over the past ten years that makes it easier to trace and verify the particular plant analyses.

The calculations and analyses examined by the inspection team followed NRC-approved procedures, and the SPC code users seemed to have a clear understanding of what these procedures were. A limitation noted by the inspectors with the SPC work was in the use of older thermal-hydraulic codes. The approved modeling approaches for BWR core thermal-hydraulics are based on approaches developed more than ten years ago and are rather simplistic and do not require a very high level of detail in the core models used. (Typically, a core can be represented with 6 to 10 channels, when in fact it consists of several hundred channels.) For a core with a relatively flat radial power distribution and having all bundles with the same fuel design, this is a reasonable approach. However, it is less satisfactory for mixed cores, where there are several different fuel designs, and more extreme local power peaking.

It should also be noted that these codes (esp. COTRAN) were not designed for core-wide stability analysis, particularly in the low flow/low power region where the various core oscillation transients have occurred. SPC is working on developing advanced methods (the STAIF code) for this type of analysis that specifically look at frequency-domain approaches which include reactivity feedback effects.

3.9.4 SPC/Customer Interface

This portion of the inspection focused on the contractual and technical interfaces between SPC and two of their BWR reload fuel customers. The review covered the vendor/customer responsibility splits and traced the process used by SPC to determine the compatibility of their current reload fuel with their existing fuel designs and with other vendors' fuel. In particular, it was questioned how large flow resistance mismatches may develop between SPC fuel designs (8x8 versus 9x9) and other vendors' (9x9 and 10x10) fuel designs.

The review started with the actual contract and resulting work orders for both WPPSS (WNP-2) and Entergy (GGN-1) and proceeded through the formal correspondence chain to the startup and operations report which is delivered after the beginning-of-cycle startup testing is evaluated. The differences that were noted between the two projects occurred because of the different contract options that the two utilities chose to exercise on a cycle reload basis. For example, Entergy chose to perform their own stability analysis for the GGN-1 reload, but contracted with SPC to perform core follow analysis calculations to monitor core performance during the operating cycle. WPPSS assumed the responsibility for loading the other vendor's Lead Test

Assemblies (LTAs) in the core and for the LOCA analysis, but chose to have SPC perform the stability analyses. The reload design team concept appears to be very important in ensuring that all aspects of the design are evaluated. This seems to be reinforced by the SPC practice of cross-training and shuffling personnel between the teams.

Although the inspection team had no contact with customer personnel, the impression gathered was that there is a wide range of technical capability and oversight ability among the different customers which SPC must accommodate. The total extent of vendor-customer interaction could not be determined from the formal documentation but it became evident that, because of their unique resources and engineering experience, SPC may wish to assume greater responsibility for the total design. The potential for increased responsibility was mentioned to SPC staff during the inspection and was emphasized during the exit meeting.

3.10 WNP-2 Event

3.10.1 Root Causes

The AIT concluded that the August 15, 1992, WNP-2 event was caused by two conditions:

1. Extremely skewed radial and axial power distributions caused by:
   - Control rod pattern selected by operator (without significant procedural control).
   - Core loading that placed highly-reactive (new) fuel in locations that were not procedurally controlled.
2. A mixed (9x9-9X and 8x8 fuel) core that was not 100% thermo-hydraulically compatible.

Other AIT findings, supported by calculations done at Oak Ridge National Laboratory, indicated that:

- Oscillations could have been out-of-phase
- A more conservative control rod pattern results in a decay ratio (DR) of 0.3 at same power/flow conditions (8/31 startup measured DR ≈ 0.2) compared to 1.05 for 8/15 startup.
- Mixed core starved flow from high-power channels and contributed to instability.
  - Full 8x8 core: DR ≈ 0.8.
  - Full 9x9-9X core: DR ≈ 0.9.

3.10.2 Corrective Actions

The following corrective actions were taken by the licensee as a result of the investigation that they performed in conjunction with SPC, and to address concerns raised by the AIT:

- WNP-2 will pre-analyze future control rod patterns for stability. Startup rod sequence and patterns for pump up-shift cannot be changed by the operator without analyses and review.
- Analyzed conditions prior to Flow Control Valve closure must result in a decay ratio less than 0.5.
- During startup, the following conditions must be met:
  - Measured core-wide decay ratio using the SPC developed Advanced Neutron Noise Analysis Software System (ANNA) system must be less than 0.6 at all times.
  - Minimum CPR must be >2.2 (was 1.9 on 8/15)
  - Maximum nodal peaking factor must be <3.4 (was 3.86 on 8/15)
  - Expected core-average axial peaking must be <1.45 (was 1.9 on 8/15)
  - Power must be <33% prior to pump upshift.
  - Feedwater temperature must be >295°F

3.10.3 Generic Implications

- Low-power maneuvering rod patterns may not be consistent with power distribution assumed in BWROG analyses.
- Adequacy of interim solution boundaries is questionable. Exercise caution near instability regions B and C per recent BWROG instructions.
- Mixed-core and fuel types increase instability concerns.

3.10.4 Inspection Team Observations

The inspection team discussed the WNP-2 instability event with the SPC BWR-NE staff involved in the WNP-2 Cycle 8 fuel reload design calculations. Based upon these discussions it was determined that:

- WNP-2 engineering staff specified a Cycle 8 core design and fuel loading configuration. SPC reviewed the proposed core design and performed their standard set of analyses to determine that all design criteria were met. Since all criteria were satisfied, SPC concurred in the WNP-2 specified core design.

- The Cycle 8 core contained some non-SPC Lead Test Assemblies (LTAs) in non-limiting locations. Although SPC was not furnished a complete description of the nuclear and thermal-hydraulic characteristics of these LTAs from other vendors, the presence and location of these LTAs was not found to be a contributor to the event.

- SPC was involved with WNP-2 engineering staff only in establishing the Cycle 8 initial cold critical rod configuration, the full-power/full-flow target rod configurations and, to some extent, with power reduction rod configurations. SPC does not review, nor were they asked to review, the rod configurations to be used between initial criticality and hot full power.

- The rod configuration specified by WNP-2 engineers at the point of the pump speed shift, where the instability event occurred, shifted the axial power shape to peak in the lower portion of the core and was the primary cause of the event.

- SPC contractual agreements with WNP-2 for the next (Cycle 9) reload require SPC to perform stability calculations early in the core design phase to aid in precluding similar events. This is also planned for future reloads.

- WNP-2 has committed, in their response to the AIT report, to use the ANNA Stability Monitoring System during future startups. This system has been previously approved by the NRC.

4 PERSONNEL CONTACTED

		J. Bonner, Engineer, BWR Nuclear Engineering
+	*	C. Brown, Manager, BWR Reload Analysis
+	*	R. Copeland, Manager, Product Licensing
+	*	L. Federico, Manager, BWR Nuclear Engineering
+	*	B. Femreite, Plant Manager, Richland
		M. Garrett, Team Leader (CECO/RWE)
+		D. Hill, Manager, Quality Control
+	*	A. Ho, Manager, BWR Reload Analysis
+	*	J. Holm, Manager, PWR Nuclear Engineering
+	*	T. Howe, Manager, Mechanical Design Engineering
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+	*	L. Maas, Manager, Regulatory Compliance
		J. Maryott, Neutronics Support
		S. Mellinger, Engineer, BWR Nuclear Engineering
+	*	J. Morgan, Vice President, Engineering
+	*	R. Nelson, Senior QA Engineer
+	*	A. Reparaz, Manager, Product Mechanical Engineering
+	*	J. Tandy, Senior QA Engineer
+		M. Valentine, Manager, Manufacturing Engineering
	*	R. Vaughn, Manager, Safety, Security and Licensing
+	*	C. Volmer, QA Manager
+	*	K. Wahlquist, QA Engineer
+	*	G. Waymire, Manager, Product Engineering
		P. Wimpy, Design Team Leader, BWR Nuclear Engineering

+ Attended Entrance Meeting on December 14, 1992

* Attended Exit Meeting on December 17, 1992

Selected Bulletins, Generic Letters, and Information Notices
Concerning Adequacy of Vendor Audits and
Quality of Vendor Products

ISSUED	TITLE
1. Information Notice 93-25	Electrical Penetration Assembly Degradation
2. Information Notice 93-33	Potential Deficiency of Certain Class 1E Instrumentation and Control Cables
3. Information Notice 93-37	Eyebolts With Indeterminate Properties Installed in Limitorque Valve Operator Housing Covers
4. Information Notice 93-42	Failure of Anti-Rotation Keys in Motor-Operated Valves Manufactured by Velan

CORRESPONDENCE RELATED TO VENDOR ISSUES

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

June 11, 1993

Docket No. 99901227

Mr. Darrell J. Moyer
Dresser Pump Division
Dresser Industries, Inc.
5715 Bickett Street
Huntington Park, California 90255-2634

Dear Mr. Moyer:

SUBJECT: RESPONSE TO 10 CFR PART 21 INQUIRY

By letter dated October 5, 1992, you requested the U. S. Nuclear Regulatory Commission's assistance regarding the understanding of particular requirements of Part 21 to Title 10 of the Code of Federal Regulations. We have addressed your four questions in Enclosure 1 to this letter.

Should you have any further questions, please contact Mr. Ronald Frahm, Jr. of my staff at (301) 504-2986 or Mr. Gregory Cwalina at (301) 504-2984.

Sincerely,

Leif J. Norrholm

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

Enclosure:
1. Response to Questions

RESPONSE TO DRESSER INDUSTRIES LETTER

QUESTION (1):

When we have imposed 10-CFR-21 in our purchase order to a vendor supplying a safety related item, are we required to verify in any way that this vendor has a program in place to satisfy the requirements of 10-CFR-21? (procedure, posting, records, etc.)

NRC RESPONSE:

No. The procuring entity's responsibility for ensuring compliance with the provisions of Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) by its contractors, suppliers, and consultants is limited to the requirement that each procurement document specifies that the provisions of 10 CFR Part 21 apply, when applicable. The NRC is responsible for evaluating the adequacy of the program.

QUESTION (2):

Is a vendor that accepts our purchase order imposing 10-CFR-21 required to have a quality program meeting the applicable portions of 10-CFR-50 Appendix B?

NRC RESPONSE:

No. A vendor that accepts a purchase order imposing 10 CFR Part 21 is not necessarily required to have a quality program meeting the applicable portions of 10 CFR Part 50 Appendix B. All "basic components," as defined by Section 21.3 of 10 CFR Part 21, are required to be manufactured and controlled under a 10 CFR Part 50 Appendix B program. Therefore, if the vendor does not have a 10 CFR Part 50 Appendix B quality program, either the procuring entity or a third party must control the safety-related activities under their own 10 CFR Part 50 Appendix B program. If the vendor is expected to supply a basic component in accordance with 10 CFR Part 50 Appendix B, this requirement should be clearly noted on the procurement document.

QUESTION (3):

Is a vendor that accepts a purchase order that imposes 10-CFR-50, Appendix B automatically required to have a procedure in place to comply with 10-CFR-21?

NRC RESPONSE:

Yes. A vendor that accepts a 10 CFR Part 50 Appendix B order has accepted a unique nuclear requirement, and would therefore be supplying a "basic component." Consequently, the regulations of 10 CFR Part 21 are automatically imposed for that procurement order. Section 21.21 of 10 CFR Part 21 requires that anyone subject to the regulations in this part adopt appropriate procedures to comply with these regulations. Section 21.31 of 10 CFR Part 21 requires that the entity procuring the basic component specify in the procurement document that the provisions of 10 CFR Part 21 apply. However, any vendor accepting a purchase order that imposes 10 CFR Part 50 Appendix B or any other unique nuclear requirement is required to have a 10 CFR Part 21 program whether or not the procurement document indicates that 10 CFR Part 21 applies.

QUESTION (4):

An organization subject to the requirements of 10-CFR-21 must comply with the posting requirements described in 21.6. Does 10-CFR-21 require training of individuals having functions described in that organizations 10-CFR-21 procedure?

NRC RESPONSE:

No. 10 CFR Part 21 does not specifically address or require training for individuals having functions described in an organization's 10 CFR Part 21 procedure. However, formal training would be an effective means to assure that all individuals fully understand and comply with the requirements of 10 CFR Part 21. Furthermore, Criterion II of 10 CFR Part 50 Appendix B requires that the quality assurance program provides for indoctrination and training of personnel performing activities affecting quality.

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