Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
October – December 1994

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001
2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328
3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Printing and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852-2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018-3308.

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

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
October - September 1994

Manuscript Completed: February 1995
Date Published: February 1995

DISCLAIMER

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

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

**Portions of this document may be illegible
in electronic image products. Images are
produced from the best available original
document.**

ABSTRACT

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from October 1994 through December 1994.

TABLE OF CONTENTS

	<u>PAGE</u>
Abstract	iii
Preface	vii
Inspection Reports	ix
Index	x
Selected Bulletins, Generic Letters, and Information Notices Concerning Adequacy of Vendor Audits and Quality of Vendor Products	153
Correspondence Related To Vendor Issues	154

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 Section (VIS) of the Special Inspection Branch reviews and inspects nuclear steam system suppliers (NSSSs), architect engineering (AE) firms, suppliers of products and services, independent testing laboratories performing equipment qualification tests, and holders of NRC 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 VIS 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.

NUREG 0040, "Licensee Contractor and Vendor Inspection Status Report," 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. NUREG 0040 also contains a

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

Correspondence with contractors and vendors relative to inspection data contained in NUREG 0040 is placed in the USNRC Public Document Room, located in Washington, D.C.

INSPECTION REPORTS

INDEX

<u>FACILITY</u>	<u>REPORT NUMBER</u>	<u>PAGE</u>
ABB Combustion Engineering, Inc. Windsor, Connecticut	99900102/94-01	1
ABB Service Company Cleveland, Ohio	99901281/94-01	48
Cardinal Industrial Products Las Vegas, Nevada	99901076/94-01	59
GE Nuclear Energy San Jose, California	99900403/94-02	79
Tennessee Valley Authority Watts Bar Nuclear Plant, Unit 1 Chattanooga, Tennessee	50-390/94-201	99

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

December 29, 1994

Mr. Donald E. Allen
Vice President, Total Quality
Combustion Engineering, Inc.
ABB Combustion Engineering
Nuclear Operations
P.O. Box 500
Windsor, Connecticut 06095-0500

SUBJECT: NRC INSPECTION NO. 99900002/94-01; 99900102/94-01

Dear Mr. Allen:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection conducted October 3 through 7, 1994, at the Fuel Operations-Windsor (CENFO) facility of Asea Brown Boveri (ABB) Combustion Engineering Nuclear Operations (ABB CENO) in Windsor, Connecticut; and the inspection conducted October 17 through 21, 1994, at the ABB CENO Fuel Operations, Nuclear Fuel Manufacturing-Hematite (CENFM) facility in Hematite, Missouri. The inspection of CENFO in Windsor, Connecticut, also included inspection of the fuel related activities of ABB CENOS Engineering Operations and Field Services organizations. The NRC inspection team, led by Steven M. Matthews, conducted a performance-based evaluation of the CENFO and CENFM management, staff, and quality programs and the implementation of those programs related to pressurized- and boiling-water reactor core reload analysis design, fuel assemblies, and other fuel-related services supplied to the U.S. nuclear industry.

The team (a) examined technical documentation, procedures, and representative records, (b) interviewed ABB CENO personnel, (c) held discussions with ABB CENO personnel, (d) listened to presentations by ABB CENO personnel, and (e) observed work activities. On the basis of this inspection, the staff determined that the implementation of the ABB CENO Topical Report, document CENPD-210A, Revision 7A, which had been approved by the NRC on February 17, 1993, as meeting the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B to 10 CFR Part 50), was deficient in certain areas and, on this basis, we are issuing a notice of nonconformance (Enclosure 1) to ABB CENO. The inspection report (Enclosure 2) contains a detailed discussion of the areas examined during the inspection and our findings.

The most important inspection finding was ABB CENOs failure, in several instances, to implement established procedures for verifying or checking the adequacy of design data and, in other instances, to establish adequate measures for the control of design interfaces. Regarding the adequacy of design data, certain unvalidated physics biases and uncertainty data were used by Arizona Public Service Company (APS) in its Palo Verde Unit 1 Cycle 4 reload safety analysis. Engineering Operations review and final verification of the reload design analysis did not recognize the use of the unvalidated data before its release to and use by APS.

Regarding control of design interfaces, ABB CENOs design interface activities were based on the assumption that data, such as core reload specification data, may be referenced as design input without verification. In the instances described in the report, ABB CENOs design interface activities and controls were not adequate to ensure that the information in the core reload specification data was consistent with the existing core analysis and safety analysis calculations for which ABB CENO is responsible.

The team observed that the joint engineering effort between ABB CENO and APS was a strength of both organizations. A significant part of certain Palo Verde reload analyses were performed by APS nuclear fuel personnel. The first reload in this arrangement was jointly analyzed by APS and Engineering Operations engineers. For the next reload, APS engineers performed the work at APS with Engineering Operations engineers reviewing the work.

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

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

D. Allen

-3-

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

Sincerely,

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

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

Docket Nos.:

99900002 (CENFO)

99900102 (CENFM)

Enclosures:

1. Notice of Nonconformance

2. Inspection Report:

99900002/94-01; 99900102/94-01

NOTICE OF NONCONFORMANCE

Combustion Engineering, Inc.
ABB Combustion Engineering
Nuclear Operations
Windsor, Connecticut, and
Hematite, Missouri

Docket Nos.: 99900002
99900102
Report No.: 94-01

On the basis of the results of U.S. Nuclear Regulatory Commission (NRC) inspections conducted on October 3 through 7, 1994, at the facilities of Fuel Operations-Windsor (CENFO), Engineering Operations, and Field Services in Windsor, Connecticut, and on October 17 through 21, 1994, at the Nuclear Fuel Manufacturing-Hematite (CENFM) facility in Hematite, Missouri, it appears that certain of your activities were not performed in accordance with NRC requirements.

- A. Section III.3, "Design Control," of the Asea Brown Boveri (ABB) Combustion Engineering Nuclear Operations (ABB CENO) Topical Report, document CENPD-210A, Revision 7A, approved by the NRC on February 17, 1993, as meeting the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B to 10 CFR Part 50), hereafter referred to as the "QA topical report," required, in part, that the procedures cover the total design process to assure that design interfaces among participating organizations are identified and controlled, and that design work be verified or checked for adequacy.

System 3, "Design Control," of the ABB CENO Quality Assurance Manual, QAM-100-NF, First Edition, Revision 2, dated June 1, 1994 (QAM-100-NF), required, in part, that the cognizant engineer shall review incoming design information to ensure that it shows evidence of approval and is adequate for the design effort; and that where the independent reviewer uses design review as the method of design verification, assumptions necessary to perform the design have been adequately described and are reasonable, or, where necessary, are identified for subsequent reverification when the detailed activities are completed.

Contrary to the above, ABB CENO failed in the following instances to implement established procedures for verifying or checking the adequacy of design and establish adequate measures for the control of design interfaces (94-01-01):

- (1) For the core reload specification data from Baltimore Gas & Electric Company (BG&E) for its Calvert Cliffs

Enclosure 1

Unit 1 Cycle 12 core reload, and from Arizona Public Service Company (APS) for its Palo Verde Unit 3 Cycle 5 core reload, Engineering Operations failed to comply with the requirements of QCP-3.6, "Design Interface Control," of QCM-101-NF, "Quality Control Manual, Fuel Engineering and Fuel Development," Revision 1, dated November 15, 1993, (QCM-101-NF). Specifically, Engineering Operations failed to comply with the requirements of QCP-3.6 by not assuring the identification of (a) the quality program under which the licensees' core reload specification data was verified and (b) the verification status of the licensees' core reload specification data. For the Palo Verde Unit 3 Cycle 5 reload design, Engineering Operations also failed to follow the interface procedure documented in QC 93-01, "Quality Plan for Activities Performed under the Arizona Public Service/ABB Engineering Partnership Agreement." Since QC 93-01 also imposed the requirements of QCP-3.6, Engineering Operations did not comply with the requirements of QC 93-01 and QCP-3.6 in that it failed to assure the identification of (a) the quality program under which the APS design specification data was verified, and (b) the verification status of the APS design specification data.

In addition, Engineering Operations design interface activities and controls were based on the assumption stated in QCP-3.3, "Design Input," of QCM-101-NF, which provides, in part, that customer fuel reload specification data may be referenced as input and assumed to be correct. Engineering Operations design interface activities and controls were based on the assumption stated in QCP-3.3 and, therefore, ABB CENOs design interface activities and controls did not ensure the information in the core reload specification data was consistent with the existing core analysis and safety analysis calculations for which ABB CENO is responsible. (Report paragraph 4.3.3.2(1))

- (2) The physics biases and uncertainties data, generated by computer code, "ROCS/NEM," and used in the Palo Verde Unit 1 Cycle 4 reload safety analysis, were not
 - (a) validated data when it was issued to APS on December 12, 1991, in reload safety analysis V-91-371 (the data was eventually validated after September 1992),
 - (b) identified in the reload safety analysis as unvalidated data as required by QCP-3.10, "Other Design Documents," of QCM-101-NF, and
 - (c) identified by Engineering Operations during the author's review and the independent reviewer's final verification of the reload design analysis before the ROCS/NEM biases and

uncertainties were released to and used by APS, as required by QCP-3.2, "Design Analysis," and QCP-3.4, "Design Document Verification," of QCM-101-NF. (Report paragraph 4.3.3.3)

- (3) The physics analysis report, "Physics Justification of TCR on 14X14 Grid Pin Pitch," used to support BG&E's Calvert Cliffs Unit 1 Cycle 12 core reload fuel mechanical design, included a neutronics analysis to determine the effect of increasing spacer grid dimensional tolerances beyond the current limits. The neutronics analysis did not document an adequate justification for borrowing the rod-bowing penalty to accommodate the increased grid spacer tolerances as required by QCP-3.2 of QCM-101-NF; which required that the methods used in an analysis be documented and justified. (Report paragraph 4.3.3.5)
 - (4) All five coefficient input errors to the computer code, "CECOR," for the Calvert Cliffs Unit 1 Cycle 11 reload and the Cycle 12 CECOR coefficient input error were missed by Engineering Operations during its review and final verification of the reload design analysis as required by QCP-3.2 and QCP-3.4 of QCM-101-NF. In addition, Engineering Operations had not established an input specification for the CECOR code inputs for Calvert Cliffs Unit 1 Cycle 11 and Cycle 12 core reload CECOR coefficients as required by QCP-3.13, "Computer Software," of QCM-101-NF. (Report paragraph 4.3.3.7)
- B. Section III.16, "Corrective Action," of the QA topical report required, in part, that conditions adverse to quality be documented and reported to cognizant parties; that corrective action be identified and implemented in a timely manner; and that procedures require identification of cause, corrective action to prevent recurrence, and verification of implementation.

System 16, "Corrective Action," of QAM-100-NF, required, in part, that conditions adverse to quality shall be identified promptly and corrected as soon as practical; that the cause of the condition be investigated and appropriate corrective action taken to preclude recurrence; and that the adverse condition, the cause or causes, and the corrective action taken be documented and reported to appropriate levels of ABB CENO management.

Contrary to the above, ABB CENO failed in the following instances, where significant conditions adverse to quality existed, to assure that the cause of the condition was determined; the adverse condition was promptly identified and corrected; or the corrective action taken precluded

repetition. In other instances cited below, ABB CENO failed to establish adequate measures to assure that conditions adverse to quality were promptly identified and corrected (94-01-02).

- (1) Zirconium (Zr) alloy, Zr4, fuel clad tubing lot DHS12R supplied to CENFO by Sandvik Special Metals Corporation (SSM) of Kennewick, Washington, was reworked by SSM because the material lots initial burst/elongation test specimens failed the circumferential elongation requirements as the result of, according to SSM, microcracks on the outside surface of the tubing. CENFO approved the lot of fuel clad tubing for use and SSM shipped the lot to CENFM without determining the cause of the microcrack condition and submitting to CENFO its assignable root cause, as required by QCP-516.1, "Corrective Action," of QCPM-500, "Quality Control Procedures Manual, Nuclear Fuel Manufacturing-Windsor," Revision 3, dated September 15, 1994 (QCPM-500). (Report paragraph 4.3.2.1(1))
- (2) Corrective Action Request (CAR) IQS-94-002 that identified that no procedures were established for the verification, control, and use of computer software for certain numerically controlled machines in the cage fabrication factory was issued with a response due date of August 8, 1994. Even though QCP-516.1 of QCPM-500 required the corrective action analyses and planning to be completed and the response returned to the originator within 30 days, the response to CAR IQS-94-002 was not written until September 20, 1994. (Report paragraph 4.3.2.3)
- (3) For the five computer code, CECOR, coefficient input errors reported for Calvert Cliffs Unit 1 Cycle 11, described in A.(4) above, Engineering Operations found the assignable root cause to be that the CECOR code at ABB CENO was different than the version at BG&E. Engineering Operations corrective action was the development of an electronic interface between the BASS and CECOR computer codes. The corrective action from Cycle 11 was the creation of an electronic interface in the form of a template file which was processed by the BASS-P5 code to create the BASS coefficients for the plant. However, the BASS-P5 code did not trap this type of input error because the CECOR coefficient input error reported for Cycle 12, was also the input parameter in error in Cycle 11.

Even though QCP-16.1, "Corrective Action," of QCM-101-NF required the corrective action to identify action taken to prevent recurrence, Engineering Operations corrective action taken for the Cycle 11 input errors

did not prevent a recurrence of an input error for the same input parameter in Cycle 12. (Report paragraph 4.3.3.7)

- (4) QCP-516.1 of QCPM-500H assigned responsibility to the cognizant quality coordinator in each of CENFMs focused factories for the issuance and control of CARs for traceable quality problems identified by P/DNs. However, QCP-516.1 did not address the responsibilities of the cognizant quality coordinators in CENFMs focused factory environment where the corrective action analysis to determine assignable root causes involve interface activities between more than one focused factory. CENFMs failure to address the interface responsibilities between cognizant quality coordinators contributed to the delay in its issuance of CAR H-1-94. (Report paragraph 4.4.1.1(2))
- (5) In October 1991, CENFM identified a malfunctioning pulse/pause timer as the assignable root cause of enrichment variations that resulted from non-homogeneous lots or batches of uranium oxide powder. CENFMs preliminary problem analysis of the fuel pellet isotopic deviations, described in B.(4) above, had determined the assignable root cause was a malfunction in the powder blenders' pulse/pause timer; a recurrence of the 1991 event. Therefore, CENFMs corrective actions taken for the 1991 event were not adequate to prevent recurrence as required by QCP-516.1 of QCPM-500H. (Report paragraph 4.4.1.1(3))

- C. Section III.5, "Instructions, Procedures, and Drawings," of the QA topical report required, in part, that activities affecting quality be prescribed by and performed in accordance with documented instructions, procedures, or drawings, and that these documents include or reference quantitative acceptance criteria for determining that the activity has been satisfactorily accomplished.

System 5, "Instructions, Procedures, and Drawings," of QAM-100-NF, required, in part, that operation sheets (OSs) shall be used to provide detailed instructions which describe the technique, equipment, tooling, and acceptance criteria to be used to perform specific manufacturing or inspection activities, and which may also provide detailed instructions for the accomplishment of other activities affecting quality.

Contrary to the above, ABB CENO failed in the following instances to prescribe certain activities that affect quality and assure that activities that affect quality are accomplished in accordance with instructions or procedures (94-01-04):

- (1) PQ-1C, Revision 7, dated October 12, 1994, "Process Qualification Plan and Report for Enrichment Blending Unit Process (UPIC)," the procedure used to requalify the enrichment blenders, addressed (a) an increase in the qualified enrichment spread and (b) the sampling plan to requalify the blenders after recovering from the problems that caused the isotopic deviations described in CAR H-1-94. However, PQ-1C did not provide guidance for a qualified procedure or establish controlled conditions to assure process qualification as required by QCP-509.2, "Control of Special Processes," of QCPM-500H and QCH-2, "Procedure for Unit Process Qualification," Revision 2, dated June 23, 1994. (Report paragraph 4.4.1.1(4))
- (2) The prestack key plan and rod pushing key plan for Part AEK000, Shop Order CC2NN6, were issued to the rod line without the approval of the cognizant quality organization as required by Operating Sheet (OS) 508, "Document Control," Revision 4, dated October 10, 1994. (Report paragraph 4.4.2.2)
- (3) The pushing table grooves were not being cleaned at the beginning of each shift as required by OS 3320, "Rod Loading - Bundle Assembly," Revision 5, dated June 8, 1994. (Report paragraph 4.4.2.4)
- (4) The pushing table water system was not being changed after approximately every 10 fuel assemblies as required by OS 3320. (Report paragraph 4.4.2.4)

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

Dated at Rockville, Maryland
this 29th day of December, 1994

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

ORGANIZATION: Combustion Engineering, Inc.,
Asea Brown Boveri (ABB) Combustion
Engineering Nuclear Operations (ABB CENO);
Fuel Operations-Windsor (CENFO); and
Nuclear Fuel Manufacturing-Hematite (CENFM)

REPORT NO.: 99900002/94-01; 99900102/94-01

CORRESPONDENCE ADDRESS: Donald E. Allen
Vice President, Total Quality
Combustion Engineering, Inc.
ABB Combustion Engineering Nuclear Operations
P.O. Box 500
Windsor, Connecticut 06095-0500

ORGANIZATIONAL CONTACT: W. Carlton Coppersmith, PhD.
Director, Total Quality

NUCLEAR INDUSTRY ACTIVITY: ABB CENO provides pressurized- and boiling-
water reactor (PWR and BWR) core reload
analysis designs, fuel assemblies, and
nuclear-fuel-related services to the U.S.
nuclear industry.

INSPECTION DATES: October 3 through 7, 1994 - CENFO
October 17 through 21, 1994 - CENFM

LEAD INSPECTOR: for Gregory C. Cavalieri 12/28/94
Steven M. Matthews Date
Vendor Inspection Section (TVIS)
Special Inspection Branch (SIB)
Division of Technical Support (DOTS)
Office of Nuclear Reactor Regulation (NRR)

OTHER INSPECTORS: David H. Brewer, TVIS/SIB/DOTS/NRR
John F. Carew, Brookhaven National Laboratory
Ramon L. Cilimberg, Parámetro, Inc.
Rodney L. Grow, Parámetro, Inc.
Edward D. Kendrick, Reactor Systems Branch
(SRXB), Division of Systems Safety and
Analysis (DSSA), NRR
Kombiz Salehi, SRXB/DSSA/NRR

Enclosure 2

REVIEWED BY:

Gregory C. Cwalina
Gregory C. Cwalina, Chief
TVIS/SIB/DOTS/NRR

12/28/94
Date

APPROVED BY:

Robert M. Gallo
Robert M. Gallo, Chief
SIB/DOTS/NRR

12/28/94
Date

1 INSPECTION BASES:

- Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50)
- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.
- Part 21, "Notification of Failure to Comply or Existence of a Defect," of 10 CFR
- Asea Brown Boveri (ABB) Combustion Engineering Nuclear Operations (ABB CENO), Topical Report, document CENPD-210A, Revision 7A, approved by the NRC on February 17, 1993, as meeting the requirements of Appendix B to 10 CFR Part 50, hereafter refereed to as the "QA topical report"

2 SUMMARY OF INSPECTION FINDINGS:

2.1 Violations

No violations were identified during this inspection.

2.2 Nonconformances

2.2.1 Nonconformance 94-01-01

This nonconformance, described in Sections 4.3.3.2(1), 4.3.3.3, 4.3.3.5, and 4.3.3.7, of this report, identifies 4 instances where the design control activities and established measures of ABB CENO failed to comply with requirements of the QA topical report and the ABB CENO Quality Assurance Manual, QAM-100-NF, First Edition, Revision 2, dated June 1, 1994 (QAM-100-NF).

2.2.2 Nonconformance 94-01-02

This nonconformance, described in Sections 4.3.2.1(1), 4.3.2.3, 4.3.3.7, 4.4.1.1(2), and 4.4.1.1(3), of this report, identifies 5 instances where corrective action activities and established measures failed to comply with the requirements of the QA topical report and QAM-100-NF.

2.2.3 Nonconformance 94-01-03

This nonconformance, described in Sections 4.4.1.1(4), 4.4.2.2, and 4.4.2.4 (2 places) of this report, identifies 4 instances where certain quality activities were not accomplished in accordance with instructions or procedures, and established measures for certain quality activities failed to comply with the requirements of the QA topical report and QAM-100-NF.

2.3 Open Items

2.3.1 Open Item 94-01-04

As described in Section 4.3.2.1(1) of this report, the NRC inspection team (team) was given a root cause analysis report for the failed burst/elongation test specimens from lot DHS12 of Zr4 fuel clad tubing supplied by Sandvik Specialty Metals Corporation (SSM) of Kennewick, Washington. However, during a subsequent inspection of SSM on December 5 through 7, 1994, the team was advised by SSM that its evaluation was not complete and that it would be submitting to ABB CENO a final report with the assigned root cause. The team requested that NRC be notified when ABB CENO has completed action on this item.

2.3.2 Open Item 94-01-05

As described in Section 4.4.1.1 of this report, the team observed ABB CENOs response to an enrichment variation event identified by CENFM when fuel pellet isotopic deviations of varying degrees were identified by rod scanner profiles. The team requested that NRC be notified when corrective action has been completed.

2.3.3 Open Item 94-01-06

As described in Section 4.4.2.1 of this report, the team observed CENFMs response to surface indications on Zr4 fuel clad tubing identified as gouges or pitting on the outside surface. The team requested that NRC be notified when the corrective action has been completed.

2.3.4 Open Item 94-01-07

As described in Section 4.4.2.1 of this report, the team learned that since July 1994 CENFM had observed tubing surface indications of blue, pink, black, and gray ink; brown stains; and white deposits. The team requested that NRC be notified when the corrective action has been completed.

3 STATUS OF PREVIOUS INSPECTION FINDINGS

Previous inspection findings were not addressed during this inspection.

4 INSPECTION FINDINGS AND OTHER COMMENTS

4.1 Background

ABB CENO was formed in October 1993, when ABB Combustion Engineering Services and ABB Combustion Engineering Nuclear Fuel were combined. According to ABB CENO, as part of the reorganization, Engineering and Development in both of the former organizations were combined into a new organization, Engineering Operations. As a result of the reorganization, the major groups within the ABB CENO organization include:

- Total Quality
- Fuel Operations (CENFO)
- Engineering Operations
- Field Services

A sister company, ABB Atom AB, located in Västerås, Sweden, is also involved in the activities of the U.S. nuclear industry, chiefly BWR related. ABB Atom is a subsidiary of Asea Brown Boveri AB of the ABB Asea Brown Boveri Group.

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

During the entrance meeting in Windsor, Connecticut on October 3, 1994, the team met with members of ABB CENO management and staff, discussed the scope of the inspection, and established contact persons for the team within the management and staff of the CENFO, Engineering Operations, and Field Services organizations. On October 7, 1994, the team held an interim exit meeting to outline to ABB CENO management and staff the major concerns identified by the team thus far.

During the entrance meeting in Hematite, Missouri, on October 17, 1994, the team met with members of CENFM management and staff, discussed the scope of the inspection, and established the team's contact persons within the management and staff of CENFM. On October 21, 1994, the team conducted an interim exit meeting to outline to CENFM management and staff major concerns identified by the team during this portion of the inspection.

During both inspection periods described above, the team conducted a performance-based inspection of the ABB CENO CENFO, Engineering Operations, and Field Services organizations through technically directed observations and evaluations of processes, activities, and documentation. The team (a) examined technical

documentation, procedures, and representative records, (b) interviewed ABB CENO personnel, (c) held discussions with ABB CENO personnel, (d) listened to presentations by ABB CENO personnel, and (e) made other observations.

The specific areas examined, the documentation reviewed, and the team's findings are described in this report. The persons who participated in and who were contacted during this inspection are listed in the Appendix to this report.

During its exit meeting in Windsor, Connecticut, on November 2, 1994, with ABB CENO management and staff, the team summarized the inspection findings, open items, weaknesses, and observations.

4.3 October 3 through 7, 1994 — Windsor, Connecticut

During this period of the inspection, the team evaluated ABB CENOs nuclear-fuel-related activities in Windsor, Connecticut.

4.3.1 Total Quality

Established by the authority of the President of ABB CENO, the quality assurance program for nuclear fuel operations is assigned to the Vice President, Total Quality. According to ABB CENO, Total Quality is the organization within ABB CENO that is responsible for the quality assurance (QA) functions of developing and maintaining quality assurance manuals and conducting independent internal quality audits. For nuclear-fuel-related activities, quality control (QC) functions are assigned to CENFOs focused factories in Windsor, Connecticut, and Hematite, Missouri; to Supply Management; and to Engineering Operations. Quality coordinators in the CENFO focused factories, Supply Management, and the Total Quality representative in Engineering Operations (analysis development) and CENFOs Fuel Mechanical Design group have unimpeded access to Total Quality and have stop-work authority from Total Quality.

Total Quality is responsible for preparing and maintaining the quality assurance program, as that program is expressed in the QA topical report and the ABB CENO Quality Assurance Manual, QAM-100-NF, First Edition, Revision 2, dated June 1, 1994 (QAM-100-NF). The following documents, prepared and maintained under the direction of Total Quality and used to implement the QA topical report and QAM-100-NF in the various ABB CENO organizations, were used by the team during its inspection of ABB CENOs nuclear-fuel-related activities:

- QSP-100-NF, "Quality Systems Implementing Procedures," Revision 1, dated June 1, 1994 (QSP-100-NF)

- QCM-101-NF, "Quality Control Manual, Fuel Engineering and Fuel Development," Revision 1, dated November 15, 1993 (QCM-101-NF)
- QCPM-500, "Quality Control Procedures Manual, Nuclear Fuel Manufacturing-Windsor," Revision 3, dated September 15, 1994 (QCPM-500)
- QCPM-500H, "Quality Control Procedures Manual, Nuclear Fuel Manufacturing-Hematite," Revision 2, dated October 1, 1994 (QCPM-500H)
- QAM-100, "Quality Assurance Manual, Nuclear Services," Revision 0, dated November 12, 1994 (QAM-100)
- QAM-101, "Quality Assurance Procedures Manual," Revision 2 (QAM-101)

4.3.2 Fuel Operations

ABB CENOs Fuel Operations maintains focused factory facilities in Windsor, Connecticut, and Hematite, Missouri.

Fuel Operations-Windsor (CENFO)

The CENFO focused factories facility in Windsor, Connecticut, fabricates components such as upper and lower end-fittings, fuel rod end caps, control element assemblies (CEAs), spacer grids, and grid cage assemblies. CEAs are shipped directly to the licensee, whereas, the end-fittings, fuel rod end caps, and completed grid cage assemblies are shipped to CENFO's facility in Hematite, Missouri. The nuclear-fuel-related activities of CENFO were conducted by the following focused factories and departments:

- Supply Management
- Grid and Cage Manufacturing
- Component Manufacturing
- Fuel Mechanical Design
- BWR Operations
- PWR Projects and Planning
- Total Quality

Fuel Operations, Nuclear Fuel Manufacturing-Hematite (CENFM)

In the CENFM facility uranium hexafluoride (UF_6) is converted to uranium dioxide (UO_2); the UO_2 is pelletized; fuel rods are loaded using components shipped from CENFO; final fuel assembly takes place using grid cages shipped from CENFO; and completed

fuel assemblies are shipped to licensees. The team's evaluation of the nuclear-fuel-related activities of the CENFM facility is described in Section 4.4 of this report.

4.3.2.1 Supply Management

The team selected the following purchase orders (Pos) to evaluate CENFOs procurement of certain materials used in fuel assemblies:

- PO 9491114-D5332, Supplement No. 3, to Teledyne Wah Chang (TWC) of Albany, Oregon, for Zr alloy strip material for spacer grids
- PO 9491170-D5332, Supplement No. 00, to Ulbrich Stainless Steel for Inconel strip material for *Guardian*TM grids for 16X16 and System 80 fuel assemblies
- PO 9392152-D5602, Supplement No. 00, to TWC for centerless ground Zr₄ rod to machine upper and lower fuel rod end caps
- PO 9491063-D5502 to SSM for Zr₄ fuel clad tubing

In each of these Pos, the requirements of 10 CFR Part 21 and QC-14-09, "Quality Assurance Program Requirements for Suppliers to ABB Combustion Engineering Fuel Operations," Revision 6, Class A, were invoked. QC-14-09 is a graded quality program; Class A is intended to meet the requirements of Appendix B to 10 CFR Part 50. The certified material test reports for each PO contained data showing that the materials met all requirements.

(1) Fuel Clad Tubing Material

For fuel clad tubing, the team reviewed the data packages for Zr₄ fuel clad tubing lots and determined that, with the exception of lot DHS12R, the material, represented by the data packages reviewed, met the requirements of CENFOs Pos. The material lots initial burst/elongation test specimens from lot DHS12 failed the circumferential elongation requirement of $\geq 12\%$ with a test value of 7%; the burst/elongation test was used to characterize the biaxial mechanical properties of the Zr₄ fuel clad tubing. According to SSMs request for approval or review 9491063-002, dated June 15, 1994, lot DHS12R, before reworking, apparently contained surface imperfections (described by SSM as microcracks) on the outside diameter (OD) of the Zr₄ fuel clad tubing. After receiving approval to rework from CENFO, SSM removed the surface imperfections from the tubing by belt-abrading the OD surface. After .0004 inches were belt sanded from the OD of the tubing to remove the surface

imperfections, burst/elongation tests was performed on 9 samples of the reworked material. Samples of lot DHS12R (R standing for reworked) met the burst test circumferential elongation requirements with values that ranged from 15.1 to 62.6%.

As stated in its June 30, 1994, memo, to SSM approving the request for approval or review, CENFO approved the reworked fuel clad tubing material on the condition that SSM perform a root cause analysis on the failed specimens. However, SSM shipped the material to CENFM as acceptable for use without performing a root cause analysis. The team noted that the root cause analysis had not been performed, as required by the corrective action requirements of paragraph 5.3.4 of QCP-516.1, "Corrective Actions," of QCPM-500. The team determined that CENFO had approved the lot of fuel clad tubing for use and that SSM shipped the lot to CENFM without determining the cause of the microcrack condition and submitting to CENFO a root cause analysis report. The team concluded that CENFO's failure to comply with QCP-516.1 by not determining the assignable root cause of the microcracks before approving the lot of fuel clad tubing for use was a nonconformance. As a result, instance (1) of Nonconformance 94-01-02 was identified during this part of the inspection.

After the team determined that no root cause evaluation had been performed, CENFO requested CENFM to place a hold tag on the lot of fuel clad tubing material. On October 20, 1994, during its inspection at CENFM, the team was given a root cause analysis report that, according to ABB CENO, was the final report of SSM's analysis.

However, during a subsequent inspection of SSM on December 5 through 7, 1994, the team was advised by SSM that its evaluation of the failed burst/elongation specimens was not complete and that it would be submitting to ABB CENO a final problem analysis with assigned root cause. Some confusion, therefore, appears to exist regarding the status of the analysis report given to the team on October 20, 1994. The team identified ABB CENO's evaluation of the failed burst/elongation specimens to determine assignable root cause as an open item of concern and requested that NRC be notified when ABB CENO has completed its problem analysis of the failed burst/elongation test specimens, identified possible causes, evaluated the possible causes, confirmed the true cause, and identified corrective actions to be taken. (Open Item 94-01-04)

(2) Rod and Strip Material

The team reviewed certain specifications for the materials used in manufacturing order (MO) 5403-94-1156, Supplement 1, to provide fuel assemblies for Southern California Edison Companys San Onofre Unit 2 core reload Batch K. The fuel assembly was prescribed in "Standard Specification for Fuel Assemblies," 00000-FMDE-0100, Revision 08; supplemental purchasing information 1-08; and bill of materials S2K-FMDE-BOM1, Revision 1.

The team reviewed CENFOs materials specifications for Zr4 rod material for upper and lower end caps (00000-PD-304, Revision 3); Zr4 strip for spacer grids (00000-PD-302, Revision 3); and Inconel strip for spacer grids (ASTM B 443-84 and SPI-5R). The Zr4 rod material specification contained requirements for documented agreement between CENFO and its supplier regarding methods and standards for evaluating materials. However, the team found that CENFO had no documented agreement for the methods to be used for analyzing hydrogen, nitrogen, and oxygen gases. The team also found a similar condition existed for Zr4 strip material.

Before the team's interim exit meeting on October 7, 1994, CENFO contacted its suppliers of Zr4 rod and strip material and obtained documentation correcting the condition described above. CENFO gave the team two request for approval or review; 9392494-2 for Zr4 rod and 9394176-1 for Zr4 strip.

4.3.2.2 Grid and Cage Manufacturing

(1) Grid Fabrication

The team observed CENFOs laser welding operations on spacer grids for the San Onofre Unit 2 core reload Batch K. The team evaluated two laser welder qualification reports to verify qualification of the processes being used to laser-weld the 16X16 Zr4 spacer grids. Two qualification procedures were required, because a second, longer, spacer grid was used to give the additional seismic stability required for fuel bundle assemblies supplied to San Onofre. The team concluded that both laser-welding qualification procedures had been developed according to CENFOs procedure and had been reviewed and approved by appropriate disciplines and levels of management. Process variables and laser motion were controlled by a computer-driven controller and an approved program. Applicable travelers and

procedures were present at the workstation, and the work observed by the team was being performed according to procedures.

Qualification of the process for laser-welding *Guardian*TM grids of Inconel material for the San Onofre order was in progress and witnessed by the team.

The team reviewed QC-17-14, "Qualification Procedure for Determining the Measuring Capability of a Non-Contact Measuring Machine - View Basic 12," and determined that QC-17-14 had been reviewed and approved by appropriate disciplines and levels of management. QC inspection of the intersection locations in grid spacers was performed by an electro-optical device referred to as the VIEW measuring machine. The VIEW measuring machine performed a 100% electro-optical measurement of the intersection locations, verified all critical dimensions within each spacer grid cell, and printed a report of the dimensions verified and conformance of the grid spacer to requirements. All grid spacers were examined by this method. The team also reviewed the adequacy of the calibration of the VIEW measuring machine and determined that standards used to calibrate the VIEW measuring machine were properly stored and calibrated and traceable to National Institute of Standards and Technology standards. The VIEW measuring machine was scheduled for calibration annually and its calibration was current.

The team also observed QC personnel visually examining the spacer grids and measuring critical dimensions with vernier calipers and gages. Applicable travelers and procedures were present at the workstation, and the inspection activities observed by the team were being performed according to procedures.

(2) Cage Fabrication

The team observed the fabrication of fuel assembly cages and their packaging for shipment to CENFM for use in fuel assemblies for Baltimore Gas & Electric Company's (BG&E's) Calvert Cliffs Unit 1. Each cage consists of five Zr4 guide tubes, spacer grids, the cast lower end-fitting, wear tubes, plugs, and flanges. The team observed the welding of guide tubes to spacer grids and bottom spacer grids to lower end-fittings and determined the welding to be performed in accordance with written procedures and travelers at the work stations. The team determined that the welders' qualifications and certifications were up to date and met

procedural requirements. Assemblers were observed inspecting spacer grid cells with plug gages according to written procedures and travelers.

4.3.2.3 Component Manufacturing

The inspectors observed the numerically controlled (NC) machining of a metal bar by a qualified machinist. The team observed that micrometers 140012, 340037, 340019, 70006, and 250014 were in use at the machining workstation and determined that these measuring instruments were within the calibration due dates. A subsequent check with the CENFO calibration technician also determined that the calibrations had been performed according to written procedures.

In its discussions with machinist, the team learned that the NC machine is controlled by computer software on a tape which the machinist installs in the machine before starting the operation. The team learned that the use of computer software in the NC machines was not controlled by procedures. The team also learned that this problem was identified as Finding 1 in Total Quality's internal audit report TQ94/MMG/006, dated July 8, 1994. Corrective Action Report (CAR) IQS-94-002 was issued for this finding with a response due date of August 8, 1994.

However, a response to CAR IQS-94-002 was not written until September 20, 1994. The team examined the CAR response and discovered that the corrective action planned would not be completed until late November 1994. QCP-516.1 of QCPM-500 required the corrective action analyses and planning to be completed and the response returned to the originator within 30 days. The team concluded that CENFO's failure to comply with QCP-516.1 and respond to the CAR in the time period required was a nonconformance. As a result, instance (2) of Nonconformance 94-01-02 was identified during this part of the inspection.

4.3.2.4 Fuel Mechanical Design

Although the Fuel Mechanical Design department belongs to the CENFO organization, the results of the team's evaluation of CENFO's mechanical design activities are described in Section 4.3.3.5 of this report.

4.3.2.5 BWR Operations

The activities of CENFO's BWR Operations with respect to nuclear-fuel-related activities were conducted by the following groups:

- Core Engineering
- Safety & Transient Analysis
- Project Management

In addition to evaluating QCPM-500 and QSP-100-NF, which CENFO followed to implement the QA program, the team also evaluated BWR Fuel Operations Quality Plan, BFO-QP-01, "Design Control for Analyses Performed in Support of Fuel Reloads, Licensing Reports, and lead fuel assembly (LFA) Programs," Revision 0, dated June 9, 1994, in conducting its evaluation of CENFOs BWR Operations.

The team listened to a presentation by CENFOs BWR Operations personnel to determine its current activity level, its organizational structure, and its interface and working relationship with ABB CENOs sister company, ABB Atom AB, located in Västerås, Sweden. From the presentation, the team learned that BWR Operations is currently working on the Cycle 12 core reload for Washington Public Power Supply Systems Washington Nuclear Unit 2 (WNP-2). During this discussion, the following key points were made:

- The CENFM facility will make the BWR UO₂ pellets and ship them to ABB Atom for use in manufacturing BWR fuel for the WNP-2 Cycle 12 core reload.
- In accordance with quality plans, reload analysis design work may be performed by either BWR Operations or ABB Atom, using the same computer systems and codes to perform analyses.
- Safety-related BWR work performed at ABB CENO will be reviewed by ABB Atom for technical content.
- Safety-related BWR work performed at ABB Atom will be reviewed by ABB Atom to the requirements of its QA program.
- All licensing submittals will be prepared and submitted by ABB CENO; however, the supporting technical analyses may be performed by either BWR Operations or ABB Atom.

4.3.3 Engineering Operations

The activities of ABB CENOs Engineering Operations with respect to nuclear-fuel-related activities were conducted by the following groups:

- Fuel Development
- Core Analysis Technology
- Systems and Transients
- Nuclear Licensing
- Total Quality

Although the Fuel Mechanical Design group belongs to the CENFO organization, the results of the team's evaluation of this area are described in this Section of the report.

Quality Control Manual, QCM-101-NF, Revision 1, dated November 15, 1993 (QCM-101-NF), used by the team in conducting its evaluation of Engineering Operations, contained the quality control procedures (QCPs) that Engineering Operations followed to implement the QA topical report and QAM-100-NF.

4.3.3.1 Organization and Management

The performance, interfaces and documentation of the reload safety analysis (RSA) processes were reviewed as part of the overall evaluation of the ABB CENO organization and management. The RSA is managed by an assigned reload project engineer who has the responsibility for producing the project deliverables and insuring that the RSA conforms to the QCPs specified in QCM-101-NF. Engineering Operations employs a matrix-like management scheme in which the managers of the technology groups also manage specified project teams that perform the RSA under the direction of the reload project engineers. As an example, the manager of the Core Analysis Technology group also manages several project teams that provide the Reload Analysis Reports (RARs) for a group of licensees. In addition, an assigned project manager provides the interface with the individual licensee.

The individual component RSAs for the RARs are performed by engineers in the technology groups that have the responsibility for the specific analysis being performed (e.g., the CEA analysis is performed by an engineer in the Systems and Transients analysis group). The computer codes and methods developers act as consultants to the project. The QCPs under which the RSAs were performed were specified in QCM-101-NF.

4.3.3.2 Licensee Interface

(1) Groundrules

In discussions with the project managers concerning the organizational interfaces both within Engineering Operations and with the licensee, the team noted that the design and operational data that provided the basis and input for the RSA was developed iteratively with the licensee over a period of time. The resulting "groundrules" document specifies the plant conditions and basic data that defines the plant analyzed and documented in the RAR.

The team concluded, therefore, that the groundrules data underlies and is fundamental to the RSA. Changes in the

groundrules data can substantially change the results and consequences of the RSA.

The team also evaluated the interface between Engineering Operations and licensees with regard to the process of reviews and approvals of (a) technical change requests (TCRs), initiated by CENFO or an external supplier, to document requested changes to technical requirements in a MO or a PO, and (b) deviation from contract requirements (DCRs), initiated by CENFO or an external supplier to request acceptance of materials, equipment, or services which do not meet requirements in a MO or a PO. The team noted that neither QCP-3.14, "Technical Change Requests," nor QCP-7.4, "Deviation from Contract Requirements," specifies requirements for the licensees' review or approval of these processes. Additionally, QCP-503.1, "Design Control," requires, in part, that the basis for a design change request should be submitted to the appropriate project manager, who will decide whether the change has been justified for the contract and whether licensee concurrence is needed. The team noted that there is no documented procedural guidance by which the project manager decides whether licensee concurrence is needed. However, this situation appears to be adequate for the current fuel reloads (i.e., ABB CENO-designed fuel reloads for ABB CENO-designed plants). The team concluded that, for future applications (e.g., mixed cores, BWRs, and other PWR-designed plants), a set of standards for licensee concurrence would formalize the process and reduce the potential for errors. The team concluded that Engineering Operations lack of establish guidance by which the project manager decide the need for the licensee to concur was a weakness in Engineering Operations reload design process.

In reviewing the BG&Es groundrules document, "Calvert Cliffs Unit 1 Cycle 12, Final Groundrules," dated August 27, 1993, the team noted that the plant data was not validated. The team also reviewed Arizona Public Services (APSS) reload specification documents for its Palo Verde Unit 3 Cycle 5, Section 5, "Unit 3 Cycle 5 Groundrules Document," which serves as a source of design input for the reload analysis and it also was not validated. The reload specification documents for both Calvert Cliffs Unit 1 Cycle 12 core reload, and Palo Verde Unit 3 Cycle 5 core reload were transmitted from the licensees to their respective Engineering Operations project managers. The team also reviewed quality plan (QP) 93-01, "Quality Plan for Activities Performed Under the APS/ABB Engineering Partnership Arrangement," Revision 0, dated July 15, 1993, and found that QCP-3.3, "Design Input," and QCP-3.6, "Design Interface Control," of QCM-101-NF apply to QP 93-01.

QCP-3.6 requires that (a) requested information identify the QA program under which the information is to be verified and (b) the status of verification of transmitted information be identified. In addition, QCP-3.3 provided that groundrules data which is documented and traceable "may be assumed to be correct."

On the basis of its evaluation of the reload specification documents described above, the team determined that Engineering Operations, for the Calvert Cliffs Unit 1 Cycle 12 core reload, and Palo Verde Unit 3 Cycle 5 core reload, had not complied with the requirements of QCP-3.6 and identified the QA program under which the reload specification documents were verified. The team also concluded that Engineering Operations had not complied with the requirements of QCP 3.6 for transmitting the information in a controlled and documented manner including the verification status. In the case of the reload specification documents for both Calvert Cliffs Unit 1 Cycle 12 and Palo Verde Unit 3 Cycle 5, the team could not find evidence that the design input data was documented and verified in a controlled manner suitable for data used as design input. The team was told by Engineering Operations personnel that they assumed the data to be correct, as provided for in QCP-3.3.

The groundrules documents reviewed did not document review, modification, verification, approval, and control in a way necessary to ensure that the groundrules documents reflect the current, licensed condition of the plant. Although the responsibility to ensure the adequacy of the groundrules documents is shared with the licensees, Engineering Operations design interface activities and controls were based on the assumption stated in QCP-3.3 that groundrules data may be assumed to be correct. Therefore, Engineering Operations design interface and controls were not adequate to ensure that the information in the groundrules document was consistent with existing core analysis and safety analysis calculations for which ABB CENO is responsible.

The team concluded that Section 3.4.3 of QCP-3.3 is not adequate to ensure the information in the groundrules document was consistent with existing core analysis and safety analysis calculations; the assumption that groundrules data is correct could produce errors in the reload analysis. The team also concluded that the instructions of QCP 3.6 were not being followed by Engineering Operations in its handling of the groundrules data and in the case of Palo Verde Unit 3 Cycle 5, the interface procedure QP 93-01 also was not followed by Engineering Operations.

Recognizing that the groundrules information is critically important and that the use of invalid groundrules data will invalidate the RSA, the team concluded that the lack of adequate procedures to assure the validation of the reload design groundrules data is a nonconformance. As a result, instance (1) of Nonconformance 94-01-01 was identified during this part of the inspection.

(2) Strengths

The team observed that a significant part of the Palo Verde Unit 3 Cycle 5 reload analysis was performed by APS nuclear fuel personnel. The team discussed this with Engineering Operations engineers to determine the impact on interface concerns and the quality of the work. The team also examined training records for the APS personnel and calculation files which were prepared by APS and reviewed by Engineering Operations personnel.

The team was told that for the first reload that was jointly analyzed by APS and Engineering Operations engineers, two APS engineers were in residence at ABB CENO for 6 months. For the next reload, APS engineers performed the work at APS with Engineering Operations engineers reviewing the work at APS or at ABB CENO. In one case, the Engineering Operations engineer performed an independent duplicate calculation and then used the duplicate analysis to validate the APS results. The team concluded that the joint engineering effort for the Palo Verde Unit 3 Cycle 5 reload analysis was a strength.

(3) Licensee Weaknesses

The team determined that a potential source of error in the engineering reload analysis process is Engineering Operations assumption that the groundrules data is correct. The team discussed the source of this data with Engineering Operations engineers and determined that this data must come from the licensee because the licensee best knows the reactor changes from cycle to cycle. The team also observed that the data contained in the output from the RSA and the RAR was documented and verified in a unique calculation folder. The output was documented and traced to its source as part of the verification, and the calculation file contained a design analysis title page. The RAR will be incorporated by the licensee in the facility change evaluation.

The team identified certain weaknesses, as described above, in the licensees' design interface activities and pointed out that Criterion 3, "Design Control," of Appendix B to

10 CFR Part 50 required, in part, that established measures include procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

Because the reload specification documents supplied by licensees contain the plant parameters that nuclear fuel vendors will use to perform the safety evaluations of fuel reloads for the licensee's reactor core, the licensee's reload specification documentation must reflect the current, licensed condition of the plant and current information and guidelines presenting cycle- or unit-specific information. To ensure the information presented in the reload specification documentation is consistent with anticipated plant configuration, operation, and licensing impact (e.g., plant technical specifications and applicable portions of the updated final safety analysis report), licensees may wish to review their methodology for the review, modification, approval, and control of reload specification data.

4.3.3.3 Reload Analysis Report

Engineering Operations performs a detailed and complete RSA for each reload core. The analysis was documented in the RAR that was given to the licensee. Core reload data and information was given to the licensee in the form of hard copy as well as computer disk. The RSA was performed with methods that were documented in topical reports that had been approved by the NRC staff. According to Engineering Operations, methods submittals were made when a fundamental change in methods was introduced or based on the nature of the uncertainties changes. Engineering Operations stated that these issues were discussed at periodic meetings with the staff. Analyses were performed for both analog and digital plants; the digital plants generally require a more complex analysis. During discussions with the team, Engineering Operations pointed out they were not presently providing fuel for cores having multiple fuel vendors ("mixed cores").

The RSAs were a very large and complex analyses efforts performed by many technical groups within the Engineering Operations organization. The analysis began with a three-dimensional coupled neutronics/thermal-hydraulics core analysis performed with the "ROCS" simulator computer code. A series of steady-state core simulations were then performed which provide input to the core limits, set point, and transient analyses. These analyses include steady-state thermal-hydraulics, CEA ejection, steamline break, loss-of-coolant accident (LOCA), CEA withdrawal, and the determination of the "CECOR" computer code input data. Each of these analyses may require several weeks to perform, and an analysis may receive input from as many as four related

analyses. The evaluations are performed according to a detailed and well-coordinated schedule to allow timely input to the component evaluations and optimize the milestone schedule. The licensee is given the RAR report and the results and data from the component analyses.

As a result of discussions with Engineering Operations staff concerning methodology submittals, the team raised a question concerning the required validation of the nodal expansion method (NEM), which has recently been incorporated in the ROCS simulator code. The ROCS/NEM bias and uncertainties associated with this new approach were documented in ABB CENO topical report CE-CES-129, "Methodology Manual for Physics Biases and Uncertainties," Revision 1-P, dated 1991. Although this report was issued in August 1991, the validation of the ROCS/NEM bias and uncertainties was not performed until after September 1992 and was not released until January 1993 in Revision 2-P of the topical report CE-CES-129. During this period, before the nuclear physics biases and uncertainties were validated, this data was transmitted to APS in the Palo Verde Unit 1 Cycle 4 reload safety analysis (V-91-371) on December 12, 1991¹. This unvalidated nuclear physics data was also released by ABB CENO in its TECHNOTE No. 92-01. The sequence of events concerning the release of this data is presented in Table 1 below.

The ROCS/NEM biases and uncertainties used in the Palo Verde Unit 1 Cycle 4 reload and included in the ABB CENOs December 12, 1991, transmittal were not identified as unvalidated data as required by Section 3.5.2 of QCP-3.10, "Other Design Documents," of QCM-101-NF. This unvalidated data was also not identified during Engineering Operations reload evaluation design review and final verification. The team concluded that Engineering Operations release of the unvalidated ROCS/NEM nuclear physics data and its failure to identify the unvalidated data during its design review is a nonconformance. As a result, instance (2) of Nonconformance 94-01-01 was identified during this part of the inspection.

¹"UIC4 RAR Revision Pages," V-91-371, letter dated December 12, 1991.

**Table 1, Sequence of Events Associated with
the Release of the Unvalidated ROC/NEM Biases and Uncertainties:**

EVENT	DATE
1. Unvalidated ROCS/NEM uncertainties and biases released in the ABB CENO topical report CE-CES-129, Revision 1-P	August 1991
2. ABB CENO transmitted the unvalidated nuclear physics data to Palo Verde Unit 1 Cycle 4	December 12, 1991
3. Unvalidated nuclear physics data released in ABB CENO TECHNOTE No. 92-01	March 9, 1992
4. ROCS/NEM nuclear physics data validated	After September 1992

4.3.3.4 Reload Design Analysis Process

A primary objective of this inspection was to evaluate Engineering Operations reload design analysis process and activities. As part of this evaluation, a detailed review of BG&Es Calvert Cliffs Unit 1 Cycle 12 (CC1-12) reload analysis was performed. The CC1-12 reload design analysis did the following:

- (a) determined the core loading (enrichment and absorber settings),
- (b) developed the groundrules agreement,
- (c) performed the licensing (transient and set point) analysis,
- (d) performed the as-built analysis

The reload evaluations included the fuel assembly and fuel rod mechanical designs, determination of the fuel stored energy and pressure, the LOCA and thermal-hydraulic analysis, analysis of the core and system transients, and the determination of the departure from nucleate boiling and linear heat rate limiting safety system set points and limiting conditions for operation. The reload design analysis results are documented in the licensing report and the core operating limits report.

After initially reviewing the approximately fifteen individual CC1-12 component reload evaluations, the team chose the CECOR coefficient library generation analysis (A-CC1-FE-0047, Revision 1), the CEA ejection analysis (A-CC1-FE-0030), and the

set point analysis (A-CC1-FE-0036) for detailed review because of the complexity of the analyses and the interfaces involved. The team reviewed each of these reload analyses for conformance with the design analysis procedures and requirements of QCP-3.2, "Design Analysis," of QCM-101-NF. These analyses were sampled and found to fully conform with QCP-3.2.

Paragraph (a) 1 of Section 3.2.5, "Performing and Documenting the Analysis," of QCP-3.2 required that the assumptions made in the RSA be stated. The assumptions section of both the CEA ejection and the set point analysis did not identify the assumptions made in these analyses. Since each of these large, complex analyses involved many assumptions, the team concluded that these analyses did not satisfy the requirements of Section 3.2.5 of QCP-3.2, and that this failure to explicitly state the assumptions was a weakness in Engineering Operations documentation of the RSA.

To determine if the quality of the reload analyses varied significantly from plant to plant, the team compared (a) the CC1-12 and the Palo Verde Unit 3 Cycle 5 (PV3-5) (A-PV3-FE-0066-CT, Revision-00) CEA ejection analysis and (b) the CC1-12 and Florida Power and Light Company's St. Lucie Unit 2 Cycle 8 (A-SL2-FE-0035) set point analysis. These comparisons indicated that the quality of all these analyses was essentially the same.

The final step in the RSA was the release of the RAR. The CC1-12 Reload Report² was issued by the project engineer and reviewed in accordance with Section 3.4, "Final Approval," of QCP-3.10.

In reviewing the Engineering Operations reload analysis process, the team discovered that no written procedures defined the step-by-step process for performing the RSA. Although the current system of using the previous cycle reload RSA as a "template" to define the process for the present cycle appears to be adequate for the current fuel reloads (i.e., ABB CENO-designed fuel reloads for ABB CENO-designed plants), the team concluded that, for future applications (e.g., mixed cores, BWRs, and other PWR-designed plants), a set of standardized detailed reload procedures appeared warranted to standardize the analyses, formalize the process, and reduce the potential for errors. The team concluded that Engineering Operations lack of establish detailed design procedures was a weakness in Engineering Operations reload design process.

²"Calvert Cliffs Unit 1, Cycle-12 Reload Design Report for 88 Assembly Reload Design," letter CC1-FE-0050, Revision 1, dated January 28, 1994.

4.3.3.5 Fuel Mechanical Design

The reload batches typically involve changes in the fuel mechanical design. As part of the mechanical design evaluation, these design changes were compared to a set of fuel design criteria (FDC) that have been established to define an envelope of design acceptance. When design parameters fall outside the FDC, a design-specific evaluation was performed. The design criteria compliance summary (DCCS) report documented the evaluation of the fuel design changes relative to the design criteria. The fuel design engineering analysis was based on the groundrules document which specified the fuel design information and assumptions.

The evaluation of the CC1-12 fuel mechanical design was performed by the fuel Mechanical Design group of the CENFO organization. The team reviewed the documentation supporting this reload analysis. The reload evaluation consisted of approximately 15 individual analyses and/or tests. The documentation of the following analyses was reviewed and found to generally conform with the requirements of QCP-3.2 of QCM-101-NF:

- (a) evaluation of the CC1-12 core reload Batch P plenum spring design
- (b) evaluation of the removal of the upper alumina spacer in the fuel rod design
- (c) evaluation of the Batch N reload fuel assembly upper end-fitting holddown spring and bundle design for incore service
- (d) power distribution transmittals from Engineering Operations to Fuel Mechanical Design

The CC1-12 reload document, "14x14 TIG Welded Zirc Grid Pitch Study," dated November 11, 1993, did not include any validation documentation. This deficiency was pointed out by the team, and Fuel Mechanical Design stated that the data from this document was not actually used in the CC1-12 reload analysis, but was included as supporting information. Nevertheless, the team concluded that the inclusion of unvalidated data in the CC1-12 reload analysis was a weakness in the CC1-12 fuel mechanical design.

The "SIGCREEP" code, version-CW1, was used to determine the guide tube growth and shoulder gap for the CC1-12 reload design. The team reviewed the SIGCREEP code documentation, which included benchmarking against measured data and other associated documentation. The team found this documentation to be consistent with the requirements of QCM-101-NF.

The team performed a detailed review of the report entitled "Physics Justification of TCR on 14x14 Grid Pin Pitch," letter

GM-FE-0125, dated September 29, 1993, which was included in the CC1-12 reload analysis. This analysis provided the neutronics calculations for the effect of increasing spacer grid tolerances beyond current fuel mechanical design criteria. The effect of this design change was to increase the local power peaking and decrease the margin to local peaking and departure from nucleate boiling ratio limits. The analysis described in the above report compensates for this nonconservative local power increase by borrowing conservatism, which was assumed to be available in the rod-bowing penalty.

On the basis of this review, the team concluded that adequate justification had not been given for using the rod-bowing penalty to accommodate the increased power peaking that results from the CC1-12 increased spacer grid tolerances. The team concluded that Fuel Mechanical Designs failure to adequately justify borrowing conservatism constitutes a nonconformance. As a result, instance (3) of Nonconformance 94-01-01 was identified during this part of the inspection.

4.3.3.6 Indoctrination and Training

The qualification and training of the Engineering Operations personnel was included as part of the evaluation of the CC1-12 reload design process and activities. The procedure in QCP-2.1, "Indoctrination and Training," of QCM-101-NF specified the training required of the engineering staff performing the reload analysis. In practice, a junior-level engineer (with a B.S. or M.S. degree and possibly several years of experience in an unrelated area) will collaborate with a senior-level engineer (acting as a "mentor") in performing the analysis and will coauthor the reload analysis report. After gaining several years of experience with the mentor, the junior-level engineer may perform the reload analysis independently. The training records of the Engineering Operations staff that were involved in the CC1-12 reload analysis were reviewed (Forms QCF 2.1-1 and QCF 2.1-2). The training records indicated the courses attended and the self-study material used. However, on all of the (approximately 30) training records examined, no record was kept of the on-the-job training (OJT), although Section 4 of QCP-2.1 required such records to be kept. ABB CENOs lack of documented OJT training was a concern to the team since OJT was the primary means of training the engineering staff and without records of this training, the staff may not be properly assigned to reload tasks.

The current system of Engineering Operations indoctrination and training appears to be adequate for the current fuel reload activities (i.e., ABB CENO-designed fuel reloads for ABB CENO designed plants). However, since the assignment of the appropriate staff to the specific reload tasks can have a

substantial effect on quality, the team concluded that, for future applications (e.g., mixed cores, BWRs, and other PWR-designed plants), the lack of OJT documentation was a weakness in Engineering Operations training process.

4.3.3.7 Error Reports

The team reviewed error reports, root cause and CARS and audit reports from audits recently performed by Total Quality, as internal audits, and by licensees or by licensees participating in the Nuclear Fuel Users Forum (NFUF) external audits. The team reviewed corrective actions to assess their effectiveness. When appropriate, the corrective actions were traced back to the originating error report or audit finding.

The team observed that five input errors were reported for the Calvert Cliffs Unit 1 Cycle 11 core reload CECOR coefficients. A review of the root cause and corrective actions report showed that the root cause of one error was that the CECOR code at ABB CENO (CDC version) was different than the version at BG&E (plant computer version). The intermediate/long-term corrective action was the development of an electronic interface between the "BASS" code and the CECOR code. The BASS code is used for on-line monitoring of thermal margin, and the BASS coefficient data was included in the CECOR input geometry deck for Calvert Cliffs.

The team observed that an error report for Calvert Cliffs Unit 1 Cycle 12 core reload reported that a CECOR input error was made. The team observed that the input parameter that was in error was also in error in the previous cycle. For Cycle 12, the 12th input value on card 102000 was 2.000, which was interpreted as 200% by the BASS/CECOR codes. The correct entry should have been 1.02, interpreted as 2%.

The corrective action for the input errors from Cycle 11 was to create a template file, which the BASS-P5 code processed to create the BASS coefficients for the plant. However, the BASS-P5 code did not trap this type of input error. The addition of a preprocessor code to the CECOR input preparation process was not an effective corrective action. Thus the corrective action from Cycle 11 did not prevent a recurrence of an input error for the same input parameter in Cycle 12. The team determined that (a) the corrective action from Cycle 11 was ineffective because the root cause had been incorrectly identified and (b) the addition of a preprocessor code to the CECOR input preparation process was not an effective corrective action. The team concluded that Engineering Operations failure to correctly determine the root cause and take effective corrective action for the input errors that were reported for the Calvert Cliffs Unit 1 Cycle 11 and Cycle 12 core reload CECOR coefficients constitutes

a nonconformance. As a result, instance (3) of Nonconformance 94-01-02 was identified during this part of the inspection.

The team reviewed the associated documentation pertaining to these CECOR input problems and discussed this material with Engineering Operations personnel. The verification and validation report for BASS-P5 was also reviewed. All five input errors in Cycle 11 and the Cycle 12 input error were missed by the design analysis and verification process, and the addition of a preprocessor code to the CECOR input preparation process was not effective. The team also determined that Engineering Operations had not established an engineering procedure or guideline for the CECOR input preparation. The team concluded that Engineering Operations failure to (a) identify the five input errors in Calvert Cliffs Unit 1 Cycle 11 core reload and the Cycle 12 during the design analysis and verification process and (b) establish an engineering procedure or guideline for the CECOR input preparation for the Calvert Cliffs Unit 1 Cycle 11 and Cycle 12 core reload CECOR coefficients constituted a nonconformance. As a result, instance (4) of Nonconformance 94-01-01 was identified during this part of the inspection.

4.3.4 Field Services

The activities of ABB CENOs Field Services organization with respect to nuclear fuel are conducted by the following groups:

- Steam Generator & Reactor Services
- NDE (AMDATA) Products & Systems
- Engineering/Products
- Total Quality

QAM-100, "Quality Assurance Manual, Nuclear Services," Revision 0, dated November 12, 1994 (QAM-100) and QAM-101, "Quality Assurance Procedures Manual," Revision 2, (QAM-101), used by the team in conducting its evaluation of Field Services, contained the quality assurance procedures (QAPs) that Field Services followed to implement the QA program.

4.3.4.1 Fuel Examination and Inspection Equipment

In reviewing the equipment used by Field Services to perform inspection and repair of fuel assemblies (e.g., failed fuel rod detection system, sipping, visual inspection, eddy current examinations, dimension measurements, oxide thickness measurements, crud sampling, gamma scanning, fuel channel inspection, control rod wear inspection, and fuel assembly reconstitution), the team found that, for the majority of fuel handling tools used by Field Services, the design control provisions of QAM-100 and QAM-101 were not applied. The design and construction of these tools were neither reviewed for their

failure modes and effects, nor were the tools considered under a foreign material exclusion program.

While the current design practices for fuel-handling tools appears to be adequate for handling ABB CENO-designed fuel assemblies because of their robust design and construction (i.e., ABB CENO-designed fuel reloads for ABB CENO-designed plants), the team concluded that, for future applications (e.g., mixed cores, BWRs, and other PWR-designed plants), a programmatic design and configuration control effort that considers failure modes and effects and that provides for foreign material exclusion may be warranted to reduce the potential for fuel-handling tool failures that may inadvertently cause the placement of loose parts in the reactor core. The team concluded that Field Services' lack of establish design and configuration control procedures was a weakness in the outage service activities of ABB CENO.

4.3.4.2 Licensee-Performed Audits

Although several licensees procure safety-related site fuel services from ABB CENO, only a few of the licensees that procure these services, in which fuel inspection and repair tools are used, also perform audits of Field Services design and configuration control activities of these tools.

4.4 October 17 through 21, 1994 — Hematite, Missouri

During this portion of the inspection, the team evaluated CENFOS Nuclear Fuel Manufacturing-Hematite (CENFM) nuclear-fuel-related activities in Hematite, Missouri. The major groups and focused factories of CENFM include:

- Chemical and Ceramic Operations
- Assembly Operations
- Regulatory Compliance
- Production Support

QCPM-500H, "Quality Control Procedures Manual, Nuclear Fuel Manufacturing-Hematite," Revision 2, dated October 1, 1994 (QCPM-500H), which CENFM followed to implement the QA program, were used by the team in conducting its evaluation of CENFM.

4.4.1 Chemical and Ceramic Operations

4.4.1.1 Powder Blending

In April 1991, CENFM switched from producing UO_2 powders with the same enrichment as the supplied UF_6 to mixing UO_2 powders with an enrichment different than, but close to, that of the supplied UF_6 (known as custom enrichment blending). In the old method of mixing the powder, the conversion process started with a fixed

UF₆ enrichment, and the resulting powder and pellets had the same enrichments as the initial enrichment of the UF₆. However, with the new method, the enrichment of the manufactured pellets is different from that of the supplied UF₆, and, therefore, the possibility existed to produce enrichment variations in powders that are not completely blended. Such an enrichment variation may occur when the process controls fail to detect a problems in the manufacturing process.

By mixing powders of different enrichments, CENFM could manufacture pellets and fuel rods of a desired enrichment with shorter lead time. This process, therefore, provided greater flexibility for licensees to determine their final power requirements by being able to delay final selection of exact fuel enrichment in response to changes in planned outage schedules; this results in improved efficiency of core reload design.

During this inspection, the team observed CENFMs response to an enrichment variation problem that started in September 1994. From its initial situation appraisal of the enrichment variation, according to CENFM, it appeared that the process controls monitoring the accuracy of the enriched powder did not detect the off-specification enriched pellets until the rod scanner rejected completed fuel rods containing the off-specification pellets.

Between September 16, 1994, and October 18, 1994, CENFMs quality coordinators in the Assembly Operations focused factory issued four problem/disposition notices (P/DNs), identifying off-specification enrichments in four different lots of powder and pellets, as follows:

- P/DN 315 was issued on September 16, 1994, on lot c635ya placing 1135 fuel rods on hold in matrix storage. The downloading of rods on hold was being accomplished per P/DN 342, and travelers had been issued for 74 rods on lot 4280f, 50 rods on lot 4287b, 50 rods on lot 4287c, and 46 rods on lot 4287d.
- P/DN 324 was issued on September 21, 1994, on lot c196sa and placed 1076 rods on hold. P/DNs 336, 337, 338, 339, 340, and 341 were issued to direct downloading of specific rods whose gamma profiles had been evaluated.
- P/DN 335 was issued on October 1, 1994, on lot c639sa and placed 510 rods on hold and directed that all rods be downloaded per P/DN 342 for return of the pellets to the ceramic factory to be recycled into the blending process.
- P/DN 357 was issued on October 18, 1994, on lot c642 and placed 510 rods on hold while the gamma profile on each rod was reviewed to determine disposition per travelers and P/DNs.

The quality coordinators' initial investigation of the enrichment variation problem did not attribute the problem to the powder blending process and, without performing a complete problem analysis, CENFMs staff attributed the problem to an isolated manufacturing event.

On October 17, 1994, over a month after the problem was initially observed, CARs H-1-94 and H-2-94 were issued. Because of CENFMs delay in generating the CARs and completing its problem analysis, four lots of pellets with enrichment variations were manufactured and loaded in fuel rods. At the time of the team's interim exit meeting with CENFM on October 21, 1994, CENFM had identified and isolated over 500 completed fuel rods (each containing about 300 pellets) with off-specification pellets. CENFMs final resolution was to reject all pellets produced from four suspected lots of pellet material.

Evaluating CENFMs response to the powder blending problem, the team formed the following conclusions:

- (1) CENFMs P/DN 315, dated September 16, 1994, reported that fuel pellet isotopic deviations of varying degrees were identified by rod scanner profiles. However, CAR H-1-94 to address the corrective action for the isotopic deviations was not written until October 17, 1994.

Considering the importance of a known and consistent enrichment to fuel operating characteristics, the team believes this delay in issuing a CAR is a weakness.

- (2) QCP-516.1 of QCPM-500H assigned responsibility to the cognizant quality coordinator in each of CENFMs focused factories for the issuance and control of CARs for traceable quality problems identified by P/DNs. However, QCP-516.1 did not address the responsibilities of the cognizant quality coordinators in CENFMs focused factory environment where the corrective action analysis to determine assignable root causes involve interface activities between more than one focused factory.

CENFMs failure to address the interface responsibilities between cognizant quality coordinators contributed to the delay in its issuance of CAR H-1-94, described in (1) above. As a result, instance (4) of Nonconformance 94-01-02 was identified during this part of the inspection.

- (3) In October 1991, CENFM identified a malfunctioning pulse/pause timer as the assignable root cause of enrichment variations that resulted from non-homogeneous lots or batches of uranium oxide powder. CENFMs preliminary problem analysis of the fuel pellet isotopic deviations, described

above, had determined the assignable root cause was a malfunction in the powder blenders' pulse/pause timer; a reoccurrence of the 1991 event.

Therefore, CENFMs corrective actions taken for the 1991 event were not adequate to prevent recurrence as required by QCP-516.1 of QCPM-500H. As a result, instance (5) of Nonconformance 94-01-02 was identified during this part of the inspection.

- (4) The latest revision (Revision 7, dated October 12, 1994) of PQ-1C, "Process Qualification Plan and Report for Enrichment Blending Unit Process (UPIC)," the procedure used to requalify the enrichment blenders, addressed (a) an increase in the qualified enrichment spread and (b) the sampling plan to requalify the blenders after CENFM recovered from the enrichment variation problem that caused the isotopic deviations described above. However, the PQ-1C did not provide guidance for a qualified procedure or establish controlled conditions to assure process qualification as required by QCP-509.2, "Control of Special Processes," of QCPM-500H and QCH-2, "Procedure for Unit Process Qualification," Revision 2, dated June 23, 1994. As a result, instance (1) of Nonconformance 94-01-03 was identified during this part of the inspection.

As of November 2, 1994, ABB CENO had neither completed its problem analysis of the enrichment variation event, as described above, to identify possible causes, evaluate the possible causes, and confirm the true cause, nor had it identified corrective actions to be taken. The team identified ABB CENOs evaluation of the enrichment variation event as an open item of concern and requested ABB CENO to notify the NRC when the corrective action has been completed. (Open Item 94-01-05)

4.4.1.2 Pelletizing

The team monitored the pelletizing process at CENFM. During the inspection, one press machine was out of service to perform a major maintenance; there were two pelletizing press machines, one in each production line. Normally, during return to service, CENFM operating staff would produce various pellets, examining their physical attributes (height, diameter, density) and observe their outer surface. However, there was no procedure to control the qualification of and the return of the equipment to full service.

In addition, the team found that no procedures were established for increases in sampling frequency and evaluation of the pellets subsequent to a major modification to the pellet press. Normally, if there was an increase in sampling frequency, it was

performed on an informal basis, rather than proceduralized. The team was concerned that lack of a specific procedure to invoke increases in sampling frequency to a specific value, could lead to inadequate sampling.

4.4.1.3 Enrichment Verification

The team monitored testing of the powders and pellets. A key device for determining accuracy of the enrichment was the Enrichment meter, (E-meter). This tool was an in-house detecting device used to check on the accuracy of the produced pellets. When the quality of the E-meter results were suspect, CENFM sent a representative sample to an outside laboratory for processing and evaluation. In addition, the outside laboratory was used for certification of each lot of pellets.

The E-meter was used to evaluate the enrichment of the finished pellets. For certification of a lot of powder, sample pellets were sent to an outside laboratory for evaluation and verification of the accuracy of the enrichment. The accuracy of the external evaluation (which used pellets) was greater than the E-meter. Further, powder samples were taken for the E-meter after mixing the powder for each individual lot, whereas, pellets were formed after another mixing step performed batch by batch. This additional mixing further improved the uniformity of the pellets as compared with uniformity of the powder after blending. For this reason pellets had a more uniform enrichment (as determined by the external evaluation) than the powder sample evaluated by the E-meter.

For the enrichment variation event described in Section 4.4.1.1 above, a powder sample failed the E-meter enrichment criterion. A separate sample of pellets was sent to the outside laboratory for verification of the E-meter results. The laboratory's results indicated that the pellets were acceptable. The laboratory's results invalidated the earlier E-meter evaluation of the powder sample. Since the outside laboratory and the E-meter did not evaluate the same sample, the outside laboratory's identification of uniform enrichment of the pellets should not have been interpreted as validating the uniform enrichment of the lot or batch. It appears that the E-meter correctly identified the enrichment variation and that CENFM failed to correctly interpret the E-meter results.

The team found the normal testing of the pellets to be acceptable. Frequent samples of powders and pellets under normal production were tested and the test equipment were properly calibrated. If a batch or a lot of powder or pellets failed the appropriate test criteria, it would be properly dispositioned.

4.4.1.4 Traceability

The team found that traceability of powders by lot numbers, pellets by boats (carriers), and scrap and press fines material met the requirements of QCPM-500H. Traceability of pellets by batch numbers remained identifiable through the sintering process. Traceability was maintained by lot numbers through the Cardex system. Since this is a key facet of CENFMs product quality, various means and process check points were established to isolate suspected products by the smallest increments.

4.4.2 Assembly Operations

4.4.2.1 Fuel Rod Fabrication

The NRC inspectors observed the fabrication of fuel rods for BG&E and Southern California Edison. The operations were performed by trained operators in accordance with approved written procedures and travelers at work stations. Much of the process is controlled automatically by computer terminals which interface with rod line operators at each work station. The inspectors observed that the computer would not accept the badge number of persons who were untrained for the operation being performed at a specific work station.

Discussions with rod line operators revealed that indications had been observed on the surface of Zr4 fuel clad tubing supplied by SSM for use by CENFM. Two of three tubes on hold at station 110 exhibited apparent indications of pitting on the outside surface. Manual Advisory Notice 75, dated October 15, 1994, described the indications as an unknown substance. P/DN 358, dated October 18, 1994, stated that "two fuel tubes have pitted areas on OD of tubes." P/DN 358 refereed the problem to technical personnel at CENFO and SSM for evaluation. The tubes from SSM lot DHU51 will not be used because sections containing the indications will be removed for photographs and metallographic evaluation.

As of November 2, 1994, ABB CENO had neither completed its problem analysis of the surface indications to identify possible causes, evaluate the possible causes, and confirm the true cause, nor had it identified corrective actions to be taken. The team identified this as an open item of concern and requested that NRC be notified when the corrective action has been completed.
(Open Item 94-01-06)

Discussions with the quality coordinator indicated that since July 1994 CENFM had experienced problems with SSM tubing involving blue, pink, black, gray, and brown stains and white deposits on various surfaces. There is evidence that the problems are being addressed as required by the CENFO QA programs, but the corrective action has not been successful in

preventing the recurrence of new but similar indications of nonconforming tubing for fuel cladding. (Open Item 94-01-07)

The rod scanner, nicknamed "Big Bertha," using a Californium source, performs gamma scans of completed fuel rods as an in-process quality check to ensure that the gamma profile of each fuel rod meets process control requirements. The rod scanner automatically rejects rods which exhibit unacceptable gamma profiles, and the computer control automatically issues an advisory notice (AN) on the nonconforming material. A P/DN is issued by the QC specialist to put the material on hold and provide documentation of corrective action to disposition the nonconforming material. An evaluation of the gamma profiles for each rod must be performed to determine whether the rod can be reworked or must be scrapped.

As a result of the enrichment variation event described in Section 4.4.1.1 above, the team observed fuel pellets being unloaded from fuel rods in accordance with written procedures and travelers. Some of the pellets were returned to pellet manufacturing for recycling into the blending process, and some of the pellets were acceptable for loading in new fuel rods.

4.4.2.2 Fuel Rod Prestacking

The team observed a rod line operator performing rod prestacking operations. The fuel assembly had been ordered by BG&E for the Batch N core reload at Calvert Cliffs. Fuel assembly C2N629 was prestacked according to Shop Order CC2NN6, Part AEK000. The prestacking plan required loading 108 fuel rods of Part FHA000 (standard fuel rod with 4.48% enrichment) using template Z5097 and then loading 68 fuel rods of Part FHB000 (burnable poison fuel rod with 4.48% enrichment) using template Z5096. The team observed the rod line operator scan the bar codes on each fuel rod. The team asked the operator to scan a rod of a type inappropriate to the template in use and the computer gave a message that the rod was inappropriate. A team member also tried to enter his visitor's badge number to see if he could be authorized computer access. The computer denied access. The team verified that component identification was complete and accurate and that only properly released components were being used to make fuel assemblies. The team's review of the operator's training records showed that he was trained in accordance with documented procedures for prestacking and that his qualifications were current.

The team also verified that measures were established and implemented to control fuel assembly rod loading patterns. Measures had been established in QCP-505.1, "Manufacturing Instructions, Procedures and Drawings," of QCPM-500H; Operating Sheet (OS) 3300, "Prestacking Rods for Fuel Bundle," and OS 3320,

"Rod Loading - Bundle Assembly," Revision 5, dated June 8, 1994. These procedures had been implemented through a computer system that provided operator guidance, limited operator actions to acceptable responses, and allowed access only to authorized personnel.

The team identified the loading pattern as defined by engineering in Drawing E-FC2-E000-A04. The manufacturing and inspection document was identified as Shop Order CC2NN6, Part AEK000, and was observed to be consistent with the engineering definition. However, the team noted that Shop Order CC2NN6, Part AEK000, did not show evidence of having been reviewed and approved by quality control personnel as required in OS 508, "Document Control," Revision 4, dated October 10, 1994. The team concluded that this was an instance of failing to follow internal procedures. As a result, instance (2) of Nonconformance 94-01-03 was identified during this part of the inspection.

4.4.2.3 Cage Assembly Receiving Inspection

The team observed an assembly operator and a QC technician performing receiving inspection on a cage assembly. The assembly operator was performing his duties in accordance with OS 3501, and the QC technician was performing his duties in accordance with OS 3503. In addition to these operating sheets, Shop Traveler P/N ADY100 was also at the workstation. After the cage assembly had been inspected and found acceptable, documentation so stating was forwarded to production control personnel for entry into the computer system. Having entered acceptable cage serial number information into the computer system, assembly operators were then authorized to tag the cage assembly as released. Only properly released and tagged cage assemblies were authorized for use in fuel assembly operations. Review of the training records of the assembly operator and the QC technician showed that they were trained in accordance with documented procedures for the inspection of cage assemblies and that their qualifications were current.

4.4.2.4 Rod Pushing and Bundle Assembly Operations

The team observed rod pushing/bundle assembly operations to verify conformance with requirements. OS 3320, paragraph 5.1.2, required cleaning of the pushing table at the beginning of each shift. On arriving, the team observed numerous loose particles on the pushing table and further observed that the table had not been cleaned for 4 days.

The team concluded that this lack of cleanliness was an instance of failing to follow internal procedures. As a result, instance (3) of Nonconformance 94-01-03 was identified during this part of the inspection.

OS 3320, paragraph 5.2.1, required the recirculating water used to flush over the assembly during rod pushing to be replaced after approximately every 10 bundles. A log of water changes at the workstation showed a recent interval of 14 bundles. The team concluded that this was an instance of failing to follow internal procedures. As a result, instance (4) of Nonconformance 94-01-03 was identified during this part of the inspection.

In other respects, bundle assembly operations conformed with requirements. Review of the training records of the assembly operators showed that they were trained in accordance with documented procedures for bundle assembly and that their qualifications were current.

4.4.2.5 Upper End-Fitting Installation

The team observed an assembler remove an upper end-fitting assembly from a wooden box and place it on the upper end of a fuel assembly. Four outer posts were inserted through the upper end-fitting assembly and threaded into the openings in the guide tubes. The assembler tightened the outer posts and torqued each one to 50 ft-lbs (± 5 ft-lbs). The team observed calibration data on the torque wrench to be current. These activities were performed in accordance with a shop traveler for 14X14 fuel assembly. Installation was performed in accordance with OS 3340. The shop traveler and the OS were present at the workstation. All activities were performed in accordance with the specified procedures. The team observed crimping operations to secure the joints between the upper end-fitting and the bundle. Crimping activity was performed in accordance with OS 3340. Crimping unit pressure was observed to be 3500 psi (± 50 psi) as required by procedure. Calibration of the pressure gage on the crimping unit was current. All assembly activity was performed according to the documented procedures. Review of the operator's training records showed that she was trained in accordance with documented procedures for upper end-fitting assembly and crimping and that her qualifications were current.

APPENDIX

PERSONS CONTACTED

The NRC staff participating in the inspections of CENFO, Engineering Operations, Field Services, and CENFM; ABB CENOs personnel contacted during both inspection periods; and the personnel attending the final exit meeting are listed below. A bullet (•) indicates that person attended the entrance meetings and a dagger (†) indicates that person attended the interim exit meetings.

October 3 through 7, 1994 — Windsor, Connecticut

Total Quality, CENFO, Engineering Operations, and Field Services:

•	†	Albert, E.L.	Machinist
•	†	Allen, D.E.	Vice President, Total Quality
•	†	Bachman, W.H.	Manager, Fuel Development
•		Baum, J.E.	Fuel Project Manager
•	†	Bennett, W.D.	Manager of Training
•	†	Blatter, A.J.	Consulting Engineer, Fuel Mechanical Design
•	†	Book, M.A.	Supervisor, Set point Analysis
•	†	Bloomquist, G.S.	Manager, Quality Assurance, Field Services
		Bosco, R.N.	Field Services Engineer
		Breckenridge, N.J.	Fuel Project Manager
		Brown, J.A.	Supervisor, Neutronics Analysis
		Burns, W.J.	Product Manager
		Case, L.R.	Assembler
		Chase, M.J.	Records Controller
		Clifford, P.M.	Sr. Nuclear Engineer
		Colflesh, J.A.	Manager, Reactor Services
•	†	Coppersmith, W.C.	Director, Total Quality
	†	Corsetti, L.V.	Supervisor of Core Materials Development
	†	Curtis, P.J.	Vice President, Engineering Operations
	†	D'Amato, L.S.	Quality Coordinator, Grid & Cage Factory
		Davis, R.	Welder
•	†	Donohue, J.P.	Total Quality Engineer, Engineering Operations
		Drumm, D.M.	Mechanical Design Engineer
	†	Ebeling-Koning, D.	Manager, BWR Licensing and Safety Analysis
		Fiero, I.B.	Sr. Consultant

--A-1--

- † Fitzgerald, R.J. Project Manager, BWR Fuel Operations
- † Freeburn, H.R. Sr. Consulting Engineer, Core Materials Development
- † Freeman, R.S. Manager, Grid & Cage Factory
- French, R.N. Assembler
- Gavin, P.H. Core Analysis Engineer
- † Glotzer, M.M. Consultant to ABB CENO
- Harris, W. Manager, Core Engineering, BWR
- Johnson, G.P. Total Quality
- † Junkrans, S.B. Vice President, CENFO
- † Kersteen, G.C. Manager, Components Factory
- † Land, R.J. Manager, Supply Management
- Leland, S.D. Project Manager, Reload Engineering
- Matheny, R.M. Project Manager, BWR Fuel Operations
- Pati, S.R. Supervisor of Core Materials Development
- Peeke, R.E. QC Engineer, Component Factory
- Perez, R. Core Analysis Engineer
- † Rodack, T. Supervisor, Mechanical & Thermal Hydraulics Development
- † Rohan, P.E. Total Quality Officer, Engineering Operations
- Rossi, J.R. Mechanical Design Engineer
- Rotondo, P.L. Manager, Fuel Mechanical Design
- Schon, J. Project Manager, BWR Fuel Operations
- Shapiro, N.L. Sr. Consultant
- † Solury, J.J. Manager, Customer Relations
- Suidek, R.S. President, ABB CENO
- † Toelle, S.A. Manager, Nuclear Licensing
- † Trapp, E.L. Director, Nuclear Fuel Projects (PWR)
- Vaslet, R.J. Supervisor, Supplier Quality
- Wagner, S.G. Supervisor, Reload Design
- Waller, K.E. Welder
- Williamson, E.A. Principal Engineer
- Wiencek, J. Quality Records Coordinator
- Wotus, D.J. Project Manager, (APS) Reload Engineering
- Wyvill, J.R. Director, BWR Operations

U.S. Nuclear Regulatory Commission:

- † Brewer, D.H. Metallurgical Engineer, TVIS/SIB/DOTS/NRR
- † Carew, J.F. Neutronics Specialist, Brookhaven National Laboratory

--A-2--

- † Cilimberg, R.L. Metallurgical and Fuel Specialist,
Parámeter, Inc.
- † Grow, R.L. Neutronics Specialist,
Parámeter, Inc.
- † Kendrick, E.D. Reactor Engineer, SRXD/DSSA/NRR
- † Matthews, S.M. Quality Assurance Specialist,
TVIS/SIB/DOTS/NRR
- † Pettis, R.L. Sr. Reactor Engineer,
TVIS/SIB/DOTS/NRR

October 17 through 21, 1994 — Hematite, Missouri

CENFM:

- † Allen, D.E. Vice President, Total Quality
- † Becquette, J.A. Rod Line Operator
- † Bennett, W.D. Manager, Training
- Black, V.L. Trainee
- Borell, S.G. Manager, Ceramics
- Brockmeyer, J.A. Rod Line Operator
- Butler, K.A. Rod Line Operator
- Coile, J.R. QC Specialist
- Coleman, J.L. Rod Line Operator
- † Coppersmith, W.C. Director, Total Quality
- Corner, R.R. Rod Line Operator
- Fadler, R.L. Trainee
- † Fromm, R.C. Director, Quality Systems
- † Gazaway, T.L. Quality Coordinator
- Hess, P.M. Process Engineering Specialist
- Hubert, P.W. Process Engineer
- † Junkrans, S.B. Vice President, CENFO
- Kondrasiewicz, W.J. QC Technician
- † Luckenbach, J.W. Quality Coordinator, Assembly
Operations
- † Luckenbach, M.J. Supervisor, Manufacturing
Information Systems
- Moore, L.E. Welder
- Nelson, H.W. Quality Engineer
- † Noacik, A.J. Operations Superintendent
- † Noel, L.M. Rod Line Operator
- † Page, G.J. Manager, Assembly Operations
- Reando, W.R. Warehouseman
- Rode, J.A. Consultant to ABB CENO
- † Sharkey, R.W. Manager, Regulatory Compliance
- † Solury, J.J. Manager, Customer Relations
- † Stokes, D.L. Manager, Production Support
- † Theodoro, G.L. Welder

--A-3--

- † Tindall, F.J. Rod Scanner Operator
- † Underwood, D.E. Manufacturing Systems Engineer
- † Uding, G.L. Quality Coordinator
- † Weaver, P.E. Manager, Material Scheduling and
 Accountability

U.S. Nuclear Regulatory Commission:

- † Brewer, D.H. Metallurgical Engineer,
 TVIS/SIB/DOTS/NRR
- † Cilimberg, R.L. Metallurgical and Fuel Specialist,
 Parámeter, Inc.
- † Cwalina, G.C. Section Chief, TVIS/SIB/DOTS/NRR
- † Matthews, S.M. Quality Assurance Specialist,
 TVIS/SIB/DOTS/NRR
- † Salehi, K. Reactor Engineer, SRXD/DSSA/NRR

November 2, 1994 — Windsor, Connecticut

ABB CENO - Final Exit Meeting:

Allen, D.E.	Vice President, Total Quality
Bachman, W.H.	Manager, Fuel Development
Colflesh, J.A.	Manager, Reactor Services
Coppersmith, W.C.	Director, Total Quality
Curtis, P.J.	Vice President, Engineering Operations
Junkrans, S.B.	Vice President, CENFO
Kersteen, G.C.	Components Factory Manager
Land, R.J.	Manager, Supply Management
Rohan, P.E.	Total Quality Officer, Engineering Operations
Solury, J.J.	Manager, Customer Relations
Suidek, R.S.	President, ABB CENO
Trapp, E.L.	Director, Nuclear Fuel Projects (PWR)
Wyvill, J.R.	Director, BWR Fuel Operations

U.S. Nuclear Regulatory Commission:

Brewer, D.H.	Metallurgical Engineer, TVIS/SIB/DOTS/NRR
Matthews, S.M.	Quality Assurance Specialist, TVIS/SIB/DOTS/NRR

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

December 5, 1994

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

SUBJECT: NRC INSPECTION NO. 99901281/94-01

Dear Mr. Tate

This letter refers to the inspection conducted by Mr. K. R. Naidu of this office on September 22-23, 1994. The inspection included a review of activities conducted at your ABB Service Center facility (ABB) in Cleveland, Ohio. At the conclusion of the inspection, the findings were discussed with you and the members of your staff identified in the enclosed report.

Areas examined during the inspection and our findings are discussed in the enclosed report. The inspector evaluated the program that ABB established and executed to implement the provisions of Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) for reporting of defects and noncompliance, and Appendix B to 10 CFR Part 50. The inspector also observed selected tests being performed on two 480 Volt circuit breakers from Cleveland Electric Illuminating Company's (CEI) Perry Nuclear Power Plant after being overhauled at your facility. The inspector also reviewed the actions taken by your staff to correct adverse findings identified in an audit that CEI had conducted. Within these areas, the inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspector.

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). Specifically, you did not document an evaluation of maintenance problems involving hardened grease in circuit breakers or bypassed disconnect switches in 5HK type circuit breakers. The violation is of concern because it is not possible to determine from records if similar defects or deviations had been identified which may generically affect the operation of safety-related circuit breakers installed at other nuclear power plants.

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 addition, during this inspection the NRC determined that the implementation of your QA program failed to meet certain NRC requirements. Specifically, the inspector identified two examples of inadequate procedures which are contrary to Criterion V of Appendix B to 10 CFR 50. The specific findings and references to the pertinent requirements are identified in the enclosed Notice of Nonconformance.

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 Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96.511.

Sincerely,

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

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

Docket No.: 99901281

Enclosures: 1. Notice of Violation
 2. Notice of Nonconformance
 3. Inspection Report 99901281/94-01

NOTICE OF VIOLATION

ABB Service Company
Cleveland, Ohio

Docket No. 99901281
Report No. 94 01

During an NRC Inspection conducted on September 22-23, 1994, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1992), the violation is listed below:

Section 21.51, "Maintenance and inspection of records" of Title 10 of the Code of Federal Regulations (10 CFR Part 21) requires (a) each individual, corporation, partnership, or other entity subject to the regulations in 10 CFR Part 21 to prepare and maintain records necessary to accomplish the purposes of 10 CFR Part 21, specifically (1) to retain evaluations of all deviations and failures to comply for a minimum of five years after the date of evaluation.

Contrary to Section 21.51, as of September 23, 1994, ABB did not document its evaluations of deviations and failures to comply and maintain them to allow representatives of the commission to inspect those records to ensure compliance. Specifically, there were no records of evaluations for two potential deviations regarding hardened grease in circuit breakers and bypassing of the "On-Off" charging motor switch in 5HK type circuit breakers. (94-01-04)

Pursuant to the provisions of 10 CFR 2.201, ABB Service Company is hereby required to submit a written statement of explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Special Inspection Branch, Division of Technical Support, Office of Nuclear Reactor Regulation within 30 days of the date of the letter transmitting this Notice of 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. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Dated at Rockville, Maryland
this 5 day of December, 1994

NOTICE OF NONCONFORMANCE

ABB Service Company
Cleveland, Ohio

Docket No. 99901281
Report No. 94 01

Based on the results of a U.S. Nuclear Regulatory Commission (NRC) inspection conducted at ABB Service Company (ABB) on September 22-23, 1994, it appears that certain of your activities were not conducted in accordance with NRC requirements as described below.

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

Contrary to the above, the attachment to Revision 2 of ABB Procedure 15.1, "Control of Nonconforming Items" dated May 20, 1991, did not have appropriate qualitative criteria to document evaluations for determining if the defect or deficiency identified in the nonconformance needs a 10 CFR Part 21 report to be filed. (99901281/94-01-03A)

Contrary to the above, the ABB test instruction used for testing low voltage metal-clad circuit breakers for safety-related applications did not specify the maximum acceptable contact resistance value, or alternatively, the maximum acceptable voltage drop at full load for the millivolt drop test on metal-clad low voltage circuit breakers. (99901281/94-01-03B)

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

Dated at Rockville, Maryland
This 5 day of December, 1994.

INSPECTION REPORT

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

ORGANIZATION: ABB Service Company
Cleveland, Ohio

REPORT NO.: 99901281/94-01

ORGANIZATIONAL CONTACT: Mr. E. Link, Quality Assurance Manager
(216) 267 2882

NUCLEAR INDUSTRY ACTIVITY: ABB Service Company services switchgear
manufactured by I.T.E/BBC

CORRESPONDENCE ADDRESS: ABB Service Company
Regional Service Company
5311 Commerce Parkway West
Cleveland, Ohio 44130

INSPECTION CONDUCTED: September 22-23, 1994,

INSPECTOR: 12/5/94
Date
Kamalakar R. Naidu
Vendor Inspection Section
Special Inspection Branch
Division of Technical Support
Office of Nuclear Reactor Regulation

APPROVED: 12/5/94
Date
Gregory C. Cwalina, Chief
Vendor Inspection Section
Special Inspection Branch
Division of Technical Support
Office of Nuclear Reactor Regulation

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

INSPECTION SCOPE: Tests on K-Line breakers, actions to correct
audit findings, 10 CFR Part 21 evaluations, and
implementation of 10 CFR 50, Appendix B in
selected areas.

PLANT SITE APPLICABILITY: Perry Nuclear Power Plant, and other plants with
I.T.E/Gould/Brown Boveri 480 Volt (V), 4160 V,
and 15kV switchgear.

1. INSPECTION SUMMARY

ABB Service Company located in Cleveland, Ohio performed overhauling services on electrical circuit breakers installed at Cleveland Electric Illuminating Company's (CEI's) Perry Nuclear Power Plant (PNPP). The circuit breakers were originally designed and manufactured by I.T.E, which subsequently changed ownerships and became known as I.T.E-Imperial, I.T.E-Gould, Gould-Brown Boveri, Brown Boveri Electric and finally Asea Brown Boveri (ABB). ABB currently manufactures metal-clad low voltage and medium voltage circuit breakers at Florence, South Carolina. ABB has established service centers to repair and service switchgear at several locations in the U.S. including Cleveland. On September 22-23, 1994, the NRC inspector witnessed selected acceptance tests on two K-Line low voltage metal-clad circuit breakers that had been refurbished, reviewed actions taken by ABB to correct audit findings, and evaluated the program that ABB established and executed to implement the provisions of 10 CFR 50, Appendix B, and 10 CFR Part 21 in selected areas.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This is the first NRC inspection of this facility.

3 INSPECTION DETAILS

3.1 Entrance and Exit Meetings

During the entrance meeting on September 22, 1994, the NRC inspector discussed the scope of the inspection and the areas to be reviewed and established the persons to contact within ABB Service Company management and staff. During the exit meeting on September 23, 1994, the NRC inspector discussed his findings and concerns with the ABB Service Company staff identified in Section 4.

3.2 Review of ABB Quality Assurance (QA) Manual

The inspector reviewed the ABB QA manual, Revision 4, dated January 22, 1993, which outlines the program to be adopted by the various ABB service facilities in the U.S. The ABB Cleveland facility developed QA procedures to implement the corporate QA manual. The inspector reviewed the QA manual and the implementing procedures and determined the following:

A current organization chart for the Cleveland facility was not available. The corporate QA manager said that one was being prepared. Pending review of the organization chart, this matter is considered an unresolved item. (94-01-01)

The job descriptions and qualifications of persons performing safety-related activities were not available. The Corporate QA manager said that this documentation is being prepared. Pending review of the job descriptions and qualifications of personnel, this matter is considered an unresolved item. (94-01-02)

Quality Assurance Procedure (QAP) 15.1, "Control of Nonconforming Items," Revision 2, dated December 18, 1991, states, in part, "Any significant defect in the Nuclear safety-related equipment shall be reported to the Corporate QA Manager in order for an evaluation to be made if a 10 CFR Part 21 Report needs to be filed." The procedure did not elaborate how this provision was to be implemented. For instance, the procedure did not attach a checklist or a form in which personnel could document the significant defect observed and forward it to designated individuals within the organization for evaluating the generic applicability. Lack of qualitative acceptance criteria to evaluate a nonconforming condition is considered contrary to 10 CFR 50, Appendix B, Criterion V. (94-01-03A)

3.3 Review of Actions Taken on a Part 21 Report

In a letter dated March 3, 1989, Indiana Michigan Power Company informed the NRC pursuant to 10 CFR Part 21 that their personnel observed failures of closing mechanisms in two BBC Brown Boveri, Inc. type 5HK250 circuit breakers installed in its D.C. Cook Nuclear Plant Unit 1. The cause of failure was determined to be aging and dirt contamination. After cleaning and lubricating the closing mechanism, the breaker functioned properly.

The plant personnel had noticed that on February 27, 1989, a similar breaker had failed close. After cleaning and lubricating the closing mechanism, the breaker functioned properly. During their investigations of the failure of the breakers, plant personnel determined that the breaker manufacturer's instructions contained in I.T.E Instruction No. 1B-8.2.7-2 pertaining to the lubrication of ABB circuit breakers were unclear. Specifically, the problem related to the following lubrication instruction contained in the instruction book for 5HK250 circuit breakers:

All other mechanism parts, bearings, pins, etc. have been lubricated with ANDEROL L757 manufactured by Tenneco Chemical, Inc. Intermediate Division. The circuit breaker requires no lubrication during its normal service life. However, if the grease should become contaminated or if parts are replaced, any lubrication should be done with NO-OX-ID or ANDEROL grease as applicable.

In a letter to the NRC dated March 21, 1989, ABB Power Distribution, Inc., responded to the 10 CFR Part 21 Report filed by the D.C. Cook Nuclear Power Plant concerning the contaminated lubricant on the operating mechanisms of two 5HK250 ABB circuit breakers resulting in their failure to close on command. In this letter, ABB stated that breakers should be periodically checked for cleanliness and if the lubricant is found to be contaminated and dry (hardened and discolored), the grease should be removed with an approved solvent and relubricated. ABB stated that to perform this activity, the circuit breaker has to be disassembled. The ABB letter did not expressly state that the breaker has to be completely disassembled to observe if the grease had been contaminated.

In a letter to the NRC dated October 3, 1994, ABB provided the following clarification:

In performing refurbishing activities on circuit breakers during the past few months, ABB has found that the condition of the lubricant (i.e. the extent of hardening and contamination) cannot be easily determined without complete disassembly of the breaker mechanism. This is due to the fact that some of the mechanism components requiring lubrication are not visible or accessible for inspection without completely disassembling the mechanism of the breaker.

Even though it is not clear from the ABB letter, the ABB personnel confirmed that ANDEROL type grease should be used on the mechanisms. NO-OX-ID type grease should be used on the primary and secondary electrical mating contacts, i.e., the contacts on the circuit breaker which mate with the stationary contacts in the breaker cubicle. The bridge pivot points should also be lubricated with NO-OX-ID type grease. The main and interrupting electrical contacts should not be lubricated with any type of grease under any circumstances. The inspector enquired if ABB personnel advised PNPP to clean and grease the contacts in the stationary cubicle and found that they had not done so in the past, but would advise them in the future.

ABB did not have any documents indicating that it evaluated this condition pursuant to 10 CFR 50 Part 21 to determine if such conditions were prevalent in ABB breakers installed at other operating plants on which they perform service at other locations. ABB's failure to document its evaluation of this potential deviation is considered a violation of 10 CFR part 21.51. (94-01-04)

3.4 Evaluation of a Potential Part 21 Item

The inspector reviewed the as-found conditions documented by ABB service personnel during the 10-year refurbishment of PNPP circuit breakers and observed that PNPP personnel had removed the motor disconnect switch on a 5HK 350 MVA, 1200 ampere circuit breaker, Serial No. 51958B-319064, bolted together the connections, and applied heat shrink tubing to the joint. ABB personnel noticed that perhaps due to non-uniform heat, the insulation on the conductor underneath the heat shrink tubing had melted and the bare conductors had been exposed.

The purpose of the "On-Off" motor disconnect switch is to control power to the motor which charges the closing strings of the breaker mechanism. When the closing springs are charged, the breaker becomes ready to close on demand. With the motor disconnect switch in the "Off" position, the charging motor does not charge the closing springs. For operational safety, the switch is placed in the "Off" position before the breaker is racked into the test position or into the cubicle. The switch is placed in the "On" position after the breaker is in either of the positions. However, when the circuit breaker is racked into the "test" position, with the switch bypassed, auxiliary power is made available to the motor as soon as the secondary disconnects of the breaker and cell connect. The motor will start and charge the closing springs, enabling the breaker to be closed if any of the interlocks

inadvertently fail while the breaker is being racked into the "connect" position. This may result in a severe flash-over when the primary contacts of the breaker meet the live contacts in the cell, endangering the operating personnel and possibly damaging the switchgear. ABB had no documents indicating that this condition was evaluated pursuant to the requirements of 10 CFR Part 21.51. The inspector observed that by removing and bypassing the motor disconnect switch, PNPP could not only endanger the safety of the personnel operating the switchgear, but also the original IE qualification of the circuit breaker. The ABB QA manager informed the inspector that he would document this matter. ABB's failure to document its evaluation of this deviation is considered another example of a violation of 10 CFR part 21.51. (94-01-04)

3.5 Review of Problems With Inadequate Service

During March 10-April 21 and April 22-May 27, 1994, inspectors from NRC Region III conducted inspections at PNPP and documented their findings in Inspection Reports (IRs) 50-440/94006 (DRP) and 50-440/94009 (DRP) respectively. IR 50-440/94006(DRP) stated that, in early 1993, PNPP sent one 4.1 kV and two 480 V circuit breakers to ABB, Cleveland, for refurbishment. ABB reported to PNPP that the "grease hardening process had begun" in the breakers. In March 1993, PNPP sent four 13.8 kV breakers for refurbishment. ABB reported that the non-critical parts of the breaker mechanism had indications of hardened grease; however, the critical portions of the breaker assemblies had sufficient lubrication. The NRC Region III inspectors reviewed the PNPP reports on the breaker maintenance and were concerned about the operability of the breakers. PNPP initiated Condition Report (CR) 94-511 to correct this matter. CR 94-511 stated that PNPP issued purchase order (PO) No. 132717 to ABB to refurbish five breakers. When ABB returned the refurbished breakers, PNPP did not remove the arc chutes and observe whether ABB lubricated the internal mechanisms and, therefore, did not detect that ABB had not lubricated them. On April 18, 1994, NRC Region III inspectors accompanied PNPP personnel during a PNPP audit of the ABB, Cleveland service shop. PNPP personnel identified deficiencies in the implementation of the quality assurance program established by ABB and documented adverse findings in the following four "Vendor Action Requests" (VARs).

VAR 1 identified that ABB had not satisfied the requirements of CEI's PO # 132717 because the requirements to perform general lubrication inspection and overhaul were not forwarded to the shop. CEI required ABB to establish a procedure to define the method of translating and transmitting customer requirements into work documents. Corrective action proposed by ABB was to revise Procedure QAP 4.4, "Sales Order Processing," to provide direction regarding the method of translating customer requirements from PO into internal ABB service work documents. The inspector reviewed QAP 4.4, Revision 1, and determined that Paragraph 3.2.2 requires resolving any inconsistencies between the purchase order requirements defining the scope of work and the ABB quote; any inconsistencies must be resolved before commencing work.

VAR 2 identified that there was no objective evidence to verify that persons performing quality-related activities had received indoctrination and training. ABB proposed to review existing records and compile the records in a central location. The ABB quality assurance manager informed the inspector that the documents related to the indoctrination and training had been retained by different managers in the past. This has now been corrected and now all the documents are being maintained in a central location.

VAR 3 identified that ABB used checklists that had not been reviewed and approved. ABB proposed to revise Procedure QAP 10.1 to require review and approval of checklists. The inspector reviewed Revision 1 to QAP 10.1 and determined that paragraph 3.4 now states, "Inspection shall be performed with checklists, drawings, instruction books or specifications as applicable. Locally prepared checklists (i.e., blank forms) shall be assigned revision numbers and/or issue dates. The service center QA manager shall review and approve such locally prepared checklists/forms, as well as any revisions, prior to use. Approval for use shall be indicated by his/her signature and date on a local document review/approval sheet which will be maintained on file at the service center."

VAR 4 identified that ABB failed to document corrective action taken to prevent recurrence in response to nonconforming conditions identified by CEI personnel on their five safety-related breakers. ABB issued a notice providing clarifications to the definitions used for service.

No adverse findings were identified in this area.

3.6 Observation of Acceptance Tests

The inspector observed ABB test personnel perform final acceptance tests on PNPP K-6005 circuit breakers Serial Nos. 812545A-010188 and 51817D-22-1135 in the presence of an PNPP engineer. The results of the tests were being documented in a checklist that had been prepared and approved. The test equipment being used to perform the acceptance test contained stickers which indicated that the calibration was current. The test results were acceptable.

The inspector observed that ABB did not specify the maximum acceptable contact resistance value, or alternatively, the maximum acceptable voltage drop at full load for the millivolt drop test value on low voltage metal-clad breakers. The inspector informed ABB personnel that the test procedure is incomplete and contrary to Criterion V, of 10 CFR 50, Appendix B because it did not contain acceptance criteria to assure that all important activities have been accomplished. (94-01-03B)

4. PERSONS CONTACTED

ABB Service Company

- * T N. Kelly, Vice President, Total Quality/Supply Management
 - * + J M. Tate, General Manager, North Central Region
 - * + J O. Webb, Quality Manager
 - * + E. Link, Manager, Cleveland Service
 - M. Rice, Product Development Manager
-

- * Individuals present at the entrance meeting on September 22, 1994.
- + Individuals present at the exit meeting on September 23, 1994.

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

December 30, 1994

Mr. Scott Akers Jr., Chief Executive Officer
Cardinal Industrial Products,
Limited Partnership
3873 West Oquendo
Las Vegas, Nevada 89118

SUBJECT: NRC INSPECTION NO. 99901076/94-01

Dear Mr. Akers:

This letter addresses the U. S. Nuclear Regulatory Commission (NRC) inspection of your facility at Las Vegas, Nevada, conducted by Messrs. U. Potapovs and L. L. Campbell of this office November 29 through December 1, 1994, and the discussions of their findings with members of your staff at the conclusion of the inspection. The inspection was conducted to evaluate Cardinal's quality assurance program and its implementation in selected areas such as (1) control of purchased material and services, (2) material and traceability control, (3) supplier audits, (4) commercial grade item dedication, and (5) a review of your program for implementing Part 21, "Reporting Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR).

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

While our review of your procurement and commercial grade item dedication activities indicated that, in several areas, your quality assurance program had established good controls for assuring product compliance to applicable specifications, the inspectors identified two areas where your program failed to meet certain NRC requirements. The first area relates to the selection of an adequate sampling plan for verification of critical characteristics. Specifically, your program does not contain a basis or objective evidence to show that the destructive sampling plan for verifying critical characteristics provides reasonable assurance that the dedicated items meet the applicable procurement document requirements. The second area concerns the adequacy of measures for evaluation and disposition of nonconforming conditions identified during the ASME Code material upgrade and commercial grade item dedication process. In at least one instance Cardinal's mechanical testing of material from a qualified supplier invalidated the supplier's certification, but no evaluation was conducted to determine the root cause of this situation and Cardinal continued to supply material for applications where at least partial reliance was placed on the supplier's certification. 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

Mr. Scott Akers Jr.

-2-

of Nonconformance. We will consider extending the response time if you can show good cause for us to do so.

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

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

Sincerely,

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

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

Docket No. 99901076

Enclosures:

1. Notice of Nonconformance
2. Inspection Report 99901076/94-01

NOTICE OF NONCONFORMANCE

Cardinal Industrial Products
Limited Partnership
Las Vegas, Nevada

Docket No.: 99901076/94-01

Based on the results of an NRC inspection conducted on November 29 through December 1, 1994, it appears that certain of your activities were not conducted in accordance with NRC requirements.

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

Contrary to the above, Cardinal had not established a documented basis to substantiate that its destructive testing sampling plan for verifying critical characteristics provides reasonable assurance that dedicated commercial grade items (CGIs) supplied met applicable procurement document requirements. (Nonconformance 99901076/94-01-03)

- B. Criterion XV, "Nonconforming Materials, Parts, or Components," of Appendix B to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, requires, in part, that measures shall be established to assure that nonconforming items are reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

Contrary to the above, Cardinal had not implemented measures for the review, evaluation, and disposition of material accepted based, in part, on the supplier's certification when Cardinal's tests on this material show this certification to be invalid. (In the example reviewed, the manufacturer's certified impact test results were found to be incorrect). (Nonconformance 99901076/94-01-04)

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

Dated at Rockville, Maryland
this 30th day of December, 1994

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
DIVISION OF TECHNICAL SUPPORT
VENDOR INSPECTION SECTION INSPECTION REPORT

ORGANIZATION: Cardinal Industrial Products
Limited Partnership

REPORT NO.: 99901076/94-01

CORRESPONDENCE ADDRESS: Mr. Scott Akers Jr., Chief Executive Officer
Cardinal Industrial Products
Limited Partnership
3873 West Oquendo
Las Vegas, Nevada 89118-3098

ORGANIZATIONAL CONTACT: Mr. David Z. Hathcock, Vice President
Quality Assurance

NUCLEAR INDUSTRY ACTIVITY: Supplies fastener products and other metallic
materials for use at nuclear power plants

INSPECTION CONDUCTED: November 29 through December 1, 1994

INSPECTOR: Uldis Potapovs 12-29-94
Uldis Potapovs, Senior Engineer Date
Vendor Inspection Section
Special Inspection Branch

OTHER INSPECTORS: Larry L. Campbell, Reactor Engineer

APPROVED: Gregory C Cwalina 12/29/94
Gregory C Cwalina, Chief Date
Vendor Inspection Section
Special Inspection Branch

INSPECTION BASIS: 10 CFR Part 21 and Appendix B to 10 CFR Part 50

INSPECTION SCOPE: To review and evaluate Cardinal Industrial Products
(Cardinal's) quality assurance program and its
implementation in selected areas such as (1) control
of purchased material and services, (2) material and
traceability control, and (4) commercial grade item
dedication.

PLANT SITE APPLICABILITY: All plants using fastener products and other material
supplied by Cardinal.

1 INSPECTION SUMMARY

Inspector Follow Up Items 99901076/94-01-01 and 99901076/94-01-02 were identified and are discussed in Sections 3.4.4.6 and 3.4.5.3 of this report.

Nonconformances 99901076/94-01-03 and 99901076/94-01-04 were identified and are discussed in Sections 3.4.6 and 3.6 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first inspection at Cardinal Industrial Products Limited Partnership (Cardinal).

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance and Exit Meetings

In the entrance meeting on November 29, 1994, the NRC inspectors discussed the scope of the inspection and established interfaces with Cardinal management. During the exit meeting on December 1, 1994, the NRC inspectors discussed their findings and observations with Cardinal management and staff.

3.2 Historical Summary and Description of Facilities

According to Cardinal management, Cardinal was established in May 1993, when Scott Akers, Jr., the current CEO and owner, purchased the company from Hub, Inc. Hub had previously acquired the company in 1987 after the previous organization (Cardinal Industrial Products Corporation {CIPC}) had declared bankruptcy. During Hub's ownership, the company operated as Cardinal Industrial Products, Inc. The location of the manufacturing and warehouse facilities has not changed as a result of these transactions.

The principal items manufactured on site are ferrous fastener products (nuts, bolts, threaded rod) however, Cardinal's scope of supply also includes ferrous and nonferrous bars, castings, forgings, plates, flanges, tubular products, structural shapes, and similar items. Cardinal has been accredited by the American Society of Mechanical Engineers (ASME) as a material manufacturer and a material supplier and holds current Quality System Certificates (QSC) in these areas.

Cardinal has in-house destructive and nondestructive testing capabilities including chemical analyses, tensile, hardness and impact testing as well as magnetic particle, liquid penetrant, and ultrasonic examination capabilities. Dimensional inspections, including thread measurement are also performed at the plant.

3.3 10 CFR Part 21 PROGRAM

3.3.1 Procedures and Implementation

The latest issue of 10 CFR Part 21 along with Section 206 of the Energy Reorganization Act of 1974 and Cardinal's implementing procedure were observed to be posted in several conspicuous locations as required by the regulation. Review of the implementing procedure identified several weaknesses. Specifically:

- In several paragraphs of the 10 CFR Part 21 implementing procedure, the term "defect" was used in place of "deviation". These terms are specifically defined in the regulation and can not be interchanged.
- The procedure did not contain provisions for informing Cardinal's customers of a deviation when Cardinal can not determine whether this deviation is also a defect.
- The procedure did not require documentation of evaluations to determine whether a deviation is also a defect.
- The procedure did not specifically require that the NRC be notified when a defect is determined to exist. The QA manager stated that under these conditions he would follow the 10 CFR Part 21 requirements.

These weaknesses were discussed with the QA Manager who indicated that the implementing procedure would be revised to address these concerns.

3.3.2 Follow-up on 10 CFR 21 report by Pennsylvania Power & Light Company (PPL) concerning cracked ASME Class 1 bolts at the Susquehanna Steam Electric Station.

On April 19, 1994, PPL reported the discovery of several cracked 1 inch ASME Section III, Class 1 hex bolts at the Susquehanna plant. These bolts were manufactured to ASME SA 193 GR.B7 requirements by CIPC from material heat no. 111890 and were obtained from Capitol Pipe & Steel Products Inc. (Capitol) in 1985. The bolts were used in various safety and non-safety applications. The bolt condition was attributed to "quench cracking" that had occurred during the original manufacturing cycle.

PPL reported that they had been in contact with Cardinal regarding the cracked bolts but had not been able to obtain any procurement or quality records because of the company's new ownership and the absence of records from the time when these bolts were manufactured. PPL determined the location and function of each safety related bolt from the suspect heat and conducted a failure mode and effects analysis. No substantial safety hazards were identified. PPL indicated that bolt replacements would be made as required.

As noted in Section 3.2, two changes in company ownership have occurred since these bolts were supplied to PPL. According to Cardinal management, many

records were lost before Hub acquired CIPC after its bankruptcy. During this inspection, Cardinal staff searched their computer files for stock inventory records related to material heat 111890 and failed to locate any information associated with this material.

3.3.3 Follow-up on nonconforming material received by Southern California Edison (SCE)

In 1993, SCE returned several orders of fasteners supplied by Nova Machine Products Corporation (Nova) to the manufacturer after their chemical overcheck analyses identified that this material was outside the applicable specification limits. The material was in the form of ASME SA 194, 1 1/4-7 heavy hex nuts and 2 inch threaded rod, which Nova manufactured from 2 1/4 inch diameter bar (Republic Engineered Steel heat 8094865) supplied by Hub. Nova's tests on the returned material indicated chemical composition within acceptable limits, however, some of the mechanical test specimens failed to meet the required yield strength minimums suggesting a nonhomogeneous heat of material. Because of Hub's prior ownership of CIPC, the inspectors requested a record search to determine if any material from Republic Engineered Steel heat 8094865 was in Cardinal's material inventory. None was found.

3.4 Cardinal Commercial Grade Dedication Program

3.4.1 Methodology

The requirements for Cardinal's commercial grade item (CGI) dedication process are prescribed in Addenda No.1, Section 8.0, "Control of Purchased Material and Services," to the Cardinal Quality System Manual, "ASME Quality System Certificate Holders," First Edition, Revision 8, dated January 8, 1994. Additionally, Cardinal Standard Practices (CSPs) provide requirements for performing and documenting various dedication activities. The following CSPs were reviewed during the NRC inspection:

- CSP No. 4.001, "Purchase Order Review," Revision 6, dated December 29, 1994,
- CSP No. 9.001, "Material Receiving Inspection," Revision 11, dated April 20, 1994,
- CSP No. 10.010, "Testing and Examination of Ferrous and Non-Ferrous Materials," Revision 9, dated July 8, 1994,
- CSP No. 11.001, "Final dimensional Inspection," Revision 8, dated October 10, 1994, and
- CSP No. 11.002, "Final Visual Inspection of Fasteners," Revision 8, dated December 29, 1992.

Cardinal selects critical characteristics for CGIs based on the testing requirements specified by the applicable ASME Boiler and Pressure Vessel Code

(Code), Section II, "Materials," (Section II) or American Society for Testing and Materials (ASTM) material specifications and any other testing requirements specified by the customer. Critical characteristics, in general, include dimensions, workmanship, and, when required by the applicable material specification, chemistry and mechanical properties and other specified tests such as an oxalic acid etch test. The verification of critical characteristics is performed using sampling plans as discussed in Section 3.4.6 of this inspection report. Additionally, the dimensional inspections performed are discussed in Section 3.4.7 of this report.

The identification and verification of critical characteristics are controlled by the Cardinal Procurement Review Sheet (PRS) and a shop traveller, identified as the Customer Manufacturing Record (CMR). The PRS, CMR, along with visual, dimensional, and testing records are collected and reviewed, and serve as the basis for Cardinal to issue a certification statement that the dedicated CGIs meet the applicable material and customer purchase order (PO) requirements.

Cardinal also uses its CGI dedication process to control the supply of products in accordance with the small products exclusion provisions of NX-2610 and the 3/4 inch and less material provisions of NCA-3800 of the ASME Code (see Section 3.4.8 of this report for additional discussion).

3.4.2 Dedication Program Strengths

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

- Cardinal selects critical characteristics based on the testing and material requirements of the applicable material specification and its customers.
- Cardinal personnel performing testing, inspection, and document review activities were knowledgeable about their work and had a positive attitude.
- Cardinal attempts to obtain material certification for each item or lot of items supplied from the manufacturer (qualified and non-qualified manufacturers) and reviews the certification, when received, for conformance with the applicable material specification.
- Cardinal has good in-house chemical, mechanical, and nondestructive testing capability.
- Cardinal performs comprehensive dimensional inspections of fasteners.

3.4.3 Dedication Program Weaknesses

The NRC inspectors concluded that Cardinal's CGI dedication program addresses the essential elements of the dedication process and that sufficient guidance for performing activities such as inspection and testing are provided. However, the NRC inspectors considered the following to be weaknesses in Cardinal's CGI dedication program:

- Cardinal's procedures do not contain provisions for verifying critical characteristics such as material strength by methods other than destructive testing. For example, Cardinal used an undocumented survey/audit to accept stress/strain charts in lieu of testing (see Section 3.4.4.7 of this report). Also, some procedures are not prescriptive in defining the various types of tests to be controlled by Cardinal's sampling plan (see Section 3.4.4.6 of this report).
- Cardinal has no documented bases, (qualitative or quantitative) for its destructive sample plan used in verifying critical characteristics (see Section 3.4.6 of this report).
- When material overchecks such as chemistry and tensile testing are performed to verify critical characteristics, there are no procedural requirements or guidance provided in Cardinal's dedication program for determining the amount that these test results may deviate from those listed on the manufacturer's certification or to require that such deviations be evaluated.
- The NRC inspectors questioned Cardinal's practice of including nonqualified supplier material certifications (stamped "QA Accepted") in documentation packages supplied to customers. These material certifications are only reviewed for compliance to the applicable material specification requirements as part of the initial screening process for the CGIs received. However, based on the "QA Accepted" stamp, the customer could assume that Cardinal had verified the test results reported in these certifications.

3.4.4 Cardinal CGI Dedication Program Implementation

The NRC inspectors reviewed several completed CGI dedications packages to determine if the critical characteristics for the items had been properly identified and verified, and if adequate procedural controls were in place. The NRC inspectors also observed in-process inspection and testing activities and the processing of CMRs. The NRC inspectors reviewed the following completed CGI dedication packages. In addition to the identified technical requirements, all customer POs invoked the requirements of Appendix B to 10 CFR Part 50 and 10 CFR Part 21.

3.4.4.1 CMR No. 54602-8, dated August 24, 1994, was for the supply of 100 heavy hex head nuts, 3/4-10, SA 194, Grade 6, in accordance with Tennessee Valley Authority (TVA) Contract No. P-92NMB-45217B/PO No. 1030568. Cardinal

purchased 125 of these nuts from a nonqualified supplier, B & G Manufacturing Company, Inc., Hatfield, Pennsylvania. Cardinal verified that markings and selected dimensions were correct, and reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements, and performed spectrochemical analyses on 2 of the 125 nuts, a proof load test on 2 of the 125 nuts, and a hardness test on 2 of the 125 nuts. The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification.

Based on the above dedication activities, Cardinal certified that the nuts were in accordance with the requirements of ASME Section II and Section III, Class 2, 1989 Edition, that NCA-3867.4 and NX-2610 applies (see Section 3.4.7 of this report for additional discussion), and that the nuts were quenched and tempered (1100 degrees F minimum). The quenched and tempered condition statement appears to be based on the nonqualified manufacturer's certification and results of hardness testing.

3.4.4.2 CMR No. 51670-19, dated May 24, 1993, was for the supply of 5000 hex head capscrews, 1/2-13 by 2 inch long, A 307, Grade A, in accordance with TVA Contract No. P-92NMB-45217B/PO No. RD348130. Cardinal purchased 5400 of these capscrews from a nonqualified supplier, L & J Fasteners, Hillard, Ohio. Cardinal verified that markings and selected dimensions were correct and reviewed the nonqualified supplier's documentation for these capscrews. The supplier provided a test report generated by INFASCO (Quebec, Canada), the nonqualified manufacturer of the capscrews. Cardinal performed a spectrochemical analyses on 13 of the 5400 capscrews, tensile (wedge test) tests on 13 of the 5400 capscrews, and hardness tests on 13 of the 5400 capscrews.

The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification. Cardinal also contracted A-1 Plating, Henderson, Nevada, to cadmium plate the capscrews. Based on the above dedication activities, Cardinal certified that the capscrews were in accordance with its customer's PO requirements.

3.4.4.3 CMR No. 54511-1, dated August 6, 1994, was for the supply of 10 hex nuts 7/8-9, A 194, Grade 2H, in accordance with Commonwealth Edison Contract No. 503085/PO No. XX176. Cardinal purchased 125 of these nuts from a nonqualified supplier, B & G Manufacturing Company, Inc., Hatfield, Pennsylvania. Cardinal verified that markings and selected dimensions were correct, and reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements, and performed a spectrochemical analyses on 2 of the 125 nuts, a proof load test on 2 of the 125 nuts, and a hardness test on 2 of the 125 nuts. Cardinal also performed 4 additional hardness tests on nuts subjected to an additional heat treatment at 1000 degrees F for 24 hours in accordance with A 194 requirements.

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

3.4.4.4 PO 552058 was issued to Castle Metals (unqualified vendor) for 77 feet of 1 1/8-inch rod to be supplied as American Iron and Steel Institute (AISI) Grade 316 stainless steel. Castle Metals supplied this material in five pieces with Mill Certification from Roldan, SA (unqualified Spanish company). Cardinal tested samples from two of the bars (tensile, chemical, hardness) and accepted all five bars based on satisfactory test results. This material was subsequently used to make threaded rod. Nine three-foot lengths of the threaded rod were later supplied to GPU Nuclear Corp. certified as complying with ASTM 193 B8M.

3.4.4.5 CMR (CIPI Stock) Requisition No. 100584, dated November 14, 1994, was for the supply of 12 heavy hex bolts, 3/4-10, by 3 inch long, SA 325, Type 1, in accordance with TVA Contract No. P-92NMB-45217B/PO No. 1045143. Cardinal purchased 500 of these bolts from a nonqualified supplier, Cordova Bolt, Inc., Buena Park, California. Cardinal verified that markings and selected dimensions were correct, and reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements. The supplier provided a test report generated by NUCOR Fastener, Saint Joe, Indiana, the nonqualified manufacturer of the bolts. Cardinal performed a spectrochemical analyses on 3 of the 500 capscrews, tensile (wedge test) tests on 3 of the 500 bolts, and hardness tests on 3 of the 500 bolts. The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification.

Based on the above dedication activities, Cardinal certified that the bolts were in accordance with the requirements of ASME Section II and Section III, Class 2, 1989 Edition, that NCA-3867.4 and NX-2610 applied (see Section 3.3.7 of this report for additional discussion), and that the bolts were quenched and tempered (minimum 800 degrees F). The quenched and tempered condition statement appeared to be based on the nonqualified manufacturer's certification.

3.4.4.6 CMR 54438-1RA, dated October 4, 1994, was for the supply of 100 stainless steel hex nuts, 3/8 inch-16 (2B), SA F 594, Grade 304, in accordance with Energy PO No. 94-G-74057, Revision 0.

Cardinal purchased 300 of these nuts from a nonqualified supplier, Bell Fasteners, Santa Fe Springs, California. Cardinal verified that markings and selected dimensions were correct, and reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements. The supplier provided a test report generated by NUCOR Fastener, Saint Joe, Indiana, the nonqualified manufacturer of the nuts. Cardinal performed a spectrochemical analyses on 3 of the 300 nuts, tensile (proof load) tests on 3 of the 300 nuts, and hardness tests on 3 of the 300

nuts. The test results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification. Cardinal also performed an intergranular corrosion test (oxalic acid etch test) on one nut. The results of this test indicated an acceptable microstructure of the nut tested. Based on the above dedication activities, Cardinal certified that the nuts were in accordance with its customer's PO requirements.

The NRC inspectors questioned why Cardinal had not verified the microstructure of 3 nuts as required by Section 8.0 of Addenda 1 to the Cardinal Quality System Manual. Cardinal informed the NRC inspectors that error was the result of its procedures not being prescriptive in the area of destructive testing and that the intergranular corrosion test should have been identified as an example of a test controlled by Cardinal's destructive sampling plan. Cardinal also informed the NRC inspectors that it was in process of revising its procedure to clarify which destructive tests are within the scope of its sample plans for destructive testing (NRC Inspector Follow Up Item 99901076/94-01-01).

3.4.4.7 CMR No. 53694-1, dated April 27, 1994, was for the supply of 20 special tube clamps, three directional, Girard Part No. 3/8-1/2 T500SS3D, in accordance with Radnor Alloys, Inc. (Radnor) PO No. RA-26732-NW, with Change Order No. 1, dated March 24, 1994. Radnor's PO, in part, invoked the following quality and technical requirements:

- Clamps are to be stainless steel and meet the requirements of ASME SA-351, Grade CF8, in accordance with ASME Section III, Subsection NF, Class 1, 1977 Edition through Winter 1979 Addenda requirements, including NCA-3800. Radnor's PO also invoked the requirements of Appendix B to 10 CFR Part 50, 10 CFR Part 21, and Baltimore Gas and Electric Specification 10280, Revision 11, dated February 25, 1994,
- Radnor's Change Order No. 1 authorized the clamps to be provided in accordance with the provisions of NX-2610.

Cardinal issued PO No. 8710901, dated April 4, 1994, to Girard Development, Inc. (Girard), a nonqualified supplier, for 25 of the clamps. Cardinal's PO to Girard did not invoke any specific quality controls nor did it invoke 10 CFR Part 21. However, the PO did require, in part, the following:

- Material must be from the same heat lot.
- Vendor must supply dimensional drawings or cut sheets and CMTRs for each lot.
- Trace codes must be stamped on each piece and traceable to the CMTR.
- Material must comply chemically to SA 351, Grade CF8.

- Supplier must provide stress/strain charts for mechanical testing, traceable to the heat code.
- Certification must accompany material and must include Cardinal's order number, trace code, material description, QSC number and expiration date or statement that material has been purchased in accordance with the above referenced manual/procedure, including revision and date, as applicable.

Cardinal performed the following activities to dedicate the commercially procured clamps and to supply them as ASME Code, Section III material.

- Cardinal verified that markings were correct and reviewed the nonqualified manufacturer's material test report for conformance with the material specification requirements.
- Cardinal performed the following dimensional checks on each clamp: a) bolt hole; b) 1/2 inch and 3/8 inch holes; c) 3/8 inch and 1/2 inch spacing; d) thickness; and e) squareness. Cardinal also performed spectrochemical analyses on 2 of 25 clamps and the results met applicable material specification requirements and were, in general, consistent with the nonqualified manufacturer's material certification.

Addenda No. 1 to Cardinal's QA manual requires verification of the physical properties of a dedicated CGI when the applicable material specification identifies such physical properties. The NRC inspectors questioned why Cardinal had not verified the physical properties of the clamps as specified by the material specification for the clamps. Cardinal informed the NRC inspectors that the physical testing was not performed because the clamps received from Girard were too small for machining a tensile test specimen. The NRC inspectors also questioned why a sub-size specimen was not machined and requested Cardinal's justification for not verifying the physical properties of the clamps.

Cardinal informed the NRC inspectors that as an alternative to verifying the clamp's critical characteristic, tensile strength, Girard had been requested to provide stress/strain charts for mechanical testing that were traceable to the heat code of the clamps supplied to Cardinal. The NRC inspectors then questioned Cardinal on accepting the stress/strain charts from Girard, a nonqualified supplier. Cardinal informed the NRC inspectors that it had audited Girard on April 8, 1994, but was unable to fully qualify them. However, Cardinal stated that based on its audit, Girard appeared to have sufficient controls in the area of generating the stress/strain charts and Girard's subcontractor, Southern Tool, Inc., Anniston, Alabama, had adequate controls for material traceability. The NRC inspectors reviewed the results of Cardinal's audit at Girard and further discussed the use of the Girard audit for accepting the stress/strain charts instead of performing testing to verify the physical properties of the clamps.

The NRC inspectors found that there are no provisions in Cardinal's QA manual or procedures to use an audit, commercial grade survey, or other methods as an alternative to performing testing to verify critical characteristics such as chemical and physical properties. Although the NRC inspectors found that the use of the stress/strain charts appeared to be acceptable, Cardinal had not generated any documentation to justify the use of the stress/strain charts as alternative to performing testing. The NRC inspectors considered this to be a weakness in Cardinal's dedication program.

3.4.5 Nuclear Utilities Procurement Issues Committee (NUPIC)

3.4.5.1 NUPIC Audit of Cardinal

During the NRC inspection, it was found that NUPIC had performed an audit of Cardinal on April 26 through 29, 1994. The NUPIC lead audit utility was the Yankee Atomic Electric Company (YAEC) with auditors from Northern States Power and Washington Public Power Supply System. A YAEC technical specialist evaluated Cardinal's CGI dedication activities by reviewing 4 CMRs (CMRs reviewed included products such as bolts, nuts, cotter pins, and round steel tubing). The technical specialist determined that all products on the CMRs were being upgraded/dedicated by performing visual and dimensional inspections and chemical and mechanical tests. The YAEC technical specialist also determined that the critical characteristics were adequate for the 4 CMRs reviewed and that CGI dedication activities were being carried out in accordance with Cardinal's procedures. One minor observation by NUPIC was that the Cardinal QA manual should reference the procedure controlling dimensional and visual inspections performed as part of receiving inspection.

3.4.5.2 NRC Inspector Comments on NUPIC's Evaluation of Cardinal's CGI Dedication Program

The NUPIC audit report states, in part, that Cardinal's dedication program is performed in accordance with Cardinal's approved and accepted procedures and that the critical characteristics for the 4 CMRs reviewed were determined to be adequate. Information, such as the following, was not provided in the NUPIC audit report and may be beneficial to NUPIC members (NRC licensees) in evaluating the use of items, dedicated by Cardinal, in safety significant/critical plant applications.

- Cardinal's dedication program uses sampling for verifying critical characteristics such as chemistry, hardness, strength, microstructure, and dimensional checks. The NUPIC audit report does not discuss the acceptability of the sample size selected for verifying certain critical characteristics using destructive testing methods. The NUPIC audit report does not address the use of sampling by Cardinal (see Section 3.4.6 of this report).
- The NUPIC audit report briefly addresses the upgrading of material by using the CGI dedication process, but does not clearly state that Cardinal supplies and certifies certain size material and products as

meeting NX-2610 and NCA-3800 ASME Section III Code requirements using its CGI dedication program (see Section 3.4.8 of this report). This information may be beneficial to licensees using this size material in critical applications.

3.4.5.3 Correspondence Between Cardinal and NUPIC

During the review of Cardinal's correspondence to Girard, the NRC inspectors reviewed correspondence between Cardinal and NUPIC that addressed NUPIC and Cardinal audits at Girard. The correspondence reviewed indicated that NUPIC had been requested to provide additional information to Cardinal relative to their inspection of Girard. The NRC inspectors informed Cardinal that they had reviewed the correspondence between Cardinal and NUPIC and would follow up on the status of this correspondence (NRC Inspector Follow Up Item 99901076/94-01-02).

3.4.6 Use of Sampling in the CGI Dedication Process

Finished fasteners, purchased commercial grade from nonqualified suppliers and dedicated as basic components, are inspected and tested in accordance with the following sample plans unless a customer specifies another sampling plan.

1. Visual inspection is performed on 100 percent of the material (Reference CPS No. 4.001).
2. Dimensional inspection is performed as follows and is based on Electric Power Research Institute (EPRI) guidelines (Reference CPS No. 4.001). It is noted that Cardinal considers the lot size to be the number of fasteners received from a nonqualified supplier in a single container, bundle, or batch.

LOT SIZE	SAMPLE SIZE	REJECT*
1-40	Entire Lot	1
41-49	45	1
50-55	47	1
56-60	53	1
61-75	59	1
76-100	68	1
101-150	73	1
201-300	78	1
301-500	83	1
501-1200	90	1
1201-3200	125	1
etc.	etc.	1

*Reject lot on one defect per sample.

3. According to Cardinal, destructive testing to verify critical characteristics such as hardness, strength, microstructure, and chemistry are performed on fasteners (bolts and nuts) and other CGIs being dedicated in accordance with the following sample plan (reference: CPS No. 4.001).

LOT SIZE	SAMPLE SIZE	REJECT*
1-280	2	1
281-500	3	1
501-1200	5	1
1,201-3,200	8	1
3,201-10,000	13	1
10,001 and greater	20	1

*Reject lot on one defect per sample.

The NRC discussed the use of Cardinal's sampling methodology in detail during the inspection and asked Cardinal to provide the bases for its destructive sampling plan as providing reasonable assurance that the fasteners not tested had the required material chemistry and strength and other properties required by the applicable material specification. According to Cardinal, this sample plan is based on the ASTM A 325 sample size for the Shipping Lot Sample. Cardinal uses its destructive sampling plan for dedicating CGIs regardless of the size or type of product, whether the CGIs were purchased directly from the manufacturer or through a distributor, the use of single or multiple production lots, or the historical performance of the manufacturer or distributor.

The NRC inspectors expressed a concern to Cardinal that the Shipping Lot Size sample plan provided in A 325 is for the manufacturer of the fasteners and is defined by A 325 as "that quantity of bolts of the same nominal size and same nominal length necessary to fill the requirements of a single purchase order." Also, the NRC inspectors identified that the quality controls in A 325 do not appear to meet the quality requirements of Appendix B to 10 CFR Part 50 and that such a sample size could include more than one production lot or heat of material.

Cardinal informed the NRC inspectors that it had not established a quantitative acceptable quality level or reject level for its sampling plan nor an associated confidence level. Cardinal's destructive sample plan appears to be based on undocumented qualitative factors such as engineering judgement and experience of its personnel in inspecting and testing fasteners, and not on using quantitative statistics. (Nonconformance 99901076/94-01-03).

3.4.7 Fastener Dimensional Inspections

Cardinal typically performs the following dimensional and visual inspections on bolts: a) threads; b) pitch diameter; c) major diameter; d) thread length; e) grip gaging length; f) body diameter; g) width across flats; h) width across corners; i) head height; j) fillet radius; k) runout of bearing surface; l) bearing surface diameter; m) bearing surface height; n) straightness; o) length; p) markings, and q) plating.

Cardinal typically performs the following dimensional and visual inspections on hex nuts: a) threads; b) minor diameter; c) width across flats; e) thickness; f) runout of bearing face; g) position of tapped holes; h) degree of chamfer; i) washer pad diameter; j) washer pad thickness; k) countersink diameter; l) marking, and m) plating.

The NRC inspectors found that the visual and dimensional inspections, including the sample size for performing dimensional inspections for the CMRs reviewed, provided reasonable assurance that the fastener dimensions were acceptable for the dedicated CGIs.

3.4.8 Cardinal's Use of CGI Dedication for ASME NCA-3800 (3/4 Inch and Less Excluded Material) and NX-2610 (Small Products)

Cardinal utilizes their CGI dedication program when supplying ASME Code material exempted from the requirements of NCA 3800 by Section III paragraph NX-2610 unless specifically prohibited by the customer's PO. This approach is described in their ASME Quality Systems Manual.

Paragraph NX-2610 of the ASME Code, Section III permits certain small products to be furnished as ASME Code, Section III, material with a Certificate of Compliance certifying that the material is furnished in accordance with the applicable material specification and the requirements of Section III. However, NX-2610 further requires that for these small products, measures to assure that the applicable material specification requirements and Section III requirements are met shall be provided by a Certificate Holder.

NCA-9000 of the ASME Code, Section III, defines Certificate Holder as an organization holding a valid N, NPT, or NA Certificate of Authorization issued by the ASME. Therefore, where the customer is not an ASME Certificate Holder, Cardinal's application of the NX-2610 exemptions for small products did not appear to fully comply with the requirements of NX-2610.

3.4.9 Observation of In-Process Inspection and Testing

During the NRC inspection, the NRC inspectors observed Cardinal personnel performing inspection, testing, and manufacturing activities. Personnel performing these activities demonstrated proficiency in accomplishing their activities and there were no abnormalities observed.

3.5 Cardinal's Program for Supplying ASME Code Material

3.5.1 Program Scope

Cardinal has been accredited by the ASME as a Material Manufacturer of ferrous and nonferrous bars, threaded fasteners, seamless fittings and flanges, and as a Material Supplier of these items as well as of castings, forgings, plates, tubular products, structural shapes, sheet, and hot rolled wire and rod. The accredited scope of supply also includes the qualification of other material manufacturers and suppliers and performance and certification of materials

testing, nondestructive examination, heat treating, and upgrading of stock material. According to Cardinal's management, fastener products compose approximately 99% of their sales. Items such as flanges, pipe, and plate are not inventoried but are purchased as required by special orders.

Cardinal purchases material from qualified sources or upgrades stock material purchased from nonqualified suppliers by performing and/or subcontracting all testing necessary to upgrade such material. Programmatic controls of these activities are described in Cardinal's ASME approved Quality Systems Manual and several referenced implementation procedures. Their material upgrading process is described in CSP No. 12.001, Revision 3, dated February 27, 1991, "Material Upgrading Process Non-qualified Vendors". This procedure is applicable to processing ASME Section III orders for products exceeding 1 inch nominal diameter and adequately addresses the Code requirements for this class of material. As indicated in paragraph 3.4.8, the inspectors identified that Cardinal's ASME Quality Systems Manual appears to be inconsistent with Section III of the Code in defining the conditions for utilization of small parts exemption (NX-2610) by organizations in the material supply chain.

3.5.2 Program Implementation

The inspectors reviewed selected POs and related documents for material that Cardinal had purchased and upgraded under their Quality System Certificates (QSC). The documentation reviewed included material purchased from suppliers on Cardinal's approved vendor list as well as material procured from nonqualified suppliers.

In reviewing the upgrading of stock material purchased from Castle Metals, the inspectors noted an example where Cardinal apparently did not assure full compliance with their procedure requirements in accepting material certification from unqualified suppliers. Specifically, in response to Cardinal's PO requirement for certification from the stock material supplier that no welding had been performed on the material, such certification was provided by a third party rather than the manufacturer and accepted by Cardinal as meeting the PO requirements. In this case, the material test report was issued by Acindar (foreign company) who manufactured and supplied the material to B.S. Livingston Co. The non-welding certification was in the form of a letter from B.S. Livingston Co. to Hy Alloy Steel Co. and was signed by a sales manager. The material in question was 1 1/4 inch AISI 4140 hexagonal bar (heat 70759) which Cardinal later used to manufacture nuts and certified them as meeting the 1983 Edition of the ASME Code, Section III, Subsection NC.

With the exception of the apparent procedural deviations noted above, based on the sample of documents reviewed, the Code procurement activities appeared to be generally conducted in accordance with the applicable Quality Systems Manual and procedural requirements.

3.6 Transfer and Utilization of Stock Material from CIPC

Cardinal's material stock includes material and finished fasteners that were acquired during the last exchange of the company's ownership. Some of this material dates back to the early 1980's, before the original company (CIPC) declared bankruptcy. In July, 1987, after CIPC's acquisition by Hub, Inc., CIPI (the new company) issued CSP No. 22.001, "Transfer of Material and Associated Documentation Between CIPC and CIPI." This procedure is currently referenced in Cardinal's Quality Systems Manual and contains a checklist of items which need to be verified before the material can be accepted for use in ASME or non-Code safety related applications.

Review of selected documentation packages of this material indicated that several heats or lots of the material were marked "rejected" on the document review checklist. Most of the rejections were based on inadequate material traceability to the manufacturer or the lack of specific documentation. It was noted that these rejections had never been dispositioned although material from these heats/lots is still being used to fill customer orders. According to Cardinal's staff, a CMR is prepared for each order. The CMR identifies all steps required to assure compliance with the customers PO. This CMR assures that all applicable documents and/or tests required to certify this material are obtained or performed. Review of several customer orders indicated that this approach was generally followed.

One lot of such material involved 2 1/4 - 8 heavy hexagonal nuts (heat 08876) which had been procured on PO 001912 from Fast-ER-Tech Trading, Osaka, Japan. The mill test report for this material was issued by Aichi Steel Works while the finished nuts were certified by Hamanaka Nut Co, Ltd. Hamanaka's certification included impact tests showing acceptable results at 0 degrees F and also at -150 degrees F.

This material was reviewed in September 1987 in accordance with CSP No. 22.001 and rejected for inadequate traceability and lack of QA signatures. The material had been used, however, to fill customer orders on several occasions after this review, the latest being June 1994. Review of CMRs associated with these orders indicated that the documentation required by these orders had been obtained, however the "reject" indicated on the CSP checklist had not been dispositioned.

It was noted that CIPI had subcontracted Eagle Mountain Testing to perform additional impact tests on this material in October 1988 apparently as a part of their sampling procedure to fill an order to ASME Code requirements. The tests failed with the impact test results being significantly lower than certified by Hamanaka Nut Co, Ltd. Although these nuts were not used to fill the particular order that prompted these tests, the material was not rejected for failure to verify the manufacturer's certification, and material from this lot was used to fill subsequent orders not requiring impact testing or, in one case, where impact testing was specified at 72 degrees F based on performing a successful retest at that temperature. The inspector expressed concern about

continued use of a heat/lot of material without adequate justification when overcheck of the certified test values produces unacceptable results.

Failure to initiate a nonconforming condition report and to evaluate the basis for continued use of this material was identified as a nonconformance with Criterion XV of 10 CFR 50, Appendix B (Nonconformance 99901076/94-01-04).

3.7 Control of Special Processes

The inspector examined Cardinal's heat treatment facilities and reviewed CSP No. 18.001, Rev. 5, which controls this activity. The inspector noted that this procedure did not provide specific parameters (time/temperature) for several of the heat treating operations routinely performed by Cardinal. According to Cardinal staff, the necessary heat treating parameters are available in uncontrolled notes and tables which are used to program the controls of their heat treatment furnace. In response to the inspector's questions concerning the use of uncontrolled documents, Cardinal's management stated that the critical parameters would be incorporated in CSP No. 18.001. A revision of this procedure containing these parameters was accomplished before the completion of this inspection.

4. PERSONNEL CONTACTED

- Scott Akers, Chief Executive Officer
 - * + David Hathcock, Vice President, Quality Assurance
 - * + Vincent Cortez, Vice President, Operations
 - * + Chuck Mapes, Materials Manager
 - * + Norman Henderson, Laboratory Manager
 - * + Earnest Roberts, Chief Quality Control Inspector
 - * + Greg Keller, Technical Services Manager
 - * + Mike Tiberio, Sales
 - * + Russ Niermayer, Sales
 - + J. Lee Conrad, Quality Assurance Technician
 - + Steven Gauthier, Laboratory Technician
-
- * Attended the Entrance Meeting
 - + Attended the Exit Meeting

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

October 11, 1994

Docket Nos. 52-004 and 99900403

Mr. Patrick W. Marriott, Manager
Advanced Plant Technologies
GE Nuclear Energy
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: NRC INSPECTION REPORT NO. 99900403/94-02

This letter addresses the inspection of your facility at San Jose, California, conducted by Richard P. McIntyre and Billy H. Rogers of the Nuclear Regulatory Commission's (NRC's) Vendor Inspection Branch, Alan E. Levin of the Reactor Systems Branch, Joseph L. Staudenmeier of the Analytical Support Group, Robert A. Gramm and Frederick R. Allenspach of the Performance and Quality Evaluation Branch, and Tim M. Lee of the Office of Nuclear Regulatory Research on June 21 through 23, 1994. The details of the inspection were discussed with you and your staff during the inspection and at the exit meeting on June 23, 1994.

The purpose of the inspection was to determine if activities performed to support the design of the Simplified Boiling Water Reactor (SBWR) and specifically, the Gravity Driven Integral Full-Height Test for Passive Heat Removal (GIRAFFE) test program performed by Toshiba in Kawasaki City, Japan were conducted under the appropriate provisions of the GE Nuclear Energy (GE-NE) 10 CFR Part 50, Appendix B, quality assurance (QA) program as implemented by NEDO 11209-04A, Revision 8, the most recent Quality Assurance Program Description that has been approved by the NRC, NEDG-31831, "SBWR Design and Certification Program Quality Assurance Plan," and NEDG-31836, "SBWR Team Organization and Procedures Manual," prepared for Department of Energy (DOE) Contract No. DE-AC03-90SF18494. The inspection also reviewed GE-NE corrective actions for certain nonconformances identified in NRC Inspection Report No. 99900403/93-01.

Areas examined during the NRC inspection and our findings are discussed in the enclosed inspection report. The inspection consisted of an examination of procedures and representative records, interviews with personnel, and observations by the inspectors. The inspection did not include a review of records and documents that are maintained by Toshiba at the GIRAFFE test facility in Japan.

The results of the inspection indicate that GE-NE failed to ensure that an adequate quality assurance program was in place and properly implemented by Toshiba for the GIRAFFE Phase 1 test activities as would be required for design basis tests by the GE-NE Quality Assurance Program Description and the

October 11, 1994

GE-NE implementing procedures. The team identified that GE-NE had not imposed NQA-1 or any other specific quality requirements in their technical agreement with Toshiba for the GIRAFFE tests. Further, GE-NE quality assurance staff had concluded in a December 1990 review that Toshiba did not have a QA program implemented for SBWR Phase 1 test activities that met NQA-1-1983. In addition, the team determined that GE-NE had not audited or reviewed the Toshiba Phase 1 testing activities during that time period and that Toshiba did not formally document a SBWR QA program until December 16, 1993. Based on the above, it appears that GIRAFFE tests were conducted as developmental tests and that GIRAFFE Phase 1 data should not be used, in other than a confirmatory mode for data generated by other tests, for TRACG code qualification as part of SBWR design certification.

In response to the above findings, GE-NE submitted letter MFN No. 087-94, dated July 1, 1994, to Docket No. STN 52-004, acknowledging the fact that GIRAFFE was a developmental test conducted in a disciplined, professional manner, but not explicitly under the requirements of NQA-1. GE-NE further stated that PANDA and PANTHERS are the principal tests to be used as design basis tests for the SBWR design certification program and it was GE-NE's intention to use GIRAFFE data to substantiate the results of PANDA and PANTHERS at another scale. Since GE-NE does not intend to use the GIRAFFE test data for code qualification as part of design certification, no nonconformances or unresolved items have been identified in this inspection report.

In addition, the team determined that GE-NE had verified that appropriate QA programs were in place or being developed for the PANTHERS (SIET) and PANDA (Paul Scherrer Institute) test facilities.

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

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

Sincerely,

Original Signed R.W. Borchardt

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

Enclosure:

1. Inspection Report No. 99900403/94-02

FOR DISTRIBUTION AND CONCURRENCE SEE NEXT PAGE

Mr. Patrick W. Marriott
GE Nuclear Energy

Docket Nos. 52-004
99900403

cc: Mr. Laurence S. Gifford
GE Nuclear Energy
12300 Twinbrook Parkway
Suite 315
Rockville, Maryland 20852

Director, Criteria & Standards Division
Office of Radiation Programs
U.S. Environmental Protection Agency
401 M Street, S.W.
Washington, D.C. 20460

Mr. Sterling Franks
U.S. Department of Energy
NE-42
Washington, D.C. 20585

Mr. John E. Leatherman
SBWR Licensing Manager
GE Nuclear Energy
175 Curtner Avenue, M/C-781
San Jose, California 95125

Mr. Frank A. Ross
Program Manager, ALWR
Office of LWR Safety & Technology
U.S. Department of Energy
NE-42
19901 Germantown Road
Germantown, Maryland 20874

Mr. Victor G. Snell
Safety and Licensing
AECL Technologies
9210 Corporate Boulevard
Suite 410
Rockville Maryland, 20805

ORGANIZATION: GE Nuclear Energy
San Jose, California

REPORT NO.: 99900403/94-02

CORRESPONDENCE ADDRESS: Mr. Patrick W. Marriott, Manager
Advanced Plant Technologies
GE Nuclear Energy
175 Curtner Avenue
San Jose, California 95125

ORGANIZATIONAL CONTACT: Mr. Kenneth W. Brayman, Manager
Quality Assurance Systems
(408) 925-6587

NUCLEAR INDUSTRY ACTIVITY: GE Nuclear Energy (GE-NE) supplies Advanced Boiling Water Reactor (ABWR) and Simplified Boiling Water Reactor (SBWR) designs worldwide. GE-NE also furnishes engineering services, nuclear replacement parts, and dedication services for commercial grade electrical and mechanical equipment.

INSPECTION CONDUCTED: June 21 through 23, 1994

TEAM LEADER: Richard P. McIntyre 9/26/94
Richard P. McIntyre Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (RVIB)

OTHER INSPECTORS: Billy H. Rogers, RVIB
Robert A. Gramm, RPEB
Frederick R. Allenspach, RPEB
Alan E. Levin, SRXB
Joseph L. Staudenmeier, ASG
Tim M. Lee, RPSB/RES

APPROVED: Uldis Potapovs 9-28-94
Uldis Potapovs, Chief Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (VIB)

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

INSPECTION SCOPE: To determine if activities performed to support the design of the SBWR and specifically, the Gravity Driven Integral Full-Height Test for Passive Heat Removal (GIRAFFE) test program performed by Toshiba in Kawasaki City, Japan were conducted under the appropriate provisions of NEDO 11209-04A, Revision 8, the most recent GE-NE Quality Assurance Program Description that has been approved by the NRC.

PLANT SITE APPLICABILITY: None

1 INSPECTION SUMMARY

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

No nonconformances were identified during this inspection.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 (Open) Nonconformance 99900403/(93-01-01)

Nonconformance 93-01-01 stated that contrary to Criterion III of Appendix B to 10 CFR Part 50 and Section 4.4.1 of Engineering Operating Procedure (EOP) 40-3.00, "Engineering Computer Programs" (ECPs), (1) the TRACG input decks used to model the gravity-driven cooling system integrated systems test (GIST) facility were not independently verified to be correct, and (2) the GE-NE Code Qualification Document (CQD), Licensing Topical Report NEDE-32177P, "TRACG Qualification," dated February 1993, which provides a description of the qualification of TRACG against various activities including the gravity-driven cooling system (GDSCS) integrated systems test, was submitted to the NRC for review and approval for referencing in licensing actions for the Simplified Boiling Water Reactor (SBWR) without receiving independent design verification or design review as required for a level 1 code used to support design basis analyses.

GE-NE committed to take certain corrective actions in response to this nonconformance. One commitment was to independently verify the TRACG input decks for GIST. A GE-NE internal audit of the corrective actions identified that the subsequent independent verification was not complete and therefore did not meet paragraph 4.2.1 of EOP 42-6.00. GE-NE has committed to complete the independent verification by September 30, 1994. Therefore, Nonconformance 93-01-01 remains open.

2.2 (Closed) Nonconformance 99900403/(93-01-02)

Nonconformance 93-01-02 stated that contrary to Criterion XVII of Appendix B to 10 CFR Part 50, EOP 42-10.00, "Design Record Files" (DRF), EOP 35-3.00, "Engineering Tests," and the GIST Program Test Plan and Procedure (TP&P) 521.1322, Revision 2, dated November 29, 1988, certain documentation required to be contained or referenced in the DRF was not included therein. Specific documents that should have been part of the DRF were: the Final Test Report (NEDO-31680) for the GIST Program; instrument calibration records, which were located in a desk drawer in another building; and final design drawings for the facility. Some drawings were found in a cabinet at the facility itself. This set of drawings did not include final numbered, approved, as-built design drawings, which are required by the QA Plan to be retained for the lifetime of the item. Also, data tapes for the GIST tests, which are part of the test records specified for inclusion in the DRF by TP&P 521.1322, were not referenced therein.

GE-NE has taken corrective actions that include the generation of as-built drawings and inclusion of those drawings in DRF A00-02917-1, the inclusion of certain post-test instrument calibration records in the DRF, the inclusion of

the Final Test Report (NEDO-31680) in the DRF, and the inclusion of documentation regarding GIST reactor pressure vessel (RPV) heat losses. These heat loss calculations were reviewed and some of the assumptions made in the calculations are open to question, such as the temperature distribution from the vessel surface to the outside of the insulation. However, the results of the calculation appear to be reasonable and come within about 50 percent of an indirect experimental measurement of heat losses.

The "as-built" drawings included in the DRF are not "blueprint"-type drawings, but rather sketches of the facility dated May 1994. These drawings do, however, contain as-built dimensions and are indicated as having been verified and checked. Post-test instrument calibration is included for the GIST wattmeter, performed November 1993, approximately four years after completion of the test. GE-NE's corrective actions satisfy the requirements of EOP 35-3.00, and EOP 42-10.00. Nonconformance 93-01-02 is closed.

2.3 (Open) Nonconformance 99900403/(93-01-03)

Not reviewed during this inspection.

2.4 (Closed) Nonconformance 99900403/(93-01-04)

Nonconformance 93-01-04 stated that contrary to Criterion XII of Appendix B to 10 CFR Part 50, Section 2.2 of EOP 35-3.20, "Calibration Control," and Section 4.1.2 of the GIST Program TP&P 521.1322, GE-NE purchased flow meters used in the GIST tests from a commercial grade supplier, not on GE-NE's approved supplier list, and accepted and used the instruments as calibrated by the supplier without further verification of the quality or traceability of those calibrations.

The team verified that GE-NE had obtained certificates of calibration from the original supplier and had filed these in DRF A00-02917-1. In addition, since the supplier had not been on the GE-NE approved supplier list, the measurements obtained from the flow meter had been verified by calculation and documented in the DRF. The team reviewed Revision 7 of EOP 35-3.20, dated July 27, 1994, and determined that it had been revised to define measurement and test equipment to include all devices or systems (from the sensing element through the output or recording device) which could affect the accuracy of test measurements. Nonconformance 93-01-04 is closed.

2.5 (Closed) Nonconformance 99900403/(93-01-05)

Nonconformance 93-01-05 stated that contrary to Criterion XI of Appendix B to 10 CFR Part 50, EOP 35-3.00, "Engineering Tests," EOP 42-10.00, "Design Record Files," and Section 4.2.4 of the GIST Program TP&P 521.1322, GE-NE failed to document in the DRF the review and disposition of anomalies in three tests, C01, D01, and D03. These tests were considered to be "invalid" as a result of incorrect valve alignment (C01) or incorrect power input to the test section (D01 and D03). For one of the tests (C01), a note was found on the folder in the DRF in which hard-copy data plots were stored, indicating that a problem existed for the test; however, the problem indicated on the folder (incorrect

power input) was not consistent with the actual reason given in NEDO-31680 for the test's invalidation (incorrect valve alignment).

The team determined that the documentation and disposition of the anomalies had been included in DRF A00-02917-1. In addition, the team reviewed Revision 11 of EOP 35-3.00, dated July 27, 1994, which clarified the requirement to document the review and disposition of anomalies in the DRF. Nonconformance 93-01-05 is closed.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Quality Assurance Program

The Quality Assurance (QA) program for the SBWR program is described in the "SBWR Design and Certification Program Quality Assurance Plan," NEDG-31831, dated May 1990. The plan was prepared for the Department of Energy, Contract No. DE-AC03-90SF18494. This QA plan commits to meeting the requirements of ANSI/ASME NQA-1-1983 and its Addenda (NQA-1a-1983). This plan is based primarily on the standard GE-NE "Quality Assurance Program Description," NEDO-11209-04A, Revision 8, dated March 31, 1989. The GE-NE topical QA program has been previously reviewed by the NRC and found to meet Appendix B of 10 CFR Part 50.

The principle difference between NEDO-11209-04A and NEDG-31831 is that the QA topical program conforms to ANSI N45.2 and the associated ANSI standards, while the SBWR QA plan conforms to NQA-1/1a-1983.

The QA plan described in NEDG-31831 contains a work element/implementing procedure matrix that contains 18 major subdivisions which correlate with the 18 criteria of Appendix B to 10 CFR Part 50; and the supplemental requirements of NQA-1a-1983. The matrix establishes the correlation between the criteria and four categories of GE-NE procedures and/or policies. These four types of GE-NE procedures/policies describe the implementation of the QA plan. These are GE Policy, GE-NE Policies and Procedures (P&Ps), BWR Engineering Operating Procedures (EOPs), and Advanced Nuclear Technology (ANT) Operations Policies and Instructions (OP&Is).

The GE-NE Policies and procedures are high level GE policies that establish overall policies and responsibilities for GE-NE. GE-NE nuclear activities are currently conducted under the responsibility of the Vice President of GE Nuclear Energy. The EOP's are a series of procedures that implement GE-NE policies and the QA program. ANT Operations Policies and Procedures deal with subjects such as schedules, budgeting, contract award, and business management.

The SBWR QA organization reports to the Manager, Services and Projects Quality. A dedicated QA group of a Quality Project Manager, Advanced Reactor Quality, and a QA engineer provides QA overview of the SBWR project.

The SBWR QA plan states that design work performed by the SBWR team will be performed in accordance with QA programs evaluated and accepted by GE-NE. The

plan additionally states that design and testing work performed by International Technical Associates (ITAs) will be performed to their internal QA programs acceptable to the regulatory authorities of their respective countries. The plan stated that by December 31, 1990, GE-NE will perform an initial review to establish that the ITA's QA programs comply to NQA-1 and its addenda. If deficiencies are found, special QA procedures will be developed and applied to attain equivalence. After initial acceptance, GE-NE plans to perform QA reviews approximately on an annual basis to assure the ITA programs are effectively implemented.

During the course of the inspection, cognizant GE-NE personnel stated that SBWR activities associated with Japanese Atomic Power Company (JAPC) were a special case that is not covered by the provisions stated in the SBWR QA plan. The justification for treating some ITAs in a manner not described within the SBWR QA Plan was not clearly evident to the team. The aspect of QA requirements imposed on ITAs is further discussed in Sections 3.4 and 3.5.

The team additionally reviewed NEDG-31836 "SBWR Team Organization and Procedures Manual." The manual was developed to provide for effective SBWR team communications and as an operational guideline to produce consistent actions and documents by team members. The manual provides an overview of the project organization, procedures for control of correspondence, and reference to applying the SBWR QA plan.

The team found that a QA program has been established for the SBWR activities. In some cases, less than effective measures had been implemented to ensure that all organizations involved in the SBWR project had appropriate QA programs.

3.2 Instruction, Procedures and Drawings

The quality requirements utilized for the SBWR are contained in the following GE-NE documents that were reviewed by the inspection team.

GE Policy

- GE QP 20.1, "Corporate Commitment"

GE-NE Policies and Procedures

- 70-1, "Nuclear Energy Quality Policy"
- 70-11, "GE-NE Quality System Requirements"
- 70-30, "Personnel Proficiency in Quality Related Activities"
- 70-50, "GE-NE Handling and Storage of Quality Assurance Records"

Engineering Operating Procedures

- 30-3.40, "Engineering Information System"
- 30-5.00, "Engineering Records Documentation Supplied by External Sources"
- 30-7.00, "Technical Design Procedures"

- 35-3.00, "Engineering Tests"
- 40-3.00, "Engineering Computer Programs"
- 40-7.00, "Design Reviews"
- 42-5.00, "Engineering Requirements Document Release"
- 42-6.00, "Independent Design Verification"
- 42-8.00, "Document Issue and Application by ERM"
- 42-10.00, "Design Record File"
- 45-4.00, "Supplier Services/Equipment Document Review"
- 55-2.00, "Engineering Change Control"
- 60-3.10, "Engineering Records"
- 66-2.10, "Safety Related Classification"

EOP 10-2.00 states that EOPs provide uniform and required practices for control of technical and or quality related activities within GE-NE components. Engineers and Project Managers performing activities governed by the requirements of the EOPs are responsible for ensuring that activities are performed in accordance with the EOPs, or if impractical, the manager initiates appropriate EOP revisions. The EOPs allow for the generation of administrative guidelines that provide detailed how-to instructions for engineers. Responsible managers are allowed, but not required, to issue such guidelines.

ANT Operations Policies and Instructions

- BR 4-117, "Project Directive"
- BR 11-100, "Material Request and Cost and Lead Time"
- BR 11-104, "Supplier Evaluations"

The OP&Is were written to provide specific instructions for the SBWR project. The instructions require that quality requirements for suppliers are to be designated on the Material Requests (MRs), either by direct incorporation or reference to other documents. QA staff are to review each MR with items subjected to NQA-1 for quality verification, access and need for QA approved suppliers. QA provides for, and evaluates, supplier quality system using pre-planned checklists.

The intent of the OP&Is appears to be to ensure appropriate level of QA involvement in material requests for supplied services to ensure translation of QA requirements into the documentation. However, this has been circumvented in practice by technical services agreements between GE-NE and other team members and ITAs as further discussed in Sections 3.4 and 3.5.

The various classes of procedures and instructions were found to provide an appropriate framework to translate the upper tier quality provisions to the working level staff.

3.3 Document Control

Document control is prescribed by P&P 70-11, "Quality Systems Requirements" and numerous EOP's such as 15-2.00, "EOP Application," 30-5.00, "Engineering Records Documentation Supplied by External Sources," 40-7.00, "Design Reviews," 42-5.00, "Engineering Requirements Document Release," 42-6.00,

"Independent Design Verification," 42-8.00, "Document Issue and Application by ERM," 55-2.00, "Engineering Change Control," and 60-6.00, "Drafting Manual Control." These procedures were found to provide an acceptable framework of document control provisions.

3.4 Control of Purchased Material, Equipment, and Services

The team reviewed the GE-NE quality assurance oversight activities related to Phase 1 of the GIRAFFE tests performed in 1989-1990 at the Toshiba test facility in Kawasaki, Japan, which GE-NE had used to support the certification of the SBWR design. GE-NE had not placed a purchase order with Toshiba, an International Technical Associate (ITA) for performance of the test, however, the testing agreement was documented in two contracts.

The first contract, "Joint Study Contract for Feature Technology of Simplified BWR (Phase 1) between the Japan Atomic Power Company (JAPC) and General Electric Company," dated March 30, 1989, included items such as use of the results, industrial property rights, liability and warranty. The document includes a general warranty provision that "GE warrants to JAPC that the study will be performed in conformance with generally accepted professional standards prevailing in the nuclear industry at the time the study is performed" but the study contract imposes no specific QA requirements such as NQA-1-1983 and NQA-1a-1983 to be imposed during the performance of the test. The Phase 1 study was a preliminary study of the containment vessel configuration, basic tests on the PCCS heat removal performance, and study of the GDCS and accumulators.

The second contract, "Agreement on Execution of Joint Study Contract for Feature Technology of Simplified BWR (Phase 1) among Hitachi, Ltd. and Toshiba Corporation and General Electric Company," dated March 30, 1989, included items such as definitions, cost share, protection and use of information, workscope description, objective of the study (test), method of the study, contents of the study (test specifics), study period, work split, study schedule, study organization, and study reports. The document includes a general warranty provision that each "partner warrants to JAPC that its portion of the study will be performed in conformance with generally accepted professional standards prevailing in the nuclear industry at the time the Study is performed," but the agreement also contained no explicit QA requirements.

GE-NE Advance Reactor Quality (SBWR/QA) requested the Toshiba SBWR QA Plan on November 29, 1989. SBWR/QA did not receive the QA Plan but did receive a list of standards and procedures on July 6, 1990, which Toshiba said applied to the SBWR work. The list of standards and procedures was presented in an informal matrix (handwritten) as opposed to a formal QA Plan.

SBWR/QA visited Toshiba on October 29-30, 1990, (the period that the GIRAFFE Phase 1 test was being performed) in order to meet the commitments of the SBWR Quality Assurance Plan NEDG-31831 which required GE-NE to review and evaluate the QA programs of the International Technical Associates. SBWR/QA "concluded that Toshiba considers its current work on the SBWR to be preliminary and has not applied a QA program that meets the applicable requirements of NQA-1-1983

and NQA-1a-1983 at this time. This will not affect preliminary work. However the direct application of Toshiba's final design work on the SBWR requires evidence of the application of a QA program that meets the appropriate parts of NQA-1-1983 and NQA-1a-1983. GE plans to review the [Toshiba] program applied to SBWR Design and Certification program in 12 to 18 months." These conclusions were documented in a letter to Toshiba, SBWR Reference No. GETO-039, dated December 10, 1990. No Corrective Action Request (CAR) was issued by SBWR/QA to ensure management attention and timely resolution. The Quality Project Manager for Advanced Reactor Quality (SBWR QA Manager) indicated to the team that he had notified the responsible people in GE-NE of his results and requested to be notified if Toshiba's activities were to be used for final design, as opposed to preliminary uses, so that he could take the required actions to verify that an adequate QA program was in place. In addition, he indicated that he had inquired about testing during the review at Toshiba but had not been advised of the ongoing activities related to the GIRAFFE test.

In the Spring of 1993, a GE-NE engineer was on site at Toshiba during the end of Phase 2 of the GIRAFFE tests. The team reviewed a series of correspondence between the engineer and the SBWR QA Manager. Two telefaxes, dated March 17, 1993 and June 1, 1993, documented the engineer's attempt to determine whether Toshiba was applying a QA program to the Phase 2 activities. After the engineer was initially unable to determine whether a QA program was in place, the QA Manager suggested in a telefax dated June 3, 1993, that Toshiba NED standard procedures may have been used. The engineer responded in a July 2, 1993, telefax that the Toshiba NED standard procedures were being used and provided documentation of implementation in several areas such as document control and inspection and test control. The engineer's conclusions were formally documented in NEDC-32215P, "GIRAFFE Passive Heat Removal Testing Program," Revision 0, dated July 1, 1993. The report stated the Toshiba quality assurance program was applied to GIRAFFE testing and provided examples of the requirements related to Document Control and Inspection and Test Control.

On July 19, 1993, the SBWR QA Manager issued a letter to the engineer, also distributed to SBWR management, which stated that the engineer's documented verification of Toshiba Nuclear Engineering Laboratories's compliance to NQA-1-1989 applicable to the GIRAFFE test provided adequate evidence that work performed under the GIRAFFE program could be used directly for SBWR final design work. The basis for making this conclusion was not evident to the team as GE-NE was aware of several facets of the Toshiba QA program that did not address NQA-1 aspects such as: evidence of design verification, development of test procedures, and conduct of audits.

On October 18, 1993, GE-NE issued a memorandum identifying that the three party agreement for the Phase II testing performed by Toshiba shall be under a QA program meeting the intent of JAEG-4101-1985. The team also reviewed an October 21, 1993, GE-NE document outlining the elements of an NQA-1 program that were not implemented during the GIRAFFE testing. These included no verification of design activities, no formal test procedures, identification and control of items was not addressed, and no formal audit program.

On December 16, 1993, Toshiba issued Document No. AS-50092, "Quality Assurance Program for Simplified Boiling Water Reactor," Revision 0. AS-50092 stated that prior to October of 1990 Toshiba was performing preliminary engineering and testing work on the SBWR program and that no QA program specific to the preliminary work was developed or documented. AS-50092 further stated that as SBWR work progressed to support the final design the appropriate quality assurance procedures were applied and that AS-50092 documented the QA program that was applied to all final design work performed by Toshiba on the SBWR program since October 1990, and was intended to meet NQA-1-1983 and NQA-1a-1983.

On March 14, 1994, the SBWR/QA visited Toshiba to review the QA program applied to the GIRAFFE tests conducted by Toshiba during the previous four years. This activity was documented in a trip report, dated April 28, 1994, which listed the comparison of GIRAFFE test activities to AS-50092. The report documented the SBWR/QA conclusion of generally adequate implementation and an example in which the flowmeters in the GIRAFFE program were used based upon the manufacturer's certification without additional verification. In an April 15, 1994, internal GE-NE memorandum, SBWR/QA documented an additional comparison of the SBWR QA Manager's review, the GE-NE engineer's review, AS-50092, and the Japan Electric Association JEAG 4101-1990, "Guide for Quality Assurance of Nuclear Power Plants," with conclusions similar to the April 28, 1994, SBWR/QA trip report. The comparison identified that initial calibration of procured instruments was not done, items were not procured from suppliers on an approved suppliers list, that while test anomaly documentation was weak, the tests were re-performed as necessary, and no test procedures were used as the system responded naturally after establishing the initial conditions.

In summary, the NRC determined that the appropriate quality requirements had not been included in the contracts with Toshiba, which established the agreement for the GIRAFFE tests, and GE-NE QA had been onsite at Toshiba in 1990 and had not audited or reviewed that ongoing Phase 1 testing activities. In addition, GE-NE QA had concluded that the SBWR activities reviewed in 1990 were preliminary, that Toshiba was not implementing a QA program which met the requirements of NQA-1-1983 and NQA-1a-1983, and further QA actions would be required to accept the SBWR activities for final design work; Toshiba documented in AS-50092 that prior to October of 1990 no QA program which met the requirements of NQA-1 was being applied to the SBWR design and testing work being performed; and that the oversight provided by the GE-NE engineer was for Phase 2 tests being performed at a time three years later than the Phase 1 test supporting SBWR certification.

The team concluded, based on a review of the documents and discussions with GE-NE personnel, that no quality requirements applicable to the GIRAFFE Phase 1 tests had been contractually imposed on Toshiba and no GE-NE verification of quality activities had been made during performance of the GIRAFFE Phase 1 tests.

3.5 GE-NE Relationships With Additional ITAs

In terms of arrangements governing other SBWR test facilities, the team reviewed the quality activities related to PANTHERS and PANDA test programs, (neither of which had yet been performed) which are intended to support the SBWR design and related computer code qualification. These programs were the PANTHERS, to be performed by Societa' Informazioni Esperienze Termoidrauliche (SIET) in Italy, and PANDA, to be performed by Paul Scherrer Institute (PSI) in Switzerland.

The team reviewed the contract (GE-NE had not used a purchase order) which established the agreement with SIET, for performance of the PANTHERS test, "Agreement among ENEL s.p.a, ENEA, ANSALDO, and GE-NE for SBWR Development," dated May 29, 1992, and determined that it contained requirements that specified the work be performed in accordance the ANSI/ASME NQA-1-1983 and NQA-1a-1983. In addition, in September 1993 GE-NE had performed a review and evaluation of the QA program in place at SIET as documented in a report dated October 29, 1993. The report concluded that the SIET Quality Plan which would be used for the SBWR test activities met the applicable requirements of NQA-1-1983 and NQA-1a-1983 and the work performed under this program could be used directly for the SBWR final design. The team concluded that GE-NE had specified the required quality assurance requirements to SIET and had taken adequate actions to verify that an acceptable quality assurance program was in place.

The team reviewed the contract (GE-NE had not used a purchase order) which established the agreement with PSI for performance of the PANDA tests, "Agreement between the Swiss Confederation, represented by the Paul Scherrer Institute and the General Electric Company and the Electric Power Research Institute, Inc. (EPRI) on Passive Decay Heat Removal and Fission Product Retention Tests and Model Qualification," dated April 1, 1991, and determined that the contract did not specify any quality requirements. GE-NE had performed a review and evaluation of the QA program in place at PSI with respect to the 18 criteria of NQA-1 as documented in a report dated October 29, 1993. The report concluded that PSI Quality Plan which would be used for the SBWR test activities met the applicable requirements of NQA-1-1983 and NQA-1a-1983 and the work performed under this program could be used directly for the SBWR final design. However, PSI was not contractually committed to using the Quality Plan which GE-NE had reviewed and accepted. GE-NE indicated to the inspection team that the required quality requirements, NQA-1-1983 and NQA-1a-1983, would be included in the Test Specification that GE-NE was currently preparing for issuance to PSI.

Further evidence of the fact that QA programs are implemented by the SBWR team members was obtained during the team's review of GE-NE evaluations of the QA programs. An October 15, 1990, GE-NE memorandum "Review and Evaluation of QA Programs Applied to the SBWR by the European Technical Associates - Trip Report" concluded that ANSALDO, NUCON, and KEMA meet NQA-1. The ENEL QA manual was found to meet NQA-1 but it had not been implemented. ENEA and ECN programs were undefined.

A similar GE-NE memorandum dated November 16, 1990, "Review and Evaluation of QA Programs Applied to the SBWR by the Japanese Technical Associates - Trip Report," documented that the Hitachi QA program was found satisfactory. An October 29, 1993, GE-NE document "Review and Evaluation of QA Programs Applied to the SBWR by the European Technical Associates - Trip Report" concluded that all of the ITAs (UTE, ENSA, KEMA, NUCON, ECN, ANSALDO, ANSALDO ACO, ENEA/SIET, and ENEL) were meeting the requirements of NQA-1-1983. The only exception was that CIEMAT was not. With the exception of the fact that NQA-1 provisions had not been explicitly invoked by GE-NE for the SBWR work performed by several Team members and ITAs, the team noted no other concerns.

3.6 Quality Assurance Records

Quality Assurance record requirements are specified by P&P 70-11, "Quality Systems Requirements", and numerous EOP's such as 35-3.00, "Engineering Tests", 40-7.00, "Design Reviews", 40-9.00, "ASME Code Design Verification", 42-6.00, "Independent Design Verification", 42-10.00, "Design Record Files", and 60-3.10, "Engineering Records Retention".

EOP 42-10.00 describes the Design Record Files (DRF's) as formal, organized accumulations of information, which provide a controlled system for retention of documented engineering activities, necessary to substantiate significant design decisions. The DRF provides a mechanism for controlling and archiving important design records, such as design verification, studies and analyses. It does not include documents, such as drawings and specifications, which are maintained under separate corporate design controls. The procedure also states that the DRF should provide for design notes, calculations, records and other supporting information, and cross-reference to related or supporting DRFs.

The team was informed by the cognizant GE-NE manager responsible for SBWR document retention that the SBWR team members and ITAs are currently holding their records rather than turning them over for GE-NE retention as outlined in NEDG-31836 section 2.6.2, that states "GE-NE is responsible for maintaining a complete physical file of all relevant design input documents received from participating organizations."

The team reviewed selected portions of DRF T15-00004, Volume 6 that pertained to QA aspects of the GIRAFFE test. This information was previously discussed in Section 3.4. Based on the limited sample of records reviewed, the inspector was not able to draw any conclusions about the QA records review conformance with applicable GE-NE requirements.

3.7 Design Control

GE-NE is responsible for maintaining records of all testing information used for SBWR design certification. GE-NE's main archive for GIRAFFE facility information and test data was the GIRAFFE Experiment and Analysis DRF T15-00004. GIRAFFE facility information and test data was also used for TRACG code qualification.

The team reviewed DRF T15-00004 for the GIRAFFE test program. GE-NE stated that it was not complete, in that key information, such as test procedures, instrument calibration records, test acceptance criteria, and as-built drawings, reside with Toshiba in Japan. However, some important design information and as-built sketches are included in the DRF. Therefore, the team review focused primarily on the technical content of the San Jose DRF. Since the conclusion of the inspection, GE-NE has submitted a letter to the NRC stating that GIRAFFE was a developmental test conducted in a disciplined, professional manner, but not explicitly under the requirements of NQA-1. GE-NE further stated that PANDA and PANTHERS are the principal tests to be used as design basis tests for the SBWR design certification program and it was GE-NE's intention to use GIRAFFE data to substantiate the results of PANDA and PANTHERS at another scale.

The information in the DRF indicated in general that GIRAFFE was a competently performed developmental engineering test program. The test facility was carefully designed, and evidence of considerable dialogue between GE-NE and Toshiba was found regarding design characteristics to be included in GIRAFFE. In regards to the conduct of the test program, it appears due attention was paid to the acquisition of good data, including characterization of the facility. Heat and mass balances were checked for the system during shakedown tests. There is some question about measurement of total power to the facility during some tests, due to the use of "microheaters" to minimize facility heat losses and the uncertainty in the heat input from these heaters.

Extensive data from the GIRAFFE test program, both in the form of plots and in test reports, are included in the DRF. Examination of this data did not show any apparent anomalous behavior. Reports from a resident GE-NE engineer on-site at Toshiba during part of the GIRAFFE program did not indicate any problems with the test data (although these reports did raise some questions concerning the implementation of a formal QA program to cover the testing). There is also substantial evidence in the DRF of information exchange between GE-NE and Toshiba during GE-NE's efforts to analyze the data, which would seem to indicate that Toshiba had the technical basis from which to address testing-related questions. There appeared to be a clear reluctance on the part of Toshiba to release to GE-NE, any information beyond which there was a clear "need to know" by GE-NE. This made it more difficult for GE-NE to analyze the data, and also made it more difficult to review the information in the DRF, since some gaps still appear to exist. While any data used in a confirmatory fashion would have to be reviewed for applicability by the NRC, it appears that the data from this developmental test can provide a useful basis for additional verification of computer code models beyond that afforded by the design certification data acquired from the PANDA and PANTHERS test facilities.

3.8 TRACG Qualification of GIRAFFE

The team interviewed the GE-NE engineer in charge of TRACG code qualification on how GIRAFFE data was being used in TRACG code qualification. The inspector discovered that the GIRAFFE input deck used in the TRACG internal design review and NRC code qualification document had been supplied by Toshiba and had not been checked by GE-NE. GE-NE did not have the calculation notebook

which contains the supporting information for the development of the input deck. GE-NE stated that such a check is not required by their engineering procedures even if GE-NE does not have the supporting information for the development of the input deck. GE-NE performed an independent verification of the GIRAFFE input deck because of the findings that were previously identified during the GIST inspection at GE-NE in August 1993. The information used in the independent verification came from the GE-NE GIRAFFE test report NEDC-32215P and information gathered from a trip to Toshiba by GE-NE and Brookhaven National Laboratory in February 1994.

The independent verification revealed discrepancies in the original input deck. The code qualification test cases were rerun and the results were shown to the chairman of the GE-NE design review team. This was documented in the TRACG DRF. Changes in the results were not significant and did not change the conclusions of the adequacy of TRACG in predicting GIRAFFE data. The team also discussed TRACG condensation modelling with GE-NE since this is the most important process for long term decay heat removal in the GE-NE SBWR design. GE-NE confirmed that the only testing of the TRACG condensation model was the GIRAFFE data. The integral GIRAFFE data is not adequate to test a local film condensation model that has strong interactions with other code models such as interfacial drag, wall drag, and the transport of noncondensable gases. Questions about the adequacy of Toshiba modelling and the adequacy of the TRACG condensation model were raised upon review of the GIRAFFE DRF T15-00004.

The GIRAFFE DRF T15-00004 included a well documented historical record of the communications that took place between GE-NE and Toshiba on the GIRAFFE test program and TRACG modelling. Documentation in the DRF made it clear that the GIRAFFE experiment should not be looked at as an SBWR simulator or proof that the long term decay heat removal system will work. It stated that the only reason the facility pressure started decreasing as early as it did was due to environmental heat losses that were greater in the test facility compared to what they would be in a perfectly scaled SBWR simulator. The GIRAFFE Phase 1 testing was also done in a facility that is nonprototypical compared to the present SBWR design. It was designed to represent an earlier SBWR design. The GIRAFFE experiment used the May-Witt decay heat curve for its power input. This is higher than the ANS-79 standard and is the Japanese regulatory licensing standard.

The DRF also documented the difficulties in modelling GIRAFFE and calculating the experimental results. An internal memorandum in the DRF recommends against modelling the GIRAFFE facility with one TRACG VESSEL component because heat losses to the environment could not be modelled properly because of code limitations. In this modelling choice the heat losses to the environment were modelled by reducing the power input to the reactor core. This will reduce the steam generation rate in the core and reduce the steam flow to the containment and the load on the PCCS. In spite of recommendations against this modelling practice, this is the model used in TRACG code qualification. A letter in the DRF from GE-NE to Toshiba stated that GE-NE could not use the Toshiba TRACG GIRAFFE model for TRACG code qualification because it would not meet GE-NE's QA requirements. This model was later used for TRACG code qualification without any explanation.

The TRACG film condensation model used in the PCCS is the University of California at Berkley (UCB) correlation which is the Nusselt correlation with multipliers that take into account the degradation due to noncondensable gases and enhancement due to shear effects. Initial evaluations of the GIRAFFE data showed that Nusselt underpredicted the heat transfer by approximately a factor of two. The TRACG code qualification document states that the UCB correlation slightly underpredicts the GIRAFFE PCCS heat transfer. A letter from GE-NE to Toshiba as late as January 1992 indicated that TRACG predicted inadequate PCCS performance. Contradictory to this Toshiba states that when the UCB correlation is used in the Toshiba TRAC, the heat removal is overpredicted. There is no clear resolution of why and how the UCB correlation was able to give acceptable predictions of the GIRAFFE data and why it was accepted for use in the code.

The above questions and concerns in Section 3.8 were discussed during the inspection with GE-NE staff and represent the type of issues that would be raised by the NRC staff during the review of the TRACG code qualification for long term containment heat removal.

3.9 Corrective Action

The team reviewed EOP 75-3.00, "Corrective Action and Audits." Corrective Action Requests (CARs) are used to document quality system problems, to request timely response to these problems, and to record objective evidence of action taken as a result. The CAR mechanism is used to assure that root causes and preventive actions are identified in response to quality problems. EOP 75-6.0, "Quality Assurance Records" specifies that audit reports and CARs are kept for 3 years.

The team reviewed several Corrective Action Requests (CARs) that had been generated either during an audit or individually. For example, on June 20, 1994, a CAR was issued as committed actions related to verification of GIST input decks had not been totally completed. Another example involved CAR ARP 94-3-1 that was issued when a number of DRFS were found non-conforming to the pertinent requirements during the course of an internal GE-NE audit. The committed corrective actions were found appropriate, they included correcting the DRFs, re-training personnel on guiding requirements, and review of other DRFs for problems.

In one case however, when QA had identified that the Toshiba quality program could not be demonstrated to meet NQA-1 in accordance with the SBWR quality program in 1990, no documented action was taken until the 1993 time frame to resolve the issue. This problem should have been a candidate for using a CAR to ensure it was rectified.

The CAR mechanism was found to assure that appropriate levels of management were aware of quality issues and that timely corrective measures were instituted and verified. One weakness was identified where an issue was not captured in the CAR system to ensure resolution.

3.10 Audits

The team reviewed EOP 70-11, section 3.13 related to audits. Audits are required to be performed annually by NQA of each staff level component reporting to the Vice President or General Manager. The audits shall comply with NQA-1 and ANSI N45.2.12 as applicable. External GE-NE audits of suppliers are required every three years. The audits are to be staffed with personnel independent of the work being audited. The program shall ensure timely corrective action for audit findings.

The following GE-NE internal audits were reviewed:

Services Quality Assurance

- Audit ABQ-88, "Quality Assurance Audit of ABWR Program by Material Services Quality of Product Quality Assurance." The audit was conducted from September 21, 1988 to April 19, 1989. The content of DRFs, one of which involved the SBWR Gravity Driven Cooling System test program, was reviewed.
- Audit ABQ-90, "PQA Audit ABQ-90 of the ABWR Programs K6/K7 and SBWR." The audit was conducted from October 10, 1990 to November 30, 1990. The audit scope examined compliance with the governing SBWR QA program, control of US Team member interfaces, indoctrination of SBWR staff, design control, and test control. An audit finding was issued for training of SBWR personnel on the quality program requirements. The audit also reviewed design files on TRACG and GDCS. The corrective action measures included documented evidence that training had been received by the appropriate GE-NE staff.
- Audit ABQ-91-3, "PQA Audit of the SBWR Program and the K6/K7 Program." The audit was conducted from December 9, 1991 to March 2, 1992. The audit scope included interface controls, design control, test control, and corrective actions from previous audits. An audit finding had been generated related to handling of Engineering Review Memoranda (ERMs).

Advanced Reactor Programs

- Audit Q93-02, "SBWR Design and Certification." The audit was conducted from June 22, 1993, to August 8, 1993. The audit scope included design interfaces, implementation of SBWR Team and Organization manual requirements, DRF file generation, design reviews, procurement control, and document control. Some audit observations included one involving administrative control of DRFs.
- Audit Q92-03, "ABWR Certification and SBWR Design and Certification." The audit scope included interface controls, design procedure implementation, corrective action on prior findings, and SBWR procurement.
- Audit 94-3, "SBWR Design and Certification." The audit was conducted from March 29, 1994, to May 19, 1994. The audit scope included DRFs, training, procurement control, and prior audit finding corrective action. A audit finding was generated as multiple DRFs were found nonconforming to requirements pertaining to: organization incomplete, missing indices, design verification is incomplete, open verifications, corrections made without

traceability. The inclusion of a technical specialist on this audit team is considered a positive approach.

The team reviewed related documentation such as: lead auditor qualification and certification records, audit plans, audit checklists, and audit exit meeting notes.

The team concluded that GE-NE was implementing an appropriate internal audit program. The conduct of GE-NE audits external to the GE-NE organization was not examined.

4 PERSONNEL CONTACTED

GE Nuclear Energy

Robert H. Buchholz, Manager, SBWR - U.S. Programs
Patrick W. Marriott, Manager, Advanced Plant Technologies
Ken Brayman, Quality Systems Manager, Nuclear Quality Assurance (NQA)
Forrest Hatch, Manager, Nuclear Services & Projects Quality
Philip Novak, Quality Assurance Manager, Advanced Reactor Programs (ARP)
Don Kaye, Quality Assurance, SBWR Project
Norman E. Barclay, Nuclear Service & Projects Quality
Jay Murray, QA Audits Manager, NQA
Terry McIntyre, Project Manager, SBWR Test Operations and Analysis
Paul F. Billig, Senior Engineer, ARP
John Leatherman, Manager, SBWR Certification
Sandra Devlin, Project Manager, SBWR Design Documentation Closure
Gordon Wingate, Engineer, SBWR Design Documentation Closure
Jim Shaug, Senior Engineer, Safety & Thermal Hydraulic Methods
Maryann Herzog, Senior Engineer, SBWR
David Foreman, SBWR Licensing
Doug Hashi, Project Manager, International Agreements
Ted Bush, Information Management Systems

Nuclear Regulatory Commission

Richard McIntyre, Team Leader, Vendor Inspection Branch (VIB)
Billy Rogers, VIB
Joseph Staudenmeier, Analytical Support Group
Robert Gramm, Performance & Quality Evaluation Branch (PQEB)
Frederick Allenspach, PQEB
Alan Levin, Reactor Systems Branch
Tim M. Lee, Office of Nuclear Regulatory Research
Andre Drozd, Containment Systems and Severe Accident Branch
Son Ninh, Standardization Project Directorate

Toshiba

Koichi Taira, SBWR Resident Engineer,
Kenji Arai, Nuclear Engineering Laboratory

Department of Energy

Kashmira Vijaiyan, Oakland Field Office

EPRI

Richard Burke, Manager, SBWR Design Certification
Hans Zimmer,

Paul Scherrer Institute

Paul Coddington, PANDA Test Facility

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

December 14, 1994

Tennessee Valley Authority
ATTN: Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Executive Officer
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: INSPECTION OF THE REPLACEMENT ITEMS PROGRAM (RIP) CORRECTIVE
ACTION PROGRAM (CAP) PLAN AT THE WATTS BAR NUCLEAR PLANT, UNIT 1
(NRC INSPECTION REPORT NO. 50-390/94-201)

Dear Mr. Kingsley:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection, conducted by Robert Pettis, Jr., Larry Campbell, Stephen Alexander, of the Special Inspection Branch, and Juan Peralta of the Quality Assurance and Maintenance Branch, of the Office of Nuclear Reactor Regulation (NRR), and Ronald Gibbs and William Bearden of NRC Region II. The inspection was conducted to review the implementation of the Tennessee Valley Authority (TVA) Replacement Items Program (RIP) Corrective Action Program (CAP) for the Watts Bar Nuclear Plant (WBN), Unit 1.

The inspection, conducted June 20 through 24, July 5 through 8, and July 18 through 20, 1994, was performed at the 75-percent completion stage of the RIP CAP Plan and was related to activities at the plant site authorized by NRC Construction Permit CPPR-91. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed inspection report. Areas examined during the inspection are identified in the report. The inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

Subsequent to the onsite portion of the inspection, TVA continued to provide information for the inspectors to review to help close out several issues that remained unresolved at the time of the exit meeting. These reviews occasioned several telephone conversations with TVA and the submission of additional information in response to questions raised during review of the supplementary submittals. The last of this information was received by the inspectors on November 7, 1994. The review of this material was completed and the appropriate report sections revised accordingly.

In general, the inspection team determined that the RIP CAP activities were effective and that significant improvements had been made. However, deficiencies regarding program implementation were identified. For example, during the review of the Material Improvement Project, numerous instances were

identified where material had been installed without performing a review for suitability, since the affected items were not tracked by the RIP data base as an outstanding work item. During the inspection, TVA initiated an evaluation which determined that 84 of a total of 158 sanitization packages (the sanitization process involved reviewing the adequacy of an item's documentation), identified for further evaluation under the RIP, had not been entered into the RIP data base and therefore, had not been evaluated for their intended safety-related applications.

The inspection team also identified several instances where inadequate commercial grade dedication was performed for installed material (i.e., circuit breakers and related components). The deficiencies appeared to reflect a less than full understanding among some procurement engineers of the design basis requirements of the various Watts Bar electrical distribution systems. The inspection team also identified the failure to perform adequate corrective action for equipment which failed in service and the failure to follow TVA procedures.

During the review of Revision 5 to the RIP CAP Plan, the inspection team determined that many elements of it were conducive to general improvement of the procurement and material control programs at WBN. However, Revision 5 itself (as had been the case with Revision 4), as a stand alone document, expresses positions (e.g., in Note 2 of the Introduction) that are not consistent with the requirements of Appendix B to 10 CFR Part 50 and the staff positions stated in NRC Generic Letters 89-02 and 91-05, where the staff positions differ from Electric Power Research Institute (EPRI) EPRI NP-5652.

As written, Revision 5 to the RIP CAP would not ensure that all material, particularly commercial grade items, would be adequately rendered suitable or verified to be suitable for specific safety-related plant applications under all design basis conditions. However, these differences from staff positions were not reflected, in general, in the implementing procedures that were actually in use for procurement and dedication activities; although the procedures did contain some less significant deficiencies as noted in this report. Therefore, the staff requests that you revise and docket a RIP CAP Plan to reflect the actual procurement practices in-place at WBN.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room. No response to this letter is required. The NRC Region II Office will issue any enforcement actions resulting from this inspection.

Mr. Oliver D. Kingsley, Jr.

-3-

Should you have any questions concerning this inspection, please contact the NRR Project Manager Peter Tam at (301) 504-1451 or the inspection team leader Robert Pettis at (301) 504-3214.

Sincerely,

A handwritten signature in dark ink, appearing to read "Steven A. Varga". The signature is fluid and cursive, with the first name "Steven" and last name "Varga" clearly distinguishable.

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

Enclosure:
Inspection Report 50-390/94-201

cc: See next page

Tennessee Valley Authority

cc w/encls:

Mr. Craven Crowell, Chairman
Tennessee Valley Authority
ET 12A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. W. H. Kennoy, Director
Tennessee Valley Authority
ET 12A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Johnny H. Hayes, Director
Tennessee Valley Authority
ET 12A
400 West Summit Hill Drive
Knoxville, TN 37902

Dr. Mark O. Medford, Vice President
Engineering & Technical Services
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. D. E. Nunn, Vice President
New Plant Completion
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. J. A. Scalice, Site Vice President
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

General Counsel
Tennessee Valley Authority
ET 11H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. R. W. Huston, Manager
Nuclear Licensing and
Regulatory Affairs
Tennessee Valley Authority
4G Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Watts Bar Nuclear Plant, Unit 1

Mr. B. S. Schofield
Site Licensing Manager
Watts Bar Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Spring City, TN 37381

TVA Representative
Tennessee Valley Authority
11921 Rockville Pike
Suite 402
Rockville, MD 20852

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW., Suite 2900
Atlanta, GA 30323

Senior Resident Inspector
Watts Bar Nuclear Plant
U.S. Nuclear Regulatory Commission
Route 2, Box 700
Spring City, TN 37381

The Honorable Robert Aikman
County Executive
Rhea County Courthouse
Dayton, TN 37321

The Honorable Garland Lanksford
County Executive
Meigs County Courthouse
Decatur, TN 37322

Mr. Michael H. Mobley, Director
Division of Radiological Health
3rd Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

Ms. Danielle Droitsch
Energy Project
The Foundation for
Global Sustainability
P.O. Box 1101
Knoxville, TN 37901

Mr. Bill Harris
Route 1, Box 26
Ten Mile, TN 37880

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

INSPECTION REPORT NO.: 50-390/94-201

CONSTRUCTION PERMIT NO.: CPPR-91

FACILITY NAME: Watts Bar Nuclear Plant, Unit 1

INSPECTION CONDUCTED: June 20 through 24, 1994
July 5 through 8, 1994
July 18 through 20, 1994

INSPECTION TEAM: Robert L. Pettis, Jr., P.E., NRR
Stephen Alexander, NRR
Larry Campbell, NRR
Juan Peralta, NRR
Ronald Gibbs, Region II
William Bearden, Region II

PREPARED BY:

Robert L. Pettis, Jr., P.E., Team Leader
Vendor Inspection Section
Special Inspection Branch
Division of Technical Support
Office of Nuclear Reactor Regulation

12/8/94
Date

APPROVED BY:

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

12/8/94
Date

TABLE OF CONTENTS

EXECUTIVE SUMMARY	i
1 INTRODUCTION	1
2 INSPECTION OBJECTIVES AND SCOPE	2
3 PROGRAM REVIEW	3
4 CURRENT AND FUTURE PROCUREMENTS	4
4.1 Procurement Package Review.....	5
5 PLANT INSTALLED ITEMS VIA PREVIOUS MAINTENANCE ACTIVITIES	
REPLACEMENT ITEMS PROGRAM (RIP)	17
5.1 QA Level I PPSP Review	18
5.2 QA Level II PPSP Review	18
5.3 QA Level III PPSP Review	23
5.4 Review of Generic Dedication Packages	28
5.5 Control of Product Contaminants	28
5.6 Replacement Items Installed by Previous Maintenance Activities ...	29
5.6.1 Maintenance History Data Base	29
5.6.2 Problem TIICs	30
5.7 RIP Hold Items	31
5.8 IMPELL Seismically Sensitive Electrically Active Items Review	34
5.9 RIP QA Level III Item Review	36
6 REPLACEMENT ITEMS INSTALLED BY PREVIOUS CONSTRUCTION ACTIVITIES	
REPLACEMENT ITEMS PROGRAM (RIP)	37
7 CURRENT WAREHOUSE INVENTORY-MATERIAL IMPROVEMENT PROJECT (MIP)	38
7.1 Implementation Review of the MIP	38
7.1.1 Acceptable Sanitization Packages	39
7.1.2 Unacceptable Sanitization Packages	40
8 EXIT MEETING	42
APPENDIX A - LIST OF DEFICIENCIES	A-1
APPENDIX B - LIST OF UNRESOLVED ITEMS	B-1
APPENDIX C - LIST OF INSPECTOR FOLLOWUP ITEMS	C-1
APPENDIX D - GLOSSARY OF SPECIFIC TERMS	D-1

EXECUTIVE SUMMARY

The Nuclear Regulatory Commission (NRC) inspection of the Tennessee Valley Authority's (TVA's) Replacement Items Program (RIP) piece parts Corrective Action Program (CAP) Plan for the Watts Bar Nuclear Plant (WBN) inspection, led by the Vendor Inspection Branch (Now the Vendor Inspection Section of the Special Inspection Branch) of the Office of Nuclear Reactor Regulation (NRR), was conducted from June 20-24; July 5-8; and July 18-20, 1994. The inspection team consisted of staff members from NRR and Region II.

The objective of the inspection was to review the implementation of Revision 5 to the TVA RIP CAP Program to determine its compliance with Appendix B to 10 CFR Part 50 (Appendix B), and related TVA commitments. The inspection team determined that many elements of Revision 5 of the RIP CAP were conducive to general improvement of the procurement and material control programs at WBN. However, Revision 5 itself (as had been the case with Revision 4), as a stand alone document, expresses positions (e.g., in Note 2 of the Introduction) that are not consistent with the requirements of Appendix B, and that take exception to certain key provisions of NRC GLs 89-02 and 91-05, where they differ from Electric Power Research Institute (EPRI) EPR, NP-5652, "Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07)."

As written, Revision 5 to the RIP CAP would not ensure that all material, particularly commercial grade items, would be adequately rendered suitable or verified to be suitable for specific safety-related plant applications under all design basis conditions. However, the positions expressed in Revision 5 were not reflected in the implementing procedures that were in use for procurement and dedication activities; although the procedures did contain some relatively less significant deficiencies as noted in the report.

In addition, the team identified some deficiencies in the implementation of these program procedures. The findings included inadequate commercial grade dedication of items installed in safety-related applications, such as General Electric (GE) 125-Vdc trip coils and spring charging motors for GE Magne-Blast circuit breakers used in Class 1E 6900V shutdown boards, Westinghouse direct trip actuators used in 480-Vac air circuit breakers and the lack of verification of the procurement requirements of Telemecanique starters. Other deficiencies included implementation weaknesses in TVA's corrective action program, the lack of evaluation of previously installed material not contained in the RIP data base, and procedures used which were not appropriate to the circumstances.

Many of the deficiencies identified by the inspection team were attributed to incomplete or ineffective program implementation or to various procedural weaknesses. In the exit meeting conducted on July 20, 1994, the inspection team discussed the inspection findings with TVA's senior management. Based on the results of this inspection, it was determined that, in general, the RIP CAP plan activities appeared to be effective, however continued management attention is required to assure proper program implementation.

1 INTRODUCTION

The Tennessee Valley Authority (TVA) began the Watts Bar Nuclear Plant (WBN) Replacement Items Program (RIP) Corrective Action Program (CAP) in 1988 to resolve deficiencies with the procurement of safety-related replacement items procured without TVA engineering review and to address TVA employee concerns and NRC audit findings from the Sequoyah Nuclear Plant. TVA's primary objective of the RIP CAP is to ensure that replacement piece parts, especially commercial grade items (CGIs) installed in safety-related basic components, are consistent with previously validated environmental or seismic qualifications, and with the component's ability to perform its intended safety function.

The scope of the RIP CAP covers commercial grade replacement item procurements for safety-related applications and procurements of replacement items for environmentally qualified (10 CFR 50.49) components. The bases for this RIP CAP plan inspection are Appendix B to 10 CFR Part 50 (Appendix B), 10 CFR 50.49, Revision 5 to TVA RIP CAP of March 1, 1994, and NRC Inspection Procedure (IP) 38703, "Commercial Grade Dedication."

The RIP CAP addressed a lack of engineering involvement in the procurement process, which resulted in unreviewed and potentially unacceptable spare or replacement parts being installed in previously qualified equipment in the plant. Revision 5 of the RIP CAP addresses the following four distinct areas:

- Current and Future Procurements: These procurements are made through TVA's current program, which requires engineering involvement in all new procurements. TVA's Procurement Engineering Group (PEG) does an engineering review of the purchases of all safety-related equipment. The PEG reports directly to Nuclear Engineering and is connected to Materials and Procurement in the organizational matrix.
- Plant Installed Items from Previous Maintenance Activities: This activity of the RIP CAP involves a historical review (for technical adequacy) of spare or replacement parts installed from the power stores inventory for work controlled by maintenance work requests or work packages. Installed material is identified by a TVA Item Identification Code (TIIC) number and is controlled by TVA Nuclear Stores requisition Form 575N.
- Replacement Items Installed by Previous Construction Activities: This activity of the RIP CAP involves a historical review (for technical adequacy) of the replacement of parts installed from the construction warehouse inventory by personnel from the construction group.
- Current Warehouse Inventory: This activity of the RIP CAP is commonly referred to as the Material Improvement Project (MIP). The MIP was initiated as a result of a 1991 TVA quality assurance (QA) assessment which identified weaknesses in commercial grade dedication. This activity is designed to strengthen the Quality Release Program by "sanitizing" all old inventory prior to release for installation to the plant. The sanitization process includes an engineering review for

technical adequacy, verification of the item, QA receipt inspection, and storage.

TVA implemented the MIP on June 5, 1991, after which replacement items, in inventory as of that date, must have a MIP sanitization package prepared and approved before the item can be issued to the plant. Items issued prior to June 5, 1991, are to be evaluated and resolved by the historical reviews discussed in the RIP portion of the inspection.

In accordance with the NRC letters to TVA regarding the RIP CAP (particularly Revision 5), the team reserved its judgement of the RIP CAP itself until it had completed a substantial portion of the review of the program procedures established under the RIP CAP and their implementation. As a result of that review conducted during this inspection, the team's assessment of the RIP CAP as written is consistent with the reservations expressed in the NRC letters to TVA regarding the RIP CAP. The team concluded that as a stand-alone document, presumed to describe and generally prescribe the practices employed by TVA in procurement and commercial grade dedication, RIP CAP, Revision 5, expresses positions that are not consistent with the requirements of Appendix B, and that take exception to certain key provisions of GL 89-02 and GL 91-05 where they differ from EPRI NP-5652.

However, the team's assessment of the actual WBN program, procedures and implementation, was that with the few exceptions noted, and contrary to some of the positions stated in the RIP CAP and associated correspondence, the program procedures appeared on the basis of the review within the scope of this inspection, to be more consistent with the applicable requirements of Appendix B, and generally consistent with the staff positions as promulgated in Generic Letters 89-02 and 91-05. The exceptions mentioned above consisted of certain programmatic deficiencies discussed below and the identified classification and dedication deficiencies.

2 INSPECTION OBJECTIVES AND SCOPE

The NRC conducted this inspection in accordance with NRC IP 38703, Appendix B, and Revision 5 to the TVA RIP CAP. The objective of this inspection was to review TVA's implementation of the RIP CAP to determine its compliance with Appendix B and related TVA commitments.

The inspection team has characterized its findings within this report as deficiencies, unresolved items, or inspector followup items. Deficiencies are either a) the apparent failure of the licensee to comply with a requirement or b) the apparent failure of the licensee to satisfy a written commitment or to conform to the provisions of applicable codes, standards, guides or accepted industry practices when the commitment has not been made a legally binding requirement. Unresolved items involve a concern about which more information is required to ascertain whether it is acceptable or deficient. Each deficiency identified during the inspection, and designated in the body of the report, is listed in Appendix A. Unresolved items and inspector followup items are listed in Appendix B and Appendix C, respectively. A glossary of TVA-WBN specific terms is included in Appendix D.

3 PROGRAM REVIEW (TI 2512/27, 35065, 35746, 35747)

Before and during the inspection, the inspection team conducted a review of the WBN site procedures which define material procurement, receipt inspection, storage and handling, and material issue and control. This review included procedures which define the current program, as well as the RIP and MIP, and included procurement, receipt inspection, storage and handling. The following procedures were included in this review:

- Site Standard Practice (SSP) SSP-10.01, "Procurement of Materials and Services," Revision 10
- SSP-10.02, "Material Receipt and Inspection," Revision 18
- SSP-10.03, "Material Handling, Storing, and Shipping," Revision 16
- SSP-10.04, "Material Issue, Control, and Return," Revision 20
- SSP-10.05, "Technical Evaluation for Procurement of Materials and Services," Revision 10
- SSP-10.06, "Materials Marking and Identification," Revision 2
- SSP-10.B, "Materials Improvement Project," Revision 5
- SSP-10.C, "Evaluation of Installed Safety-Related Replacement Items," Revision 1
- Site Administrative Instruction (SAI) SAI-10.B, "Materials Improvement Project Sanitization process," Revision 1
- Quality Assurance Instruction (QAI) QAI-10.03, "Material Sanitization QA Program," Revision 4
- QAI-10.04, "Material Receipt Inspection," Revision 11
- QAI-10.05, "PPSP-Review and Inspection," Revision 1
- QAI-10.06, "Inspection of Storage Facilities," Revision 3

The team verified that procedures were established and implemented for key parts of the material control program. The team verified the following attributes of the requirements:

- inclusion of technical and QA requirements in procurement documents, including environmental qualification and seismic requirements, testing and inspection requirements, documentation requirements, TVA access to vendors facilities and records for audit purposes, and the basis for quality classification of materials.
- control of procurement documents to ensure incorporation of engineering design requirements, as well as, engineering review and approval of the purchases and changes during the procurement process.
- inclusion of 10 CFR Part 21 requirements, including applicant dedication of CGIs where appropriate.
- establishment, maintenance, and use of an approved suppliers list (ASL) including the basis for qualification of suppliers such as by audit and source inspection.
- request for receipt and review of vendor documentation, including Certificates of Compliance/Conformance (COCs) and Certified Material Test Reports (CMTRs).

- performance of receipt inspection of materials including verification of the adequacy of vendor documentation, vendor compliance with purchase order requirements, and performance of special inspections specified by engineering. The team also reviewed the receipt inspection facilities, tools and equipment used by quality control (QC) inspectors, and receipt inspection records which include programs for identification and resolution of receipt inspection deficiencies, and the control of conditional releases for materials not yet inspected.
- means to separate acceptable material from material not yet accepted, and from nonconforming material (i.e., tagging, identification, segregation, and storage).
- identification and implementation of storage requirements and storage levels, including segregation of non-conforming materials, and requirements for inspection of storage areas by TVA QC personnel.

The inspection team inspected the storage areas for proper storage of materials and verified the adequacy of storage conditions, access controls, identification and retrievability of stored materials, and segregation of nonconforming materials. Except for the procedural deficiencies identified with SSP-10.C (Section 5.1 of this report), the procedures reviewed appeared satisfactory in implementing the RIP CAP.

4 CURRENT AND FUTURE PROCUREMENTS (TI 2512/27, 35065, 35746, 35747)

The requirements for the preparation of TVA's current procurement practice and processes are prescribed in SSP-10.01, Revision 10, "Procurement of Materials and Services" and in SSP-10.05, Revision 10, "Technical Evaluation for Procurement of Materials and Services." These procurements are made to TVA's current program which requires engineering involvement by TVA's PEG for all new safety-related procurements.

In a letter of April 5, 1994, the NRC staff documented its review of Revision 5 of the WBN RIP CAP plan of March 15, 1994, and stated that it could not substantiate the acceptability of the commercial grade dedication portion of the program. Accordingly, the NRC stated that an inspection would be performed to determine the acceptability of controls and procedures governing commercial grade dedication activities at WBN. TVA's decision to take exception to the guidance in Generic Letter (GL) 89-02 (in Note 2 of the Introduction) was one of the reasons for the staff's decision not to reach a conclusion on the acceptability of the RIP CAP program until an inspection of its implementation was performed. In a letter to TVA on July 27, 1992, the staff stated that since TVA takes exception to the GL and does not state an acceptable alternate position, the staff cannot conclude that the RIP CAP plan is acceptable for commercial grade dedication. As stated previously, the staff recommends that TVA revise the RIP CAP plan to reflect the actual procurement practices in-place at WBN.

4.1 Procurement Package Review

Prior to the on-site inspection, the NRC inspection team requested that TVA provide lists of PEG packages addressing all TVA QA Level I materials (Appendix B, safety-related); all TVA QA Level II materials (dedicated CGIs); and all TVA QA Level III materials (materials used in safety-related systems that do not have a safety function), which have been purchased since June 5, 1991, the inception of TVA's new procurement program.

The team selected a sample of PEG packages from these lists for review and organized the packages by discipline into electrical and instrumentation, mechanical, and materials (including lubricants). The PEG packages selected were to contain the appropriate purchasing, engineering, and receipt inspection documentation. TVA provided associated commercial grade survey reports in those instances when it used acceptance method 2, "Commercial-Grade Survey of Supplier," of Electric Power Research Institute (EPRI) NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07)." The inspection team selected the following PEG packages for review:

QA Level I

- PEG Package 9400023342 (RIMS No. T49940401828), Revision 0, April 1, 1994, for an ITT Barton transmitter, part number 764.
- PEG Package 9400018340 (RIMS No. T49940311858), Revision 0, March 11, 1994, for Class 1E motor inspection/repair services.
- PEG Package 4730-1-01, Revision 8 (purchase order (PO) 93N3F-41936D-01, RD 363172 and RD 1025437), for metallic pipe.
- PEG Package 9300002498 (RIMS No. T4992113835), November 23, 1993, for a 90 degree, 1.5-inch, 3000 pound, pipe elbow supplied in accordance with the requirements of Section III, Class 2, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.
- PEG Package 9300006087 (RIMS No. 49930120800), for TVA PO No. P-93PJX-36732H-000, April 22, 1993, for various shapes and sizes of structural steel products purchased from Mid-South Nuclear, Incorporated.
- PEG Package 9400020408 (RIMS T49940317808), March 17, 1994, for ASCO solenoid valves.
- PEG Package 9400018339 (RIMS T49940312803), March 11, 1994, for General Electric (GE) power supply.
- PEG Package 9308024 (RIMS No. T49930924807), September 23, 1993, for various WBN System 31 instruments.
- PEG Package 9400002182, March 31, 1994, for various types and quantities of molded-case circuit breakers, manufactured by United Controls, Incorporated, used for both safety-related and nonsafety-

related applications.

QA Level II

- PEG Package 9300016028 (RIMS No. T49940312824), Revision 0, March 12, 1994, for an ITT Barton flow switch, part number 288A.
- PEG Package 9400018459 (RIMS No. T49940309810), Revision 0, March 9, 1994 for Amphenol connector, part number MS 3106A-14S-7S.
- PEG Package 922236 (RIMS No. T49930415858), Revision 0, April 14, 1994, for a Sorrento Electronics, Incorporated, flow meter part number 02815067-001.
- PEG Package 9400012490 (RIMS No. T49940203922), Revision 0, February 3, 1994, for Mallory capacitors.
- PEG Package 9300009828 (RIMS No. T49930318808), March 4, 1993, for a safety relief valve.
- PEG Package AND037P (RIMS B26910820640), August 1990, for a rotary oil pump for circulating oil through the control room chiller compressors.
- PEG Package 9400004823 (RIMS T49940202969), February 2, 1994, for a Gould-Shawmut fuse.
- PEG Package 912816 (RIMS T49920131812), January 31, 1992, for a Foxboro saturable reactor.
- PEG Package 9300000184 (RIMS T49940302805), March 1, 1994, for a Gould-Shawmut fuse holder (Block).
- PEG Package 920013, June 9, 1992, for 125-Volt dc (Vdc) closing spring charging motors, part number 0105C9393P002, for GE model AM-7.2-500-6HD Magne-Blast circuit breakers used in 6.9-kV, Class IE service.

QA Level III

- PEG Package 9400023282 (RIMS No. T49940402832), Revision 0, April 2, 1994, for Sorrento Electronics, Incorporated, read-out module, part number 03573000-004.
- PEG Package 9400005161 (RIMS No. T49940409852), Revision 0, April 8, 1994, for a GE transmitter, part number 50-555-111BDAA4WBE.
- PEG Package 9400021420 (RIMS No. T49940322854), Revision 0, March 22, 1994, for an ASCO solenoid valve, part number 8316G54E.
- PEG Package 9400011803, Revision 0, (PO P-90NJC-44792B-001, Release No. 1010689) for a ball valve.

- PEG Package 9400013143 (RIMS No. T49940212841), February 12, 1994, for valve parts for gate valves.
- PEG Package 9400012190 (RIMS No. T49940201847), February 1, 1994, for a stem assembly with disc for a 2-inch angle valve.
- PEG Package 9400014204 (RIMS No. T49940216841), February 16, 1994, for a machined shaft sleeve key for safety injection pumps.
- PEG Package 4821302, May 22, 1992, for non-critical Limitorque parts. The part in question is a metallic clutch ring in the clutch mechanism that allows manual or electric operation of the actuator.

The team reviewed the information contained in the PEG packages in order to establish that technical and quality requirements had been adequately translated into the appropriate procurements documents, to verify that the bases for the safety classification of the structures, systems, and components could be traced to the corresponding design bases documents, and to confirm that dedication activities conformed to the applicable regulatory criteria and guidance. The inspection team found that certain packages did not include information of sufficient detail to support the conclusions stated. The team confirmed that the PEG packages complied with TVA procedures and the appropriate regulatory requirements except as discussed below.

- PEG Package 9400018459 - This QA Level II package, March 9, 1994 (RIMS No. T49940309810), documented TVA's dedication of a standard, straight plug shell, three-contact, Amphenol connector (part number MS 3106A-14S-7S). The package documented the evaluation of indeterminate material stored in TIIC AXB4794 which had been identified as a suitable replacement for unique identifier (UNID) 1-SM-046-0057B-S. The PEG package described the "Host Equipment Description/UNID(S)" as "Various Class 1E and Non-1E Instruments and Devices, UNID 1-SM-046-0057B-S Typical," and the "Host Equipment Safety function(s)" was documented as "Provide Signal."

The "Critical Characteristics For Item Functions" portion of the Technical Evaluation Form (TEF) identified the following as critical characteristics for this item: part number, as stated; configuration, vendor catalog pin location, nominal dimensions are acceptable; the presence of the Amphenol logo on a blue insert; and insulation resistance, short circuit across pins, open circuit pin and case. All critical characteristics were to be verified during receipt inspection and the bases for accepting these critical characteristics in establishing the ability of the connector (or according to the procurement package, the "Host Equipment") to "Provide Signal" were identified as "Contract," "Catalog Excerpts," and "Engineering Judgement," respectively.

The manufacturer's published catalog referenced in the package stated that standard Mil-Spec contacts are silver plated and have pre-tinned solder pockets, the 97 prefix indicates an Amphenol design, and they may have special purpose contacts, variations from

Mil-Spec requirements, or a feature which has been assigned a Mil-Spec. However, these differences in design appear not to have been considered by the engineer when he concluded, on page 6 of 22 of the PEG package, that part number MS 3106A-14S-7S is the same as 973106A-14S-7S, except it's built to a Mil-Spec, and that both are acceptable. The basis for this conclusion was not documented in the package, nor was there documentation to support the conclusion that the connector was a "like-for-like" replacement.

Furthermore, since EPRI NP-5652, acceptance Method 2, was not part of the dedication, TVA could not assume that the connector(s) conformed to the design and performance and fabrication requirements of a given Mil-Spec as the manufacturer's published catalog stated. The inspection team concluded that the procurement package included no information on (1) the frequency and amplitude of the signal expected to be carried through the connector, (2) how the value of the impedance measured across the contacts or pins correlated to the transmission ability of the connector, (3) the function performed by the connector within the system (later identified as the EG-R actuator in the Woodward governor control circuit of the turbine driven auxiliary feedwater pump), (4) the design bases documents or sources that were reviewed or consulted to ensure that the design integrity of the host system was not degraded below an acceptable level and, (5) the basis for the engineer's determination that the part number differences were acceptable.

During the inspection, the inspection team asked TVA to clarify the basis for the engineer's conclusions that a resistance test yielding a short circuit across the connector's pins would demonstrate the ability of the connector to carry the required safety-related signal, and that the Amphenol connector being replaced was a "like-for-like" replacement of the original. TVA agreed to amend the package to resolve these issues. As a result, this item is designated as Inspector Followup Item 94-201-11.

PEG Package 9400018340 - This QA Level I package (RIMS No. T49940311858), Revision 0, March 11, 1994 (which supersedes RIMS No. T49911127964), documents TVA's procurement of refurbishment services from TVA's Power Service Shops (PSS), Muscle Shoals facility, for a GE 250-horsepower, 460-Volt-ac (Vac), 3-Phase, Class 1E Motor used in the main control room chiller package A-A. Page 2 of 4 of the TEF stated that the motor shown on the attached form was reworked by PSS on shop order 92-7609 and that the PEG originating document for transfer of the motor was RIMS No. T49911127964. It was also stated that the motor, UNID 0-MTR-31-80/2-A, failed after installation and initial run tests. This PEG package returned the subject motor to PSS for disassembly and examination to determine the cause of motor failure and to initiate any needed repairs.

Examination of RIMS No. T49911127964 (PEG Package No. 920953, November 27, 1991), revealed that the technical and quality requirements for this service had been established and documented under RIMS No.

B26891031760 (Nuclear Engineering - Contract Engineering Group, EPG Package 8903107E, October 31, 1989). Attachment 1 to PEG Package 920953 states, The motor specified for transfer to Muscle Shoals for rewind (see attached 2652) is identical to the motor transferred for rewind by PEG Package B26891031760 (which was supplemented by B26901207779). All characteristics and specifications contained in B26891031760 and B26901207779 still apply. Page 12 of 20 of EPG Package 8903107E documents the analysis upon which the engineer established how seismic qualification of the refurbished motor would be maintained and listed motor performance acceptance criteria which, once confirmed by testing at Muscle Shoals, would confirm that the rewound motor meets NEMA Design B performance requirements. The documented results of the tests would be evaluated before receipt inspection at WBN.

The inspection team could not identify from the information provided by TVA, that the test and weight data required to be furnished by Muscle Shoals in accordance with EPG Package 8903107E had been "evaluated prior to receipt inspection" by the responsible organization. Additionally, it could not be verified that once the motor had prematurely failed after installation and initial run tests that an examination or evaluation had been performed by the responsible organization to determine the cause of motor failure and to initiate corrective actions accordingly.

During the inspection, Problem Evaluation Report (PER) WBNPER940354, Revision 0, July 14, 1994, was initiated by TVA to address these deficiencies. TVA's failure to review test results that were part of safety-related activities intended to assure that design integrity of a Class 1E motor was maintained; provide documentation to support that appropriate corrective actions had been taken once a failure occurred shortly after post-installation tests; and to demonstrate suitability of application, was identified as Deficiency 94-201-D1.

- PEG Package 9400002182 - Under Contract No. P94N3F-102173-000, TVA procured for WBN from United Controls Incorporated (UCI) several types and quantities of molded-case circuit breakers (MCCBs), some for nonsafety-related applications, designated QA Level III, and some for safety-related applications designated QA Level I, which were all evaluated under different parts of this PEG package.

The QA Level III portion of the package covered contract Item 1 (three Westinghouse LA3400PR MCCBs, TIIC BTN245B), Item 3 (two Westinghouse JA3175W MCCBs, TIIC BVA124F) and Item 4 (one Westinghouse EHB2060 MCCB, TIIC BTW598A). In UCI COC for shop order 10QX002645, dated January 19, 1994, UCI stated that the testing was done in accordance with UCI's nuclear qualification test procedure NQTP-1816, which was to have been prepared per the TVA PO based on TVA/WBN Maintenance Instruction (MI) 57.27, Revision 11. Upon review of this portion of the PEG package, deficiencies in the apparent test methods and documentation provided with a COC from UCI were identified. However, because the items concerned were not installed, nor intended for safety-related service, and as verified by warehouse inspection were appropriately tagged, the

inspection team reviewed only those items on this PEG package (and contract) that were intended and tagged for safety-related applications.

Item 2 (one ITE/Siemens FXD63B100, TIIC BTK256M), and Items 5 through 8 (two ITE/Siemens ED63A050, TIIC BRH828J; six Square D FA26020AC, TIIC BWF356Q; three ITE/Siemens ED63A025, TIIC BTK204A; and three ITE/Siemens ED63A010 MCCBs, TIIC BVB013A) of TVA Contract P94N3F-102173-000, were all designated QA Level I and tested by UCI using the same procedure as the QA Level III MCCBs according to the associated UCI COCs (same shop order, but dated December 30, 1993) provided for these contract items.

During review of the UCI data sheets in the package (marked NQTP-1816.1.2), the TVA-approved UCI test procedure NQTP-1816.1.1, Revision 4, dated February 9, 1993, and the TVA/WBN test specification, MI-57.27, on which the NQTP was based, deficiencies were noted that indicated that MI-57.27 did not adequately translate design basis requirements for the intended MCCB applications into design output documents. As a result, a proper review for suitability of application would not have been performed.

The following are examples of inadequate translation of design basis requirements into design output documents in that the TVA test specification (MI-57.27, Rev. 11) on which the UCI test procedure was based would not be adequate to verify that the MCCBs would perform their safety functions under all design basis conditions:

- UCI's test procedure called for testing shunt trips at 120 Vac, but did not address 125 Vdc or degraded voltage conditions (as low as 90 Vdc at some plants) and equalizer battery charge voltage (as high as 140 Vdc). TVA stated in response that "no degraded voltage should exist when the shunt trip is used." However, during this inspection, TVA did not provide documented objective evidence to demonstrate that no shunt trips in MCCBs tested under the UCI test procedure (or any others in the plant tested in a similar manner) would ever be required to operate to perform their safety function under degraded (or excess) voltage conditions.
- Manufacturers' time-current characteristic curves for the inverse-time thermal overload trip response of the breakers under test were not identified, nor were curves from TVA/WBN's TI-108 curve book as required by MI-57.27. Therefore, the source of the acceptable trip times for the thermal overload trip tests listed on the UCI data sheets was not documented. This was a concern particularly with ITE/Siemens breakers because of past problems with this manufacturer as discussed in NRC Information Notice 85-16, "Time/Current Curve Discrepancy of ITE/Siemens-Allis Molded Case Circuit Breaker," dated February 27, 1985, and in NRC Information Notice 89-21, "Changes in Performance Characteristics of Molded Case Circuit Breakers," February 27, 1989. TVA is verifying that proper curves were used for the WBN design basis and will document the results.

- The full-load hold-in test of 30-minute duration was not consistent with industry standards (e.g., NEMA AB 4-1991 or UL-489) or practice in which MCCBs would normally be loaded 1 or 2 hours, depending on rating. TVA stated that the 30-minute test was reasonable based on the existing loads relative to the continuous current rating of the breaker or the nominal rating of its overload trip unit. However, there was no documented analysis to show adequate margin between breaker ratings and existing loads, accounting for worst-case service conditions, nor was the question of future additional loads addressed.
- Step 1.2.6, the instantaneous magnetic trip test instructions, of the test procedure stated that test current values are "typically" 1.5 times the trip setting of an adjustable magnetic-only MCCB or 15 times rated current for a thermal-magnetic MCCB and "shall be within the range stated on the trip curves or 15 times the rated current." According to WBN design specification WB-DC-30-28, "Low and Medium Voltage Power Systems" (Revision 9, December 29, 1993), the [safety] functions of MCCBs in the Class 1E 480-Vac shutdown boards is to provide fault (short circuit) protection for their associated branch circuits (cables), to provide selective coordination of their instantaneous trips with the short time trips of their upstream feeder breakers (Westinghouse DS type metalclad air circuit breakers), and to allow normal starting of their loads without premature tripping.

However, the procedures and acceptance criteria in the test procedure (which reflected those of TVA MI-57.27, Revision 11) did not provide for testing the adjustable magnetic trips at both the low and high ends (or recording this data) or even at the expected setting values based on end use. They did not provide for verifying that the MCCBs would not trip at current values lower than the lower tolerance limit of the trip curves with which they were selected, and did not in all cases ensure that the MCCBs would trip before the feeder breaker (i.e. trip in less than 70 milliseconds (ms) at current values near the pickup of the feeder short time trip). Also, the test current value of 15 times rated current was too high to provide meaningful performance data to verify that the MCCBs would trip within their design current ranges and times indicated on the trip curves, over their adjustable range where applicable (with appropriate tolerance for field testing). Therefore it was not clear, for example, that in all cases, insulation thermal limits of associated protected cabling would not be exceeded if MCCB ratings and settings for cable protection were selected on the basis of the trip curves.

The procedural deficiencies cited above were attributed to the failure to adequately translate design basis requirements into design output documents (i.e., instructions, procedures and drawings). As stated previously, TVA-provided specification, MI-57.27, Revision 11, formed the basis for UCI test procedure NQTP-1816. TVA's failure to adequately translate design basis requirements (e.g., system design voltage ranges

or variations) into design output documents (i.e., instructions, procedures and drawings), resulted in an inadequate review for suitability for application. As a result, this item is identified as Deficiency 94-201-D2.

In addition to the deficiencies in that test procedure described above, which were reflective of provisions of the MI, the MI itself contained the following discrepancies:

- The note below Paragraph 5.7 stated that inrush current withstand capability (e.g., during motor starts) is verified by so-called circuit function test (presumably a post installation test). This statement appeared to be a justification for not performing the magnetic trip test in a manner that would verify that the breaker will not trip below the lower tolerance limit of its magnetic trip band or setting during an inrush transient. The post-installation test is normally performed to provide a final check of MCCB reliability and for making fine adjustments to the magnetic trip setting for adjustable MCCBs as required (providing a sufficient number of starts are performed to eliminate the random asymmetrical current (DC offset) factor). However, an MCCB procurement acceptance test is supposed to verify the breaker to be performing within its design parameters before releasing it to the plant for testing and cycling plant equipment unnecessarily or requiring excessive adjustment.
- Caution below Step 6.6.2 [18] to stop test if breaker doesn't trip in ten seconds is too long to wait at that high current level.
- The 10-percent increase adjustment in Step 6.2.2 [21] was expressed as a "factor of 0.1" instead of 1.1.
- Section 7.0, Step [3] at the end of the test procedure calls for verifying that all test equipment is within its calibration period. Checking this after the test instead of before may result in having to perform the test again.
- The "trip free" function of the breakers was not checked.

The procedural deficiencies cited above involved wording that would not as written, and with the presumed verbatim compliance, accomplish their intended purposes or were deviations from accepted industry practice without apparent reason. The instructions used for this procurement were not appropriate to the circumstances, lacking appropriate qualitative and quantitative acceptance criteria to ensure that the breaker would perform its safety-related function and not fail in a manner adverse to safety under all design basis conditions. Accordingly, these procedural deficiencies are identified as part of Deficiency 94-201-D9.

In response to the above concerns, TVA provided information that satisfactorily addressed the issue of premature tripping for future testing with its revised procedure. TVA also provided analysis to show that selective coordination should be maintained with the particular design and conditions at WBN. It was still not clear, however, that TVA had demonstrated that all protected circuits would in fact be protected under all design basis conditions.

Although many of the problems found with Revision 11 of MI-57.27 were corrected in Revision 12, the test procedure used to test these breakers (and presumably, others in previous procurements) was based on Revision 11, and therefore, UCI did not adequately test the breakers described in PEG Package 940002182 for either specific or general purpose Class 1E service at WBN. At the time of this inspection, Revision 12 of MI-57.27 was already in use. However, the team did not address within the scope of this inspection the question of the adequacy of testing of other breakers tested under Revision 11.

In addition to the apparent deficiencies identified in PEG Package 940002182 described above, the following are instances in which objective evidence of quality furnished by the supplier was not adequately evaluated. TVA did not in all instances ensure that purchased material, equipment and services met the original procurement (design) requirements.

- No documentation in the package that breaker nameplate data is fully recorded by vendor as called for in the test specification.
- The information referred to as the Underwriter's Laboratories (UL) "rating" is not a rating per se and referring to it as such could be misleading. The information that is checked as one indicator of authentic components is the UL "listing" number and "issue" number, normally printed on the MCCB label decal. The issue number is traceable to the general design and type of the MCCB. The issue number should correspond to a period of time following original certification of the design, during which periodic followup testing is done, and covers date codes as should be identified in the manufacturer's records.
- Paragraph 1.2.3.2 of test procedure NQTP-1816 required that the tester record the applicable data for insulation resistance (IR) tests, yet no data was recorded; only subjective "Y" or "N" check blocks are provided on the data sheet.

It was also noted that the UCI data sheet gave the units of maximum acceptable and actual trip time in terms of Hertz, the unit of frequency or cycles per second; whereas the appropriate units in which instantaneous trip times are expressed are a number of cycles or a time interval in milliseconds (ms).

- Review of the data sheets for this job indicated that the minimum current for the magnetic trip test was to be set at 15 times rated current (not within the trip curves) for thermal-magnetic MCCBs. Although this current level is itself too high for a valid test of circuit fault protective function, in several cases the actual test current applied to thermal-magnetic MCCBs was more than 1.5 times the 15-times rated current value. For example, for serial number 2645-10-6, a 20-amp thermal-magnetic MCCB, minimum amps were listed as 300, with 476 amps applied. For some adjustable magnetic-only MCCBs, test currents of 15 times the setting, far above the tolerance band, were applied. For example, for serial number 2645-12-1, a 10-amp, adjustable magnetic-only MCCB, minimum amps were specified at 150 with 151 applied. These values, for either types of MCCBs, were not only much too high for meaningful results, but also indicated a lack of adherence based on an apparently inadequate understanding of the UCI test procedure or the TVA MI on which it was based.

The above instances in which objective evidence of quality furnished by the supplier was not adequately evaluated, and in which TVA did not in all instances ensure that purchased material, equipment and services met the original procurement (design) requirements, are identified as Deficiency 94-201-D3.

- PEG Package *EPEG 920013 - (*EPEG was an interim designation). This QA Level II package dedicated 125-Vdc closing spring charging motors, GE part number 0105C9393P002 (or 2B also acceptable per PEG Package 927254) for GE model AM-7.2-500-6HD Magne-Blast circuit breakers used in 6.9-Kv, Class 1E service at WBN. Five of these motors were procured at the old QA Level II (CGIs, by part number only) under TVA contract 85PN8-357877-02, dated January 17, 1985, a copy of which was in the package. The PEG package identified one additional previous replacement parts contract, 79P19-270009, but did not state the quantity of motors received or their disposition, nor was a copy of the contract included.

According to Form 575N issue document number A118749 in the file, one of the five motors from contract 85PN8-357877-02 was installed under MI 57.1 (no work order number given) in breaker serial number 256A4605-015 designated UNID "2-BKR-211-Comp 8." As part of the RIP CAP plan review discussed later in this report, Previous Procurement Substantiation Process (PPSP) 27724 downgraded this motor from QA Level II to QA Level III rather than dedicating it. The original QA Level II TIIC, AFT403B, was changed to a QA Level III TIIC, BPL195W, per PEG package 90-2728 for motor brushes. However, as part of the dedication process, EPEG called for changing the QA Level III TIIC for the motors back to QA Level II TIIC, AFT403B, and stated that future procurements would be QA Level I.

EPEG 920013 included a copy of work order (WO) 92-05903-00, October 15, 1992, which documented the dedication testing of six motors; five from nuclear stores and one installed in breaker 0-BKR-569-4603017-S, designated with a UNID of 1-BKR-211-A/5-A. Only five motors had been procured on contract number 85PN8-357877-02 and one of those had been

installed in 2-BKR-211-Comp 8 which for an unknown reason had not been included in the testing. The inspection team reviewed WO 91-00149-00 which documented the installation of TIIC, BPL195W, charging motor issued on Form 575N B05606 in breaker 1-BKR-211-A/5-A, in compartment 5 of 6900V shutdown board 1A-A, GE breaker serial number 256A4603-017. The issue record for this WO, 575N No. B05606, August 27, 1991, identified the same breaker location and showed procurement document number 357877. This indicated that four of the motors tested (three in stores and one in 1-BKR-211-A/5-A) had been procured under 85PN8-357877-02.

Therefore, presumably the other two of the motors in stores were from contract 79P19-270009; although, EPEG 920013 did not specifically identify their origin. The EPEG 920013 folder also contained an on-hand inventory adjustment form, document number 62-94-16(9) (last digit appears to be 9), January 29, 1994, that indicated that two AFT403B motors were to be surplused to close inspection report WBN-SWEC-R93.2, but this disposition was not explained or mentioned elsewhere in the file.

Although the dedication of the motors under EPEG 920013 appeared to be satisfactory, the motor installed in 2-BKR-211-Comp 8, erroneously downgraded to QA Level III in PPSP 27724, had at the time of the inspection not yet been dedicated (Refer to report section 5.3, Deficiency 94-201-D6). Furthermore, the question of the origin of two of the motors tested and the status of the two surplused remain unconfirmed and the existence of one or more additional procurements or additional unaccounted for motors procured under Contract 79P19-270009, which the inspectors did not review while on site, has not been ruled out.

One additional discrepancy identified was that the breaker with UNID 1-BKR-211-A/5-A, in which one of the tested motors had been installed, was described in the file as the supply breaker to 480V shutdown transformer 1A-A (in compartment 5 of 6900V shutdown board 1A-A); whereas, configuration control drawing 1-45W724-1, Revision 12, dated September 21, 1990, reportedly the effective, as-built, revision, showed the UNID of the breaker in compartment 5 and with this same load, but with a system designator of 212 (480V Class 1E distribution) instead of system 211 (6900V Class 1E distribution system) used in the PEG package. Although the 1990 drawing designation is consistent with TVA's practice of redesignating breakers with the system designator of their loads rather than the system designator of the distribution system of which they are a part, the file did not agree with what was supposed to be the as-built drawing.

The concern therefore remains unresolved regarding the disposition of any other charging motors that may have been procured in a similar manner, installed in safety-related breakers that must operate more than once to perform their safety functions, but not dedicated due to misidentification and/or misclassification. As a result of this ambiguity and the potential for more unaccounted for motors, this item

is designated as Unresolved Item 94-201-U1 pending further review by the licensee.

- PEG Package 4821-3-02 - This QA Level III package, "Limiterque Parts/Non Critical," May 22, 1992, was supplemented by PEG Package 9400024180 for Limitorque parts. The part in question was a clutch ring which functions as part of the clutch mechanism that allows manual or electric operation of the actuator. When the manual operation lever is depressed and latched, the actuator is set up for manual handwheel operation. Return to electric motor operation is effected automatically by the clutch mechanism upon the next motor operation, which at the same time disengages the handwheel. This and other limiterque spare parts were purchased on TVA Indefinite Quantity and Time (IQT) contract 86XNQ-838100. The item in question was classified nonsafety-related by TVA based on its "non-critical" designation on Limitorque's "Critical Component Document," EC-0001, Revision 2, October 9, 1990.

According to a telephone conversation with Limitorque, TVA and members of the inspection team during the inspection, the clutch ring is considered to have an active safety function in that it helps to effect the automatic return to motor operation after manual operation. However, Limitorque explained that the clutch ring is a sturdy piece of well lubricated steel that is under very low stress and therefore is considered by Limitorque to have no credible failure modes that would impair its safety function, nor is its application very sensitive to dimensional tolerances or material considerations. On this basis, Limitorque determined it to be non-critical.

This rationale is not inconsistent with the guidance in Appendix B of EPRI NP-5652 on part classification. EPRI NP-5652 was conditionally endorsed by the NRC in GL 89-02 without taking exception to this reasoning. Therefore it would not be inconsistent with the promulgated NRC staff position on this subject to accept the Limitorque rationale in this particular case only. The NRC inspection team pointed out to TVA that they have the ultimate responsibility for proper safety classification; and that while it is not unreasonable to start with a manufacturer's functional part classification basis as a basic guideline, the specific plant application requirements that the component manufacturer may not consider must always be taken into account.

- PEG Package 9300006087 - This package, RIMS No. 49930120800, TVA PO P-93PJX-36732H-000, April 22, 1993, was for various shapes and sizes of structural steel products purchased from Mid-South Nuclear, Incorporated (MSN). The same TVA PO was also reviewed by the NRC during its inspection at MSN in April 1994 (NRC Inspection Report 99901270/94-01, dated March 8, 1994) and concluded that MSN did not adequately dedicate structural steel items supplied to TVA. TVA actions to address the NRC's findings at MSN are as follows:

- Following its review of the NRC inspection report, TVA performed an audit at MSN in March 1994, and issued findings against MSN's CGI dedication activities. Several of these findings identified discrepancies in MSN's selection of critical characteristics and the lack of adequate documented technical justification to reasonably assure that the material furnished met the designated material specification requirements. TVA also performed an evaluation of the dedication activities by MSN and requested additional testing and information from MSN for the material supplied.
- TVA accepted MSN's response and corrective action to its findings and informed the NRC that MSN has improved and upgraded its CGI dedication activities to a level acceptable to TVA.
- TVA's audit checklist for Appendix B audits contains specific requirements to review CGI dedication activities if performed by the supplier. TVA performed its initial audit at MSN in July 1992 and qualified MSN to supply ASME Code and non-code safety-related material. MSN's QA program, including its controls for CGI dedication, were reviewed by TVA during this audit.
- TVA's implementation audit of MSN's QA program in July 1993 included a review of its CGI dedication activities. However at the time, MSN was only processing and dedicating sheet metal. During the audit TVA reviewed several sheet metal dedication activities and MSN's ASME Section III Code material upgrade activities, but did not review any other CGI dedications.
- TVA informed the NRC inspection team that it has developed a new standard procurement clause applicable for steel products such as those supplied by MSN. The new clause would require that the supplier assure that the material meets the requirements of the applicable material specification.

Several items supplied by MSN on TVA PO P-93JX-36732H-00 were reviewed during the inspection including verification of heat numbers, material traceability, and observation of hardness testing of MSN products. No deficiencies were identified.

Within the procurement package review part of the inspection, the inspection team identified four deficiencies, one unresolved item and one inspector followup item.

5 PLANT INSTALLED ITEMS VIA PREVIOUS MAINTENANCE ACTIVITIES REPLACEMENT ITEMS PROGRAM (RIP) (TI 2512/27, 35065, 35746, 35747)

In 1989 WBN implemented a Quality Release Program that required all material released to the plant to have a PEG evaluation package. Prior to the implementation of this program, the current inventory of material was evaluated for acceptability. The TVA RIP CAP contractor evaluated the procurement (and dedication where applicable) of the subject items per Site Procedure SSP-10.C and documented each evaluation in a file called a Previous

Procurement Substantiation Process Package (PPSP). However, due to the large backlog of evaluations, it was determined that material could be released before it had been evaluated and that this released material would be tracked using a Release Tracking Log (RTL) and evaluated at a later date. The RTL was in effect from about June 1989 through May 1990. In order to determine the adequacy of selected PPSPs, particularly those in the RTL, the inspection team reviewed TVA POs, vendor documentation, receiving inspection reports, audit and surveillance reports, material issue records, engineering evaluations, and quality records and also discussed the contents of these packages with appropriate TVA personnel in each discipline.

5.1 QA Level I PPSP Review

The following PPSPs were reviewed and found acceptable for evaluating the item's specific use at WBN:

- PPSP 28824, 40854 and 40693 - Metallic bar
- PPSP 27152 - Limitorque parts (clutch, sliding sleeve and shaft)
- PPSP 28194 - Butterfly valve
- PPSP 29210 - Valve stem with disc assembly
- PPSP 27725 - Speed changer motor for a Woodward governor
- PPSP 28613 - Voltage regulator
- PPSP 28819 - GE, 6.9-Kv Magne-Blast circuit breakers
- PPSP 29380 - Limitorque limit switch gear assembly
- PPSP 28453 - Solid state relay stored in TIIC ARV824E

5.2 QA Level II PPSP Review

The following QA Level II PPSPs were reviewed and found acceptable for evaluating the item's specific use at Watts Bar:

- PPSP 27747 - Pipe clamp
- PPSP 29107 - Carbon steel plate
- PPSP 28630 - 7/8-inch diameter continuous threaded rod
- PPSP 29277 - Water pump assembly
- PPSP 20376 - Limitorque operator parts (gear and shaft)
- PPSP 29227 - W-2 type cell switch
- PPSP 40853 - Cartridge fuse stored in various TIICs
- PPSP 29102 - Westinghouse Type W-2 rotary switches
- PPSP 28079 - Limitorque limit switch gear assembly
- PPSP 28921 - Temperature transmitter

The following QA Level II PPSPs were reviewed and found to be deficient:

- PPSP 27975 - November 6, 1992, for a printed circuit board for a GEMAC power supply, TIIC AJW144B. The critical characteristics for acceptance related to normal operability for this item were reasonable and appeared to be satisfactorily verified by the post-installation test in accordance with the technical manuals. However, the seismic similarity analysis for this CGI was weak and illustrates a weakness in the RIP equivalency determination methodology (and implementation) as prescribed by SSP-10.C.

The seismic argument was based on two factors: (1) the assertion that the circuit board "does not provide seismic quality to the host," and (2) an equivalency (like-for-like) determination. The first point does not conclusively substantiate seismic adequacy (apparently using an approach similar to that of the Seismic Qualification Utility Group's Generic Implementation Procedure, GIP-2) because it neglects the potential for the item's introducing seismic fragility or sensitivity into the host device if, for example (assuming for this discussion equivalency to the original, presumably seismically qualified part), the replacement part has any loose parts or weak connections or solder joints, or the circuit board plug-in socket is not tight enough.

Second, the inspection team questioned the adequacy of the equivalency determination itself. The equivalency determination form, page 3 of 5 of Appendix B of SSP-10.C (Revision 1), lists several factors to be considered in the determination, but does not provide guidance on how to apply them, e.g, what to verify if the part number had not been the same (first factor), or what is required or can be used if the item was not procured from the original manufacturer or supplier (second factor, although not checked in this PPSP). Also, procuring a "commodity item" to a "national standard" (fourth factor, also not checked) would not necessarily ensure consistency of product attributes without some knowledge and verification of the manufacturer's controls on batch or lot homogeneity and traceability.

Finally, the third listed factor (also not checked in this PPSP) was the statement, "TVA had no reason to believe there were changes in the design, materials, or manufacturing processes of the item since the original item was procured." It was not clear, first, how a RIP evaluator could reasonably make such a determination. This assertion does not provide any conclusive, objective evidence that there were no such changes. NRC GL 91-05 promulgated the staff position that even for an item of the same part number (which may or may not indicate changes) that was not procured from the same source at the same time, a review of design, material and process changes would need to be made. That a purchaser merely "had no reason to believe" that there were changes clearly would not serve for such a review to support a like-for-like or equivalency determination.

The equivalency determination form then listed, appropriately, vendor contact and document review (and "other") as methods of determination, but again, no further guidance was provided. Hence, in this case, the reviewer simply listed a contract number as the sole reference document, then marked the rest of the form, which called for an equivalency determination to be made, and the Appendix completed, "NA." It was not clear how review of one contract, unless the item was received under an RD from an original IQT contract, could support an equivalency determination. The deficiencies with the Attachments to Procedure SSP-10.C discussed above, that it does not logically accomplish its stated purpose and is inconsistent with the guidance in GL 91-05, render that procedure inappropriate to the circumstances. Therefore they are identified as part of Deficiency 94-201-D9. In addition, as the PPSP

did not contain explanations of the identified technical concerns, this issue is designated as Inspector Followup Item 94-201-I2.

- PPSP 20406 - Dated October 12, 1993, for Limitorque motors, TIIC AVA550G. At the time of the inspection, it was not clear from documentation in the file whether the prescribed "stroke test" per the cited maintenance request (MR) ensured that the motor met design requirements (e.g., full-flow cutoff differential pressure capability). It was also not evident how the production dedicated motors were demonstrated to be similar to test specimens. Therefore, this item is considered Unresolved Item 94-201-U2.

- PPSP 27732 - Dated November 22, 1992, TIIC ALQ229B, for 125-Vdc trip coils, GE part number 6174582G01 (also used as closing spring release coil according to GE Instruction Book GEI-88763F), for GE AM-7.2-500-6HD Magne-Blast circuit breakers in WBN Class 1E 6900V shutdown boards. Five of the coils were procured QA Level II under TVA Contract 78P13-234973, received on GE Switchgear Products Department (Chattanooga) Packing Slip 19141, and receipt inspected on February 17, 1978 (Receipt Inspection Report (RIR) 152014). The PPSP associate, the contract number with what it calls a procurement reference item, N3-219-1, but the relationship of these two numbers was not evident. According to Form 575N 628809626, dated September 13, 1988, in the PPSP, one of the TIIC (ALQ229B) coils from contract N3-219-1 was installed under MR A605597 in breaker Serial No. 256A4603-013, UNID 2-BKR-211-2B/10B. The other issue document in the PPSP, Form 575N A118747, December 1, 1989, indicated that one of these coils, also from contract N3-219-1, was installed as a trip coil in breaker with serial number 256A4605-015, UNID 2-BKR-211/ERCW-DA (Note that this is the same breaker (also called 2-BKR-211/Comp 8) in which an undedicated charging motor was installed and downgraded under PPSP 27724).

The PPSP technical evaluation form (Appendix B of the package) listed voltage range (design criterion 70 to 140 Vdc) as a critical characteristic for design, citing TVA Specification 1765. The list of critical characteristics for acceptance (Appendix C) appropriately included voltage range, also specifying 70 to 140 Vdc for the acceptance criterion. The prescribed acceptance method was post-installation test and the PPSP took credit for MI 57.99.4 performed under MR A-605597 for verification that this critical characteristic was met. The copy of MR A605597 in the PPSP references Form 575N 628809626 and indicates that the coil was used to replace the breaker's closing coil in accordance with MI 57.99.4, Revision 11.

The data sheets from this MI are included in the MR package, but it was only stated that the breaker operated satisfactorily and the control voltage used was not mentioned. Also, there was no documentation in the file to dedicate the trip coil installed in 2-BKR-ERCW/D-A. The 575N which documented the issue of the trip coil for this breaker listed MI-57.1. TVA was unable to produce a copy of any revision of MI 57.99.4 to determine what control voltage, if any, had been specified by the procedure. The current version of MI-57.1, which superseded 57.99.4,

was reviewed in accordance with which the other coil had been installed. Revision 21 of MI 57.001, "6900 Volt Breaker Inspection," May 13, 1994, did not specify any particular control voltage to use when testing the trip coils or closing spring release coils of the breakers. Finally, neither the list of critical characteristics for design nor for acceptance included coil insulation or winding resistance.

Failure to perform an adequate review of the coils for suitability of application and the failure to verify that the item meets the procurement requirements expressed in the PPSP is designated as Deficiency 94-201-D4. The deficiency in MI-57.1 discussed above (lacking appropriate quantitative acceptance criteria) rendered it inappropriate to the circumstances. Accordingly, it is identified as part of Deficiency 94-201-D9.

PPSP 27732 also contained two more examples of the use of UNIDs that were inconsistent with the UNIDs on the drawings said by TVA to reflect as-built conditions. The breaker in which the closing coil was installed was identified in the PPSP as it was on the issue and work documents as 2-BKR-211-2B/10B; whereas, Configuration Control Drawing 1-45W724-4, "Wiring Diagrams, 6900V Shutdown Board 2B-B, Single Line," Revision 12, September 22, 1990, this breaker (supply to the motor of auxiliary feedwater pump 2B-B) as UNID 2-BKR-3-128B. The other issue document in the PPSP, but not mentioned in the PPSP procedure forms, identified the breaker in which the trip coil was installed as 2-BKR-211-ERCW/D-A (essential raw cooling water). This breaker (also called 2-BKR-211/Comp 8 in PPSP 27724), supply to the motor for ERCW pump D-A), on Configuration Control Drawing 1-45W724-3, "Wiring Diagrams, 6900V Shutdown Board 2A-A, Single Line," Revision 12, dated 9-22-90, was shown as 0-BKR-67-40-A.

- PPSP 27739 - Dated October 21, 1992, was for Telemechanique A203C48, NEMA Size 1, full-voltage, non-reversing, 3-pole, open configuration, motor starters (each comprised of a contactor, an overload relay, and an auxiliary switch) with 480-Vac coils. Four of the starters (TIIC AJP795T) had been purchased QA Level II on TVA Contract 87NLC381927, dated December 29, 1986, with two issued on Form 575Ns. The PPSP indicated that both of these had subsequently been replaced, leaving two, the disposition of which was not apparent in the PPSP. The PPSP indicated that eight AJP795Ts were later dedicated, qualified and supplied by Farwell & Hendricks (F&H), a third-party dedicater, under TVA Contract 91NNA42765C, dated February 26, 1991, Change 1, March 25, 1991 (item 3, QA Level I). According to F&H COC No. 80056.1, F&H Technical Procedure 3-001, Revision 0, was used and the results were recorded in Data Package 80056.1, said to be on file at F&H.

However, the PPSP did not show how F&H dedicated these CGIs. Although F&H had been audited by a Nuclear Procurement Issues Council (NUPIC) joint utility team, the PPSP did not indicate that WBN engineering had reviewed this specific data package. In particular, system interface requirements were not specified. For example, the control voltage used was not shown therefore it could not be determined whether the units

were tested with worst-case degraded/elevated voltage, highest ambient temperature, and other WBN-specific service conditions. Also, it was not clear how the production units were verified to be similar, particularly in terms of seismic performance/sensitivity to the prototype test specimens. The units' Seismically Sensitive Electrically Active (SSEA) status was not given.

The inspectors' concerns about the details of the F&H dedication arise from experience in recent years that indicates an increasing number of instances in which the dedications performed by third-party dedicators (who supply the material under Appendix B QA programs and accept 10 CFR Part 21 reporting responsibility) are not adequate to assure that the items are suitable for safety-related service under all of the customers' plant-specific design-basis conditions. Purchasers often have not completely or accurately communicated all the specific application requirements to the dedicator, typically ordering, as in this case by part number only. Nor have dedicators always adequately taken these plant-specific factors into account in identifying critical characteristics, verification methods, and acceptance criteria that are relevant to the particular plant application for which the items are being dedicated.

In several responses to inquiries on this matter, TVA did not provide information to address the inspector's specific questions (as documented in licensee transcriptions of the inspector's verbal questions) such as (1) How did F&H dedicate the starters (i.e., critical characteristics, verification methods, acceptance criteria, and results of tests and inspections) (2) How did F&H demonstrate similarity of the production units to the seismically tested prototype, (3) what was the extent of TVA's review of Data Package 80056.1, (4) what control voltage was used by F&H for testing the starters (pull-in and drop-out tests); and was this voltage consistent with or lower than the worst-case degraded voltage expected at WBN? Instead, TVA continued to restate its position that F&H was an audited and approved Appendix B/Part 21 supplier and therefore, subject to receipt inspection, the Telemechanique starters, were acceptable.

Review of the data sheets eventually sent to the inspector from F&H Data Package 800056.1 revealed that the pull-in and drop-out voltages were measured. Although the voltages appeared to the inspector to be adequate for most credible plant conditions, TVA would be expected to review them in light of actual WBN service conditions to confirm adequacy. A comparison of WBN service conditions to the manufacturer's specifications may already been done when the original starters were specified, but the actual tested starter performance apparently was never reviewed heretofore by TVA. Review of the data sheets also revealed that contrary to manufacturer's and general industry guidance, the critical characteristics of full-load hold-in capability and starter insulation resistance (of contactors, coil, and auxiliary devices) were not verified by F&H.

In addition, although, F&H may have verified the dimensions and resistance of separately dedicated overload heaters, measurement of these parameters for the heater elements installed in the overload relays in the starters in question was not documented in the data sheets provided for review. Furthermore, although the data sheets did indicate that the overload trip function was checked by passing three-times rated current through the heater and verifying a trip, the required time delay for the class of overload relay/starter was not checked. Based on this review, the inspector concluded that in this case, F&H had not adequately dedicated the starters for general purpose Class 1E service in a nuclear power plant.

Finally, the inspector reviewed the copy of the audit checklist used in the NUPIC audit of the vendor. The checklist was requested to supplement the reports of the audit and the supplemental surveillance (that only covered a Telemecanique overload heater) provided previously. The starters were being ordered, in addition to quality assurance (QA) and 10 CFR Part 21 (part 21) requirements, by part number only. Therefore, the inspector evaluated the audit reports to determine whether they verified that the vendor's program to dedicate the commercial grade starters (or other CGIs) included verification of all the design specifications associated with that model or part number upon which the licensee was relying to ensure suitability for safety service at WBN. The inspector could not conclude from this review that such verification was either programmatically required or routinely performed, or included in this case. Therefore, on this basis, the inspector determined that the audit was not adequate to support procurements in which no additional technical requirements other than part or model number and minimal description were imposed in procurement documents.

TVA may have reviewed the technical data on the model starters in question for suitability of application, and finding this acceptable, ordered the starters purely on the basis of part number, with Appendix B QA and Part 21 specified. However, it was apparent that TVA did not verify, either during the audit or surveillance of the vendor, nor by examination of products upon delivery, that all critical characteristics pertinent to WBN that were being relied upon on the basis of the design characteristics associated with the part number were being verified. This failure to verify that safety-related equipment meets the procurement requirements is designated as Deficiency 94-201-D5.

5.3 QA Level III PPSP Package Review

The following QA Level III PPSP packages were reviewed and found to be properly classified as nonsafety-related items. It is noted that these classifications were found acceptable only for their specific end use and if used in similar applications in other components and systems, such classification may change.

- PPSPs 27992 and 27905 - Stainless steel wire used to secure equipment
- PPSP 29149 - Pump wear ring
- PPSP 28019 - Limitorque operator handwheel clutch pinion
- PPSP 27067 - Rotary control switch stored in TIIC ACN683G.
- PPSP 27977 - Positioner/controller module stored in TIIC ARV417V.

The following QA Level III PPSP packages were reviewed and found to be inadequately dedicated:

- PPSP 28039 - Lockwire used to secure the set screw for motor pinions in certain Limitorque operators. During the inspection, TVA generated PER 940337 to document the inadequate technical evaluation of this PPSP. Based on the following discussion, the inspection team expressed a concern that failure of the lockwire could potentially affect the host component from performing its safety-related function and that the PPSP package should have been more perceptive with justification for not classifying the lockwire as QA Level II and dedicating it as a basic component. The PPSP did not specifically address the end use or function of the lockwire as discussed in Limitorque Maintenance Update 89-1 (TVA Document No. WBN-VTD-L200-0280, Revision 0, October 3, 1991). Part 1B, "Item Data," of Appendix B, Technical Evaluation, to PPSP 28039, states in part, that the purpose of the lockwire is to prevent nuts from becoming undone due to vibration; that the lockwire does not perform a safety-related function; and that the failure of the lockwire does not influence the ability of the host equipment (the Limitorque actuator) to perform its safety-related function.

The PPSP stated that the end use of the lockwire was to prevent a set screw, not a nut, from backing out. The set screw is used to fix the motor pinion once it is properly orientated on the actuator motor shaft. The inspection team also reviewed the Limitorque Maintenance Update and participated in a conference call with TVA and Limitorque to discuss the safety functions of the lockwire and the set screw. Limitorque confirmed that the motor pinion gear is attached to the motor shaft by means of a key (to transmit the rotary force) and a set screw (to prevent axial movement of the gear on the shaft) and stated that the set screws are normally set by staking, which for the size - 000 actuators, is sufficient to retain the set screw.

However for all other size actuators, the set screws cannot be staked due to the hardness of the steel motor pinion gears. To prevent these set screws from backing out, Limitorque recommends the use of thread locking compounds or lockwire inserted through grooves. Limitorque also stated that it intended to review its critical parts list and evaluate the need for classifying the lockwire and set screw as critical.

Page 17 of TVA Document WBN-VTD-L200-0280, lists the lockwire as 0.047-inch diameter wire. PPSP 28039, identifies the lockwire as 0.06-inch diameter (TIIC No. AXR 685K), however the correct TIIC for the 0.047-inch diameter stainless steel lockwire, according to TVA, should have been AXR 684M. TVA stated that it would document the use of the incorrect lockwire on the PER and appropriately disposition the PER once the Limitorque evaluation of the proper classification of the set screw and lockwire is completed. As a result of the misclassification of the lockwire, the inspection team selected two additional

QA Level III PPSP packages for wire and found that they were properly classified as QA Level III (nonsafety-related) and performed no safety function for their identified end use. Satisfactory resolution of the lockwire classification concerns is identified as Inspector Followup Item 94-201-I3.

- PPSP 27724 - Dated April 6, 1992, for a closing spring charging motor, GE part numbers 0105C9393P002 and 2B (original QA Level II TIIC AFT403B changed to QA Level III TIIC BPL195W per PEG package 90-2728 for motor brushes), for AM-7.2-500-6HD GE Magne-Blast circuit breaker, installed in breaker serial number 256A4605-015, designated UNID "2-BKR-211-Comp 8," per Form 575N A118749 and MI 57.1. This PPSP, on the same basis as the PEG package for the brushes, downgraded the motor from QA Level II (the old QA Level II at which it and four others were originally purchased under TVA contract number 85PN8-357877-02, dated January 17, 1985) to QA Level III.

The rationale documented was that (1) the motor was not supposed to impact automatic function of the safety-related (Class 1E) breaker, (2) the motor is isolated from Class 1E power supply by fuse, and (3) closing springs may be charged manually. This rationale did not support the assertion that the motors were not needed for the breaker to perform its safety function for several reasons. First, although the breaker will be capable of closing one time automatically once the closing spring has been charged by the motor, it may be required under certain design basis event (DBE) scenarios within the licensing basis of WBN to close more than once automatically. A normally open bus feeder breaker with a charged closing spring (exhibiting a spring charged white light on the switchboard), could be reasonably expected to close the breaker once upon demand such as an engineered safety feature (ESF) actuation signal, and although it may trip upon a subsequent loss of offsite power, it should not normally be required to reclose automatically during the DBE because the 6.9-Kv shutdown board should be re-energized under these circumstances by the emergency diesel generator (EDG). Therefore in such an instance, the closing spring charging motor would not be required to operate during the DBE and so would have no safety-related function.

However, a normally-open breaker supplying a Class 1E load from the 6.9-Kv bus would need to shut automatically upon an ESF actuation, but would be required to trip upon a subsequent loss of offsite power and then reclose automatically with a signal from the EDG load sequencer after the EDG had re-energized the bus. In this case the load breaker's motor would be required to operate during the DBE and perform a safety function and would be itself classified 1E accordingly.

Although the PPSP did not establish that the former scenario was the case for the installed motor, TVA's response to the inspection team's inquiry as to the purpose/function of breaker 2-BKR-211-Comp-8 was that it was the alternate power supply to 6.9-Kv shutdown board 2A-A from station supply transformer CSST D. This would justify downgrading the installed motor to QA Level III as was done by this PPSP. However, upon review of the associated 6.9-Kv distribution diagram, "Wiring Diagrams, 6900V Shutdown Board 2A-A, Single Line," drawing number 1-45W724-3, Revision 12, dated September 22, 1990, the inspector found that the breaker in compartment 8 (indicated by the "comp 8"

in the UNID) was in fact breaker 0-BKR-67-40-A, supplying motor 0-MTR-67-40-A for essential raw cooling water pump D-A. This breaker was a normally-open Class 1E load breaker and therefore its charging motor did have a safety function.

Second, the fact that closing springs may be charged manually is irrelevant because if the springs should be required to be recharged in order for the breaker to reclose during a DBE to perform a safety function, the motor would be required to operate to meet system safety function requirements and credit for manual charging could not be taken. Also, even for a motor that is determined to be non Class 1E, the PPSP did not establish that the fuses were proper Class 1E isolation devices. This charging motor was procured as a CGI, but never dedicated for the safety-related application in which it was being used. The improper quality level (safety class) downgrading by PPSP 27724 compounded the error. The misclassification resulted in failure to review the item for suitability of application as well as failure to verify that the item meets the procurement requirements. Accordingly, this issue is identified as Deficiency 94-201-D6.

- PPSP 28973 - Dated November 14, 1992, for a Westinghouse 592C114G01 direct trip actuator (DTA) for tripping Westinghouse Type DS-206 480-Vac metalclad air circuit breakers in response to signals from the breaker's Amptector overload trip unit. Two of these DTAs (TIIC AMN016P) were purchased QA Level II under contract 86PJH-357168 from Mills & Lupton Supply Company (M&L). The Westinghouse packing list, shipment no. 23771791, (referencing Westinghouse General Order C072644, the M&L order and the TVA contract) indicated the DTAs were shipped direct from the Westinghouse warehouse in Charlotte, North Carolina, to WBN, yet the PPSP stated the items were received from the distributor, not Westinghouse.

According to issue document Form 575N A132952, dated April 5, 1990, one of the DTAs from contract 357168 was issued for installation under MI 57.2 (no WO number given) in a breaker designated UNID 1-BKR-212-A1/8A-A, serial number 010, located in compartment 8A of Class 1E 480V shutdown board 1A1-A. Although configuration control (as-built) drawing no. 1-45W749-1 (Revision 20, dated 9-22-90), "Wiring Diagrams 480V Shutdown BD 1A1-A, Single Line," shows the breaker in compartment 8A as UNID 0-BKR-239-1-A, the 1992 PPSP still referred to it by its previous UNID using the system 212 designator instead of the load system 239 designator as apparently is the current practice. System 239 is WBN's 250-Vdc battery power supply system and this breaker is the alternate feeder to one of the 250-Vdc battery chargers.

However upon further review of the package, the inspectors identified that the justification for downgrading the DTA to QA Level III was in error. The rationale as documented in the PPSP was that although system 212 which the breaker was in is a safety-related system, its load in system 239 (250-Vdc battery charger) was a nonsafety-related load. Therefore, the PPSP preparer reasoned incorrectly that on this basis the breaker, and hence the DTA, could be considered nonsafety-related. This was incorrect because it did not take into account the fact that this circuit breaker must act as a Class 1E isolation device to clear faults on the nonsafety-related load and associated buswork and cabling protected by the breaker in order to protect the system

212 Class 1E 480-Vac bus. Fault tripping in this breaker is accomplished by means of the DTA; therefore the DTA must be considered to have a safety function. The fact that this PPSP downgraded the installed DTA to QA Level III without dedicating it means that it compounded the error of installing it without proper dedication initially, leaving an undedicated commercial grade item DTA of indeterminate quality in a safety-related application. Failure to review the item for suitability of application and failure to verify that the item meets the procurement requirements is identified as Deficiency 94-201-D7.

Subsequent to the inspection, TVA submitted information to the NRC that described its dedication plan for the DTA in question. Proposed Revision 1 of PPSP 28973, dated August 4, 1994 (through the independent reviewer, but not at that time approved), was included in the information submitted and was reviewed with some discrepancies noted:

- Attached to the revised PPSP was Work Request (WR) C152372, dated August 4, 1994. This WR, which only identified the breaker in which the DTA in question was installed as the component to be worked on, called out several tests. First, it specified testing in accordance with the vendor manual, Section 11-7, which would exercise the DTA. Then, it also called out testing insulation resistance (IR) and so-called "coil voltage." However, while an IR test of the DTA appeared to be appropriate, it was not clear from the WR exactly which component's IR was to be tested. The revised PPSP referenced MI-57.2 (Revision 15), used to test the breaker, ostensibly after installation of the replacement DTA, but the MI only covered insulation resistance measurement of the breaker's main poles. Section "6.10" of MI-57.2 was referenced in two places, but there was no Section 6.10 in the data package which only went up to Section 6.3.3. The only section with a similar number was Section 6.1.10, but it had nothing to do with the DTA which is operated in the overload trip tests of Section 6.13. In addition, the "coil voltage" test specified could not be found as such in either the referenced vendor manual, WBN-VTD-W120-3023, or in MI-57.2. It was not clear what coil voltage was to be tested, since the DTA is only energized by the Amptector trip unit in the breaker under sufficient overload or fault conditions.
- Page 4 of the revised PPSP, the like-for-like determination form, indicated (block checked) that "TVA had no reason to believe there were changes in the design, materials, or manufacturing process of the item since the original item was procured, yet Section 3-1.6 of the referenced vendor manual described several changes in the DTAs supplied, and it was not clear that all changes involved a change in part number. Also, the various possible DTAs are of different colors, slightly different terminal or pigtail configurations (one is not seismically qualified), yet the PPSP did not mention checking these attributes to ensure a DTA fully compatible with the Amptector trip unit in the breaker, particularly if the Amptector had been replaced.
- Page 5 of the revised PPSP, the technical evaluation form, listed part number, manufacturer, dimensions/configuration, coil resistance or continuity, and coil voltage as critical characteristics for design.

The list of critical characteristics for acceptance listed part number, manufacturer, dimensions/configuration, and continuity/coil voltage. First, for coil resistance, having been identified as a critical characteristic ("for design") there was no justification for not using it for acceptance, which is not consistent with GL 91-05. Also, it was not clear how "continuity/coil voltage" together as a critical characteristic for acceptance describes a meaningful test. Again, coil voltage was not explained. And finally, the insulation resistance test specified in the WR supposed to be used to complete the dedication, was not listed at all.

It is standard practice to measure IR among phases and to ground from each phase either with the breaker closed, or if open, on the load side and the line side, followed by a test of each pole from line to load (open). However, according to the completed MI in the PPSP, this test was performed with the breaker tripped (Step 6.1.13RR), yet only one set of readings for each pole to ground and to each other was recorded, and there were no IR readings recorded from line to load. This procedure did not appear to adequately check IR of the breaker and it would not ensure checking of DTA insulation resistance at all. These deficiencies rendered the MI inappropriate to the circumstances, lacking appropriate qualitative and quantitative acceptance criteria. Accordingly they are identified as part of Deficiency 94-201-D9.

5.4 Review of Generic Dedication Packages

The inspection team reviewed the list of bulk commodities listed in TVA Procedure SSP 10.C and based on discussions with RIP and PEG personnel, determined that TVA presently performs no generic dedications and that in its evaluation of plant installed items, only one generic technical evaluation was used for accepting installed fasteners. TVA stated that since the late 1970s both commercial and safety-related fasteners used at WBN were examined and tested to assure compliance to specification requirements.

TVA also performed overchecks and screened lots/heats of nonconforming fasteners as documented in nonconformance reports, Institute of Nuclear Power Operations (INPO) assessments, NRC inspection reports, TVA's response to NRC GL 87-02, fastener test results from TVA's test laboratories, and internal TVA memorandums for the period 1979 through 1989.

The inspectors also reviewed TVA Construction Specification G-53, "ASME Section III and Non-ASME Section III (including AISC, ANSI/ASME B31.1 and ANSI B31.5) Bolting Material." Selective revisions were reviewed and it was determined that the specification required overchecks of all fasteners since its issuance in 1978. Based on this review and discussions of the engineering justification of the WBN fastener program, it was concluded that TVA had an adequate basis for not testing fasteners under the RIP CAP plan.

5.5 Control of Product Contaminants

TVA-WBN through its Process Specification 4.M.1.1 and PF material purchase specifications identifies technical requirements for chemical limitations for elements such as halogens, sulfur and water leachable halides, for products

used in austenitic stainless steel components and systems. TVA has established a policy that based on satisfactory historical performance of suppliers of these type of items, no audits, surveys, surveillance of supplier or overchecks of the items received is necessary to ensure that these items meet PF chemistry limitation requirements.

The inspectors discussed with TVA the requirements of American National Standards Institute (ANSI) N45.2.13-1976, "Quality Assurance Requirement for Control of Procurement of Items and Services for Nuclear Power Plants," as endorsed by NRC Regulatory Guide 1.123-1977. TVA stated that Section 10.2, "Certificate of Conformance," was a permitted method for ensuring that an item meets specification. It was pointed out to TVA that Subparagraph F of Section 10.2 provides additional requirements and states, in part, that means shall be provided to verify the validity of supplier certificates and the effectiveness of the certification system. The inspectors concluded that even though the past performance of these suppliers may indicate satisfactory implementation of their QA program, Appendix B to 10 CFR Part 50 requires assessment of the effectiveness of the control of quality of these suppliers at some interval based on the importance, complexity, and quantity of the product received.

TVA General Specification G-29, Volume IV, Process Specification 4.M.1.1, "Material Fabrication and Handling Requirements for Austenitic Stainless Steel," Revision 17, dated March 25, 1994, and several lower tier process specifications such as PF-1060, "Purchase Specification for Gaskets," Revision 4, dated November 23, 1993, and PF-1061, "Purchase Specification for Lubricants Unrestricted Use on Stainless Steels Primary and Secondary Systems," Revision 3, dated July 31, 1992, permit the use of a COC from a supplier, not on TVA's ASL, certifying that actual chemistry test results meet the required product limitations required by the applicable TVA specification for certain products used in safety-related austenitic stainless steel systems and components. TVA representatives stated that they will review the policy for consistency with Section 10.2 of ANSI N45.2.13-1976, and will determine the need to revise its policy based on the results of that review. As a result, this item is identified as Inspector Followup Item 94-201-I4.

5.6 Replacement Items Installed by Previous Maintenance Activities

5.6.1 Maintenance History Data Base

To address replacement items installed by previous maintenance activities, WBN developed the Maintenance History Data Base (MHDB). This data base was developed by reviewing some 91,000 work control documents, surveillance instructions, and operating instructions. The work control documents were generated for (1) corrective maintenance, (2) preventative maintenance, (3) maintenance performed during surveillance testing, and (4) maintenance performed during plant modifications. The MHDB is no longer in use. However, it was useful to the 10 CFR 50.49 review task because of the extensive amount of data it contained which related to MRs and work requests (WRs), as well as the cross references between documents such as contract numbers and material issues tickets. The MHDB is now held in inactive status by the Information Systems Group.

These work control documents were issued in the time period between original system(s) turnover and January, 1989, the date when the Quality Release Program was implemented. The review of the work control documents was performed to determine (1) did the maintenance activity install parts? (2) Was the host equipment safety-related? and (3) Was the host equipment subject to 10 CFR 50.49 requirements. A "yes" answer to questions number 1 and 2 meant the work document would be added to the data base along with other information such as (1) MR Number, (2) Contract Number, (3) TIIC number, (4) 575N Number, and (5) Quality Classification. This data base made it possible for use in several cross reference checks. A "yes" answer to question number 3 allowed the grouping of the 10 CFR 50.49 equipment subset for later use.

The advantage of the data base information being available up front to the installed items evaluations is that the process started with a specific "end use," the unique identification number for the host equipment, and works backward via the work document and the issue document (Form 575N) to the procurement documents. Work control or 575Ns referencing unclassified equipment, or equipment designated by a temporary identification number, would precipitate a Pro-TIIC designation in the MHDB.

Items on the MHDB were reviewed, consolidated, evaluated and dispositioned. Old items identified as installed in host equipment subject to the requirements of 10 CFR 50.49 were evaluated for acceptability. The MHDB also identified a total of 139 consolidated safety-related potentially problem items that were not subject to the requirements of 10 CFR 50.49.

5.6.2 Problem TIICs

The population of maintenance installed items reviewed by TVA consisted of several incomplete TIIC items listed in the MHDB called Problem or Pro-TIICs. These items identify inconsistencies with the correlation of document identifiers such as the TIIC number on the MR or the Form 575N number. These items resulted from in-process reviews of data by the clerks reviewing the documents for inclusion in the data base. There were inconsistencies that required a detailed evaluation by an engineer knowledgeable in the process to resolve exactly what the TIIC number was, or what the material really is. Many involved the simple transposition of the alpha-numeric elements of the TIIC number so that what shows up on the MR, Form 575N, or contract is an invalid TIIC number, or a TIIC number associated with material obviously incompatible with the description and other data on the MR. Other examples of difficulties were that the TIIC number was not legible, not complete, or missing.

A total of 139 Pro-TIICs were identified in the MHDB as being safety-related, but not installed in host components subject to the environmental qualification requirements of 10 CFR 50.49. The RIP group evaluated these items and determined that 116 items were duplicate items or nonsafety-related. The remaining 23 items were included and evaluated by RIP. All MHDB Pro-TIICs installed in host components subject to 10 CFR 50.49 were captured and added to RIP, and subsequently evaluated as part of the "50.49 EQ Task."

5.7 RIP Hold Items

A large number of items had previously been rejected by the Material Improvement Project (MIP) (See Section 7.1) because they were unable to qualify existing material in inventory. However in many cases some of the affected material had been already been installed in the plant further complicating the issue. The inspection team selected a sample of 15 of these items from the RIP On Hold database. This database contained 2246 separate TIIC numbers that had been placed on hold pending further action necessary to resolve the issue. PPSP packages were selected for review to determine the adequacy of applicant corrective actions for each of the selected items. PPSP packages reviewed included the following:

- PPSP 40858 - Identified electrical connectors procured as QA Level III items during July 1990. The items were included under TIIC BBF564V. TVA determined that the items had been correctly procured as QA Level III and had not been used in any safety related application in the plant.
- PPSP 40831 - Identified an ITT Conoflow pressure regulator procured under TVA Contract 92MNA-43589C, dated December 10, 1991. The item had been procured as QA Level II without proper dedication. The item was included under TIIC BQE436R and had been installed in the plant as control air pressure regulator, 0-PREG-065-0028B. TVA stated that corrective actions associated with this item have not yet been completed and the PPSP is considered open. It was noted that WR C152395 had been written to remove/replace the regulator with a properly procured item. Additionally it was noted that PER CHPER930033 had been issued to address reportability and other related issues associated with this problem.
- PPSP 41033 - Identified a time delay relay which had been installed in the control circuitry for the component cooling water booster pump 1A-A under MR 631548 on July 12, 1990. The item should have been procured as QA Level I but did not have adequate documentation. The item was included under TIIC AHP725N. TVA stated that corrective actions have not yet been completed for this item and the PPSP is considered open. It was noted that WR C152300 had been written to replace the relay with a properly procured item.
- PPSP 41016 - Identified 10K resistors that had been installed in 125-Vdc vital battery power system panels under MR A479681 on April 26, 1985. The resistors should have been procured as QA Level I or Level II but did not have adequate documentation. The resistors were included under TIIC AVC377E and an incorrectly referenced contract number may have been the reason for lack of traceability of the material. TVA stated that corrective actions have yet not been completed for this item and the PPSP is considered open. It was noted that WR C152390 had been written to replace the resistors with a properly procured items.
- PPSP 40782 - Identified various bearings in stock without adequate documentation or known receiving date. The bearings were included under TIIC APB197E and had been placed in RIP hold status due to lack of traceability of the material. The applicant had determined that a total of three items were present in stock and no records could be found which would show that any of

the items had been used in the plant. It was verified through the Material Management System (MAMS) program that the three bearings had been surplused and the TIIC had been deleted.

- PPSP 40944 - Identified the possible incorrect use of insulator washers procured under TVA Contract, 256489, dated April, 1979. The items were included under TIIC ART495C and had been procured as QA Level III. However two of the items had been issued for use and installed in plant equipment under TVA Form 575-628609643 during July 1986. Both washers had been installed on radiation monitor, 1-RM-090-0010. TVA evaluated the washer as not a basic component and failure of the item would not affect the ability of equipment to perform its intended function. The remainder of the items were surplused.

- PPSP 40018 - Identified the possible lack of adequate documentation for a Crosby, three-inch, relief valve bellows assembly used in relief valve, 1-RFV-74-505 for the residual heat removal system. The bellows had been procured as QA Level I. However the item was placed on RIP hold status to resolve a concern about lack of proper documentation. Crosby had been an approved supplier and a COC had been furnished along with the item. However, TVA's original contract had specified the bellows as one of the various replacement parts that was required to meet specifications for pressure retaining components. This would have required a CMTR for material used to manufacture the bellows. Crosby had not furnished a CMTR. Subsequently the applicant evaluated the item as not requiring a CMTR. The original contract wording had been in error as the bellows was not actually a pressure retaining component. The bellows was acceptable as QA Level I.

- PPSP 40957 - Identified the possible improper usage of thermocouple wire in the plant. The thermocouple wire was included under TIIC AXX798H and had been procured under TVA Contracts 322205, 362363, and 371807 from 1981 through 1985 as QA Level II without proper dedication. The wire was placed on RIP Hold Status pending resolution of the issue. No stock remained in stores. TVA evaluated the wire as requiring downgrade to non-quality related and determined that all wire under this TIIC had been used in nonsafety-related applications (heat tracing), which was not required to function during an accident. It was verified through the MAMS program that no stock was on hand and that the TIIC had been deleted. Additionally it was noted that a new TIIC (BQR841P) had been issued for future material which was classified as QA Level 0.

- PPSP 40293 - Identified possible usage of bourdon tubes in the plant without proper dedication. Six bourdon tubes had been procured under TVA Contract RD-619204 as QA Level II but no supporting documentation was required by the contract. Five of the items had remained in stores but the sixth item could not be accounted for. The items were included in TIIC ABW859C and were placed on RIP hold status pending resolution of the issue. TVA evaluated the issue as acceptable. The five items known to be in stores were surplused and with no record of issue or other transaction history for the item existed it was assumed that the sixth item was lost. It was verified through the MAMS program that no stock was on hand and that the TIIC had been deleted.

- PPSP 40967 - Identified the usage of piping hanger material in the plant that had an improper TIIC designation. One hundred pipe hangers were originally included under TIIC BBH991X and had been procured under various TVA Contracts during 1988 and 1989 as QA Level I material. However the material had been incorrectly placed under a TIIC number which reserved for QA Level II material. There had not been a problem with the actual qualification of the material. Seventy of the pipe hangers had been installed in the plant and 30 remained in stores. The pipe hangers were placed on RIP hold status pending resolution of the issue. TVA evaluated the pipe hangers as acceptable for QA Level I. It was verified through the MAMS program that no stock was on hand for TIIC BBH991X and that the TIIC had been deleted and that all remaining pipe hangers in stores were transferred to a new TIIC Number BHQ340A which properly designated the material as QA Level I.

- PPSP 40974 - Identified the possible usage of improperly procured capacitors in the plant. Various electrolytic capacitors included under 18 separate TIICs had been procured under various TVA Contracts from Foxboro during 1989 and 1990. The capacitors were procured as QA Level II material but had originally been thought not to meet COC requirements and were placed in RIP Hold Status pending resolution of the issue. A number of the capacitors had been installed in various nuclear steam system supply instrumentation located in the plant. The remainder of the material had reached end of shelf life and had been surplused. TVA evaluated the capacitors as like-for-like replacements purchased from the original supplier of host equipment and therefore acceptable for QA Level II without further dedication. This determination was based on the existence of an original COC provided by Foxboro who, at the time, was on TVA's ASL.

TVA had conducted an audit at the suppliers facility prior to that time. This Audit, 89V-97, evaluated the suppliers program for control of quality required by TVA contracts for providing commercial grade modules and spare parts. The inspection team reviewed the audit report for this applicant vendor audit and noted that review of capacitors as spare parts had been included as part of the audit.

- PPSP 41039 - Identified the possible usage of improperly procured conduit bushings in the plant. Three hundred, 1.5-inch galvanized electrical bushings under TIIC AWD041V had been procured under TVA Contract 43501B during 1989 as QA Level III material. Two hundred fifty one of the bushings had been surplused and the remainder installed in the plant. No bushings procured under this contract remained in stores. Because some of the bushings were used in safety related applications the issue was placed on RIP Hold Status pending resolution of the issue. TVA evaluated the bushings as acceptable for QA Level III. This determination was based on the bushings not being a basic component. Bushings were installed in various conduit applications and conduit does not serve a safety related function.

- PPSP 41040 - Identified the possible usage of improperly procured cable in a safety-related application in the plant. A special purpose cable assembly had been procured as QA Level II without proper dedication and installed in Module, 1-LPT-24-69, under MR A619637 on February 21, 1990. The item was originally included under TIIC BHX678Y. However the modified module (TIIC

ARD287N) containing the special cable assembly was subsequently determined to be non qualified and returned to stores under a TVA Form 575-A031233 for credit. It was verified from the MAMS program and by a review of PEG package 01493 that the module that contained the cable assembly had been downgraded to QA Level 0.

- PPSP 41009 - Identified a possible discrepancy in usage of electrical cable. Following return of cable (Form 575N credit) from cable reel WB14735 a discrepancy of 352 feet of cable existed. Two thousand eighty two feet had been previously received satisfactorily and documentation for only 1730 feet which had been returned for credit. No other documentation for any other issuance existed. The returned 1730 feet was later surplused. The cable had been procured under TVA Contract 290617-2 as QA Level I and the qualification of the cable was never in question only the usage. The 4AWG cable was included under TIIC AXK180W and the issue placed in RIP hold status pending resolution of the issue. TVA evaluated the issue as acceptable based on a review of the Computerized Cable Routing System (CCRS). This review revealed that this type cable was not listed or referenced and therefore not installed in any safety-related applications.

- PPSP 40853 - Identified various cartridge type fuses for which procurement documentation could not be located. These fuses were included under a series of 54 separate TIICs and had been procured as QA Level II. However documentation was not available to support their use in safety-related applications. Because some of the fuses had been used in safety related applications the issue was placed on RIP hold status pending resolution of the issue. TVA had resolved this issue based on corrective actions which had been included under WBFIR910192112 and as part of TVA's Master Fuse List (MFL) Special Program. TVA had verified that all fuses in Class 1E circuits identified on the MFL have a solid path of traceability. This included both installed and non-installed fuses placed in plant storage areas.

Non-traceable fuses were discarded or returned to stores to be surplused. Inplant storage facilities for fuses were established and must meet the requirements of SSP-10.04, "Material Issue, Control, and Return." Additionally, TVA's MFL Special Program was reviewed by the NRC as documented in NRC Inspection Reports 50-390, 391/92-27 and 93-31. During those reviews the staff determined that fuses listed on the MFL were observed to have the correct identification, location, manufacturer, type, and size of fuse.

For each of the 12 selected RIP on hold items that were closed, it was determined that corrective actions had been adequate and that any related installed items were properly procured. For the three remaining RIP on hold items which were still considered as open, the inspection team verified that documentation existed to ensure that removal of the related incorrectly procured items would occur.

5.8 IMPELL Review of Seismically Sensitive-Electrically Active (SSEA) Items

The inspection team reviewed the status of TVA's corrective actions associated with Significant Corrective Action Report WBP8900634SCA, which was initiated as the result of a previous material evaluation performed for TVA by IMPELL

Corporation, between August and October 1989. The SCAR remains open pending completion of corrective actions. During the evaluation, SSEA devices in stock and installed in safety-related applications associated with 416 TIICs were evaluated. As the result of that review, installed SSEA devices associated with 56 TIICs were determined to be inadequate. These electrical devices had been released from stores inventory for safety related applications and were not procured to requirements of the Institute of Electrical and Electronics Engineers (IEEE) Standard IEEE-344 1975, "Seismic Qualification Requirements for Nuclear Power Generating Stations," nor were they subsequently determined to satisfy those requirements. This failure was also reported to the NRC under Construction Deficiency Report (CDR) 390/91-31. Because there can be multiple installations associated with the same TIIC number, the 56 TIIC items resulted in a total of 66 separate installations and classified under 56 Case Numbers (separate case number for each affected TIIC). The affected devices included relays, breakers, motor starters and magnetic contacts. When corrective actions are complete each of the affected installations was to be replaced with a properly qualified component.

The inspection team was provided a copy of a matrix used by TVA personnel to identify a specific work document such as a design change notice (DCN), WO, WR or other document which was intended to ensure the replacement of each of the 66 devices. The team reviewed the matrix and determined that TVA's corrective actions were complete for all but three of the 66 devices which had originally been determined to be inadequate. In order to assess the adequacy of TVA's corrective actions in this area, TVA was requested to provide documentation to support corrective actions for the following five devices:

- Case 14 - Identified motor starters procured under TVA Contracts N3-865-1 and 322172-1 which were not procured with any unique nuclear requirements for use as basic components. TVA had determined that one motor starter had been requisitioned under TVA Form 575-628602989 and installed under MR 498392. That starter had been installed in 1MCC-215-1B2/2B-B for the EDG Room 1B-B Exhaust Fan, 1-MTR-30-453B. It was verified that the affected motor starter had been replaced with a qualified device under WO 90-16222-63.
- Case 16 - Identified a relay requisitioned under TVA Form 575-628501768 for installation under MR 412375. The relay had been procured under TVA Contract N3-632-1 and was never certified as a basic component for use in a Class 1E circuit. However the MR which was worked during 1984 to troubleshoot an oil pressure fluctuation on Chiller O-CHR-031-0049/2-B, does not indicate that the relay was actually used in the system as a spare part. MR A622272 had been written by TVA to replace the relay but was canceled when TVA determined that the chiller control panel did not contain this type relay. Additionally TVA personnel inspected the associated chiller control panel and did not identify any evidence that relay replacement had occurred.
- Case 23 - Identified a circuit breaker procured under TVA Contract N3-2578-1 which was not procured as a qualified Class 1E device. TVA had determined that the circuit breaker had been requisitioned under TVA Form 575-628706537 and installed under MR 511626. That circuit breaker

had been installed in IMCC-214-A1-A Compartment 2D. This application requires the use of a Class 1E circuit breaker so that the nonsafety-related primary makeup water pump 1A circuitry may be isolated from the Class 1E power source during an accident. It was determined that replacement of the affected circuit breaker had been included under DCN M-19745-A as implemented by WP D-19745-12. Additionally the inspection team verified that WP D-19745-12 was field complete on October 25, 1993.

- Case 43 - Identified a relay requisitioned under TVA Form 575-628504307 for installation under MR 408313. The relay had been procured under TVA Contract 303615 and was never certified as a basic component for use in a class 1E circuit. However the referenced MR is not available on microfilm and could not be traced. Contract 303615 also stated that the subject relay had been procured for use in the Incore Instrument System. TVA reviewed the Nuclear Instrumentation System (NIS), Incore Thermocouple System (ITS) and Incore Flux Mapping System (IFMS) to determine if this type relay was used in any safety-related applications. The IFMS is not considered safety-related and the electronic processing portion of the ITS and the source and intermediate range portions of the NIS have been replaced since TVA Form 575N was issued. Additionally no relay of this type exists in the power range portions of the NIS. TVA determined that since they did not identify any record or work document which indicated that the relay was actually used in the system as a spare part, that this issue had been adequately dispositioned.
- Case 54 - Identified a circuit breaker procured under TVA Contract 468257 which was not procured as a qualified Class 1E device. TVA had determined that the circuit breaker had been requisitioned under TVA Form 575-628705034 and installed under MR 413626. That circuit breaker had been installed as the power supply for the 120 Vac vital inverter 2-IV which supplies power to the vital instrument board. It was verified that the affected circuit breaker had been replaced with a qualified device under WO 90-16570-29.

As a result of this review, the inspection team determined that corrective actions associated with the above selected cases were adequate. This issue will receive further review during the followup review associated with CDR 390/91-31.

5.9 RIP QA Level III Item Review

The inspection team reviewed "Final Report, Replacement Items Program Task 4A, Revision 1, issued June 30, 1994," which was prepared by TVA. The scope of this task was to investigate the QA Level III population in TVA's computerized Material Management System (MAMS) to determine if items had been installed into safety-related applications. As part of this effort, TVA performed a sample review of QA Level III items to determine if these items had been inadvertently installed into safety-related applications and thus negated environmental or seismic qualifications or degraded the capability of the host equipment to perform its intended safety function.

TVA used MIL-STD-105E as a guide for selection of the sample lot and a predetermined rejection rate of no more than five. This would provide a 95 percent level of confidence that at least 95 percent of the installations were not adversely affected by the replacement parts installed as QA Level III. This review was performed on a sample lot of 315 items out of a total of 5215 QA Level III items that were installed in the plant. Because there can be multiple installations associated with the same TIIC number, the 315 TIIC items selected as the sample population resulted in 1417 installations.

The inspection team reviewed the listing of 315 sample QA Level III TIIC items that had been the subject of this review. The sample list included items such as light bulbs, conduit, identification markers, etc., that would not have been expected to serve any safety-related application along with items that would have required detailed reviews to ensure that the items were not used in safety-related applications. The team selected several items from TVA's sample listing and requested that they provide documentation to establish the extent of their review effort. The team determined that, for each of the items selected, TVA had ensured that the related items had not been installed in a safety-related application.

As the result of TVA's review, three items were found to have been installed into safety-related applications but were declared to be of indeterminate quality because the documentation was not sufficient to demonstrate traceability. As a result, the items were replaced with acceptable piece parts and the team agreed that this rejection rate met the acceptance criteria. Based on the above review, the team determined that TVA's review had been adequate to ensure that the 95 percent confidence level had been achieved.

6 REPLACEMENT ITEMS INSTALLED BY PREVIOUS CONSTRUCTION ACTIVITIES REPLACEMENT ITEMS PROGRAM (RIP) (TI 2512/27, 35065, 35746, 35747)

The RIP CAP plan states that its scope includes the review and evaluation of spare or replacement parts from the construction warehouse inventory installed in WBN by the construction organization. The purpose of the NRC's review of this area during this inspection was to provide an independent determination of the scope and depth of the TVA review completed thus far in the area of construction installed materials at WBN. It was not intended to evaluate the disposition of TVA's findings but rather to summarize TVA's review and provide NRC management with information necessary to evaluate the suitability of materials used in the construction of WBN.

In its initial response to concerns raised regarding the adequacy of this portion of the RIP CAP plan, TVA contracted Digital Engineering, Incorporated (DE), to develop the methodology and conduct a review of this material. The results of this review were published in DE's preliminary report, "WBN Construction RIP, Phase 1 Summary Report," dated October 3, 1989. On August 17, 1991, TVA Nuclear Quality Assurance (NQA) issued the report of its audit of RIP, QA Audit Report No. WBA92211. The report contained three recommendations (numbers 2, 3, and 4) regarding construction installed materials. Recommendation 2 was that because the review completed thus far had only addressed the ledger card recorded items (about 55,000) and not the

so-called direct charge items (about 19,000), RIP should review the direct charge items. Recommendation 3 was that RIP should disposition ledger card inventory adjustments made without issues on Form 575Ns. Recommendation 4 was that associated 575Ns be sampled for verification of the validity of the information thereon.

In a subsequent audit report, NQA Assessment Report NQA-WB-92-028, dated September 23, 1992, NQA accepted the RIP proposal, in response to Recommendation 2 of WBA92211, to sample a homogeneous group of direct charge items listed in their "System 3" computer inventory system, stating that the completion assurance group will follow up on the acceptability of the sample and its results. On the basis of procedural controls in-place requiring the use of Form 575N for all material issues, NQA accepted the RIP explanation that the no-issue inventory adjustments were most likely the result of breakage or theft. Finally, NQA rejected RIPs taking credit for confidence in the information contained on the forms because the response addressed only maintenance installed items and not construction installed items.

The revised response to Recommendation 4 agreed to be provided by RIP as a result of this disapproval, was submitted in a memorandum from the TVA site vice president, dated October 19, 1992, which transmitted to R.F. Driscoll, the WBN responses to the recommendations in NQA-WB-92-028. In the revised response (Attachment F to the memo), several additional factors in support of substantiating the validity of Form 575N information were presented. Finally, on January 25, 1993, RIP issued "Final Report, Replacement Items Program Task 3, Construction Installed Replacement Items, Revision 1." Most notably this report provided the results of the review of direct charge materials as well as additional supporting documentation on ledger card materials.

On the basis of this review, the inspection team did not find any obvious significant deficiencies in the logic, methodology, or conclusions of the final report. However to date, the formal review by TVA of this report has not been documented. Since TVA needs to make a formal, final submittal of its review and proposed final disposition of this issue, this item is identified as Inspector Followup Item 94-201-I5.

7 CURRENT WAREHOUSE INVENTORY-MATERIAL IMPROVEMENT PROJECT (MIP) (TI 2512/27, 35065, 35746, 35747)

7.1 Implementation Review of the MIP

As a part of the RIP CAP plan, TVA committed to the sanitization of the current inventory of materials purchased prior to June 5, 1991, before these materials were released for installation in the plant. Sanitization was the term used to define a process of review for technical adequacy and re-receipt inspection of these materials. This effort was referred to in the RIP CAP as the Material Improvement Project or "MIP." This part of the inspection was conducted to review the implementation of the MIP and was conducted primarily by an evaluation of the adequacy of the sanitization packages which had been created during the process.

The sanitization packages included the following information: a walkdown sheet which identified the on hand stock; a PEG package which provided the engineering evaluation of the technical and QA requirements for the material; the original contract/purchase order/request for delivery; the old receipt inspection records, if available; COCs/CMTRs and other vendor documentation; the sanitization receipt inspection; a plant issue history from the MAMS computer data base for the material; and verification that the vendor was on the ASL, as applicable. The inspection team conducted an independent review of all of these documents to verify technical adequacy and inclusion of appropriate regulatory requirements.

7.1.1 Acceptable Sanitization Packages

Many of the items reviewed by the team were surplused during the sanitization process. For each of these items the inspection team verified that there was an entry on the computer data base, which indicated that the material had in fact been surplused. The team also verified that there was no issue history for the material, and where possible reviewed inventory sheets, which verified that the material had been moved to TVA's surplus yard. The following packages were determined to be adequate:

SANM 14347	SANM 13295	SANM 13234	SANS SP0070
SANM 13199	SANM 12585	SANM 11030	SANE 2949
SANM 13289	SANM 13968	SANM 15093	SANE 3092
SANM 14061	SANM 11545	SANM 14345	SANE 3648
SAN SP0023	SANE 4623	SANE 2130	SANE 4480
SANE 3368	SANE 3973	SANE 3934	SANE 3755
SANE 4123 (partial surplus)			

The following items reviewed by the inspection team successfully completed the sanitization process and were maintained in inventory by TVA:

SANM 13126	QA level I	Pump collar
SANM 15684	QA level I	Ball bearings
SANM 10337	QA level I	Valve pin
SANM 10901	QA level I	Metal Bar
SANE 4468	QA level I	Penetration repair kits
SANE 3718*	QA Level I	ASCO solenoid valves
SANE 3736	QA Level II	Gould fuse
SANM 12193	QA level II	Structural angle

* Determined acceptable after TVA corrective action reported subsequent to inspection

The following items had been rejected during the sanitization process. Review of these packages by the inspection team determined that the materials had an issue history, and therefore, required further engineering evaluation. Further review of additional documentation determined that these materials had been referred to engineering for an evaluation of the installed items:

SANE 4601	QA level II	Digital microcircuits
SANE 3407	QA level II	Rotary control switches
SANE 4013	QA Level II	Resistors (TIIC AQD7SOB)

7.1.2 Unacceptable Sanitization Packages

The inspection team's review of sanitization packages identified the following deficiencies:

- SANE 2953, SANMs 13888, 14894 and 14175 - These sanitization packages rejected stock material and, due to a material issue history, required an engineering evaluation of the specific installation application in the plant (i.e., the items had been referred to RIP for evaluation). The team requested copies of these engineering evaluations (PPSP packages) for review and was advised that no packages had been developed for these items. Additionally, the team's review of the RIP computer data base determined that the items were not carried on this data base as outstanding work items for the RIP project. Note: TIIC numbers in these packages requiring engineering evaluation were as follows: SANE 2953 (TIIC Numbers AKN-613T, AKN-712Q, AMN-341Q, AMN-414L, AKN-912G); SANM 13888 (TIIC Numbers AVX-142J, BAY-349Q); SANM 14894 (TIIC Number AVF-311X); and SANM 14175 (TIIC Number BRK-844A, Heat # M7780).

This issue was identified to TVA as a problem area requiring an extent of condition evaluation and resolution. As a result of this finding, TVA initiated WPER 940351. As a part of the corrective action to the PER, TVA reviewed 885 sanitization packages in an attempt to identify the extent of the problem. Of the 885 packages, 158 packages were identified that were to have been referred to the RIP project for further evaluation. Of the 158 referred packages, 84 were not found on the RIP computer data base and no additional engineering reviews had been performed. TVA's extent of condition, evaluation and corrective actions in this area were ongoing at the completion of this inspection.

TVA Procedure SSP-10.B, "Materials Improvement Project," Revision 5, requires that: "Material evaluated as unacceptable for use at WBN will be investigated to determine if any of the material has been issued from inventory. If none has been issued, then future issues will be denied. Material issued, but not installed, will be recalled to the warehouse. In both cases, the material will be surplus, dedicated, or discarded. If an item has been installed, all pertinent information will be logged on Appendix D and forwarded to the appropriate Engineering Group for evaluation and resolution." Since TVA personnel did not follow this procedure, previously installed material was not evaluated for suitability. Failure to follow procedures is identified as Deficiency 94-201-D8.

- SANE 3718 - This package was for solenoid operated valves (SOVs) manufactured by the Automatic Switch Company (ASCO), model NP8320A182E (TIIC BEE211G). These SOVs are referred to in ASCO's product catalog as NP-1 valves, environmentally qualified, yet the procurement master data sheet (PMDS) was checked "no" under 10 CFR 50.49 applicability (a

typical use is MSIV pilot valves). Also, the 10 CFR Part 21 block was not checked and the shelf life was marked "NA." It was not clear where one SOV from the each of the two contracts covered was used. The stock verification sheet has shop order marked "NA," yet there are shop order numbers associated with the SOVs.

The printout from MAMS of the associated V300 item history screen indicated five SOVs which were surplused or possibly scrapped due to damage. Attachment A to release RD 111893 of IQT Contract 89NNR-34440B, Test Report AQR67368, Revision 1, possibly by Wyle Laboratories, had no associated environmental qualification (EQ) binder number reference, e.g., "WBN-SOV-XXX," and pages 2 and 4 of the ASCO COC were missing. The requested temperature profile attached to the COC was the wrong type. It was an EQ accident profile (temperature in containment versus time); whereas, the profile intended was the chart of temperatures at different locations on the SOV when it is continuously energized, which is needed to perform qualified life calculations. The activation energies of the limiting components and their bases were not evident. The COC for Contract 85PN7-360966, Shop Order 65076N (quantity 4), supplied through M&L did not reference TVA's PO number, yet this COC was accepted. The other contract in the package, SOVs supplied through another ASCO distributor, Power & Control, Incorporated, did have the TVA RD number on the COC.

However, in another SAN package, TVA rejected a COC for this deficiency. In response to these concerns, and subsequent to the inspection, TVA submitted copies of the missing COC pages and first page with the proper contract reference it had obtained from ASCO. The correct temperature profile diagram was on the missing page 4. Also submitted was a new PEG package (9400042184) with which TVA supplemented SANE 3718 with the required information. TVA also reported that the deficient PMDS is under review to incorporate the proper requirements. These corrective actions were considered acceptable.

- SANE 3845 - This package was for QA Level I, Dunham-Bush motor starters, TIIC AQQ696J. The package indicated that the material was retained, yet inventory adjustment sheet number 62-92-357 indicated the material "failed SAN" and "code 95 TIIC AQQ696J." It was also not clear from the package how the items were dedicated by the supplier, Southern Testing Services (STS), and how production units sent to WBN were verified to be similar to the STS seismic test specimen.

In response to these concerns, TVA reported subsequent to the inspection that three starters were procured under contract 90LK-85122B from STS. One was used for the electrical board room chiller (Form 575N 254779) and when found not operating properly, it was replaced by another one from stock (Form 575N 245741). PPSP 35545 identified the rejected starter as being put on RIP hold and then surplused. TVA reported that the remaining unit was still being held in "indeterminate material" and would require a PEG package to put it in stock. In response to the inspection team's questions on dedication and seismic qualification, TVA provided a copy of STS Report S254-RP-02, dated October 5, 1990. This

report consisted of an analysis for additional starters in question for seismic qualification similarity to starters seismically tested by STS for WBN as reported in STS Report S254-RP-01, dated September 21, 1990, under the same TVA contract. The similarity analysis was based on visual examination of the starters and the fact that they were of the same part number as the original units supplied as well as the seismic test specimen and had been determined to have been all manufactured within a 3-week period in the same factory.

Concluding that the subsequently supplied starters were similar, or even "like-for-like" to those qualified on the basis described above is consistent with the staff position promulgated in NRC GL 91-05. However, the similarity analysis was intended only for seismic qualification. Since seismic qualification is a design verification process, and is only one critical characteristic in a dedication process, it cannot substitute for a complete dedication. Although STS energized a sample starter with nominal 240 Vac during some of the seismic test runs, verification of other critical characteristics considered standard by the industry, such as insulation resistance, individual pole resistance, full load hold-in capability, overload trip performance, and pull-in and drop-out voltages (degraded voltage conditions) was not documented in the packages reviewed by the team.

In addition, the original seismic report contained a notice of anomaly describing 10- to 12-millisecond (ms) chatter in the normally closed auxiliary switch of the seismic test specimen starter during the first run in which it was deenergized and in one of four horizontal orientations. The description of the test and the procedure indicated that there had been no other runs with this or another starter in the same orientation (apparently the unit's most seismically sensitive axis) in which the NC auxiliary switch did not chatter in order to rule out common mode failure. It was not clear that any acceptance or rejection criterion had been established for this performance parameter, yet STS determined, apparently unilaterally, that this condition would not adversely affect the starters' performance.

As far as could be determined from the documents reviewed, this determination was apparently acceptable to TVA because the WBN Civil Engineering group had accepted the report for seismic qualification, yet it was not clear how or whether the 10- to 12-ms contact bounce of the NC auxiliary switch had been evaluated for its effect on the safety function(s) of the starter, or its parent equipment or system as installed at WBN. Accordingly, this issue is identified as Unresolved Item 94-201-U3.

8 EXIT MEETING

On July 20, 1994, the inspection team conducted an exit meeting which summarized the inspection scope and the findings. The inspection team described the areas inspected and discussed in detail the inspection results. The following people were in attendance.

TVA

D. Nunn, Vice President
J. Scalice, Vice President
R. Johnson, Acting Engineering Manager
W. Clothier, Jr., Manager, Nuclear Stores
R. Brown, Licensing
T. Whittemore, Materials Program Administrator
W. Rogers, PEG Manager
J. Lewis, Material Supervisor
V. Patuzzi, Quality Assurance Specialist
J. Seeley, RIP Project Manager
P. Pace, Compliance Licensing Manager
D. Malone, Manager
B. Schofield, Licensing Manager
L. Spiers, Site Quality Assurance Manager
G. Pannell, Manager Special Programs
W. Elliott, Engineering and Modifications Manager
D. Koehl, Technical Support Manager
D. Moody, Plant Manager
P. Phillips, Senior Technical Lead Auditor
M. Fecht, Manager Quality Programs
J. Overturf, RIP Project Manager

United Energy Services Corporation

F. Yurich, Consultant to TVA
W. Craig, Consultant to TVA

Nuclear Regulatory Commission

R. Pettis, Jr., P.E., Team Leader, NRR
U. Potapovs, Section Chief, NRR
P. Frederickson, Construction Branch Chief, RII
L. Campbell, Reactor Engineer, NRR
S. Alexander, EQ and Test Engineer, NRR (via telephone)
J. Peralta, Operations Engineer, NRR
R. Gibbs, Reactor Engineer, RII
W. Bearden, Project Engineer, RII
P. Van Doorn, Senior Resident Inspector-Operations, WBN
G. Walton, Senior Resident Inspector-Construction, WBN

Appendix A

List of Deficiencies

<u>Number</u>	<u>Title</u>	<u>Section/Page</u>
94-201-D1	Weaknesses in Corrective Action Program Implementation - PEG Package 9400018340	4.1/9
94-201-D2	Inadequate Translation of Design Basis Requirements into Design Output Documents Which Resulted in an Inadequate Review for Suitability of Application for QA Level I Molded Case Circuit Breakers - PEG Package 9400002182	4.1/12
94-201-D3	Failure to Ensure that QA Level I Molded Case Circuit Breakers Meet Procurement Requirements - PEG Package 9400002182	4.1/14
94-201-D4	Inadequate Review for Suitability of Application and Failure to Verify Compliance to the Procurement Requirements for GE 125-Vdc Trip Coils for Magne-Blast Circuit Breakers used in Class 1E 6900V Shutdown Boards - PPSP 27732	5.2/21
94-201-D5	Failure to Verify the Procurement Requirements of Telemechanique Starters Evaluated Under PPSP 27739	5.2/23
94-201-D6	Inadequate Commercial Grade Dedication of a GE Magne-Blast Circuit Breaker Spring Charging Motor - PPSP 27724	5.3/26
94-201-D7	Inadequate Commercial Grade Dedication of a Westinghouse Direct Trip Actuator for a 480-Vac Air Circuit Breaker Evaluated Under PPSP 28973	5.3/27
94-201-D8	Failure to Follow TVA Procedures for Previously Installed Items Not Captured in the RIP Data Base - SANE 2953	7.1.2/40
94-201-D9	Inadequate TVA Procedures Associated With the Evaluation of PEG Package 9400002182, PPSP 27975, PPSP 27732 and PPSP 28973	4.1/12, 5.2/19, 5.2/21, 5.3/28

Appendix B

List of Unresolved Items

<u>Number</u>	<u>Title</u>	<u>Section/Page</u>
94-201-U1	QA Level II PEG EPEG 920013 for GE 125-Vdc Closing Spring Charging Motors That Must Operate More Than Once to Perform Their Safety Function	4.1/16
94-201-U2	Questions Relating to Dedication Testing of Limitorque Motors Evaluated Under PPSP 20406	5.2/20
94-201-U3	Questions Concerning QA Level I Dunham-Bush Motor Starters Evaluated Under SANE 3845	7.1.2/42

Appendix C

List of Inspector Followup Items

<u>Number</u>	<u>Title</u>	<u>Section/Page</u>
94-201-I1	Several Issues Requiring Amendment to PEG Package 9400018459 for Amphenol Connectors	4.1/8
94-201-I2	Documentation Contained in PPSP 27975 for a GEMAC Power Supply Did Not Contain Explanations of the Technical Concerns Identified by TVA	5.2/20
94-201-I3	Failure to Perform an Adequate Technical Evaluation for Lockwire Used in Certain Limitorque Operators - PPSP 28039	5.3/25
94-201-I4	Questions Concerning TVA's Process Specification 4.M.1.1 and PF Material Purchase Specifications Regarding Control of Contaminants	5.5/29
94-201-I5	Questions Concerning the Evaluation of Replacement Items Installed by Previous Construction Activities (RIP)	6/38

Appendix D

Glossary of Specific Terms

BKR	Breaker (UNID Abbreviation)
BFN	Browns Ferry Nuclear (Plant)
CAP	Corrective Action Program
CCRS	Computerized Cable Routing System
COMP	Switchboard Compartment (UNID Abbreviation)
CSST	Combined Station Supply Transformer
DCN	Design Change Notice
DE	Digital Engineering (RIP Contractor)
EPEG	Equipment Procurement Engineering Group
EPG	Engineering Procurement Group (Contract Procurement Group)
IQT	Indefinite Quantity and Time (Contract)
ITS	Incore Thermocouple System
IFMS	Incore Flux Monitoring System
MI	Maintenance Instruction
MR	Material Requisition
MAMS	Material Management System
MTR	Motor (UNID Abbreviation)
MIP	Material Improvement Project
MFL	Master Fuse List
MHDB	Material History Data Base
NIS	Nuclear Instrumentation System
PER	Problem Evaluation Report
PEG	Procurement Engineering Group
PPSP	Previous Procurement Substantiation Process (RIP)
PSS	Power Service Shops (TVA Repair Facility, Mussel Shoals, LA)
PMDS	Procurement Master Data Sheet
QAI	Quality Assurance Instruction
RTL	Release Tracking Log (PPSPs)
RIR	Receipt Inspection Report
RIP	Replacement Items Program
RIMS	Record Information Management System
SAN	Sanitization (MIP)
SANE	Electrical Sanitization Package
SANM	Mechanical Sanitization Package
SSEA	Seismically Sensitive-Electrically Active
SQN	Sequoyah Nuclear
SSP	Site Standard Practice
SAI	Site Administrative Instruction
TIIC	TVA Item Identification Code
TEF	Technical Evaluation Form (SSP-10.05 and -10.C)
UNID	Unique Identifier (Equipment Tag Number)
V300	MAMS Material History Screen
WBN	Watts Bar Nuclear
WO	Work Order
WR	Work Request
575N	Material Requisition/Issue Form

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

<u>DOCUMENT TYPE AND NUMBER</u>	<u>TITLE</u>
Information Notice 94-71	Degradation of Scram Solenoid Pilot Valve Pressure and Exhaust Diaphragms
Information Notice 94-76	Recent Failures of Charging/Safety Injection Pump Shafts
Information Notice 94-85	Problems With the Latching Mechanisms in Potter and Brumfield R10-E3286-2 Relays
Information Notice 94-86	Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag

CORRESPONDENCE RELATED TO VENDOR ISSUES

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

November 4, 1994

Mr. Jim Fitzwilliam
Quality Assurance Manager
Nova Machine Products Corporation
P.O. Box 30287
Middleburg Heights, Ohio 44130

SUBJECT: REQUEST FOR INTERPRETATION: NRC INFORMATION NOTICE NO. 86-21

Dear Mr. Fitzwilliam:

By letter dated August 19, 1994, you requested the U.S. Nuclear Regulatory Commission (NRC) to provide an interpretation of Information Notice (IN) No. 86-21, "Recognition of American Society of Mechanical Engineers Accreditation Program for N Stamp Holders," dated March 31, 1986, including its Supplements 1 and 2, issued on December 4, 1986, and April 16, 1991, respectively. IN No. 86-21 and its supplements provided no new requirements, but provided guidance and clarification on existing NRC positions that may be used by purchasers of certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, "Rules for Construction of Nuclear Power Plant Components," items to verify that the ASME-accredited suppliers of the items are effectively implementing their quality assurance (QA) program.

Your August 19, 1994, letter to the NRC identified a method, other than those discussed in IN No. 86-21, Supplement 2, for verifying that an ASME-accredited supplier's QA program is being effectively implemented and you requested an interpretation by the NRC for its acceptability. The following is our understanding of your interpretation request.

I. BACKGROUND

Purchase orders (POs) received by Nova from utilities (NRC licensees) and support groups such as Architect Engineers, Nuclear Steam System Suppliers, Original Equipment Manufacturers, and pump and valve manufacturers, invoke requirements for Nova to verify that ASME-accredited suppliers are effectively implementing their QA program.

II. METHOD PROPOSED BY NOVA FOR VERIFYING THAT ASME-ACCREDITED SUPPLIERS ARE EFFECTIVELY IMPLEMENTING THEIR QA PROGRAMS

1. Through a review of the ASME publication, "Companies Holding Nuclear Certificates of Authorization," which lists suppliers accredited by ASME and their scope and expiration date, Nova will establish that ASME has performed an audit and accepted a supplier's QA program and issued the supplier a certificate.

2. The supplier is then placed on Nova's approved supplier's list (ASL) for the scope identified in the ASME publication with a restriction that prior to procurement from the supplier an implementation audit will be performed.
3. Nova is informed that Nuclear Procurement Issues Committee (NUPIC) performed an audit at the supplier and issued the supplier a NUPIC audit report verifying that the supplier is effectively implementing its the ASME-accredited scope activities.
4. Nova requests and the supplier forwards a copy of the NUPIC audit report. The audit report identifies the audit participants, the ASME certificate number(s), QA program revision, and details the audit results.
5. Nova evaluates the NUPIC audit report and concludes that the NUPIC audit verified the supplier is adequately implementing its QA program for the ASME-accredited scope. Nova maintains its evaluation of the NUPIC audit report and the audit report.
6. Based on its evaluation, Nova uses the NUPIC audit as the means to verify that the supplier is effectively implementing its ASME-accredited scope QA program and removes the restriction from its ASL to perform a QA program implementation audit of the supplier. Nova's justification for not performing its own audit of the supplier is that a NUPIC member, an NRC licensee, has performed the verification audit and, therefore, the intent of the requirements discussed in IN 86-21 have been met.

III. INTERPRETATION REQUEST

Nova's specific question presented to the NRC in its August 19, 1994, letter is as follows:

"Under the above noted conditions, can Nova procure materials certified under an ASME Certificate holder's program and, meet the intent of IN 86-21 without a) performing an audit of the ASME Certificate holder's program, or b) performing any type of program verification?"

IV. NRC EVALUATION OF NOVA'S QUESTION

BACKGROUND

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

Additionally, in NRC Regulatory Guide (RG) 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants," Revision 1, dated September 1980, the NRC has provided a position on the use of joint audits. Regulatory Position C.5 of RG 1.144 provides additional guidance on performing audits and supplements the provisions of Section 1.3 of ANSI/ASME N45.2.12-1977, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants."

Section 1.3 of ANSI/ASME N45.2.12-1977 states, in part, that the work of establishing practices and procedures and providing the resources in terms of personnel, equipment and services necessary to meet the requirements of this standard may be delegated to other organizations and such delegation shall also be documented. Regulatory Position C.5 of RG 1.144 states, "Where more than one purchaser buys from a single supplier, a purchaser may perform an audit of the supplier on behalf of more than one purchaser in order to reduce the number of external audits of the supplier. The results of this audit should be distributed to all purchasers for whom the audit was conducted."

Also, RG 1.144 acknowledges that differences may exist between the requirements of ANSI/ASME N45.2.12 and Section III of the ASME Code. Regulatory Position C.2 of RG 1.144 addresses these differences and states, in part, that subject to the exceptions of this regulatory position, ANSI/ASME N45.2.12 should be used in conjunction with Section III of the ASME Code for auditing QA programs where the ASME Code does not address activities covered by ANSI/ASME N45.2.12.

NRC RESPONSE

Nova's proposed method for verifying that ASME-accredited suppliers are effectively implementing their QA program does not appear to meet the requirements of Appendix B to 10 CFR Part 50, the provisions of ANSI/ASME N45.2.12-1977, or the guidance provided in Regulatory Position C.5 of RG 1.144. In order for a supplier to use a NUPIC audit the supplier would have to perform the following.

- The supplier would have to delegate to NUPIC the responsibility to perform audits on its behalf and this delegation would have to be documented as provided for in Section 1.3 of ANSI/ASME N45.2.12-1977.
- The results of the NUPIC audit should be distributed to the supplier using the audit by NUPIC, not the company audited by NUPIC, as provided for in Regulatory Position C.5 of RG 1.144.

J. Fitzwilliam

-4-

Should you have any further questions, please contact Mr. Larry L. Campbell of this office at (301) 504-2976.

Sincerely,

A handwritten signature in black ink, appearing to read "R. M. Gallo", written in a cursive style.

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

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

November 28, 1994

Mr. Roger F. Reedy, President
Reedy Associates, Inc.
15951 Los Gatos Blvd., Suite 1
Los Gatos, California 95032

SUBJECT: REQUEST FOR INTERPRETATION: 10 CFR PART 50, APPENDIX B, INTERNAL
AUDITS PERFORMED BY VENDORS

Dear Mr. Reedy:

We have reviewed your November 2, 1993, letter to the U.S. Nuclear Regulatory Commission (NRC) that provided additional information with respect to your previous inquiry to the NRC, dated July 27, 1993. You expressed specific opinions in your November 2, 1993, letter concerning the degree of self audits (internal audits) necessary to satisfy the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR). Additionally, in this letter you stated that Appendix B does not state that Criteria XVIII, "Audits," was mandatory for consulting firms such as yours. Also, in your November 2, 1993, letter you proposed an alternative in-lieu of internal audits performed by your firm, viz., "licensees should be able to accept an audit of a vendor performed by another licensee, provided the audit results are reviewed and accepted by the first licensee prior to procuring products or services from this vendor. (Please note that these audit results are also reviewed and evaluated by us as equivalent to internal audits)." Our response to your inquiry is as follows. The information provided in the enclosure to this letter supports our response to your inquiry.

Licensees are responsible for determining the quality and technical requirements applicable for a contractor's safety-related design service. Certain design services may require extensive quality assurance design controls through all stages of development and implementation while others may require only a limited quality assurance involvement in selected phases of development and implementation. The licensee specifies the quality assurance controls to be used in safety-related design services which it contracts for in its procurement documents for such services.

It is the NRC staff's expectation that (a) where the licensee has determined, pursuant to Criteria I, III, and IV of Appendix B, that the activities performed by its contractor must be subjected to an internal audit by the contractor in accordance with Criterion XVIII, and (b) the licensee has imposed such compliance on its contractor through the licensee's procurement documents, then the contractor is responsible for ensuring the required internal audits are performed. Contractors performing safety-related design services that have accepted procurement documents that invoke the requirements of Appendix B to 10 CFR Part 50, or ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants," or NQA-1, "Quality Assurance Requirements for Nuclear Power Plants," or other similar criteria that require audits, should perform internal audits of their safety-related design activities. These internal audits would serve to verify that the contractor's

quality assurance program, including design control, for the contractor's work scope was being effectively implemented.

There are two alternatives to a contractor actually performing the internal audits. First, the contractor could advise licensees that the contractor declines to perform the auditing described in Appendix B, and that the licensee must have total responsibility for compliance with these provisions. Alternatively, the contractor can authorize and delegate, in writing (as provided for in Criterion I and IV), the responsibility for the performance of the internal audits to another organization. However, because the performance of internal audits is a safety-related service, the contractor would have to assume the responsibility for meeting the requirements of Criterion VII, "Control of Purchased Equipment and Services," of Appendix B by ensuring that the organization (licensee or contractor) performing internal audits on behalf of the contractor was qualified to do so. Additionally, applicable requirements from procurement documents issued to the contractor by licensees would have to be passed on to the organization performing the internal audits. Also, the contractor would have to clearly establish, in writing, that the internal audit was performed on behalf of Reedy and the organization performing this quality service would have to assume certain responsibilities for performing a safety related function. Finally, even though the contractor has delegated the performance of the audit to a third party, the contractor is nonetheless ultimately responsible for the adequacy of the audits.

If, for whatever reason, your firm does not wish to have the responsibility for complying with the auditing provisions of Appendix B, you must: (a) explicitly advise any and all licensees whom you are providing or intend to provide safety-related design services that your firm will not comply with the auditing requirements of Criterion XVIII, such that the licensee will have total responsibility for meeting the pertinent audit provisions of Appendix B and must be responsible for any programmatic and regulatory burden thereof, and (b) assure that the licensee's procurement document retaining your firm's services does not contain requirements for compliance with the auditing provisions of Appendix B.

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

Sincerely,

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

Enclosure: As stated

INQUIRY DISCUSSION

BACKGROUND

Appendix B to 10 CFR Part 50 identifies the quality assurance (QA) criteria for nuclear power plants. The pertinent requirements of this appendix apply to all activities affecting the safety related functions of structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public: these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. The NRC staff evaluates the adequacy of QA programs used by both holders of construction permits and operating licenses with respect to the requirements of Appendix B to 10 CFR Part 50. Additionally, when the NRC staff conducts inspections at nonlicensed vendors, the bases for the inspection are 10 CFR Part 21 and the quality and technical requirements contractually imposed on the vendor by its customers.

DISCUSSION ON TEXT OF THE REEDY ASSOCIATES, INC. NOVEMBER 2, 1993, LETTER

In your November 2, 1993, letter to the NRC, you stated, "Criterion I points out that the permit holder or licensee responsible for delegating to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but must retain responsibility for quality assurance." We agree that Criterion I of Appendix B permits the licensee to delegate the responsibilities to others, however Criterion I also states, in part, that the authorization for delegating the performance of these responsibilities (activities) shall be clearly established and delineated in writing, including both the performing functions of attaining quality objectives and the quality assurance function. Criterion I continues by stating "The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed..." Therefore, a contractor accepting the responsibility of performing safety-related activities is also responsible for assuring that its QA program controlling these activities is effectively executed.

In your letter, you also stated, "Criterion IV requires permit holders and licensees to have measures that assure that applicable regulatory requirements necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, or services. It also requires that, to the extent necessary, procurement documents require contractors or subcontractors to provide a quality assurance program consistent with the provisions of Appendix B," and that "Criterion VII requires permit holders and licensees to assess the effectiveness of the quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services." Licensees determine the importance, complexity, and quantity of services performed by

consultants such as Reedy Associates, Inc. (Reedy), and based on their evaluation of the consultant's capabilities and its QA program and implementation, invoke the applicable quality and technical requirements in their procurement documents. If a licensee invokes the requirements of Appendix B in its procurement document, consultants such as Reedy would be contractually obligated to comply with all applicable requirements including the performance of internal audits required by Criterion XVIII, "Audits."

Although contractors who perform safety-related design activities are not explicitly mandated by Criterion XVIII of Appendix B to perform internal audits, as previously discussed, Criterion I, "Organization," and Criterion IV, "Procurement Document Control," requires that procurement documents specify and impose applicable requirements to assure the adequate quality of the delivered item or service. Therefore, procurement documents are the vehicle by which the licensee specifies the necessary aspects of Appendix B which a contractor such as Reedy must comply.

It is the NRC staff's expectation that (a) where the licensee has determined, pursuant to Criteria I, III, and IV of Appendix B, that the activities performed by its contractor must be subjected to an internal audit by the contractor in accordance with Criterion XVIII, and (b) the licensee has imposed such compliance on its contractor by the licensee's procurement documents, then the contractor is responsible for ensuring the required internal audits are performed.

APPLICABLE REGULATORY GUIDES AND ANSI/ASME STANDARDS

The NRC, through its regulatory guides (RG) has endorsed several ANSI/ASME standards, as acceptable methods for meeting the requirements of Appendix B. In some instances, the NRC has taken exception to portions of the ANSI/ASME standards or provided additional positions not addressed in the standards. Most licensees have committed to these ANSI/ASME standards, as endorsed by the NRC RGs, as part of their licensing commitments (some licensees may have proposed alternatives to parts of these RGs and standards, but in general, they form the bases for licensee's QA program commitments).

ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants," as endorsed by RG 1.28-1972, "Quality Assurance Requirements (Design and Construction)," and ANSI N45.2-1977 as endorsed by RG 1.28, Revision 2, dated February 1979, provide a general set of requirements for meeting Appendix B and refer to the ANSI N45.2 series standards to provide details for implementing selective portions of Appendix B. In August 1985, the NRC issued Revision 3 to RG 1.28 endorsing ANSI/ASME NQA-1-1983, "Quality Assurance Requirements for Nuclear Power Plants," and the ANSI/ASME NQA-1-1a-1983 Addenda. NQA-1 incorporates the requirements of ANSI N45.2 and selected ANSI N45.2 series standards. Although not a complete listing, selective text from several of documents that address your inquiry are provided below.

1. Section 1.2, "Scope," of ANSI N45.2-1971 states, in part, that this standard applies to the plant owner as well as other organizations participating in activities affecting quality. Other standards have similar text (e. g. NQA-1, states in part, that the organization upon which this standard, or portions thereof, is invoked shall be responsible for complying with the specified requirements). Both ANSI N45.2 and NQA-1 impose similar requirements for internal audits.
2. Section 11, "Audits," of ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants," as endorsed by RG 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants," clearly describes provisions for internal audits of organizations performing design activities.
3. Section 1.2, "Applicability," of ANSI N45.2.12-1977, "Requirements for Auditing Quality Assurance Programs for Nuclear Power Plants," as endorsed by RG 1.144-1980 states, in part, that the requirements of this standard apply to both internal and external audits performed by or for the plant owner, contractors, or other organizations participating in safety-related activities. Section 1.3, "Responsibility," continues by addressing the interface aspect where the licensee and a contractor both have the responsibility to perform audits and states "Where multiple organizational arrangements (for auditing) exist, the interface responsibilities for each organization shall be clearly defined and documented. In no way shall the performance of audits by an organization diminish the responsibility of the audited organization or contractor of his designated portion of the quality assurance program or the quality of his product or services."
4. Section 1.3, "Responsibility," of ANSI N45.2.12-1977 states, in part, "The work of establishing practices and procedures and providing the resources in terms of personnel, equipment, and services necessary to meet the requirements of this standard may be delegated to other organizations and such delegation shall also be documented."
5. NRC Regulatory Position C.5 of RG 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants," Revision 1, dated September 1980, states: "Where more than one purchaser buys from a single supplier, a purchaser may perform an audit of the supplier on behalf of more than one purchaser in order to reduce the number of external audits of the supplier. The results of this audit should be distributed to all purchasers for whom the audit was conducted."

Although this position addresses external audits, its provision that the audit be performed on behalf of a purchaser would also be applicable for internal audits (the audit would have to be performed on behalf of the contractor).

6. Section 1.2, "Responsibility," of ANSI N45.2.13-1976, "Quality Assurance Requirements for the Control of Procurement of Items and Services for

Nuclear Power Plants," as endorsed by RG 1.123-1977, states, in part, that the purchaser shall assure that the quality assurance requirements incorporated in its procurement documents satisfy the requirements of ANSI N45.2 and applicable supplementary standards as applicable to the items or services procured. The appendix to ANSI N45.2.13-1976 provides an example of a supplier performing design reviews and the degree of quality assurance that is expected for compliance with ANSI N45.2. The example indicates that it is appropriate to expect the supplier to perform audits. Also, Section 7.1, "General," of Section 7, "Verifications to be Performed by the Purchaser," of ANSI N45.2.13, requires, in part, that the purchaser's verification activities are not intended to relieve the supplier of its responsibilities for verification of quality requirements.

The NRC, through its endorsement of standards such as those referenced in this letter, has established a position that there is a need for both licensees and contractors to perform internal audits of their activities. However, both the licensee and its contractors, as provided for in Criterion I of Appendix B, may delegate the performance of internal audits to others.

Contractors performing safety-related design services that have accepted procurement documents that invoke the requirements of Appendix B to 10 CFR Part 50, or ANSI N45.2, or NQA-1, or other similar criteria that require audits, should perform internal audits of their safety-related design activities. These internal audits would serve to verify that the contractor's quality assurance program, including design control, was being effectively implemented for the contractor's work scope.

GENERAL COMMENTS

A licensee's procurement documents for safety-related services normally impose the requirements of 10 CFR Part 21 and Appendix B to 10 CFR Part 50 on the contractor performing these services and require that these requirements be passed onto subcontractors performing safety-related services for the contractor.

It is not uncommon for contractors, such as Reedy, that employ a small number of personnel to qualify and contract an independent organization to perform its internal audits.

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

November 15, 1994

EA 94-049

Mr. Steve M. Quist, President
Rosemount Nuclear Instruments, Incorporated
8200 Market Boulevard
Chanhassen, Minnesota 55317

Dear Mr. Quist:

SUBJECT: NOTICE OF VIOLATION (NRC INSPECTION REPORT 99900271/93-01 & OFFICE
OF INVESTIGATIONS CASE No. 4-90-009)

This letter concerns the Rosemount Nuclear Instruments, Incorporated (Rosemount) presentation given to NRC staff at a June 23, 1994, enforcement conference and the NRC staff conclusions based on NRC review of and deliberation on the circumstances in this matter. Our deliberations took into consideration all of the information obtained and developed by NRC staff from 1988 to the present.

In 1993, NRC staff conducted an inspection to follow up on an NRC concern regarding an oil-loss problem identified in Rosemount Model 1150-series pressure transmitters that could have resulted in undetectable degraded operation of Rosemount nuclear safety-related pressure transmitters. Based on that inspection and on the findings of a related Office of Investigation (OI) investigation, certain of your activities were found to be in violation of 10 CFR Part 21 (Part 21) requirements, as specified in the enclosed Notice of Violation (Notice). The violation is of concern because Rosemount did not fulfill its Part 21 responsibilities to its customers between 1984 and 1988, when Rosemount became aware of degraded operation of its 1150-series transmitters in nuclear safety-related applications due to oil-loss problems in the transmitter's sensor cell. Although the initiating events and Rosemount's initial Part 21 violation occurred a considerable time ago, they represent a significant weakness in the implementation of Rosemount's 10 CFR Part 21 and 10 CFR Part 50, Appendix B programs.

The NRC staff determined that in the mid-1980s, Rosemount provided incomplete and inaccurate information to NRC licensed facilities regarding the scope of the Rosemount transmitter oil-loss problem and that some Rosemount staff were not candid about providing comprehensive information regarding the oil-loss matter. Several licensees subsequently used this information, in part, for their evaluation of the deviation pursuant to 10 CFR Part 21. The documents and testimony that were reviewed by NRC staff indicated that Rosemount limited the oil-loss information it disclosed and the information disclosed was typically site specific. Further, Rosemount did not describe the sensor cell manufacturing and test weaknesses which could have allowed transmitters that exhibited slow leaks to be shipped to NRC licensees. These weaknesses in the manufacturing and test processes preceding the late 1980s were potentially generic, could have caused degraded transmitter operation in any of

Rosemount's 1150-series transmitters, and should have been sent to NRC licensees for evaluating their specific transmitter safety-related applications pursuant to 10 CFR Part 21.

Based on OI's investigation and the NRC staff's inspection, the NRC has concluded that Rosemount acted in careless disregard of 10 CFR Part 21 requirements and its own procedures by failing to adequately evaluate or to inform its customers of the potential for degraded transmitter operation as a result of sensor cell oil-loss. We have determined that Rosemount's actions in this period between 1984 and 1988 - the multiple opportunities to have recognized the generic implications of numerous 1150 series transmitter problems, the repeated failure to recognize these problems by experienced Rosemount personnel, and the reluctance of Rosemount personnel to allow candid communications in order to adequately inform customers of the deviations - reflect careless disregard for the requirements of 10 CFR Part 21. Therefore, in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), 10 CFR Part 2, Appendix C, the failure to perform an adequate Part 21 review and to inform your customers of the potential for degraded operation of Rosemount's 1150-series transmitters in safety-related applications is being classified as a Severity Level II violation. Your past performance in this matter was unacceptable. A civil penalty has not been proposed inasmuch as we were not able to identify any Rosemount Director or Responsible Officer subject to 10 CFR Part 21 who knowingly and consciously failed to provide the notice required by 10 CFR 21.21. In addition, the general standard for imposition of a civil penalty against a licensee under Section 234 of the Atomic Energy Act of 1954, as amended, which is less specific and stringent than the "knowingly" and "consciously" standard for imposition of a civil penalty against a vendor pursuant to 10 CFR Part 21, does not apply in these circumstances.

The failure of the Rosemount staff to candidly inform NRC licensees of the potential oil-loss failure mode in the four years that Rosemount was aware of the problem constitutes a very significant regulatory concern and indicates significant weaknesses in your implementation and compliance with 10 CFR Part 21. The NRC staff does acknowledge that after the Part 21 responsibilities were discussed with Rosemount during the 1993 inspection effort, Rosemount undertook corrective action to preclude recurrence. For example, Rosemount has revised, and provided NRC staff with, its 10 CFR Part 21 procedure DP-N-1626, "Implementation of 10 CFR Part 21 for Deviations and Failures to Comply in Nuclear Products." The NRC staff also notes that the current Rosemount staff interactions with NRC seem more open and candid than in the past.

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 take in order 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. The NRC staff will review the adequacy of Rosemount's corrective actions during a future inspection.

Further, as a result of the inspection and investigation activities conducted by the NRC, the staff concluded that Rosemount provided inaccurate and incomplete information to the NRC during an April 13, 1989, public meeting regarding the failure experience of the Rosemount Model 1152 transmitter. However, the NRC staff found that Rosemount did not deliberately supply the NRC staff with inaccurate and incomplete information at this meeting. The NRC has not proposed any enforcement action for the inaccurate and incorrect statements made to the NRC and its licensees because the applicable regulation (10 CFR 50.9) applies to applicants and licensees and not an entity like Rosemount under the circumstances of this case. However, the staff has substantial concerns about this matter and the staff must emphasize that Rosemount's actions resulting in the submittal of inaccurate or incomplete information to the NRC and licensees are wholly unacceptable. The NRC expects all licensee and vendor communications to be complete and accurate, and to properly reflect situations that could have implications to public health and safety, especially where an evaluation pursuant to 10 CFR Part 21 is involved. Therefore, in addition to your response to the Notice, you are requested to provide, under oath or affirmation, a written description of those actions that Rosemount has taken or intends to take to ensure that information provided to the NRC or its licensees is complete, candid, and accurate in all material respects. The response to this request should be provided within 30 days of the date of this letter.

Additionally, it is noted that the NRC staff received a Rosemount letter dated September 28, 1994. In this unsolicited letter, Rosemount stated that it agreed with the NRC "views" expressed at the June 23, 1994 enforcement conference on the importance of 10 CFR Part 21 and was "pleased with the conclusion... that no deliberate violation occurred." Rosemount also stated, however, that it cannot concur in the view that Rosemount acted in careless disregard by failing to adequately identify and report potential defects in its Model 1153 pressure transmitters prior to December 1988. The NRC staff noted that Rosemount attached a 40 page enclosure to its letter that takes exception to a number of statements and conclusions delineated in NRC Inspection Report 99900271/93-01 which provided some of the bases for the enforcement action that is set forth in the attached Notice of Violation. The NRC staff has reviewed the information in the Rosemount letter and has determined that the letter does not provide new information or arguments that would cause any change in the staff's position or enforcement action.

Mr. Steve M. Quist

4

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

Sincerely,

A handwritten signature in black ink, reading "Frank J. Miraglia". The signature is written in a cursive style with a large, stylized "F" and "M".

Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

Docket No. 99900271

Enclosure:
Notice of Violation

cc: See next page

cc: Mr. Kenneth E. Ewald, Business Unit Manager
Rosemount Nuclear Instruments, Incorporated
12001 Technology Drive
Eden Prairie, Minnesota 55344

Mr. Paul Blanch
135 Hyde Road
West Hartford, Connecticut 06117

Mr. Ernest Hadley
414 Main Street
Post Office Box 3121
Wareham, Massachusetts 02571

NOTICE OF VIOLATION

Rosemount Nuclear Instruments,
Incorporated
Chanhassen, Minnesota 55317

Docket No. 99900271
EA 94-049

During an NRC inspection conducted between February 1 and 4, and March 8 and 12, 1993, and an NRC investigation that was conducted between February 1990 and November 1993, the staff identified a violation of NRC requirements. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1993), the violation is listed below:

- A. Section 21.21, "Notification of failure to comply or existence of a defect," of the version of 10 CFR Part 21 in effect from January 1978 until July 1991, stated, in part:

Each individual, corporation, partnership, or other entity subject to the regulations in this part shall adopt appropriate procedures to (1) provide for (i) evaluating deviations or (ii) informing the licensee or purchaser of the deviation in order that the licensee or purchaser may cause the deviation to be evaluated unless the deviation has been corrected.

Rosemount Nuclear Instruments, Inc. is subject to 10 CFR Part 21.

Contrary to the above, the Measurement Division of Rosemount, Incorporated (Rosemount), Quality Implementation Procedure (QIP) 126(N), "Potential Defect or Deviation in Products for Nuclear Application," issued March 18, 1981, was not adequately established and implemented to ensure that deviations, as defined in 10 CFR 21.3(e), about which Rosemount did not have adequate knowledge of each of the applications at NRC licensee facilities, were either evaluated or transmitted to the applicable customer or licensee for evaluation. Specifically, between 1984 and December 1988, Rosemount failed to properly inform licensees of a potential for a sensor cell oil-loss problem that could occur in its nuclear-safety-related 1150-series pressure transmitters which could have caused safety limits to be exceeded or caused substantial safety hazards to exist at licensee facilities.

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

Pursuant to the provisions of 10 CFR 2.201, Rosemount Nuclear Instruments, 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. 20555-0001 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: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time. Under the authority of Section 182 of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Dated at Rockville, Maryland
this 15th day of November 1994.

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

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

NUREG-0040
Vol. 18, No. 4

2. TITLE AND SUBTITLE

Licensee Contractor and Vendor Inspection
Status Report
Quarterly Report
October - December 1994

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	1995

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Quarterly

7. PERIOD COVERED (Inclusive Dates)

October -
December 1994

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

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

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

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from October through December 1994.

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

Vendor Inspection

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE