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LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT

QUARTERLY REPORT
OCTOBER 1985 - DECEMBER 1985

UNITED STATES NUCLEAR REGULATORY COMMISSION

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OCTOBER 1985 - DECEMBER 1985

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Division of Quality Assurance, Vendor and Technical Training Center Programs
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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PREFACE

A fundamental premise of the Nuclear Regulatory Commission's (NRC) nuclear facility licensing and inspection program is that a licensee is responsible for the proper construction and safe operation of nuclear power plants. The total government-industry system for the inspection of 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 accordance with requirements prescribed by, or consistent with, NRC rules and regulations. The NRC inspects to determine whether its requirements are being met by a licensee and his contractors, while the great bulk of the inspection activity is performed by the industry within the framework of ongoing quality verification programs.

In implementing this multilayered approach, a licensee is responsible for developing a detailed quality assurance (QA) plan. This plan includes the QA programs of the licensee's contractors and vendors. The NRC reviews the licensee's and contractor's QA plans to determine that implementation of the proposed QA program would be satisfactory and responsive to NRC regulations.

In the case of the principal licensee contractors, such as nuclear steam supply system designers and architect engineering firms, the NRC encourages submittal of a description of corporate-wide QA programs for review and acceptance by the NRC. Upon acceptance by NRC, described QA programs provide written bases for inspection on a generic basis, rather than with respect to specific commitments made by a particular licensee. Once accepted by NRC, a corporate QA program of a licensee's contractor will be acceptable for all license applications that incorporate the program by reference in a Safety Analysis Report (SAR). In such cases, a contractors's QA program will not be reviewed by the NRC as part of the licensing review process, provided that the incorporation in the SAR is without change or modification. However, new or revised regulations, Regulatory Guides, or Standard Review Plans affecting QA program controls may be applied by the NRC to previously accepted QA programs. The status of NRC review of QA topical reports submitted by the principal contractors is shown in Table 1.

When design and construction activities were high, firms designing nuclear steam supply systems, architect engineering firms designing nuclear power plants, and certain selected major equipment vendors were inspected on a regular basis by NRC to ascertain through direct observation of selected activities whether these design firms and vendors were satisfactorily implementing the accepted QA program. However, with the substantial decline of new plant design activities, the inspection of QA program implementation has been deemphasized. Instead, the NRC vendor inspection focus has been shifted to vendor activities associated with nuclear plant operation, maintenance, and modifications. Inspection emphasis in now placed on the quality of the vendor products including hardware fabrication, licensee-vendor interfaces, environmental qualification of equipment, and equipment problems found during operation and corrective action. If nonconformances with NRC requirements and regulations are found, the inspected organization is requested to take appropriate corrective action and to institute preventive measures to preclude recurrence. If generic implications are identified, NRC assures that affected licensees are expeditiously informed.

In the past, NRC issued confirming letters to the principal contractors to indicate that NRC inspections have confirmed satisfactory implementation of the accepted QA programs. Licensees and applicants could, at their option, use the letters to fulfill their obligation under 10 CFR 50 Appendix B, Criterion VII, that requires them to perform initial source evaluation audits and subsequent periodic audits to verify QA program implementation. However, based on the above described change in nuclear plant design and construction activities, NRC will no longer issue confirming letters to principal contractors since future NRC vendor program inspections will focus on selected areas rather than addressing the implementation of their respective QA programs. Therefore, confirming letters that have already exceeded their three year effective period will not be renewed. Confirming letters issued less than three years ago will remain in affect until the stated effective period expires. Therefore, as the confirming letters expire, licensees and applicants will no longer be allowed to take credit for the NRC acceptance of the implementation of a principal contractor's QA program. Licensees continue to be responsible for the conduct of initial source evaluation audits and subsequent periodic audits to verify QA program implementation. The NRC Division of Quality Assurance, Vendor and Technical Training Center Programs will continue to review revisions to principal contractor QA programs when submitted and, when approved, will list the latest approved revision number and date of the approval letter in Table 1 of the next edition of the White Book.

The White Book will continue to be published and will contain copies of all vendor inspections issued during the calendar quarter specified. The vendor inspection reports list the nuclear facilities to which the results are applicable thereby informing licensees and vendors of potential problems. In addition, the affected NRC Regional Offices are notified of any significant problem areas that may require special attention.

The White Book contains information normally used to establish a "qualified suppliers" list; however, the information contained in this document is not adequate nor is it intended to stand by itself as a basis for qualification of suppliers.

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

Copies of the White Book may be obtained at a nominal cost by writing to the National Technical Information Service, Springfield, Virginia 22161.

ORGANIZATION: COMPANY, DIVISION
CITY, STATE

REPORT NO.: Docket/Year Sequence	INSPECTION DATE(S):	INSPECTION ON-SITE HOURS:

CORRESPONDENCE ADDRESS: Corporate Name
Division
ATTN: Name/Title
Address
City/State/Zip Code

SAMPLE PAGE
(EXPLANATION OF FORMAT AND TERMINOLOGY)

ORGANIZATIONAL CONTACT: Name/Title
TELEPHONE NUMBER: Telephone Number

NUCLEAR INDUSTRY ACTIVITY: Description of type of components, equipment, or services supplied.

ASSIGNED INSPECTOR: Signature ________________________ ________
Name/VPB Section

OTHER INSPECTOR(S): Name/VPB Section

APPROVED BY: Signature ________________________ ________
Name/Section/Vendor Program Branch

INSPECTION BASES AND SCOPE:

A. BASES: Pertain to the inspection criteria that are applicable to the activity being inspection; i.e., 10 CFR Part 21, Appendix B to 10 CFR Part 50 and Safety Analysis Report or Topical Report commitments.

B. SCOPE: Summarizes the specific areas that were reviewed, and/or identifies plant systems, equipment or specific components that were inspected. For reactive (identified problem) inspections, the scope summarizes the problem that caused the inspection to be performed.

PLANT SITE APPLICABILITY: List docket numbers and plant name of licensed facilities for which equipment, services, or records were examined during the inspection.

ORGANIZATION: ORGANIZATION
CITY, STATE

REPORT NO.:	INSPECTION RESULTS:	PAGE 2 of 2

A. VIOLATIONS: Shown here are any inspection results determined to be in violation of Federal Regulations (such as 10 CFR Part 21) that are applicable to the organization being inspected.

B. NONCONFORMANCES: Shown here are any inspection results determined to be in nonconformance with applicable commitments to NRC requirements. In addition to identifying the applicable NRC requirements, the specific industry codes and standards, company QA manual sections, or operating procedures which are used to implement these commitments may be referenced.

C. UNRESOLVED ITEMS: Shown here are inspection results about which more information is required in order to determine whether they are acceptable items or whether a violation or nonconformance may exist. Such items will be resolved during subsequent inspections.

D. STATUS OF PREVIOUS INSPECTION FINDINGS: This section is used to identify the status of previously identified violations, items of nonconformance, and/or unresolved items until they are closed by appropriate action. For all such items, and if closed, include a brief statement concerning action which closed the item. If this section is omitted, all previous inspection findings have been closed.

E. INSPECTION FINDINGS AND OTHER COMMENTS: This section is used to provide significant information concerning the inspection areas identified under "Inspection Scope." Included are such items as mitigating circumstances concerning a violation or nonconformance, or statements concerning the limitations or depth of inspection (sample size, type of review performed and special circumstances or concerns identified for possible followup). For reactive inspections, this section will be used to summarize the disposition or status of the condition or event which caused the inspection to be performed.

F. PERSONS CONTACTED: Typed, Name, Title

*present during exit meeting

SAMPLE PAGE
(EXPLANATION OF FORMAT AND TERMINOLOGY)

TABLE 1

PRINCIPAL CONTRACTORS WITH APPROVED QA PROGRAM TOPICAL REPORTS

CONTRACTOR	TOPICAL REPORT DESIGNATION	REVISION	DATE OF LATEST NRC APPROVAL LETTER
Babcock & Wilcox	BAW 10096A	Revision 4	April 9, 1982
Bechtel	BQ-TOP-1	Revision 3A	August 28, 1984
Black & Veatch	BVTR-1-D	Revision 0A	August 1, 1983
C. F. Braun	21A	Amendment #5	July 16, 1980
Brown & Root	B&R-002A	Revision 3	April 8, 1980
Burns & Roe	B&ROE-COM-1-NP	Revision 3A	June 15, 1984
Combustion Engineering	CENPD-210-A	Revision 3	October 16, 1984
Ebasco Services, Inc.	ETR-1001	Revision 12	August 10, 1983
Framatome	FRA-QP/85 0782 NR	Revision 2A	Under Review
General Atomic	GA-A13010A	Amendment #8	October 15, 1984
General Electric Co.	NEDO-11209-04A	Revision 5	April 19, 1985
Gibbs & Hill, Inc.	GIBSAR 17-A	Amendment 7	August 21, 1984
Gilbert/Commonwealth	GAI-TR-106	Revision 3	August 9, 1984
Ralph M. Parsons	P-TOP-QA1	Revision 3A	August 26, 1985
Sargent & Lundy Engineers	SL-TR-1A	Revision 6	April 14, 1983
Stone & Webster	SWSQAP 1-74A	Revision E	February 6, 1986
United Engineers & Constructors	UEC-TR-001	Revision 6	September 16, 1982
Westinghouse NTD	WCAP-8370/7800	Rev. 10/6A	August 29, 1984

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

(ADDRESSEE)

Gentlemen:

A series of Nuclear Regulatory Commission (NRC) inspections have been conducted to review your implementation of the quality assurance program applicable to NRC applicants or licensees who have contracted for services from the (applicable corporate entity). These inspections consisted of selective examination of procedures and representative records, interview of personnel, and direct observation by the inspectors. As a result of these inspections, the NRC has concluded that the QA program described in Topical Report ________ is being implemented satisfactorily. Neither this conclusion nor the remainder of this letter applies to manufacturing activities or construction-related activities conducted at reactor sites.

Licensees and applicants that have referenced the above Topical Report in their Safety Analysis Reports (or have adopted the total quality assurance program described in that Topical Report) may, at their option, use this letter to fulfill their obligation under 10 CFR Part 50, Appendix B, Criterion VII, that requires them to perform initial source evaluation/selection audits and subsequent periodic audits to assess the quality assurance program implementation.

The NRC expression of satisfaction with the implementation of your quality assurance program does not assure that a specific product or service offered by you to your customer is of acceptable quality, nor does it relieve the applicant or licensee from the general provision of Criterion VII which requires verification that purchased material, equipment, or services conform to the procurement documents. It is recognized that in some cases this assurance can be made by the applicant or licensee without audits or inspections at your facility.

Continuing acceptability of implementation of your quality assurance program is contingent upon your maintaining a satisfactory level of program implementation, certified through periodic NRC inspection, throughout all corporate organization units and nuclear projects encompassed by your program. Should your program implementation at any time be found unacceptable you will be notified by letter and requested to correct the deficiencies promptly. In the event you fail to correct the deficiencies promptly, or if the record of deficiencies is such as to indicate generally poor program implementation, you and the applicants and licensees who have referenced your quality assurance program will be notified that the generic implementation of your program is no longer acceptable to the NRC. All of the audit/inspection requirements of Criterion VII, Appendix B, 10 CFR Part 50, must then be implemented by the applicants or licensees. The NRC will reinstate its letter of acceptability of implementation of your quality assurance program only after our inspectors have concluded, based on reinspection, that you have again demonstrated full compliance.

Except as noted above, the conclusions expressed in this letter will be effective for 3 years from the date of issue of the letter. At that time, program performance over the previous 3-year period will be evaluated and this letter reissued, if appropriate.

The results of our inspections are published quarterly in the Licensee Contractor and Vendor Inspection Status Report (NUREG 0040), which is made available to NRC facility applicants, licensees, contractors, and vendors as well as to members of the public, by subscription.

Sincerely,

Director
Division of Quality Assurance, Vendor
and Technical Training Center Programs
Office of Inspection and Enforcement

x

ORGANIZATION: AIR BALANCE INCORPORATED
WESTFIELD, MASSACHUSETTS

REPORT NO.: 99901005/85-01	INSPECTION DATE(S): 7/11-12/85 and 8/5-9/85	INSPECTION ON-SITE HOURS: 88

CORRESPONDENCE ADDRESS: Air Balance Incorporated
Division of Reed National Corporation
ATTN: Mr. S. B. Reed - President
260 North Elm Street
Westfield, Massachusetts 01085

ORGANIZATIONAL CONTACT: Mr. Randy Wright - Assistant Product Manager
TELEPHONE NUMBER: (413) 568-9571

PRINCIPAL PRODUCT: Fire Dampers

NUCLEAR INDUSTRY ACTIVITY: Approximately 2%. Current in-house nuclear orders: Millstone 3, Braidwood, St. Lucie, Vogtle, Shearon Harris, and Robinson.

ASSIGNED INSPECTOR: Joseph J. Petrosino 12/5/85
J. J. Petrosino, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): E. L. Burns, Brookhaven National Laboratory

APPROVED BY: E. W. Merschoff 12/6/85
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: (1) Obtain information in regard to curtain type fire damper deficiencies, (2) evaluate the Air Balance quality assurance program for adequacy and implementation of applicable requirements.

PLANT SITE APPLICABILITY: Beaver Valley 1 & 2 (50-334/412); Braidwood 1 & 2 (50-456/457); Clinton (50-461); Comanche Peak 1 & 2 (50-445/446); D.C. Cook 1 & 2 (50-315/316); Crystal River (50-302); Davis Besse (50-346);

ORGANIZATION: AIR BALANCE INCORPORATED
WESTFIELD, MASSACHUSETTS

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PLANT SITE APPLICABILITY: (continued) Duane Arnold (50-331); Farley 1 & 2 (50-348/364); Fitzpatrick (50-333); Enrico Fermi 2 (50-341); Grand Gulf 1 & 2 (50-416/417); Haddam Neck (50-213); Indian Point 2 & 3 (50-247/286); Kewaunee (50-305); Limerick 1 & 2 (50-353/352); Millstone 1, 2 & 3 (50-245/336/423); Monticello (50-263); Nine Mile Point 1 & 2 (50-220/410); Oyster Creek (50-219); Palisades (50-255); Peach Bottom 2 & 3 (50-277/278); Perry 1 & 2 (50-440/441); Pilgrim 1 (50-293); Point Beach 1 & 2 (50-266/301); Quad Cities 1 & 2 (50-254/265); Rancho Seco 1 (50-312); River Bend 1 & 2 (50-458/459); Robinson 2 (50-261); San Onofre 1, 2 & 3 (50-206/361/362); Seabrook 1 & 2 (50-443/444); Shoreham (50-322); St. Lucie 1 & 2 (50-335/389); Summer (50-395); Susquehanna 1 & 2 (50-281/387); Three Mile Island 1 & 2 (50-289/320); Vermont Yankee (50-271); Vogtle 1 & 2 (50-424/425); Waterford 3 (50-382); Watts Bar 1 & 2 (59-390/391); Washington Nuclear 1, 2 & 3 (50-460/397/508).

A. INSPECTION ISSUES:

1. Determine if failures of curtain type fire dampers to close under certain flow conditions, as reported by Ruskin Manufacturing Company (RMC), apply to Air Balance (ABI) supplied fire dampers.

2. Review the ABI quality assurance program for adequacy and implementation in regard to NRC regulations.

B. INSPECTION FINDINGS:

1. The failure of fire dampers to close under certain flow conditions is applicable to the ABI curtain type fire damper. Although ABI does do some testing of curtain type fire dampers under duct flow conditions, ABI's Assistant Product Manager stated that the ratings provided on the specification sheets are not guaranteed and therefore it is possible that damper closure may not occur under all duct flow conditions. This issue also affects other similar designed curtain type fire damper manufacturers, as discussed below.

2. The ABI quality assurance manual (QAM) adequately addresses all 18 criteria of Appendix B to 10 CFR Part 50, and ANSI N45.2, as required. However, the QA program implementation is inadequate in several areas as discussed below. A lack of ABI management support for the QA program was also apparent.

C. SUPPLEMENTARY INFORMATION:

1. An NRC inspection at the Ruskin Manufacturing Company (RMC) offices in February 1985 in conjunction with discussions with personnel from RMC, Air Balance, Incorporated (ABI), and Underwriters Laboratories indicates that both ABI and RMC curtain type (CT) fire dampers (FD) could fail to close under certain flow conditions. The failure frequency of the CT fire dampers to close under flow conditions was determined to be relative to one or more of the following factors:

   a. Size of the individual fire damper - As the size of the fire damper increased, the flow velocity at which it could close decreased.

   b. Velocity of air flow - The test reports indicated more closure difficulties under higher flow velocities.

   c. Horizontally installed dampers - RMC's horizontal damper test report results indicated lower flow velocity rates at which the CT-FD's would fail to close, than the vertically installed dampers.

   d. Negator springs - Curtain type fire dampers without any negator springs to assist the closure would be the most susceptible to failures during closure under flow.

   However, potential failures of the curtain type fire dampers to close under certain flow conditions cannot be limited to just ABI or RMC. This appears to be a generic issue which could affect any manufacturers' similarly designed curtain type fire damper, which will be used in nuclear safety related applications.

   Currently, there are no mandatory industry wide functional test requirements to assure that the curtain type fire dampers will operate under specific flow conditions.

   There is one industry testing requirement with which all fire damper manufacturers comply. It is the Underwriters Laboratories Standard (UL), Number 555, "Standard for Fire Dampers and Ceiling Dampers". UL-555 testing methods are implemented to verify the fire hour rating of specific dampers. The tests determine the acceptability of fire damper assemblies for use where fire resistance of a specified duration of time is required.

Section 1.6 of UL-555 states, in part: "Closing reliability of fire dampers...is evaluated on the basis that...ventilating systems are automatically shut down when a fire occurs... Therefore, the ratings are applicable to fire dampers... installed in systems where air movement is effectively stopped at the start of a fire."

Therefore, the failure of fire dampers to close under certain flow conditions may possibly affect all nuclear plant systems, if the specific system design does not require air movement to be stopped at the start of a fire.

2. A total of two violations and several nonconformances were identified within the ABI quality assurance program. Implementation of the ABI quality assurance program has not been adequately performed. The ABI QA Manager appears to be the only person within the ABI organization that is implementing or cognizant of the quality assurance program.

D. VIOLATIONS:

1. Contrary to Section 21.21 of 10 CFR Part 21, appropriate procedures to evaluate deviations or inform the licensee of the deviation had not been adopted by ABI (85-01-01).

2. Contrary to Section 21.6 of 10 CFR Part 21, copies of 10 CFR Part 21 or an explanatory notice describing the regulations/procedures was not posted (85-01-02).

E. NONCONFORMANCES:

1. Contrary to Criterion I of Appendix B to 10 CFR Part 50, and ANSI N45.2, the ABI QA Manager does not have adequate organizational freedom or sufficient independence from cost and schedule, and the QAM organizational chart does not accurately depict the current organization as indicated by the following examples:

   a. The QA Manager is also the ABI Purchasing Agent for the entire manufacturing facility at Wrens, Georgia (85-01-03).

b. The QA Manager reports to the Wrens, Georgia, Plant Manager approximately 90% of the time while acting as the Wrens facility Purchasing Agent (85-01-03).

c. Annual performance evaluation of the QA Manager is performed by the Wrens facility Plant Manager (85-01-03).

d. Only seven out of eleven management positions indicated on the organizational chart, contained in the ABI-QAM, had their responsibilities delineated (85-01-04).

e. Two ABI management personnel had job titles for which responsibilities and authorities were not delineated. These titles were Vice President-Engineering and Assistant Product Manager (85-01-04).

2. Contrary to Criterion II of Appendix B to 10 CFR Part 50, and Section 2.4 of the ABI-QAM (85-01-05):

   a. The ABI President did not annually review and approve the ABI QA program.

   b. The management of other organizations participating in the QA program did not review their applicable part of the program for status and adequacy.

   c. ABI did not perform QA program indoctrination for any management personnel at either facility.

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50, and Sections 7.2.4 and 11.4 of the ABI-QAM (85-01-06):

   a. ABI Project Engineering has not performed its function of issuing the ABI "shop traveler" in accordance with the ABI-QAM.

   b. ABI QA/QC fire damper final inspection procedures had not been issued for use by inspection personnel.

4. Contrary to Criteria X and XVII of Appendix B to 10 CFR Part 50 and Sections 11.1 and 11.3 of the ABI-QAM (85-01-07):

   a. No QA/QC in-process inspection activities for any nuclear orders had been documented.

b. No in-process sampling inspections, based on MIL-STD-105D had been implemented or documented for any ABI nuclear order.

5. Contrary to Criteria XV and XVI of Appendix B to 10 CFR Part 50 and Section 16.1 of the ABI QAM (85-01-08):

   a. QA/QC hold tags are not utilized for control of nonconforming items as observed on a in-process nuclear order.

   b. QA/QC hold tags have not been issued to QA/QC inspection personnel, nor could any ABI hold tags be produced for the NRC inspectors.

   c. No ABI nonconformance report log had been established to log nonconformance reports.

6. Contrary to Criterion XVIII of Appendix B to 10 CFR Part 50 and Section 19.1 of the ABI-QAM, the annually required QA program audits have been performed only once in the last five years, at the Wrens, Georgia, facility (85-01-09).

7. Contrary to Criterion VII of Appendix B to 10 CFR Part 50 and Sections 8.1 and 8.2 of the ABI-QAM, an accurate and current approved vendors list was not maintained (85-01-10).

F. OTHER FINDINGS OR COMMENTS:

1. Quality Assurance Program

It was observed that the ABI QA Manager is stationed at the Wrens, Georgia, facility where his main duties are as the Purchasing Agent. Discussions indicated that 90% of his time was spent as Purchasing Agent and the remainder as QA Manager. Additionally, the QA Manager functions as the QC inspector at Wrens. However, it was learned that annual performance evaluations for the QA Manager, for possible salary increases, are completed by the ABI manufacturing Plant Manager. A major portion of the responsibilities of the Purchasing Agent is procurement of manufacturing materials, equipment, services and the scheduling of production, which renders the QA Manager insufficiently independent from cost and schedule. Since the QA Manager and Purchasing Agent functions are performed by the same person a conflict is apparent.

A review of the QA Manual indicates it adequately addresses Appendix B requirements. However, implementation and updating of certain administrative portions of the manual is necessary for compliance to the regulations. Discussions with ABI personnel in conjunction with the QAM review appeared to indicate that other than the QA Manager, no ABI personnel are cognizant of the QA program or their responsibilities. This was evidenced by:

a. Lack of adequate records of management review of the QA program.

b. Personnel lack of familiarity with their QA program responsibilities.

c. Several QA program implementation deficiencies.

It was observed that numerous quality related functions assigned throughout the QAM were not being performed at all.

Personnel were querried concerning the amount or type of QA program indoctrination or training that they had received. It was concluded that this area of personnel introduction to the ABI QA program has not been accomplished.

QA personnel do not become involved with ABI nuclear engineering or design activities until the documents are sent to Wrens, Georgia, for fabrication of the products.

2. <u>Audits and Approved Vendors List</u>

ABI corporate headquarters personnel indicated that no internal audits are performed on the Wrens, Georgia, manufacturing facility and subsequent record reviews revealed that in a five year period only one audit was performed. However, annual audits have been performed at the Westfield offices. Corrective actions identified on these audits are, in part, unresolved.

The Approved Vendors List (AVL), which is a part of the QAM, was found to be inconsistent in regard to the requirements for listing the vendors on the AVL. Section 8.1 of the QAM indicated that vendors would be placed on the AVL after an audit of their QA program. However, three out of a sample of eight vendors on the AVL did not have a QA program.

3. Plant Tours

A facility tour on July 11, 1985 at the corporate offices in Westfield, Massachusetts, and a manufacturing plant tour on August 5, 1985 at the Wrens, Georgia, facility were conducted. Several deviations from the ABI QA program were observed.

The manufacturing process of nuclear fire dampers is controlled in part by the use of a shop traveler, which accompanies a shop "cut sheet" print out, which delineates all measurements and "cut" locations. The shop traveler is required to be generated by the Project Engineering department with QA review for possible modifications and approval. However, the shop travelers were found to be generated and approved by the QA Manager.

4. Design and Testing

The ABI nuclear application fire damper is a curtain-type device identified by the model number prefix 319. The Assistant Product Manager stated that the model 319 is a modified version of the 1-1/2 hour fire-rated commercial damper model 119, which has been enhanced with additional rivets, a change in sill angle, and the addition of a mullion strip in order to achieve a 3-hour fire rating. A review of engineering drawing control sheet 10-1822 dated May 11, 1978, revealed that no modifications have been made to the model 319 basic damper since its original adaptation from the model 119 damper. The model 319 fire damper is equipped with either an electro-thermal-link (ETL) or a fusible link, depending on damper size, for ensuring closure. ABI personnel stated that no complaints or reports of malfunction had been received concerning performance of this product. However, the Assistant Product Manager did disclose that ratings provided on model 319 (marketing) specification sheets are 'not guaranteed,' and therefore it is possible that damper closure may not occur in all applications under all duct flow conditions. Testing of this damper is frequently conducted, primarily for satisfying procurement contract requirements, and as a result considerable free area, flow, and leakage data is available for various model 319 sizes for use in the horizontal as well as vertical application. Since the ABI laboratory facilities are not equipped to perform large-scale, elevated-velocity flow evaluations, damper testing for licensee acceptance is typically conducted by a subcontractor, American Warming & Ventilation Company. A review of a typical performance test conducted for Pullman Sheet Metal Works on behalf of CP&L/Shearon Harris dated November 1984, revealed the use of a comprehensive procedure addressing static pressure, total air flow velocity, maximum allowable leakage, and a requirement for three (3) consecutive closures under maximum simulated conditions. From the review of this document it was apparent that under certain conditions, the model 319 fire damper will perform as required. However, there was no documentation correlating the test conditions to operating conditions.

It was also noted that ABI has made extensive efforts for ensuring proper damper installation and for long-term operability verification by the end user. For example, ETL instructions, which accompany all dampers so equipped, provide inspection notes and precautions for installation which are intended primarily to prevent damper failure due to human error. In addition, a review of correspondence from ABI to Bechtel Corporation, on behalf of FP&L/Turkey Point Units 3 & 4, dated December 30, 1982 provided recommendations for damper installation, annual inspection, and periodic maintenance.

5. <u>10 CFR Part 21</u>

Observations of the ABI bulletin boards to assure that adequate 10 CFR Part 21 posting was accomplished, revealed that only a modified version of Section 206 of the Energy Reorganization Act of 1974 was posted.

Subsequent discussions with ABI personnel indicated that they were not familiar with the requirements of 10 CFR Part 21. Section 21.21 of Part 21 was discussed, since it concerns the evaluation of deviations and notification requirements. The personnel were also not cognizant of these requirements. Current copies of Part 21 and Section 206 were provided to both ABI locations. Section 21.21 was briefly explained by the NRC inspectors. No procedures to evaluate deviations and notify the end users had been generated or adopted by ABI.

ORGANIZATION: AIR BALANCE INCORPORATED
WESTFIELD, MASSACHUSETTS

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Concern was expressed to ABI that correspondence with licensees may not be adequately screened for potential 10 CFR Part 21 defects, which are identified to ABI through their normal customer service channels.

During customer document package reviews it was revealed that many of the licensees had not imposed Part 21 upon ABI for the manufacture of fire dampers. However, Quality Air Design PO #22269, dated October 28, 1983 and PASNY PO Specification dated January 16, 1985 imposed Part 21 on ABI, but ABI did not impose Part 21 on their suppliers.

G. PERSONS CONTACTED:

W. Jennings, Vice President - Engineering (Westfield)
R. Wright, Assistant Product Manager (Westfield)
A. Ondik, Wrens, Georgia Plant Manager
M. Bekanon, Purchasing Agent and QA Manager (Wrens)

H. DOCUMENTS EXAMINED:

The documents listed below were reviewed by the inspectors to the extent necessary to satisfy the objectives of the inspection.

1. Quality Assurance Documents

QA Manual, Revision 1, 7/15/82
QA-QC audit report letter to ABI, 6/25/85
ABI audit of Elsie Manufacturing Co., 2/4/85
ABI audit of S&R Products Co., 2/5/85
ABI audit of Law Engineering/Testing, 2/22/85
Internal ABI audit report - manufacturing, 2/11/81
Stone & Webster audit report letter, 6/17/81
Internal ABI audit report (QA Manager to R. Wright/W. Jennings, 2/16/85
ABI audit letter to Law Engineering/Testing, 2/28/85

2. Procurement and Associated Correspondence

PASNY PO Specification No. JAF-85021-01, 1/16/85
Pullman PO to Addco (St. Lucie) #32926, 1/26/83
ABI PO to Pullman (St. Lucie) #44613, 12/31/82 (N942862)
ABI PO to Addco (St. Lucie) #46915, 3/11/83 (N944629)
Pullman PO to ABI (St. Lucie) #12948, 3/4/85 (N964856)
Pullman PO to ABI (Braidwood) #36240, 10/16/84
IDM-Pullman to Braidwood, 10/30/84 (N961001/1002)

Braidwood PO to ABI, 3/1/85 (N964400)
Pullman to Addco PO, 11/19/84 (N965777)
Braidwood related drawing, BRCC-507, 6/16/84
Crystal River PO Specification, SP5833, 2/4/77
PEC procurement report letter of ABI-QAM, 6/14/85
FP&L PO, F9025676D, 9/5/84
ABI material receipt of Edgcomb Metals, P05756, 7/23/81
CMTR, Edgcomb, T16579, 8/5/81
ABI PO to Edgcomb, 5756, 7/17/81
CMTR Edgcomb, V18257, 9/4/81
ABI PO to Edgcomb, 5852, 8/17/81 (V18257)
ABI PO to Edgcomb, 5814, 8/5/81
CMTR, Edgcomb, T16385, 8/19/81
ABI PO to Edgcomb, 5754, 7/17/81
Pullman PO to ABI, 35569, 6/18/85 (Shearon Harris)
Pullman PO to ABI, 36240, 10/16/84 (Braidwood)
Pullman PO to ABI, 35569, 11/14/84 (Shearon Harris)
QAD PO to ABI, 22269, 10/28/83 (River Bend)
IDM R. Wright to M. Bekanon (N950686), 10/18/83, specifies Part 21 applies
ABI PO to Edgcomb, 10933, 6/6/85, Part 21 imposed
Pullman PO to ABI, 12948, 3/4/85 (St. Lucie)

3. Other Documents

ABI Drawing 21285-1, 2/26/85, 319 ALV-UL design
ABI Drawing 21285-5, 2/26/84, fire damper schedule
Cygna Letter to ABI, 85021-011, 3/14/85, seismic report
Ebasco Seismic Test Report #JO 0801, 4/29/83, Chin Shan Nuclear
Gage Lab certification of gage, #1682002, 9/28/84
Gage Lab certification of meter, #1682001, 9/10/85
ABI PO to Gage Lab, Caliper #1682003 recal., 7/15/85
Purchasing Record Log/Card Catalog, PASNY #85-866
ABI Drawing Control Sheet, Model 319, 5/11/78
Closure Tests for Horizontal & Vertical Damper Model #319
S&R/ABI Instructions for ETL Installation
ABI letter to Bechtel, 12/82, Turkey Point
ABI Drawing, DSK-12618, Model #119, Free Area Chart, 5/24/83
ABI Drawing, C-11097, Model #119 Blade Chart, 2/10/77
Underwriters Laboratory Standard #UL-555, dated 5/14/79 (6/1/79)

ORGANIZATION: BABCOCK & WILCOX
LYNCHBURG, VA

REPORT NO.: 99900400/85-01	INSPECTION DATE(S): July 23, 1985	INSPECTION ON-SITE HOURS: 6

CORRESPONDENCE ADDRESS: Babcock & Wilcox
Nuclear Power Division
ATTN: Mr. T. R. Stephens, Quality Assurance Manager
P. O. Box 1026
Lynchburg, Virginia 24506

ORGANIZATIONAL CONTACT: C. Armentrout
TELEPHONE NUMBER: (804) 385-3138

PRINCIPAL PRODUCT: Nuclear Steam Supply Systems

NUCLEAR INDUSTRY ACTIVITY: Less than 1%.

ASSIGNED INSPECTOR: Naidu 10/30/85
K. R. Naidu, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S):

APPROVED BY: E W Merschoff 10/31/85
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

A. Bases: 10 CFR Part 21 and 10 CFR 50 Appendix B

PLANT SITE APPLICABILITY: 50-312 Rancho Seco Nuclear Power Plant.

ORGANIZATION: BABCOCK & WILCOX
LYNCHBURG, VA

REPORT NO.: 99900400/85-01	INSPECTION RESULTS:	PAGE 2 of 6

A. Inspection Issues

On June 10, 1985, the Rancho Seco nuclear power plant (Rancho Seco) notified the Nuclear Regulatory Commission (NRC) of an unusual occurrence (PNO-V-85-33A). During functional testing of recently refurbished Reactor Trip Breakers (RTB), one of the six RTBs failed to trip. The purpose of this inspection was to review the quality assurance records associated with recent refurbishment activities and obtain additional information to assist in evaluating the cause of failure.

B. Background Information

The reactor trip systems on all commercial nuclear power reactors must be single-failure proof and highly reliable. NUREG - 1000 describes the generic implications of the Anticipated Transient Without Scram (ATWS) events which took place at the Salem nuclear power plant on February 22 and 25, 1983. The NRC issued Generic Letter 83-28 dated July 8, 1983, which outlined the actions, including maintenance of RTBs, to be taken by licensees of operating power plants, applicants for an operating license, and Construction Permit holders. Babcock & Wilcox supplied the Nuclear Steam Supply System (NSSS) including the reactor trip breakers to Rancho Seco.

B & W arranged for the return of the Rancho Seco breakers to General Electric Company (GE), the manufacturer, for refurbishment which included the replacement of the trip arm bearings and roller bearing latch assembly. GE originally supplied these breakers as commercial grade with B & W performing the dedication to upgrade the breakers from commercial grade to safety related following procedures established by B & W.

C. Inspection Findings and Other Comments

1. Review of Receipt Inspection Procedures

B & W uses procedures 51-1156-268-00 and 51-1156-269-00 to perform receipt inspection of breakers used in a.c. and d.c. circuits respectively. Review of these procedures indicated that the maintenance instructions furnished in the GE Power Circuit Breaker booklet, GEI-50299E, were followed. The procedures are implemented by using a Receipt Inspection check list in which the following attributes are verified and documented for each breaker:

a) Visual examination
b) Dimensional checks
c) Physical characteristics
d) Rear-view arrangement
e) Undervoltage trip device (UVD) check
f) Trip time tests for UVD and shunt trip
g) Auxiliary contact function test
h) Dielectric strength test
i) Measurement of the torque on the trip shaft

2. Review of Quality Assurance Records

The quality assurance records pertaining to the six Rancho Seco breakers were reviewed, including the receipt inspection checklists and electrical test reports. No unacceptable findings were identified in the documents reviewed, indicating that the breakers were operable with respect to the documented attributes.

3. Review of B & W Records on the Qualification of Circuit Breakers

B & W contracted Wyle Laboratories (Wyle), Huntsville, Alabama to qualify a GE-AK-2-25 type circuit breaker to the requirements of IEEE-344 (seismic qualification). The circuit breaker's electrical characteristics are: 600 volts, 60 Hertz, 600 amperes, 3-poles, with one auxilary switch containing 5 normally open and 5 normally closed contacts, a shunt trip device, and an instanteneous under voltage trip device. The circuit breaker was mounted in a GE-AKD-5 metal enclosure. One Struthers Dunn FC-406 type relay (Reactor Protection System buffer relay) was also mounted in the same enclosure. B & W document No. 58-007600 dated October 27, 1975, indicates that Wyle performed the tests on August 4 and 5, 1976. Tests included resonance searching in two principal horizontal and vertical axes and proof level tests. Proof level tests were performed using 30 second random noise transients at four levels: 1/4 level generic (for TVA plants), 1/2 level generic and 3/4 level generic (for Toledo Edison), and generic. Each of these test levels was applied first in one principal horizontal, simultaneously with the vertical direction, then in the other principal horizontal, simultaneously with the vertical direction. The circuit breaker was tripped a minimum of three times at each level.

ORGANIZATION: BABCOCK & WILCOX
LYNCHBURG, VA

REPORT NO.: 99900400/85-01	INSPECTION RESULTS:	PAGE 4 of 6

The following results were observed:

1/4 Level
The response spectra were met in all directions. No structural damage or chatter on the relay or breaker was observed. The breaker drop-in time was well within the permissible 80 milliseconds.

1/2 Level
The response spectra were met in all directions. No structual damage or chatter on the relay or breaker was observed. The breaker drop-in time was satisfactory.

3/4 Level
The response spectra was met in all directions in the vertical and front-to-back test. No structural damage or chatter on the relay or breaker was observed. The relay functioned but the breaker did not trip. The problem was identified to be a maladjustment in the linkage between the undervoltage coil and the breaker. This was corrected and no further problems were encountered with the linkage. In the vertical and side to side test, no chatter on the relay or breaker occurred and the breaker tripped satisfactorily.

Structural damage to the cabinet lower corners occurred during the test. Several welds on both x-y-z brackets sheared. Permanent deformation of the cabinet frame occurred when several bolts were ripped from the cabinet, however, no projectiles or structural damage was observed that could have prevented the breaker from performing its safety related function. The welds and brackets were repaired prior to the resumption of the tests.

B & W report LR:74:6383:-01:4 dated September 27, 1974 documents an analysis performed to qualify the Ranch Seco RTB cabinets based on the qualification test discussed above.

4. Review of B & W Dedication Process

B & W purchased the RTBs as commercial grade circuit breakers from GE because GE had not qualified the AK-2-25 type circuit breakers to the requirments of IEEE-344 (seismic qualification). B & W contracted Wyle Laboratories to conduct the seismic qualification in 1975. The B & W dedication process for the RTBs consists of establishing that the RTB has physical and electrical characteristics identical to the specimen breaker qualified by test at Wyle Laboratories. Specifically, the process consists of the following:

a. Comparing the physical characteristics of each breaker with the photographs of the qualified specimen breaker to establish similarity. B & W had photographed the specimen breaker from different views to reveal the various accessories.

b. Performing the following mechanical tests.

(1) Verify that the trip shaft torque with the circuit breaker open is between 2 to 6 inch-ounces.

(2) Verify that the trip shaft torque with the circuit breaker closed is 16 inch-ounces or less.

c. Verifying the following electrical characteristics.

(1) Measure the time to open the circuit breaker by actuating the shunt trip.

(2) Measure the time to open the circuit breaker by actuating the undervoltage trip.

(3) Measure the voltage required for the undervoltage trip device to pick up.

(4) Measure the voltage required for the undervoltage trip device to drop out.

B & W performed the dedication described above on all six breakers and issued a "Certificate of Conformance" to Rancho Seco certifying that the breakers meet the class 1E requirements.

5. Conclusion

The quality assurance records reviewed indicated that, with respect to the parameter measured, the Rancho Seco RTBs were in acceptable operating condition prior to shipment to the Rancho Seco nuclear power plant.

ORGANIZATION: BABCOCK & WILCOX
LYNCHBURG, VA

REPORT NO.: 99900400/85-01	INSPECTION RESULTS:	PAGE 6 of 6

E. Persons Contacted

Babcock and Wilcox (B & W) Lynchburg, Virginia

R. Boven	Principal Engineer
C. Armentrout	Manager, QA Audits and Programs
S. Dasgupta	Manager, Procurement & Quality Control Surveillance
H. B. Prasse	Supervisor, Technical Support
T. R. Stevens	Manager, Quality Assurance
H. Stevens	Principal Engineer

F. Exit Interview

The inspector met with individuals identified in Section E and discussed the scope and findings of the inspection.

ORGANIZATION: COMBUSTION ENGINEERING
POWER SYSTEMS GROUP
WINDSOR, CONNECTICUT

REPORT NO.: 99900401/85-02	INSPECTION DATE(S): 11/18-22/85	INSPECTION ON-SITE HOURS: 122

CORRESPONDENCE ADDRESS: Combustion Engineering, Inc.
Power Systems Group
ATTN: Mr. Evan Woollacott, Vice President
Quality and Administrative Services
1000 Prospect Hill Road
Windsor, Connecticut 06095

ORGANIZATIONAL CONTACT: Mr. P. D. Ford, Supervisor, Group QA
TELEPHONE NUMBER: (203) 285-9210

PRINCIPAL PRODUCT: Nuclear Steam Supply Systems.

NUCLEAR INDUSTRY ACTIVITY: The Power Systems Group Combustion Engineering (CE), had contracts for 16 domestic reactor units to date, of which four (4) are in the design and construction phase. In addition, CE has modification/repair/service contracts for 16 reactor units.

ASSIGNED INSPECTOR: R P McIntyre 1/30/86
R. P. McIntyre, Special Projects Inspection Section (SPIS) Date

OTHER INSPECTOR(S): P. J. Prescott, SPIS
D. W. Golden, EG&G Idaho
W. C. Townsend, EG&G Idaho

APPROVED BY: John W Craig 1/31/86
John W. Craig, Chief, SPIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 50, Appendix B and CE Topical Report CENPD-210-A.

B. SCOPE: The purpose of this inspection was to review recent design modifications for (CE) facilities and to review the implementation of the CE system for the distribution and evaluation of computer code error reports.

PLANT SITE APPLICABILITY: Multiple including Palo Verde (50-528, 529, 530), Calvert Cliffs (50-317, 318), St. Lucie (50-335, 389), and Maine Yankee (50-309).

ORGANIZATION: COMBUSTION ENGINEERING
POWER SYSTEMS GROUP
WINDSOR, CONNECTICUT

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A. VIOLATIONS:

None.

B. NONCONFORMANCES:

1. Contrary to CE Quality Assurance of Design Procedure (QADP) 5.2, Section 6.0, "Computer Code Error Reports," one manager and two supervisors on the CESEC Computer Code distribution list could not produce any documented evidence that they had circulated the latest three CESEC error reports to the code users within their groups. (85-02-01)

2. Contrary to CE QADP 5.2, Section 1.2, the Reactor Vessel Level Monitoring System (RVLMS) Water Drainage calculation (19367-LOCA-026) for St. Lucie 1 and the RVLMS Phase III Prototype Testing procedure included supporting information within the calculations which was not properly referenced. (85-02-02)

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Open) Nonconformance (84-02): No internal audits have been performed on error reports pertaining to the CESEC computer code.

   As of the date of the inspection no internal audits had been performed on error reports pertaining to the CESEC computer code. Engineering Quality Assurance (EQA) is in the process of completing an internal audit including error reports on the CESEC computer code. This will be reviewed during a future inspection.

2. (Closed) Nonconformance (84-03): Computer Code FATES3A Verification analysis (0000-TH-186) was found to have insufficient information concerning the test problems to evaluate the intent or adequacy of the verification runs.

   The inspector reviewed the FATES3A original code verification file and the Revision 1 verification file as well as the code verification file for FATES3. The FATES3A verification analysis (0000-TH-186, Revision 1) included:

1. A summary description of the test problems

2. A description of the basis for selecting the test problems

3. A more complete discussion of the results of the test problems related to the verification intent and needs.

This item is considered closed.

3. (Open) Nonconformance (84-03): No verification calculations were available for the STRIKIN II computer code.

Not inspected during this inspection.

4. (Open) Nonconformance (84-03): The verification calculations performed for the CELDA and HCROSS computer codes were not independently reviewed.

Not inspected during this inspection.

5. (Open) Nonconformance (84-03): A modification implemented in the 78226 version of the CELDA computer code was not tested and verified.

Not inspected during this inspection.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Recent design modifications at CE plants: During this inspection several recent design modifications at Palo Verde and Calvert Cliffs were reviewed. These modifications included hardware changes as well as revisions to the FSAR, Technical Specifications, and analyses when appropriate.

   a. Loss of Auxiliary Pressurizer Spray System (APPS) (Palo Verde Nuclear Generating Station (PVNGS) Unit 1):

      On September 12, 1985, during a loss of load test at approximately 55% power, and the generator supplying onsite loads, the plant did not perform as expected. The scenario began when on loss of load the generator failed to provide power to onsite loads and included:

- turbine trip
- loss of all offsite power to non-essential loads when automatic transfer did not occur (including reactor coolant pumps)
- Reactor Coolant Pump (RCP) trip (caused by low bus voltage)
- Reactor trip (caused by projected low DNBR resulting from RCP coastdown)
- ECCS initiation (resulting from low RCS pressure and projected low DNBR)
- Chemical Volume Control System (CVCS) volume control tank (VCT) supply to charging pumps was drained due to failure of VCT level instrumentation, which
- required manual alignment of charging pumps to the Refueling Water Tank (RWT) and the restarting of charging pumps due to gas binding
- RCP's restored in about 1 hour

As a result of this event, Arizona Nuclear Power Project (ANPP) has proposed to the NRC three design modifications to Palo Verde. The objectives are to: improve the operators' ability to operate the charging/auxiliary spray system from the control room; provide an automatic function to reduce the amount of required operator action; and to improve the reliability of control grade level instrumentation on the volume control tank. These modifications are: (1) provide power to two critical alignment valves from 1E-Motor Control Center (MCC); (2) enhance the automatic realignment to the refueling water tank; (3) and enhance the volume control tank level instrumentation.

These proposed modifications and enhancements have been submitted to the NRC for review and approval. As of the week of the inspection these modifications had not been approved by the NRC.

The inspector reviewed the revision to the transient analysis for the steam generator tube rupture accident (SGTR) with a loss of offsite power and a fully stuck open atmospheric dump valve (ADVS). The revised SGTR analysis indicates that auxiliary pressurizer spray is not needed in the first two hours to mitigate the consequences of the accident. Main pressurizer spray is supplied via the reactor coolant pumps. The auxiliary pressurizer spray is supplied via the charging pumps included in CVCS. Also, the results of this analysis show that the response of the plant was acceptable, including off-site dose calculations which were within the acceptance criteria of 10 CFR Part 100.

The Palo Verde FSAR and Technical Specifications addressing loss of condenser vacuum and loss of load transients were reviewed with respect to the requirements of the auxiliary spray system. The review revealed that credit is <u>not</u> taken for the auxiliary spray system and that rapid primary coolant system pressurization is limited by the pressurizer safety valves. CE has provided the revision to the (SGTR) analysis as requested by Arizona Nuclear Power Project.

b. <u>Auxiliary Feedwater System (Palo Verde)</u>

Licensee Event Report (LER) No. 85-008-00 was submitted to the NRC on March 6, 1985 concerning an unanalyzed safety condition related to the auxiliary feedwater system (AFW) at PVNGS Unit 1. The FSAR assumes that the maximum AFW flow rate to the steam generators following an automatic activation is 1750 GPM. A recent analysis and close examination of pump head flow curves indicated that AFW flow rates could exceed 1750 gpm for some accidents. It was then assumed that operator action would prevent this from occurring.

After meetings and discussions between ANPP, the Architect-Engineer, Bechtel, and CE it was determined that operator action could not be guaranteed to prevent occurrence of an increased flow rate for certain types of accidents, i.e., main steam line break. CE performed subsequent analyses for a main steam line break transient and incorporated an increased AFW flow rate and different valve hysteresis characteristics. The analyses confirmed that an increase in

AFW flow rate does not result in a DNBR less than that calculated in the limiting analysis and presented in the PVNGS FSAR, therefore there is no decrease in the safety margin of the analysis.

The inspector reviewed the analysis for technical content and compliance to CE Quality Assurance of design procedures and found it to be adequate.

c. Reactor Vessel Level Monitoring System (RVLMS) Calvert Cliffs

The RVLMS is intended to provide the operator advisory information on liquid level in the upper plenum region during accidents such as the small break loss of coolant accident. The RVLMS is based on the use of heated junction thermocouple pairs located at a number of axial positions in the reactor vessel upper plenum. Each thermocouple pair consists of a heated junction and an unheated junction. When the water level in the probe, which is intended to closely represent the collapsed liquid level in the vessel, drops below the heated junction it will dry out and heat up relative to the unheated junction. When the temperature difference between the heated and unheated junction reaches the setpoint temperature the liquid level is determined to be at the level of the heated junction.

Two areas of the RVLMS program were reviewed during this inspection. First, the Phase III test program was reviewed as it applies to Calvert Cliffs. Secondly, the performance calculation for Calvert Cliffs was reviewed. Technical and procedural aspects of both areas were reviewed. The documents reviewed included RVLM Phase III test request, test requirements, test procedure and test report; Phase II test procedure; Calvert Cliffs water drainage calculation for RVLMS; St. Lucie 1 RVLMS water drainage calculation and the verification file for the RVLMS utility code.

The RVLMS Phase III testing was performed to provide design verification and determine the setpoint temperature difference at which a heated junction thermocouple pair will indicate the reactor vessel liquid level. The test program was found to provide a range of test conditions which bound the conditions which have been predicted for Calvert Cliffs during a small break loss of coolant accident.

The Calvert Cliffs RVLMS water drainage calculation was reviewed. The Calvert Cliffs calculation incorporated the St. Lucie 1 calculation. The purpose of the calculation was to establish that this system would perform acceptably under conservative conditions. The calculation was found to be technically adequate. It was noted that a properly verified computer code (RVLMS) was used. Independent review of the calculation was noted.

Failure to follow CE QAPP requirements for referencing information was found in two documents. First, in the St. Lucie 1 Reactor Vessel Level Monitoring System (RVLMS) water drainage calculation, the splash guard area was incorporated from another calculation. The reference given for the splash guard area was to the author of a supporting calculation and not to a specific calculation as required by the QAPP. Further the supporting calculation for the splash guard area was not provided in the reference list of the above cited calculation. It was noted that the independent reviewer of the cited calculation was able to provide the appropriate references in a timely manner.

Second, the "RVLMS Phase III Prototype Testing" document specified a range of depressurization rates from 0.0 to 10.0 psi/sec via bottom blowdown. No specific reference for these depressurization rates was provided. However, the reference was listed in the reference list.

Nonconformance 85-02-02 was identified during this part of the inspection.

d. <u>Calvert Cliffs Main Steam Safety Valve (MSSV) Setpoints</u>:

In November 1984 Baltimore Gas and Electric (BG&E) requested CE to perform analyses to support an expansion of the MSSV setpoint range and increase the limiting steam pressure from design pressure to 110% of design pressure. BG&E made this request to eliminate the LERs which were being written at the end of cycle due to the MSSVs drifting outside the allowable setpoint range between the beginning and end of cycle. This requires two changes to the technical specifications for Calvert Cliffs Units 1 and 2. The first is for the change in MSSV setpoints and the second is for increasing the limiting steam pressure.

The inspector reviewed correspondence between CE and BG&E regarding the MSSV setpoint issue. The inspector also reviewed CE analyses in support of the MSSV setpoint change. These analyses included complete loss of load, asymmetric steam generator operation and a Small Break Loss of Coolant Accident calculation with reduced High Pressure Injection (discussed in item e) as it applies to the MSSV setpoints. The inspector reviewed CE analyses to support application of the setpoint change to other than the reference cycle and the portion of the reload licensing package submitted to BG&E by CE for Unit 1 Cycle 8 as it applies to the change in MSSV setpoint and increasing the limiting pressure. A letter from BG&E to the USNRC requesting the technical specification change for Unit 2 was also reviewed. The inspector found that analyses were performed for changing the MSSV setpoints to allow for in cycle drift. Independent reviews were noted.

No violations or nonconformances were found in this part of the inspection.

e. High Pressure Safety Injection System (Calvert Cliffs)

This item involves a request from BG&E to evaluate applicable safety analyses with reduced High Pressure Safety Injection (HPSI) System flow, and thereby provide a leeway between actual HPSI flow and technical specification requirements in the event of reduced flow.

The analysis considered a 0.1 ft^2 LOCA transient with the HPSI system flow reduced by 35 gpm (at every system performance point) as specified by BG&E. The analysis takes credit for the charging flow whereas the original SAR analyses does not credit the charging flow. The analysis of the 0.1 ft^2 LOCA also incorporated a reduced peak linear heat generation rate as specified by the reload analyses guidelines.

The Quality Assurance requirements for design procedures were properly followed for the re-analysis, reload analysis guidelines and the revised FSAR and technical specification submittals to BG&E.

2. Availability Data Program InfoBulletin 85-07 - RCP Motor Anti-Reverse Rotation Device

During RCP maintenance at St. Lucie 1, the utility pulled the RCP motor rotor. This uncovered the Anti-Reverse Rotation Device (ARRD) when it was noted that a number of the pins which serve to lock the ratchet ring in the ARRD were stuck inside their cavities in the rotating disc. The specific ARRD involved consists of 36 pins, about 4½" long, housed in blind holes in a large metal disc attached to the motor shaft. These pins are free to drop by gravity onto a stationary toothed ratchet ring. A number of pins and teeth are arranged such that, essentially, no reverse motion of the rotating assembly can occur since the pins engage when the pump stops.

The ARRD prevents reverse rotation of the RCP in the case of reversed power leads or in cases where reverse fluid flow could result. CE performed a conservative calculation that indicates that only 4 of the 36 pins are required to prevent reverse rotation under worst-case loading. This criterion was met for the ARRD at St. Lucie. The RCP motor manufacturer at St. Lucie 1 and 2 is Allis Chalmers and the ARRD is a ratchet ring and pins type. St. Lucie is the only CE plant with this type of ARRD.

CE stated that Maine Yankee is also having problems with its ARRD. Maine Yankee also has an Allis Chalmers RCP motor but it has a Marland ARRD. This item will be reviewed during a future inspection.

3. Computer Code Error Reports

The NRC inspector conducted a review of the CE system for distributing and evaluating computer code error reports. QADP 5.2, Section 6.0, "Computer Code Error Reports," defines the procedures to be followed for distribution and evaluation of these error reports. The last three error reports for the computer code CESEC III were chosen for review.

The inspector interviewed one manager and two supervisors who are on the CESEC distribution list. They were not sure if they had received all the error reports and whether they had been circulated to the code users within their group. They could not provide the inspector with any documentation proving that this had been accomplished. QADP 5.2 requires managers and supervisors to circulate the error notice within his group to be signed and dated by code users on the response receipt section of the computer program error notification. This was not being accomplished within the groups reviewed.

Nonconformance 85-02-01 was identified during this part of the inspection.

4. <u>AE and Vendor/Supplier Interface</u>

Records and procedures involving vendor/supplier interface with CE were reviewed by the NRC inspector. The three (3) major areas involved in the review were; (1) vendor evaluation and the approved vendor list (Procedure QAP 7.1), (2) external audits (Procedure QAP 18.2), and (3) surveillance (Procedure QAP 10.1).

The NRC inspector reviewed CE's Approved Vendor List (AVL) and selected the files from two (2) of CE's suppliers. The first file reviewed was for Dresser Industries (File No. 37) to determine if the records were in compliance with CE procedure QAP 18.2 (external audits) and procedure QAP 7.1 (Vendor Evaluation). The second file reviewed was for the Crosby Valve Co. No. (PO 9601526) to determine if the records were in compliance with procedure QAP 10.1 (Surveillance).

All documents reviewed by the NRC inspectors were found to be in compliance with the applicable procedures and no nonconformances or violations were noted.

5. <u>Training</u>

Section 17.2.5 of the Quality Assurance Manual, and attendance records for training courses attended by personnel in the fluids and components department were examined by the NRC inspector.

All attendance records were found to be up to date and in compliance with Section 17.2.5 of the quality assurance manual. All associated logs and files were found to be accurate and in compliance with Section 7.0 of the quality assurance of design procedures manual.

6. <u>Internal Audits</u>

Three (3) procedures were reviewed by the NRC inspectors for the Engineering Quality Assurance (EQA) internal audits: QA of design procedures Section 9.0 quality assurance instruction section 18.01 and quality assurance manual Section 17.18. The EQA audit file (No. E-PE-0484-01) was also reviewed for compliance with QA of design procedures manual Section 7.0. All records and files examined were found to be accurate and in compliance with the applicable procedures.

ORGANIZATION: DIETRICH STANDARD CORPORATION
BOULDER, COLORADO

REPORT NO.: 99901034/85-01	INSPECTION DATE(S): 10/21-24/85	INSPECTION ON-SITE HOURS: 38

CORRESPONDENCE ADDRESS: Dietrich Standard Corporation
ATTN: Mr. James L. Benke
President
Post Office Box 9000
Boulder, Colorado 80301

ORGANIZATIONAL CONTACT: M. G. Anderson, QA Manager
TELEPHONE NUMBER: (303) 530-9600

PRINCIPAL PRODUCT: Flow Measurement Systems

NUCLEAR INDUSTRY ACTIVITY: A small portion of Dietrich Standard's flow elements/measurement systems are manufactured for use in nuclear facilities (approx. 3%).

ASSIGNED INSPECTOR: RP Correia 1/31/86
R. P. Correia, Special Projects Inspection Section, (SPIS) Date

OTHER INSPECTOR(S): P. J. Prescott, SPIS

APPROVED BY: John W Craig 1/31/86
John W. Craig, Chief, Special Projects Inspection Section Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: The inspection consisted of an evaluation of quality assurance and engineering activities in general and specifically those related to the design, procurement, manufacturing, inspection and testing of a safety-related flow element for the Ft. Calhoun Station component cooling water system.

PLANT SITE APPLICABILITY: Ft. Calhoun (50-285), Millstone 3 (50-423), Vogtle (50-424, 425)

REPORT NO.: 99901034/85-01	INSPECTION RESULTS:	PAGE 2 of 6

A. Violations

None.

B. Nonconformances

1. Contrary to 10 CFR Part 50, Appendix B, Criterion XV, and procedures in Dietrich Standard Corporation's QA Manual, Sections 10, 12, and 13, Dietrich Standard QA inspection personnel released from the segregated QA inspection area the nonconforming Ft. Calhoun safety-related flow element for a nondestructive examination prior to the completion of a nonconforming material report by engineering, project administration/procurement and quality assurance personnel. (85-01-01)

2. Contrary to 10 CFR Part 50, Appendix B, Criterion III, the Dietrich Standard flow element calculation for the Ft. Calhoun component cooling water system did not include seismic qualification as required by Omaha Public Power District specifications. (85-01-02)

3. Contrary to 10 CFR Part 50, Appendix B, Criterion XVII and Dietrich Standard QA Manual procedure section 20.2.1, the required documentation for nonconformances, repair or rework and inspections resulting from the disposition of a nonconformance report were not included in the Quality Assurance master file (No. 12302) for a Northeast Nuclear Energy Company procurement of safety-related flow elements. (85-01-03)

C. Unresolved Items

None.

D. Status of Previous Inspection Findings

This is the first Vendor Program Branch inspection of Dietrich Standard Corporation.

E. Inspection Findings and Other Comments

1. Inspection of the Manufacturing Facility

The Dietrich Standard Corporation's manufacturing facility was inspected to observe processes by which a safety-related annubar-type flow element, similar to the one which Dietrich Standard is supplying to Ft. Calhoun, is manufactured. Areas included in the inspection were material receipt and inspection, standard storage (for commercial grade materials) and bonded storage (for nuclear grade materials), machining facilities, welding, assembly and testing, and quality assurance inspection.

During the inspection of quality assurance activities, the NRC inspector examined the flow element for the Ft. Calhoun component cooling water system. Production and inspection records, the Nuclear Annubar Traveller and inspection checklists were examined. The Nuclear Annubar Traveller identifies the customer, invoice/control number, model number, drawing number, serial number, Authorized Nuclear Inspector (ANI) reviewer, preparer, quality assurance inspector and all manufacturing operations, required production activities, and inspection and ANI sign-offs. The Nuclear Annubar Traveller, when completed, becomes the permanent record of the flow element's production operations and inspections performed. During examination of the Ft. Calhoun flow element and associated records, the NRC inspectors observed a nonconformance tag on the flow element. The flow element had been rejected as a result of a critical dimension of the length of the flow element extrusion tube/transition piece being out of tolerance. This nonconformance was recorded by the QA Inspector on a nonconformance report as required by Dietrich Standard's Quality Assurance Manual (Revision 12, dated June 1, 1985) Section 13, Paragraph 13.2.1.1, "Nonconforming Material Report" (NMR).

Sections 10 and 13 of the Dietrich QA Manual "Examination and Inspection," and "Control of Nonconforming Material," respectively, require, in part, that NMR's be dispositioned and authorized by engineering, project administration/procurement and quality assurance personnel. Also, the nonconforming item must be retained in the quality assurance controlled area until such time that the dispositioned NMR is duly authorized. Section 12, "Nondestructive Examination" requires that NDE's be performed when the assembly reaches the NDE hold point.

The NDE hold point is identified as the next item on a Nuclear Annubar Traveller after the assembly is fit-up, welded, critical dimensions verified and welds visually inspected. Dietrich Standard QA personnel did not comply to the above QA requirements. Prior to the final authorization of the NMR, the Ft. Calhoun flow element was released from the QA control area for a NDE test to be performed by a contracted examiner.

Nonconformance Item 85-01-01 was identified as a result of this finding.

2. Flow Element Calculations

The NRC inspector examined the Dietrich Standard calculations and associated engineering, production, project administration/ procurement and quality assurance documentation which substantiated the selection of materials, parts and applicable code compliance for the Ft. Calhoun flow element. The inspector also reviewed the Ft. Calhoun "Specification for Replacement Flow Element for the Component Cooling System" which is part of the Omaha Public Power District's purchase order to Dietrich Standard, No. 08005, dated September 20, 1985. The NRC inspector noted that this specification listed seismic accelerations as a design parameter and that the governing code was ASME III ND Class 3, current edition and addenda. Neither a Dietrich Standard calculation nor any other documentation existed which could verify that the Ft. Calhoun flow element would operate safely as intended during a seismic event. After questioning several engineers involved with the Ft. Calhoun flow element design, the Dietrich Standard Engineering Manager explained to the NRC inspector that a seismic analysis had not been performed at that time. An unverified calculation summary sheet was presented to the NRC inspector prior to the exit meeting. The unverified summary indicated that the flow element stresses were all within the allowables prescribed by the ASME III Boiler and Pressure Vessel code.

Nonconformance Item 85-01-02 was identified in the area.

3. Quality Assurance Records

During the inspection, a review of Dietrich Standard's documentation for procurement, production and inspection procedures was conducted. For this review, the NRC inspector requested QA records of safety-related flow elements which had previously been supplied for use in a nuclear facility.

The QA manager referred the NRC inspector to the Dietrich Standard Quality Assurance Master Files, which as described in the Dietrich Standard QA Manual Section 20.2.1, "Record Retention," is required to contain all documents pertaining to the design, materials, manufacturing, testing and examinations for each order processed to ASME code requirements.

The first master file reviewed was No. N10115 SP-2 for Georgia Power Company. During the examination of this file, the NRC inspector noted on an inspection checklist, that due to a linear indication on the surface metal, a dye penetrant test was rejected for three (3) of the six (6) safety-related flow elements to be procured by Georgia Power Company. Further review determined that the Authorized Nuclear Inspector (ANI) had signed off an inspection hold point for this test on the Nuclear Annubar Traveller referencing the dye penetrant test rejection.

A nonconforming material report (NMR) No. 1663, was written to document that linear indications were found, and the NMR was dispositioned to repair the defects. The parts traveller form, used as an attachment to the NMR, documented the repair and reinspection. However, this method of documenting disposition of NMRs is not included in Dietrich Standard's QA manual.

Section 9 of the Dietrich Standard QA Manual, "Manufacturing Control" specifies usage of the parts traveller when large quantities of items are to be fabricated and control is to be maintained for various machine shop and inspection operations.

The second master file reviewed was No. 12302 for Northeast Nuclear Energy Company. The NRC inspector examined NMR No. 1703, which documented nonconformances with two (2) safety-related flow elements. The first nonconformance was dispositioned "use-as-is" and the second nonconformance required further rework. The documentation for the rework specified on the second nonconformance could not be found in the QA master file area. The QA Manager retrieved the original NMR No. 1703 from another file. The missing parts traveller containing a description of the rework and inspection documentation was photocopied on the back of the original NMR. The QA Manager deduced that when copies were made of the original documents for the master file, the parts traveller was misplaced.

The NRC inspector also noted during further review of master file No. 12302, that several NMRs referenced on various inspection checklists and nuclear travellers were not included in the file.

Nonconformance Item 85-01-03 was identified as a result of this review.

ORGANIZATION: DIETRICH STANDARD CORPORATION
BOULDER, COLORADO

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4. Other Documents Examined

The NRC inspectors examined several procurement and quality assurance records to determine whether Dietrich Standard procedures and their implementation met the requirements of 10 CFR Parts 21 and 50, Appendix B as specified on Omaha Public Power District's purchase order for the Ft. Calhoun flow element.

Purchase orders and certificates of compliance for several components of the flow element procured from sub-vendors by Dietrich Standard were examined. All sub-vendors used for these purchases appeared on Dietrich Standard's Approved Vendor's List, dated August 26, 1985. Required documentation was in place and Dietrich Standard QA Manual procedures for use of such documents had been implemented as required.

ORGANIZATION: DRESSER INDUSTRIES, INC.
ALEXANDRIA, LOUISIANA

REPORT NO.: 99900054/85-01	INSPECTION DATE(S): 09/30/85-10/04/85	INSPECTION ON-SITE HOURS: 50

CORRESPONDENCE ADDRESS: Dresser Industries, Inc.
ATTN: B. G. Bronson, QA Manager
Post Office Box 1430
Alexandria, Louisiana 71301

ORGANIZATIONAL CONTACT: Mr. B. G. Bronson, QA Manager
TELEPHONE NUMBER: (318) 640-2250

PRINCIPAL PRODUCT: Nuclear Safety and Safety Relief Valves

NUCLEAR INDUSTRY ACTIVITY: Less than 5% of Dresser Industries (Alexandria) business is supplying valves for nuclear facilities.

ASSIGNED INSPECTOR: J. C. Harper, Reactive Inspection Section (RIS) — 12-6-85 Date

OTHER INSPECTOR(S): L. D. Vaughan, Program Coordination Section (PCS)

APPROVED BY: E. W. Merschoff, Chief, RIS — 12-13-85 Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: To ensure that valves, model 7150 and 7250, Class 1 and 2 are being supplied to Fort Calhoun in accordance with established commitments. In addition, to ensure that valves and valve spare parts are supplied by Dresser to the nuclear industry in accordance with established QA procedures, applicable codes (ASME Section III) and standards (Appendix B to 10 CFR Part 50.)

PLANT SITE APPLICABILITY: Fort Calhoun, 50-285; Perry Nuclear, 50-440, 50-44.; Davis-Besse, 50-346; Diablo Canyon, 50-275, 50-323.

REPORT NO.: 99900054/85-01	INSPECTION RESULTS:	PAGE 2 of 7

A. Inspection Issues

The inspection was conducted to determine whether models 7150 and 7250 Class 1 and 2 valves are being fabricated and supplied to Fort Calhoun in accordance with the established Dresser QA and QC commitments. Dresser is supplying these valves as part of the Fort Calhoun outage work package. During the inspection, observations were made on the fabrication of other valves and valve spare parts that are to be supplied to the nuclear industry. The survey consisted of ensuring that there is proper implementation of Dresser QA procedures, applicable codes (ASME Section III) and standards (Appendix B to 10 CFR Part 50.)

B. Inspection Findings

1. Violations

Contrary to section 21.21 of 10 CFR Part 21, evaluations of defects are not being performed adequately to determine if they merit reporting in accordance with 10 CFR 21. A review of Dresser's Part 21 procedure (#003.00 dated March 28, 1984) and Part 21 evaluation files for 1984 and 1985 concluded that as a result of an inadequate evaluation on file no. 84-01, Dresser failed to notify the NRC or its customers of a reportable Part 21 item (85-01-01).

2. Nonconformances

a. Contrary to Criterion VII of Appendix B to 10 CFR Part 50, and Dresser Industries QAM, Section 6 paragraph 6.1 and 6.2, Dresser procured services from two sources that were not on an approved vendors list (85-01-02).

b. Contrary to Criterion VII of Appendix B to 10 CFR Part 50, and Dresser Industries procedure QTI-13, paragraph 3.6, Dresser failed to perform receiving inspection on 4 containers of type 7018 nuclear welding rods (85-01-03).

c. Contrary to Criterion XII of Appendix B to 10 CFR Part 50, and Dresser Industries procedure QIT-33, paragraph 9.0b, calibration of the Charpy V-Notch Impact Testing Machine has not been performed during the last five years (85-01-04).

d. Contrary to Criterion XII of Appendix B to 10 CFR Part 50, and Dresser Industries procedure QTI-76, paragraph 2.1, the calibration certifications (#1343, #6141) for torque devices had no documentation of a standard serial number. Therefore, the calibration load cell and readout devices used as the standard for calibration cannot be traced as required (85-01-05).

e. Contrary to Criterion XII of Appendix B to 10 CFR Part 50, and Dresser Industries QAM, Section 12, paragraph 5.1, calibration of the WR-12 Carbon Determinator (used for analysis of carbon and sulfur content) was not traceable to national standards or equipment manufacturer's recommended standards (85-01-06).

f. Contrary to Criterion XII of Appendix B to 10 CFR Part 50, and Dresser Industries procedure QTI-13, paragraph 3.1, Dresser quality control failed to certify calibration completed on June 6, 1984 for the nuclear welding rod oven bi-metal thermometers TG-2, TG-4, TG-6, TG-8, TG-11, TG-20, TG-21 (85-01-07).

g. Contrary to 10 CFR 21, paragraph 21.21, and Dresser's Part 21 procedure, #003.00, paragraph 3.2, Dresser did not identify/list the pertinent data to substantiate an investigation for Part 21 file no. 85-01 (85-01-08).

C. Unresolved Items

None.

D. Other Findings and Comments

The NRC inspector reviewed the parts and travelers for Class 1 and 2 Dresser/Hancock 7150 and 7250 model valves to verify that heat number stamping, nondestructive testing, correct calibration of gages, subcontractor agreements were met, and appropriate sign offs were completed according to Dresser commitments.

The nonconformances found during the inspection did not involve the fabrication of valve models 7150 and 7250.

ORGANIZATION: DRESSER INDUSTRIES, INC.
ALEXANDRIA, LOUISIANA

REPORT NO.: 99900054/85-01	INSPECTION RESULTS:	PAGE 4 of 7

1. 10 CFR Part 21 Requirement

a. A review of Dresser's Part 21 files for 1984 and 1985 noted that Part 21 file no. 84-01 was not adequately evaluated in accordance with 10 CFR Part 21 Section 21.21. File no. 84-01 identified a problem with the failure of the disc collar on model 3707RA safety valves. The file stated this problem was not reportable since "The failure is an isolated case. Since 1971, Dresser QA system has improved to preclude this failure mode." However, a letter dated July 2, 1984 in the same file states this failure has happened twice since 1971 once in August 1981 at Toledo Edison/Davis-Besse and again at Diablo Canyon in May 1984.

Information in file no. 84-01 indicates that in May 1984, during blowdown testing by Wyle Laboratories, Huntsville, Alabama, on the main steam safety valves from Pacific Gas & Electric/Diablo Canyon 2 (Dresser valve type 3707RA), valve S/N BN 1741 lifted with simultaneous shearing of the disc collar/spindle threads and cotter pin. This occurred during the first actuation, at approximately 1065 psig. The disc and spindle deflected sidewise upon closing and the disc became wedged between the nozzle seat area and the bottom of the disc holder. The valve became mechanically jammed and could not open. It also leaked severely because the disc was not seated properly.

Dresser was asked about the blow-down problems that they are presently experiencing on the 3700 series safety relief valves undergoing testing at Wyle Labs in Huntsville, Alabama for Toledo Edison/Davis Besse. Specifically, Dresser was asked whether an investigation/evaluation was done or is planned to compare the problem experienced with the 3700 series safety relief valves to the problems documented in file no. 84-01. Dresser's response was "no evaluation has been performed." During a subsequent conference call on October 10, 1985, Dresser (Mr. J. Watz and Mr. B. Brunson) stated that "Dresser has performed an evaluation/comparison of file no. 84-01 and the valves being tested by Wyle Labs and determined the failure modes were similar. File no. 84-01 will remain closed, however, a new file, no. 85-04, will be initiated. This file (85-04) will reference file no. 84-01 and will be reported as a Part 21".

Violation 85-01-01 was identified during this part of the inspection.

b. A review was made of Dresser's Part 21 evaluation and reporting procedure "Evaluating and Reporting of Deviations and/or Noncompliances Affecting Safety Related to NRC Regulation 10 CFR Part 21" dated March 28, 1984 and Part 21 evaluation files for 1984 and 1985. This review noted that Dresser did not identify/list the pertinent data called for in their procedure to substantiate the investigation for Part 21 file no. 85-01.

This Part 21 file identified a problem with Dresser's model 3050 diaphragm valves sticking partially open or closed. The problem was reported to the NRC by the Cleveland Electric Illuminating Co., Perry Nuclear Power Plant. According to Dresser file no. 85-01, the problem was not reportable per 10 CFR 21 since Dresser had "no knowledge in power plant system design." However, Dresser's customers were notified. File 85-01 did not identify/list the information called for in paragraph 3.2 of procedure no. 003.001 to substantiate an investigation.

In addition, a review of the purchase order and specifications for valves ordered by the Perry Nuclear Power Plant was performed by the NRC inspector. This review identified documents which listed valves as "active" and "non-active", "safety related" and "non-safety related" and "ASME Class 1, Class 2 or Class 3" valves. With this information Dresser should have been able to determine the safety significance of the deviation.

Nonconformance 85-01-08 was identified during this part of the inspection.

2. Purchase Requirements

An examination was made of 20 purchase orders (P.O.) from Dresser for outside calibration services for gages, testing and measuring devices. Dresser was found to have purchased calibration and certification services for thermocouples TG-2, TG-4, TG-6, TG-8, TG-11, TG-20, TG-21, from two vendors that were not on Dresser's approved vendors list. Calibration of the thermocouples was performed by Honeywell-Houston on August 13, 1980 - Dresser order #26229-6; on October 19, 1982 - Dresser order #48879-6; and, on December 4, 1984 - Dresser order #47671-6. Likewise, Honeywell - Ft. Washington, PA performed calibration and certification of the thermocouples on February 25, 1982 - Dresser order #41201-6. These thermocouples were used to monitor the temperature of the heat treating furnaces.

Nonconformance 85-01-02 was identified during this part of the inspection.

3. Control of Special Processes

Heat treating and weld rod ovens were inspected and were found to be adequately calibrated and functioning according to Dresser procedures. The nuclear welding rod storage area was inspected. Weld rod containers were stored in either locked cabinets or locked weld rod ovens. The Section III weld rod ovens were marked "Nuclear." All rods were stored in the metal containers received and were marked with rod specification, lot and/or heat number to maintain traceability. However, four cases were found where containers of Type 7018, nuclear welding rods, lot 3c504Y0Z, heat #76175, were not stamped (verified) by the receiving inspection personnel. Receiving Inspection is essential in verifying that cans are properly sealed, and that identification numbers, material type, and quantity ordered are correct.

Nonconformance 85-01-03 was identified during this part of the inspection.

4. Manufacturing Process Control

The NRC Inspector reviewed the parts, travelers, and drawings for four shop orders and verified that specified requirements such as: (a) heat number stamping, (b) nondestructive testing, (c) authorized nuclear inspector sign off, (d) final inspection, (e) calibration of gages, (f) serial number assignment, and (g) heat treatment had been completed and correctly documented. No nonconformances were noted in this area.

5. Calibration

The NRC inspector found that torque devices calibration certifications #1343 and #6141 had no documentation of a calibration standard serial number. Therefore, there is no objective evidence that the torque devices were checked using a traceable load cell and readout device as the standard for measurement.

Nonconformance 85-01-05 was identified during this part of the inspection.

The NRC inspector found that calibrations of nuclear welding rod oven bi-metal thermometers (TG-2, TG-4, TG-6, TG-8, TG-11, TG-20, and TG-21) completed on June 7, 1984 were not certified by quality control. Accurate continuous monitoring of the nuclear weld rod ovens (between 250° and 300° F) is essential for low hydrogen electrodes. The temperature range of 250° to 300° F has to be maintained in order to reduce moisture or hydrogen absorption of the weld rods.

Nonconformance 85-01-07 was identified during this part of the inspection.

The NRC inspector reviewed the calibration and procedure requirements involved in the operation of the WR-12 Carbon Determinator. Dresser was not able to show that the calibration standards used for calibrating the carbon determinator are traceable to national standards or the equipment manufacturer's recommended standards. The WR-12 Carbon Determinator is used to determine material carbon and sulfur content. For applications where high levels of carbon and sulfur are detrimental to impact strength and low levels of carbon are desirable (i.e., some corrosive environments), accurate analysis of carbon and sulfur level is essential.

Nonconformance 85-01-06 was identified during this part of the inspection.

10 Plant Tour

A tour of Dresser's facility was performed by the inspectors. The activities observed were receiving, nuclear material storage area, nuclear welding rod storage area, heat treating furnaces, stock rooms, valve assembly clean room, calibration records area, and the metallurgical laboratory. The storage, laboratory and work areas were neat, clean and free of extraneous materials. Operations observed appeared to be well planned and progressing in an orderly fashion.

ORGANIZATION: ELGAR CORPORATION
SAN DIEGO, CALIFORNIA

REPORT NO.: 99900871/85-01	INSPECTION DATE(S): 09/17-19/85	INSPECTION ON-SITE HOURS: 44

CORRESPONDENCE ADDRESS: Elgar Corporation
ATTN: Mr. P. A. Zecos
Vice President and General Manager
9250 Brown Deer Road
San Diego, California 92121

ORGANIZATIONAL CONTACT: Mr. C. B. McVicker - QA Manager
TELEPHONE NUMBER: (619) 450-0085

PRINCIPAL PRODUCT: Uninterruptible Power Supplies

NUCLEAR INDUSTRY ACTIVITY: Approximately 2%

Current in house nuclear orders: Vogtle, Indian Point and WPPSS

ASSIGNED INSPECTOR: E. W. M[illegible] 12/27/85
For J. J. Petrosino, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): E. H. Yachimiak, RIS

APPROVED BY: E. W. M[illegible] 12/27/85
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: (1) Obtain information in regard to Elgar inverters originally sold to TVA (Hartsville) which were recently purchased by Ft. Calhoun Station; (2) Review an Elgar problem evaluation concerning a recent River Bend fuse block stud problem, which was identified (continued on page 2)

PLANT SITE APPLICABILITY: Elgar model #UPS-253-1 (fuse block stud problem): Beaver Valley #2 (50-412), Comanche Peak 1 & 2 (50-445/446), Millstone #3

ORGANIZATION: ELGAR CORPORATION
SAN DIEGO, CALIFORNIA

REPORT NO.: 99900871/85-01	INSPECTION RESULTS:	PAGE 2 of 5

B. SCOPE: (continued)
on an Elgar 25 KVA inverter; and (3) Evaluate the Elgar quality assurance program for adequacy and implementation of applicable requirements.

PLANT SITE APPLICABILITY: (continued)
(50-423), Nine Mile Point #2 (50-410), River Bend #1 (50-458), Seabrook #1 (50-443), South Texas #1 & 2 (50-498/499), Vogtle #1 & 2 (50-424/425) and WPPSS #3 (50-508)

A. INSPECTION ISSUES:

1) Obtain information to support an NRC inspection of refueling outage modifications at the FT. Calhoun nuclear station.

2) Review Elgar's evaluation of a fuse block stud problem which was reported to the NRC by the River Bend station as a 10 CFR Part 50.55(e) concern.

3) Evaluate the adequacy of Elgar's implementation of its Quality Assurance Program.

B. INSPECTION FINDINGS:

1) Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 2, "Quality Assurance Program", of ANSI N45.2-1977, no quality assurance manual requirements or records were in evidence to assure that all Elgar personnel performing activities affecting safety were indoctrinated as to the QA program requirements (85-01-01).

2) Contrary to Criteria III and V of Appendix B, to 10 CFR Part 50, a review of the circumstances surrounding a 25 KVA inverter fuse block stud deficiency revealed the following (85-01-02):

a) No documents were in evidence to assure that the design basis for current carrying conductors associated with the P-266D fuse holder were correctly translated into specifications, drawings, procedures, or instructions.

b) No instructions, procedures, or drawings were in evidence to assure that the interconnection assembly of a 400 ampere fuse holder (858-P266D), a 400 ampere shunt resistor (857-PR4), and a bus bar (943-390-20) were satisfactorily accomplished for Elgar static inverter model 253-1 filter panel assembly.

C. OTHER COMMENTS:

1) Discussions with Elgar Corporation (Elgar) personnel and a review of selected records concerning Elgar's 7.5 KVA (model #752-1) and 10 KVA (model #103-1), were performed. Areas that were discussed included; seismic testing, possible equipment requalifications, pre-operational testing, structural attachments, and storage. General Electric (GE) was TVA's agent for the original procurement of the inverters, which were purchased in 1980 for the Hartsville nuclear station.

Omaha Public Power District (OPPD) recently purchased the inverters from TVA for use in their uninterruptible power supply systems (UPS). The inverters will be installed at the Fort Calhoun nuclear station during the current refueling outage. The UPS system will contain two 7.5 KVA inverters and one 10 KVA inverter for each power train, for a total of six inverters.

Copies of Elgar and GE specifications, purchase orders, procedures, requirements, and other associated records were obtained from Elgar. These will be utilized by an NRC inspection team at Fort Calhoun station for a selective verification to assure the equipment conforms to all applicable requirements and is traceable back to Elgar.

2) The NRC was notified of a deficiency located inside class 1E uninterruptible power supplies furnished by Elgar and installed at the River Bend nuclear station. The deficiency involved a loose stud on a 450-600 Amp rated fuse block located on a "filter panel assembly", inside a 25KVA Elgar static inverter, model number OPS 253-1, which is part of the class 1E UPS system. The loose stud was identified on the side of the fuse block which was mechanically connected to a bus bar, which subsequently was connected to a shunt type resistor. This involved a straight line connection of the components, with the bus bar in between the resistor and the fuse block.

During discussions with assembly line personnel and observations of how the components were actually assembled, it was revealed that no fabrication drawings or instructions had been generated for that activity.

Subsequent discussions with engineering and quality assurance personnel revealed that engineering had not performed any design analysis to allow for the nontypical utilization of the fuse block terminal stud as a current carrying conductor.

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Normal industry practice is to have direct physical contact between the bus bar and the fuse, whereas Elgar's configuration utilizes the stud and stud hardware as a current carrying conductor.

Discussions with the fuse block manufacturer determined that the fuse block stud is a copper alloy which is tin plated, while the nuts and washers could be stainless steel or plated steel alloys. The manufacturer also indicated that the bus bar and fuse should make direct mechanical contact to reduce high heat conditions created by using the stud hardware as a current carrying conductor (see page 1 for affected plants).

The safety significance of this problem is that during a loss of offsite power, an inverter failure could result in the loss of a 120 volt ac class 1E power supply for plant control and instrumentation.

Elgar Corporation is currently evaluating all the circumstances surrounding this problem pursuant to 10 CFR Part 21. However, it was noted by the NRC inspector that Elgar's evaluation of the deficiency did not take into consideration any root cause areas other than licensee induced stud damage. Currently, Elgar is evaluating the lack of design documents and lack of in-process manufacturing controls as a potential root cause of the reported deficiency (Nonconformance 85-01-02).

3) A brief quality assurance (QA) program implementation review was conducted. Areas that were specifically reviewed included; measuring and test equipment, and training and indoctrination.

A sample of approximately twelve electrical crimping tools were examined for unique identification, calibration control, records of calibration, and traceability back to the National Bureau of Standards. All aspects of this area were satisfactory.

The QA manual adequately addressed the 18 criteria of Appendix B to 10 CFR 50 and ANSI N45.2. Within the individual sections, the area of indoctrination and training of personnel who perform activities affecting safety was reviewed. It was noted that an effective program of training and indoctrination was documented and appeared to be adequately implemented for the QA/QC and manufacturing personnel. However, no program was in place which would indoctrinate other personnel who performed activities affecting quality. Areas for which the program did not address indoctrination were engineering, design, procurement, customer services, and all management positions other than QA (Nonconformance 85-01-01).

4) Plant Tour:
A plant tour was conducted which included all manufacturing aspects of Elgar's facility at San Diego. Material receipt inspection, material control, in-process quality control functions, and wave soldering processes, were some of the areas which were observed.

No deficiencies were noted during this part of the inspection.

D. PERSONS CONTACTED:

Name	Title	Company	
Susan Pritzl	Program Administrator	Elgar	**
Ed Noble	Supervisor QA Test/Insp.	Elgar	
Clyde B. McVicker	QA Manager	Elgar	***
Mike Murray	QA Engineer	Elgar	***
Gilbert Cota	QA Inspector	Elgar	
Vernon Lawson	Supervisor - Magnetics	Elgar	
Clydine Ford	Supervisor - Boards	Elgar	
Sue Zorich	QC Inspector - Boards	Elgar	
Steve Sedio	Engineering Manager	Elgar	
Josie Smith	Training Specialist	Elgar	
Phyllis Kelly	Supervisor - Production	Elgar	
Debbie Nason	Human Resources	Elgar	
Fred Welch	Sr. QA Technician	Elgar	
P. A. Zecos	Vice President/Gen. Manager	Elgar	**
Tom Erickson	Vice President Human Res.	Elgar	**
George Seibert	Electrical Engineering Dept.	Stone & Webster (1)	
Thomas Crouse	QA Manager - River Bend Station	Gulf States Utl. (1)	
Robert Stafford	Director - Quality Services	Gulf States Utl. (1)	
Al Wilkinson	Manager Applications Engineer	Gould Inc. (1)	

**Exit meeting only
***Entrance and exit meeting
(1) telephone contact only

ORGANIZATION: GENERAL ELECTRIC COMPANY
NUCLEAR ENERGY BUSINESS OPERATIONS
SAN JOSE, CALIFORNIA

REPORT NO.: 99900403/85-01	INSPECTION DATE(S): 3/4-6/85	INSPECTION ON-SITE HOURS: 25

CORRESPONDENCE ADDRESS: General Electric Company
Nuclear Energy Business Operations
ATTN: Mr. W. H. Bruggeman, Vice President and General Manager
175 Curtner Avenue
San Jose, California 45125

ORGANIZATIONAL CONTACT: Mr. J. J. Fox, Senior Program Manager
TELEPHONE NUMBER: (408) 925-6195

PRINCIPAL PRODUCT: Nuclear Steam System, Services and Fuel.

NUCLEAR INDUSTRY ACTIVITY: General Electric Company (GE) Nuclear Energy Business Operations (NEBO), has a work force of approximately 4500 assigned to domestic power plant activity.

ASSIGNED INSPECTOR: RP Correa for R L Pettis Jr — 12/30/85
R. L. Pettis, Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): P. Sears, SPIS
W. Shier, BNL
W. Banister, EG&G

APPROVED BY: RP Correa for J W Craig — 12/30/85
John W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: GE Topical Report No. NEDO-11209-04A and 10 CFR 21.

B. SCOPE: The inspection was conducted to review and obtain copies of selected GE Service Information Letters (SILs); review the status of previous inspection findings and Potentially Reportable Condition (PRC) files; and review various reactive items.

PLANT SITE APPLICABILITY: Limerick 1 (50-352); Fermi (50-341); Hope Creek (50-354); LaSalle 1 and 2 (50-373/374); Monticello (50-263); Oyster Creek (50-219); Perry 1 and 2 (50-440/441); Shoreham (50-322); Vogtle 1 and 2 (50-424/425).

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Closed) Nonconformance (84-02): Contrary to Engineering Operating Procedure (EOP) 42-10.00, Section 4.2.d.4, concerning Design Record Files (DRFs), the DRFs that supported the verification of computer calculations for SAFER02 computer code (DRFs No. A00-01249, A00-1320, and E00-137) did not identify the reviewer and date when performed.

   The DRFs supporting the SAFER02 verification calculations have been independently reviewed. In addition, two actions have been taken by GE to prevent recurrence of this type of nonconformance: The Manager, Core and Fuel Technology has issued a letter to all engineers responsible for Engineering Computer Programs reiterating the DRF requirements for verification of calculations; in addition, a Quality Assurance Newsletter (dated August 1984) has been issued to all engineers and managers that includes a "DRF Closeout Checklist" with reminders about signing and dating DRF entries.

2. (Closed) Nonconformance (84-02): Contrary to EOP 42-1.00, Section 3.3.2, regarding design control, no documentation was available for the analyses described in GE Topical Report HEDE 23785-1-P, Vol. 11, and NEDE 24984. These topical reports were submitted to the Office of Nuclear Reactor Regulation for review.

   The two topical reports referenced above describe the analytical basis for two safety-related computer codes (SAFER02), NEDE 23785-1-P and (ODYN04) NEDE 24984. A review of the extensive verification programs for these computer codes has indicated that sufficient testing and comparison of code calculations with other analytical and experimental results was performed to preclude the need for additional DRFs.

3. <u>(Closed) Nonconformance (84-04)</u>: Contrary to GE Quality Assurance Topical Report NEDO-11209, Rev. 4, Section 3.12, "Design Change Control," Engineering Operating Procedures (EOP) 40-3.00 "Engineering Computer Programs" (ECPs), does not require that Control Components (responsible engineers for ECPs) define other design documents affected by computer code changes or errors, or coordinate these changes with other responsible engineers whose documents are affected. Further, Section 4.1 of the same procedure (EOP 40-3.00) does not require that the Control Component interface with responsible engineers affected by a computer code error, and assess effects of computer code errors on designs, past and present.

   EOP 40-3.00 has been revised (Change Notice A, December 19, 1984) to require that responsible engineers for ECPs document all errors in approved Level 2, 2R and 3 computer programs. In addition, these errors will be classified according to their potential impact on previous analyses. This documentation is then reported to all User Design and Development Component Managers for evaluation. These managers also acknowledge receipt to the responsible engineer.

   The inspector reviewed an example of the implementation of the procedure. This included the ECP error description and potential impact evaluation, the distribution to the component managers, and the acknowledgement of receipt returned to the ECP responsible engineer.

4. <u>(Closed) Nonconformance (84-04)</u>: Contrary to EOP 40-3.00, "Engineering Computer Programs," the Design Record File (DRF) for the CRNC-04 computer code (No. A00-01619) did not include all of the code testing specified in the Software System Specification.

   The Software System Specification for CRNC-04 has been revised to indicate that the code verification testing will include a comparison of results with results from previous versions of the code. This is considered sufficient since the current version of CRNC-04 does not include any significant analytical model modifications or editions.

5. <u>(Closed) Nonconformance (84-04)</u>: Contrary to EOP 42-6.00, "Independent Design Verification," the verification of calculations described in GE Topical Report NEDE 25518 was not completed until after issuance of the report.

No action regarding the design record file for NEDE-25518 was required. However, as part of the action taken to prevent recurrence of this type of nonconformance, a memorandum was written emphasizing the requirement for completion of the independent review and verification prior to issuance of reports.

6. (Open) Nonconformance (84-04): Contrary to EOP 42-10.00, "Design Records Files," the DRF for the PANACEA Core Design System (No. 670-0005) did not always identify the originator, reviewer, or date performed.

   GE will review DRF entries to assure that originators, reviewers and dates of entries are adequately identified. A record of this review will be incorporated into DRF 670-0005. In addition, the Manager, Core and Fuel Technology will issue a letter to all engineers responsible for ECPs reminding them of their responsibility under the referenced EOP. These requirements will also be emphasized in QA training course documents related to DRFs. This item will be reviewed during a future inspection.

7. (Open) Nonconformance (84-04): Contrary to Section 3.10 of the QA topical report NEDO-21109-04A, application of the SAP4G07 code was not fully verified in the following areas:

   a. Two options of the beam element (fixed end forces and shear deformation analysis) and one option of the pipe element (the ASME code analysis) had no verification provided.

   b. One nodal point option (slaved degrees of freedom) and one option of the beam element (released degrees of freedom) had verification for the latest version only. However, an earlier version of the SAP4G07 code (which is a Level 3 program), is still available for use on safety-related designs.

   GE stated, in their May 22, 1985 written response to the NRC, that the SAP4G07 code is a fully verified Level 2 computer program, meaning it satisfies GE's design review process requirements for independent verification with results comparable to those from either experimental data or their alternate solution techniques, as per EOP-40-3.00. The response also stated this documented design review fully conforms to the verification requirements of Regulatory Guide 1.64 and NEBO commitments outlined in NEDO-11209. This item will be reviewed during a future inspection.

8. (Open) Nonconformance (84-04):

Contrary to EOP 42-6.00, the method from which analytical results were obtained in the SAP4G07 computer program verification problems 4.1, 4.2, 5.1, 8.1, and 14 was not referenced, nor were any hand calculations included.

As stated in GE's written response to the NRC regarding Nonconformance 7 above, the SAP4G07 computer code has been fully and independently verified by a NEBO design review team and judged to be adequate for its intended purpose. This item will be reviewed during the next inspection.

9. (Closed) Nonconformance (84-04): Contrary to EOP 40-3.00, "Engineering Computer Programs," users had been reporting potential computer code errors verbally to the responsible engineer without the required documentation. GE personnel stated that no potential computer code errors had been discovered since the previous NRC inspection and that proper procedures will be followed in the event of future code errors.

10. (Closed) Unresolved Item (84-04): This item concerned errors that were discovered in the RVRIZ02 computer code that is used by GE in containment system and piping design calculations. The computer code has also been distributed to utilities for their own use. During this inspection it was determined that:

    a. The various utilities who obtained the computer code have been notified of the error and advised of potential consequences;

    b. A survey of GE users of RVRIZ02 determined that no additional safety-related code applications had been performed;

    c. The computer code has been removed from the approved Level 2 status (i.e., not approved for safety-related analyses).

    As a result of the corrective action taken by GE, this item is considered closed.

ORGANIZATION: GENERAL ELECTRIC COMPANY
NUCLEAR ENERGY BUSINESS OPERATIONS
SAN JOSE, CALIFORNIA

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E. OTHER FINDINGS OR COMMENTS:

1. Sodium Pentaborate Curve Error

A potential deficiency in the Standby Liquid Control System (SLCS) at the Fermi 2 plant was the subject of a GE Field Deviation Disposition Request (FDDR) on January 9, 1985. This deficiency was related to an error that was discovered in the sodium pentaborate concentration data supplied in SLCS system specifications for FERMI. This item was reviewed with the GE cognizant engineer, for the SLCS, who determined that the error was within the margin included in the system design and did not jeopardize the plant's ability to achieve safe shutdown.

2. Control Rod Drive Filters

Movable inner filters for the control rod drive mechanisms (CRDMs) at the Monticello Nuclear Plant were supplied by GE as spare parts with incorrect mesh size, 2 mil instead of 10 mil. This situation occurred once in 1974 and twice in 1984. The new 10 mil replacement design is "stationary" in contrast to the former which was "movable," i.e., moved along with the index tube during a reactor scram.

The old design may have resulted in screen clogging; the new 10 mil design allows for the passage of larger particles, thus reducing the possibility of clogging. GE stated that this new filter is easily recognized by the fact its screen material appears on the outside of the filter casing rather than the inside as in the case of the earlier 2 mil design.

GE personnel stated that this improved design was a response to slow scram times experienced at Oyster Creek in late 1971. This was accomplished by the issuance of Product Service Information Letter 71-21, dated December 29, 1971, advising customers to convert to the new 10 mil filter design. In reviewing another incident involving excessive scram times, at Monticello in December, 1984, the major cause was attributed to the paper purge dam material (DESOLVO) and corrosion/cleaning byproducts in the system, and not the incorrect size filters furnished by GE.

Oyster Creek, the only other plant using 2 mil filters, is presently converting its remaining 12 CRDs to the 10 mil filters. At present, all 2 and 10 mil filters are in storage at GE and have been quarantined pending disposition instructions.

GE's Potentially Reportable File (PRC 84-62) classified the CRD filters as a non-safety-related component which would not impact scram time performance even if totally plugged.

3. Defective Circuit Breakers at Vogtle

Approximately 239 defective GE circuit breakers (models AKR30 and AKR50) were identified at Vogtle. The defective breakers were originally identified to GE by the A/E, Bechtel, on February 6, 1984. The breakers were in the Plainville, Connecticut, warehouse scheduled to be shipped to the Hope Creek Generating Station. They had apparently been reworked per GE Service Advices 175-9.6, 175-9.7, and 175-9.11 by factory personnel who had not been previously involved in rework programs and consequently was accomplished without adequate retesting and reinspection. Per letter by GE-Contractor Equipment Business Operation (CEBO) dated February 23, 1984, all subsequent rework and generic reinspection were accomplished. GE-CEBO notified NRC on February 24, 1984, of a potential safety hazard. Since GE-CEBO is a subcontractor to the A/E, GE-NEBO did not have responsibility for this problem. This item will be reviewed with GE on a future inspection of the Plainville facility.

4. Neutron Monitor Power Supply Failures on Limerick 1

As a result of a -20 Vdc power supply failure for the Intermediate Range Neutron Monitors (IRM) during which the reactor failed to trip, GE-NEBO was requested to investigate. GE concluded that the system worked as designed and met all system specifications. However, Philadelphia Electric Company requested a change in the hardware design to provide information to the operator if the -20 Vdc supply should fail. This new design was documented in Field Deviation Disposition Request (FDDR) HH1-4460, Rev. 0, dated December 6, 1984.

5. Dikkers Safety Relief Valve Equipment Qualification Test Failure

A Potentially Reportable Condition file (PRC 83-12) was initiated by GE on April 19, 1983, concerning a test failure of actuator solenoids which initiate valve operation for the Automatic Depressurizer System. This item was subsequently determined not reportable to the NRC pursuant to 10 CFR Part 21. This conclusion was based on the performance capability of the equipment in accordance with the original GE specifications and requirements in effect prior to the TMI event. Subsequently, the failure of the actuator solenoids was reported to the NRC by GE's customers, Niagara Mohawk Power Corporation and Cleveland Electric Illuminating Company, June 1, 1983 (Region I) and July 29, 1983 (Region III), respectively. These interim reports were issued pursuant to 10 CFR 50.55(e).

A separate file, PRC 84-44, on the same basic problem was reviewed for its contents concerning the Dikker Safety Relief Valves. There was no cross-reference between the two PRC files despite their almost identical problem. PRC 84-44 was opened on June 26, 1984, and reopened on November 11, 1984, and again determined not reportable under 10 CFR Part 21, but was found to be a condition germane to safety. The NRC was notified of this conclusion on November 28, 1984.

6. Perry Feedwater System Pipe Rupture Analysis

Gilbert/Commonwealth (G/C), the A/E for Perry, made a 10 CFR 21 report which indicated design forces and other data originally calculated from G/C's Feedwater System (FW) pipe rupture analysis may possess unconservative assumptions, in light of a recent reanalysis performed by the NSSS supplier, GE, at the request of Cleveland Electric & Illuminating Company (CEI).

GE's reanalysis postulated a pipe break per NUREG-800 and calculated jet impingement loadings for selected target locations, including the effects of fluid thermodynamics and state, pipe friction, and the reactor vessel's contribution to such jet forces. The erroneous assumptions made by G/C considered only the FW pump side of the rupture, while ignoring the contribution of the reactor vessel to the total force. This assumption produced jet forces and shape profiles which underestimated the total pipe rupture effect.

GE's final report, DRF B21-00306 dated November 14, 1984, indicated a total jet force 25% larger than that calculated by the G/C analysis. In addition, the jet shape resulting from such a postulated rupture yielded an entirely different configuration than the G/C analysis, which may result in the failure of protection devices for essential equipment. A further review by G/C indicated the nonconservate assumptions existed only for the FW system.

CEI's final report to the NRC, dated February 14, 1985, indicated that the incident was reportable to the NRC pursuant to 10 CFR 50.55(e). Their evaluation of the safety implications revealed increased jet forces affected four target locations thus requiring additional or modified equipment shielding. However, only two of the four targets have safe shutdown functions since they affect the control rod drive tubes at the bioshield wall interface. This report further states that following a design basis pipe rupture, on the reactor side of the FW pump, a loss of control rod insertion capability coupled with an inoperable Standby Liquid Control System, may impair the ability to achieve safe shutdown.

Corrective action is to include modification of all four target locations, which for Unit 1 have already been performed. Modifications for Unit 2 will be completed consistent with its construction schedule.

7. GE Supplied Steam Leak Detection System

An integrated electrical test was conducted at Shoreham in which offsite power was cut-off to initiate the test. When the diesel generators picked up the load, it was discovered that both the Reactor Water Cleanup (RWCU) and High Pressure Coolant Injection (HPCI) systems had been isolated by temperature instruments in the Steam Leak Detection (SLD) system furnished by GE. The cause of the isolation was attributed to incorrect settings of the time-delay relays contained inside the circuitry of the Riley, Model 86, Temperature Switch Modules, which measure ambient and/or differential temperatures in the Emergency Core Cooling (ECC) systems equipment areas.

When power is first applied to these relays, they receive a temperature trip signal before achieving a steady-state condition. This same condition was previously discovered at Limerick and several other BWRs. Corrective actions taken involved either a GE modification to the internal wiring of the module or a relay replacement in the isolation circuits for systems containing Agastat time-delay relays.

The NRC inspector reviewed GE's evaluation of this incident at Shoreham, PRC File 84-47, which concluded that the condition was not reportable to the NRC, pursuant to 10 CFR Part 21, since the plant could be brought to a safe shutdown without the availability of the HPCI or RCIC systems, therefore, no significant safety hazard would exist. GE based their conclusion on a five plant FSAR review which indicated the availability of at least two ECCS pumps following any single failure in addition to isolation of the HPCI system. In addition, GE's review stated it could be shown that with one pump available, sufficient make-up flow would exist to provide adequate core cooling.

On January 15, 1985, GE issued Service Information Letter (SIL) No. 416, "Riley Temperature Switches" to all BWR/4 operating plants. GE's recommendation was for owners of BWR/4 plants and LaSalle (BWR/5) to review their temperature switch designs based on information contained in the notice. FSAR licensing calculations may have to be updated to reflect the modified set of available systems, should affected plants refrain from modifying SLD circuitry.

8. Ground Break Relay Deficiency in Class 1E Units at Hope Creek

Eight Model TGSR-12 ground break relays manufactured by GE were found to have a defective component. The relays were supplied by one of GE's non-nuclear operations under a subcontract to the A/E, Bechtel. GE-NEBO has not received GE Service Advice Letter 175-9.2, which describes this deficiency to affected non-nuclear customers, therefore, they assume it is not in the NSSS area of responsibility. Bechtel has replaced the defective relays with acceptable units provided by GE and documented the deficiencies and corrective action.

No nonconformances or violations were identified during this part of the inspection.

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E. PERSONS CONTACTED:

*M. Blich
*B. Smith
N. Barclay
*J. Fox
*J. Case
*G. Stramback
R. Hill
*E. Giambalvo
R. Waldman
B. Simon
E. Chu
*F. Hopkins
*J. Wood
D. Saxena
R. Valencia
R. Gridley
H. Hwang
J. Atwell
N. Barker
A. Amiri
R. Bloomstrand
T. Herczeg
R. Siemer
C. Canham

*Attended exit meeting.

F. DOCUMENTS EXAMINED:

1. Procedure, dated December 19, 1984, Engineering Computer Programs Change Notice A for EOP 40-3.00.

2. Internal memo, dated November 9, 1984, J. Fox to distribution, memo concerning Engineering Computer Program (ECP) Error Control.

3. Report, document no. DRF-B21-00306, Rev. 0, dated November 13, 1984, Feedwater Line Postulated Break Analysis, Perry Unit 1.

4. Report, document no. PRC 84-47, Shoreham Leak Detection System.

5. Report, document no. PRC 84-62, CRD Inner Filter (Monticello).

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6. Letter, dated February 23, 1984, from GE Contractor Equipment Business Operations (GE-CEBO), Phillip Piqueir, to Claude Turnbow, Bechtel Power Corp., Hancock, NJ.

7. Letter, document no. B015, dated February 24, 1984, from GE-CEBO, David Dixon, Manager QA, to NRC, Richard C. DeYoung.

8. Specification, document no. FDDR HH1-4460, Rev. 0, dated December 6, 1984, SURNMS Elementary Diagram.

9. Letter, dated November 21, 1984, Licensee Event Report - Failed 20 Volt SRM/IRM Supply Preventing RPS Actuation.

10. File, document no. PRC 83-12, dated March 6, 1985, GE file on Dikkers Safety Relief Valve (SRV).

11. Letter, Revision 0, dated April 19, 1983, from J. Jacobsen to G. G. Sherwood, Potentially Reportable Condition of the Electro - Pneumatic Actuation Assembly on the Dikkers Main Steam Safety Valve (SRV) to perform its Class 1E Function under NUREG-0588, Category 1 Qualification Requirements.

12. Letter, Revision 0, dated May 13, 1983, from G. G. Sherwood to J. Jacobsen - PAC 83-12, Electro-Pneumatic Assembly on the Dikkers Safety/Relief Valve (SRV).

13. Letter, Revision 0, dated February 24, 1984, from G. G. Sherwood to J. Jacobsen - same subject.

14. File, document no. PRC 84-44, dated March 6, 1985, Dikkers Solenoid Valve Failure During EQ Testing.

15. Letter, document no. HE 84-3., dated June 26, 1984, from H. Ehson to G. Sherwood, Potentially Reportable Condition Failure of the Dikkers Safety Relief Valve Solenoid during the NUREG-0588 Category I EQ Testing.

16. Internal memo, dated November 26, 1984, PRC 84-44, Dikkers Solenoid Valve Failure During Environmental Qualification Testing.

ORGANIZATION: GESELLSCHAFT FUR NUKLEAR SERVICE
ESSEN, WEST GERMANY

REPORT NO.: 99901025/85-01	INSPECTION DATE(S): 9/16-27/85	INSPECTION ON-SITE HOURS: 80

CORRESPONDENCE ADDRESS: Gesellschaft fur Nuklear Service MBM
ATTN: Dr. Klaus Janberg
Director
Goethestr. 88
4300 Essen, West Germany

ORGANIZATIONAL CONTACT: Mr. Reinhard Bittner, QA Manager
TELEPHONE NUMBER: (0201) 7220-160

PRINCIPAL PRODUCT:

NUCLEAR INDUSTRY ACTIVITY: Waste conditioning, packaging, engineering, cask development and transport.

ASSIGNED INSPECTOR: J. T. Conway 12.24.85
J. T. Conway, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): S. K. Iskander, NRC Resident Engineer

APPROVED BY: RP Correa for 12/27/85
John W. Craig, Chief, Special Projects Inspection Section, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: The purpose of this inspection was to conduct a programmatic evaluation of the implementation of Gesellschaft fur Nuclear Service's (GNS) QA program pertaining to the fabrication of the CASTOR V dry spent fuel storage/transport casks for Virginia Electric & Power Company (VEPC); review the QA records for the CASTOR No. 2 cask; and witness the manufacturing and testing activities at four GNS subcontractor's

PLANT SITE APPLICABILITY: Surry Unit Nos. 1 and 2 (50-280/281).

ORGANIZATION: GESELLSCHAFT FUR NUKLEAR SERVICE
ESSEN, WEST GERMANY

SCOPE: (continued)

facilities. The subcontractors were Gontermann-Peipers (GP) and Siempelkamp (SK) which casts the cask body, Boschgotthardshutte (BGH) which forges the primary and secondary lids and trunnions, and Kraftwerk Union (KWU) which performs final machining and assembly.

A. Nonconformances:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 3.2 of GNS Procedure No. PV 97, and Sections 4.3.1, 4.3.2 and 4.3.4 of GNS Quality Assurance Handbook (QAH), a review of nine purchase orders (PO) to subcontractors for items and services pertaining to the CASTOR V casks for VEPC did not specify that the manufacturing activity should be conducted under an approved QA Program or that the requirements of 10 CFR Part 21 applied (85-01-01).

   The nine subcontractors were GP and SK (cast body), KWU (final machining & assembly), Butting Metallwerk (fuel basket), BGH (forged lids and trunnions), Schmidt & Clemens (fasteners), Cefilac (metallic seals), Pennekamp Huesker (neutron moderator material), and Von-Roll (nickel plating).

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Sections 17.3, 18.2.1, 18.2.2, and 18.3.1 of the QAH, a review of six external audit reports and QA records pertaining to personnel qualifications revealed the following (85-01-02):

   a. Audits of three subcontractors were overdue in that during the interval in which material/services were being controlled, the most recent audits of Butting Metallwerk (BW) and KWU were in July 1984 and SK in September 11, 1984.

   b. Checklists were not used in any of the six audit reports reviewed.

   c. Qualifications of the auditing personnel were not documented and retained as QA records.

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 5.1.1 of the QAH, a review of GNS Procedure No. 91 "Hydrostatic Test Procedure for Dry Spent Fuel Storage Cask" dated October 23, 1984, and GNS Procedure No. PV 32-1 "Helium Leak Test Procedure for Dry Spent Fuel Storage Cask" dated June 28, 1984

revealed that neither procedure referenced the requirement for the sequence of tightening bolts, the tightening torque and subsequent documentation on a test report (85-01-03).

4. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 9.3.1 of the QAH, and SNT-TC-1A, a review of records for four NDT personnel (Nos. 2, 6, 7, and 9) revealed the following (85-01-04):

   a) GNS did not have a written practice or procedure for the various disciplines of NDT.

   b) Eye examinations and current copies of written examinations were not maintained on file for the four examiners.

   c) Certifications for Level II ultrasonic testing (examiner No. 7) and for Level II leak testing (examiner No. 9) were missing.

   d) Statements indicating satisfactory completion of training were missing for the four examiners.

5. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 4.5 of GNS Procedure No. 32-1, and Sections 5.5 and 5.7 of GNS Procedure No. 91, witnessing of helium leak testing and hydrostatic testing and a review of calibration records at GNS and KWU revealed the following (85-01-05):

   a) Pressure gauge No. 6 which was used for the leak test did not have a calibration sticker, and there was no evidence of records to show that the gauge was calibrated.

   b) Pressure gauge S/N 31 which was used for the hydrostatic test was last calibrated in October 1984 by KWU. When this anomaly was identified during the inspection, the gauge was calibrated by KWU on September 20, 1985, but calibration records for the dead weight tester used to calibrate the gauge were noted to be missing after April 1982.

B. Other Findings or Comments:

1. Gesellschaft fur Nuclear Service (GNS)

GNS is the affiliated company of STEAG Kernenergie (SKE), VEBA Kraftwerke Ruhr (VKR) and Deutsche Gesellschaft fur Wiederaufarbeitung von Kernbrennstoffen (DWK). SKE is engaged in all stages of the nuclear fuel cycle and performs design and project work in the planning and construction of nuclear facilities. VKR is the owner and operator of several coal-fired power stations. DWK was founded by twelve German nuclear-based utilities for the construction and operation of installations of the nuclear fuel cycle. GNS is involved in the fields of mobile waste conditioning, associated engineering and services, and cask development and transportation.

Personnel Contacted

Dr. K. Janberg, Director
Dr. M. Baatz, Director
R. Bittner, QA Manager
D. Methling, Engineering Manager
A. Bonifacio, GNSI Project Manager

a. 10 CFR Part 21

The requirements of 10 CFR Part 21 were imposed upon GNS by VEPC for the design and manufacture of the CASTOR V dry spent fuel storage casks. The NRC inspector noted that GNS had written Procedure No. PV 97 "Procedure for Notifying US NRC or Customer of Defects and/or Noncompliances per 10 CFR Part 21" dated June 22, 1984, to satisfy this requirement. A Notice, written in both German and English, described the regulation and identified the QA Manager as the individual to whom reports may be made. This Notice was posted in the GNS engineering department and was also sent to seven GNS subcontractors. GNS failed to send a copy of this Notice to two subcontractors, Von-Roll (VR) which nickel plates the cask body and Pennekamp Huester (PH) which supplies the neutron moderator material, for subsequent posting at their facilities. Although GNS had sent this Notice to seven of nine subcontractors, the 10 CFR Part 21 requirements were not documented on specific POs sent to each of these subcontractors (See 85-01-01).

b. Nondestructive Examination (NDE)

The NRC inspector reviewed nine NDE procedures pertaining to ultrasonic testing (UT) and liquid penetrant testing (PT) of the cask body and forged lids, trunnions, and fasteners and leak testing (LT) of the assembled cask. It was noted that the procedure for LT did not require a sequence for tightening bolts nor did it specify the tightening torque values (See 85-01-03).

All of the NDE procedures required personnel performing the tests to be qualified per SNT-TC-1A. In a June 13, 1984 letter to the GNS QA Manager, Deutsche Gesellschaft fur Zerstorungsfreie Prufung (DGZP), the German Society for Nondestructive Testing, stated that the "Recommendation for the Qualification and Certification of NDT personnel" dated December 1983 was equivalent to SNT-TC-1A of the American Society for Nondestructive Testing.

The certification records for four GNS examiners were reviewed to verify that all personnel performing NDE were qualified. With the exception of missing certificates for examiner No. 7 (UT-Level II) and examiner No. 9 (LT-Level II), documents from DGZP stated that the QA Manager was certified to Level III for PT, UT, and magnetic particle (MT); examiner No. 7 to Level III for radiographic, PT, and MT; examiner No. 6 to Level III for PT; and examiner No. 9 to Level II for PT, UT, and MT.

It was noted that written examinations for Level II disciplines are given in a specific region by DGZP, and the Level III examinations are given in Berlin. Copies of the written examinations or evidence of successful completion of the examination as well as eye examinations for all four examiners were missing. GNS had failed to generate a procedure addressing the training, examination and certification of NDE personnel. Further GNS failed to designate the individual who would function as the company's Level III. Nonconformance 85-01-04 was identified in this area of the inspection.

c. Testing

The CASTOR V No. 2 cask successfully passed a leak test and a hydrostatic test on September 17 and 18, 1985, respectively. The leak test was performed in two stages to check the seal between the cask body and the primary and secondary lids. Helium leak test Procedure No. 32-1 dated June 28, 1984 was used. This procedure was reviewed and was found to be in compliance with Article 10 (T-1060) of Section V of the ASME Code. The test was witnessed by a representative from Bundesanstatt fur Materialsprofung (BAM) and a representative from VEPC. GNS examiner No. 9 conducted the test, but a copy of this individual's certification to LT-Level II was missing from the files (See 85-01-04).

The hydrostatic test, which is a VEPC requirement, was performed in accordance with Procedure No. 91 dated October 23, 1984. The test was in compliance with Part UG-99 of Section VIII of the ASME Code. A representative from VEPC witnessed the test.

The torque wrench used to tighten the bolts on the primary and secondary lids was stamped with MPA 2743. Calibration records indicated that the wrench was calibrated on April 23, 1985 by Versuchs-und Prufanstalt fur Werkzenge Remscheid (VPWR). However, certification No. 138122 743 from VPWR did not identify the reference standard used and that calibration was traceable to Physikalische Techmische Braunschweige (PTB). PTB is the German counterpart to the US National Bureau of Standards. Neither of the two procedures used in the tests identified a requirement for the tightening torque values or the sequence of tightening bolts (See 85-01-03). In addition, pressure gage No. 6 which was used on the leak test, and pressure gage S/N 31, which was used on the hydro test, were not calibrated prior to testing (See 85-01-05).

d. Control of Purchased Material & Services

GNS utilized nine subcontractors/vendors for components with a safety function, and machining and plating services related to the manufacture of the CASTOR V casks for VEPC. The nine subcontractors were GP and SK (cask body), BGH (forged lids and trunnions), Schmidt & Clemens (bolts), Cefilac (metal seals), PH (neutron moderator material), BM (fuel basket), KWU (machining and final assembly), and VR (nickel plating).

e. Purchase Orders

POs to the nine subcontractors were reviewed to assure that technical and quality requirements imposed upon GNS by VEPC were properly implemented and passed on to the material manufacturers and service vendors. The POs specified: the scope; listed the operations to be performed; referenced the technical specifications (e.g., Material Data Sheets and NDE procedures); identified the documents to be submitted both prior to start of fabrication (e.g., fabrication and QC, welding, cleaning and packaging plans) and after completion (e.g., material certifications, test reports, and manufacturer's acceptance and test report certificates) to GNS; specified GNS's right of access for onsite surveillance/inspection and review of QA records; and established the delivery schedule. Although all the POs were initialed by a representative of the GNS QA department, it was noted that GNS failed to pass on the requirements of 10 CFR Part 21 and did not require the fabrication of the item or performance of the service to be in accordance with a specific QA program that had been reviewed and approved by GNS (85-01-01).

f. QA Manuals

GNS had copies of QA Manuals for six vendors. With the exception of KWU's manual, the manuals/handbooks for GP, SK, BGH, Schmidt and Clemens (SC) and BM all contained a document from Technischer Uberwachungs-Vereif (TUV) stating that the QA program meets the requirements of "AD-Merkblatt WO/TRD100" which is a document similar to the ASME Code. There was no documented evidence that GNS had reviewed and approved each manual/handbook for use in the manufacture of the CASTOR V casks for VEPC.

The manuals included GP's "QA Handbook" dated February 2, 1981; SK's "QA Handbook" dated June 30, 1983; BGH's "QA Handbook" dated December 22, 1980; SC's "QA Handbook" dated March 1983; and BM's "QA Manual" dated January 23, 1984. KWU's manual was supplemented with a document titled "QA Requirements for Fabrication of CASTOR(s)" dated January 20, 1984. The document addressed six areas: Quality Planning, Document Control, Receipt Inspection, In-process Inspection, Documentation, and Identification of Departments Responsible for Quality. GNS did not have a copy of the manuals for PH and VR. Further, GNS is currently waiting for Cefilac's (CL) manual to be translated from French into German and English.

g. Vendor Audits

Six external audit reports whose summaries had been transcribed into English were reviewed. All the audits were performed by the GNS QA Manager. BM and KWU were audited in July 1984, SK in September 1984, GP in November 1984, and BGH in January 1985. There was no date on the audit report for SC. The 1985 annual audits of BM, KWU, and SK were overdue. It was noted that the audits were performed without the use of a checklist, and the qualifications of the auditor were not documented and retained as a QA record (See 85-01-02).

h. Documentation Packages (DP)

The QA records for the CASTOR no. 2 cask for VEPC were reviewed, and DPs for the eight major vendors consisted of:

° Parts List - identifies the individual components by material specification, dimension and DIN No. (if applicable) and references the drawing No. This document is prepared by GNS-Engineering.

° Material Data Sheet (MDS) - lists the chemical, mechanical and testing requirements and the required documents (e.g. DIN No. which establishes who performs, witnesses, and signs off the test). GNS-Engineering prepares the MPS, and it is approved by GNS-QA and BAM.

° Fabrication Control Plan (FCP) - lists specific fabrication and control steps (e.g., cleaning, identification marking, inspection and testing). The FCP is prepared by the vendor but is reviewed and approved by both GNS and BAM prior to the start of fabrication. In-process inspection is performed in accordance with the requirements contained in the individual FCPs. Test results are verified against procedures, drawings, codes, and standards; and the results are recorded. The following activities are typical of witness hold points to be inspected, witnessed, and verified by the noted organizations:

UT - Cask body by GP, GNS, and BAM.
Primary and secondary lids by BGH, GNS, and TUV.
Bolts by SC, GNS, and TUV.

PT - Trunnion housing and sealing surfaces on cask body, trunniors, primary and secondary lids, and bolts by KWU, GNS, and TUV.

LT - Completed cask (primary and secondary lids) by GNS and BAM.

- Supplemental Documents - includes certified material test reports (CMTR) and Certificate of Conformances (CC) from the subtier vendors; identification of transfer marking for material; welding, cleaning, and packaging plans; material lists showing item traceability to heat number; and dimensional inspection and NDE reports. With the exception of the CMTRs and CCs, the remaining documents are generated by the subcontractor and are approved by a QA representative from GNS.

2. Castor V Cask - Fabrication

A finished cask goes through the following manufacturing steps: The nodular iron body is cast by GP or SK. Following a visual examination and UT of the rough casting, a dimensional check and layout is performed. The outer surface and cavity are machined, and 100% UT is conducted. The body is then shipped to KWU.

KWU machines the trunnion housing, drills and taps the holes. Holes are also drilled and tapped in the bottom and lid area followed by the boring of holes in the outer perimeter to accommodate the neutron absorber rods. Samples from the extracted rods are used for mechanical property testing. This is followed by an intermediate assembly and dimensional checkout.

The body is then shipped to VR which nickel plates the internal diameter of the cask body and the housing for the trunnions. The plated body is then shipped back to KWU.

The trunnions, and primary and secondary lids are forged by BGH. The individual products are heat treated and rough machined. The trunnions and lids are sent to KWU. The bolts are forged by SC followed by machining to semi-finished bolts which are shipped to KWU. The lids, trunnions, and bolts are final machined at KWU.

At KWU, the internal diameters of the nickel plated cask body is final machined, and the sealing surfaces of the cask lid area are finished and then PT examinations are performed.

Moderator material, a fuel basket, and metallic seals are fabricated by PM, BM, and CL respectively; and the furnished items are shipped to KWU for final assembly. At KWU, the unit is assembled, and a helium leak test and hydrostatic test are conducted. Painting and trunnion installation are the final steps prior to shipment to the customer.

The fuel basket is fabricated by BM in Wittingen-Knesbeck, West Germany from hot-rolled sheets out of Radionox A 18 (XCr Ni 1913). The GNS material specification No. BS 05 included the following German DIN and GNS standards: 50 049 - Certificates of Material Testing, 17 440 - Stainless Steel, 50 145 - Tensile Test, 50 125 - Test Coupons (Type E), 1543 - Dimensional Check, and GNS MDS No. WB 15/1. The MDS was signed off by GNS inspector No. 2 and BAM inspector No. 1.52. The Welding Plan was approved by GNS No. 9, and GNS No. 7 approved the Cleaning and Packaging Plans. It was noted from the FCP that activities such as analysis of boron in each sheet and material stamping were witnessed by both GNS-QA and TUV. Identification marking transfer, CMTRs from KRUPP, Creusot Coire and Thyssen, and a dimensional inspection report were stamped by GNS No. 8 and a representative from VEPC.

The neutron moderator rods are fabricated by PM in Vreden, West Germany in accordance with a Work Guide Line (designated AV) which was approved by GNS prior to the start of fabrication. The AV addressed areas such as purchasing of raw material, material identification during manufacturing, inspection and certification. GNS MDS No. WB 23 specified the raw material ("LUPOLEN 5261 Z") which was supplied by BASF. The manufacturing steps included forming sheets under pressure and temperature from raw material in granulated form, cutting the sheets into rods, and machining into circular rods. The FPC indicated that GNS No. 7 verified material identification and a dimensional check. Material certifications (3) from PH indicated that the rods meet the requirements of "3.1B-Abnahme."

Nickel plating of the internal diameter of the cask body and the housing for the trunnions is performed by VR in Klus, Switzerland in accordance with Procedure HV-3 "Galvanic Nickel Plating" which is approved by GNS-QA. Cleaning, electrolyte composition check, and etching are performed per internal VR procedures

(AV-V.R1/R2/R31/R32) which have been approved by GNS and BAM. Internal checks such as nickel thickness testing, hardness testing, and final visual examination are in accordance with Procedure No. HV-3.

The high-tensile bolts are fabricated by SC in Lindlar, West Germany from forged rods using Material No. 1.4313 (X5 Cr Ni 134) as identified in GNS material specification No. BS 07. Time/temperature charts for the heat treatment, CMTRs and PT and UT reports were submitted by SC. Tensile, charpy impact, and hardness testing were performed per DIN 50 145, 50 125, 50 115, and 50 351. Visual examination and dimensional verification were in accordance with GNS Procedure PV 23-1. PT and UT, which were witnessed by GNS-QA and TUV, were in accordance with GNS Procedures PV 23-2 and PV 13, respectively.

Rods for trunnions and bolts and disks for primary and secondary lids are forged by BGH from Material No. 1.4313 identified in GNS specification BS 06. For the three finished products, BGH submits a preliminary plan for heat treatment and specimen location for tensile and charpy impact tests for subsequent approval by GNS-QA and TUV. Tensile and impact tests are performed in accordance with DIN 50 145 and 50 115, respectively. Time/temperature charts and heat treatment certifications, CMTRs and UT reports were submitted by BGH. TUV Inspection Certificates reference dimensions and mechanical properties. Visual examination and dimensional verification are performed in accordance with GNS procedures PV.22-1 (trunnions), PV 21-1 (lids), and PV 23-1 (bolts). Ultrasonic testing was witnessed by GNS No. 7 and TUV No. 4, and UT was performed in accordance with PV 12 (trunnions), PV 11 (lids), and PV 13 (bolts).

The nodular iron body is cast by either of two foundries: SK or GP. The CASTOR No. 2 body was cast by GP. A review of the FCP showed that UT of the machined casting was witnessed by GNS No. 6 and BAM No. 6.21, and a dimensional check and PT were signed off by GP-QA and GNS No. 7.

The UT report dated March 27, 1985 indicated the test was performed by a Level II and a Level III examiner, and the results were accepted by a Level III examiner from VEPC. Two certifications from GP were also reviewed. One was for chemical analysis and the other for mechanical properties. Mechanical testing of core samples removed by KWU was performed by BAM.

Final machining of components, testing, and final assembly is performed at KWU. Two FPCs, one prior to nickel plating and one after nickel plating were reviewed by the NRC inspector. Other documents reviewed included an acceptance test Certificate for the bored holes for the neutron moderator rods which was signed off by QA personnel from KWU and GNS; a dimensional inspection report for trunnion seats which was signed off by personnel from KWU, GNS, and TUV; a PT report for trunnion seats which was witnessed by TUV No. 4 and a certification to verify moderator length which was signed off by KWU and GNS-QA.

Additional documents reviewed included a nickel plating acceptance test certificate and a dimensional check after nickel plating both of which were signed off by KWU-QA and GNS No. 7. In addition, a surface roughness check of the nickel plated surfaces and a PT report of the sealed surfaces following machining was signed off by KWU-QA, GNS No. 7 and TUV No. 4.

For the primary and secondary lids, two FPCs and the following documents were reviewed: material identification certifications; dimensional inspection reports signed off by KWU-QA, GNS, and a individual from VEPC, and a PT report witnessed by GNS No. 7 and TUV No. 4.

3. Subcontractors

The facilities of four subcontractors were visited to assure that each manufacturer had adequate QC commitments and was implementing an effective QA program. During the plant visits, three of the manufacturers were not engaged in work related to a CASTOR V cask.

GP-Siegen, West Germany

Personnel Contacted

Dr. K. Schroeder, QA Manager
H. Emami, QA Supervisor - Foundry
H. Mehlau, NDE Examiner

GP produces spin, continuous, and heavy mold castings. The melting shops consist of four electric arc and two induction furnaces. Weights, times, and temperatures are monitored using modern measuring and testing methods to guarantee uniform quality. The production process, from receipt of incoming raw materials to final inspection, is subject to stringent quality controls. Inprocess QC consists of a variety of hardness testing techniques and special UT methods. Final inspection includes checks for dimensional accuracy and surface finish by roughness testing, MT and PT. Chemical composition of the raw materials and finished products are controlled, and mechanical properties are verified with the aid of static and dynamic testing equipment.

The time/temperature chart for the heat treatment of the CASTOR No. 2 cask was reviewed. It was noted that the Ni-Cr-Ni thermocouples attached to the cask body and located in the furnace ceiling were calibrated with a Pt-Rh-Pt thermocouple (transfer standard). There was no documented evidence that the standard was ever calibrated, and the NRC inspector was told that the Pt-Rh-Pt thermocouples were purchased from a sole source supplier and that CMTRs were not required. A review of calibration certificates revealed that the mechanical testing equipment (e.g. tensile, hardness, charpy impact) was calibrated by Staatliches Materialprufungsanct (MPA) on an annual basis.

The NDE certifications from the DGZP for the Level II and Level III examiners who performed UT of the CASTOR No. 2 cask were reviewed and found acceptable.

SK - Krefeld, West Germany

Personnel Contacted

M. Rosenau, QA Manager
H. Muting, Foundry Manager

SK casts products with unit weights up to 150 tons from gray and nodular cast iron and alloyed grades. An independent QC department is integrated into the manufacturing process, and each stage of the process is overseen by the QC department. Testing for mechanical properties, dimensional checks, surface inspections, and UT are performed by the QC department. Optical and electron microcopy are used in the laboratory, and a mass spectrometer is used for chemical analysis of both raw material and finished products.

The plant tour consisted of the pattern shop, core production and sand molding areas, melt shop (four induction heated furnaces), batch mixing area, and metallurgical and testing laboratories. A review of calibration certificates for mechanical testing equipment (tensile, hardness, and charpy impact) indicated that the equipment was calibrated by MPA on an annual basis. However, there was no evidence that the reference standards were traceable to PTB.

KWU - Muehlein-Ruhr, West Germany

Personnel Contacted

K. Muller, QA Manager
H. Kaufer, Electro Technician
F. Schlegel, TUV Representative

KWU is responsible for the final machining of several components (i.e., cask body, primary and secondary lids, trunnions, and bolts) as well as testing and final assembly. The NRC inspector witnessed a successful leak and hydrostatic test on the CASTOR No. 2 cask (See B.1.c). The UT calibration block of nodular cast iron containing three flat bottom holes in three orthogonal directions was examined.

A visit to the metrology laboratories at two different sites revealed that Siemens, the parent company of KWU, calibrates the reference standards used by KWU to calibrate measuring and test equipment. Both laboratories had posted a "Level B Test Laboratory Certificate" from Siemens which indicated that each lab meets all the requirements in controlling measuring and test equipment according to a Siemens manufacturing guideline. A review of individual certifications from Siemens did not indicate that the standards were traceable to PTB. It was noted that dead weight tester S/N 207-909/910 was used in October 1984 to calibrate pressure gage No. 31 which was used on the hydrostatic test. Test certificate No. C3/1 from Siemens indicated that the dead weight tester was calibrated in April 1982 and was due for calibration in April 1984. There was no documentation to show that the dead weight tester was calibrated after April 1982 (See B.1.c).

Another CASTOR cask body (designated GNS No. 02) from SK was on blocks, and two core samples for mechanical testing were being machined from the holes which will accommodate the neutron moderator rods. All mechanical testing of the cask body is performed by BAM. A decision will be made whether or not to use the cask body in a finished CASTOR cask for drop tests or for another CASTOR cask for VEPC.

BGH - Siegen-Weidenan, West Germany

Personnel Contacted

Dr. Biener, QA Manager
J. Dilgert, QA Engineer
H. Giesler, Supervisor - NDE
E. Ohrndorf, Marketing Manager

BGH forges the primary and secondary lids, the trunnions and stock for the bolt fasteners. For all four products, the process consists of melting the steel and pouring it into ingots at the steel works. At the forging plant, the ingot is warmed-up, pre-compressed in a 2000 ton press, and transferred via a roller gear bed to the forging machine. After forging, the product is heat treated in an automated annealing, hardening and tempering plant. The product then enters a hydraulic straightening press before being sand blasted. Following a hardness check and an in-process UT examination, the final production steps include cutting and machining. In the case of CASTOR V products, QC consisted of 100% UT examination. Mechanical testing (i.e., tensile and charpy impact) is witnessed by a TUV representative. It was noted that material identification is by impression marking of the heat number and material identification (e.g., W 310 alloy) on all products. Traceability is maintained from the ingot, to the forged bar, through intermediate products to the final product which is shipped to the customer.

There are seven gas fired furnaces each with five monitored zones. Ni-Cr-Ni thermocouples are used in one furnace and Pt-Rh-Pt thermocouples are used in the other six furnaces. The thermocouples are calibrated every six months against a Pt-Rh-Pt standard (two of which are in-house) which in turn is calibrated against Pt-Rh-Pt standard No. 657 which is calibrated by Eichamt Hannover No. 2 (EM-2) which is a department/division of PTB. A review of log books for each furnace verified that the thermocouples were calibrated at the required frequency, and certificate No. 65//85 from EH-2 indicated that reference standard No. 657 was calibrated on July 18, 1985. The printer charts in the control room are calibrated twice a year. A remote controlled panel is used to check the temperature in the five furnace zones against the actual print-out in the control room. The thermocouples are spaced equidistant apart in the furnace ceiling.

The inspector reviewed a calibration certificate dated 1980 from the PTB for the measuring instrument "Symetra 2D" used for calibrating the thermocouples. The instrument is calibrated every two years and the most recent certificate from TUV-Rheinhand was dated September 24, 1982. TUV had calibrated the instrument per a contract with the PTB as noted in the certificate. No other documentation could be produced which suggests that the instrument is approximately one year overdue for calibration.

The testing laboratory was toured, and a review of QA records indicated that the hardness testers (2), tensile testers (2) and charpy impact testers were all calibrated by the MPA. A review of records for the testing performed on the items for CASTOR V indicated that the actual testing was witnessed by TUV. The UT units are serviced and calibrated by Kraut Kramer once a year. Certifications for four NDE examiners - one Level III and three Level II were in compliance with the certifying agency - DGZP.

ORGANIZATION: ILLINOIS FABRICATORS, INC.
BRADLEY, ILLINOIS

REPORT NO.: 99901036/85-01	INSPECTION DATE(S): 11/6-8/85	INSPECTION ON-SITE HOURS: 40

CORRESPONDENCE ADDRESS: Illinois Fabricators, Inc.
ATTN: Mr. R. A. Hawker, President
265 South Kinzie
Bradley, Illinois 60915

ORGANIZATIONAL CONTACT: Mr. Charles R. Hawker, Vice President
TELEPHONE NUMBER: (815) 939-3551

PRINCIPAL PRODUCT: Custom fabricated steel, specializing in battery trays and instrument control panels.

NUCLEAR INDUSTRY ACTIVITY: None since November, 1980

ASSIGNED INSPECTOR: [signature] 17 Jan 86
E. H. Trottier, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): Jeffrey B. Jacobson, RIS

APPROVED BY: [signature] 24 Jan 86
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was performed to evaluate the allegation received by the Region III office of the Nuclear Regulatory Commission on September 25, 1985 concerning fabrication of two safety-related containment ventilation control panels without benefit of a QA program.

PLANT SITE APPLICABILITY: Clinton Power Station (50-461).

REPORT NO.: 99901036/85-01	INSPECTION RESULTS:	PAGE 2 of 5

A. Inspection Issues

The issue that resulted in this inspection was the allegation received by Region III of the Nuclear Regulatory Commission on September 25, 1985. The allegation related to the quality assurance controls in effect at Illinois Fabricators, Inc. where two containment ventilation control panels were fabricated. The panels were fabricated for MCC Powers who designed, engineered and installed the completed control panels at Illinois Power Company's Clinton Power Station near Clinton, Illinois. This inspection sought to establish whether an appropriate quality assurance program was in place at the Illinois Fabricators, Inc., the company that fabricated the panels.

B. Inspection Findings

Although Illinois Fabricators did not have a quality assurance program in place that would meet the requirements of 10 CFR 50 Appendix B, the safety related work was performed under the control of MCC Power's quality assurance program and, as such, was properly controlled.

Specifically, the inspector verified that Illinois Fabricators received and accepted all quality-related requirements referenced by and attached to the purchase order for fabrication of the panels as follows:

1. Packaging

Equipment and material were required to be shipped in containers "... in keeping with good commercial practices to prevent damage during shipment and storage at buyer's [MCC Powers, Inc.] warehouse."

The inspector reviewed the Receiving Inspection Report for each panel (IPL43JA and B). The overall appearance of each panel was inspected by a member of the MCC Powers QA staff and accepted on 12/2/80. In addition to overall appearance, the Receiving Inspection Report verified conformance to four other criteria. These criteria were:

a. Fabrication in accordance with approved drawings.

b. No evidence of distortion from cutting/punching operations.

c. No evidence of significant paint defects.

d. No missing hardware, gaskets or accessories.

The inspection accepted each criterion separately and noted acceptance by individual check marks in appropriate boxes. The Receipt Inspection Report was signed by a member of the MCC Powers QA staff.

2. Source Inspection

The panels were "... subject to source inspection at the supplier's [Illinois Fabricators] plant by MCC Powers Quality Assurance representative, ... one week before fabrication ...." began.

The inspector reviewed the Vendor Surveillance Report (source inspection) performed by the Manager of Quality Assurance for MCC Powers on November 11, 1980 at the Illinois Fabricators facility in Bradley, Illinois. The summary section of the report cites the MCC Powers purchase order and job numbers, and clearly identifies the ultimate destination of the panels as "the Clinton Nuclear Power Station, ..." The summary section concludes with the following statement: "It is the opinion of MCC Powers that Illinois Fabricators is an established, viable, and competent manufacturer fully capable of supplying MCC Powers with an acceptable product line with appropriate design support and documentation evidence."

The final section of the source inspection report is titled "Objective Evidence." In it, the Manager of QA for MCC Powers summarized his plant tour, reviewed panel base material purchase orders (including Certificates of Conformance), and welder qualification records.

3. Certifications and Test Reports

Copies of applicable documents required by MCC Powers under this item were to be "... signed by a responsible member of the supplier's [Illinois Fabricators] Quality Assurance function." and forwarded to MCC Powers.

The inspector reviewed the "QW-484 Manufacturer's Record of Welder or Welding Operation Qualification Tests" for both Illinois Fabricators welders designated to undergo qualification. The inspector was advised by Illinois Fabricators that although the lead welder had been selected and did perform the welding, a backup or alternate welder was also qualified. The inspector's review of each welder's QW-484 form verified that all information was entered and the form signed by the Vice President of Illinois Fabricators.

The inspector reviewed both the "QW-482 Welding Procedure Specification (WPS)" documented on ASME Form E-6, and the "QW-483 Procedure Qualification Record (PQR)" for the welding procedure used on the panels. Each form was completed as required, with the physical tests associated with the PQR performed by Pittsburgh Testing Laboratory and signed by their manager of the Chicago District.

4. Certification of Conformance

Illinois Fabricators was required to "... certify that all materials and furnished items supplied under this order were inspected and/or tested and conform with all requirements of the published specifications and requirements of this [purchase] order."

The inspector reviewed the statement of certification by Illinois Fabricators, Inc. that "... all materials and finished items supplied under P. O. 377-10870 conform with specifications received with the purchase order." The certification was signed by the President of Illinois Fabricators, Inc.

In addition to verifying Illinois Fabricators' conformance to the requirements established by MCC Powers, the inspector reviewed the Contract Specification governing HVAC controls. The Contract Specification was written "... by Baldwin Associates on March 22, 1978 by and between MCC Powers, a unit of Mark Controls Corp., ... and Baldwin Associates, ... for HVAC Controls, complete for Clinton Station."

The inspector found no weld-related requirements in Article 112, Codes and Standards, Division 1 - General Requirements. It was noted, however, that the subarticle dealing with control panels found in Division 3 - Technical Requirements, states, "Each panel shall be 3/16 inch steel plate, all welded-construction and built and reinforced in such a manner that no deflection will occur due to the weight of instruments, other apparatus, etc."

A review of the material referenced in the Welder Qualification Tests (QW-484) and the Welding Procedure Specification (QW-482) referenced in item 3 above, verified the thickness range as "1/16 - 3/8 [inch]," with the actual test plate being "5/16 [inch] thick." The base metal specified in QW-482 is "Commercial Quality Mild Steel," and "carbon steel plate" in QW-483. Thus, all available records indicate that Illinois Fabricators, Inc. fully satisfied the material and performance requirements specified by MCC Powers, Inc.

ORGANIZATION: ILLINOIS FABRICATORS, INC.
BRADLEY, ILLINOIS

REPORT NO.: 99901036/85-01	INSPECTION RESULTS:	PAGE 5 of 5

C. Other Observations and Comments

The inspector noted that Illinois Fabricators, Inc. established a comprehensive Quality Assurance Program in support of expected participation in the construction of the now inactive Marble Hill Nuclear Generating Station. Revision 0 to the QA manual that was to have been used is dated 6-13-83 and contains 19 sections that appropriately address the eighteen criteria of Appendix B to 10 CFR 50.

The inspector examined purchase orders and other documents for items that would be required to support the fabrication and supply of safety-related components for the Marble Hill Station. These documents included calibration certificates for measuring and testing equipment, and welder certifications. No deficiencies were noted.

ORGANIZATION: JOHNSTON PUMP COMPANY
CHATTANOOGA, TENNESSEE

REPORT NO.: 99901023/85-01	INSPECTION DATE(S): 7/15-19/85	INSPECTION ON-SITE HOURS: 64

CORRESPONDENCE ADDRESS: Johnston Pump Company
Nuclear Service Division
ATTN: Mr. Raymond L. Clark
2601 East 34th Street
Chattanooga, Tennessee

ORGANIZATIONAL CONTACT: C. Tommy Craig
TELEPHONE NUMBER: (615)629-1415

PRINCIPAL PRODUCT: Sales and Repair Services for Centrifugal Pumps

NUCLEAR INDUSTRY ACTIVITY: Approximately eighty percent of total billing is related to sales and repair of centrifugal pumps used in domestic nuclear power plants.

ASSIGNED INSPECTOR: [signature] 3/20/85
E. H. Trottier, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): Thomas F. Burns, Technical Specialist
Brookhaven National Laboratory

APPROVED BY: [signature] 12/27/85
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was performed to review the conduct of repair and fabrication of nuclear related centrifugal pumps and to review the administrative mechanism used to comply with the reporting requirements of 10 CFR Part 21.

PLANT SITE APPLICABILITY: Edwin I. Hatch 1/2, 50-321;366

ORGANIZATION: JOHNSTON PUMP COMPANY
CHATTANOOGA, TENNESSEE

REPORT NO.: 99901023/85-01	INSPECTION RESULTS:	PAGE 2 of 9

A. Inspection Issues

The issue that resulted in this inspection was Johnston Pump Company's emergence as a major participant in after-market refurbishing of pumps used at nuclear power stations.

B. Inspection Findings

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Quality Assurance Procedure JCP-VE-21, objective evidence of the required eye test for one non-destructive examination technician could not be produced to support qualification of a Level II Liquid Penetrant Examiner in 1983.

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50, ASME Code Section IX and Quality Assurance Manual Section 2-G, a Johnston Pump Company welder was improperly qualified.

3. Contrary to ASME Code Section III, Division I, weld repair of an 18-inch pump bowl was performed by a Johnston Pump Company vendor using weld filler metal of indeterminate chemical and physical properties.

4. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Quality Assurance Manual Section 2-C, all attendees of seven training sessions held in 1984 and 1985 did not initial course attendance sheets to signify participation in training.

5. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Quality Assurance Manual Section 16-A, the Corrective Action log did not contain sections or columns that facilitated the classification of nonconformities by type, responsibility and corrective action taken.

6. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 2-A of the Quality Assurance Manual, the semi-annual review of nonconformances and corrective actions that should have occurred in early 1985 was not conducted until June 27, 1985.

7. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 15-A of the Quality Assurance Manual, many nonconformances that were dispositioned "repair" contained no technical justification to substantiate the adequacy of such repairs. Instead of the required technical or engineering justification, nine of the 11 the NCRs examined described the repair activity needed to disposition the NCR.

C. Other Findings and Comments

1. Part 21 Program

While touring Johnston Pump Company shop and warehouse areas, the inspector verified that both 10 CFR Part 21 and Section 206 of the Energy Reorganization Act were prominently posted. In addition, these documents were found posted in the office area.

The inspector reviewed Johnston Pump Company's Part 21 reporting procedure, which is procedure number JCP-10 CFR 21. It was found to contain adequate direction regarding evaluation of deviations and notification requirements.

2. Plant Tour

A tour was conducted of the Johnston Pump Co. facility to observe and evaluate the operations and practices conducted. At the time of the inspection, the level of commercial nuclear activity was very low, with only one order presently being filled (for several pump shafts). The activities observed were shaft straightening and various machining operations. The facility was found to be clean and orderly, with appropriate segregation and identification of raw materials to be used for nuclear applications.

3. Qualification of Nondestructive Examination Personnel

This activity is governed by Quality Assurance Manual Section 2-F, and the respective nondestructive examination procedures. Section 2-F of the manual directs that qualification steps shall be detailed in a "Written Practice" that shall meet the requirements of SNT-TC-1A-1980 and the ASME Code. The written practice is identified as QAP JCP-7, Rev. 0, Qualification and Certification of Nondestructive Examination Personnel. This document delineates the qualification requirements for Level I, II and III personnel for the following test methods:

a. Liquid Penetrant Examination
b. Magnetic Particle Examination
c. Visual Examination

Johnston Pump Co. has qualified three inspectors as Level II liquid penetrant by education, experience, training and examination. An examination of the qualification files for these three individuals revealed that the vision test and results could not be located for one inspector for the year 1983. All other examination and test requirements were found to be in compliance with the written practice.

4. Welder Qualification

The qualification of Johnston Pump Co. welders is governed by QAM Section 2-G, which invokes the "applicable ASME Code Section". The applicable ASME Code Section is IX, Welding and Brazing Qualifications. Compliance with this Code Section is mandatory to fulfill the requirements of ASME Code III.

Johnston Pump Co. had only one qualified welder in 1983 (who was eventually terminated due to declining business). A review of this welder's qualification tests revealed the following descrepancy that is contrary to the requirements of ASME Section IX, paragraph QW 410.16:

Johnston Pump Co. had considered this welder as being qualified to deposit weld filler metal without a restriction on progression direction based on his having performed a qualification test using only an "upward" progression for vertical welding. The restriction imposed by ASME Section IX, paragraph QW 410.16 is that a welder may not change the progression direction of weld metal deposited from that used during the qualification test. Since this welder only used an "upward" progression during the qualification test, he was not qualified to weld in the "downward" direction. Johnston Pump Co. should have made note of this restriction in the welder's qualification record.

5. Weld Filler Metal Certification

The requirements that govern weld filler metal testing requirements for use on ASME III, Division 1, Class 3 components are found in Subarticle ND 2420. The testing detailed in this section of the Code is specified to be performed on each lot or each heat of the weld filler metal to be used. This testing would necessarily have to be performed on a test coupon of the actual weld filler metal to be used in the component fabrication, rework or repair.

An 18 inch cast steel pump bowl in the segregated material storage area was randomly selected to verify material control and traceability. During this activity it was discovered that this bowl had been weld repaired in three locations by the casting vendor. A thorough review of the documentation supplied revealed that the casting vendor had submitted a "typical" certificate of conformance for the weld filler metal used for the repair. To comply with the requirements of the Johnston Pump Co. purchase order and the above referenced Code Section, a certified material test report on the actual weld filler metal used is required. This bowl was purchased from Fisher Cast Steel Products of West Jefferson, Ohio, on Johnston Pump Co. Purchase Order TE 4175.

The pertinent pump bowl data is:

a.	Item Identification	Replacement Bowl Assembly for RHR Service Water Pumps
b.	End use	Georgia Power Co., Plant Hatch
c.	Code/Date	ASME III, 1971 (no addenda)
d.	JPC P.O. (Date)	TE 4175 (5/14/84)
e.	Part No.	N54087-B
f.	Heat No.	85B-072
g.	S/N	85B-0601
h.	Job No.	SNRV 6504/05
i.	Material	SA 351 Grade CF-3M
j.	Repair Date	3/12/85
k.	Weld Filler Metal	E316L-16

The welding procedures used by the casting vendor were reviewed and approved by Johnston Pump Co. prior to the repair. A review by the inspector confirmed that these procedures were qualified in accordance with the requirements of ASME Code Section IX. Also, the repair areas were nondestructively examined (liquid penetrant) and found to be free of rejectable indications. No other findings were noted in this area.

6. Repair Activities

The Chattanooga facility has been designated as the only Johnston Pump Co. plant that will perform the fabrication and repair of ASME Code Section III, Class 3 pumps. The restriction of this responsibility to one facility has greatly enhanced the control of all activities that affect the quality of these components. Although Johnston Pump Co. proposes to perform major repair and/or rework on pumps manufactured and "N-stamped" by other pump manufacturers, it has not yet done so. Since gaining their N and NPT stamps (August 1983), the only work they have performed on pumps supplied by others has been to replace non-pressure retaining parts. The replacement of pressure retaining parts has been limited to pumps of their own manufacture. The exception to the above was the replacement of the pump bowl assemblies for the RHR Service Water Pumps for Iowa Electric Light and Power Company (Duane Arnold Energy Center). However, these pumps were not N-stamped components. Consequently, the original manufacturer and the Code authorities had no concern over the activity. Also, the replacement of the bowl assemblies was a total replacement, whereby everything below the column flange was replaced. In essence, the pump was now a Johnston pump. Extensive discussion with engineering and quality assurance staff members revealed that their intent in the repair/replacement market is to basically sell a "new" pump, designed and rebuilt (to ASME III) by Johnston Pump Co. Original design and operational parameters will be used, unless a change is warranted by failure analysis or "state-of-the-art" improvements. Company officials stated that they did not contemplate entering the Class I and II pump business at this time.

7. Design Control

Design control is governed by QA Manual Section 3-A. Verification of compliance to the requirements of this section was accomplished by examination of those activities undertaken in fulfilling a typical nuclear order. The purchase order selected was Iowa Electric Light and Power Company Purchase

Order 009689. This P.O. was issued on 10/24/83 for two, Model 16 GMC 6 stage pump bowl assemblies for the RHR Service Water Pumps at the Duane Arnold Energy Center. (Seismic requalification also was included in this P.O.) The specification invoked for this item was 7884-MRS-M010-4, which is a modification of the original Bechtel specification used to procure the original pumps installed in the plant. The applicable design and fabrication code for these replacement pumps was ASME III, Division I, Class 3, except that the pumps were not to be "N" stamped. Drawings 70060-CN and 70061-CN were reviewed and found to be in compliance with QAM Section 5-A, Rev. 0, Instructions, Procedures, and Drawings, and Section 3-A, Rev. 3, Paragraph 3.7, Preparation of Drawings. The required design report was prepared by Johnston Pump Co. and was verified by an independent registered professional engineer. The registered professional engineer previously had been evaluated for appropriate qualifications by Johnston Pump Co. The seismic stress analysis also was performed by the same professional engineer. The inspector reviewed the hydrostatic test report and found it to be complete with the identification of pressure, time, test gauges, calibration results and test results. There were no findings in this area of the inspection.

8. Control of Measuring and Test Equipment

QA Manual Section 12-A and QA Procedure JCP-12 represent the administrative controls that govern measuring and test equipment at Johnston Pump Co. The inspector randomly selected the following tools and gauges to verify conformance with the requirements found in the above-referenced documents:

Tool/Gauge	Type	I.D.	Cal. Date
Starrett	2"-3" mic.	MK-03	05/20/85
B&S	578-1 vern.	VK-01	05/20/85
MTI	dial ind.	ID-04	06/17/85
Mitutoyo	6"-7" mic.	MJ-07	07/08/85
-	6" dial col.	VH-02	07/08/85
Mitutoyo	4" std.	EJ-04	07/08/85
Mitutoyo	1" std.	EJ-01	07/08/85
-	Ring Gauge	GJ-03	02/20/85
Starrett	23"-24" mic.	MJ-38	03/11/85
-	Level	LJ-01	03/11/85
MTI	Dial Ind.	IJ-09	05/20/85
MTI	Dial Ind.	IJ-11	05/20/85

The inspector noted that dial indicator IJ-09 and IJ-11 were repaired at the time of the calibration. Johnston Pump Co. personnel produced a copy of the repair record from the calibration service organization that detailed the repairs made, parts replaced and subsequent calibration. Also, the repair record was reviewed upon return of the gauge to determine if gauge accuracy had been outside the specified tolerance and a nonconformance report required. In this instance, the repairs were cleaning, oiling and crystal replacement, and were not the result of inaccuracy. Thus, a nonconformance report and evaluation of prior work was not justified.

The tools and gauges tabulated above were also examined for marking, calibration tagging, cleanliness, correct function, control and suitability of storage facilities. All tools were found to be clean, marked and tagged in accordance with the QA Manual, and in good working order and stored in a controlled area.

The records governing the initial survey and subsequent audits of the calibration services vendor were examined and found to be in compliance with the QA Manual. The calibration vendor has been audited annually as required.

There were no findings in the area of calibration and control of measuring and test equipment.

9. Weld Procedure Specifications

Weld procedure specifications (WPS) were developed and qualified by Johnston Pump Co. in accordance with the requirements of ASME Code Section IX. (Qualification in accordance with Section IX is required by the QA Manual.) The following weld procedures and procedure qualification records (PQR) were reviewed:

WPS	Material	PQR	Date
JCP-SM-1, Rev. 0	P1 to P1	SM-1	06/07/83
JCP-SM-18, Rev. 0	P8 to P8	SM-18	02/15/84
JCP-SM-19, Rev. 0	P1 to P8	SM-19	08/22/83

These procedures and the supporting procedure qualification records were found to be in compliance with the QA Manual and ASME Code Section IX. There were no findings in this area of the inspection.

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10. Identification and Control of Welding Materials

Although no welding was currently being performed, the facility for the storage and control of weld filler metal was examined for compliance with the QA Manual Control of Welding Materials. The area for storage was found to be clean and orderly, with weld material stored in a secure, heated oven. The materials were segregated, with the operating temperature of the oven high enough to preclude damage to low hydrogen content electrodes (at least 250°F as specified in AWS D1.1, paragraph 4.5.2). When low hydrogen content electrodes were issued for use, their exposure time was limited to a maximum of four hours (AWS D1.1 paragraph 4.5.2.1). Electrodes in hermetically sealed, unopened containers were found to be segregated by type and size and maintained in a secure storage area having controlled access. There were no findings in this area of the inspection.

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CORRESPONDENCE ADDRESS: Joseph Oat Corporation
ATTN: Mr. John Benckert
Quality Control Manager
2500 Broadway
Camden, New Jersey 08104

ORGANIZATIONAL CONTACT: Mr. John Benckert, QC Manager
TELEPHONE NUMBER: (609) 541-2900

PRINCIPAL PRODUCT: Nuclear Heat Exchangers, Fuel Storage Racks, Defueling Canisters.

NUCLEAR INDUSTRY ACTIVITY: Approximately 75% of Joseph Oat Corporation's activity is devoted to the commercial nuclear industry.

ASSIGNED INSPECTOR: R. L. Cilimberg 12/10/85
R. L. Cilimberg, Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): C. Abbate, SPIS J. Thomas, TMI Program Office
B. Brown, EG&G
R. Pettis, SPIS

APPROVED BY: John W. Craig 12/10/85
John W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Verify the implementation of the Joseph Oat QA program and compliance with the requirements of 10 CFR Part 21 during the fabrication of defueling canisters for TMI-2. These QA requirements and the applicability of 10 CFR Part 21 were specified in Bechtel purchase order TC-022111, Rev. 0 dated August 7, 1985, and Bechtel Technical Specification 15737-2-M-101A(Q), Rev. 2 dated June 18, 1985.

PLANT SITE APPLICABILITY: TMI-2 (50-320)

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A. VIOLATIONS:

Contrary to Sections 21.21(a)(1) and (2) of 10 CFR Part 21 Joseph Oat procedure SP-1522, "Procedure for Reporting Defects and Non-Compliances," is not appropriate in that it (a) fails to adequately address a procedure to be followed by Joseph Oat personnel in informing the licensee or purchaser of a deviation in order that it may be evaluated, and (b) fails to address informing a director or responsible officer of deviations on failures of conditions reportable to the NRC, Oat purchasers and/or licensees for equipment/services previously supplied by Joseph Oat.

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

B. NONCONFORMANCES:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50, and Section 4.2.2 of the Oat Quality Assurance Manual (QAM), Revision 8, dated June 26, 1985, lifting lugs were installed and removed on canister 2476A1 without a description in the traveler of the operation steps for the lifting lugs.

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50, and Section 4.2.2 of the Oat QAM, the shell of canister 2476A2 was formed with a steel bar and a hammer to achieve a concentric fit-up between the shell and the lower head without a description in the traveler for this operation step in the manufacturing process.

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 4.3.3.1 of Bechtel Technical Specification 15737-2-M-101A, Revision 2, dated June 19, 1985, Turco Dy-Check Remover No. 3 was being used for cleaning canister parts and welds before obtaining Bechtel's approval for the procedure which permits the use of this cleaner.

4. Contrary to Criterion XVI of Appendix B to 10 CFR 50, Special Condition 19 to Bechtel Purchase Order TC-022111, and Section 9.1 of the Oat QAM, a condition adverse to quality existed in that conflicting documentation left the status of the recombiner elements in the canister lower heads indeterminate and no action had been taken to identify and correct the apparent nonconformity.

C. UNRESOLVED ITEMS:

1. The chemical reaction of the concrete resin with water during refueling requires evaluation relative to the impact on TMI-2 defueling activities and subsequent transportation safety implications. This item is discussed in E.1 below.

2. The quality of the catalyst in the lower heads supplied by NES is an unresolved item based on indications of overheating, weight discrepancies, and absence of receipt inspection criteria as discussed in E.2 below.

3. Missing radiographs for a number of the longitudinal welds on the canister shells is an unresolved item. This item is discussed in E.10 below.

4. The method of pouring the concrete resin into the canister assembly is an unresolved item. The present procedure does not adequately address whether continuous pouring is required. The present method of intermittent pouring may have an effect on the overall curing and behavior of this material. This item is discussed in E.11 below.

5. The applicability of Joseph Oat invoking the provisions of 10 CFR 21 upon Air-Oil Systems Incorporated, Purchase Order 19515, is an unresolved item. This item is discussed further in E.3 below.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not applicable.

E. OTHER FINDINGS AND COMMENTS:

1. Durability of Concrete Resin - Cleaning of excess concrete resin from the surfaces of the canister was observed approximately 24 hours after the concrete resin was poured into the void between the boral shroud and the shell of fuel canister 2476A1. This cleaning was performed by wiping the excess concrete resin with a water-dampened cloth as required by section 5.1 of procedure QC-2476-82 dated August 1, 1985. The inspectors observed that the concrete resin was easily removed. This raised questions

concerning potential deterioration of the concrete resin when the canister is immersed in the water in the TMI-2 reactor and during subsequent long term storage. A sample of concrete resin (which had been poured approximately 24 hours previously) was placed in a cup of water and observed to interact with the water and weaken within one hour. There were no specifications for curing and installation of the resin, and the suitability of the concrete resin in fuel canisters when exposed to water without meeting the curing recommendations of the resin manufacturer have not been reviewed. Unresolved item number C.1 was identified in this area.

2. Inspection of Recombiner Catalyst in Lower Heads

The NRC inspectors noted that the recombiner elements had been installed in 30 of the fuel canister lower heads. These elements were installed by NES Manufacturing prior to shipment to Joseph Oat. Bechtel specification 15737-2-M-101A requires that the recombiner elements be handled in clean work areas with gloved hands and that care be exercised to protect the recombiners from contamination during canister fabrication. However, the Bechtel specification gives no indication of what foreign materials pose a threat of catalyst contamination.

The inspectors verified that the recombiners were adequately protected from contamination by dirt or debris from the shop environment at Oat and that Oat had provided adequate procedural controls during fabrication to assure the cleanliness as specified in the Bechtel specification. The inspectors selected four canister lower heads from stock and inspected the installed recombiners, four of which are welded into each head. The wire screens covering the catalyst pellets in all four of the elements in one head were contaminated with cloth fibers. Further review, determined that the shop traveller used as the basic canister fabrication procedure contained requirements to clean the inside of the head prior to its installation. The Quality Control Manager at Oat indicated that this would be done with a vacuum cleaner.

Visual examination of the recombiner elements in one of the heads revealed that the wire screen covering the elements exhibited a high degree of discoloration probably the result of welding the screens to the recombiner element. The catalyst pellets beneath the discolored area of the screen also appeared to be discolored. The pellets in this area, which comprised 40% to 50% of the pellets in each element, were much lighter in color. Oat had performed a visual inspection of the recombiner elements for shipping damage upon receipt, and relied upon a letter from Bechtel certifying that the elements installed by NES met the design specifications. The inspectors questioned whether the heat of welding, which had been performed by NES Manufacturing may have damaged the silicon coating on the catalyst pellets in these elements and impaired their operability. Oat did not detect the discoloration and had not inspected for this characteristic since Bechtel had not provided any information or specifications on what the characteristics of the catalyst should be.

Based on the observed discoloration of catalyst pellets and lack of inspection criteria and the catalyst weight problems discussed below the quality of the catalyst is in question. Unresolved item number C.2 was identified in this area.

3. Procurement Document Control

The inspectors reviewed purchase documents issued by Joseph Oat for procurement of material for fuel canister fabrication. Joseph Oat is in the process of fabricating 30 fuel canisters (A-1 through A-30) using material supplied by Bechtel. This material was shipped from NES Manufacturing's facility in Greensboro, NC, to Oat. In addition, Oat has been contracted to fabricate approximately 30 additional fuel canisters (A-31 through A-60) using Oat procured material. The inspectors verified that an acceptable Bill of Material (BOM) had been prepared by Oat from the Bechtel purchase order, technical specification, and drawings.

The BOM had been prepared in accordance with and met the requirements of the Oat Quality Assurance Manual (QAM). The BOM was applicable to fuel canisters A-1 through A-60, however, it served as the reference for generation of material procurement documents for only the second 30 canisters. Some, but not all, purchase orders have been issued by Oat for material for canisters A-31 thru A-60. The inspectors examined several of those purchase orders to determine whether Oat had incorporated the appropriate regulatory, technical, and QA program requirements. Bechtel specified that the requirements of 10 CFR 50 Appendix B and ANSI N45.2 were to be met by Joseph Oat as described in Technical Specification 15737-2-M-101A(Q) for the defueling canisters. The requirements for Certified Material Test Reports (CMTR) in the Technical Specification were subsequently clarified and relaxed in a letter from the buyer dated August 12, 1985. For the POs reviewed, in all cases where Bechtel specified that a CMTR was required, the Joseph Oat purchase order also specified that a CMTR be provided and invoked the quality assurance requirement of 10 CFR 50 Appendix B and ANSI N45.2 on the subtier supplier. The inspectors verified that these purchase orders were issued only to vendors currently on the Active Qualified Supplier List or were issued contingent upon Oat completing a vendor audit and placing the vendor on the Active Qualified Supplier List. Oat did not invoke the requirements of 10 CFR 50 Appendix B and ANSI N45.2 upon suppliers for material not requiring a CMTR (determined by Bechtel) but did specify that the subtier vendor provide a Certificate of Compliance certifying that the material was supplied in accordance with the purchase order and the applicable drawings and references. This met the requirement of the QAM since these vendors hold Quality Systems Certificates issued by ASME or other equivalent certificates.

Bechtel Technical Specification 15737-2-M-101A(Q) also required that the reporting requirements of 10 CFR 21 were to be met by Joseph Oat. The inspectors found that Oat had not invoked the requirements of 10 CFR 21, "Reporting of Defects and Noncompliance," on Air-Oil Systems, Incorporated (purchase order 19515). This purchase order was for four quick-disconnect fittings, two of which were considered commercial grade and not subject to the provisions of 10 CFR 21. However, the two remaining catalog items were modified by Air-Oil Systems thus removing their exemption from 10 CFR 21. Unresolved item number C.5 was identified in this area.

4. <u>10 CFR Part 21</u>

Joseph Oat Standard Procedure SP-1522, "Procedure for Reporting Defects and Non-Compliances," dated January 15, 1979, was reviewed by the NRC inspector and found not to be in compliance with 10 CFR 21. The NRC inspector determined that the procedure, SP-1522, was inadequate in that it did not contain appropriate procedures to assure that a director or responsible officer of the Joseph Oat Corporation is informed of deviations or failures as required by 10 CFR 21.21(a)(2). In addition SP-1522 failed to adequately address measures to make necessary notifications, by Joseph Oat personnel, of licensees or purchasers of deviations or failures in basic components supplied them in order that such deviations may be evaluated. SP-1522 is a general statement of policy rather than a procedure to implement such a policy. Violation A.1 was identified in this area.

Interviews with the Quality Control Manager and Vice President of Engineering indicated that Oat has never made a Part 21 report, nor were they aware of any instance where material or services furnished by them was the subject of a Part 21 report issued by either an Oat customer or a licensee.

The inspector also verified that the posting requirements of 10 CFR 21 were satisfied both in the Manufacturing and Engineering work areas. Posting included 10 CFR 21 and Section 206 of the Energy Reorganization Act of 1974.

5. Lifting Lugs

The NRC inspectors observed that two stainless steel plates containing holes (lifting lugs) for chain hooks were welded on opposite sides of the outer surface of canister shell 2476A1 to facilitate handling of the canister. After pouring the resin was completed the lifting lugs were removed. Traveler No. 2476A1 did not contain an operation description for installation or removal of the lifting lugs. Oat personnel stated that a procedure had been written to cover removal of temporary fixtures but this procedure had not been approved by Bechtel and had not been referenced in the traveler. Bechtel required that Oat adhere to the General Welding Requirements (G300), Revision 6, dated March 20, 1981, which is attachment 3 to Bechtel Specification 15737-2-M-101A. Section 11.1.2 of attachment 3 requires that Oat not fabricate until a weld table or map has been reviewed by Bechtel. Bechtel had not approved a weld map to cover Oat's welding of the lifting lugs on the canister shells. Nonconformance B.1 was identified in this area.

6. Audits

The inspectors reviewed Joseph Oat's internal QA audit program. The Oat QAM requires that the General Manager have the Quality Control Department audited once during each twelve month period and that the Quality Control Manager have all other portions of the QA program audited once each twelve months. The Quality Control Manager had prepared a schedule of audits providing for the requirements of one section of the QAM to be audited each month in such a manner that all of the twelve sections were audited once per year. These monthly audits were all performed by the lead auditor who currently is the Quality Control Manager. While this individual does have responsibility in many of the areas being audited, a comprehensive audit of all areas of the QAM is performed by the General Manager annually.

The General Manager did not have direct responsibility in the audited areas which meets the requirements of ANSI N45.2. The audits were performed using written checklists that provided space for the auditor to write descriptive comments on deficient areas. Audit reports were reviewed by management and documentation was provided of deficient areas and corrective actions taken. Corrective actions were documented, reviewed and approved by the auditor who had identified the deficiency and were subsequently reviewed by management.

The NRC inspectors reviewed Joseph Oat's program for qualification of suppliers which requires audits to be documented on a checklist which indicate the attributes of the supplier's QA program and the acceptability of the areas reviewed. One supplier qualification audit was reviewed.

The inspectors concluded that Joseph Oat's audit program met the requirement of the QAM.

7. Canister Cleanliness

Canister shells were stored indoors at the Oat facility under clear plastic sheets with rubber end caps on the empty shells. Cleaning operations are described on the traveler used during fabrication and Job Procedure No. JP-2476-1 is referenced for all cleaning.

This Oat procedure outlines the requirements for in-process and final cleaning of the fuel canisters and specifies permissible cleaning agents, mechanical cleaning limitations, and cleaning provisions for top head weldment assembly, vessel assembly, and cement pouring. However, the inspectors observed that weld wire was cleaned in a plastic container which did not identify the solvent. The NRC inspectors questioned whether the solvent could dissolve the plastic and coat the weld wire thus affecting the quality of the welds. Oat personnel were unable to evaluate the effect of the plastic containers on the welding operation and the plastic containers were promptly replaced with metal containers.

The NRC inspectors determined that Turco Dy-Check Remover No. 3 was being used extensively for cleaning canister shells, weld wire, welds, and other stainless steel parts. This cleaner is not listed on the Bechtel approved cleaning procedure JP-2476-1 and therefore should not be used for cleaning canisters. Revision 2 of procedure JP-2476-2 includes this cleaner, but revision 2 has not been approved by Bechtel. The NRC inspectors note that the cleaner is approved by Bechtel for removing dye penetrant used to perform inspections of the canisters. Nonconformance B.3 was identified in this area.

The NRC inspectors observed that tap water was used to remove the excess concrete resin discussed in Item 1 above, but the operator immediately switched to deionized water when questioned by the inspector. The QC Manager committed to revise QC-2476-82 to require deionized water.

8. Control of Special Processes

The NRC inspectors reviewed the Oat QAM and 12 Oat procedures to determine whether special processes were being conducted by qualified personnel using qualified procedures and equipment. A review of two travelers relating to fabrication of two fuel canisters revealed that all individual operations were properly initialed and dated. In addition, the hold points for witnessing by the Bechtel site inspector were signed or initialed and dated. The traveler package contained a weld and heat sketch and record sheet which identified bill of material item numbers, heat numbers, weld numbers, welding procedure specification numbers, welder stamp, and filler metal heat numbers. Operation descriptions in the traveler described fabrication steps and identified the approved drawings which contained weld notes and joint details.

While observing the fit-up (Operation Number 20) for welding the lower head to the shell of canister 2476A2, the NRC inspectors observed that the shell had been distorted by thermal stresses which had been initiated by welding the impact plate to the shell (Operation Number 16). The Oat welder achieved a concentric fit-up between the shell and the lower head by "tapping" the shell with a steel bar and a hammer. The Oat QAM, Section 4.2.2, requires that each operation be described on the shop traveler. The "tapping" forming operating was not covered by an approved procedure (Operation Number) on Traveler 2476A2. Nonconformance B.2 was identified in this area.

9. Training/Qualification

The NRC inspectors reviewed applicable sections of the QAM, and training records from 1983 to the present for 16 employees (8-manufacturing, 2-purchasing, 2-engineering, 3-QA/QC, 1-General Manager) to determine whether personnel performing and verifying

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activities affecting quality had received the necessary training and qualifications. Qualification records were evaluated for these personnel and it was determined that they met the requirements of Section 12 of the QAM for Indoctrination and Training. Training was given in various disciplines of the QAM and quality procedures. QA inspectors performing inspections, examinations and tests were required to pass a written examination and were certified as Level I, II, or III. Oat procedures were found to be in compliance with the guidelines of SNT-TC-1A.

Qualification records for three welders who are scheduled to work on the defueling canisters were reviewed. All welders were qualified to weld using procedures WPS-8303 "GTAW" and WPS-4304 SMAW. The qualifications were signed off by the welding engineer and the QC Manager. A current qualification Maintenance Chart is maintained and a file documenting that each welder had welded using a process to maintain their qualification in accordance with Section IX of the ASME Code is located in the QC Manager's office. A review of three Procedure Qualification Records (PQR) for WPS-8303 and three PQR for WPS-4303 and six test reports indicated that all testing had been performed as required in accordance with Section IX of the ASME Code. The disposition of welding filler metal and electrodes appeared to be in compliance with Section 5.1.5 through 5.1.9 of the QAM.

10. Nondestructive Examination

Nondestructive examination (NDE) of the TMI-2 defueling canister is to be performed in accordance with the requirements in the 1983 edition of the ASME Code Section VIII, Subsection UW (lethal), paragraphs UW-50, UW-51, and UW-53 as required by Bechtel Specification 15737-2-M-101A(Q).

The NRC inspectors determined that Oat has three certified NDE examiners currently participating in canister NDE examinations. John Benckert, QC Manager, is certified Level III in radiographic (RT), penetrant (PT), and visual (VT) methods; Charles Leonard, Assistant QC Manager, is certified Level II in RT, PT, and VT; and William Badaili is certified Level II in PT and VT. The Oat procedures for qualification and certification of NDE personnel, as well as the individual training and certification records including eye examination results for the above NDE examiners were found to be in compliance with the guidelines of SNT-TC-1A.

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All radiography is currently being performed under subcontract by Eastern Testing and Inspection, Inc. (ETI) of Pennsauken, NJ, with radiograph interpretation and final acceptance by Oat Level II and III examiners. The ETI procedures for qualification and certification of NDE personnel, including individual training and certification records with current eye examination results were reviewed by the NRC inspectors and found to be in compliance with the guidelines of SNT-TC-1A.

The NRC inspectors reviewed Oat procedures QC-2476-20, Radiographic Examination of Welds, Revision 1, dated August 12, 1985; QC-2476-10, Liquid Penetrant Examination of Welds, Revision 0, dated July 25, 1985; QC-2476-60, Visual Examination of Welds, Revision 1, dated August 12, 1985; SP-1579, Requirements for Qualification and Certification of NDE Personnel (J. Oat), Revision 4, dated September 1, 1983; and CP-101, Procedures for Qualification and Certification of NDE Personnel (ETI), Revision 3, dated June 1, 1981. All of these procedures were determined to meet the applicable ASME code requirements.

The NRC inspectors reviewed Oat procedure QC-2476-70, Ultrasonic Inspection for Circumferential Pipe Weldments per ASME Section VIII and V. This procedure is not on the List of Approved Procedures and, therefore, should not be used for canister inspection. This procedure can be confusing to the examiners in that paragraph 3.4.3 states that "Instrument settings during calibration shall not be changed during actual test." This is contradictory to paragraphs 4.1.1, 4.2.1, 4.2.2, and ASME code requirements (Section V, Article 5, paragraphs T-546-2.2.2, T-546-2.2.3, and T-546-2.2.4) which state that "Scanning shall be performed at a gain setting at least two times the primary reference level." Procedure QC-2476-70, paragraphs 4.2.1 and 4.2.2 for angle beam examinations, also states that the search units shall be manipulated so that sound passes through the required volume of weld and adjacent base metal. These paragraphs are incomplete unless they are expanded to include the ASME Section V, Article 5, requirements of paragraph T-524.1 which states that each pass of the search unit shall overlap a minimum of 10% of the transducer dimension perpendicular to the direction of the scan, and paragraph T-524.2 which requires that the rate of search unit movement for examination shall not exceed six inches/second unless calibration is verified at scanning speed.

The NRC inspectors reviewed the radiographs, technique sheets, and inspection report for the head to canister weld on canister No. 2476A-1 which was the only weld of this type that had been radiographed by Oat at the time of this inspection. No rejectable indications were noted during review of the radiographs and inspection reports and the radiographic technique, quality, and densities were determined by the NRC inspectors to meet the ASME code requirements.

The NRC inspectors reviewed radiographs and inspection reports for the longitudinal welds on the canister shells that were supplied to Oat by NES. The radiographs had been supplied initially by Armco Steel, Wildwood, Florida. Review of these radiographs by the NRC inspectors did not identify any indications on the radiographs that were different than those reported by Armco and the quality of the radiographs was determined to meet the applicable requirements of the ASME code.

The identification numbers of the shells for which radiographs and inspection reports were reviewed are as follows:

95P1	95P2
24P1	24P2
109P1	109P2
129P1	129P2
137P1	137P2
120P1	120P2
130P1	130P2
89P1	89P2
44P1	44P2
110P1	110P2
117P1	117P2
127P1	127P2
124P1	124P2
126P1	126P2
92P1	92P2

The NRC inspectors observed the dye penetrant examination (PT) of impact plate to shroud welds on canister numbers 2476-A2 and 2476-A3 and the PT of the lower head to canister root pass weld on canister number 2476A2 and concluded that the techniques, materials, and documentation for the examinations were in conformance with approved procedures and the inspection personnel (Level II) and materials were certified.

ORGANIZATION: JOSEPH OAT CORPORATION
CAMDEN, NEW JERSEY

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During the inspection, the NRC inspectors were informed by Oat that Bechtel had advised Oat that some radiographs were missing for the longitudinal welds on the canister shells before the shells were shipped to Oat from NES. During Oat's inspection of packing lists provided by NES, Oat determined that 40 welds had one view missing and one weld had no radiograph. Oat issued Deviation Notice number 2956 dated September 24, 1985 per Oat QAM section 3.2.6. The NRC inspectors determined that Oat's action was acceptable in that missing radiographs will be replaced. Unresolved item number C.3 was identified in this area.

11. Concrete Resin Pouring

The NRC inspector witnessed the concrete resin pour for fuel canister assembly No. 1. Oat Procedure QC-2476-82, dated August 1, 1985, describes this activity.

This low density (62.5 pounds/cubic foot) light-weight concrete consists of a mixture of 60% refractory cement, 11% glass bubbles and 29% demineralized water, by weight, which fills the cavity between the square inner boral shroud and the circular outer shell of the canister assembly thus providing continuous lateral support to both components. This distributed loading function is intended to minimize instantaneous displacements in the overall shape of the boral shroud during all postulated accident conditions to successfully meet the criticality criteria.

The inspector observed the Joseph Oat technician preparing the concrete which was supplied by the Advanced Ceramics Division of Babcock & Wilcox (B&W). This material consisted of 148 boxes of pre-measured LICON ultralight concrete certified by B&W in a Certificate of Compliance, dated April 9, 1985, as meeting the design specification. It could not be determined by Oat personnel, or the NRC inspector whether the pre-mix furnished by B&W contained the proper proportions (60/11 percent by weight) of refractory cement and glass bubbles. The quality of the mix is questionable in light of the observation that the resin loses structural properties when allowed to react with water as discussed in E.1 above.

As the placing of the concrete continued, the NRC inspectors noticed that the Oat procedure did not clearly address the method of pouring the resin, i.e., continuous or intermittent. It is uncertain whether this potential lack of bonding of the resin may have any affect on the structural analysis which was performed to substantiate the structural integrity of the fuel canister. Unresolved item number C.4 was identified in this area.

12. Calibration of Measuring and Test Equipment

Section 6 of the Oat QAM, "Tool, Gauge, and Equipment Control," and calibration records (Gauge Control Records and Calibration Cards) for several different pressure gauges, micrometers. vernier calipers and densitometer were reviewed. All tools, gauges and equipment are identified with a unique number and recorded on a Master Tool List, with the exception of tapes, rules and scales which are calibrated only if so specified by contract. A "periodic check" as defined in Section 6.1.3, of the Oat procedure, is also performed each time an instrument is used to ensure its continued accuracy. All instruments observed during the inspection possessed a calibration "sticker" which contained the date calibrated, next calibration due date, device identification number and the individual performing the calibration.

Several instruments, used primarily for receipt inspection of canister shells, were reviewed for compliance. These instruments were a 0-1 inch tube wall micrometer (No. J-20) and a 0-24 inch vernier caliper (No. K-7). In addition calibration data for an X-rite Model 301 densitometer, used in conjunction with the reading and interpretation of radiographs, was also reviewed and found to be satisfactory. As an extra precaution, the Oat inspector calibrated item No. J-20 at 0, 25, 50, 75, and 100 percent of range, with item No. K-7 calibrated at 0, 25, and 50 percent. This additional step ensures calibration is also maintained between intervals. All calibration activities, reviewed by the NRC inspector, were performed by the Quality Control Manager or his assistant.

13. Control of Purchased Material

No material purchased by Oat had been received at the time of the inspection. Material at Oat was provided by Bechtel and shipped from NES Manufacturing to Oat. The inspectors verified that the material had been receipt inspected and entered into Oat's material control program. As material was received, the material receiving report (MRR) was filled out listing the items received, heat numbers when applicable, whether proper documentation was received, and whether there was apparent shipping damage. As subsequent receipt inspections, such as dimensional checks, were performed, the MRR was annotated showing the area to be inspected was completed. However, the specific items checked were not identified on the MRR. The receipt inspections being performed and the completed documentation relating to the receipt inspections met the requirements of the QAM.

Activities performed and observed by the NRC inspectors during receipt inspection of the canister shells shipped from NES by Bechtel included a dimensional check of the shell (OD, ID, straightness and length), in addition to a visual examination for cracks and damage as required by Oat Procedure SP-1532, Revision 0, dated October 10, 1979. Oat personnel observed that shells furnished by Bechtel from NES did not possess the weld preparation necessary for full penetration welding of the lower head and bulkhead components to the fuel canister shells. Oat arranged for a local contractor to machine a J-bevel in the end of the shells to facilitate fit-up for welding the lower heads to the shells.

During receipt inspection of the boral shrouds, the NRC inspectors determined that a wall thickness inspection was not performed by Oat as required by Section 3.2.3 of the Oat QAM. The failure to identify specific characteristics inspected was identified as a weakness in Oat's receipt inspection activities.

During the NRC inspection the Oat QC inspector inspected the dimensions of shrouds in addition to committing to incorporating complete dimensional measurements of all shrouds in future inspections.

The receipt inspection activities performed on the shrouds did not follow a prescribed checklist of items to be verified, but rather an overall check of dimensions in accordance with manufacturing drawings. The consistency of measurements is questionable when measurements to be checked and acceptable tolerances are not specifically prescribed. Further, the blue-line drawings used as a basis for dimensional check during receipt inspection were poor quality reproductions.

The inspectors reviewed a random sampling of CMTRs and Certificate of Compliances for material supplied by Bechtel. During this review, the inspectors noted that Bechtel had supplied a letter as a Certificate of Compliance indicating that the canister lower heads and recombiner catalyst met all design specifications. The canister lower heads had been formed by a subcontractor to NES Manufacturing and the catalytic recombiner elements had been installed by NES Manufacturing. Bechtel then supplied these units to Oat with a Bechtel Certificate of Compliance. However, the inspector found in the documentation package a Suppliers Deviation Disposition Request (SDDR) issued by NES Manufacturing to Bechtel indicating that when the recombiner catalyst was weighed and bagged from bulk supply individual inspection verification of weights was not performed. There had been no indication of resolution of the potential discrepancy on the weights of catalyst actually installed in the elements. The NRC inspectors found that Oat had not evaluated this SDDR which appears to be in conflict with the Certificate of Compliance on the recombiner elements. Bechtel issued the Certificate of Compliance without determining whether NES had corrected the discrepancy on the weights of the catalyst actually installed in the lower heads. Nonconformance B.4 was identified in this area.

14. Material Identification and Control

The NRC inspectors reviewed the procedures and the functioning of the material control system. Material markings, identifying each component by job, item and heat number, were found etched into the material. Transfer of markings prior to cutting each piece was not verified since Joseph Oat's contract primarily calls for canister assembly. Nonconforming canister shells were properly segregated from production material and all observed to possess a "Hold" tag.

ORGANIZATION: JOSEPH T. RYERSON & SON, INC.
PHILADELPHIA, PENNSYLVANIA

REPORT NO.: 99900876/85-01	INSPECTION DATE(S): August 12-16, 1985	INSPECTION ON-SITE HOURS: 93

CORRESPONDENCE ADDRESS: Joseph T. Ryerson & Son, Inc.
ATTN: Mr. Francis W. Thorley
General Manager
5200 Grays Avenue, Box 7349
Philadelphia, Pennsylvania

ORGANIZATIONAL CONTACT: Mr. Raymond DeLuca, QA Manager
TELEPHONE NUMBER: (215) 724-0700

PRINCIPAL PRODUCT: Steel, aluminum, plastics

NUCLEAR INDUSTRY ACTIVITY: Less than 1% of Joseph T. Ryerson & Son, Inc. business is supplying nuclear grade material.

ASSIGNED INSPECTOR: J. C. Harper, Reactive Inspection Section (RIS) 11-22-85 Date

OTHER INSPECTOR(S): N. J. Miegel, RIS
T. F. Burns, Brookhaven National Laboratory (BNL)

APPROVED BY: E. W. Merschoff, Chief, RIS, Vendor Program Branch 11/22/85 Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was made as a result of a 10 CFR Part 21 notification made by Calvert Cliffs Nuclear Power Plant on May 28, 1985 concerning a shipment of type 316 stainless steel rather than the 17-4 PH stainless steel ordered. The inspection addressed the areas of material traceability as well as the adequacy and implementation of J. T. Ryerson (Philadelphia) Quality Assurance Program in accordance with Appendix B to 10 CFR 50.

PLANT SITE APPLICABILITY: Calvert Cliffs, 50-317, 50-318, Susquehanna 50-387, 50-388, Limerick 50-352, 50-353, Palo Verde-50-528 50-529, 50-530

REPORT NO.: 99900876/85-01	INSPECTION RESULTS:	PAGE 2 of 10

A. Inspection Issues

The inspection was conducted in response to a Part 21 report received from Baltimore Gas and Electric, which stated that Joseph T. Ryerson & Son, Inc. - Philadelphia, PA (J. T. Ryerson) had supplied the Calvert Cliffs Nuclear Power Station with Type 316 stainless steel rather than the 17-4 PH stainless steel ordered. The 17-4 PH stainless steel was to be used for body to bonnet studs for the pressurizer spray valves.

B. Inspection Findings

1. Violations:

a. Contrary to Section 21.6 of 10 CFR Part 21, J. T. Ryerson did not meet the posting requirements of 10 CFR Part 21. This is a Severity Level V violation (85-01-01).

b. Contrary to Section 21.31 of 10 CFR Part 21, J. T. Ryerson failed to specify 10 CFR 21 as an applicable requirement on purchase orders for "Basic Components" issued to Morris Wheeler & Company, Inc. (J. T. Ryerson work order No. 9W16/648A2 dated 1/13/84) and Lehigh Testing Laboratory (J. T. Ryerson work order No. 9W14648A3 dated 1/13/84). This is a Severity Level V violation (85-01-02).

2. Nonconformances:

a. Contrary to Criterion I of Appendix B to 10 CFR Part 50, and J. T. Ryerson document "Critical Requirement Material Instruction," (J. T. Ryerson No. 9W14648A1, dated 1/13/84) the same employee was identified by signature as both the salesperson and the QA coordinator (85-01-03).

b. Contrary to Criterion I of 10 CFR Part 50, Appendix B, the Quality Assurance Manager at the Philadelphia facility is also designated as the "Credit/Office Manager". As Credit/Office Manager, he is responsible for the financial aspects of claims settlement (85-01-03).

c. Contrary to Criterion II of Appendix B to 10 CFR Part 50, and Section 4.8 of the J. T. Ryerson QAM, there was no objective evidence to indicate that a formal QA training program for employees assigned quality related duties had ever been established and/or implemented (85-01-04).

d. Contrary to Criterion VII of Appendix B to 10 CFR Part 50 and Section 5.1 of the J. T. Ryerson QAM, J. T. Ryerson procured material and/or services from two firms that were not listed on the "Ryerson Quality" sources for Critical Applications (85-01-05).

e. Contrary to Criterion VIII of Appendix B to 10 CFR Part 50, Section 5.2 of the J. T. Ryerson QAM, and Form 856.07-8 "Receiving and Identification", there were several bundles of material in stock without receiving tags (85-01-06).

f. Contrary to Criterion VIII of Appendix B to 10 CFR Part 50, Section 5.2 of the J. T. Ryerson QAM and the "Receiving Instruction Manual" loose plates were observed by the inspector which were not marked with the size, grade, or heat number as required (85-01-06).

g. Contrary to Criterion VIII of Appendix B to 10 CFR Part 50, and J. T. Ryerson policy and procedure 18.400.07, Sections I and II, material was in stock without a color code marking, with incorrect color code markings and the color coded end of some stock was not accessible without removing the material from the rack (85-01-07).

h. Contrary to Criterion X of Appendix B to 10 CFR Part 50, foreman and operators who were responsible for production functions were also delegated the responsibility for the receiving inspection (85-01-08).

i. Contrary to Criterion XVII, Appendix B of 10 CFR Part 50, the data recorded on various discrepancy reports were insufficient to identify the discrepant item, its origin/vendor, disposition, QA review and approval of the disposition, and accomplishment of the corrective action. Furthermore, DRs are routinely voided without explanation and are not identified as accountable documents (85-01-09).

j. Contrary to Criterion XVIII of Appendix B to 10 CFR Part 50 and Sections 6.1 and 6.4 of the J. T. Ryerson QAM (85-01-10):

1) There was no objective evidence that corrective action, followup, and close out of all deficiencies noted in the 1984 management audit of J. T. Ryerson Philadelphia had been completed.

2) The 1985 internal audits completed as of August 16, 1985 were performed by a lead auditor not yet qualified per ANSI N45.2.23.

C. Unresolved Items

None.

D. Other Findings and Comments

On May 28, 1985, Baltimore Gas & Electric notified the NRC that the Calvert Cliffs Nuclear Station received a shipment of Type 316 stainless steel rather than the ASTM A-564, Type 630, Condition A, 17-4 PH stainless steel ordered. The material was to be used for body to bonnet studs on the CV-100 E&F pressurizer spray valves. The heat involved was #656045 and was ordered from the J. T. Ryerson - Philadelphia Plant. However, the shipment of the questionable material originated from the Joseph T. Ryerson - Chicago facility according to the original P.O., Order No. 67864-GX.

1. 10 CFR Part 21 Requirement

J. T. Ryerson upgraded stock material to meet the requirements of ASME III, Class 2 by sending material samples to the Lehigh Testing Laboratory for chemical and mechanical testing. However, the J. T. Ryerson purchase order requirements to Lehigh Testing Laboratory did not indicate the applicability of 10 CFR Part 21. The work was completed by Lehigh and the results used in the certifications were supplied by J. T. Ryerson to the purchasers. The two subject purchase orders examined were:

P.O.	Item	Date	For
8031-F-61459	SA 5156R 70-BAR	7/12/83	Philadelphia Electric
6064	SA 240 Typ. 316L	12/8/83	Arizona Public Service

Violations 85-01 and 85-02 and Nonconformance 85-01-05 were identified in this area of the inspection.

2. Quality Assurance Program Organization

The Quality Assurance Manager at each J. T. Ryerson plant is assigned the function and responsibilities of the Credit/Office Manager at his location. The responsibilities of the Credit/Office Manager include activities which are directly related to cost and schedule requirements such as settlement of claims with suppliers and purchasers of J. T. Ryerson products (paragraph 4.3.2.2 of J. T. Ryerson QAM) as well as customer credit obligations.

During this inspection, it was noted that the demands on the Credit/Office Manager were predominantly in the area of settlement of claims and cost/schedule requirements. Minimum time was spent in the Quality Assurance Area. Virtually all of the quality related functions were delegated to the Quality Assurance Coordinator. The participation of the Quality Assurance Manager in this inspection was minimal.

Nonconformance 85-01-03 was identified in this area of the inspection.

3. Training

Section 4.8 of the J. T. Ryerson QAM was reviewed and current (August 19, 1985) employee training programs at J. T. Ryerson were discussed with the General Superintendent and the QA Coordinator. Personnel training in quality assurance and quality related concerns was performed on an as-needed basis determined by either the General Superintendent or the QA Coordinator. Attendance at training sessions was not recorded prior to August 8, 1985. The inspector reviewed an attendance record for a training session held on August 15, 1985. This training covered color coding, receiving and heat numbers. Thirty-one employees attended the session.

Nonconformance 85-01-04 was identified during this area of the inspection.

4. Independence of Inspection Personnel

There are no qualified "inspection" personnel or "inspectors" in the Quality Assurance organization at the Philadelphia facility of J. T. Ryerson. The inspection functions at the plant are performed by the foremen, stockmen, and/or unloaders.

The J. T. Ryerson QAM provides for additional quality related activities in Section 5.5 when the order is identified as "nuclear" or "safety related". This Section is titled "Order Processing Control-Critical Requirement Orders." These processing requirements are additional requirements to the normal "commercial" items when involved in processing material orders. However, these additional steps do not effectively separate the quality and production functions to the extent required by Criterion X of 10 CFR 50 Appendix B. The actual inspection is still carried out by production personnel as detailed in paragraphs 5.2.2.2, 5.5.2.3, and 5.5.2.4 of the QAM. The additional activities involve a verification function by the Quality Assurance Coordinator rather than an actual physical inspection (except as provided for "special requriements" in paragraph 5.5.2.5). It appears that more specific physical involvement by quality assurance or quality control personnel is necessary to maintain the independence of the inspection activity.

Nonconformance 85-01-08 was identified during this area of the inspection.

5. Discrepant Material Control Program

An examination was made of a total of 89 discrepancy reports (DR) during the inspection. Numerous deficiencies were noted regarding the way these documents are controlled and processed by J. T. Ryerson. Those deficiencies are:

	Item	DRs Involved
a.	No vendor identified	7
b.	No recommended disposition	16
c.	No statement of problem	1
d.	No disposition accomplished	22
e.	DR was "voided" without explanation	17
f.	Signed by unidentified person for QA Mgr.	1

In addition, the DR generally lacked such information (blocks were without entries) as: J. T. Ryerson order number, mill invoice number, mill order number, shipping point, date shipped, car number or carrier process number, shipping notice number, packing number, and heat number.

The Quality Assurance Coordinator reviews each DR and signs as approving the disposition in the capacity of "Claim Manager." This is the only individual who acts with approval authority on the recommended disposition, and it is unclear whether the action taken is a result of financial considerations (Claim Manager) or quality considerations (Quality Assurance Coordinator). A provision is made on the DR for disposition action by the "National Quality Assurance Coordinator (NQAC)" who is headquartered in the J. T. Ryerson General Offices in Chicago, Illinois. However, no copies with this signature were available for examination at the Philadelphia facility. It is questionable that any real useful purpose is accomplished by the NQAC approval since it is a long "after the fact action." This was evident since DRs 185820 and 185922 (original copies) were still in Philadelphia (without NQAC approval) but were signed off as "dispostion accomplished" on 9/14/84.

Discussions with J. T. Ryerson quality assurance personnel revealed that the DR is not considered an "accountable document." Therefore the actual status of individual DRs was not always clear. For example, DRs would be issued in bulk (15-25 copies) to various production department personnel who would then issue them on an as-needed basis when a discrepant condition was identified. Thus, many DRs were listed as issued in the DR log without further information. This particular practice was not documented in the J. T. Ryerson QAM or in their policy and procedure instructions. The J. T. Ryerson QAM does not make provision to "VOID" a DR once it is initiated. However, this action was routinely performed. Since January 1985 a total of seventeen DRs were noted as "VOID" without explanation for this action. It was not clear which individual made the determination to "VOID" the discrepancy reports.

Proper use and control of the discrepancy report system will provide meaningful data to enable material suppliers such as J. T. Ryerson to evaluate the effectiveness of their own internal operations and monitor the quality level being supplied by its vendors.

Nonconformance 85-01-09 was identified during this area of the inspection.

6. QA Resolution of Discrepant Conditions

The QA Manager at the Philadelphia facility is given the responsibility to resolve discrepant conditions with the "source." At J. T. Ryerson, the QA Manager has had a minimum involvement in the resolution process. In fact, he is rarely involved with the discrepancy reporting and resolution activity. The QA Manager functions as the Credit/Office Manager at this facility. This latter responsibility has, to a large degree, removed him a considerable distance from the day-to-day quality assurance activities. This area of multiple responsibilities has resulted in a situation where it is unclear whether the quality function is being performed as intended by the J. T. Ryerson Quality Program.

Nonconformance 85-01-03 was identified during this area of the inspection.

7. Failure of Color Code Identification System

J. T. Ryerson has a very detailed color code system used to maintain material identification in its storage facility. Personnel at individual J. T. Ryerson plants are not permitted to make any changes in the color code system (Policy and Procedure 18.400.01, III, A, 2) and are directed to affix the assigned color code to materials prior to placing them in storage (Policy and Procedure 18.400.07, I, B, j).

The implementation of this system was observed and determined to be inadequate. This conclusion was the result of the following observations:

a. Approximately 30% of aluminum structural shapes in storage were unmarked and some aluminum bar was unmarked.

b. Numerous items (bars, tubes, shapes) were marked with additional colors as a result of inventory activities.

c. Carbon steel bars and rod material were marked with the incorrect color code. Some carbon steel bars and rod material were unmarked.

d. Marking of sheet steel was inconsistent (some marked, most not marked).

Specific Examples

4130 annealed bar	- no color affixed
4140 hex stock	- wrong color affixed
1" 8620 round stock	- wrong color affixed
5/8" cold drawn round stock	- no color affixed
8" 6061 aluminum bar	- no color affixed
ASTM A607 sheet	- marked green/white
ASTM A607 sheet	- no color affixed

The identification and control of materials within the J. T. Ryerson warehouse is absolutely vital to assure that the material shipped is what the customer has ordered. This activity at J. T. Ryerson is governed by the two documents Policy and Procedure 18.400.01 and 18.400.07, and they appear adequate. Assurance can be provided that the system is operating correctly by marking the materials in accordance with the established color codes upon receipt (before placement in storage) and purging the existing stores of unmarked and unidentifiable materials.

Nonconformances 85-01-06 and 85-01-07 were identified during this area of inspection.

8. Mixed Sheet Material In Storage

J. T. Ryerson has not stated how it will identify sheet steel. This action is necessary since the current practice is inconsistent and various plant personnel opinions exist regarding "unwritten policy". Sheet material was observed in storage which was color coded and stenciled with identification Other sheet material was found in close proximity to marked material which lacked any identification. In fact, two stacks of material were found with both marked and unmarked material. In addition, stainless steel (300 series) sheet was found intermixed with aluminum (unidentified as to type or grade) sheet.

Specific Examples

Stack #1

2 sheets 304 stainless - identified
1 sheet 316L stainless - identified
2 sheets aluminum - unidentified
several sheets - 3003 H14 aluminum - identified
several sheets - 5454 H32 aluminum - identified

Stack #2

Several sheets 310 stainless - identified
Several sheets 304 stainless - identified
Several sheets 5052 aluminum - identified
Several sheets 5005 aluminum - identified

9. Internal Audits

Internal audit reports for 1985 completed as of August 16, 1985 were reviewed. Audit reports of Project Engineering Warehouse and Merchandise Department dated from 1980 through 1984, and the J. T. Ryerson management audits for 1982, 1984 and 1985 were also reviewed. Objective evidence of these audits, such as the auditor's handwritten notes, were available for all reports. The head auditor's qualifications were also verified. The 1984 management audit identified five items which the auditor identified as minor deviations. There was no objective evidence of corrective action, followup and closeout for these items. The J. T. Ryerson QAM makes no distinction between minor deviations requiring no action and major deviations, those which require formal corrective action. J. T. Ryerson personnel acknowledged this condition but could not elaborate on the occurrence.

Nonconformance 85-01-10 was identified during this area of the inspection.

10. Plant Tour

A tour of the J. T. Ryerson facility was made to assess the quality aspects of their activities. The activities observed were shipping and receiving, placement of material in storage, cutting and packaging. The storage areas were neat, clean, and free of extraneous material. Operations observed appeared to be well planned and progressing in an orderly fashion.

ORGANIZATION: JOSEPH T. RYERSON & SON, INC.
CHICAGO, ILLINOIS

REPORT NO.: 99901039/85-01	INSPECTION DATE(S): 09/16-20/85	INSPECTION ON-SITE HOURS: 103

CORRESPONDENCE ADDRESS: Joseph T. Ryerson & Son, Inc.
ATTN: Mr. Edward A. Mullin, Vice President and General Manager, Chicago Plant
16th and Rockwell Street
Box 8000-A
Chicago, Illinois, 60680

ORGANIZATIONAL CONTACT: Mr. John Bingham, Chicago QA Manager
TELEPHONE NUMBER: (312) 762-2121

PRINCIPAL PRODUCT: Steel, aluminum and plastics.

NUCLEAR INDUSTRY ACTIVITY: Less than one percent of current business is raw metal for commercial nuclear application.

ASSIGNED INSPECTOR: [signature] 6Feb86
E. H. Trottier, Reactive Inspection Section (RIS) — Date

OTHER INSPECTOR(S): J. C. Harper, RIS
T. F. Burns, BNL

APPROVED BY: [signature] 6 Feb 86
E. W. Merschoff, Chief, RIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was performed as a result of the 10 CFR Part 21 notification made by Calvert Cliffs Nuclear Power Plant on May 28, 1985. The inspection addressed the area of material traceability, as well as the adequacy of J. T. Ryerson's Quality Assurance Program and its implementation in accordance with Appendix B to 10 CFR Part 50.

PLANT SITE APPLICABILITY: Calvert Cliffs Nuclear Power Plant 1/2 (50-317, 318).

ORGANIZATION: JOSEPH T. RYERSON & SON, INC.
CHICAGO, ILLINOIS

REPORT NO.: 99901039/85-01	INSPECTION RESULTS: 09/16-20/85	PAGE 2 of 13

A. Inspection Issues:

On May 28, 1985, Baltimore Gas and Electric Company submitted a 10 CFR Part 21 notification to NRC Region I. The subject of this Part 21 notification was the receipt of incorrect stainless steel for pressurizer spray valve bonnet studs at Calvert Cliffs Nuclear Power Plant, Unit 2. The incorrect material was ordered through the Philadelphia warehouse of Joseph T. Ryerson & Son, Inc.

On August 8-12, 1985, the Vendor Program Branch of the NRC performed an unannounced inspection of the Joseph T. Ryerson & Son, Inc. warehouse facility in Philadelphia, PA (Inspection Report 99900876/85-01). In addition to two violations and seven items of nonconformance, the inspectors noted that it is the policy of Joseph T. Ryerson & Son, Inc. to supply all nuclear-grade material from its central facility in Chicago, Illinois, and that the subject material received by Calvert Cliffs did, in fact, originate from that facility. A review of the Certified Material Test Report submitted by the mill (ARMCO) verified that Joseph T. Ryerson supplied Heat #656045 (17-4 PH stainless steel) to Calvert Cliffs as ordered.

This inspection was conducted to determine the circumstances contributing to incorrect steel being received and installed in the reactor coolant systems at Calvert Cliffs Nuclear Power Plant. In addition, the inspectors concentrated on the findings identified during the Philadelphia inspection to determine if similar problems existed at the Chicago facility.

B. Inspection Findings

1. Violations:

a) Contrary to Section 21.6 of 10 CFR Part 21, J. T. Ryerson-Chicago did not meet the posting requirements of 10 CFR Part 21. (85-01-01)

A Severity Level V violation was issued in this area.

b) Contrary to Section 21.31 of 10 CFR Part 21, J. T. Ryerson-Chicago failed to specify Part 21 as an applicable requirement on numerous work orders associated with "Basic Components" issued to Charles C. Kawin Testing Company in 1984. (85-01-02)

A Severity Level V violation was issued in this area.

2. Nonconformances:

a) Contrary to Criterion VIII of 10 CFR Part 50, Appendix B, sheet steel was purchased and placed into storage without markings that established identification and control of this material. (85-01-03)

b) Contrary to Criterion VIII of Appendix B to 10 CFR Part 50 and Joseph T. Ryerson Policy and Procedure 18.400.07, material was in stock without color code, material had been placed in stock with incorrect color code, and material was placed in storage compartments where the color code could not be seen without removal of the material from its stored location. (85-01-04)

c) Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 5.5 of the QA Manual, the Quality Assurance Coordinator failed to verify special requirements on the Critical Requirement Material Instruction Form for numerous orders placed by the Zack and Bahnson companies. (85-01-05)

d) Contrary to Criterion XII of Appendix B to 10 CFR Part 50 and Policy and Procedure 18.220.02, measuring and test equipment was not uniquely identified and was not calibrated in accordance with the prescribed calibration schedule. Further, no acceptance criteria (allowable deviation) had been established for this equipment. (85-01-06)

e) Contrary to Criterion XVIII of Appendix B to 10 CFR Part 50 and Section 6.1 of the QA Manual, seven annual internal audits of at least five Ryerson plants were not performed. (85-01-07)

f) Contrary to Criterion XVIII of Appendix B to 10 CFR Part 50 and Section 6.2 of the QA Manual, deficient areas noted in the Detroit plant internal audit had not been closed out at the time of this inspection (September 20, 1985). The Detroit facility committed to close out these audit findings by April 29, 1985. (85-01-08)

g) Contrary to Criterion XVIII of Appendix B to 10 CFR Part 50 and Section 6.4 of the QA Manual, three lead auditors were qualified without evaluation of their communication skills, and two lead auditors were qualified without having performed a minimum of five QA audits. Two other lead auditors were qualified without having performed a minimum of five QA audits within three years prior to their qualification date. (85-01-09)

h) Contrary to Criterion XV of Appendix B to 10 CFR Part 50 and Section 5.3 of the QA Manual, a Discrepancy Report for nonconforming material received from a supplier was not completed. No evidence could be found to substantiate the actions taken to properly disposition the discrepancy. (85-01-10)

C. Other Findings and Comments

1. Identification and Control of Materials

Ryerson does not have procedures to mark sheet material adequately, because this product form (three-sixteenths inch thick and less) does not lend itself to edge marking. Thus, sheet material was found unmarked, yet no provision is made for this exclusion in the QA Manual or in the Policy and Procedure Bulletins. (85-01-03)

Ryerson uses an extensive color coding system to identify materials by type and grade within its storage facilities. Generally, the material is ordered with the color code applied by the mill, and inspection of the color code is made at the time the shipment is received. If the material has not been color coded, or has been incorrectly color coded by the mill, Ryerson personnel affix the required identification before the material is placed in storage.

A comprehensive examination was made of all storage facilities at the Chicago location (including the Plate Plant). Products and product forms examined included the following:

a. Carbon steel bars
b. Alloy steel bars
c. Carbon steel plates
d. Alloy steel plates
e. Steel tube (and structural steel shapes)
f. Stainless steel (plate, rod, bar and tube)
g. Nickel (rod and tube)
h. Aluminum (plate, sheet, rod, bar, tube and structural shapes)

The implementation of the color code program lacks adequate control, based on the number of instances where materials were found to be either unmarked or marked incorrectly. Due to conflicting identification methods on some materials, it was not always possible to determine the material identity without more extensive examination, or use of additional test methods. This conclusion is the result of the following observations:

Plate Material - A random selection of 89 plates was made to determine compliance with the established color code (identification procedure). Of these 89 plates:

- 49 plates were identified and marked correctly
- 38 plates were not color coded
- 2 plates could not be identified with certainty
- 5 plates (of the above) were stored in the wrong compartment

Details and inadequacies of implementation of the identification procedure are presented below:

Material Examined	Color Code	Storage Location
12 stainless steel plates (304, 304L, 316L)	Color coded correctly	Stored in correct location
6 stainless steel plates (including "shorts")	No color code	Stored in correct location

Material Examined	Color Code	Storage Location
37 carbon steel plates (A36)	Color coded correctly	Stored in correct location
28 carbon steel plates	No color code	Stored in correct location
1 plate-marked as "A36" on plate	Pink (abrasion resistant steel)	Stored in carbon steel plate area
4 plates-carbon steel	No color code	Stored in A285 compartment
1 plate-carbon steel	Marked as A285	Stored in A515 compartment

Aluminum (all product forms) - The stocks of aluminum material were found to be almost entirely unmarked. It is estimated by the inspector that the percentage of this material not marked with the prescribed color code exceeded 90 percent. A representative sample of the product forms examined are listed as follows:

Material Examined	Color Code	Storage Location
290 pcs. tube	No color code	Aluminum storage area
4 bundles-squares	No color code	Aluminum storage area
177 pcs-angles	No color code	Aluminum storage area
42 pcs-beams	No color code	Aluminum storage area
46 pcs-channel	No color code	Aluminum storage area
7 stacks-plates (shorts)	No color code	Aluminum storage area

Bar/Rounds/Tube - A random selection of material types and shapes were made for examination to determine the extent of compliance with color code requirements. Twenty two bundles (exceeding 25 pieces per bundle) of 321 rounds and 33 tubes were examined with the following results:

- 196 pieces had no color code
- 16 bundles had no color code
- 6 bundles of material requiring a dual color code were incorrectly marked
- 122 pieces were tagged with identification that differed from the color code applied
- 36 pieces were correctly color coded and tagged

Details on the bar, rounds and tube shapes examined are listed as follows:

Material Examined	Color Code	Storage Location
2 Bundles 1" Round (tagged 4140)	One-half bundle solid black, one-half bundle solid blue	Receiving area
1 Bundle 1" Round (tagged) 4940)	One-half bundle-solid black, one-half bundle-solid white	Receiving area
28, 1" x 9" bars (tagged A36)	Marked green (1018, 1020)	Carbon steel bar area
36, 1" x 9" bars (tagged A36)	Marked pink and brown (A36)	Carbon steel bar area
33, 1" x 9" bars (tagged A36)	No color code	Carbon steel bar area

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Material Examined	Color Code	Storage Location
10, 1" x 9" bars (tagged A36)	Marked blue and white: (there is no carbon steel material allowed with this code- only 410 stainless	Carbon steel bar
8, 1" x 9" bars (tagged A36)	Marked blue (1035)	Carbon steel bar area
2 Bundles 1½" round tagged "Stressproof"	One-half bundle-solid brown, one-half bundle-solid yellow	Carbon steel bar area
6 Bundles (tagged M1020)	No color code	Carbon steel bar area
130, 7/8 x 8" bar (tagged A36)	No color code	Stocked with M1020 material
1 Bundle 7/16" round (tagged 203EZ)	One-half bundle-solid red, one-half bundle-solid black	Stainless compartment
33 tubes 2½ x 18 ga. A269T304	No color code	Stainless tube storage location
52 pcs 1½" round B408	Marked white (Ni 800)	Nickel storage area
16 pcs 1" x 8" bar (tagged A36)	Marked white (Cor-Ten)	Carbon steel bar area
10 bundles 1" x 4" bar	No color code	Carbon steel bar area

It is recognized that a considerable quantity of material is identified both by ink marking and hard (stamp) marking of heat numbers and receiving ticket numbers, in addition to the color code scheme. Although these steps add to the ability to identify material with certainty, the workman in the shop places his reliance on the color code when selecting material to fill an order. (85-01-04)

2. Control of Measuring and Test Equipment

Control of measuring and test equipment at Joseph T. Ryerson & Son, Inc. is governed by Policy and Procedure Bulletin (P.&P.B) 18.220.02. This document establishes the procedure for testing the accuracy of measuring devices and test instruments at specified time intervals. A review of P.&P.B 18.220.02 revealed that it is thorough and encompasses all those tools and devices used by Ryerson employees in their daily activities, with two exceptions. The procedure does not establish a specific maximum deviation that is acceptable for continued use of the instrument, and the procedure does not provide for identification of test equipment such that traceability of the calibration status of each piece of equipment is established.

The inspector found a significant failure by Ryerson personnel to comply with the requirements of P.&P.B 18.220.02. This failure to comply was due, in part, to the fact that many persons responsible for using this equipment were not aware of the procedural requirements in regards to the calibration schedule, equipment covered and records retention.

The specific areas examined and the results of that examination are as follows:

QA Department Equipment

Device	Cal. Freq	Cal. Not Done
Tensile Tester S/N R48071	Yearly	1980, 1981
Master Tape (50') (Lufkin)	Yearly	Not performed since purchase (1977)

Device	Cal. Freq	Cal. Not Done
Wilson Hardness Tester S/N 1413	Yearly	1977, 1979, 1980, 1982, 1984
Brinell Tester S/N 75105	Yearly	1980, 1981
Starrett Micrometer Set S/N 436 CPSZ	Yearly	1979

North Plant Equipment

Device	Cal. Freq	Cal. Not Done
Olsen Ductility Tester	Yearly	No records available except for 1980
Hardness Tester S/N 1SS430	Yearly	1982-85
Hardness Tester S/N 1JR954	Yearly	1982-85
Tapelines, Slide Calipers and Steel Squares	Monthly	No records available
Micrometer and Vernier	Daily	No records available

Center Plant Equipment

Device	Cal. Frequency	Cal. Not Done
Master Block Set	Yearly	All years since purchase except 1985 (see note)
Tapelines, Slide Calipers, Steel Squares	Monthly	All months prior to 1985 (no records available)
Micrometer and Vernier Calipers	Daily	All months prior to 1984

South Plant Equipment

Device	Cal. Freq.	Cal. Not Done
Starrett End Measuring Rods (used for caliper checking)	No frequency stated	No certificate for any period
Tapelines, Slide Calipers Steel Squares	Monthly	No records availble prior to August 5, 1985
Micrometer and Vernier Calipers	Daily	No records available prior to August 5, 1985

East Plant (Plate) Equipment

Device	Cal. Freq.	Cal. Not Done
Ultrasonic Test Machine (2 units)	Before each test	No records for any period
Ultrasonic Test Machine (2 units)	Yearly	No records for any calibration except unit S/N 1107/T39, which was calibrated on 11/5/84

West Plant Equipment

Device	Cal. Freq.	Cal. Not Done
Tapelines, Slide Calipers	Monthly	No records available prior to 1985
Micrometer and Vernier Calipers	Daily	No records available prior to 1985

The master block set in use in the Center Plant was sent to a calibration laboratory on August 26, 1985 for accuracy testing. This is the only record available for the master block set. The set is estimated to have been in use for five to seven years. The calibration laboratory recommended that 28 blocks of the 75 piece set be replaced due to excessive error. Joseph T. Ryerson & Son, Inc. replaced two of the most used blocks (one inch and three inch).

Ryerson Policy and Procedure Bulletin 18.220.02 also provides specific instructions regarding the steps to be taken when discrepancies beyond acceptable limits are found at the time of calibration. Those instructions specify removal of the instruments from active use until corrective action is accomplished. Also, material which has been previously examined with such discrepant instruments shall be reviewed to establish whether the applicable requirements have been met.

During the examination of calibration records, it was discovered that three instruments had been noted by the calibration laboratory as follows: "Above unit was received in a damaged and non-useable condition and could not be calibrated before repair." Those instruments were:

a. Starrett 3"-4" Outside Micrometer, S/N 236 RL-4
b. Brown & Sharpe 4"-5" Outside Micrometer, S/N 599-50-19
c. Starrett Mul-T-Anvil Micrometer, S/N 220

No evaluation or corrective action was indicated as having been taken by Ryerson personnel as a result of this notification. (85-01-06)

3. Control of Nonconforming Material

Section 5.3 of the Ryerson QA Manual describes the actions to be taken when nonconforming material is found at the time of receipt inspection. Additional guidance is provided by Ryerson Policy and Procedure Bulletin 12.200.03.

The inspector noted that Receiving Ticket No. 40450 and attendant Discrepancy Report No. 190213 for 27 pieces of "bent and twisted" steel were not documented as required. In addition, Receiving Ticket No. 56195 and associated

Discrepancy Report No. 196031 failed to document the disposition of nonconforming tubular material received from a supplier. No evidence could be found to establish the action taken to resolve these matters.

In conversations with warehouse employees, the inspector was advised that control of material on shifts other than the day shift is a problem. Inadequate shift instructions and the absence of a clear flow chart to remind second shift workers of the correct steps to process such paperwork were cited as problems. In the case of Receiving Ticket No. 56195 cited above, the material is believed to have been sold "as is". (85-01-10)

ORGANIZATION: NATIONAL TECHNICAL SERVICES
HARTWOOD, VIRGINIA

REPORT NO.: 99900914/85-02	INSPECTION DATE(S): 10/30-31/85- 11/01/85	INSPECTION ON-SITE HOURS: 37

CORRESPONDENCE ADDRESS: National Technical Services
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Division Vice President
State Route 748, Box 38
Hartwood, Virginia 22471

ORGANIZATIONAL CONTACT: Mr. W. C. Hartman, Quality Control Manager
TELEPHONE NUMBER: (703) 752-5300

PRINCIPAL PRODUCT: Testing Laboratory

NUCLEAR INDUSTRY ACTIVITY: Approximately 15 percent of the National Technical Systems (NTS) total business (dollar value) is a result of testing of equipment for the nuclear power industry.

ASSIGNED INSPECTOR: Uldis Potapovs 1-3-86
for R. N. Moist, Equip. Qual. Inspec. Section (EQIS) Date

OTHER INSPECTOR(S): M. J. Jacobus, Sandia National Laboratories

APPROVED BY: Uldis Potapovs 1-3-86
U. Potapovs, Chief, EQIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection consisted of: (1) a technical evaluation of equipment qualification (EQ) test activities for safety-related equipment; (2) verification of implementation of corrective action (CA) on the nonconformances identified in NRC Inspection Report No. 99900914/85-01 (3) verification of implementation of the quality assurance (QA) program; and (4) witnessing Post Loca Functional Tests.

PLANT SITE APPLICABILITY: Palisades 50-255

A. Violations:

None.

B. Nonconformances:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and paragraph 2.5.1 and 2.6.1 of the Quality Control Manual (QCM), (a) The job instruction package for Master Job Order (MJO) 558-1711 did not contain a procedure describing the detailed operation of the steam supply system used for qualification testing and (b) The circuit used for monitoring the Resistance Temperature Detector (RTD) output signals during the LOCA test was not described in Test Procedure (TP) 558-1711.

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and paragraph 2.5.1 and 4.2.1(e) of the QCM, only one measurement was recorded on the data sheet by the test technician when two or more readings were required by TP 558-1711 to be recorded for the Post LOCA Triple point calibrations functional test.

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and paragraph 10.3.2 of the QCM, NTS calibration label was not affixed to a stripchart recorder when the outside laboratories calibration interval was different from NTS official equipment list calibration interval.

4. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Appendix A, page A-27 of the QCM, the provisions of 10 CFR Part 21 were not specified by reference on the Purchase Order 631-054814C.

5. Contrary to Criterion XVII of Appendix B to 10 CFR Part 50 and Appendix A, page A-23 of the QCM; (a) NTS was unable to identify the source of the activation energy value for ethylene propylene as stated in TP 558-1711 for MJO 558-1711, (b) there was no documented record showing the chemical spray solution was mixed as prescribed in the test data calculations and (c) two blue traces that measured temperature parameters during LOCA testing for MJO 558-1711 overlapped on the stripchart preventing clear identification of which trace was which.

C. Unresolved Items:

None.

D. Status of Previous Inspection Findings:

1. (Closed) Nonconformance (85-01, Item B.1): The test technician had not initialed and dated corrections to recorded data on data sheets for MJO 558-1572. The NRC Inspector reviewed several data sheets from two MJO Files to verify that corrections to recorded data are now being initialed by the test technician.

2. (Closed) Nonconformance (85-01, Item B.2): The test technician had not initialed the appropriate column of Job Traveler Forms (JTFs) for MJO's 558-1720, 558-1686, and 557-1382 when it had been ascertained that a test had been conducted and completed in accordance with applicable specifications. The NRC Inspector reviewed JTFs for two MJO Files to verify that the appropriate column of the JTF are now being initialed by the test technician after test had been conducted and completed.

3. (Closed) Nonconformance (85-01, Item B.3): The audit report for the May 1984 corporate quality internal audit was not issued within thirty days of the conclusion of the audit, did not request a response date for corrective action, and there was no documented evidence to indicate that the required follow-up had been performed by the lead auditor. The NRC Inspector verified that the 1985 corporate audit report was issued in a timely manner and that a follow-up was made by lead auditor of the corrective actions taken in response to the 1984 corporate audit.

4. (Closed) Nonconformance (85-01, Item B.4): NTS test plans for test procedures did not list test equipment requirements for MJO 558-1572 or list all test equipment accuracies for MJOs 558-1572 and 557-1382. The NRC Inspector reviewed two MJO Files and verified that test equipment requirements and their accuracies are now listed as an attachment to the test procedure.

E. Other Findings or Comments:

1. Observation of testing activities MJO-558-1711 - Post-LOCA functional testing was being conducted for Consumer Power on three Rosemount RTDs for use in Palisades Nuclear Power Plant. The NRC Inspector and Sandia consultant (NRC inspection team) witnessed portions of the opening of the chamber, removal of the specimens and performance of post-LOCA functional tests. The post-LOCA functional tests consisted of a three-point calibration check (triple point of water, boiling point of water corrected for atmospheric pressure, and the freezing point of lead). During this witnessing, the NRC inspection team identified the nonconformance described in paragraph B.2.

   With respect to B.2, the NRC inspection team reviewed selected data from the post LOCA functionals as well as data from all previous functional tests. No problems were identified with this data. However, the functional tests prescribed in TP 558-1711 were not followed precisely by the test technicians. TP 558-1711 required that measurements during the three-point calibrations should be taken every minute until two consecutive readings differed by 0.02 ohms or less, indicating stabilization. Only one reading was recorded at each of the calibration points. However, interviews with the test technicians indicated that they always waited for a stable reading before taking the data.

2. Technical Evaluation - The NRC inspection team performed a technical evaluation of MJO 558-1711 for qualification testing of Rosemont RTDs. The NRC inspection team reviewed the EQ process prescribed in TP 558-1711 and reviewed test results, including the bases for accelerated thermal aging and radiation and verified calculations. TP 558-1711 and related engineering documents were examined to verify the following:

   a. Adequate test instrumentation and their accuracies were described and used to meet the requiremens of IEEE-STD-323/1974.

   b. Equipment interfaces were addressed.

   c. Test acceptance criteria were established as described in the test specification or in the design engineering documents, such as calculations and engineering letters to meet the requirements of IEEE-STD-323/1974.

d. Same equipment was used for all phases of testing.

e. Environmental conditions were established and described (e.g., pressure and temperature profiles, and thermal aging factors were consistent with those outlined in the test specifications or test plan).

f. Test results were adequately reduced and evaluated against established acceptance criteria described in customer test specifications or purchase orders.

g. All prerequisites for the given tests as outlined in the test specification had been met.

h. Test equipment included a description of all materials, parts, and subcomponents.

i. Notice of Deviation Forms were properly documented.

j. Appropriate margins were applied.

k. 10 CFR Part 21 imposed on procurement documents.

During the above review and evaluation, the NRC inspection team identified the nonconformances in paragraph B.1.a, B.1.b, B.4, B.5.a, B.5.b, B.5.c.

With respect to B.1.a, the NRC inspection team requested a copy of NTS's detailed operating procedure for the steam supply system used for EQ testing. Interviews with the Nuclear EQ Manager disclosed that no procedure existed.

With respect to B.1.b, the NRC inspection team visually examined a resistance bridge arrangement which was used for monitoring the RTD outputs during LOCA testing. The outputs of the RTDs were compared at two points to the chamber temperature and the reference temperature which was recorded by a stripchart recorder. The method of connection of the RTDs through the resistance bridge arrangement to the stripchart recorder was not documented in TP 558-1711.

With respect to B.4, the NRC inspection team reviewed Purchase Order 631-054814C for stress analysis and determined that the provisions of 10 CFR Part 21 were not specified by reference on the purchase order as required by Appendix A, page 27 of the QCM. Interviews with the Nuclear Manager indicated however, that 10 CFR Part 21 was included as an attachment to the purchase order. A copy of the attachment was obtained from purchasing and provided to the NRC inspection team.

With respect to B.5.a, the NPC inspection team determined that NTS did not provide any references or documentation supporting their choices of activation energies. Specific concern was expressed by the NRC inspection team for a choice of 1.22 ev for an activation energy for ethylene propylene. (EPRI) Electric Power Research Institute data indicates a range of activation energies of 0.70-1.28 ev for various forms of ethylene propylene with a conservative choice of 0.90-0.95 ev based on specifics given for the different formulations.

With respect to B.5.b, the NRC inspection team reviewed a data calculation sheet showing the amount of each chemical used in mixing the chemical spray solution. The NRC inspection team requested documentation which would verify that the chemical spray solution used during design basis event testing was mixed as prescribed in the data calculation sheet. The Nuclear Manager was unable to provide the requested documentation.

With respect to B.5.c, the NRC inspection team reviewed a stripchart recording taken during the LOCA test. The stripchart recorder used identical blue pens which recorded two temperature measurements. It was determined by the NRC inspection team that the two blue traces that overlapped on the stripchart prevented clear identification of which trace was which.

3. <u>Calibration of Test Equipment</u> - The NRC inspection team inspected several pieces of measuring and test equipment used for data acquisition during the LOCA test of the RTDs. The NRC inspection team observed that the calibration interval established on the calibration label of the outside laboratory, was different from the calibration interval established in NTS's official equipment list for a Gould stripchart recorder. This nonconformance is identified in paragraph B.3. NTS responsible personnel corrected this nonconformance on the spot by placing a NTS calibration label on the stripchart recorder with the proper calibration interval identified.

ORGANIZATION: NUCLEAR ENERGY SERVICES
DANBURY, CONNECTICUT

REPORT NO.: 99900762/85-01	INSPECTION DATE(S): 9/16-19/85	INSPECTION ON-SITE HOURS: 66

CORRESPONDENCE ADDRESS: Nuclear Energy Services
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ORGANIZATIONAL CONTACT: C. E. Anderson, Quality Assurance Manager
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PRINCIPAL PRODUCT: Engineering Services for the Nuclear Power Industry

NUCLEAR INDUSTRY ACTIVITY: 100% of NES' activities

ASSIGNED INSPECTOR: RP Correia 11-6-85
R. P. Correia, Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): R. P. McIntyre, SPIS
A. W. DuBouchet, Consultant

APPROVED BY: John W. Craig 12/6/85
John W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21, 10 CFR Part 50, Appendix B

B. SCOPE: The inspection consisted of an evaluation of quality assurance and engineering activities related to the design, procurement, manufacturing, testing and installation of steam generator nozzle dams for Ft. Calhoun.

PLANT SITE APPLICABILITY: Ft. Calhoun (50-285)

ORGANIZATION: NUCLEAR ENERGY SERVICES
DANBURY, CONNECTICUT

REPORT NO.: 99900762/85-01	INSPECTION RESULTS:	PAGE 2 of 1C

A. Violations

Contrary to the requirements of 10 CFR Part 21, Section 21.31, NES did not specify on purchase order No. N57605 that the provisions of 10 CFR Part 21 apply to Reno Machining Company who were to procure material to be used for fabricating clamps used in the steam generator nozzle dam assemblies. (85-01-01)

B. Nonconformances

1. Contrary to 10 CFR Part 50, Appendix B, Criterion III, NES design calculation for the steam generator nozzle dam assembly did not encompass all aspects of the following: a) materials selection and suitability, b) diaphram/seal sub-assembly stress analysis, c) all possible loading conditions (hydraulic, pneumatic, and seismic), d) the correct subsection of the ASME III code per contractural commitments, and e) consideration for dimensional tolerances. (85-01-02)

2. Contrary to 10 CFR Part 50, Appendix B, criterion XVI, NES Document No. 80A9010 "Computer Code Documentation Control Procedure" did not have provisions for handling computer code error reports. Computer code error reports received from vendors supplying computer code services were not promptly identified and corrective action to assure that conditions adverse to quality for past and present safety related components were not determined, documented and reported to appropriate levels of management. (85-01-03)

3. Contrary to 10 CFR Part 50, Appendix B, Criterion XVIII, NES had used Control Data Corporation's (CDC) services for computer codes used for safety related component analyses and had failed to comply with criteria in the aforementioned section of Appendix B by: (85-01-04)

    a) NES had not performed an audit of CDC to verify it's compliance with all aspects of the quality assurance program.

    b) NES did not have a planned audit of CDC scheduled.

    c) CDC was not on the NES approved vendor's list (dated 9-16-85)

C. <u>Unresolved Items</u>

1. Based upon observations made during steam generator nozzle dam testing, questions were raised concerning the adequacy of test equipment, procedure and classification of hardware. (85-01-05)

D. <u>Status of Previous Inspection Findings</u>

There were no findings on the previous inspection.

E. <u>Other Findings or Comments</u>

1. <u>Documentation Review</u>

The NRC Inspectors reviewed the following purchase orders used to procure services and materials for the steam generator nozzle dam assemblies:

P.O. No.	From	To	For	Date
7234	OPPD	NES	Steam Generator Nozzle Dams	05/16/85
7234 (sup. #1)	OPPD	NES	Steam Generator Nozzle Dams	07/15/85
7234 (sup. #2)	OPPD	NES	Steam Generator Nozzle Dams	08/12/85
N 51646	QualCorp (NES)	Presray Corp	Diaphram Assemblies	06/26/85
N 51624	QualCorp (NES)	Quality Castings	Dam Castings	06/03/85
N 57605	QualCorp (NES)	RENO Mach. Co.	Fabrication of Dams	09/05/85

The NRC Inspectors noted that the applicability of 10 CFR Part 21 was not indicated or OPPD's P.O. No. 7234 and supplements No. 1 & No. 2. However, P.O. No. 7234, supplement No. 2, page 2 of 2 did state in part, "This material/service is nuclear safety related..." NES P.O.'s N 51646 and N 51626 did indicate the applicability of 10 CFR Part 21 to the Presray Corporation for the diaphram assemblies and the Quality Castings for the dam castings. NES P.O. N 57605 to Reno Machining Company for the fabrication of clamp assemblies and the machining of the dam castings did not specify the applicability of 10 CFR Part 21. This same purchase order did specify that material certifications for material procured by Reno Machining Company would be required.

One violation, 85-01-01, was identified in this area of the inspection.

Design basis documentation used for the stress analysis of the steam generator nozzle dams was reviewed. OPPD's contract No. 1453, Section H, dated March 12, 1985, and NES technical proposal 8560-103, section 2.4, dated April 1985, required that a stress analysis be performed in accordance with the guidelines of section III, NB Class 1 of the ASME Boiler and Pressure Vessel Code. The NES calculation, "The structural design calcs for Fort Calhoun dams," project 5273, task 140, dated June 3, 1985, referenced ASME section III, NF, 1983 with addenda. Subsection NF addresses component supports, however the steam generator nozzle dams act as a reactor coolant system pressure boundary element governed by subsection NB of the ASME code.

The calculation did not address the acceptability of the aluminum/stainless steel interface between the aluminum clamps and the stainless steel bolts used to restrain the dam. The clamp analysis also did not determine the maximum stresses of the bolt hole threads but rather calculated the allowable load. OPPD contract No. 1453 specified that maximum stress and minimum safety margin would be indicated. Also, the load imposed by the bolt, which is to be torqued during installation of the dam, was not addressed in the clamp analysis. The aluminum clamp body, detailed on page 7 of the calculation, depicts a curved shape design. However, the analysis employed a straight beam theory and no consideration was given to the effects of holes and shape factors in the stresses in the clamp.

The NES stress analysis of the dam assembly castings was performed by evaluating portions of a cross section as independent structural elements rather than as a whole, integral piece. Only the loading imposed by the hydrostatic pressure was used to determine the local bending effects on the casting. The loads imposed by the seal pressure and the clamping forces, the fabrication tolerance for the casting thickness and the complex stresses in the casting flange were not addressed in the analysis.

The calculation also failed to include a statement of the method used for the analysis, an evaluation of the materials selection and suitability, any analysis of the diaphram/seal assembly portions of the dams, and seismic loading conditions. During the inspection, NES committed to perform a complete, three-dimensional computer stress analysis of the nozzle dam and to include in the calculation all items noted as deficiencies above.

Nonconformance 85-01-02 was identified in this area of the inspection.

2. <u>Computer Code Error Handling Procedures</u>

The NRC inspector reviewed Document No. 80A9010, "Computer Code Documentation Control Procedure," and found that no procedures exist for the handling of computer code errors received from computer service bureaus such as Control Data Corporation (CDC) or for reported errors on internally developed computer codes.

The original NES service contract with CDC did not impose the requirements of 10 CFR Part 21 on CDC. On July 10, 1984, NES requested CDC to amend their contract to include the provisions of 10 CFR Part 21 and Part 50, Appendix B and to send notification of all errors reported for four specified computer codes; ANSYS, STARDYNE, PIPESD, and UNIPLOT. CDC accepted the NES proposed amendment by letter on October 3, 1984.

On October 18, 1984, NES received a complete list of all error reports known, to date, for the specified codes. Up to this point, NES had received only a limited number of error reports from CDC. NES is in the process of revising the Computer Code Documentation Control Procedure to include procedures for the handling of computer code errors and are also reviewing computer code error reports received from CDC for their impact on past and present safety-related design analyses. This review is scheduled to be completed September 30, 1985.

The NRC inspectors reviewed the NES Approved Vendors List and noted that CDC was not on the list. It was further determined that NES has not performed any audits of CDC in the past nor was an audit scheduled for the near future. During the inspection, NES' QA Manager committed to plan and perform a quality assurance audit of Control Data Corporation (CDC).

Nonconformances 85-01-03 and 85-01-04 were identified in this area of the inspection.

3. Other Information

The NRC inspectors toured the Reno Machining Company of Newington, Connecticut with representatives of NES on September 19, 1981, to observe current activities on the machining of nozzle dam components and the general operation of the facility. The inspectors observed several machining operations including the milling of the steam generator nozzle dam clamps. Reno Machining Company makes extensive use of numerical control machine tools and has an in house computer department in which programs are used to generate control tapes which direct the machining processes. The inspection and calibration area was visited to observe calibration equipment, records and the general layout. Additionally, an interview with the quality control manager was also conducted to discuss Reno's quality assurance and control program and commitments.

The NRC inspector observed testing of the steam generator nozzle dam assembly conducted at the Reno Machining Company on October 3, 1985.

A 32 in. diameter steam generator nozzle dam assembly for the Ft. Calhoun Station and a control console were prepared for testing as part of the contract between Omaha Public Power District and NES. The dam assembly was mounted to a test fixture which was fabricated to simulate the steam generator nozzle. It was then connected to the control console which is used to control and monitor air supplies to the inflatable seals annulus components of the nozzle dam diaphram assembly. The seals were then inflated to 60 psig and maintained at that pressure for approximately 20 minutes without any leakage detected. The test fixture was then filled with water, vented, and pressurized to approximately 15 psig and held at that pressure for 20 minutes, again without any leakage detected. The hydrostatic pressure was then increased to the maximum required test pressure, 25 psig, held at that pressure for 20 minutes and no leakage was detected.

As a means of determining the adequacy of the redundancy of the sealing system, the wet seal (i.e., seal in first line contact with the water) was deflated. A leak was detected: the air supply connection for the annulus section of the diaphram did not maintain its bond and water leaked through the diaphram, filled the space between the diaphram and the aluminum dam support segments and subsequently leaked through the seams between the aluminum dam support segments. As a matter of course, the secondary or dry seal was deflated to determine the integrity of the passive seal or the portion of the diaphram which is compressed between the nozzle and the aluminum dam support assembly. Because of the aforementioned leak of the air supply connection, it was difficult to determine visually if the passive seal did in fact not leak. This concluded the testing of the 32 in. nozzle dam assembly.

NES decided that all diaphrams for both the 32 in. and 24 in. nozzle dams would be returned to the manufacturer Presray Corp., for revulcanizing and that a retest of all dam assemblies would be performed.

The following items were noted during the nozzle dam test:

1) When the 32 in. dam assembly was mounted to the test fixture, three people performed all necessary procedures and no timing of the mounting was recorded. The actual assembly of the dam inside the steam generators at Fort Calhoun will be performed by one (1) person and must be completed in eight (8) minutes or less.

2) The control used to control and monitor the nozzle dam test did not have its gages calibrated.

3) The air supply lines, connectors, check valves and clamps were purchased by NES as commercial grade. Also, the control console assembly (gages, instrumentation, hardware etc.) was considered non-safety-related. No documentation for the selection and use of these items was available.

Based on the above, the test procedure, monitoring and regulating equipment and the commercial grade hardware used on the nozzle dam assembly do not appear to meet the requirements of the Quality Assurance Criteria of 10 CFR Part 50, Appendix B for a safety-related component.

Unresolved item 85-01-05 was identified in this area.

REPORT NO.: 99900762/85-01	INSPECTION RESULTS:	PAGE 8 of 10

F. Persons Contacted

Louis J. Zezza, Mgr. Des/Eng., Eng. Prods, NES
Craig E. Anderson, Quality Assurance Mgr., NES
W. J. Manion, President of NES,
Mary-Ellen Alling, Quality Assurance Engineer, NES
H. J. Larson, Sr. VP, NES
Mark Weiner, Project Manager, NES
George Hamilton, Sr. VP, NES
Louis J. Barbieri, Sr. QA Engr., NES
J. Shah, Engineer, NES
R. D. Stefano, Mgr. OPS. Eng., NES
T. Kettler, Systems Eng., NES
Jack Atashian, Q.C.M., Reno Machine Co., Inc.
Mark Cchialini, Production Manager, Reno Machine Co., Inc.
Arnold Gundersen, VP Engineering, NES
Albert Uziel, Gen. Mgr. Eng'd ProD's, NES

G. Documents Examined

1. Q. A. Manual, No. 80A9002, Rev. 7, dated 11/23/83, NES Division/Engr. Operations/QA Manual.

2. Procedure No. 80A9007, Rev. 7, 06/15/82, NES Procurement Control Procedure.

3. Procedure No. 80A9010, Rev. 3, dated 12/01/83, Computer Code Documentation Control PROC.

4. Procedure No. 80A9022, Rev. 8, dated 06/02/84, QA Audit Procedure.

5. Procedure No. 80A9004, Rev. 5, dated 12/01/83, Calculation Notebook Procedure.

6. P.O. No. N 51646, dated 06/26/85, for diaphram assemblies/NES to Presray Corp/Pawling, NY.

7. P.O. No. N 57605, dated 09/05/85, to fabricate & deliver S.G. nozzle dams/NES to Reno Machining Co./Newington, CT.

8. P.O. N 51624, dated 06/03/85, to fabricate S.G.Nozzle Dams castings/NES to Quality Castings/Greensboro, NC.

ORGANIZATION: NUCLEAR ENERGY SERVICES
DANBURY, CONNECTICUT

REPORT NO.: 99900762/85-01	INSPECTION RESULTS:	PAGE 9 of 10

9. P.O. 7234, dated 05/16/85, steam gen nozzle dams and mock up for Ft. Calhoun/Omaha Public Power District (OPPD) to NES.

10. Letter, MSM-809, dated 05/30/85, NES to OPPD/Acceptance of order from OPPD for SG nozzle dams.

11. List, NQA-1923, Rev. 2, dated 09/16/85, Approved Vendors List for category 1 & 2.

12. Report, QAA-320, dated 06/07/85, Audit report/vendors survey of Quality Castings.

13. Internal Memo, GES-2246, 09/16/85, Computer code error report handling meeting.

14. Letter, GES-2131, dated 07/10/85, NES to CDC for the modification of services contract to assure NES receives errors for safety related codes.

15. Internal Memo, 85-7, dated 04/29/85, meeting minutes on computer code error handling.

16. Letter, EO-144110-VI, dated 10/03/84, CDC to NES for the amendment to CDC agreement for addition of 10 CFR 50 Appendix B & 10 CFR Part 21.

17. LOG, computer code usage log of Ansys, Stardyen and Uniplot.

18. Report, dated 08/01/85, CDC to NES software problem report for Ansys errors.

19. List, dated 07/30/85, Administrative Procedures.

20. Contract P.O./SPEC, No. 1453, dated 03/12/85, OPPD contract 1453.

21. Specification No. 8560-103, dated 04/85, NES proposal to the OPPD for the supply...for Ft. Calhoun station contract No. 1453.

22. Specification No. 80A9503, Rev.1, 07/09/81, General Specification For The Fabrication of Safety-Related Special Tools and Equipment For Nuclear Applications.

23. Q.C. Manual, dated 06/07/83, Q.C. manual for the Reno Machine Co., Inc.

24. Q.A.Manual, QAM-2, Rev. D, dated 05/15/84, Presray QAM.

25. NES DWG 83E2364,5, noz. dam. diaphram details & ass'y.

26. Design calculation for Ft. Calhoun st. gen. nozzle dams, project 5273, task 140.

27. Code, dated 07/01/83, sect.III Rules for Construction of Nuclear Power Plant Components, Division 1-sub.-sec. NB, Class 1 components.

28. Catalog, 1/85-SM, dated 1985, Presray-seal catalog.

29. Catalog, 1979 cat'l, Presray-seal.

ORGANIZATION: NUS CORPORATION
GAITHERSBURG, MARYLAND

REPORT NO.: 99900516/85-01	INSPECTION DATE(S): 11/18-20//85	INSPECTION ON-SITE HOURS: 45

CORRESPONDENCE ADDRESS: NUS Corporation
ATTN: Mr. Donald L. Couchman
Senior Vice President,
Administrative Services
910 Clopper Road
Gaithersburg, Maryland 20878

ORGANIZATIONAL CONTACT: Mr. M. R. Booska, Director, Corp. QA-Division Operations
TELEPHONE NUMBER: 301-258-6000

PRINCIPAL PRODUCT: Engineering Consulting Activities

NUCLEAR INDUSTRY ACTIVITY: NUS provides training and technical consulting services to the nuclear industry.

ASSIGNED INSPECTOR: ______________________ 6 Feb 86
E. H. Trottier, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): Jeffrey B. Jacobson, RIS

APPROVED BY: ______________________ 6 Feb 86
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was conducted to review the NUS quality assurance program with emphasis on divisions that provide technical training to nuclear power plant personnel.

PLANT SITE APPLICABILITY: Perry 1/2 (50-440, 441).

ORGANIZATION: NUS CORPORATION
GAITHERSBURG, MARYLAND

REPORT NO.: 99900516/85-01	INSPECTION RESULTS: 11/18-20/85	PAGE 2 of 4

A. Inspection Issues

In August 1985, the Nuclear Regulatory Commission learned that a portion of a containment integrated leak rate testing (ILRT) course provided by a nuclear consulting company, contained some material that appeared to suggest or condone practices that could mislead NRC inspectors. This inspection sought to establish whether NUS Corporation includes information of this type in training programs it offers. This inspection was also performed to establish whether NUS Corporation is aware of their responsibilities under 10 CFR Part 21.

B. Inspection Findings

1. Violations

   a. None.

2. Nonconformances

   a. Contrary to Criterion III of Appendix B to 10 CFR Part 50 of the NUS Enviromental Services Division Quality Assurance Manual, changes were made to NUS Design Control Document ASD-5492-3 without proper review or control. Changes were made in the field to this document by Tracer Technology (a subcontractor), without prior approval or authorization from proper NUS personnel. The changes were, however, documented in a letter dated October 21, 1985, from Tracer Technology to NUS, and pertained to Tracer Technology having to substitute different units in a formula used to calculate atmospheric dispersion characteristics at the Perry Nuclear Power Station.

   b. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Chapter 5 of the Corporate QA Policy Manual, semiannual reports on the status of each operating unit's QA Program have not been prepared. The inspector noted that one report was prepared in 1982 and 1984, but no report was found for 1983. In addition, no report for 1985 had been issued as of the date of this inspection. (However, the 1985 report was in preparation.)

      In discussions with the NUS Corporate QA Staff, the inspector was advised that either the requirement for such semiannual QA status reports will change, or the content and format of the reports will be modified to enhance preparation by computer (charts, tables, and graphs).

C. Other Findings and Comments

1. Part 21 and 10 CFR 50 Appendix B Programs.

Various divisions within NUS were inspected for compliance with 10 CFR Part 21 and 10 CFR Part 50 Appendix B requirements. The divisions inspected had adequate quality assurance programs including all necessary requirements for evaluating and reporting Part 21 deficiencies. Although NUS has not reported any Part 21 type deficiencies recently, in-house procedures have been invoked to evaluate potential Part 21 deficiencies. In each case reviewed by the inspector, the NUS determination of Part 21 reportability appeared to be adequate.

2. Failure to Properly Pass Down Part 21 Requirements

On July 13, 1984, NUS signed a contract with Control Data for Control Data to provide information concerning errors discovered in safety related computer programs supplied for use by NUS. The term of this contract was for six months and has since expired. Control Data is therefore not contractually obligated to supply NUS information concerning errors discovered in computer programs that may have been used in analyzing safety related systems. NUS acknowledged this discrepancy and indicated it would rewrite their contract with Control Data. Documentation was provided showing that although not contractually required, error reports were still being received from Control Data.

3. Training Divisions

All site specific technical training of nuclear power plant personnel by NUS is developed and conducted by NUS employees at each particular training site. No significant examples of site specific type training materials were available at NUS for review. Non specific, generic type training materials were reviewed and found to contain no material that would be considered inappropriate.

4. Internal Audits

The inspector noted that internal audits of the NUS Consulting Divisions have not been consistently conducted as required by the Division QA Manual. In discussions with the Division QA staff, the inspector was advised that this issue was the subject of a previous Division or Corporate QA audit finding. Since recent internal audits (since 1983) of this Division had been conducted as required by the Division QA Manual, corrective action for this finding appears to be effective.

ORGANIZATION: PACIFIC SCIENTIFIC COMPANY
ANAHEIM, CALIFORNIA

REPORT NO.: 99900255/85-01	INSPECTION DATE(S): 8/12-13/85	INSPECTION ON-SITE HOURS: 17

CORRESPONDENCE ADDRESS: Pacific Scientific Company
Kin-Tech Division
ATTN: Mr. Edward R. Thomsen, Manager
Quality Systems and Services
1346 S. State College Boulevard
Anaheim, California 92803

ORGANIZATIONAL CONTACT: Mr. Edward R. Thomsen, QA Manager
TELEPHONE NUMBER: (714) 774-5217

PRINCIPAL PRODUCT: Inertially operated restraint systems and components.

NUCLEAR INDUSTRY ACTIVITY: Mechanical Shock Suppressors (snubbers) utilized for seismic restraint of piping systems.

ASSIGNED INSPECTOR: Robert L Pettis Jr. 12-4-85
R. L. Pettis, Jr., Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): E. Trottier, Reactive Inspection Section (RIS)
M. Subudhi, Brookhaven National Lab

APPROVED BY: John W Craig 12/5/85
John W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review of technical information and QA procedures relative to snubber testing, design, manufacturing, and maintenance activities.

PLANT SITE APPLICABILITY: Multiple including Perry Unit 1 (50-440).

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED SAFETY ISSUES:

None.

D. OTHER FINDINGS OR COMMENTS:

1. Background

The Pacific Scientific Company (PSC), Kin-Tech Division, is the sole manufacturer of inertially operating mechanical snubbers used for seismic support of piping systems in nuclear power plants. As a result of the overall decline of new nuclear business, there was no production activity to examine at the PSC facility. Consequently, the inspection focused primarily on snubber testing and maintenance services and PSC's relationship with utilities maintaining already installed PSC snubbers.

2. Snubber Repair and Test Services

a. Repair

PSC snubber repair and testing activities for various size Pacific Scientific Arrestor (PSA) units are performed in accordance with the following instruction manuals:

PSC Model #	Rated Load	PSC Document #
PSA-1/4, 1/2	350, 650 lbs.	PS 192
PSA-1, 3, 10	1500, 6000, 15000	PS 193
PSA-35, 100	50000, 120000	PS 194

These documents included detailed descriptions of the snubber repair procedure which also included: spare parts procurement; functional testing instructions to an acceleration level of 0.02g for both extension and retraction (used to verify restraining action); lost motion test for free play of all assembled parts; Dead Band (limited to 0.04 inches); and Drag/Breakaway Force Test (which verifies the minimum force applied to extend or retract the snubber, and the force required to maintain movement at a constant velocity).

The above tests are performed on PSC's Shock Arrestor Test Stand, Model 524, which is capable of testing all seven models of PSC snubbers. During the plant visit, the NRC inspectors observed an actual test of a PSA-100 model, undergoing testing to demonstrate activation level.

b. <u>Testing</u>

The Shock Arrestor Test Stand previously mentioned is designed exclusively by PSC and accurately measures and records all functional parameters required to demonstrate snubber operability. As of this inspection, repair and testing for PSC is performed at PSC corporate headquarters in Anaheim, California. Westinghouse Electric's Spartanburg Service Center is the only other facility, authorized by PSC, to perform similar repair and functional testing on PSC snubbers, especially those in high radiation areas. Over 50,000 PSC snubbers are now in operation with approximately 2000 in the 35-100,000 pound capacity range, or commonly referred to as "Large Bore."

According to PSC, clients are not supplied spare parts for performing site repair without first having sent representatives to PSC training class held in Anaheim. According to PSC's records as of August 13, 1985, 17 customers (made up primarily of vendors and utilities) have completed the course which covers snubber design, operation, maintenance, testing and repair. Graduates also receive a copy of the entire training session on video tape.

The NRC inspectors observed a demonstration of the PSC Validator, an in-situ snubber tester, which verifies both activation level and breakaway force functional parameters. This device, unlike the Model 524 test stand, is a portable, self-powered, and lightweight device suitable to test the snubber while in the installed position.

3. Snubber Qualification

The following qualification testing reports were reviewed by the NRC inspector. These reports documented the ability of PSC snubbers to meet their qualification test requirements.

Model #	Report Date	Report #
PSA-1/4	December 12, 1979	TR 839
PSA-1/2	December 12, 1979	TR 840
PSA-1	January 21, 1980	TR 841
PSA-10	January 25, 1980	TR 843
PSA-35	February 7, 1980	TR 845
PSA-100	May 12, 1980	TR 846

4. Snubber Specific Deficiencies

A study, conducted by Brookhaven National Laboratory, under a contract funded by the NRC, identified PSC snubbers, specifically the smaller PSA-1/4 and 1/2 models (350,650 pound rated load respectively), as being sensitive to capstan spring tang failures. PSC attributes this problem to abusive actions taken by maintenance personnel such as stepping on installed units and overall improper installation practices.

However, PSC did acknowledge that several years ago a batch of capstan springs manufactured by an approved vendor were improperly heat treated. The spring tangs were formed after they were hardened followed by a stress relieving process which may have left residual stress in the root of the spring tang. Proper corrective action was implemented for those snubbers with defective springs, followed by PSC cancelling the contract with the vendor. A previous NRC inspection report, (83-01), addressed this issue, which was later satisfactorily resolved by PSC.

5. 10 CFR Part 21

PSC's standard operating procedure No. 01.07, "Compliance with 10 CFR 21, "Reporting of Defects and Noncompliances," dated May 9, 1985, was reviewed and observed to be adequate by the NRC inspector.

The most recent Part 21 notification, reported to the NRC by PSC on June 13, 1985, raised concerns over the improper installation of pipe clamps installed by Cleveland Electric Illuminating Company at Unit 1 of the Perry Nuclear Station. These clamps were supplied to PSC by Basic Engineers, a division of National Valve and Manufacturing Company (NAVCO). The NAVCO report stated that improper installation may cause the pipe clamp to slip circumferentially when a compressive load, less than the faulted rated load (ASME III-Level D), is applied to the clamp at an angle of five degrees to the clamp centerline, as permitted by design drawings.

Correction of the problem required an increase in torque values for the clamp bolts which in some cases required the installation of new bolts and nuts. All affected customers were notified in writing by PSC.

6. Procurement Document Control

Chapter 7, "Preparation and Control of Procurement Documents for Materials," and Section 7.1.2 of the Pacific Scientific Company Quality Assurance Manual Approved Supplier's List (ASL), were reviewed to determine the criteria used by PSC to qualify vendors. Selected supplier evaluation reports were reviewed to determine whether the required documents to support incorporation onto the ASL were contained in the file. The following represents a list of those reviewed by the inspectors:

Company	Audited & Approved	Criteria Used	PSC Auditor
Don Rickett Co.	August 14, 1984	Supplier Evaluation Report	Charest
Carpenter Technology	May 14, 1985	Supplier Evaluation Report	Hartford
Mills Alloy Steel Co.	February 24, 1985 March 24, 1987	QSC324	N/A

The suppliers chosen for review were found to meet the requirements for inclusion on PSC's current Approved Suppliers List.

7. Control of Special Processes

Chapter 16 of the Quality Assurance Manual, "Examination and Testing," was reviewed to determine the qualification requirements for NDE personnel. Records for the following PSC employees who perform NDE were reviewed to verify satisfactory and current NDE qualifications:

Name	NDE Level	PSC Review Date
R. Reina NDE:	VT, II	September 18, 1984
G. J. DeGrave	VT, III	April 3, 1985; November 2, 1984
R. Reina	MT, II	November 8, 1984

The training and qualification records for the above listed PSC employees support their performance of NDE testing.

8. Control of Measuring and Test Equipment

Chapter 13 of the Quality Assurance Manual, "Measuring and Testing Equipment," was reviewed with respect to the requirements of calibration control at PSC. The following was randomly selected from equipment in the shop to verify current calibration.

Equipment Type/Name	Calibration Control No.	Calibration Performed	Calibration Due
Lost Motion Transducer	TE-17-41	June 4, 1985	October 4, 1985
Snubber Test Machine	TE-18-3	June 4, 1985	October 4, 1985
Optical Comparator	TE-19-66	July 2, 1985	November 2, 1985
Hardness Tester	TE-17-11	July 8, 1985	October 1, 1985
Height Gauge	TE-17-6	February 19, 1985	February 19, 1985
Dial Indicator	TE-18-17	July 16, 1985	October 16, 1985

In summary, the test equipment calibration program at PSC was found to be organized, properly maintained, and in conformance with the QA program commitments.

E. PERSONS CONTACTED:

*E. Thomsen
*F. Frederickson
*W. S. Wright
*P. Hadnagy
P. M. Zatezalo
C. J. Charest
J. Kowalski

*Attended exit meeting

ORGANIZATION: PACIFIC SCIENTIFIC COMPANY
ANAHEIM, CALIFORNIA

REPORT NO.: 99900255/85-01	INSPECTION RESULTS:	PAGE 7 of 8

F. DOCUMENTS EXAMINED:

1. Letter, document no. 6000-QS-163-85, dated June 18, 1985, Part 21 notification to Region V.

2. Procedure, document no. 01.07, Revision 5-9-85, dated March 6, 1978, "Compliance with 10CFR21, "Reporting of Defects and Noncompliance."

3. Letter, document no. 6000-QS-177-85, dated July 19, 1985, Potential Problem with Pipe Clamps.

4. Book, 10 CFR 21 Log.

5. Qualification Test, document no. TR 839, dated December 12, 1979, Qualification Testing on PSA - 1/4.

6. Qualification Test, document no. TR 840, dated December 12, 1979, Qualification Testing on PSA - 1/2.

7. Qualification Test, document no. TR 841, dated January 21, 1980, Qualification Testing on PSA - 1.

8. Qualification Test, document no. TR 843, dated January 25, 1980, Qualification Testing on PSA - 10.

9. Qualification Test, document no. TR 845, dated February 7, 1980, Qualification Testing on PSA - 35.

10. Qualification Test, document no. TR 846, dated May 12, 1980, Qualification Testing on PSA - 100.

11. Quality Assurance Manual, document no. QAM, Revision 10, dated April 8, 1985, Pacific Scientific Kin-Tech Division Quality Assurance Manual.

12. Quality Assurance Manual, Revision 10, April 8, 1985, Pacific Scientific Quality Assurance Manual.

13. Quality Supplier Control, document no. N-1198, dated August 17, 1984, N-Stamp Authorization for Class 1, 2, & 3 & MC Component Supports (thru August 9, 1987).

14. Approved Suppliers List, document no. 6000-QE-157-85, Revision F, dated August 12, 1985, Approved Suppliers List.

15. Form 0198, Revision June 1984, Supplier Evaluation Report for Don Rickett Co. on August 14, 1984.

16. Form 0198, Supplier Evaluation Report for Carpenter Technology on May 31, 1985.

17. Quality Service Control, dated February 24, 1985, QSC for Mills Alloy Steel Co., February 24, 1985 to March 24, 1987.

18. Quality Control Document, dated September 19, 1984, NDE Qualification & Certification Record for R. Reina.

19. Procedure, Revision G, dated November 16, 1984, Penetrant Examination of Parts & Materials for Nuclear Power Plant Products.

20. Procedure, Revision J, dated November 16, 1984, Magnetic Particle Examination of Parts & Materials for Nuclear Power Plant Products.

ORGANIZATION: PAUL-MUNROE
ENERGY PRODUCTS DIVISION
ORANGE, CALIFORNIA

REPORT NO.: 99900337/85-01	INSPECTION DATE(S): 8/14-16/85	INSPECTION ON-SITE HOURS: 26

CORRESPONDENCE ADDRESS: Paul-Munroe
Energy Products Division
ATTN: Mr. Mark P. Schneider, P.E.
Corporate Director of Quality Assurance
1701 W. Sequoia Avenue
Orange, California 92668

ORGANIZATIONAL CONTACT:
TELEPHONE NUMBER: (714) 978-9600

PRINCIPAL PRODUCT: Design and manufacturer of fluid power components.

NUCLEAR INDUSTRY ACTIVITY: Major supplier of hydraulic shock suppressors (snubbers) and related testing services.

ASSIGNED INSPECTOR: RG Correa 1-3-86
FOR R. L. Pettis, Jr., Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): E. Trottier, Reactive Inspection Section (RIS)
M. Subudhi, Brookhaven National Lab

APPROVED BY: RG Correa 1-3-86
FOR J. W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 50, Appendix B; 10 CFR Part 21.

B. SCOPE: Follow-up on previously reported deficiencies involving shock suppressor design, testing and maintenance. Review of 10 CFR Part 21 system, reports of corrective action and the procurement process.

PLANT SITE APPLICABILITY: Sequoyah, Units 1 and 2 (50-327/328); Arkansas, Unit 2 (50-368).

ORGANIZATION: PAUL-MUNROE
ENERGY PRODUCTS DIVISION
ORANGE, CALIFORNIA

REPORT NO.: 99900337/85-01	INSPECTION RESULTS:	

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

1. Contrary to Paul-Munroe (PM) Quality Assurance Manual (QAM) Section 7.4(f), "Control of Purchased Items and Services," documentation was unavailable to support quarterly reviews of Nonconforming Material Reports (NMR) for the second and fourth quarters of 1984, and the first and second quarters of 1985. (85-01-01)

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

None.

E. Other Findings or Comments:

1. Anker-Holth and E-Systems Affiliation

Recently, PM acquired the engineerng and design rights to the Anker-Holth (AH) snubber line, which was earlier affiliated with McDowel-Wellman, in addition to expanding their snubber repair and spare parts business activities. As a result, all snubbers previously manufactured by AH will be serviced by PM with damaged units to be replaced with the PM design.

In addition to AH, PM has a similar contract with E-Systems, whose snubber assemblies were primarily supplied to BWR facilities designed by the General Electric Company. According to this agreement, PM will provide all service and sales for the E-System product line.

PM has been supplying the nuclear industry with primarily large capacity hydraulic snubbers used in conjunction with the seismic support of such equipment as steam generators and reactor coolant pumps.

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Most parts and products used in the design are manufactured at REMCO Hydraulics outside San Francisco, California, while the marketing, engineering, and assembly are performed at PM's headquarters in Orange, California.

PM snubbers utilize TEFZEL, a unique seal material qualified for a 40 year life, in contrast to other manufacturers who use EPR or VITON material, which PM suggests should be replaced at 5 year intervals. In the case of large capacity snubbers (1000-2200 kips), the fluid reservoir is a remote type which supplies several snubbers used as supports for such equipment as steam generators and reactor coolant pumps. In these applications, tubing is used to deliver hydraulic fluid to each snubber and is supported seismically according to normally accepted engineering practice, and without the aid of a rigorous seismic analysis.

Earlier PM snubber units utilized a Reed valve design which experienced fluid leakage problems. This problem has been corrected by PM with the incorporation of a control valve. According to PM, the only operating plant which may still have older PM units incorporating the Reed valve design is the Sequoyah Nuclear Station.

2. Review of PM Stress Analysis

An overall review of the stress analysis supporting the design of the PM shock suppressor was performed by the NRC inspector for a typical 1000 kip (1,000,000 pound load capacity) unit. Calculations were performed by PM using standard empirical equations which normally result in conservative values. Static seismic calculations were performed since it was demonstrated that the snubber assembly possessed no natural frequency below 33 cycles per second with snubber overall buckling demonstrated at the faulted load (ASME Level D) condition.

The PM units incorporate an integral fluid reservoir, located inside the main body, when used primarily as a seismic support for piping systems.

3. Repair and Test Services

PM is active in the repair and functional testing of snubbers to meet the needs of the nuclear industry. To accomplish such testing PM developed, TESTAN, a portable console unit which connects hydraulically to an installed snubber. This equipment is capable of testing the entire range of PM snubbers and is used to verify parameters such as performance under faulted load conditions, and spring, bleed and lockup rates. In addition, mobile laboratories dedicated to snubber testing and repair for on-site outage services, have also been developd by PM and toured by the NRC inspectors.

4. 10 CFR Part 21

The procedure adopted by PM to comply with 10 CFR Part 21, Quality Assurance Manual (QAM) Section 20.0 "Code of Federal Regulations, 10 CFR 21," was reviewed and found adequate by the NRC inspector. The posting of this procedure, 10 CFR Part 21, and Section 206 of the Energy Reorganization Act of 1974 were also verified.

Although the documentation reviewed was adequate, the NRC inspector found instances where it was difficult to clearly identify potential Part 21 issues and follow their final disposition.

No violations or nonconformances were identified during this part of the inspection.

5. Refurbishment of Arkansas #2 Snubbers

A review of PM's 10 CFR 21 records indicated the failure of several spherical bearings during functional testing performed by REMCO Hydraulics, in April 1985.

The bearing, which failed at full faulted load of 250 kips during the bleed test, was observed to have experienced a failure of the outer race. Further visual examination indicated the crack may have been in the bearing for some time.

At the request of AP&L, all 16 units tested were outfitted with new bearings, manufactured by the Torrington Company, rated at 400 kips each. The failed bearings were shipped to AP&L, at their request, so a complete metallurgical evaluation could be performed.

Snubber bearing failures were discussed in NRC Information Notice 84-73, dated September 14, 1984, "Downrating of Self-Aligning Ball Bushings Used in Snubbers." This notice pertained to both mechanical and hydraulic snubber types utilizing spherical ball bushings and reported end housing stresses, in some cases, exceeding the material yield. As a result, load reductions were initiated by the manufacturer, the Torrington Company, based on a more conservative analysis than that previously used to establish the bearings load rating.

6. Snubber Maintenance

Snubber seal and fluid problems were discussed with the PM staff for both PM and Anker-Holth models. Several weaknesses were identified with respect to the Technical Instruction Manual for seal and fluid maintenance of these snubbers. At present, the manual is not current with respect to normally accepted industry standards and lacks specificity with regard to overall snubber maintenance.

As a result, PM is planning to notify all owners of snubbers of such concerns prior to revising the Technical Manual. Emphasis will be placed on maintenance measures which will increase seal life through the periodic investigation of snubber fluid, reservoir level, viscosity, cleanliness and water content of the snubber fluid.

The main thrust will be to recommend to PM customers that seal replacement be performed at 5 year intervals, since experience has demonstrated that degradation of EPR and VITON A seal material occurs much faster than earlier predicted. PM believes increased seal degradation is due to such factors as compression set, aging and service temperature.

PM's snubbers incorporate a TEFZEL seal material integrated with stainless steel springs inside a plastic jacket, referred to by PM as the "LIFE-OF-PLANT" seal design and guaranteed for 40 years without replacement. AH snubbers upgraded with this seal design in addition to PM designed units will be exempt from concerns identified above.

7. Procurement Process

The inspector selected three vendors from the Paul-Munroe Approved Vendor List (AVL) for review of their qualifications. One vendor had a current and valid QSC certificate with the remaining two vendors being recently audited by Paul-Munroe. In addition, Section 7.4 of the PM QAM was reviewed which states, "The Quality Assurance engineer reviews the Nonconforming Material Report (NMR) log quarterly." The results of this review by the NRC inspector produced the following:

ORGANIZATION: PAUL-MUNROE
ENERGY PRODUCTS DIVISION
ORANGE, CALIFORNIA

REPORT NO.: 99900337/85-01	INSPECTION RESULTS:	PAGE 6 of 7

Year	Quarter	Review Status of NMR Log
1985	2	Review of NMR log not conducted
1985	1	Review of NMR log not conducted
1984	4	Review of NMR log not conducted
1984	3	Third quarter, 1984, reports were reviewed for significant trends on October 5, 1984.
1984	2	Review of NMR log not conducted
1984	1	First quarter, 1984, reports were reviewed for significant trends on January 29, 1984.

Nonconformance 85-01-01 was identified during this part of the inspection.

A new logging and tracking system for NMRs was developed by Paul-Munroe in July, 1985. To date, 15 entries have been made, but insufficient information existed to establish any trends.

F. PERSONS CONTACTED:

*J. M. Raymont, Jr.
*M. P. Schneider
*W. F. Holub
F. U. Erlach
*R. C. Fisher
A. Shelcoviz
R. Galantine
R. Estes

G. DOCUMENTS EXAMINED:

1. QAM, dated December 18, 1984, Paul-Monroe QA Program Manual.

2. IOM, Tech Instruction Manual for 1200K Hydraulic Shock Suppressors.

3. IOM, dated July 1978, Snubber & Accessories - Recirculation System (E-Systems).

REPORT NO.: 99900337/85-01	INSPECTION RESULTS:	PAGE 7 of 7

4. Report, document no. PA87776, Revision B, dated November 16, 1985, Design Report PMH-2200.

5. Report, document no. A-690734, Revision 0, dated September 7, 1977, Stress Report, SAYAGO, 1000K Snubber.

6. Report, document no. A-690623, Revision 0, dated October 27, 1976, Stress Report, Multiplant 11, 1000K Snubber.

7. Report, document no. A-690624, Revision 0, dated January 29, 1977, Stress Report, Multiplant 11, 1300K.

8. QAM, Revision A, December 18, 1984, QA Program Manual.

9. AVL, Revision 0, dated May 10, 1985, Approved Vendor List.

10. INM, NDE Training Records (Qualifications).

11. INM, Equipment Calibration Records.

12. Report, Nonconforming Material Report (NMR) 1984 & 1985.

13. INM, General (New Employee) Training Records.

14. P.O., document no. E-19121, dated June 24, 1985, AC Fasteners.

15. P.O., document no. E-15163, dated June 21, 1985, Kosmos Engineering.

16. P.O., document no. E-15280, dated July 16, 1985, Bowman Plating.

17. P.O., document no. E-15476, dated August 8, 1985, A&G Engineering.

18. P.O., document no. E-15096, dated June 17, 1985, Bowman Plating.

19. P.O., document no. E-15105, dated June 18, 1985, Bowman Plating.

20. P.O., document no. E-14052, dated May 6, 1985, Dragon Valves.

ORGANIZATION: POWER INSPECTION, INC.
WEXFORD, PENNSYLVANIA

REPORT NO.: 99901033/85-01	INSPECTION DATE(S): 10/15-17/85	INSPECTION ON-SITE HOURS: 40

CORRESPONDENCE ADDRESS: Power Inspection, Inc.
Post Office Box 216
12330 Perry Highway
Wexford, Pennsylvania 15090

ORGANIZATIONAL CONTACT: Kris Kumar, President
TELEPHONE NUMBER: (412) 935-7111

PRINCIPAL PRODUCT: Nondestructive Examination Services

NUCLEAR INDUSTRY ACTIVITY: Eddy Current testing constituted approximately 10% of 1984 sales.

ASSIGNED INSPECTOR: J. T. Conway — 11-29-85
J. T. Conway, Reactive Inspection Section (RIS) — Date

OTHER INSPECTOR(S): E. Yachimiak, Jr. (RIS)

APPROVED BY: J. T. Conway /for — 11-29-85
E. W. Merschoff, Chief, RIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 50 and 10 CFR Part 21.

B. SCOPE: This inspection was made as a result of the receipt of an allegation pertaining to certification documents for the calibration of eddy current testing equipment.

PLANT SITE APPLICABILITY: Palisades (50-255) and Beaver Valley 1 (50-334).

ORGANIZATION: POWER INSPECTION, INC.
WEXFORD, PENNSYLVANIA

REPORT NO.: 99901033/85-01	INSPECTION RESULTS:	PAGE 2 of 7

A. VIOLATION:

1. Contrary to Sections 21.6 and 21.21 of 10 CFR Part 21:

   a. Copies of 10 CFR Part 21 and Section 206 of the Energy Reorganization Act were not posted (85-01-01).

   b. Appropriate procedures to evaluate deviations or inform the licensee or purchaser of the deviation did not exist (85-01-02).

B. NONCONFORMANCES:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Procedure No. PI-A-04, it was noted that eddy current testing equipment and calibration services were obtained from Zetec in 1984 and 1985, but procurement documents were not prepared, processed and approved for the purchase of these items (85-01-03).

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 5.4 of Procedure No. PI-A-IV, there was no documented evidence that Power Inspection (PI) had indoctrinated and trained any personnel since the company was incorporated in 1982 (85-01-04).

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 5.6 of Procedure No. PI-A-04 and Section 5.3.1 of Procedure No. PI-A-12, it was noted that Zetec, a supplier of eddy current testing (ET) equipment and calibration services, was never surveyed or audited by PI; and internal audits of the QA Program have never been performed by PI (85-01-05).

4. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Procedure No. PI-A-06, it was noted that eddy current instruments (2), magnetic tape recorders (2), strip chart recorders (4), vector analyzer (1), and M-17 mixers (2) were never calibrated in 1985; and a Master Index of M&TE (Form No. 2026) and M&TE Data Sheets (Form No. 2027) did not exist for any of the items requiring calibration (85-01-06).

5. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 6.0 of Procedure No. PI-A-07, and SNT-TC-1A, a review of NDE records for one-Level III, three-Level II, and five-Level I examiners revealed (85-01-07):

a. The records for all nine examiners did not contain a statement showing completion of training in accordance with PI Procedure No. PI-A-07.

b. There was no eye examination for 1982, and it was overdue for 1985 for the Level III. One Level II was missing eye examinations for 1984 and 1985, and a 1985 examination was overdue for another Level II.

c. Copies of examinations given in 1983 for the Level III and in 1985 for one Level II were missing.

6. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 9.7.3 of SNT-TC-1A, it was noted that PI Procedure No. PI-A-07 "Certification of NDT Personnel" did not address the area of duration of interrupted services requiring re-examination and re-certification (85-01-08).

C. OPEN ITEMS:

None.

D. OTHER FINDINGS OR COMMENTS:

1. Persons Contacted

*K. Kumar, President
*J. Lint, Vice President
*F. Lovate, QA Manager

*denotes those attending the exit interview.

2. Allegation

In September 1985, the NRC Region V office received a phone call alleging that PI was using ET equipment which had not been properly calibrated.

The inspector reviewed Procedure No. PI-A-06 "Control of Measuring and Test Equipment" dated December 1, 1981 which described the calibration and certification system for M&TE used by PI personnel. It was noted that the individual(s) who prepared, reviewed, and approved the procedure was not identified on the title page.

Calibration records for two eddy current instruments, two magnetic tape recorders, four strip chart records, one vector analyzer and two M-17 mixers used on testing for Duquesne Light Company (DLC) in 1984 and 1985 and Consumer Power Company (CPC) in 1982, 1983, and 1984 were reviewed. The testing equipment (identified by S/N) consisted of oscilloscopes (B012079 and B118167), detector amps (072, 073, 224, and 234), frequency drivers (040 and 082), magnetic tape recorders (011 and 016), strip chart recorders (11557, 13643, 15395, and 19321), vector analyzer (009), and M-17 mixers (016 and 055).

The equipment was used on ET of control room air conditioning condensers, component cooling water heat exchangers in the reactor and turbine plants, and other heat exchangers (diesel generator, recirculation spray, and blowdown) at Beaver Valley Unit No. 1, and steam generators and a main condenser at Palisades.

It was noted that a Master Index of M&TE (Form No. 2026) did not exist at PI. In addition, an M&TE Data Sheet (Form No. 2027) for each item of the test equipment requiring calibration was nonexistent (see Nonconformance B.4). The NRC inspector was told by the President of PI that the purchase or lease of ET equipment from Zetec as well as calibration services performed by Zetec were handled on a verbal basis, and POs were not generated (see Nonconformance 85-01-03).

The only documentation at PI to verify that a particular item was calibrated is a certification and invoice from Zetec for the services performed. A review of the invoice file and a June 19, 1985 letter from Zetec to PI which listed the outstanding invoices indicated that Zetec had calibrated three strip chart records (S/N 19321, 13643, and 11557) in October 1984. A magnetic tape recorder (S/N 011) was calibrated in January 1985 (Invoice No. 14009 dated February 4, 1985). Calibration certifications were also in the files to confirm the above calibrations.

A further review of calibration certifications indicated that a magnetic tape recorder (S/N 016); detector amps (S/N 072, 073, 224, and 234), and a magnetic tape recorder (S/N 011); and a M-17 mixer (S/N 055) were calibrated in February, June, and July 1984 respectively. However, there were no Zetec invoices in the files for the above items.

There was no documentation to show that frequency drivers (S/N 040 and 082), oscilloscopes (S/N B012079 and B118167), a M-17 mixer (S/N 016), and a vector analyzer (S/N 009) were ever calibrated; or the other items noted above, with the exception of the magnetic tape recorder (S/N 011) were calibrated in 1985.

Based on the inspector's review and evaluation of the QA records pertaining to the calibration of M&TE, the inspector substantiated the allegation.

3. <u>NDE</u>

The NRC inspector reviewed the qualification and certification records of NDE personnel (one-Level III, three-Level II, and five-Level I) to determine whether the individuals performing ET were certified to SNT-TC-1A. The written practice of PI for all phases of certifying NDE personnel was also reviewed. The title page of Procedure No. PI-A-07 "Certification of NDT Personnel" was dated December 1, 1981, but there were no initials or signatures to indicate who prepared, reviewed and approved the document. With the exception of failing to address the area of interrupted service vs. re-examination/re-certification, the procedure appeared to be consistent with SNT-TC-1A (see Nonconformance 85-01-08).

The 10 DLC POs (eight in '84 and two in '85) to PI required NDE personnel to be certified to SNT-TC-1A (June 1975 Edition). Section 14 of CPC Contract No. NDT-82-01 "Consulting and Nondestructive Testing Services with Power Inspection, Inc." dated March 25, 1982 required that SNT-TC-1A be the recommended practice for PI's written practice. CPC's nine POs (three in '82, four in '83, and two in '84) to PI referenced Contract No. NDT-82-01.

Records for the Level III included a Personnel Certification Summary (PCS) document dated September 3, 1981. The PCS gave test scores, educational and experience background and was signed by a Level III examiner, but it did not contain a specific certification statement (e.g., to SNT-TC-1A). In addition, there were no records in the files to verify the certification of the Level III examiner. Copies of the general, specific, and practical examinations were all dated September 2, 1981. It was noted that the general examination with a score of 92 percent was an identical copy (with the exception of name, date, location, and instructor in the information block) to copies of a general examination in the file of two Level IIs

(Polaski and McGregor). A copy of page one of the general examination with spaces for "name, date, location, and instructor" in the information block whited out was also in the file.

A March 31, 1983 letter from a Level III consultant to the President of PI stated that PI's Level III successfully completed his written examination and was certified to Level III-ET. However, copies of the three examinations, as well as scores, were missing, and there was no documentation on the certification of the Level III consultant. The eye examination for 1982 was missing and the examination for 1985 was overdue by two months.

A PCS dated January 3, 1983 for Level II (Polaski) was signed by a Level III examiner, but there was no documentation attesting to the certification of the Level III examiner. Four PCSs dated in January and May 1983 and March 1984 and 1985 did not indicate to what requirements the individual was certified. The "discrepant" general examination was dated October 12, 1981 with a score of 92 percent.

Four PCSs dated December 1982, May 1983, and March 1984 and 1985 for Level II (McGregor) did not contain a specific certification statement. The eye examination for 1985 was overdue by four months. The "discrepant" general examination was dated October 12, 1981 with a score of 89 percent.

A PCS dated October 15, 1985 for Level II (Williams) did not contain a certification statement, and there were no copies of any examinations. Eye examinations for 1984 and 1985 were also missing.

For all five Level I examiners, copies of examinations were missing and the PCSs did not contain a certification statement. For one examiner (Griter), there were no eye examinations.

Nonconformance 85-01-07 was identified in this area of the inspection.

4. Reporting of Defects

The 10 POs from DLC and the nine POs from CPC imposed the requirements of 10 CFR Part 21 upon PI for the ET testing at Beaver Valley Unit No. 1 and Palisades. It was noted that PI failed to have a procedure for reporting defects and deviations; and failed to post the appropriate documents as required by 10 CFR Part 21 (see Violations 85-01-01 and 85-01-02).

5. QA Program

Although Section 5.4 of Procedure No. PI-A-IV "Quality Assurance Program-Administrative Policy" required that personnel performing activities affecting quality be indoctrinated and trained via group lectures or personal instructions with subsequent record of attendance being maintained, there was no documented evidence that any employee had been trained and indoctrinated (see Nonconformance 85-01-04).

Procedure No. PI-A-12 "Audits" requires that audits of QA activities be conducted by PI personnel to ensure compliance to QA program requirements. There was no documented evidence that audits had ever been conducted of the specific elements in PI's QA program. In addition, external audits of vendors supplying equipment and calibration services were never conducted. Nonconformance 85-01-05 was identified in this area of the inspection.

ORGANIZATION: ROBERT-JAMES SALES, INC.
BUFFALO, NEW YORK

REPORT NO.: 99901002/85-01	INSPECTION DATE(S): 3/25-28/85	INSPECTION ON-SITE HOURS: 56

CORRESPONDENCE ADDRESS: Robert-James Sales Inc.
ATTN: Mr. Robert Boker
President
269 Hinman Avenue
Buffalo, New York 14216

ORGANIZATIONAL CONTACT: Mr. Robert Boker, President
TELEPHONE NUMBER: (716) 874-6300

PRINCIPAL PRODUCT: Pipe, tubing, flanges, fittings, and valves.

NUCLEAR INDUSTRY ACTIVITY: Less than 0.1 percent of the FY 1984 sales.

ASSIGNED INSPECTOR: J. T. Conway 4-18-85
J. T. Conway, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): J. J. Petrosino, RIS

APPROVED BY: E T Baker for 4/22/85
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was made as a result of the receipt of an allegation pertaining to remarking foreign fittings and selling them as domestic fittings.

PLANT SITE APPLICABILITY: Part 21 requirements: Nine Mile Point (50-220), Fermi 2 (50-341), and James A. Fitzpatrick (50-333).

10 CFR 2.790 INFORMATION HAS BEEN DELETED

ORGANIZATION: ROBERT-JAMES SALES, INC.
BUFFALO, NEW YORK

REPORT NO.: 99901002/85-01	INSPECTION RESULTS:	PAGE 2 of 4

A. VIOLATIONS:

1. Contrary to Sections 21.6 and 21.21 of 10 CFR Part 21:

   a. Copies of 10 CFR Part 21 and Section 206 of the Energy Reorganization Act were not posted.

   b. Appropriate procedures to evaluate deviations or inform the licensee or purchaser of the deviation did not exist.

2. Contrary to Section 21.31 of 10 CFR Part 21, it was noted that purchase order (PO) Nos. 132-47, J-141-362, and J-141-383 from and PO No. 91190 from to Robert-James Sales (RJS) specified 10 CFR Part 21 as an applicable requirement, but RJS POs to (No. 6581), (No. 6561), (No. 6582), (No. 6555), (No. 3232), and (Nos. 3156, 3233, and 3924) did not similarly specify that 10 CFR Part 21 would apply.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. OTHER FINDINGS OR COMMENTS:

1. Compliance with 10 CFR Part 21 Requirements - An inspection of the shop area noted that RJS had not complied with the posting requirements of 10 CFR Part 21 (see Violation A.1.a). In addition, RJS had not developed a procedure for reporting defects and noncompliances (see Violation A.1.b).

2. Documentation Packages - Two hundred eight-nine documentation packages for pipe, flanges, fittings, and valves ordered by utilities and manufacturers were reviewed. The orders were for fiscal years (FY) 80 (36), 81 (37), 82 (48), 83 (80), 84 (60), and 85 (28). Documentation packages consisted of customer POs; POs to suppliers/manufacturers, work orders, shipping invoices, and Certificate of Conformance (CC); and CCs and/or Certified Material Test Reports (CMTR) from suppliers/manufacturers.

10 CFR 2.790 INFORMATION HAS BEEN DELETED

The majority of the 209 nonnuclear orders were for fossil type electrical generation facilities, and the 80 nuclear orders were for (60), (15), (3), and (2). A review of the 80 nuclear orders revealed the following:

a. Seventy-six orders were for non safety-related items and referenced ANSI or ASTM for the material specification.

b. POs J-141-362 and J-142-383 dated August 31 and September 7, 1982, respectively, were the only POs that referenced the requirements of Section III of the ASME Code.

c. POs 132-47 dated October 10, 1980 and J-141-362 and J-141-383 from were the only POs that specified the requirements of 10 CFR Part 21.

d. PO 91190 dated November 28, 1979, from was the only PO that required RJS to have a QA program meeting the requirements of Appendix B to 10 CFR Part 50. In addition, the PO was stamped "Note: This is an order for Safety Related Materials...All requirements of NRC Regulation 10 CFR 21 apply as outlined in draft letter #1 August 10, 1977." August 10, 1977 letter to RJS states, in part, "...inform subtier vendors that they must inform you of defects and deviations from procurement documents."

RJS ordered the items for the customer POs (3 from and one from) identified above from six material suppliers and manufacturers. It was noted that applicable POs 3924 (December 10, 1979), 3156 (September 7, 1982), and 3233 (September 16, 1982) to ; 6561 (October 10, 1980) to 6581 (October 14, 1980) to ; 6582 (October 14, 1980) to ; 6555 (October 14, 1980) to ; and 3232 (September 20, 1982) to did not identify the applicability of 10 CFR Part 21 (see Violation A.2). It was also noted that RJS POs did not include or reference QA program requirements (e.g., Appendix B to 10 CFR Part 50) or indicate that the material would be used on a nuclear project. There was no indication that the POs had been reviewed and approved by a QA representative.

ORGANIZATION: ROBERT-JAMES SALES, INC.
BUFFALO, NEW YORK

REPORT NO.: 99901002/85-01	INSPECTION RESULTS:	PAGE 4 of 4

The fittings designated as Section III/Class 2 on POs J-141-362 and J-141-383 were ordered from on POs 3233 and 3156 and on PO 3232. At the time of the orders both and were holders of an ASME Quality System Certification (Materials). In addition, all the items on the 3 POs were shipped directly from the manufacturer's facility to the Fermi-2 nuclear site. The safety-related items for PO 91190 were also shipped from facility direct to the Nine Mile Point 1 facility.

3. <u>Allegation</u> - In April 1983, an individual alleged to the NRC Region I Office that RJS was remarking foreign fittings and selling them as domestic fittings. The allegation did not specifically address any material that may have been furnished to a nuclear facility.

The NRC inspector toured RJS's warehouse facility at various times during the inspection. It was noted that stainless steel pipe, flanges and fittings were segregated according to size and alloy type. A visual inspection of markings on both foreign and domestic fittings showed no indication that original markings were altered or changed. The inspector also reviewed approximately 290 POs from utilities and suppliers/manufacturers of items to the nuclear industry from FY 1980 to the present. Only 80 POs were for items that were ordered for nuclear facilities. Of the 80 POs only 4 POs were for items considered "safety related" in that Section III and/or Part 21 requirements were imposed upon RJS.

As noted in D.2 above, all the items on the 4 POs were produced by domestic qualified manufacturers, and all the items were shipped directly from the manufacturer's facility to the nuclear site.

Based upon the inspector's review of nuclear orders from FY 1980 to the present and an indepth evaluation of stored items in the warehouse, the inspector could not substantiate the allegation.

10 CFR 2.790 INFORMATION HAS BEEN DELETED

ORGANIZATION: ROCHESTER INSTRUMENT SYSTEMS
ROCHESTER, NEW YORK

REPORT NO.: 99900222/85-01	INSPECTION DATE(S): 8/26-27/85	INSPECTION ON-SITE HOURS: 9

CORRESPONDENCE ADDRESS: Rochester Instrument Systems
ATTN: Mr. S. Rogoff
President
255 North Union Street
Rochester, New York 14605

ORGANIZATIONAL CONTACT: A. Wayne Engbrecht
TELEPHONE NUMBER: 716-263-7735

PRINCIPAL PRODUCT: Monitoring instruments such as undervoltage relays and square root extractors.

NUCLEAR INDUSTRY ACTIVITY: Less than 5%.

ASSIGNED INSPECTOR: K. Naidu 12/06/85
K. R. Naidu, Reactive Inspection Section, (RIS) Date

OTHER INSPECTOR(S):

APPROVED BY: E. W. Merschoff 12/13/85
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review of records related to square root extractors supplied to the Trojan nuclear power plant; review Rochester Instrument Systems' (RIS) evaluation of a Part 21 report by Stone & Webster related to undervoltage relay set point drift problem identified at Shoreham nuclear power plant; review of corrective action taken on findings documented in Inspection Report 99900222/77-01

PLANT SITE APPLICABILITY: Trojan Nuclear Power Plant, 50-344; Shoreham Nuclear Power Plant, 50-322.

ORGANIZATION: ROCHESTER INSTRUMENT SYSTEMS
ROCHESTER, NEW YORK

REPORT NO.: 99900222/85-01	INSPECTION RESULTS:	PAGE 2 of 6

A. Inspection Issues

1. Rochester Instrument Systems (RIS) supplied nine square root extractors, type SC-1330-C, to Portland General Electric Company (PGE) for installation in the Trojan nuclear power plant (Trojan). The objective of this inspection was to determine the adequacy of backup documentation to support the certificate of conformance supplied with the square root extractors.

2. Stone & Webster, the architect engineer for Shoreham nuclear power plant, reported to the NRC in a letter dated July 22, 1985, that type PR-2035 undervoltage relays manufactured by RIS demonstrated a tendency to drift from their calibrated set points. Long Island Lighting Company, the owner of Shoreham, reported the same defect in a letter dated July 23, 1985. The objective of this inspection was to determine whether RIS adequately evaluated the problem.

B. Background Information

RIS manufactures monitoring instruments for nuclear power plants, commercial power plants, and chemical processing plants at several locations in the USA, Canada, and England. RIS implements quality assurance programs QA-100 and QA-200 during the manufacture of commercial grade and nuclear grade items respectively. The QA personnel scrutinize all sales orders received and assign, as appropriate, the QA program to be implemented.

C. Corrective Action Taken on Part 21 Report

Stone and Webster informed the NRC in a letter dated July 22, 1985 that undervoltage relays, type PR-2035, installed at Shoreham demonstrated a tendency to drift from the calibrated set points. Long Island Lighting Company (LILCO) also informed the NRC in a letter dated July 23, 1985 of a similar problem and shipped the relays to RIS. RIS received the relays, tested them, and confirmed that there was a drift in the set points. In a letter dated September 25, 1985, RIS provided a preliminary notification to the NRC of a potential defect relative to PR-2035 type undervoltage relays. In a final report dated October 11, 1985, RIS informed the NRC that the defective relays returned by Shoreham were tested in the "as found" condition. The test results indicate that the stated minimum deadband tolerance was unobtainable. The deadband is the voltage difference between the undervoltage resets. For undervoltage units with P1 type input modules, the calculated minimum deadband adjustment should have been 0.5% of the nominal input voltage or 0.6

VAC (0.5% x 120 VAC). Undervoltage units shipped from Shoreham were found to have the following deadband adjustments:

Serial Number 71232-2	1.4 volts A.C.
Serial Number 71232-3	0.7 volts A.C.
Serial Number 71232-6	0.9 volts A.C.

RIS revised both the undervoltage circuitry and its Product Bulletin. RIS stated that the existing RR-2035 relays will be replaced in the following nuclear power plants within the next 6-9 weeks.

Long Island Lighting Company	17 units
Virginia Electric Power Company	59 units
Pacific Gas and Electric Company	18 units
Public Service Electric & Gas Company	21 units

D. Inspection Findings and Other Comments

1. Shop Tour

The inspector, accompanied by the QA Manager, toured the manufacturing facilities and observed the assembly of components on printed circuit boards for various instruments. Workmanship procedures were available at the work stations. Receipt inspections and in process inspections were being conducted as appropriate. All the test and measuring equipment were observed to have current calibration stickers. No nonconformances were identified in the above areas.

2. Review of Purchase Order Processing

The inspector reviewed the Portland General Electric Company (PGE) Purchase Order (PO) No.-29319, dated February 22, 1985 and supplement 1 dated March 21, 1985, to Branon Instruments (BI) Portland, Oregon. BI forwarded the PO to RIS for the supply of nine Model SC-1330-C type Square Root Extractors (SRE). The PO described the technical requirements and furnished the environmental conditions, such as temperature, pressure, humidity, and radiation, in which the SREs would be installed. Supplement 1 of the PO deleted the radiation requirements and stated that the SREs need not be qualified to IEEE-323 environmental qualification requirements. The PO did not require the vendor to implement a quality assurance program meeting the requirements of 10 CFR 50 Appendix B, however the PO required the vendor to conform to 10 CFR Part 21 requirements. The RIS QA manual requires all purchase orders for small instrument type orders to be reviewed by a quality committee prior to release to manufacturing. Review of the records indicates that the PGE PO was stamped "Nuclear QA-200 Program." The QA manager stated that the personnel who process incoming orders are trained to recognize an order intended for installation in a nuclear power plant and are required to stamp them "QA-200" even if the PO does not specifically require the implementation of a 10 CFR 50 Appendix B program. The PGE PO was routed through a distributor and contained statements relative to 10 CFR Part 21, Certificate of Compliance, and background radition which alerted RIS personnel to recognize that the SREs were intended for installation in a nuclear power plant. The QA manual requires the issuance of a Certificate of Compliance for all QA-200 orders.

The relevant shop orders and test records were readily retrievable. Review of the records indicates that the RIS QA-200 program was implemented during the manufacturing and testing processes.

3. Review of QA Records

   a. The QA records pertaining to the nine Square Root Extractors, type 5C-1330-C, with serial numbers 750781-1 to 9, were reviewed. The records indicate that the RIS Nuclear Quality Assurance Program QA-200 was imposed during the manufacture of the items. Test reports indicate that each instrument was tested and determined acceptable. The Seismic Qualification Report, A-295-80 dated April 30, 1980, prepared by Corporate Consulting and Development Company, Raleigh, North Carolina, was available to substantiate the validity of the certificate of conformance issued by RIS to Portland General Electric Company.

   b. Long Island Lighting Company (LILCO) purchase order (PO) 347758, dated August 21, 1979, requested RIS to supply eight Class 1E undervoltage relays, type PR-2035-P1-T1-0. The relays were specified to be manufactured to the RIS QA-200 program. LILCO subsequently issued PO 373386, dated

September 9, 1981, for the supply of three additional identical undervoltage relays, also to be manufactured under RIS QA-200 program. LILCO imposed 10 CFR Part 21 reporting requirements in the PO. The test reports indicate the relays were satisfactory.

No nonconformances were identified in the above areas.

3. Observation of Instrument Testing and Calibration

At the NRC inspector's request, RIS inspection personnel demonstrated typical tests on one undervoltage relay and one square root extractor. Documented test procedures were used. Test equipment had current calibration stickers.

No nonconformances were identified in the above area.

E. Corrective Action Taken on Previously Identified Findings

IE report 99900222/77-01 identified four items which required corrective action. In their letter dated July 14, 1977, RIS outlined the corrective action taken. During this inspection, the inspector verified the implementation of this corrective action.

1. Item 1 identified that the mission of various organizations performing activities which affect nuclear safety related functions was not defined. The current revision of the QA manual, dated February 1985, defines the mission adequately.

2. Item 2 identified that the job descriptions of the QA personnel were inadequately defined. The Manager of Industrial Relations had documented the job descriptions of four QA positions.

3. Item 3 identified that the design control procedure was inadequate. The Engineering Department developed Procedure 30-1 which appears to be adequate.

4. Item 4 identified that a procedure to assure selection and suitability of parts was not developed. Procedures G-1 and F have been developed which adequately address the control of purchased material and control of materials parts and components respectively.

F. Persons Contacted

Rochester Instrument Systems
S. Rogoff, President
*A. W. Engbrecht, Quality Assurance Manager
P. Shah, Product Manager
*D. W. Seward, Quality Assurance Audit Leader

*Denotes those individuals present at the exit interview.

G. Exit Interview

The inspector met with individuals identified in Section F at the conclusion of the inspection and discussed the scope and results of the inspection.

ORGANIZATION: SQUARE D COMPANY
PERU, INDIANA

<table>
<tr><td>REPORT
NO.: 99900367/85-01</td><td>INSPECTION
DATE(S): 10/1-3/85</td><td>INSPECTION
ON-SITE HOURS: 15</td></tr>
<tr><td colspan="3">CORRESPONDENCE ADDRESS: Square D Company
Power Equipment Division
ATTN: Mr. L. West
Quality Assurance Manager
252 North Tippicone
Peru, Indiana 46970
ORGANIZATIONAL CONTACT:
TELEPHONE NUMBER: (317) 472-3382</td></tr>
<tr><td colspan="3">PRINCIPAL PRODUCT: Motor Control Centers

NUCLEAR INDUSTRY ACTIVITY: Less than one percent of total effort.</td></tr>
<tr><td colspan="3">ASSIGNED INSPECTOR: K. R. Naidu, Reactive Inspection Section (RIS) 1/9/86 Date

OTHER INSPECTOR(S):

APPROVED BY: E. W. Merschoff, Chief, RIS, Vendor Program Branch 1/15/86 Date</td></tr>
<tr><td colspan="3">INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was made to obtain additional information on a Part 21 report issued by Long Island Lighting Company on July 23, 1985 to the NRC; to review documentation on Motor Control Centers supplied to Shoreham, San Onofre Units 2 & 3, and V. C. Summer; and to review corrective action taken on inspection findings documented in NRC inspection report 99900367/81-01.</td></tr>
<tr><td colspan="3">PLANT SITE APPLICABILITY: Shoreham (50-322), San Onofre 2 and 3 (50-361 and 50-362), V. C. Summer (50-395)</td></tr>
</table>

A. Inspection Issues

On July 23, 1985, Long Island Lighting Company (LILCo), the owner of Shoreham nuclear power plant, reported a potential 10 CFR Part 21 defect relative to the size 1 starters installed in Motor Control Centers. LILCo stated that they may have size 1 motor starters with operating coils which are only capable of picking up at 85% of the rated voltage instead of the required 77.5%. The objective of this inspection was to obtain additional information relative to this potential defect.

B. Background Information

Square D Company (Square D) located in Peru, Indiana, manufactured and supplied the 480 Motor Control Centers (MCCs) to Shoreham nuclear power plant. A combination of various sizes of circuit breakers and motor starters are installed in each MCC. Specification SH-1-115 developed by Stone and Webster (S & W), the architect engineer for Shoreham, provides the technical details for each MCC. Square D also supplied MCCs to San Onofre Units 2 & 3 and V. C. Summer nuclear power plants.

C. Inspection Findings and Other Comments

1. Review of Technical Specifications.

S & W issued specification SH-1-115 for the MCCs. This specification required the starter coils to be capable of pickup and operation at a minimum voltage of 85% of the rated voltage (the rated voltage is 120 volts). Subsequently, S & W determined that the degraded voltage could be as low as 77.5% of rated voltage. Correspondence between S & W and Square D indicates that Square D tested the starter coils for operation with 77.5% of rated voltage and determined that all starter coils except size 1 reversing starter coils picked up and operated at 77.5% of the rated voltage. The confidence level for the operating coils of size 1 reversing starters to operate at 77.5% rated voltage was low since the coils had to overcome the additional burden of interlock mechanisms and auxiliary contacts. Subsequently, Square D developed a coil capable of pickup and operation at 77.5% rated voltage in size 1 reversing starters. At that time, several MCCs had already been shipped to the Shoreham site.

Square D shipped the special coils type 31041-400-41 to Shoreham and requested S & W to arrange the replacement of the existing coils type 31041-400-42 installed in size 1 reversing starters. The Specification SH-1-115 Revision 1 dated November 13, 1980, on page 1-10 specifies that the coils for combination starters for all class 1E MCCs are to be tested for pickup at 77.5% of rated voltage. Furthermore, the coils were required to withstand 110% of rated voltage continuously without damage.

2. Review of Documentation

a. Requirements

The specification required Square D to furnish the following documents:

(1) Design test report documentation for class 1E MCCs

(2) Certified factory test reports for class 1E MCCs

(3) Statements of compliance with referenced specifications, codes and procedures

(4) Seismic testing documentation

(5) Certificate of seismic compliance

(6) Calculations of Class II equipment

(7) Anchorage systems.

b. Documentation package 12-01219-58 for MCC marked 1R 24 MCC 1128 was reviewed. The documentation package consisted of the following:

(1) A quality control inspection checklist which verified that the following attributes complied with data sheet drawing 12-01219-58A 1, structural key sheet 12-01219-58:

(a) Physical inspection of the enclosure, doors, gasketing, internal barriers and other hardware.

(b) Inspection of the size of the bus system including the neutral, horizontal ground joints, short circuit bracing, and other electrical hardware.

(c) Inspection of the general wiring.

(d) Location of the starters and circuit breakers are as specified in the layout drawings.

(e) Verification of the voltage ratings of the devices used in the assembly are correct.

(f) Verification that the various specified electrical tests were performed.

The above checklist indicated that the inspection was completed on 10/13/76 and identified no unacceptable findings.

(2) Assembly plant work sheets.

(3) The Certificate of Compliance dated October 18, 1976, stated that the equipment was constructed in compliance with those specifications, codes, and procedures referenced in specification SH 1-115.

(4) The Certificate of Seismic Compliance stated that equipment of similar design was tested in accordance with the seismic requirements in SH 1-115 and the test results were approved by Stone and Webster per letter dated November 17, 1975. The summary of test method stated that a combination of testing and analysis was used in the qualification of the equipment. Structures with devices installed were tested and data obtained from the test results were used to evaluate the response levels of a typical structure. Devices were also tested individually.

(5) The Certification of Factory Tests dated October 12, 1976, stated that the following tests were performed:

(a) Dielectric test per ANSI C 19.1 section 15.15-65B operation and mechanical adjustment test per NEMA IC.1-2.40.

(b) Molded case circuit breaker production test per NEMA AB1-2.22.

(c) Continuity test on all wires from termination to termination.

c. Review of Documentation Package 12-01219-67 for MCC marked 1R 24-MCC-112Y. This package contains all the documents mentioned in the previous paragraph including a checklist, which was revised on November 23, 1976, to include the 77½% pickup voltage test.

3. Review of Test Procedure for Contact Pickup Voltage

Quality Control Procedure (QCP) #167 dated 12/19/79 was developed to test all starters intended for Shoreham nuclear power plant. QCP 167 adequately describes the test to verify that the coil picks up and operates at 77½% of the rated voltage and lists the test equipment to be used. A table furnishes the values of fixed resistances to be used in series with coils of various starter sizes. The series resistor was selected to simulate a "hot coil" assuming that the coil under test was at room temperature.

4. Review of Seismic Qualification Records

The specification SH-1-115 specifies that the MCC should be capable of withstanding the following events: Operating Basis Earthquake g=0.26 horizontal, g=0.20 vertical; Design Basis Earthquake g=0.48 horizontal, g=0.28 vertical. (g=acceleration as a fraction of acceleration due to gravity.)

Documents indicate that seismic withstandability tests were performed to determine the characteristics and limits of eleven common devices used in Model 4 Motor Control Center. The tests were conducted in accordance with test plan #8998-10.02 at Dayton T. Brown Inc. of Bohemia, Long Island, New York from October 17 through November 6, 1973. The eleven common devices consisted of five different types of circuit breakers, four starters - NEMA size 1 through 4, and two relays.

All the specimens performed as anticipated. The number of mounting screws which fasten these starters to the enclosure were changed from three to five for size 3 and size 4 starters.

Agastat relays were qualified on June 22, 1981. A seismic simulation test was performed to determine the dynamic characteristics of Model 4 MCC with various unit and short circuit bracing levels. These tests were conducted in accordance with the test plan identified as 108-1.01 dated 02/20/74 at Wyle Laboratories in Huntsville, Alabama from May 13 through May 22, 1974. A total of 229 test runs were conducted.

Conclusions

The test demonstrated that the Model 4 MCC would operate properly in nuclear power plants under a variety of earthquake conditions. The test plan was designed to be conservative by incorporating several worst case conditions such as:

a. Deceleration response and contact chatter were monitored on the unit mounted in the uppermost position.

b. Single and multiple frequency test inputs were used; thus, the control center was subjected to more simulated seismic excitation than it would be expected to experience during its lifetime.

The seismic withstandability parameters of the devices monitored during this test were determined in phase I of the seismic test program. These devices were size 1 through size 4 starters, circuit breakers and relays. The seismic performance of any device in the event of an earthquake can be determined by comparing the dynamic environment of each device mounting location to the acceleration qualification level "AQL" of each device.

The difference in the natural frequencies of structures braced for 65 Kilo Amperes (KA) and 42KA and those for a standard 22KA braced structure is minor and was not considered.

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5. Review of Inspector Qualification Records

The Quality Assurance (QA) staff consists of a QA manager, five Quality Control inspectors and eight Quality Assurance analysts.

Review of the qualification records of five QC inspectors and three QA analysts identified no unacceptable findings.

6. Review of Documentation on MCCs Supplied to San Onofre Nuclear Generating Station

Specification SO 23-302-4 dated August 16, 1974, issued by Bechtel Power Corporation, Norwalk, California, established the following test requirements for MCCs.

(1) Paragraph 4.6.3.2 requires the completely assembled MCC to be tested in accordance with UL Standard for Safety 845 and NEMA publication No. ICS-1970.

(2) Paragraph 4.6.3.2 requires dielectric tests to be made on each assembled unit at the vendor's factory prior to shipment.

(3) Paragr ph 4.6.3.3 requires the vendor to demonstrate that endurance tests were performed in accordance with NEMA ICS-1970 on circuit breakers.

b. Review of the documents associated with MCC 3BJ supplied for the San Onofre nuclear generating plant indicates that tests were performed to satisfy the specification requirements enumerated in the preceding paragraph. The documents were filed in two separate folders, one to comply with Bechtel's requirements and the other to reflect Square D manufacturing requirements.

The review identified no unacceptable findings.

7. Review of Documentation on MCCs Supplied to V. C. Summer Nuclear Generation Station

a. Requirements for MCCs.

Specification SP-555-044461-000 dated August 14, 1974 issued by Gilberts Associates, specifies the requirements for 480 volt MCCs. The following are the highlights relative to the starters and circuit breakers:

(1) Paragraph 2:05.7.2.a requires the holding coil to be rated for 120 volts 60 hertz per ICS.2-110.41. In addition, the coil shall have a drop out voltage of less than 65% of the rated voltage.

(2) Paragraph 2:07.2 specifies the following tests:

(a) With all of combination starters and circuit breakers in place, the equipment shall successfully pass the dielectric test for 600 volt equipment performed in accordance with NEMA ICS Part 1-109.05.

(b) All testing requirements specified in items 2:03 and 2:04 shall be carried out and documented. (2:03 lists several Gilbert requirements; 2:04 lists several applicable codes and standards).

(c) Copies of all test documents shall be submitted to the OWNER and ENGINEER before shipment of the equipment.

(d) The bidder shall submit with his proposal a list of all design and production tests to be performed on the equipment quoted.

b. Seismic requirements for MCCs.

The specification describes the seismic requirements which the MCCs should withstand without deleterious effects.

(1) Operating Base Earthquake (OBE) shall mean that earthquake which is of sufficient probability of occurrence to require its resulting ground accelerations at a site to be considered for operational loadings. The maximum horizontal ground acceleration for OBE is 0.10 g for foundations in rock and 0.15 g for foundations in soil. The corresponding vertical acceleration is 2/3 of the horizontal acceleration.

(2) Design Basis Earthquake (DBE)

Maximum horizontal ground accleration for DBE is 0.15 g for rock and 0.25 g for soil. Vertical acceleration is 2/3 of horizontal acceleration.

c. The inspector reviewed two documentation packages identified as 12-01219-07 and 12-01219-058. The following documents stated that each MCC was acceptable.

(1) A QC inspection checklist verified that all the attributes in the MCC were in compliance with the applicable drawings.

(2) The Certificate of Compliance stated that the MCC is in compliance with the requirements of the purchase order, including all acknowledged revisions and deviations.

(3) The Certificate that Factory Tests were performed indicated that no unacceptable conditions were identified.

8. Inspection Results

The results of the inspection indicate that Square D performed adequate tests on the replacement starters supplied to Shoreham to assure that the coils in size 1 reversing starters would pick up and operate at 77½% rated voltage. Square D shipped coils separately to Shoreham for installation in size 1 reversing starters which had been furnished prior to the 77½% rated voltage operation requirement. Review of the documentation packages for MCCs supplied to San Onofre and VC Summer nuclear power stations identified no unacceptable findings.

D. Action Taken on Previous Inspection Findings

The inspector reviewed action taken by Square D on the nonconformances identified during an inspection conducted during November 30 through December 4, 1981, and documented in Inspection Report 99900367/81-01.

1. (Closed) Item A identified that Quality Control Procedure #187 dated February 18, 1980 was not completely followed. The reportability portion of the form had not been completed for Problem Report No. 1980-2 which related to coating on rubber bumpers for size 3 starters which was initially reported on February 28, 1980. The Master Form was revised on October 3, 1981 to clarify the reporting section.

2. (Closed) Item B identified that design control measures were not adequately implemented for field changes related to relocation of switchgear within cabinets to eliminate the problem of pinched wires. The corrective action was in the form of a letter dated 02/24/82 to the Square D Utility Sales Group and the Design Engineering Group to stress the importance of documenting reviews of all design changes including field changes.

3. (Closed) Item C identified that contrary to Square D Standard Practice Bulletin (SPB) 512.406 dated December 6, 1976, corrective action had not been reviewed within the prescribed time limits for each finding identified in the QA Program Audit conducted during July 13 and 14, 1978. The corrective action taken was to place all audit finding reports on a 30 day or less followup schedule.

4. (Closed) Item D identified that contrary to SPB 521.307 dated November 10, 1977, Paragraph III.B, Route Change Form No. PE-1014 was not being used to change Master Record Routings. Instead, the inspector was informed that the Operational Routing Manual, dated September 28, 1981, (which was not an approved document at that time) was being used to change Master Record Ratings. Corrective action taken was to review the Standard Practice Bulletins and require internal audits to review these procedures for correctness and accuracy.

5. (Closed) Item E identified that contrary to SPB 500.020 dated September 20, 1976 and Quality Control Procedure 200-9, Revision C, dated October 15, 1981, personnel had not satisfactorily performed the electrocoat paint process to comply with applicable procedures and the records maintained on the electrocoat paint process did not verify control of equipment settings, chemical tests and gauge readings. Corrective action taken was to revise the control record to document the set up ranges rather than the upper and lower limits for the temperatures at various stations, the pressure, the ph and the free alkali.

6. (Closed) Item F identified that contrary to SPB 500.12 dated June 22, 1973, the location of a drawing removed from the Master File was not identified prior to removal and revision marks were not placed on some drawings to indicate the latest changes made on those drawings. SPB 500.12 was revised and now requires a reproducible copy to be placed in the Master File whenever Master drawings are removed. The

reproduced copy is to be stamped and will contain the design notice number, date and the name of the draftsman or engineer. The revised procedure requires that the revision marks not be removed.

7. (Closed) Item G identified that contrary to SPB 512.407 dated May 11, 1976, neither the Peru Plant QC Supervisor nor his designate had reviewed each written QCP annually as evidenced by the lack of records of QCP reviews. Corrective action taken was to revise SPB 512.407 to simplify the review documentation by providing for the reviewer to sign on the QCP contents page that the review was completed.

E. Persons Contacted

A. B. Sagersee, Manager, Utility Marketing

L. West, Manager, Quality Assurance
R. B. Wiley, Product Qualification Engineer
D. Rogers, Product Specialist
S. Higgins, Q.C. Supervisor
A. Birkmire, Q.C. Analyst

F. Exit Interview

The inspector met with the Quality Assurance Manager at the conclusion of the inspection and discussed the scope and findings.

ORGANIZATION: VALLEY STEEL PRODUCTS COMPANY
ST. LOUIS, MISSOURI

REPORT NO.: 99901019/85-01	INSPECTION DATE(S): 11/20/85	INSPECTION ON-SITE HOURS: 16

CORRESPONDENCE ADDRESS: Valley Steel Products Company
ATTN: Mr. R. Guthrie
Vice President - Operations
Post Office Box 503
St. Louis, Missouri 63166

ORGANIZATIONAL CONTACT: Mr. G. R. Mergel, QA Manager
TELEPHONE NUMBER: (314) 231-2160

PRINCIPAL PRODUCT: Ferrous Seamless Pipe and Tubing

NUCLEAR INDUSTRY ACTIVITY: None since November 1982.

ASSIGNED INSPECTOR: J. T. Conway
J. T. Conway, Reactive Inspection Section (RIS) — 12-16-85 — Date

OTHER INSPECTOR(S): J. C. Harper (RIS)

APPROVED BY: E. W. Merschoff
E. W. Merschoff, Chief, RIS, Vendor Program Branch — 12-27-85 — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was made as part of an NRC initiated review of compliance by material manufacturers and suppliers with Section III, Subsection NCA-3800 requirements of the ASME Code.

PLANT SITE APPLICABILITY: Not identified during the inspection.

REPORT NO.: 99901019/85-01	INSPECTION RESULTS:	PAGE 2 of 5

A. Violation

Contrary to Section 21.21 of 10 CFR Part 21, appropriate procedures to evaluate deviations or inform the licensee or purchaser of the deviation did not exist (85-01-01).

B. Nonconformance

Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Sections 11.2 and 11.3 of the Quality Assurance Identification and Verification Program, a review of calibration records and vendor audits from 1978 through 1983 indicated that Weber Gage, Radiatronics, and Ehrhardt Tool and Machine Company (ETMC), had performed calibration services for Valley Steel, but Weber Gage and Radiatronics were not on the Qualified Vendor List (QVL). Although ETMC appeared on the QVL, there was no documented evidence that Valley Steel performed surveys or audits of any of the three vendors (85-01-02).

C. Unresolved Items

None.

D. Other Findings or Comments

1. Personnel Contacted

*G. Ray Guthrie, Vice President Operations
*G. R. Mergel, QA Manager

*Denotes those attending the exit meeting.

2. 10 CFR Part 21

A review was conducted to verify that Valley Steel Products (VSP) had complied with the posting and procedural requirements of 10 CFR Part 21. The NRC inspector reviewed a "Notice" which the QA Manager said was posted at VSP's warehouse in Sparta, Illinois. The Notice, which was attached to Section 206 and 10 CFR Part 21, indicated that suspected noncompliances were to be reported directly to the QA Manager, VSP or the Vice President-Engineering, Valley Industries. There was no documented evidence that VSP had written a procedure relating to the reporting of defects (See violation 85-01-01).

3. Control of Purchased Material

The NRC inspector reviewed all nuclear orders (five) placed with VSP. The five orders were placed from August 1979 thru November 1982 and included the following:

a) Western Piping & Engineering - San Francisco, California

PO No. 100086 dated August 15, 1979 was for 400 ft. of 10" carbon steel pipe ordered to Section III, Class 3 of the ASME Code.

b) Tube Turns Division - Louisville, Kentucky

PO No. 38778 dated October 20, 1980 was for 12 items of 6" x 16½ ft. long carbon steel pipe ordered to Section III, Class 2 of the ASME Code.

c) A. B. Murray - McKeesport, Pennsylvania

PO No. P79249-K1 dated September 11, 1981, was for four items of 18" x 6 ft. long carbon steel pipe ordered to Section III, Class 3 of the ASME Code.

d) Capitol Pipe & Steel (CPS) - Bala Cynwyd, Pennsylvania

PO No. D-24347-OON dated January 27, 1982 was for 2 ft. of 20" carbon steel pipe ordered to Section III, Class 2 of the ASME Code. The requirements of 10 CFR Part 21 were imposed, and the pipe was to be shipped to Associated Piping & Engineering in Compton, California.

e) CPS - Bala Cynwyd, Pennsylvania

PO No. D-29878-OON dated November 8, 1982 was for 230 ft. of 12" carbon steel pipe ordered to Section III, Class 2 of the ASME Code. The requirements of 10 CFR Part 21 were imposed, and the pipe was to be shipped to CP&S in Perland, Texas.

All of the seamless pipe in the above orders was ordered to Grade B of specification SA 106. The VSP work orders for the five nuclear orders were stamped "ASME III-Nuclear."

The work orders identified processing/coating instructions (e.g., blast/clean per VSP-CL-500 and final inspection per VSP-INSP-101) and were initialed and dated by QA for heat treat number verification both prior to pulling stock and after processing.

The material for the five orders was purchased from United States Steel (USS) in Lorain, Ohio on VSP PO No. A8318-RH dated August 31, 1978. Ten specific items to ASTM/ASME SA106-B were identified on the PO which was stamped "ASME III-Nuclear" and initialed and dated by the QA Manager. The PO also contained a statement that: (a) the material was to be manufactured in accordance with a quality program audited and approved by VSP on May 9, 1978 as conforming to NCA-3800 of ASME Section III, (b) 10 CFR Part 21 applied, and (c) no weld repair was allowed.

VSP inspection reports for the five orders were reviewed. Both in-process (before cutting) and final (after cutting) inspections were performed by a QC inspector per Procedure No. VSP-INSP-101.

The Certified Material Test Reports (CMTR) for the five items from USS referenced the VSP PO, 10 CFR Part 21, and Section III certification. VSP CMTRs referenced the same heat numbers and mechanical and chemical properties as the applicable USS CMTR. Copies of both the VSP and USS CMTR were sent to the customer when the items were shipped.

It was noted that VSP received a QSC (Materials) in May 26, 1978 as a "Material Supplier of Carbon & Low Alloy Seamless Pipe & Tubing" at their Sparta, Illinois warehouse. The certificate was renewed (No. 335) in May 1981, and was allowed to expire in May 1984.

4. Indoctrination & Training

The VSP training log was reviewed by the NRC Inspector. The log described the training session date, class duration, instructor, attendees, and subject matter. Three QC inspectors were trained in ASME Section III requirements, upgraded QA program, inspection techniques, calibration control system and implementation of the QA manual.

5. Calibration of Measuring and Test Equipment (M&TE)

The NRC inspector reviewed records for M&TE and certifications for reference standards calibrated by outside vendors. Wall micrometers (S/Ns 5001, 5002, and 5004), a D-meter (S/N 610095), and and measuring rods were properly calibrated in accordance with VSP procedures. It was noted that VSP received calibration services from Weber Gage, Radiatronics and ETMC from 1978 through 1983, but there was no documented evidence that VSP required these companies to have a QA program, or that a pre-award evaluation and post-award audits were conducted on each vendor by VSP (See Nonconformance 85-01-02). Radiatronics calibrated the D-meter in 1982 and 1983, ETMC calibrated gage block set No. 77112 in 1978, and Weber Gage calibrated gage block set No. 0800 in 1978.

6. External Audits

The NRC inspector reviewed applicable sections of the QA manual and vendor qualification audit reports. VSP audited USS in Lorain, Ohio on May 9, 1978, and USS was added to the QVL for nuclear material. The audit of USS was very comprehensive. The composition of the audit included a checklist as well as a detailed narrative on process control, documentation on plant observations, and traceability of the product through the mill.

There were no records of Vendor Qualification Audits performed on the vendors who supplied VSP with calibration services (See Nonconformance 85-01-02).

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR TECHNOLOGY DIVISION
FOREST HILLS, PENNSYLVANIA

REPORT NO.: 99900900/85-02	INSPECTION DATE(S): 9/16-17/85	INSPECTION ON-SITE HOURS: 26

CORRESPONDENCE ADDRESS: Westinghouse Electric Corporation
Nuclear Technology Division
ATTN: Mr. J. L. Gallagher, General Manager
Post Office Box 355
Pittsburgh, Pennsylvania 15230-0355

ORGANIZATIONAL CONTACT: Mr. P. T. McManus, QA Manager, NTD
TELEPHONE NUMBER: (412) 825-7988

PRINCIPAL PRODUCT: Functional and environmental testing of nuclear power plant equipment.
NUCLEAR INDUSTRY ACTIVITY: Westinghouse Nuclear Technology Division (W-NTD) Forest Hills test laboratory performs developmental, verification and qualification testing of both nuclear and non-nuclear power plant components. Loss-of-coolant accident (LOCA)/thermal aging equipment qualification testing of nuclear power plant safety-related equipment comprises approximately 10% of the facility's work.

ASSIGNED INSPECTOR: S D Alexander 11/13/85
S. D. Alexander, Equp. Qual. Inspec. Section (EQIS) Date

OTHER INSPECTOR(S): M. Jacobus, Sandia National Laboratories (SNL)

APPROVED BY: Uldis Potapovs 11-13-85
U. Potapovs, Chief, EQIS, Vendor Program Branch Date

INSPECTION BASIS AND SCOPE:

A. BASIS: 10 CFR Part 50, Appendix B.

B. SCOPE: This inspection consisted of observation of a high energy line brea(k) (HELB) test on a large pump motor (LPM), review of related documentations and follow up on issues raised during the previous inspection.

PLANT SITE APPLICABILITY: 50-412/Beaver Valley-2, 50-423/Millstone-3.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR TECHNOLOGY DIVISION
FOREST HILLS, PENNSYLVANIA

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A. Violations:

None.

B. Nonconformances:

None.

C. Unresolved Items

None.

D. Status of Previous Inspection Findings:

1. (Open) Unresolved Item (85-01, Item C.1): W-NTD had not received from its irradiation services subcontractor all the specific data required by procurement specifications for test program 83-0296. The NRC inspector examined a letter from W-NTD to the subcontractor written since the last inspection requesting the missing data and examined the material provided by the subcontractor in response. The inspector noted that while the data were provided, there was no evidence that the data sheets had been reviewed as required by the procurement specification.

E. Other Findings and Comments:

1. Observation of Testing

A 600 HP electric motor (LPM) was being tested for a HELB environment. The test was intended to be generic in scope, although several specific applications at two plants had been identified. The NRC inspectors witnessed portions of the first 24 hours of the test. Prior to the HELB transient, the motor was run for a minimum of two hours in accordance with the test plan. The HELB was begun at about 2:45 p.m. on 9/16/85. The following test plan deviations/anomalies were noted:

a. The test plan specified a ramp time of ten seconds to 220°F. The actual ramp time achieved was approximately 100 seconds. The remainder of the profile was met or exceeded through 65 minutes into the test. The rest of the test data for the initial 24-hour period were not readily available for inspection, but checks were made at several times to verify compliance with the test plan.

b. The test plan called for a relative humidity (RH) of 100% for 65 minutes to 7 days. The normal value for RH on the instrumentation parameter list was specified as 90% with a warning given at 75%. Westinghouse personnel stated that 90% was the typical value observed in the pretest runs, but 100% was the target value. The values observed during the test were in the 120% range. W-NTD personnel stated that a post-test calibration will be performed on the humidity instrument to verify consistency with pre-test values, but the anomalous values observed probably resulted from exceeding instrument capabilities in the moisture saturated, HELB test chamber environment.

During the parts of the test witnessed by the inspectors, motor phase currents remained relatively stable and no anomalous conditions of the motor were observed.

2. The NRC inspectors reviewed equipment qualification documentation related to the HELB on the LPM with the following comments:

a. Much of the documentation was not available for inspection because it has not yet been received at Forest Hills. This documentation, which will be reviewed in a future inspection, includes the thermal aging data, the mechanical cycling data (the motor was started 2000 times), and the functional test data taken between phases of the test.

b. The thermal aging oven had a specified accuracy of control of ±3°C and the thermocouples recording the temperature had a tolerance of ±4°F, for a maximum possible error of about 5.2°C, a difference which could make the aging time change significantly. Westinghouse compensated for this problem in two ways. First, the oven temperature was monitored by four thermocouples, and their average used for control, thus reducing the expected error in temperature measurement. Secondly, 15°F margin was added to the anticipated service temperature, making the errors relatively inconsequential.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR TECHNOLOGY DIVISION
FOREST HILLS, PENNSYLVANIA

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c. The only part of the motor thermally aged was the stator since all other parts were made of metal.

d. The test plan specified that the motor be cleaned before reassembly after thermal aging. Westinghouse justified this by stating that cleaning is part of normal reassembly procedures (the motor was disassembled for thermal aging) and that the instruction manual requires periodic cleaning of the motor.

e. Several deviation notices were reviewed and found to contain adequate disposition actions.

f. The oil used in the motor during the HELB was to be aged and irradiated according to the test plan. Westinghouse stated that the oil had been irradiated but because of problems with thermal aging, the oil was aged by bubbling air through it for 3 days at ambient temperature to simulate oxidation.

g. Most subcontractor data were not available for review except those from the irradiation facility. Review of the letter report from this subcontractor indicated that irradiation had been conducted in accordance with the test plan. However, the package was found to be lacking some items called for in the procurement specification including photographs and a description of the test facility and set up. This area will be reviewed in a future inspection.

h. The inspectors reviewed the temperature and pressure plots on the strip chart recorder output and the data logger printouts which had been generated during the first 24 hours of the test. The strip chart recorder output (covering the initial transient conditions) was consistent with data logger printouts with the exception that (1) the strip chart recorder output was anotated with the wrong test program number (83-0292 instead of 82-0292) and (2) the initial chart speed used was 20 in./min., as specified in the procedure, while it was annotated as 4 in./min. W-NTD corrected these discrepancies stating that they had not yet had time adequately to review the data themselves.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORP.
MURRYSVILLE, PA

REPORT NO.: 99901031/85-01	INSPECTION DATE(S): 10/7-9/85	INSPECTION ON-SITE HOURS: 48

CORRESPONDENCE ADDRESS: Westinghouse Electric Corp.
Nuclear Services Integration Division
Post Office Box 78
Pittsburgh, PA 15230

ORGANIZATIONAL CONTACT: Gordon E. Michel, Quality Programs, NSID
TELEPHONE NUMBER: (412) 256-6474

PRINCIPAL PRODUCT: Equipment/parts/services for operating nuclear plants.

NUCLEAR INDUSTRY ACTIVITY: Nearly all of Westinghouse Nuclear Services Integration Division (NSID) activities are related to supplying nuclear plants with replacement components and services.

ASSIGNED INSPECTOR: RP Correia 1-23-86
R. P. Correia, Special Programs Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): P. J. Prescott, SPIS

APPROVED BY: John W Craig 1/23/86
John W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: The inspection consisted of an evaluation of quality assurance and engineering activities related to the design, procurement, manufacturing, inspection and testing of 4160/480 volt transformers procured by Omaha Public Power District for the Fort Calhoun station.

PLANT SITE APPLICABILITY: Ft. Calhoun (50-285)

REPORT NO.: 99901031/85-01	INSPECTION RESULTS:	PAGE 2 of 5

A. Violations

None.

B. Nonconformances

1) Contrary to 10 CFR Part 50, Appendix B, Criterion III and Westinghouse Document No. DTT-1, "Dry Type Transformers Commercial Dedication Process" used in the dedication process for the Ft. Calhoun 4160/480 volt transformers, Westinghouse failed to document the selection and the review for suitability of application of materials and parts that were determined to be essential to the safety-related functions of the transformers. (85-01-01)

2) Contrary to 10 CFR 50, Appendix B, Criterion III, Westinghouse failed to establish measures to control design interfaces and coordination among participating organizations who determined which components of the Ft. Calhoun transformers were critical to the safety-related operation of the transformers. (85-01-02)

3) Contrary to 10 CFR Part 50, Appendix B, Criterion VII, and Westinghouse Document No. DTT-1, "Dry Type Transformers Commercial Dedication Process", Westinghouse failed to document evidence of quality furnished by the manufacturer on the 4160/480 volt transformers which were being procured by Omaha Public Power District for the Ft. Calhoun Station. This evidence includes source evaluation or information from the manufacturer or other customers of the manufacturer on field experience. (85-01-03)

C. Unresolved Items

None.

D. Status of Previous Inspection Findings

This is the first Vendor Program Branch inspection of the Nuclear Services Integration Division of Westinghouse.

E. Other Findings or Comments

1. Dedication of Items Procured as Commercial Grade

Westinghouse NSID personnel present at the entrance meeting with the NRC inspectors outlined the process by which the 4160/480 volt transformers procured by Omaha Public Power District for the Ft. Calhoun Station were being manufactured at a

Westinghouse facility in South Boston, Virginia as commercial grade items and in turn would be dedicated by NSID for use as nuclear safety-related components. Commercial grade items are defined in 10 CFR Part 21 as being (1) not subject to design or specification requirements that are unique to facilities or activities licensed pursuant to Parts 30, 40, 50, 60, 61, 70, 71, or 72 of chapter 1 of 10 CFR and (2) used in applications other than facilities or activities licensed pursuant to the aforementioned parts and (3) to be ordered from the manufacturer/supplier on the basis of specifications set forth in the manufacturer's published product description.

Westinghouse Procedure No. WCAP-10859 (Rev. 0, dated 6/85), "Renewal Parts Dedication Process" is the base document used to dedicate commercial grade items. This is a general procedure which requires that each type of component have a set of procedures specific to its dedication for use as a nuclear safety-related component. Document No. DTT-1, "Dry Type Transformers Commercial Dedication Process", defines the scope and activities by which a series of disc wound, polyester-encapsulated dry-type transformers are to be dedicated. These activities are then implemented by an "Engineering Control Instruction" (ECI). The ECI outlines specific instructions by which a single series of transformers of a specific electrical rating are to be physically inspected, dimensionally, materially and energized operability checked. These three documents (WCAP-10859, DTT-1 and an ECI) are the ones by which the 4160/480 volt transformers procured by Omaha Public Power District for Ft. Calhoun are to be dedicated.

During discussions between the NRC inspectors and Westinghouse engineers involved with the Ft. Calhoun transformer procurement, the portions of the dedication process which had been completed were reviewed:

a. Vendor/Manufacturer Selection

Westinghouse selects a vendor/manufacturer of commercial grade transformers which follows a quality control program based on commercial industry practice and manufactures a transformer which meets the electrical requirements specified by the customer. If possible, the vendor would provide as much available literature on the transformer and any information from other customers on field experience with similar type transformers.

Westinghouse NSID had selected a Westinghouse transformer manufacturer in South Boston, Virginia to supply the commercial grade transformers to fill Omaha Public Power District purchase orders via the dedication process. The basis of this selection was Westinghouse NSID's knowledge of the history of the South Boston facility's production of transformers and that this facility used materials for their transformers that have Westinghouse established material specifications. Westinghouse NSID had not documented this information. See Nonconformance Items 85-01-01 and 85-01-03.

b. Determination and Qualification of Critical Parts

Westinghouse engineers determine which parts of the transformer are critical to its safe operation during all plant conditions as identified and required by the customer. Once this has been established, the critical parts are analyzed, tested and inspected to assure that the requirements of 10 CFR Part 21 and Part 50, Appendix B as specified by the customer are met and that they would perform as intended.

Determination and evaluation of critical parts of the transformer and their subsequent analyses, tests and inspections were decided during meetings held by members of a review committee. Verification of the operability of the critical parts are considered adequate by Westinghouse upon completion of the inspection and testing of the transformers. The only documentation that existed for this evaluation process was the minutes of these meetings. Nonconformance item 85-01-02 was identified as a result of this finding.

2. Testing and Inspection

During discussions between NRC inspectors and Westinghouse engineering personnel, NRC inspectors learned how the transformers were to be tested and inspected as part of the verification method used to check the adequacy of the evaluations which determined the components classified as critical to the safe operation of the transformers. A computer-aided seismic structural analysis of the transformer frame would be performed, and upon delivery of one of the eight transformers being manufactured by the South Boston facility to Westinghouse's test facility in Large, Pennsylvania, a shake-table test would be performed in accordance with IEEE 344-1975.

The shake-table test would demonstrate both functionality of the transformer and the integrity of the supporting and enclosing structures. Also, the transformer's environmental qualification will be based on a comparison analysis to components similar in construction, and materials prequalified by Westinghouse's aging/qualification testing programs.

Upon completion of the manufacturing of the seven transformers for installation at Ft. Calhoun, Westinghouse engineering, quality assurance and inspection personnel were to inspect and test each transformer to the requirements specified for this type of transformer in accordance with "Engineering Control Instruction No. TRC-100485.01." This instruction specifies physical, dimensional, material and energized operability checks are to be performed after the manufacturer has tested, (witnessed by NSID personnel) accepted and certified completion of industry standard production tests. Upon completion and acceptance of inspection and testing the transformers will be readied for shipment to Ft. Calhoun.

3. Inspection of Westinghouse Seco Road Facility

The NRC inspectors visited Westinghouse's NSID inspection and test facility at Seco Road. This facility is normally used to inspect and test components purchased commercially to be dedicated for nuclear safety-related use. Areas of the facility inspected were the receipt, storage, assembly, testing and calibration. The NRC inspectors examined the facility's QA manual and records of components previously tested and inspected for dedication.

During discussions between NRC inspectors and Seco Road personnel, the method Westinghouse procures, receives, inspects, assembles, and tests commercial grade components was reviewed. Previously, the components tested at the Seco Road facility for use as basic components have been small in size and the entire component has been inspected and tested for dedication. The Ft. Calhoun transformers are the first commercial grade items procured by Westinghouse NSID to be dedicated for safety-related use in which only specific parts of the unit were deemed critical to safety and subsequently, these parts were tested and inspected.

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

REPORT NO.: 99900902/85-03	INSPECTION DATE(S): 09/30/85-10/1-4/85	INSPECTION ON-SITE HOURS: 68

CORRESPONDENCE ADDRESS: Wyle Laboratories
Scientific Services and Systems Group
ATTN: Mr. W. W. Holbrook, General Manager
Eastern Test and Engineering Operations
7800 Governors Drive
Huntsville, Alabama 35807

ORGANIZATIONAL CONTACT: Mr. E. W. Smith, Director, Contracts and Purchasing
TELEPHONE NUMBER: (205) 837-4411

PRINCIPAL PRODUCT: Research, engineering, and test operations

NUCLEAR INDUSTRY ACTIVITY: Wyle Laboratories; Huntsville, Alabama, provides a variety of nuclear services to the industry. These services include environmental and seismic qualification testing of safety-related equipment, refurbishment and recertification of valves, valve and component flow testing, mechanical and hydraulic snubber testing, decontamination, and repair.

ASSIGNED INSPECTOR: Randolph N Moist 11-7-85
R. N. Moist, Equipment Qualification Inspection Section (EQIS) Date

OTHER INSPECTOR(S): E. H. Richards, Sandia National Laboratories (SNL)
J. Grossman, SNL

APPROVED BY: Uldis Potapovs 11-7-85
U. Potapovs, Section Chief, EQIS Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection consisted of: (1) a technical evaluation of equipment qualification (EQ) test activities for safety-related equipment (2) witnessing EQ testing; and (3) verification of implementation of the quality assurance (QA) program.

PLANT SITE APPLICABILITY: Browns Ferry 1, 2 & 3 (50-259, 50-260, 50-296), Sequoyah-1, 2 (50-327, 50-328), Watts Bar 1, 2 (50-390, 50-391), Bellefonte 1, 2 (50-438, 50-439), Nine Mile Point 1 (50-220)

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A. Violations

None.

B. Nonconformances

None.

C. Unresolved Items

None.

D. Other Findings or Comments

1. Observation of Cable Testing Activities

A high energy line break (HELB) Environmental Qualification test was being performed on various cables for the Tennessee Valley Authority (TVA) for use in Watts Bar and Sequoyah Nuclear Power stations. Twenty five cable specimens representing 8 manufacturers and several different insulation systems were included in the test program.

Prior to the test, the NRC inspector and the Sandia consultants (NRC Inspection Team) reviewed Wyle's Environmental Qualification Plan (QP) 17360-44 to verify that; (a) test instrumentation was adequately described, (b) test acceptance criteria was established, (c) environmental conditions were established and described (Pressure and temperature profiles and thermal aging), (d) all prerequisites for the given test had been met, (e) appropriate margins were applied and (f) QP was approved by TVA.

The NRC inspection team reviewed the test results of the dry runs which were performed using masses to simulate the cables.

The inspection team reviewed the test set-up to verify that: (a) specimens were located in the chamber as specified in the QP, (b) Instrumentation was calibrated, (c) accuracies of instrumentation were consistent with the requirements of the QP, (d) themocouples and pressure transducers were located in the chamber as specified in the QP., (e) Wyle Quality Assurance (QA) test monitor reviewed test set-up and stamped off the test log, and (f) functional tests were performed prior to the test.

The inspection team observed the initial ramp of the test on October 1, 1985 and made periodic checks during the remainder of the inspection. The ramp time for temperature and pressure was to be on a best effort basis. Approximately three minutes into the test one specimen failed, causing an open circuit.

The specimen was removed from the circuit and the circuit was reenergized. The on-site TVA representative authorized Wyle to extend the high temperature plateau an additional twelve minutes to compensate for the time the circuit was deenergized. This test anomaly was annotated on a Notice of Anomaly Form by the Wyle test engineer. Analysis and evaluation of the specimen will be conducted by Wyle and TVA at the end of the test when the test chamber is opened. The NRC inspection team will follow up on the disposition of this anomaly during a future inspection. The NRC inspection team reviewed the functional test data after the first 26.4 hours and determined that the test data met the requirements of the QP.

2. Other Testing

An accident (HELB/LOCA) simulation test was being performed on GE EB-5/GE EB-25 terminal Blocks for Niagara Mohawk Power Company (NMPC) for use in Nine Mile Point Unit One. This test was performed to verify results of a previous analysis performed by NMPC. Two ramps were performed, the first to verify the analysis, the second for informational purposes.

The NRC inspection team reviewed Wyle Test Procedure (TP) 17655-PRO-3 to verify that: (a) test instrumentation was adequately described, (b) test acceptance criteria was established and (c) environmental conditions were established and described (pressure and temperature profiles).

The inspection team observed the set up prior to the performance of the second ramp to verify that: (a) instrumentation was calibrated, (b) accuracies of instrumentation were consistent with the requirements of the TP and (c) functional tests were performed prior to the test.

The NRC inspection team observed the second saturated steam ramp which was more severe than the first ramp on October 2, 1985. The second ramp was modified from the TP with spray introduced later in the profile which was approved by the customer. A leakage current of approximately 50 ma was measured in one of the two instrument circuits during the ramp. NMPC and Wyle are planning to analyze the results. The NRC inspection team will review the first and second ramp test data on a future inspection.

3. Visual Inspection of Various Cables and Terminal Blocks

The NRC Inspection Team visually inspected various cables and viewed photographs of terminal blocks that had completed the qualification sequence as specified in QP 17460-45. The cables are used inside containment at Sequoyah and Watts Bar Nuclear Power Plants. Five cable specimens representing four manufacturers were included in the test program.

The cables were still wrapped on the same mandrels used during qualification and were identified. The cables showed no significant damage. The NRC inspection team viewed photographs of the terminal blocks taken after the accident test. Ten of the twenty terminal block terminations were coated with DOW RTV 3140. The terminal blocks showed no significant damage.

A test report was being prepared and will be reviewed during a future inspection. The NRC inspection team however did review QP 17460-45 to verify that: (a) test instrumentation was adequately described, (b) acceptance criteria were established, (c) environmental conditions were established and described (pressure and temperature profiles and thermal aging) and (d) appropriate margins were applied.

4. Technical Evaluation

The NRC Inspection Team performed a technical evaluation and review of test program 17521-1 for qualification of motor insulation systems used inside containment for use in Nuclear Power Plants for TVA. Documents examined were: Test report (TR), QP, Data Sheets and a Letter. The NRC Inspection Team reviewed the qualification prescribed in the QP and reviewed test results, including the basis for accelerated thermal aging and radiation. The TR and QP and related engineering documents were examined to verify the following:

a) Adequate test instrumentation and their accuracies were described and used to meet the requirements of IEEE-STD-323/1974.

b) Equipment interfaces were addressed.

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

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c) Test acceptance criteria were established as described in the test specification or in the design engineering documents, such as calculations and engineering letters to meet the requirements of IEEE-STD-323/1974.

d) Same equipment was used for all phases of testing and represented a standard production item.

e) Environmental conditions were established and described (e.g., pressure and temperature profiles, and thermal aging factors were consistent with those outlined in the test specification or test plan.)

f) Test results were adequately reduced and evaluated against established acceptance criteria described in customer test specifications or purchase orders.

g) All prerequisites for the given tests as outlined in the test specification had been met.

h) Test equipment included a description of all materials, parts, and subcomponents.

i) Notices of Anomalies were properly documented.

j) Appropriate margins were applied.

No nonconformances were noted during this review.

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

REPORT NO.: 99900902/85-04	INSPECTION DATE(S): 11/18-22/85	INSPECTION ON-SITE HOURS: 78

CORRESPONDENCE ADDRESS: Wyle Laboratories
Scientific Services and Systems Group
ATTN: Mr. W. W. Holbrook, General Manager
Eastern Test and Engineering Operations
7800 Governors Drive
Huntsville, Alabama

ORGANIZATIONAL CONTACT: Mr. E. W. Smith, Director, Contracts and Purchasing
TELEPHONE NUMBER: (205) 837-4411

PRINCIPAL PRODUCT: Research, engineering, and test operations.

NUCLEAR INDUSTRY ACTIVITY: Wyle Laboratories; Huntsville, Alabama, provides a variety of nuclear services to the industry. These services include environmental and seismic qualification testing of safety-related equipment, refurbishment and recertification of valves, valve and component flow testing, mechanical and hydraulic snubber testing, decontamination, and repair.

ASSIGNED INSPECTOR: Randolph N. Moist 5FEB86
R. N. Moist, Equipment Qualification Inspection Section (EQIS) — Date

OTHER INSPECTOR(S): R. H. Lasky, EQIS
J. Grossman, Sandia National Laboratories

APPROVED BY: Uldis Potapovs 2-6-86
U. Potapovs, Chief, EQIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection consisted of: (1) a technical evaluation of equipment qualification (EQ) test activities for safety-related equipment (2) witnessing EQ testing; and (3) verification of implementation of the quality assurance (QA) program.

PLANT SITE APPLICABILITY: Sequoyah-1, 2 (50-327, 50-328), Watts Bar 1, 2 (50-390, 391), Nine Mile Point 1 (50-220)

A. Violations

None.

B. Nonconformances

None.

C. Unresolved Items

None.

D. Other Findings or Comments

1. Technical Evaluation of Test Results

The NRC inspectors and Sandia Consultant (NRC inspection team) evaluated test results of two cable tests performed under Qualification Plan (QP) 17460-44. The cable tests were conducted for Tennessee Valley Authority (TVA) for use in Watts Bar and Sequoyah Nuclear Power Stations.

Twenty five cable specimens representing eight manufacturers and several different insulation systems were included in the test program for the first test. The NRC inspection team had previously witnessed a portion of a high energy line break (HELB) environmental test on October 1, 1985. Approximately three minutes into that test, one specimen (39-2) failed, causing an open circuit. Since the NRC's last visit, Wyle personnel performed a failure analysis and determined that a defect at the point where the jacket material was cut for splicing was the cause of failure. Failure was attributed to damage done to the insulation during installation and not as a result of the test exposure. This cable was subsequently retested during the second cable test and performed as required.

Sixteen cable specimens representing nine manufacturers and several different insulation systems were included in the test program for the second test.

Review of the test results by the NRC inspection team showed that two cable specimens (8-1, 8-2) lost voltage during the HELB environmental test. Failure analysis by Wyle for specimen 8-1 (20 year accelerated aging) determined that a faulty instrumentation

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cable through the penetration and a defect at the point where the jacket material was cut for splicing on the cable specimen was the cause of failure. However, it could not be determined which fault occurred first. Failure was attributed to damage done to the insulation during installation and not as a result of the test exposure. Wyle retested specimen 8-1 during this visit. Observation of the test is discussed in paragraph D.2 of this report.

Wyle test personnel began isolating conductors on specimen 8-2 (40 year accelerated aging) and experienced continued failures. Upon inspection by Wyle personnel, a hole was found in the jacket with indications that as a conductor failed it damaged adjacent conductors causing them to fail subsequently. Wyle and TVA are still analyzing and evaluating this cable failure.

Two other cable specimens (24-2, 42-2) had to be retested due to test personnel failing to perform the last power check during the HELB test. During retest both cable specimens would not maintain voltage before heat up of the chamber. Steam was then introduced into the chamber and temperature was maintained at 104°F for two hours. Both specimens would not hold voltage. A failure analysis was conducted by Wyle for speciment 42-2 (40 year accelerated aging) however, the cause of failure was not determined. Wyle's customer requested a post design basis accident hi-pot test to be performed on specimen 42-2. The specimen failed the hi-pot and Wyle's customer indicated specimen 42-1 (20 year accelerated aging) will be tested for qualification. Failure analysis of cable specimen, 24-2 (40 year accelerated aging) by Wyle determined that a faulty instrumentation cable through the penetration caused the failure. Failure was attributed to damage done to the insulation during installation and not as a result of test exposure. Specimen 24-2 was put back into the chamber at 104°F and held voltage for one hour. No further testing was anticipated by Wyle.

The NRC inspection team reviewed other test results for both tests which included the normal plus accident radiation exposure and thermal aging. No nonconformances were noted. The test report for this testing was being prepared by Wyle and will be reviewed during a future inspection.

The NRC inspection team visually inspected various cables from both tests that had completed the qualification test sequence as described in QP 17460-44. The cables were still wrapped on the same mandrels used during qualification and were identified. The cables showed no significant damage.

2. Observation of HELB (Retest)

A HELB environmental qualification test was being performed on cable specimen 8-1 (20 year accelerated aging) for TVA. This specimen was originally powered at 528V-18 amps during first HELB test conducted on October 1, 1985 which subsequently failed. The failure analysis for this specimen was discussed in paragraph D.1. This retest was performed as prescribed in QP 17460-44 except the cable specimen was powered at 133V-1A. Wyle's customer determined that most applications of the cable were at 133V-1A.

The NRC inspection team observed the test set-up prior to the performance of the ramp to verify that, (a) instrumentation was calibrated, (b) accuracies of instrumentation were consistent with the requirements of the QP, and (c) Wyle Quality Assurance test monitor reviewed test set-up and stamped off the test log.

The NRC inspection team observed the initial ramp of the test on November 21, 1985 and made periodic checks during the remainder of the inspection. Prior to the NRC inspection team departure on November 22, 1985 the cable specimen was performing as prescribed in the QP. Test results will be reviewed during a future inspection.

3. Visual Inspection of Terminal Blocks

The NRC inspection team viewed photographs of GE EB-5/EB-25 terminal blocks that had completed the qualification sequence as specified in test procedure 17655-PRO-3. The terminal blocks were tested for Niagara Mohawk Power Company for use in Nine Mile Point unit one.

The GE EB-5 terminal block showed no visual evidence of corrosion on the terminal screws, however, a very light, powdery rust colored residue covered the terminal connections.

The GE EB-25 terminal blocks showed evidence of external corrosion on the terminal hold-down screws. The marker strip was also discolored, brittle and physically cracked.

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A preliminary test report had been prepared and submitted to Niagara Mohawk Power Company. The NRC inspection team will review the final test report during a future inspection. With respect to the test results, Wyle personnel indicated that the current of approximately 50 ma measured in the output of an instrument circuit (nominal reading of 12 ma) and discussed during a previous NRC inspection was not a leakage current as originally suspected. The cause of the high current output had not been determined by Wyle during this visit.

4. Technical Evaluation

The NRC inspection team performed a technical evaluation and review of test program 17460-39 for qualification of a Masoneilan Electropneumatic valve positioner for TVA for use in Sequoyah Nuclear Power generating plants units 1 and 2. Documents examined were: Test Report (TR), Qualification Plan (QP), data sheets and a purchase order. The NRC inspection team reviewed the qualification prescribed in the QP and reviewed test results, including the basis for accelerated thermal aging and radiation. The TR and QP and related engineering documents were examined to verify the following:

a) Adequate test instrumentation and their accuracies were described and used to meet the requirements of IEEE-STD-323/1974.

b) Equipment interfaces were addressed.

c) Test acceptance criteria were established as described in the test specification or in the design engineering documents, such as calculations and engineering letters to meet the requirements of IEEE-STD-323/1974.

d) Same equipment was used for all phases of testing and represented a standard production item.

e) Environmental conditions were established and described (e.g., pressure and temperature profiles, and thermal aging factors were consistent with those outlined in the test specification or test plan.)

f) Test results were adequately reduced and evaluated against established acceptance criteria described in customer test specifications or purchase orders.

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

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g) All prerequisites for the given tests as outlined in the test specification had been met.

h) Test equipment included a description of all materials, parts, and subcomponents.

i) Notice of Anomaly reports were properly documented.

j) Appropriate margins were applied.

No nonconformances were noted during this review.

ORGANIZATION: YARWAY CORPORATION
BLUE BELL, PENNSYLVANIA

REPORT NO.: 99901012/85-01	INSPECTION DATE(S): 10/30/85	INSPECTION ON-SITE HOURS: 6

CORRESPONDENCE ADDRESS: Yarway Corporation
ATTN: Mr. Richard Rose
Vice President, Manufacturing
Blue Bell, Pennsylvania 19422

ORGANIZATIONAL CONTACT: Mr. Frank Peszka
TELEPHONE NUMBER: (215) 825-2100

PRINCIPAL PRODUCT: Valves

NUCLEAR INDUSTRY ACTIVITY: Less than 5%.

ASSIGNED INSPECTOR: J. C. Harper — 2/11/86
J. C. Harper, Reactive Inspection Section (RIS) — Date

OTHER INSPECTOR(S): P. Cortland

APPROVED BY: E. W. Merschoff — 2/11/86
E. W. Merschoff, Chief, RIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 21

B. SCOPE: To review the technical aspects of reported valve stem problems with Yarway valves.

PLANT SITE APPLICABILITY: Midland Plant, Units 1 & 2, 50-3294, 50-330; Grand Gulf Nuclear Station Units 1 & 2, 50-416 & 50-417; Susquehanna Units 1 & 2, 50-387 & 50-388; V. C. Summer Nuclear Station 50-395; Clinton Power Station Units 1 & 2, 50-461 & 50-462.

ORGANIZATION: YARWAY CORPORATION
BLUE BELL, PENNSYLVANIA

REPORT NO.: 99901012/85-01	INSPECTION RESULTS:	PAGE 2 of 3

A. Inspection Issues

A Part 21 notification was made to the NRC on September 26, 1985 concerning cracked valve stems on 1/2 inch and 3/4 inch weldbond valves. Although the cracked stems were not involved in nuclear plant service, stems from the same heat were sold to multiple nuclear plants.

Yarway concluded that the stem cracking and subsequent leakage "...is caused by a void in the bar stock used to manufacture the stems...." Therefore, they consider this incident an isolated case.

B. Background Information

On September 26, 1985, Yarway Corporation issued a Part 21 notification concerning a cracked valve stem in 1/2 inch and 3/4 inch weldbond valves. Specifically, a Houston Light & Power (HL&P) fossil power plant reported leakage of a newly installed valve to Yarway in July 85, and five other stems cracked during a non-nuclear hydrotest at Yarway. All stems were manufactured from the same heat of material. Yarway determined that additional stems from this heat were sold to Grand Gulf, Susquehanna, V. C. Summer, and Clinton. Yarway has notified these plants of the potential problem and has recommended replacement of the valves. As of December 1985, Yarway has not received any reports of stem leakage from the identified nuclear facilities.

The bar stock used to manufacture the stems was 5/8" round bar, martensitic stainless steel type 416, ASTM A582-75 condition T, heat number 93876. This grade contains a relatively high sulfur content in order to improve machinability. The bar stock originated from the Al Tech Specialty Steel Corporation where an oversize bar is heat treated (1850°F - 1 hr. - oil quenched) and tempered (1025°F - 6 hrs. - air cool) according to ASTM 582-75, eddy current tested and ground down approximately 1/32 inch to size. Subsequently, the bar stock was supplied to P.A. Frasse and Co., Inc., then to Yarway who threads and inspects the stems for surface finish.

The stem hardness was within specification at Brinell 302. Mr. Bill Toter of Yarway indicated that transverse microhardness testing across the cross section of the stem revealed uniform hardness properties which were within specification. The chemical analysis for carbon, manganese, sulphur, and silicon were all within specification. Both the carbon content and hardness were at the upper limit of the specification.

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Conclusions

Upon visual examination of the cracked stems, the NRC inspectors found a crack running the entire length of the stem. No bulk elongation was evident. Micrographic analysis of the stem transverse cross section at 50X and 100X revealed a martensitic grain structure with uniform randomly spaced spheriodized manganese sulfide inclusions. There was no apparent evidence that the cracks preferred initiation at the inclusions. There was no evidence of stringers or banding. Evaluation of the microstructure revealed that the heat treatment appeared to be adequate.

The cracks were viewed at 100X and generally appeared to be straight with little or no branching. The space between crack faces appeared very tight at the outside diameter and progressively wider at the inside diameter. Therefore, it appears that the crack initiation occurred internally.

From the given information, and assuming the heat treatment was carried out as certified, it appears that internal inherent flaws combined with severe internal residual stresses caused the cracks to initiate at the flaws and propagate. The source of these residual stresses may be from cold working such as thread machining (or excessive thread machining) and hydrotesting. As a result of the crack appearance and the normal microstructure for this material and heat treatment, the failure appeared not to be a result of material selection or heat treatment. These conclusions are consistent with Yarway's determination of the problem.

E. Persons Contacted

Roy G. Chew - Manager of Quality Assurance, Yarway Corp.
Frank Peszka - Manager of Quality Systems, Yarway Corp.
William F. Toter - W..ding Engineer, Yarway Corp.
George Papson - Product Manager, Yarway Corp.
Jim Wiggin - Region I, US NRC
Hal Gregg - Region I, US NRC
Ed Daily - Phone Contact-Al Tech Speciality Steel Corp.

F. Exit Interview

At the conclusion of the inspection, the inspectors met with the persons identified in Section E (with the exception of Mr. E. Daily) and discussed the scope and findings of the inspection.

ORGANIZATION: ZETEC, INCORPORATED
ISSAQUAH, WASHINGTON

REPORT NO.: 99901037/85-01	INSPECTION DATE(S): 12/9-13/85	INSPECTION ON-SITE HOURS: 56

CORRESPONDENCE ADDRESS: Zetec, Inc.
ATTN: Mr. Clyde Denton
General Manager
Post Office Box 140
Issaquah, Washington 98027

ORGANIZATIONAL CONTACT: Mr. A. L. Lucero, QA Manager
TELEPHONE NUMBER: (206) 392-5316

PRINCIPAL PRODUCT: Eddy current test systems

NUCLEAR INDUSTRY ACTIVITY: Approximately 50 percent

ASSIGNED INSPECTOR: J. T. Conway — 1-17-86
J. T. Conway, Reactive Inspection Section (RIS) — Date

OTHER INSPECTOR(S): J. C. Harper, RIS

APPROVED BY: E. W. Merschoff — 1-30-86
E. W. Merschoff, Chief, RIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was made to assess the implementation of the QA program and to review QA records pertaining to calibration services performed by Zetec on eddy current testing (ET) equipment.

PLANT SITE APPLICABILITY: Not identified during the inspection.

ORGANIZATION: ZETEC, INCORPORATED
ISSAQUAH, WASHINGTON

REPORT NO.: 99901037/85-01	INSPECTION RESULTS:	PAGE 2 of 9

A. VIOLATIONS:

1. Contrary to Section 21.6 of 10 CFR Part 21, copies of Section 206 of the Energy Reorganization Act and Procedure No. ZAG-16 "Reporting of Safety Hazards" were not posted along with a copy of 10 CFR Part 21 posted in the main building, and none of the documents were posted in another building containing the Machine Shop and Specialty Shop. (85-01-01)

2. Contrary to Section 21.31 of 10 CFR Part 21, Zetec did not pass on the requirements of 10 CFR Part 21 to Westinghouse Specialty Metals Division who supplied inconel tubing to Zetec in August 1982. The tubing was fabricated into a calibration standard and shipped to Rochester Gas & Electric (ref. PO N-EG-45426 dated September 5, 1984 which imposed Part 21 requirements upon Zetec). (85-01-02)

B. NONCONFORMANCES:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 2.4.3 of the Quality Assurance Manual (QAM), there was no documented evidence that the QA Manager reviewed two purchase orders (PO) containing QA requirements. The POs were from Westinghouse (MM-22051-M-XX dated June 3, 1985) and Rochester Gas & Electric (N-EG-45426 dated September 5, 1984). (85-01-03)

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Sections 4.4.2.4 and 4.4.2.5 of the QAM, a review of 17 POs for major components revealed the following (85-01-04):

   a. PO No. EL 13185 to Gould for strip chart records and PO No. EL 92485 to Allen Engineering for camera systems did not contain QA requirements, and there was no documented evidence of a QA Manager review.

   b. Fifteen POs did not reference the "QA Requirements" attachment. The POs were to Hewlett-Packard (30485, 82185, and 102485); Allen Engineering (61485); Koyo International of America (71185); American Music (51685); Standard Power (82884-B); Tektronix (91785, 92884, 102585, and 111585); and Advanced Digital Information (21185, 32185, 32585, and 51685).

ORGANIZATION: ZETEC, INCORPORATED
ISSAQUAH, WASHINGTON

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3. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 18.4.1B of the QAM, there was no documented evidence that a field service audit was performed on field inspection No. 840324 at the Yankee Rowe nuclear facility in April and May 1984. (85-01-05)

4. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 7.4.3.3 of the QAM, during an inspection of the manufacturing facility, it was noted in a holding area that 29 containers with camera systems from Allen Engineering and four containers with graphics printers from Hewlett-Packard were not appropriately tagged following receipt inspection approval. (85-01-06)

5. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 7.6.2 of the QAM, a review of POs for major components indicated that three items were purchased from suppliers who were not on the Approved Supplier List dated January 25, 1985. The suppliers were American Music - tape recorders (PO 51685), Allen Engineering - TV cameras (POs 61485 and 92485), and Koyo International of America - video monitors (PO 71185). (85-01-07)

6. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 8.4.3.1 of the QAM, it was noted that Burn-in tags for four items (i.e., M1Z-12 remote/amp and power supply, OMB II, and SM4) on Rochester Gas & Electric PO N-E6-45426 did not have "N/A" identified in not applicable blanks. (85-01-08)

7. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 18.6.1 of the QAM, there was no documented evidence to show that the individual who audited XTEX on an annual basis from 1980 thru 1984 was trained or qualified to be an auditor. (85-01-09)

8. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 2.3.3.1 of the QAM, Section 9.6.1 of SNT-TC-1A, and Section 7.4.1 of Procedure No. Z-QA 101, a review of the qualification records of 32 NDE personnel revealed that the qualification records did not contain a statement indicating satisfactory completion of training in accordance with Procedure No. Z-QA 101. (85-01-10)

9. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 4.4 of Procedure No. Z-QA 101, there was no record of an eye examination for inspector Nissley for the period of December 26, 1980 to February 5, 1981 during which time the inspector was on a job at the Millstone nuclear facility. (85-01-11)

10. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 17.4.1 of the QAM, there was no record of a calibration certificate (Form No. Z-QA 8A) for instrument FM 22-4 (S/N 016) which had been calibrated for Power Inspection in February 1984. (85-01-12)

11. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Section 2.3.1.1 of the QAM, Section 13 of ANSI N45.2, and Section 3.0 of Procedure No. SSP-TCS, there was no documented evidence to show that the three gage blocks (S/Ns 810135, 770198, and 800145) in the Specialty Shop were ever calibrated against certified equipment traceable to the National Bureau of Standards. (85-01-13)

C. OPEN ITEMS:

None.

D. OTHER FINDINGS OR COMMENTS:

1. 10 CFR Part 21

Procedure ZAG-16 relating to the reporting of defects and failures was reviewed, and the implementation of the procedure in regard to posting requirements was evaluated by inspecting the shop areas. It was noted that a copy of 10 CFR Part 21 was the only document posted on a bulletin board in Zetec's main building. Another building, remote from the main building, contained the Machine Shop and Specialty Shop, but there were no documents (i.e., procedure ZAG-16, Section 206 of the Energy Reorganization Act, and Part 21) posted. (See Violation 85-01-01)

2. Training/Qualification

The training records for fabricators, NDE personnel, and auditors were reviewed to assure that personnel performing and verifying activities affecting quality were trained and qualified. The records of eight fabricators who work in the manufacturing of ET equipment indicated that each individual had successfully completed an eight hour course in "Soldering Technology." The training course completion record was signed off by the QA Manager.

Zetec's personnel qualification and certification procedure Z-QA 101 and qualification records for 32 examiners (four-Level III, three-Level I, and 25-Level IIA) were reviewed. The Level IIA certification is comparable to a Level II, but the individual is also certified to analyze data tapes. The training and certification of personnel in ET complies with SNT-TC-1A, 1980 Edition. With the exception of one individual who performed work at Millstone during a period when his annual eye examination was overdue, the physical examinations of the remaining examiners were satisfactory (see Nonconformance 85-01-11). Although the qualification records contained certifications and copies of the examinations, none of the records contained a statement that each individual had satisfactorily completed training in accordance with procedure Z-QA 101 (see Nonconformance 85-01-10).

The QA Manager trained three individuals using reference material (i.e., QA Manual and ANSI N45.2 and N45.2.12) and certified them as auditors in September 1984. There was no documented evidence that one individual who audited XTEX, a calibration service vendor, on an annual basis from 1980 thru 1984 was trained or had the qualifications to be an auditor (see Nonconformance 85-01-09). None of the individuals were qualified in accordance with ANSI N45.2.23 or similar criteria to organize and direct audits, report audit findings, and evaluate corrective action.

It was also noted that a formal and documented indoctrination into the aspects of the QA program was not given to any of the individuals performing quality affecting activities (e.g., purchasing, engineering, calibrating, manufacturing, and inspecting/testing).

3. Major Components

Beginning in 1985, major components requiring high quality levels were procured with written POs and went through a receiving inspection. The major components include computers, tape recorders, and graphics printers from Hewlett-Packard; strip chart recorders from Gould; tape recorders from American Music; camera systems from Allen Engineering; video monitors from Koyo International of America; power supplies from Standard Power; data cartridge recorders from Advanced Digital Information; and oscilloscopes from Textronix.

Seventeen Zetec POs for major components were reviewed, and it was noted that two POs (EL 13185 to Gould and EL 92485 to Allen Engineering) did not identify any QA requirements or show evidence of a review by the QA Manager (see Nonconformance 85-01-04). In addition, 15 of the POs failed to reference the "QA Requirements" attachment (see Nonconformance 85-01-05). It was noted that three of the eight vendors were not on an Approved Vendor List dated January 25, 1985 and signed by the QA Manager (see Nonconformance 85-01-07). Pre-award evaluations and audits of all eight vendors were not performed by Zetec.

Receipt inspection of the major components was performed by two individuals from the QA organization in accordance with Procedure No. QAP-6. It was noted that the inspection of all the components purchased on the 17 POs was documented on a Receiving Inspection Report (RIR). The suppliers/manufacturers were requested to supply a Certificate of Compliance (CC) with each shipment. Although item No. 5 "Documentation: Certificates" of the RIR was signed acceted, it was noted that CCs were missing for four computers from Hewlett-Packard (PO EL 30485), 25 camera systems from Allen Engineering (PO EL 61485), and five oscilloscopes from Textronix (POs EL 10285, 111585, and 91785). Following acceptance of the component, an "Accepted Tag" is initiated and attached to the component or its container. During an inspection of the manufacturing facility, it was noted that 29 containers with camera systems from Allen Engineering and 4 containers with graphics printers from Hewlett-Packard had been accepted at receipt inspection but were not appropriately tagged in a hold area (see Nonconformance 85-01-06).

4. Field Service Work

The inspector reviewed 12 files related to field service work performed by Zetec at nuclear power plants. The sample of jobs included eight in 1984 and four in 1985. Documents in each file consisted of customer POs, Equipment Check Lists, personnel certifications, invoices, and inspection reports, where applicable.

On job No. 84060, eight individuals were supplied to CON AM for work at Zion Unit No. 1. Allen Nuclear Associates performed ET at Trojan using nine individuals and leased equipment from Zetec (job No. 840644). Job No. 850734 consisted of supplying ten individuals and a ET analyst system to Baltimore Gas & Electric for work at Calvert Cliffs. Zetec performed ET of steam generators Nos. 2, 3, and 4 at Yankee Rowe on job No. 840324. Eight examiners were sent to Combustion Engineering (CE) on job No. 850843 at Maine Yankee. Equipment and 12 examiners were used at Millstone Unit No. 3 by Northeast Utilities (job No. 850525). Job Nos. 841062 and 850313 included equipment and personnel to CE for work at Sequoyah Unit No. 2 and Arkansas Unit No. 2, respectively. Westinghouse utilized Zetec personnel at North Anna Unit No. 2, Turkey Point Unit No. 4, and Salem Unit No. 1 on job Nos. 840856, 840533, and 840426, respectively. CE used examiners at Fort Calhoun on job No. 840537.

5. Audits

Three types of audits are performed by Zetec. They include: (a) facility audit performed twice each calendar year and documented on Z-QA 16, (b) field service audit reported on Z-QA 4, and (c) vendor audits. For facility audits, the NRC inspector reviewed the 1985 Audit Log Book, the September 1985 Audit Plan, and the "Zetec Facility Audit Checklist" Form No. Z-QA 16. Internal audits for December 1984 and February and September 1985 were reviewed and found adequate. It was noted that internal audit reports prior to 1983 did not exist. Although the latest revision of the QA Manual requires that audit reports be retained for a minimum of ten years, it was pointed out by the QA Manager that prior to 1983 there was no requirement to retain audit reports for any period of time.

Twenty-six field service audits were reviewed. They were performed of field inspections at the following nuclear facilities: St. Lucie No. 1 (six in November), Ginna (one in March 1984), Calvert Cliffs (seven in April and May 1985), and Millstone No. 3 (12 in May and June 1985). There was no documented evidence that field service audits were conducted at Yankee Rowe in April and May 1984 for field inspection job No. 840324 (see Nonconformance 85-01-05). Personnel performing field inspections and not complying with "In-Plant Audit Report" was addressed in corrective action request No. QA-2 dated October 2, 1985.

It was noted that Zetec only performs external audits on vendors providing calibration services for Zetec test equipment. The audit check list used for external audits contained four categories - Equipment Certification Records, Voltage Calibration

Source, Frequency Calibration Source, and General Workmanship. The only vendor who calibrates test equipment is XTEX Corporation, and records show that they were audited on an annual basis starting in 1980.

Zetec uses a number of suppliers/manufacturers to furnish tubing which is fabricated into reference standards by Zetec. Some of the vendors include Tube Sales, Ducommon, Kilsby Tube Supply, Alaskan Copper & Brass, H. M. Hillman, Tech-Metals, PAC Stainless, and Westinghouse - Speciality Metals Division. Zetec has never performed a pre-award evaluation and/or annual audits of these tubing vendors.

6. Calibration of Measuring & Test Equipment (M&TE)

The NRC inspector checked for up-to-date calibration of randomly selected instruments in the Zetec shop area. A Tektronix oscilloscope (S/N B011334), MIZ-12 Display Module (S/N 215), MIZ-12 Timer/Driver (S/N 081), and a 3968AZ FM tape recorder (S/N 006) were all adequately calibrated, and the calibration was traceable to the National Bureau of Standards (NBS).

Ten master controlled copies of procedures (CSP-ZQA8, CSP-ARC, CSP-EM3300, CSP-BR220, CSP-HP3968, CSP-FRQ12-2, CSP-FD17, CSP-DA17, CSP-MIX17, and CSP-HP26716) were all checked against corresponding calibration procedures at selected work stations. All the stations checked had the current revision of the applicable calibration procedures.

During a previous inspection at another company who performs ET, the NRC inspector obtained a copy of a Zetec calibration certificate for instrument FM 22-4 (S/N 016). Upon cross checking Zetec's records for the original of this certification, the original could not be located (see Nonconformance 85-01-12).

During an inspection of the Specialty Shop where tubing is fabricated and certified as ET tubing standards, it was noted that several M&TE are used to measure the various dimensions of a calibration standard. The standard is manufactured from tubing the same size and material as will be examined in the vessel. Eight M&TE ranging from a linear gauge to a tape recorder are used. With the exception of three gauge blocks, the remaining seven items were tagged with calibration stickers, and a review of records in the calibration shop verified that the seven items were calibrated traceable to the NBS (see Nonconformance 85-01-13).

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7. Documentation Packages - Equipment

Twelve documentation packages from nuclear customers for ET equipment were reviewed. The nuclear customers were CE, Omaha Public Power District, Westinghouse, Rochester Gas & Electric, General Atomic, Babcock & Wilcox, and Florida Power & Light. The documentation packages consisted of a customer PO, shipping paper, packing check-off list, manufactured tags, final inspection report, packing paper, and an invoice.

It was noted that Westinghouse PO MM-22051-M-XX dated June 3, 1985 and PO N-EG-95426 dated September 5, 1984 from Rochester Gas & Electric imposed 10 CFR Part 21 and ANSI N45.2 requirements upon Zetec. Neither of these POs was reviewed by the QA department to assure Zetec compliance with the quality requirements (see Nonconformance 85-01-03). In addition, manufactured tags which control burn-in and check-out of equipment were not adequately documented in accordance with the "Tag System" procedure (see Nonconformance 85-01-08).

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VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

PLANTS / VENDORS	ARKANSAS 1 & 2	BEAVER VALLEY 1 & 2	BELLEFONTE 1 & 2	BRAIDWOOD 1 & 2	BROWNS FERRY 1 2 & 3	CALVERT CLIFF 1 & 2	CATAWBA 1 & 2	CLINTON 1 & 2	COMANCHE PEAK 1 & 2	COOK 1 & 2	CRYSTAL RIVER 3	DAVIS BESSE 1	DIABLO CANYON 1 & 2	FERMI 2	FORT CALHOUN 1	GRAND GULF 1 & 2
AIR BALANCE INCORPORATED		X		X				X	X	X	X	X				
BABCOCK & WILCOX NUCLEAR POWER DIVISION																
COMBUSTION ENGINEERING (MULTIPLE INCLUDING)						X										
DEITRICH STANDARD CORPORATION															X	
DRESSER INDUSTRIES, INC.												X	X		X	
ELGAR CORPORATION		X							X							
GESELLSCHAFT FUR NUKLEAR SERVICE																
GENERAL ELECTRIC COMPANY NEB OPERATIONS														X		---
JOSEPH T. RYERSON & SON CHICAGO,IL						X										
JOSEPH T. RYERSON & SON PHILADELPHIA, PA						X										
ILLINOIS FABRICATORS, INC								461								

X-APPLIES TO ALL PLANTS DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	ARKANSAS 1 & 2	BEAVER VALLEY 1 & 2	BELLEFONTE 1 & 2	BRAIDWOOD 1 & 2	BROWNS FERRY 1 2 & 3	CALVERT CLIFF 1 & 2	CATAWBA 1 & 2	CLINTON 1 & 2	COMANCHE PEAK 1 & 2	COOK 1 & 2	CRYSTAL RIVER 3	DAVIS BESSE 1	DIABLO CANYON 1 & 2	FERMI 2	FORT CALHOUN 1	GRAND GULF 1 & 2
JOHNSTON PUMP COMPANY																
JOSEPH OAT CORPORATION																
NUS CORPORATION																
NATIONAL TECHNICAL SERV. HARTWOOD																
NUCLEAR ENERGY SERVICES															X	
PAUL-MUNROE ENERGY PRODUCTS DIV	368															
POWER INSPECTION, INC.		334														
PACIFIC SCIENTIFIC CO (MULTIPLE INCLUDING)																
ROBERT JAMES SALES, INC.														X		
ROCHESTER INSTRUMENT SYSTEMS																
SQUARE D COMPANY																

X-APPLIES TO ALL PLANTS DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	ARKANSAS 1 & 2	BEAVER VALLEY 1 & 2	BELLEFONTE 1 & 2	BRAIDWOOD 1 & 2	BROWNS FERRY 1 2 & 3	CALVERT CLIFF 1 & 2	CATAWBA 1 & 2	CLINTON 1 & 2	COMANCHE PEAK 1 & 2	COOK 1 & 2	CRYSTAL RIVER 3	DAVIS BESSE 1	DIABLO CANYON 1 & 2	FERMI 2	FORT CALHOUN 1	GRAND GULF 1 & 2
VALLEY STEEL PRODUCTS CO (NO UNITS IDENTIFIED)																
WESTINGHOUSE NUC SER INT DIV															X	---
WESTINGHOUSE NUC TECH DIV		X														
WYLE LABORATORIES HUNTSVILLE,AL 85-03			X		X											
WYLE LABORATORIES HUNTSVILLE,AL 85-04																
YARWAY CORPORATION								X								X
ZETEC INCORPORATED (NO UNITS IDENTIFIED)																

X-APPLIES TO ALL PLANTS   DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	HATCH 1 & 2	HOPE CREEK 1 & 2	JA FITZPATRICK	LASALLE CONNTY 1 & 2	LIMERICK 1 & 2	MAINE YANKEE	MIDLAND 1 & 2	MILLSTONE 1 2 & 3	MONTICELLO	NINE MILE PT 1 & 2	OYSTER CREEK	PALISADES	PALO VERDE 1 2 & 3	PERRY 1 & 2	RANCHO SECO 1	SAN ONOFRE 1 2 & 3
AIR BALANCE INCORPORATED																
BABCOCK & WILCOX NUCLEAR POWER DIVISION															X	
COMBUSTION ENGINEERING (MULTIPLE INCLUDING)						X							X			
DEITRICH STANDARD CORPORATION								423								
DRESSER INDUSTRIES, INC.														X		
ELGAR CORPORATION								423								
GESELLSCHAFT FUR NUKLEAR SERVICE																
GENERAL ELECTRIC COMPANY NEB OPERATIONS		X		X	X				X		X			X		
JOSEPH T. RYERSON & SON CHICAGO,IL																
JOSEPH T. RYERSON & SON PHILADELPHIA, PA					X								X			
ILLINOIS FABRICATORS, INC																

X-APPLIES TO ALL PLANTS   DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

PLANTS / VENDORS	HATCH 1 & 2	HOPE CREEK 1 & 2	JA FITZPATRICK	LASALLE CONTY 1 & 2	LIMERICK 1 & 2	MAINE YANKEE	MIDLAND 1 & 2	MILLSTONE 1 2 & 3	MONTICELLO	NINE MILE PT 1 & 2	OYSTER CREEK	PALISADES	PALO VERDE 1 2 & 3	PERRY 1 & 2	RANCHO SECO 1	SAN ONOFRE 1 2 & 3
JOHNSTON PUMP COMPANY	X															
JOSEPH OAT CORPORATION																
NUS CORPORATION														X		
NATIONAL TECHNICAL SERV. HARTWOOD												X				
NUCLEAR ENERGY SERVICES																
PAUL-MUNROE ENERGY PRODUCTS DIV																
POWER INSPECTION, INC.												X				
PACIFIC SCIENTIFIC CO (MULTIPLE INCLUDING)														440		
ROBERT JAMES SALES, INC.			X							220						
ROCHESTER INSTRUMENT SYSTEMS																
SQUARE D COMPANY																361 362

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VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	HATCH 1 & 2	HOPE CREEK 1 & 2	JA FITZPATRICK	LASALLE CONNTY 1 & 2	LIMERICK 1 & 2	MAINE YANKEE	MIDLAND 1 & 2	MILLSTONE 1 2 & 3	MONTICELLO	NINE MILE PT 1 & 2	OYSTER CREEK	PALISADES	PALO VERDE 1 2 & 3	PERRY 1 & 2	RANCHO SECO 1	SAN ONOFRE 1 2 & 3
VALLEY STEEL PRODUCTS CO (NO UNITS IDENTIFIED)																
WESTINGHOUSE NUC SER INT DIV																
WESTINGHOUSE NUC TECH DIV								X								
WYLE LABORATORIES HUNTSVILLE,AL 85-03										220						
WYLE LABORATORIES HUNTSVILLE,AL 85-04										220						
YARWAY CORPORATION							X									
ZETEC INCORPORATED (NO UNITS IDENTIFIED)																

X-APPLIES TO ALL PLANTS DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	SEQUOYAH 1 & 2	SHOREHAM	ST. LUCIE 1 & 2	SUMMER	SURRY 1 & 2	SUSQUEHANNA 1 & 2	THREE MILE IS 1 & 2	TROJAN	TURKEY POINT 3 & 4	VOGTLE 1 & 2	WATTS BAR 1 & 2
AIR BALANCE INCORPORATED											
BABCOCK & WILCOX NUCLEAR POWER DIVISION											
COMBUSTION ENGINEERING (MULTIPLE INCLUDING)			X								
DEITRICH STANDARD CORPORATION										X	
DRESSER INDUSTRIES, INC.											
ELGAR CORPORATION											
GESELLSCHAFT FUR NUKLEAR SERVICE					X						
GENERAL ELECTRIC COMPANY NEB OPERATIONS		X								X	
JOSEPH T. RYERSON & SON CHICAGO, IL											
JOSEPH T. RYERSON & SON PHILADELPHIA, PA						X					
ILLINOIS FABRICATORS, INC											

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VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

PLANTS / VENDORS	SEQUOYAH 1 & 2	SHOREHAM	ST. LUCIE 1 & 2	SUMMER	SURRY 1 & 2	SUSQUEHANNA 1 & 2	THREE MILE IS 1 & 2	TROJAN	TURKEY POINT 3 & 4	VOGTLE 1 & 2	WATTS BAR 1 & 2
JOHNSTON PUMP COMPANY											
JOSEPH OAT CORPORATION							320				
NUS CORPORATION											
NATIONAL TECHNICAL SERV. HARTWOOD											
NUCLEAR ENERGY SERVICES											
PAUL-MUNROE ENERGY PRODUCTS DIV	X										
POWER INSPECTION, INC.											
PACIFIC SCIENTIFIC CO (MULTIPLE INCLUDING)											
ROBERT JAMES SALES, INC.											
ROCHESTER INSTRUMENT SYSTEMS		X						X			
SQUARE D COMPANY		X		X							

X-APPLIES TO ALL PLANTS   DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

VENDORS \ PLANTS	SEQUOYAH 1 & 2	SHOREHAM	ST. LUCIE 1 & 2	SUMMER	SURRY 1 & 2	SUSQUEHANNA 1 & 2	THREE MILE IS 1 & 2	TROJAN	TURKEY POINT 3 & 4	VOGTLE 1 & 2	WATTS BAR 1 & 2
VALLEY STEEL PRODUCTS CO (NO UNITS IDENTIFIED)											
WESTINGHOUSE NUC SER INT DIV											
WESTINGHOUSE NUC TECH DIV											
WYLE LABORATORIES HUNTSVILLE,AL 85-03	X										X
WYLE LABORATORIES HUNTSVILLE,AL 85-04	X										X
YARWAY CORPORATION				X		X					
ZETEC INCORPORATED (NO UNITS IDENTIFIED)											

X-APPLIES TO ALL PLANTS   DOCKET NO.-APPLIES ONLY TO THE IDENTIFIED UNIT

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NUREG-0040
Volume 9, No. 4

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Quarterly Report -- October 1985 - December 1985

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
December	1985

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
February	1986

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Office of Inspection and Enforcement
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Washington, D.C. 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

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October 1985-December 1985

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This periodical covers the results of inspections performed by the NRC's Vendor Program Branch that have been distributed to the inspected organizations during the period from October 1985 through December 1985. Also included in this issue are the results of certain inspections performed prior to October 1985 that were not included in previous issues of NUREG-0040.

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