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Vol. 10, No. 4

LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT

QUARTERLY REPORT
OCTOBER 1986 - DECEMBER 1986

UNITED STATES NUCLEAR REGULATORY COMMISSION

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LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT

QUARTERLY REPORT
OCTOBER 1986 - DECEMBER 1986

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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, DC 20555

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PREFACE

A fundamental premise of the Nuclear Regulatory Commission's (NRC) nuclear facility licensing and inspection program is that licensees are responsible for the proper construction and safe operation of their 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. 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 addition to the above, the Vendor Program Branch has begun inspections at licensee facilities covering the areas of procurement of replacement parts for use in safety-related systems and licensee/vendor interface programs as requested in Generic Letter 83-28. This edition of the White Book contains copies of the inspection reports of inspections completed to date. Subsequent issues will contain those reports that are issued in the quarterly report period covered by that White Book.

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 effect 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 also contains copies of I&E Information Notices, concerning vendor issues released during the calendar quarter.

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.

ORGANIZATION: COMPANY, DIVISION
CITY, STATE

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

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

ORGANIZATIONAL CONTACT: Name/Title
TELEPHONE NUMBER: Telephone Number

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

ASSIGNED INSPECTOR: _____ _____
Name/Vendor Program Branch Section Date

OTHER INSPECTOR(S): Name/Vendor Program Branch Section

APPROVED BY: _____ _____
Name/Chief - Section/Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: Pertain to the inspection criteria that are applicable to the activity being inspected; 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 plant name and docket numbers 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 of 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

NRC APPROVAL STATUS OF QA PROGRAM TOPICAL REPORT REVISIONS

PRINCIPAL	TOPICAL REPORT	REVISION	DATE OF LATEST NRC REVISION APPROVAL LETTER
Babcock & Wilcox	BAW 10096A	Revision 4	December 30, 1983
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 4A	March 14, 1986
Combustion Engineering	CENPD-210-A	Revision 6	*
Ebasco Services, Inc.	ETR-1001	Revision 12	May 4, 1984
Exxon Nuclear Company	XN-NF-1A	Revision 8	March 6, 1986
Framatome	FRA-QP/85 0782 NP	Revision 2A	March 14, 1986
General Atomic	GA-A13010A	Amendment #8	October 15, 1984
General Electric Co.	NEDO-11209-04A	Revision 6	July 30, 1986
Gibbs & Hill, Inc.	GIBSAR 17-A	Amendment 8	February 27, 1985
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

*Accepted on 6/27/86 through the provisions of 10 CFR 50.54(a)(3)(iv)

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

SAMPLE QA TOPICAL REPORT REVISION APPROVAL LETTER

(ADDRESSEE)

Gentlemen:

Your letter of _____ transmitted Revision _____ of your QA topical report _____, "_____ Quality Assurance Program." The report describes your QA program for design, procurement, fabrication, and testing activities.

We have reviewed Revision _____ of the report against the acceptance criteria in Section 17.1 of the NRC's Standard Review Plan for nuclear power plants (NUREG-0800, July 1981). Based on our review and evaluation of Revision _____, we find that the criteria in Appendix B to 10 CFR Part 50 are met. Revision _____ of your QA topical report is, therefore, acceptable, and should be implemented for safety-related applications. Our evaluation is enclosed.

Should regulatory criteria or regulations change such that our conclusions about this topical report are invalidated, we will notify you. You will be given the opportunity to revise and resubmit it should you so desire. We note the commitment in the Foreward of the report to keep the NRC informed of changes to the QA program description.

Please incorporate this acceptance letter and its enclosed evaluation into the topical report, identify the report as Revision _____A (the "A" indicating that it has been found acceptable by the NRC), and transmit a copy to:

Document Control Desk
ATTN: Vendor Program Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

By way of letters to the same address, please keep us informed of the nuclear units to which this QA topical report applies. Should you have any questions regarding our review or if we can provide assistance, please contact us on (301) 492-

Sincerely,

Chief
Vendor Program Branch
Division of Quality Assurance, Vendor
and Technical Training Center Programs
Office of Inspection and Enforcement

INSPECTORS REPORTS

ORGANIZATION: BABCOCK & WILCOX
LYNCHBURG, VIRGINIA

REPORT NO.: 99900400/86-02	INSPECTION DATES: 10/16/86	INSPECTION ON-SITE HOURS: 12

CORRESPONDENCE ADDRESS: Babcock & Wilcox, A McDermott Company
Nuclear Power Division
ATTN: J. H. Taylor
Manager, Licensing Services
Post Office Box 1260
Lynchburg, Virginia 24506-0935

ORGANIZATIONAL CONTACT: T. Stevens, Manager, Quality Assurance
TELEPHONE NUMBER: (804) 385-3138

NUCLEAR INDUSTRY ACTIVITY: Design and engineering services for B&W plants requesting reanalysis and modifications to existing systems, components and structures.

ASSIGNED INSPECTOR: R. P. Correia
R. P. Correia, Special Projects Inspection Section (SPIS) — 12-24-86 Date

OTHER INSPECTOR(S): P. D. Milano, SPIS

APPROVED BY: P. D. Milano for
J. W. Craig, Chief, SPIS, Vendor Program Branch — 12-29-86 Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: The purpose of the inspection was to review the Babcock & Wilcox (B&W) system for providing information to their nuclear customers pertaining to installation and operation of B&W supplied equipment.

PLANT SITE APPLICABILITY: Arkansas 1 (50-313), Belefonte 1 & 2 (50-438 & 439), Crystal River 3 (50-302), Davis-Besse 1 (50-346), Oconee 1, 2, & 3 (50-269, 270 & 287), Rancho Seco 1 (50-312), Three Mile Island 1, (50-289), Washington Nuclear 1 (50-460).

ORGANIZATION: BABCOCK & WILCOX
LYNCHBURG, VIRGINIA

REPORT NO.: 99900400/86-02	INSPECTION RESULTS:	PAGE 2 of 3

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

A review of previous inspection findings was not within the scope of this inspection.

E. INSPECTION REPORT SUMMARY:

During the NRC inspection, Babcock & Wilcox (B&W) staff involved with field service information discussed with the NRC inspectors the various programs in place at B&W. B&W has both formal and informal programs in which B&W nuclear customers are supplied with information applicable to the operation and/or installation of B&W supplied equipment. B&W has established a close relationship with it's nuclear licensees during the last three years and much of the information being provided has been done thru the B&W Owners Group (BWOG).

The B&W system for the receipt and evaluation of deficiencies potentially reportable under 10 CFR Part 21 provides the BWOG with a quarterly report of Potential Safety Concerns (PSC). The report contains the status in which current issues, new issues received since the last report and reported PSC's to the licensees are included. PSC's reported under 10 CFR Part 21 are matrixed in the quarterly status reports. Informal feedback from licensees addressing their actions and evaluations of PSC's are compiled and included in the PSC status reports. If a new PSC appears to be of high concern, B&W will issue a letter to the licensees notifying them and will continue with the evaluation. There is also an annual meeting held in which PSC's are reviewed by the BWOG.

The Transient Assessment Program (TAP) provided a reporting system in which B&W plant trips occuring between 1980 and present were evaluated by B&W and the licensees. The evaluations resulted in recommendations being

made to the licensees in order to prevent a trip reccurrence. The TAP reports were issued to licensees for information only until 1984. At that time, a more formal program came into being, the Trip Reduction Program. The program was maintained by a BWOG subcommittee which evaluated trips in order of priority by safety significance and made recommendations to prevent recurrence. After the Davis-Besse incident, another formal program, the Safety and Performance Improvement Program (SPIP), was formed by the BWOG.

This program is currently active in evaluating TAP reports and also conducts interviews with operators at B&W plants. Results of these evaluations and operator interviews will be reported in the Recommendation Tracking System (RTS). BWOG will track receipt, evaluation and implementation of resultant RTS recommended facility changes. At the time of the inspection, B&W felt that as many as 500 recommendations could evolve as a result of SPIP.

The Nuclear Parts Advisory System addresses information on B&W procured or manufactured parts in the form of Nuclear Parts Advisory Bulletins. The program provides for technical reviews and analyses of replacement parts B&W supplies to the nuclear industry. If the results of the reviews and analyses show that replacement parts may be necessary, a bulletin is issued. The bulletins address specific parts or assemblies and include information such as the particular component the part is in, the B&W plant applicability, background information as to why the bulletin is being issued and specific recommendations for effected utilities to follow.

B&W also has issued Service Bulletins (SB) and Site Instructions (SI). SB's and SI's provide to B&W plant owners engineered technical documents, procedures, inspections, tests and/or changes. SB's and SI's are issued to licensees by letter with feedback requested. BWOG activities have taken over much of the SB and SI activity and therefore, they are rarely issued.

B&W's assessment of their programs in 1986, specifically the SPIP, has resulted in a determination that of the 13 trips at B&W plants in 1986 (as of the date of the inspection), seven could have been prevented and two could have been less severe if SPIP recommendations had been known and implemented at the particular facilities.

ORGANIZATION: EBASCO SERVICES INCORPORATED
NEW YORK, NEW YORK

REPORT NO.: 99900505/86-01	INSPECTION DATES: 8/11-15/86	INSPECTION ON-SITE HOURS: 108

CORRESPONDENCE ADDRESS: Ebasco Services Incorporated
ATTN: Mr. Charles R. Healy
Director, Quality Assurance
Two World Trade Center
New York, New York 10048

ORGANIZATIONAL CONTACT: Mr. Herman J. Kunis, Jr., QA Engineering
TELEPHONE NUMBER: (201) 460-6345

NUCLEAR INDUSTRY ACTIVITY: Major projects include: Shearon Harris Units 1 and 2; South Texas Project Units 1 and 2 (Constructor); and Comanche Peak Units 1 and 2 (cable tray and conduit support effort).

ASSIGNED INSPECTOR: Robert L. Pettis Jr. 12/30/86
R. L. Pettis, Jr., Special Projects Inspection Section (SPIS) Date

OTHER INSPECTOR(S): R. P. McIntyre, SPIS
K. C. Leu, SPIS

APPROVED BY: John W. Craig 12/31/86
John W. Craig, Chief, SPIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Parts 21 and 50 Appendix B, Ebasco Topical Report ETR-1001, and Ebasco prepared Comanche Peak Topical Report ETR-1001.

B. SCOPE: Status of previous inspection findings, procurement process, design control, indoctrination and training of personnel, audits and computer program error reporting, and evaluation with respect to Ebasco's cable tray and conduit support design verification efforts on Comanche Peak Units 1 and 2.

PLANT SITE APPLICABILITY: Comanche Peak Units 1 and 2 (50-445/446).

BACKGROUND

General

On March 12, 1984, the U.S. Nuclear Regulatory Commission's (NRC) Executive Director for Operations established a program to ensure that technical concerns and allegations related to Comanche Peak Steam Electric Station (CPSES) design and construction activities would be addressed in a coordinated and integrated manner by the NRC staff. As a means of executing its assigned responsibilities, the NRC formed a Technical Review Team (TRT). The TRT initiated a series of onsite inspections and evaluations in July 1984, and documented the results of these activities in five supplements to the Comanche Peak Safety Evaluation Report (NUREG-0797, "Safety Evaluation Report Related to the Operation of the CPSES, Units 1 and 2," Supplements 7-11).

In response to the concerns and issues identified by the TRT, Texas Utilities Electric Company (TUEC) formed the Comanche Peak Response Team (CPRT) to address various construction and design issues. The CPRT Program Plan indicates that all cable tray supports in both units will be "as built verified" by the CPSES personnel and "design verified" by both the Ebasco Services and Impell Corporation. TUEC has also developed a program to review conduit supports. Under this program, TUEC has retained Ebasco Services to perform the design verification for all Unit 2 conduit supports. The design verification of a sample of Unit 1 and common conduit supports selected in accordance with Appendix D of the CPRT Program Plan will also be performed by Ebasco.

Cable Tray and Conduit Supports

TUEC will have 100 percent verification reviews of the cable tray support designs performed. All Unit 2 analysis/design verification work will be performed by Ebasco. Unit 1 work will be divided between Ebasco and Impell. In addition, a CPRT third-party overview of this effort is being conducted by the TERA Corporation to provide assurance that the objectives of the plan in the cable tray and conduit support area are being achieved. This program includes: the development of as-built drawings for all cable tray supports; analysis and design review of all cable tray supports to criteria that are responsible to external source design concerns and in compliance with CPSES licensing commitments; testing to verify and/or establish specific component or system behavior characteristics; and hardware modifications as necessary to ensure final acceptability of all supports.

ORGANIZATION: EBASCO SERVICES INCORPORATED
NEW YORK, NEW YORK

REPORT NO.: 99900505/86-01	INSPECTION RESULTS:	PAGE 3 of 11

Dynamic analysis methods are employed in the design verification process for cable tray supports for Unit 2. Both Ebasco and Impell will utilize their own design procedures for performing the work; however, design methods and criteria will be consistent. For both units, loading combinations and stress limits will be checked in accordance with acceptance criteria conforming to CPSES licensing commitments. For any support that fails to meet these criteria, design modifications will be prepared that result in acceptable qualification. A testing program has been defined to support the cable tray support verification effort. Modifications will be prepared for all cable tray supports that fail to meet specified acceptance criteria. All supports will be qualified in their existing state, or required modifications will be implemented.

CPRT Quality Assurance Program

The Comanche Peak Response Team (CPRT) Quality Assurance (QA) program is described in the CPRT Program Plan. The objective of the CPRT QA program is to ensure the quality of CPRT Program Plan activities and results. The CPRT Senior Review Team (SRT) directed that the QA program be developed based on appropriate criteria in 10 CFR 50, Appendix B.

This inspection is the first in a series of inspections to be performed at the Architect-Engineers (A/E) (Ebasco, Stone & Webster, and Impell) engaged in performing design verification activities for piping and pipe supports, cable tray and conduit supports for both Unit 1 and 2 of the Comanche Peak facility.

These inspections will review the implementation of the A/E's QA programs as modified to address the concerns identified in the CPRT Design Adequacy Program.

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEM:

During the review of a Potentially Reportable Condition file, pursuant to the provisions of 10 CFR 50.55(e) for Comanche Peak Unit 2 cable tray and conduit support effort, documentation was not available which could demonstrate that an evaluation had been performed for applicability to Ebasco projects, other than Comanche Peak, in relation to P-Delta STRUDL computer code errors. (86-01-01)

This item will be reviewed during a future inspection.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Open) Nonconformance (84-02-01): Ebasco procedures do not define a system for reporting and evaluating computer program and system errors.

   Ebasco stated in its response to this nonconformance that Procedure A-30, "Computer-Based Safety Related Programs - Development and Control" is undergoing a revision which will more fully define a system for the reporting and evaluation of computer program errors.

   The inspector reviewed the latest issue of A-30, dated March 26, 1986 and A-37 dated May 31, 1985 "Control of Operating Systems Software Problems Affecting Computer-Based Safety-Related Programs," developed by Ebasco. A-30 now includes Section 10.0, "Errors and Defects," and Section 11.0, "Problems & Errors and Defects in Supplier Developed Programs." These sections define a system for reporting and evaluating computer program errors. However, they do not completely address the dissemination of computer program error reports and the evaluation for potential effect of errors on current and previous design calculations. This is particularly evident in Section 11.0 for errors in supplier developed codes.

   This item will remain open pending a further review of Ebasco's commitment to further revise Procedure A-30.

2. (Open) Nonconformance (84-02-02): No formal method is available for the dissemination of computer program and system error reports and the evaluation of these reports for the effect on current and previous design calculations.

   The response for Nonconformance 84-01 also addresses this nonconformance. This item will remain open pending a further review of Ebasco's revision to Procedure A-30.

3. (Open) Nonconformance (84-02-03): Contrary to Section QA-II-6, "Nonconformances," of Ebasco Services Topical Report ETR-1001, nonconforming conditions being detected by Vendor Quality Assurance Representatives are not being documented on nonconformance reports to ensure that both product and process corrective actions are being completed.

   This item was not reviewed during this inspection.

E. OTHER FINDINGS OR COMMENTS:

1. Cable Tray and Conduit Support Design Verification Review

During this inspection, samples of design calculations for cable tray hangers and conduit supports for Comanche Peak Unit 1 and 2 were selected for review. The review included verification of compliance to "Comanche Peak Manual of Procedures," "Ebasco Nuclear QA Program Manual Topical Report" and applicable codes and criteria.

In general, the inspectors reviewed the technical adequacy of the design calculations and stress analyses related to cable tray/conduit support member stresses, weld design, base plate and anchor bolt design. In addition, seismic and static modeling, STRUDL (Structural Design Language) input and output, loading combinations and conduit span frequency calculations were also reviewed. In addition, the inspectors examined the references, assumptions and drawings associated with the design and analysis.

a. Design Verification of Cable Tray Supports

The following documents from Comanche Peak Unit 1 and 2 were selected and reviewed by the NRC inspectors.

CTH-2-7388
CTH-2-7388 J4177A
*CTH-2-9895
CTH-2-10801
CTH-2-12997
CTH-2-7520
CTH-2-7502
CTH-1-1893

*CTH-2-9895 represents design calculations for another thirty cable tray hangers which include the following: CTH-2-9871, 9872, 9874, 9875, 9877, 9878, 9885, 9886, 9888, 9889, 9891, 9894, 9900, 9901, 9945, 9946, 9951, 9968, 9970, 9971, 9973, 9974, 9979, 9980, 9982, 9983, 9985, 9986, 9988, and 9989.

The STRUDL computer program was used in both static and dynamic analyses. The design and analysis of cable tray support members, including base plate and anchor bolts was performed by STRUDL internally. However, weld design was performed by hand calculations.

b. Conduit Supports and Span Verification Calculations

The inspector selected the following support and span documents for review and examination: SUPT-1226, SUPT-1024, SAG CP 10-6/86, and SPAN-1122

A review of conduit span design calculations was performed which included span frequency and conduit stress calculations. In addition, accelerations for conduit support design and clamp capacity analysis were also reviewed. The STRUDL computer program was used in both conduit support design and span analysis calculations.

As a result of the review, the NRC inspectors determined that the general design requirements and procedures were being followed. However, two items were identified as observations with respect to general references in design calculations. These two items are as follows:

Item No. 1

As shown on Sheet No. 41 of design calculation No. CTH-2-9895, a combined member stress coefficient of 0.48 was being compared to a value of 0.75 without reference to the basis for such comparison. Ebasco personnel stated that this 0.75 value is used to account for the effects of bolt holes, in lieu of performing a rigorous design calculation. The design basis of 0.75 governing the bolt hole effect should have been referenced in the calculation so that the reviewer is able to understand the basis of the analysis and verify the adequacy of the results.

Item No. 2

The NRC inspector observed that the peak "g" values for Safe Shutdown Earthquake (SSE) and Operational Basis Earthquake (OBE) were selected from a refined response spectrum without additional documentation to support such practice. The ratio of SSE to OBE was used to determine the governing loading condition for member stress, weld, and anchor bolt design, as referenced on Sheet No. 3 of design calculation No. CTH-1-1893. Additional documentation to support this determination was not available in the calculation.

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

2. P-Delta STRUDL Computer Code Error

The NRC inspector reviewed Ebasco correspondence concerning deficiencies identified in an Ebasco procured computer program provided by the Power Computing Corporation (PCC), formerly known as UCCEL. The errors were identified in the 0385 version of the P-Delta STRUDL program used in the design verification of safety-related cable tray hangers and conduit supports for Comanche Peak Unit 2. Two errors were found by Ebasco in the code checking portion of the program while performing design verification activities. The errors consisted of (1) the incorrect use, by the program, of an allowable stress factor for axial loads which resulted in improper code checking of several sections along the members length and (2) the program analyzing structural tubing as a tee section. Ebasco committed to perform a 100 percent review of the hangers through a revision to Appendix K of their Comanche Peak procedure. Prior to the error correction, Ebasco had used the program to verify the design of approximately 2600 cable tray and 200 conduit supports.

Ebasco determined that the program produced unconservative results when checking the combination of two-directional bending stresses. These results, if left uncorrected, would result in unreliable and unconservative design verification conclusions. In September 1985, Ebasco Services reported the condition to Texas Utilities Generating Company (TUGCO) as potentially reportable under the provisions of 10 CFR 50.55(e). TUGCO later reported the problem as a potential deficiency to the NRC Region IV office on September 18, 1985. After Ebasco's review of over 1000 calculations, TUGCO notified the NRC, on February 28, 1986, that the potential deficiency earlier reported has now been determined not reportable under the provisions of 10 CFR 50.55(e).

Documentation was not available during the inspection to support a similar evaluation for reportability under the provisions of 10 CFR 50.55(e), or 10 CFR 21, as applicable, for other Ebasco projects.

Unresolved Item (86-01-01) was identified during this part of the inspection.

3. Ebasco Use of an Unverified Computer Code

A review of an Ebasco internal QA audit of the Comanche Peak site, performed during the last quarter of 1985, revealed that Ebasco engineers in the Field Damage Study Project used the computer code

"SETS" to perform high energy line break analysis used by Ebasco to confirm earlier results performed by the original Architect-Engineer, Gibbs and Hill. During Ebasco's analysis, it was discovered that the program was unverified and not suitable for nuclear use in its present form. However, Ebasco decided to continue such program use concurrent with the verification process being performed through UCCEL.

During this verification process, Ebasco attempted to verify the program using an alternate fault-tree analysis, which produced assurance that the code would perform satisfactorily after final verification. The studies generated by this group would continue pending further results from UCCEL.

UCCEL, in its attempt to verify the program, contracted a third party consulting organization to help with the code verification and later concluded that the code met UCCEL's QA program and was certified for nuclear use. This certification came approximately three months after the problem had been identified to UCCEL by Ebasco.

The NRC inspector reviewed Ebasco audit report No. E1606/833 performed at UCCEL during November 5-7, 1985, in which UCCEL was cited by Ebasco for allowing SETS to become part of Ebasco's QA library of computer programs without undergoing the proper baseline testing. QA library programs, known as QALIB, are those programs which are considered to meet the UCCEL QA requirements for nuclear safety-related applications. The "BASE 29A" version was to be tested against the most current installed version (which appeared to be "BASE 26") using all inputs available from UCCEL files. Testing documentation reviewed by Ebasco demonstrated the UCCEL input files were run twice for "BASE 29A" only, and not compared to the earlier version.

Since the nature of the work performed by the Ebasco Field Damage Study Project was confirmatory in nature, the use of this unverified program did not affect hardware already installed.

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

4. Ebasco Topical Report Reference to 10 CFR 21

Ebasco Nuclear QA Program Manual ETR-1001, adapted for Texas Utility Generating Company CPSES Project, was reviewed by the NRC inspectors. It was observed that reference to the reporting of defects and noncompliances as required by 10 CFR 21 was not addressed, however Ebasco's implementing procedure N-23 does state this commitment as referenced in Table I-1.2, page 7, Section QA-I-1 "Matrix of

Compliance of Principal Implementing Procedures to 10 CFR 50 Appendix B." Table I-1.3, page 10, "Principal Implementing Procedures" also addresses this reference. Section QA-III-6, paragraph 3.3 "Nonconformances," which deals with activities affecting quality performed at construction sites, addresses this issue with respect to the overall reporting requirements of 10 CFR 50.55(e).

Ebasco stated they would review this item for possible inclusion into their overall Corporate Topical and CPSES Project Specific Report.

5. Procurement of Computer Services

Ebasco contracted with UCCEL, which later became Power Computing (PCC), in January 1986, after being purchased by Babcock & Wilcox, for Data Processing Services. These services include the accessing of safety-related computer programs which are part of the UCCEL/PCC library. One of these UCCEL/PCC computer codes is P-DELTA STRUDL, which Ebasco is using as part of their verification program of cable tray/conduit supports for Comanche Peak.

The inspector reviewed the original purchase contract with UCCEL, No. D84046 dated September 27, 1984 and Ebasco Specification No. 862-84. This specification establishes the QA requirements for suppliers of nuclear safety-related computer services and software, and is the document which imposes 10 CFR Part 50 Appendix B, and 10 CFR Part 21 on the service bureau or code supplier. The specification defined a defect, as applied to software or services and is restricted to those deviations in delivered software or services from technical requirements that could, on the basis of the suppliers evaluation, create a substantial safety hazard.

The contract left the determination of whether a computer program error could create a substantial safety hazard up to the supplier, based upon his evaluation. When Specification No. 862-84 was revised to No. 862-86 in 1986, the definition of a defect was changed to remove this statement concerning supplier evaluation. Ebasco stated that they interpreted this change to mean that all defects are to be sent to Ebasco.

Purchase Contract Supplement No. D-84046, dated May 15, 1986, was revised to include, among other changes, a mailing address for computer program error notices.

A new Purchase Contract No. D-86039, which was written with PCC on May 26, 1986, also referenced Ebasco Specification No. 862-86 for QA requirements. This is the contract which is current for the accessing of computer programs such as P-DELTA STRUDL from the PCC library.

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

6. Computer Program Error Reports

A review was conducted of Ebasco Administrative Procedure No. A-30, "Computer Based Safety-Related Programs - Development and Control," to determine if the revisions made in response to Nonconformances (84-02-01) and (84-02-02) adequately address the control of computer program error reports. This includes the methods established to evaluate the impact of these errors on present and previous designs.

The inspector reviewed the sections of A-30 for both Ebasco developed programs as well as supplier developed computer programs. Section 10.0 "Error and Defects" defines the process by which computer program errors of Ebasco developed programs shall be distributed and evaluated for impact on current or previous calculations. This section adequately defines this process, except for the overall control and review of these documented evaluations.

Section 11.0 "Problems & Errors and Defects in Supplier Developed Programs," defines the system by which a Project Engineer and holder of a user's manual for a safety-related program acknowledges receipt of a program error using the Supplier Developed Program Error and Defect Notice (SEDN). However, the section has no provisions for the evaluation of these errors to determine their impact on current and previous design calculations, as well as documenting and reviewing of these evaluations.

Previous nonconformances (84-02-01 and 84-02-02) which address this area will remain open.

7. Training

The inspector reviewed Section QA-I-3, "Personnel Indoctrination and Training," of the Nuclear QA Program Manual (Adapted for the TUGCO Comanche Peak SES Project). This section describes the program for indoctrination and training of Ebasco personnel engaged in activities affecting quality with respect to Comanche Peak SES.

The NRC inspector reviewed the Comanche Peak SES QA Training Matrix which lists the design activities performed on the Comanche Peak effort and the appropriate procedural training which is required to perform this activity. Also reviewed was the QA training/directories of all personnel in the New York and Lyndhurst, New Jersey offices. These directories are broken down by discipline listing personnel and their job title.

The inspector reviewed training records for eight Ebasco employees from both the Cable Tray and Conduit Support Group, to assure they had received the training required for their particular position and job activity. This group represents both Ebasco offices and the personnel reviewed had received the required training for work on Comanche Peak. A review of these records demonstrated compliance to the Ebasco procedure with respect to Quality Related Training.

Ebasco stated that some special technical training was given "conference room style," but this training was not documented and no attendance sheets were circulated. There is no requirement for technical training, however Ebasco stated that technical training is received via on-the-job training.

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

ORGANIZATION: EXXON NUCLEAR COMPANY
NUCLEAR FUELS DEPARTMENT
RICHLAND, WASHINGTON

REPORT NO.: 99900081/86-01	INSPECTION DATE: 10/27-30/86	INSPECTION ON-SITE HOURS: 54

CORRESPONDENCE ADDRESS: Exxon Nuclear Company
Nuclear Fuels Department
ATTN: Mr. C. J. Volmer, QA Manager
2101 Horn Rapids Road
Richland, Washington 99352

ORGANIZATIONAL CONTACT: Mr. C. J. Volmer, QA Manager
TELEPHONE NUMBER: (509) 375-8257

NUCLEAR INDUSTRY ACTIVITY: Nuclear fuel assembly supplier for Westinghouse, General Electric, and Combustion Engineering designed reactors.

ASSIGNED INSPECTOR: Ramon L. Cilimberg 12/1/86
R. L. Cilimberg, Special Projects Inspection Section (SPIS) Date

OTHER INSPECTOR(S): C. M. Abbate, SPIS
W. Shier, Consultant

APPROVED BY: 12/1/86
John W. Craig, Chief, SPIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review fabrication, testing, and computer code (CC) design activities, follow-up on previous inspection findings and fuel problems reported at St. Lucie Unit 1, Yankee Rowe, and Big Rock Point.

PLANT SITE APPLICABILITY: PWR facilities with fuel supplied by Exxon Nuclear Company (ENC).

ORGANIZATION: EXXON NUCLEAR COMPANY
NUCLEAR FUELS DEPARTMENT
RICHLAND, WASHINGTON

REPORT NO.: 99900081/86-01	INSPECTION RESULTS:	PAGE 2 of 12

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

1. Contrary to Section 3.0 "Design Control," of the Exxon Nuclear Company, Incorporated Quality Assurance Program Topical Report for Nuclear Fuel Design and Fabrication, XN-NF-1A, Revision 7, dated January, 1985, and Section 1.2.12 of Quality Assurance (QA) Procedure XN-NF P00,002, the verification calculations for the addition of the Moody critical flow model in the RELAP5/MOD2 computer code (version U0CT85) did not adequately test the implementation of the new modelling with the theoretical basis. (86-01-01)

2. Contrary to Section 9, "Control of Special Processes" of XN-NF-1A, Revision 7, an operator used an unapproved procedure to operate the x-ray machine while performing fluoroscopy on fuel rods in the manual mode. (86-01-02)

3. Contrary to Section 9 of XN-NF-1A, Revision 7, an operator used an unapproved coordinates chart to read the values on a current ultrasonic testing (UT) trace being used for the zero settings on the Gould ES 1000 recorder. (86-01-03)

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Closed) Nonconformance (85-01-01)

Contrary to Section 2.2 of ENC QA Procedure QAP No. 2, Licensing and Safety Engineering personnel, engaged in performing safety-related computer code calculations, are not formally indoctrinated and trained into the various fuel reload computer programs used by ENC.

ENC has prepared a package of documents which contains the basic requirements for computer code control Software Development Records (SDR), calculation files, and nuclear safety hazard reporting. The documents are distributed to all new employees

and have been distributed to all current employees involved in safety-related analyses in the Licensing and Safety Engineering (L&SE) organization. In addition, ENC initiated a training program for L&SE personnel that addresses various aspects of PWR and BWR analyses including the methodology of the computer codes used in reload analyses. The training program was initiated during November 1985 and has continued on a schedule of approximately one per week. ENC also committed to maintain this program on an ongoing basis such that new employees will enter the current program schedule. This item is considered closed.

2. (Closed) Nonconformance (85-02-02)

Contrary to Section 3.6 of ENC Topical Report XN-NF-608, Revision 5, "Procedure for Control and Administration of Computer Codes for Engineering Design Calculations,"

a. ENC failed to comply with the reporting requirements which states that written notice of code errors shall be submitted to the Manager, Fuel Engineering and Technical Services (FE&TS) from the cognizant Section Manager for appropriate action.

b. In order to prevent further use, ENC users of ECCS computer code TOODEE-2 were not formally notified of the existence of a coding error, found on St. Lucie Unit 1, soon after its discovery by ENC.

c. A review of the SDR for TOODEE-2 failed to produce written notices pertaining to previously reported code errors.

ENC has transmitted letter WVK:029:85 to all L&SE personnel which describes the TOODEE-2 computer code error. The Manager of FE&TS has issued letter GLR:86:221 to all section managers reaffirming the requirement of ENC procedure XN-NF-608, Revision 6, to provide written notice of computer code errors to the Manager of FE&TS. A letter has been sent to every member of the PWR analysis group which formally notifies them of the TOODEE-2 coding error and the discontinued use of augmentation heat transfer factors. Written notices of the TOODEE-2 coding error have been entered in the SDR. This item is considered closed.

3. (Closed) Nonconformance 85-01-03)

Contrary to 10 CFR 50 Appendix B, Criterion III, and ENC Quality Assurance Procedure XN-NF-P00-002, Section 3.3.2, ENC's documentation and independent review of analyses performed were deficient in the following areas:

a. Adequate documentation such as a calculation notebook or other form of design analysis documentation was not maintained concerning the safety-related analysis described in the ENC Topical Report XN-NF-82-20, Supplement 4. This topical report describes a modification to the ENC ECCS methodology and includes calculations performed to verify changes implemented in the TOODEE-2 computer code.

b. The independent review of the TOODEE-2 computer code input data for the St. Lucie Unit 1 ECCS Analysis (E7380-963-D) was not adequately documented.

ENC has documented and completed an independent review of the analysis described in XN-NF-82-20, Supplement 4. This analysis is identified as E-T21D-887, dated February, 1986. ENC has completed an independent review of the input parameters used with the TOODEE-2 code in analysis 7757-963-1 which supersedes analysis E7380-963-D. Additionally, the review of 7757-963-1 included the input of earlier data associated with E7380-963-D and the documentation is complete. This item is considered closed.

4. (Closed) Nonconformance (85-01-04)

Contrary to ENC Quality Assurance Topical Report XN-NF-608, Revision 5, Section 1.2.7, the verification and qualification calculations performed for the UJUL84 version of the TOODEE-2 computer code were not included in the SDR and were also not retrievable.

ENC has updated the TOODEE-2 SDR with documentation of the verification and qualification calculations that were performed for the UJUL84 code version. Additionally, XN-NF-874, "Procedure for Computer Code Development and Documentation" was issued with a checklist that identifies the SDR content and requires review of the SDR and signature of the code developer's manager. This item is considered closed.

5. (Closed) Nonconformance (85-01-05)

Contrary to Criterion V of 10 CFR 50 Appendix B and Section 6.0 of ENC Analytical Procedure, "Spectrochemical Determination of Impurities in Uranium," Revision 7, dated November 28, 1983, ENC laboratory technicians, responsible for oxidizing uranium, failed to raise samples to a 900°C state for a minimum period of one hour (one minute vs. one hour), as required by procedure.

ENC issued P69269, "Spectrochemical Determination of Impurities in Uranium," Revision 8, dated September 3, 1985, which now requires heating samples to 900°C with no holding time. ENC has performed tests which indicate that a one hour holding time is not necessary to ensure complete oxidation of the sample. This item is considered closed.

6. (Closed) Nonconformance (85-01-06)

Contrary to Section 7.0 of ENC Analytical Procedure, "Calibration of the Quantometer," Revision 4, dated November 28, 1983, ENC failed to formally approve detection limits and calibration curves prior to placing equipment in service.

The NRC inspectors determined that P69268, "Calibration of the Quantometer," Revision 5, dated September 3, 1985, was issued to include an approval form and changed the approver from Spectroscopist to Chemist or Laboratory Manager. The required approvals have been documented. This item is considered closed.

7. (Closed) Nonconformance (84-01-03)

The Software Development records for the REFLEX and TOODEE-2 computer codes were incomplete with regard to identification of purpose, preparer, and independent review.

The SDR for TOODEE-2 has been updated to conform to the requirements of Section 1.2.7 "Software Development Record." However, during this inspection, ENC made a commitment to update the SDR's for REFLEX and other safety-related codes. This item will be reviewed during a future inspection.

The SDR for TOODEE-2 and REFLEX have been updated with the requirements of Section 1.2.7 "Software Development Record." This item is considered closed.

8. (Open) Nonconformance (84-02-01)

No requirements concerning necessary action to report significant errors in Structural Dynamic/Heat Transfer computer codes such as NASTRAN or ANSYS have been established by ENC.

ENC has requested error reports for NASTRAN and ANSYS from UCCEL, an ENC computer services supplier. An incomplete set of error reports from the past several years has been received for ANSYS. In addition, ENC has established a Special Code Coordinator to review and evaluate the effects of these error reports on previous safety-related analyses. However, none of the error reports received to date were determined to be applicable for evaluation on previous analyses. This item will be reviewed during a future inspection.

ENC is receiving computer code error reports for NASTRAN from the Macreal-Schwendler Corporation and from UCCEL. Additionally, a log has been maintained which contains a review of the applicability of the errors reported on ENC safety-related analyses. The error reports for the ANSYS code are being documented, but the written evaluation of the applicability of the code errors on the ENC analyses is incomplete. This item will be reviewed during a future inspection.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Entrance and Exit Meetings

Exxon management representatives were informed of the scope of the inspection during the entrance meeting. The inspection findings and observations were summarized during the exit meeting on October 30, 1986.

2. Computer Codes

The NRC inspectors reviewed safety-related analyses and SDR for computer codes. The results of this review are summarized below.

a. Computer Code Development and Documentation

The review of XN-NF-874, "Computer Code Development and Documentation," Revision 0, dated February 1986, determined that the procedure contains the SDR content, reviews, and approvals needed for computer code development. The forms

and checklists that are required for an SDR are also in XN-NF-874. This procedure will be used in describing the development of new computer codes and the implementation of modifications to existing codes. No items of nonconformance or unresolved items were identified.

b. Computerized Code Development System

ENC has developed a new Computerized Code Development System (CCDS) that is currently being used to provide automation to maintain developmental and production code versions, code modifications, and test problem data. The CCDS maintains a log of all activities in the code evolution and controls the development of new codes. ENC is converting existing codes to be compatible with the CCDS. The CCDS should provide for an improved control of code development and allow for retrieval of previous code versions while preserving the history of code modifications. The new system provides for the traceability of code changes in response to the discovery of errors or other problems. No items of nonconformance or unresolved items were identified.

c. RELAP5/MOD2

The NRC inspectors reviewed nine versions of the development of the RELAP5/MOD2 computer code. ENC received RELAP5/MOD2 from the Idaho National Engineering Laboratory (INEL) and implemented the INEL version in July 1985. Subsequent versions were implemented by ENC using the CCDS and have followed XN-NF-874. The code methodology is intended for use in small break loss of coolant accident analyses and has been submitted for NRR review. No items of nonconformance or unresolved items were identified.

d. The Moody Model

During the review of the RELAP5/MOD2 version designated as UOCT85, the NRC inspectors noted that the implementation of the Moody critical flow model was not adequately verified. The verification calculations included only the case of all steam flow and showed that the code calculation agreed with a theoretical solution. The calculation for that part of the Moody model which includes two phase flow conditions

important for small break calculations was not verified. The NRC inspectors determined that all options within the new modelling are required to be verified.

Nonconformance 86-01-01 was identified in this area.

e. UJUN86

The NRC inspectors reviewed the calculation notebooks associated with four qualification calculations performed for RELAP5/MOD2. Each notebook included a description of the analysis and a detailed independent review. However, all calculations were not performed with the latest version of the code identified as UJUN86. A detailed review was made of the calculation results which determined that the analyses were either performed with FORTRAN coding equivalent to the latest RELAP5 version or the accident analyses code modification was not pertinent to the calculation. No items of nonconformance or unresolved items were identified.

f. RELAP5 Upgrades

ENC has established a contract with INEL for ENC participation in the RELAP5 Newsletter Service. This was initiated in order to obtain timely access to INEL RELAP5 upgrades that enhance the code or correct reported errors. ENC has received the newsletters and performed informal reviews. The NRC inspectors observed that ENC does not have a system to document the review of the newsletters with emphasis on review of error reports prior to performing licensing calculations with RELAP5/MOD2. No items of nonconformance or unresolved items were identified.

g. H.B. Robinson - COLAPS

The NRC inspectors reviewed the calculation notebooks associated with the XCOBRA code for the steamline break analysis for the H. B. Robinson nuclear plant. The calculation notebooks were complete and indicated that an independent review had been performed. However, the data analysis used the COLAPS preprocessor code to prepare input parameters for the XCOBRA code. The H. B. Robinson analysis was verified under a version of XN-NF-608 which did not require COLAPS to be reviewed in detail. The procedures described in the current version of XN-NF-608, Revision 6 define COLAPS as a "Peripheral Code" which requires a more detailed

review by ENC. The review of Peripheral Codes will be included in future NRC inspections. No items of nonconformance or unresolved items were identified.

3. Nondestructive Testing (NDT)

The NRC inspectors reviewed ENC procedure P66,789, "Radiographic Procedure for Through Rod X-Ray," Revision 0, dated June 28, 1985 and ENC procedure P66,785, "Radiographic Procedure for Fixed X-Ray Unit," Revision 0, dated June 25, 1985. The operators followed the steps of both procedures until reaching section 4 of P66,789 which requires fluoroscopy of the pellet stack in the manual mode. The NRC inspectors observed the operator performing operating steps in the manual mode which were contained in a document with a blue cover dated December 7, 1984. The document exhibited the title "Draft Operations Manual" and was not referenced in procedure P66,789. The use of this unapproved procedure was brought to the attention of ENC management. Nonconformance 86-01-02 was identified in this area.

The NRC inspectors reviewed two ENC procedures for UT of zircaloy tubing in accordance with ENC specification XN-NF-535055, "Zircaloy Tubing for Fuel Rod Cladding," Revision 0, dated April 30, 1986. The ENC procedures are P69023, "Fuel Cladding Ultrasonic Test Stations," Revision 19, dated April 29, 1983, and P69027, "3D Gage," Revision 11, dated August 26, 1985. The operator followed the steps of both procedures during the UT of 40 tubes from ENC Lots 0000-2000-26027 and 26028. Section 9.6 of P69027 requires that a current trace or graph be made of tubing standards to provide values to be used for the zero settings on the Gould ES 1000 recorder. The NRC inspector observed the operator using a coordinates chart to improve the readability of the values exhibited by the current trace. The coordinates chart was not a part of procedure P69027 and was not referenced by any approved procedure. The use of this unapproved chart was brought to the attention of ENC management. Nonconformance 86-01-03 was identified in this area.

4. Fuel Rod Manufacture

The NRC inspectors reviewed the ENC Manufacturing Order/Follower for fuel rods. The follower contains each step performed during the manufacturing process, the operation number, work center, and a place for operator signature and date. The follower also lists the acceptance criteria for weld overhang, Total Indicator Reading (TIR) and weight, and has an area where lot and identification numbers are recorded for traceability.

The NRC inspectors observed each step in the fuel rod manufacturing process to assure the rods were being manufactured according to written procedures by qualified operators. The following manufacturing steps and procedures were reviewed: P66,417, "Tube Cleaning," Revision 12, dated December 26, 1984; P66,770, "K-Lower End Weld," Revision 0, dated May 14, 1985; Process Qualification XN-NF-PQ-454, "Jet Pump Lower End Closure Weld for .486 inch Nominal Clad OD (K-Welder)," Revision 4, dated May 30, 1986; P66,557, "Line 1 and Line 2 Rod Loading Stations," Revision 6, dated May 9, 1986; Process Qualification XN-NF-PQ-397, "BWR Jet Pump Upper End Rod Weld (.486 inch Clad OD) H and J," Revision 6, dated May 13, 1986; P66,787, "Helium Leak Check, Horizontal Test Station," Revision 7, dated October 7, 1985; P66,788, "Operating Instructions for Model II Rod Scanner," Revision 1, dated July 22, 1986; P66,433, "Final Etching UO_2 Rods," Revision 17, dated May 9, 1986; P66,529, "Autoclaving Rods and Components," Revision 14, dated June 18, 1986; and Quality Control (QC) Standard XN-NF-P68,537 "Rod Inspection," Revision 11, dated May 1, 1986.

The operators are qualified to work at a number of stations and the NRC inspectors observed several different operators at each work station. The operators were knowledgeable of the procedures being used and followed the procedures. No items of nonconformance or unresolved items were identified.

5. Fuel Incidents at Reactor Sites

The NRC inspectors reviewed selected information concerning fuel incidents which have occurred at reactor sites which may have been caused by steps in the manufacturing process at ENC. Information was obtained during meetings with site personnel and by review of ENC documents. The results are listed below.

a. Failed Rods at Big Rock Point

Between December 1983 and May 1984, it was determined that 26 rods in three fuel assemblies had failed and 84 rods in four fuel assemblies exhibited outer diameter (O.D.) surface corrosion. ENC examined the areas which could have caused the failures and corrosion. These areas included the Quality Control records and shop travelers for each defective rod, and reactor events which occurred during the fuel cycle which could have affected the fuel. As a result of these investigations, ENC issued a report, XN-NF-85-40(P), "Findings on Big Rock

Point Reload H-2 Fuel Failures," in May 1985, which discussed the findings. ENC discovered that all of the defective rods contained the same lot of cladding/tubing supplied to ENC by a West German vendor who is now out of business. ENC was unable to determine the exact cause of failure, but concluded that the failures probably resulted from high corrosion susceptibility of some of the cladding in the lot, outside contamination during manufacturing, and changes in reactor chemistry during the fuel cycle.

b. Failed Rods at St. Lucie Unit 1

In November 1985, six failed and two damaged rods were identified in four fuel assemblies. The failures and damage were a result of fretting caused by debris in the reactor or spacer springs, and secondary hydriding, as described in the ENC report, XN-NF-86-29 (P), Revision 1, dated April 1986 "St. Lucie Unit 1 Fuel Repair, Inspection and Evaluation."

As a result of the ENC evaluation, several manufacturing corrective action measures have been implemented at ENC to decrease the possibility of recurrence. These are: (1) increased inspection of assemblies for loose rods which could fret, (2) a new spring design to decrease fretting, (3) more training of site personnel on handling fuel assemblies, (4) a new bottom end cap design, and (5) new top and bottom spacer designs. Another area being examined by ENC to further decrease fretting is a debris resistant tie plate.

c. Failed Pin in Yankee Rowe Assembly

In December 1985, a failure of one rod was identified at the Yankee Rowe Nuclear Power Station. The NRC inspectors were informed by ENC staff that the failure consisted of one rod, and the fuel was in its second of two cycles, therefore no investigation was performed to determine the cause of the single failure.

6. Fuel Assembly Operations

The NRC inspectors observed the fuel assembly process for manufacturing order 7681KEW001, Revision 2, dated October 14, 1986, being produced for the Kewaunee nuclear plant. Some of the steps performed

for receipt and inspection of spacer components, assembly and welding of spacers, skeleton fabrication, insertion of fuel rods in the skeleton, and final inspection of the fuel assembly were observed.

Activities were performed utilizing the following documents: Drawing XN-303,908, "Spacer Inspection," Revision 4, dated July 7, 1986, Drawing XN-305,763, "Spacer Assembly," Revision 1, dated August 7, 1986, P66775, "Spacer Assembly, Welding and Inspection," Revision 2, dated June 18, 1986, P66,778 "Generic Cage Assembly," Revision 0, dated July 24, 1985, P66779, "Guide Tube Insertion - Skeleton Assembly," Revision D, dated June 5, 1985, P66780, "Guide Tube Flaring - Skeleton Assembly," Revision 0, dated June 5, 1985, P69602, "PWR Bundle Rod-to-Rod Inspection," Revision 6, dated August 15, 1985, and P69049, "Measuring PWR Assembly Envelope and Perpendicularity," Revision 1, dated June 7, 1984.

The steps of the written procedures were properly followed and the required documentation was completed as specified.

No items of nonconformance or unresolved items were identified.

F. PERSONS CONTACTED:

R. Andress
S. Atkins
D. Atkisson
*G. Busselman
H. Chow
D. Cooper
*R. Copeland
C. Davis
R. Delpozo
K. Ford
C. Franklin
B. Fryer
R. Hancock
*N. Harbinson
R. Hill
*J. Holm
S. Hoyem
G. Judson
R. Kantonen
D. Kolesar
D. Lewis
J. Manus
A. Mercado
*W. Nechodom
*D. Perry
A. Price
M. Prince
*G. Ritter
K. Sickle
B. Stavig
M. Stricker
J. Tandy
A. Tarantino
C. Thoren
*M. Valentine
*C. Volmer
D. Westover
*H. Williamson
*W. Yale

*Attended exit meeting.

ORGANIZATION: FISHER CONTROLS INTERNATIONAL, INC.
MARSHALLTOWN, IOWA

REPORT NO.: 99900105/86-01	INSPECTION DATE: 10/7-10/86	INSPECTION ON-SITE HOURS: 24

CORRESPONDENCE ADDRESS: Fisher Controls International, Inc.
ATTN: Mr. Gary Houser
Plant Manager
Post Office Box 190
Marshalltown, Iowa 50158

ORGANIZATIONAL CONTACT: Mr. D. Rowley
TELEPHONE NUMBER: (515) 754-3011

NUCLEAR INDUSTRY ACTIVITY: Manufacturers of ASME Class 1, 2 and 3 valves and controllers and distribution of the associated spare parts.

ASSIGNED INSPECTOR: Edward Yachimiak 11/21/86
E. Yachimiak, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): T. Tinkel, Consultant

APPROVED BY: 11/21/86
Ellis W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: Appendix B to 10 CFR 50, 10 CFR Part 21, and ANSI N45.2.

B. SCOPE: As a result of an interim 10 CFR Part 50.55(e) construction deficiency report from Vogtle electric generating plant concerning failed manual valve actuators, an inspection of the manufacturer, Fisher Controls International, Inc., was performed. The inspection mainly covered the Vogtle report and its similarity to a 10 CFR

PLANT SITE APPLICABILITY: Multiple: all nuclear power plants that use Fisher Model 1073-1076 manual valve actuators.

ORGANIZATION: FISHER CONTROLS INTERNATIONAL, INC.
MARSHALLTOWN, IOWA

REPORT NO.: 99900105/86-01	INSPECTION RESULTS:	PAGE 2 of 6

B. SCOPE (continued):

Part 21 report submitted by Fisher Controls International, Inc. (FCI) in February 1981. Additional areas reviewed were FCI's finished parts receiving inspection department and FCI's Part 21 procedures and implementation plan.

A. VIOLATIONS:

1. Contrary to Section 21.21, "Notification of failure to comply or existence of a defect," of 10 CFR Part 21, FCI failed to adopt appropriate procedures to provide for (i) evaluating deviations or (ii) informing the customer (licensee) so that the deviations may be evaluated for parts, components, or products which are intended for nuclear use and are processed under the Standard Quality Assurance Manual (SQAM). (86-01-01)

2. Contrary to Section 21.31, "Procurement documents," of 10 CFR Part 21, FCI failed to assure that procurement documents for two (2) safety-related Limitorque motor operators specified that the provisions of 10 CFR Part 21 were applicable. (86-01-02)

B. NONCONFORMANCES:

1. Contrary to Criterion V, "Instructions, Procedures, and Drawings," of Appendix B to 10 CFR Part 50, FCI failed to establish procedures and instructions for quantitative or qualitative acceptance criteria for finished parts receiving inspection activities. (86-01-03)

2. Contrary to Criterion XV, "Nonconforming Materials, Parts, or Components," of Appendix B to 10 CFR Part 50, FCI failed to notify customers that the output torque available from the size 2 cast iron gear segment for type 1073-1076 manual valve actuators was derated to 8,000 inch-pounds maximum torque. (86-01-04)

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Closed) Nonconformance (81-01, Item A):

Contrary to Paragraphs 4.2.1.b and 4.2.1.d of FCI's ASME accepted Quality Assurance Manual, FCI did not assure that two (2) open items were signed off in the Summary of Resolution section of Form 3113 before Specification and Design Review Form 3119 was signed off by Technical Consultants for Project 78P174.

A review of these documents showed that Form 3113 was still unsigned. Further investigation revealed, however, that Stone and Webster, Specification No. 2472.110-185, Addendum 1, contained the changes that the two open items of Form 3113 represented. Since the information in Addendum 1 shows that the two open items of Form 3110 were, in fact, closed, the FCI Manager for Planning and Control signed off Form 3113 in the presence of the NRC representative.

2. (Closed) Nonconformance (81-01, Item B):

Contrary to Paragraph 2.2.12 of FCI's QAM and Subarticle NC-4620, Subsection NC, Section III of the ASME Code, FCI did not verify that a qualified procedure had been used for elevated temperature post weld heat treatment of castings.

Review of all pertinent documents showed that qualified weld procedure 05-P1W-1A, Revision 1, had been used instead of the mistakenly reported procedure 05-P1W-1, dated 1-17-74. A re-issued Certificate of Conformance for the affected castings was reviewed. This indicated that the correct weld procedure had been used.

3. (Closed) Nonconformance (81-01, Item C):

Contrary to Paragraph 15.3.1.f. of FCI's QAM, a completed Machine Routing (traveler) for a four inch ASME Section III, Class 1 Vee-Ball valve contained a signature which indicated that only 50% of a required hardfacing operation had been performed.

As per FCI's corrective action response to the NRC, dated July 31, 1981, the above Vee-Ball valve was reworked in accordance with the applicable drawings and procedures, and documented per the QAM.

E. OTHER FINDINGS AND COMMENTS:

1. Type 1073 Manual Valve Actuator Failures

On May 23, 1986, Georgia Power Company (GPC) notified the NRC that handwheels for some manually operated butterfly valves were found to rotate freely without moving the valve disc. The actuator gear boxes (FCI Model 1073) were examined for the valves that failed to operate. Teeth were found to be broken off from the gear sectors of these valves, allowing free rotation of the handwheel and rendering the valves inoperable.

There were two sizes of actuators that failed, a size 1 and a size 2. The size 1 actuator gear sector failed, according to GPC, because a partial tooth on the end of the gear was not strong enough to withstand the normal stopping force when the valve reached the full open or closed position. This tooth would hit the travel stop and break causing the next tooth to impinge on the stop, at an oblique angle, and also break. FCI claims that size 1 gear sectors should not fail if normal force is applied to the handwheel. Only under abusive conditions will the gear sectors fail. Review of FCI engineering drawings indicate that a revision to the size 1 worm gear sector drawing was made on February 2, 1983. This revision was made to include a note requiring partial teeth to be ground flush with the root diameter of the gear teeth. While it is true that a properly manufactured gear sector should not fail under normal loads, a gear with a partial tooth can fail under design conditions. FCI now ensures that size 1 gear sectors do not contain partial teeth.

The details of the size 2 actuator gear sector failure are more complex than that of the size 1. On February 25, 1981, FCI issued a Part 21 report notifying the NRC that a defect existed in the Model 1073 and 1074, size 2, manual valve actuators. A number of plants/utilities were notified by FCI: Callaway 1, Wolf Creek 1, Waterford 3, and the Commonwealth Edison Company. This notification was for valve actuators that had already been shipped. It was also only applicable for valve actuators that were required to produce more than 8,000 inch-pounds of torque.

For valves that had not been shipped, FCI internal memo from G. R. Icenogle to L. Montione, dated 1/12/81, recommended the following: (1) valve actuators that are required to deliver more than 8,000 inch-pounds of torque will be modified, (2) customers that receive valve actuators that are not required to output more than 8,000 inch-pounds of torque will be notified of this new maximum torque limit.

The modification to enable the size 2 actuator to deliver more than 8,000 inch-pounds of torque consisted of a material design change of the worm gear sector from a 3/4" cast iron to a 1-3/8" manganese-bronze. This increased the maximum torque rating to 16,000 inch-pounds.

The problem that occurred at Vogtle was due to FCI's failure to inform Vogtle that a new design limitation was being put into place. Valve data sheets for the failed valves were reviewed and it was found that many contained expected maximum torque ratings from the operator of 13,050 inch-pounds, the FCI catalogue value for a size 2, Type 1073 actuator. Nonconformance 86-01-04 addresses FCI's failure to notify their customers as per their planned corrective action.

2. 10 CFR Part 21 Procedure

During the review of FCI's 1981 Part 21 submittal to the NRC, a concern developed surrounding FCI's Part 21 procedure for evaluating and reporting defects in nuclear, non-ASME components. While section 16.6 of the FCI nuclear quality assurance manual (NQAM) describes the evaluation and reporting process for defects in ASME Code items, the standard quality assurance manual (SQAM) makes no mention of 10 CFR Part 21. The SQAM is written to comply with ANSI N45.2 and is used for non-code nuclear components and parts.

Violation (86-01-01) addresses this item.

3. 10 CFR Part 21 Procurement Document Specification

Contrary to Section 21.31 of 10 CFR Part 21, FCI did not assure that procurement documents for two (2) safety-related Limitorque motor operators specified that the provisions of 10 CFR Part 21 were applicable. On FCI purchase order (PO) 5184115, dated 9/27/85, Part 21 was not imposed for two (2) SMB-000-2 operators. Millpower Supply Company (Duke) PO K50386-73, dated 11/7/84, stated that Part 21 was applicable. On FCI PO 5183957, dated 8/2/85, Part 21 was also not imposed upon Limitorque. Bechtel PO PAV2-42, originally dated 10/2/78, specified that Part 21 was applicable.

Violation (86-01-02) addresses this item.

4. Finished Parts Receiving Inspection Procedures

Review of FCI's finished parts receiving inspection area revealed that personnel performing inspections did not have instructions or procedures to perform inspection activities. A sample inspection process was witnessed during which the FCI inspector stated that the part drawing was the only document used for quantitative or qualitative acceptance criteria. In addition, the back of the drawing was used as the document for recording acceptable lots and this was done with only the date. No acceptance criteria or lot sizes were specified.

Nonconformance (86-01-03) addresses this area.

F. PERSONS CONTACTED:

D. Rowley, Manager QA Documentation
L. Metz, Quality Doc. Analyst
R. Head, Sr. QA Specialist
H. G. Rich, Manager Product Engineering
C. Herdlick, Product Engineer
J. Dresser, Engineer Qality and Analysis Group
H. Dougall, QA Analyst
T. Inslaud, Manager Engr. Admin.
C. Colwell, Manager Project Planning & Control
E. J. Munked, Applications Engineer

ORGANIZATION: GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA

REPORT NO.: 99900403/86-02	INSPECTION DATES: 07/14-16/86	INSPECTION ON-SITE HOURS: 20

CORRESPONDENCE ADDRESS: General Electric Company
Nuclear Energy Business Operations (NEBO)
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

NUCLEAR INDUSTRY ACTIVITY: General Electric Company (GE) NEBO, currently is the central design, engineering and nuclear plant services center for all of the domestic GE nuclear activities regarding commercial nuclear power generating plants.

ASSIGNED INSPECTOR: J. S. Petrosino, Reactive Inspection Section (RIS) 11/7/86 Date

OTHER INSPECTOR(S): None.

APPROVED BY: E. W. Merschoff, Chief, RIS, Vendor Program Branch 11/7/86 Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: To review the corrective actions performed by GE on two vendor related deficiencies.

PLANT SITE APPLICABILITY: Big Rock Point (50-155); Browns Ferry 1/2/3 (50-259/260/296); Clinton 1 (50-461); Cooper Station (50-298); Dresden 2/3 (50-237/249); (continued on next page)

PLANT SITE APPLICABILITY (continued): Duane Arnold (50-331); Fermi 2 (50-341); Fitzpatrick (50-333); Grand Gulf 1/2 (50-416/417); Hatch 1/2 (50-321/366); Hope Creek 1 (50-354); LaCrosse (50-409); LaSalle 1/2 (50-373/374); Limerick 1/2 (50-352/353); Monticello (50-263); Nine Mile Point 1/2 (50-220/410); Oyster Creek 1 (50-219); Peach Bottom 2/3 (50-277/278); Perry 1 (50-440); Pilgrim 1 (50-293); Quad Cities 1/2 (50-254/265); River Bend (50-458); Shoreham (50-322); Susquehanna 1/2 (50-387/388); Vermont Yankee 1 (50-271); and Washington Nuclear 1 (50-397).

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

1. Contrary to Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B to 10 CFR Part 50, GE failed to perform effective source evaluation surveys at Conax Buffalo and Conax Florida to assure that components procured for SLC explosive valve replacement parts were properly controlled during fabrication (86-02-01).

   Specific examples of improperly procured components are listed by GE purchase order (PO) number as follows:

a.	PO-205-XN109	Fitzpatrick	3-SLC explosive valve replacement kits
b.	PO-205-XN510	Fitzpatrick	3-SLC explosive valve replacement kits
c.	PO-205-XK789	Perry	2-SLC explosive valve replacement kits
d.	PO-205-XN337	Shoreham	6-SLC explosive valve replacement kits
e.	PO-205-XN338	Grand Gulf	8-SLC explosive valve replacement kits

f.	PO-205-XN381	Limerick	3-Explosive chamber devices
g.	PO-205-XN733	Limerick	2-SLC explosive valve replacement kits
h.	PO-205-83L460	Limerick	10-SLC explosive valve replacement kits
i.	PO-205-XN642	Pilgrim	3-SLC explosive valve replacement kits
j.	PO-205-XN722	Hatch	8-SLC explosive valve replacement kits
k.	PO-205-XN723	Nine Mile Point 2	3-SLC explosive valve replacement kits
l.	PO-205-83L418	Susquehanna	7-SLC explosive valve replacement kits
m.	PO-205-83L438	Susquehanna	12-SLC explosive valve replacement kits

2. Contrary to Criterion XVI, "Corrective Actions," of Appendix B to 10 CFR Part 50, GE failed to take adequate measures to assure that the vendor performed corrective action to establish a nuclear QA program for its SLC explosive valve component manufacturing. Specifically, the vendor was permitted to use a MIL-I-45208 inspection and test program instead of the required ANSI N45.2 quality assurance program (86-02-02).

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not addressed during this inspection.

E. OTHER FINDINGS OR COMMENTS:

1. Standby Liquid Control (SLC) System Explosive Valve Components

GE's disposition of a January 1984 nonconformance regarding the failure of a nuclear basic component manufacturer to implement a nuclear QA program was reviewed and discussed with GE personnel.

a. Background

The Vermont Yankee (VY) nuclear plant recently discovered that both pathways of the SLC system failed to fire and had been inoperable for approximately 18 months. The root cause of the failure was incorrectly wired explosive primer chamber (EPC) assemblies which were manufactured by the Conax Corporation. VY later determined that it had received six incorrectly wired EPC assemblies from Conax. Therefore, the NRC conducted an inspection at the Conax Corporation facilities to review the circumstances of the failure and to determine how incorrectly wired EPC assemblies were able to go undetected through several tiers of product verifications within the vendor's required nuclear quality assurance program.

The Conax Buffalo Corporation (CBC) facility was inspected by the NRC on April 14-18, 1986 and the inspection was continued on April 28-30, 1986 at the Conax, St. Petersburg, Florida (CF) facility (reference NRC report #99900323/86-01).

During the inspection it was revealed that Conax had not established a nuclear QA program to control the manufacture of nuclear basic components at its Florida facility. Instead, Conax Florida imposed a military inspection and test program on the control of its nuclear items. Additionally, it was revealed that Conax Buffalo did not impose its ANSI N45.2 QA program on all of its SLC replacement component manufacturing. Rather, if the component part number was prefixed by four numbers, no ANSI N45.2 program was imposed. Those components that were prefixed by an "N" were manufactured under an ANSI N45.2 program.

It was noted during the NRC inspection at the CBC facility that GE had previously identified the failure of Conax to control the manufacturing of certain safety-related components in accordance with the required ANSI N45.2 QA program, but did not take adequate actions to ensure that Conax corrected the problem. GE identified this nonconformance on GE deviation disposition request (DDR) numbers 30021-30024, dated January 31, 1984. However, no vendor corrective actions to prevent recurrence were delineated on the DDRs. Therefore, this issue was scheduled for a follow-up at GE's offices.

b. GE Documents Related to SLC Issue

1. GE Supplier deviation disposition request (DDR) numbers 30021-30024, written 2/2/84, approved 2/6/86, Conax Buffalo Corporation (CBC), subject: Vendor manufacturing explosive valve replacement kits, MPL #C41-F004, without imposing the required ANSI N45.2 nuclear QA program requirements.

2. GE Nuclear Power Systems Engineering Department Memo, dated 2/1/84, from G. L. Moore, subject: Trip report to CBC on 1/25-27/84 - to resolve questions in connection with the SLC explosive valve.

3. GE Nuclear Power Systems Engineering Department Memo, dated 1/27/84, from J. Boseman to H. Ehsen, subject: Potential 10 CFR Part 21 reportable condition regarding SLC explosive valve replacement kits (MPL #C41-F004), subject: SLC explosive valve replacement kits were not manufactured in accordance with ANSI N45.2 contract requirements.

4. GE Quality Assurance Engineered Equipment & Installation Department QA Systems, Audit Report, dated 10/3-6/83, Conax Corporation - Buffalo, New York and St. Petersburg, Florida, subject: An ANSI N45.2 and Appendix B to 10 CFR Part 50 audit which determined that "Conax was in compliance with quality practices as stated in their QA manual."

5. GE Quality Assurance Engineering & Services Memo, dated 7/29/86, from C. L. Lewis to Conax Supplier File, subject: Clarification of Conax audit report of 10/3-6/83 - "The October 1983 intent of subject audit report was the approval of only the Conax Buffalo facility."

c. Discussion

A review was performed of the documents listed above and it was found that GE was aware that Conax was not supplying the SLC system replacement components in conformance with its procurement requirements. The 10 CFR Part 21 reportability memo (b.3) listed above indicates that 70 SLC system replacement components were not manufactured in accordance with an ANSI N45.2 QA program or equal. Instead, the replacement components were manufactured to a MIL-I-45208 military inspection and test program which addresses only a portion of a typical nuclear QA program. A review of several Conax certificates of conformance during the Conax inspection indicated that a MIL-I-45208 program was used in lieu of the required ANSI N45.2 QA program.

General Electric identified on DDR #30021-30024, item b.1 above, that Conax failed to control the manufacturing operation of its SLC explosive valve replacement parts (except for part numbers that were prefixed by "N") in accordance with the required ANSI N45.2 QA program. The DDRs accepted the deviating hardware on a "use-as-is" disposition based on a statement that SLC replacement hardware was acceptable as recommended by Conax because the QA program "employed" at Conax Florida contains the same program elements required by ANSI N45.2, and that the attached GE trip report shows acceptability of the QA program that was implemented. However, the NRC inspector determined that both of the documents that GE based their hardware acceptance on, (b.2 and b.4 above) were in error. Specifically:

1. GE memo, b.2 above, improperly concluded that the program implemented at the Conax Florida facility ..."has all of the ANSI N45.2 elements in place," based on GE-Conax discussions at Buffalo without any GE inspection or visit to the Florida facility.

2. GE audit report, b.4 above, concluded that Conax Buffalo and Conax Florida was in compliance with quality practices as stated in their nuclear QA manual. However, NRC-GE discussions determined that only Conax Buffalo was in compliance and that Conax Florida was inadvertently added to the audit report. A GE clarification letter (b.5 above) was subsequently submitted to the GE files and the NRC inspector.

Additionally, GE memo, b.3 above, indicates that even though the explosive valve components were not manufactured in accordance with an ANSI N45.2 program the hardware is acceptable. The memo states that "it is apparent that adequate quality assurance

provisions are provided by the MIL-I-45208 program when manufacturing the functional explosive element of the valve." However, the MIL-I-45208 program does not address several important ANSI N45.2 attributes such as: (1) establishment and execution of a QA program; (2) QA/QC inspectors independence from cost and schedule; (3) design change control; (4) qualitative and quantitative acceptance/rejection criteria; (5) document review, approval and control; (6) control of special processes; (7) inspections performed by individuals other than those who performed the work; (8) inspection hold/witness points; (9) indoctrination and training; and (10) internal audits to verify adequacy/execution of program. However, the GE conclusion in the above memo (b.3) was incorrect as evidenced by 12 nonconforming explosive primer chambers that went undetected through the Conax quality assurance test inspection and verification system at Florida.

Ultimately, GE's disposition of this matter resulted in Conax Florida's continuing to manufacture nuclear components without a QA program.

d. Conclusion

The failure of Conax to comply with the GE procurement document ANSI N45.2 QA program requirements was adequately identified by GE. However, GE failed to procure nuclear basic components in accordance with their purchase order document QA requirements, and did not take adequate steps to ensure that Conax performed corrective action to prevent recurrence at the Conax facilities.

1. Elma Power Supplies:

A review was performed to determine if GE had addressed workmenship, overheating and other similar problems that have been identified with Elma's ferroresonant power supplies. GE has adequately addressed and has taken corrective actions to prevent field failures of the power supplies.

F. PERSONS CONTACTED:

G. P. Ferguson, QCE-QAE&S, GE/NEBO
K. Utsumi, EDE, Sr. Engineer, GE/NEBO
R. A. Luoma, EDE, Sr. Engineer, GE/NEBO
R. T. Kern, EDE, Engineer, GE/NEBO
J. J. Boseman, Principal Engineer, GE/NEBO

F. PERSONS CONTACTED (continued)

C. L. Lewis, QCE, GE/NEBO
T. Gachowski, Manager, GE/NEPO
E. Y. Gibo, System Engineer, GE/NEBO
H. P. Williams, Manager, GE/NEBO
J. J. Fox, Senior Project Manager, GE/NEBO
G. B. Stramback, Safety Evaluation Manager, GE/NEBO

ORGANIZATION: LIMITORQUE CORPORATION
LYNCHBURG, VIRGINIA

REPORT NO.: 99900100/86-01	INSPECTION DATES: 09/22-26/86	INSPECTION ON-SITE HOURS: 12

CORRESPONDENCE ADDRESS: Limitorque Corporation
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ORGANIZATIONAL CONTACT: Mr. P. McQuillan, QA Administrator
TELEPHONE NUMBER: (804) 528-4400

NUCLEAR INDUSTRY ACTIVITY: Manufacturer of motor and manual operated valve actuators.

ASSIGNED INSPECTOR: J. T. Conway — 11-4-86
J. T. Conway, Reactive Inspection Section (RIS) — Date

OTHER INSPECTOR(S): J. J. Jacobson, RIS
S. Alexander, EQIS
O. Gormley, Equipment Qualification Inspection Section (EQIS)
T. Tinkle, Consultant

APPROVED BY: E. W. Merschoff — 11/7/86
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 as a result of the receipt of 10 CFR Part 21 reports from: (1) Portland General Electric Company (PGEC) relating to defective motor insulation in H. K. Porter/Peerless motors, (2) Georgia Power Company (GPC) pertaining to damaged internal wiring in
(continued on next page)

PLANT SITE APPLICABILITY: Defective motor insulation - Trojan (50-344); Damaged internal wiring - Vogtle 1 & 2 (50-424/425); Cracking of limit switch rotors - Rancho Seco (50-312); and Unqualified terminal strips - Arkansas Nuclear 1 & 2 (50-313/368).

ORGANIZATION: LIMITORQUE CORPORATION
LYNCHBURG, VIRGINIA

REPORT NO.: 99900100/86-01	INSPECTION RESULTS:	PAGE 2 of 21

B. SCOPE (continued):

Type SMB-000 operators, (3) Sacramento Municipal Utility District (SMUD) on cracking of limit switch rotors, and (4) Arkansas Power & Light (AP&L) on unqualified terminal strips.

A. VIOLATIONS:

1. Contrary to Section 21.31 of 10 CFR Part 21, Limitorque did not pass on the requirements of 10 CFR Part 21 to WESCO (Puchase Order (PO) No. 1825 dated April 18, 1986) for Rockbestos wire and to Barker Jennings Corporation (PO No. 3579 dated September 19, 1986) for calibration services. (86-01-01)

2. Contrary to Section 21.51(b) of 10 CFR Part 21, Limitorque was unable to provide documented evidence that an adequate review and evaluation was performed for possible 10 CFR Part 21 reporting on Type SMB-000 motor operators containing wiring that can be potentially damaged by the energization of the installed space heaters as determined by the following (86-01-02):

   a. A GPC 10 CFR 50.55e report to the NRC dated March 20, 1986 described damaged (burnt) wiring on a number of safety-related Limitorque motor operators.

   b. An NRC inspector at Vogtle Nuclear Power Plant on May 20, 1986 observed burnt wiring on a number of Limitorque operators.

B. NONCONFORMANCES:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Subsection A.1 of Section I of the Quality Assurance Manual (QAM), and Section 5 of ANSI N45.2-1977, it was noted that Limitorque failed to impose the requirements of a QA Program (ref. PO No. 3M4515 dated January 28, 1985) upon H. K. Porter for nine Class 1E DC motors which were used on safety-related operators sent to Portland General Electric Company (PGEC) for use at the Trojan facility. (86-01-03)

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 6 of ANSI N45.2, written instructions/procedures did not exist for current practices followed by the (a) Parts Department to ensure correctness and completeness of spare part orders, and (b) Parts Department, Application Engineering, and Quality Control to initiate, review, and issue Certificates of Compliance (CofC) for spare part orders. (86-01-04)

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 2 of ANSI N45.2, Limitorque did not process AP&L Company PO No. 51965 for terminal strips under controlled conditions to assure that prerequisites affecting quality of the order were identified and accomplished. It was noted that Limitorque incorrectly advised AP&L personnel that Buchanan 724 terminal strips were a correct replacement for qualified Buchanan 524 terminal strips, and issued a CofC that certified that Buchanan 724 terminal strips were environmentally qualified in accordance with tests documented in Limitorque Report B0119, Revision 1, dated November 10, 1982. As a result, AP&L was supplied with Buchanan 724 terminal strips that were, in fact, not qualified. (86-01-05)

C. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Closed) Violation 85-01-01: Limitorque did not notify the NRC that unqualified Baldor motors were used in safety-related valve actuators installed in the WNP-2 and Salem Nuclear Power Plants.

   Limitorque corrective action commitments made in a letter dated July 15, 1986, were reviewed. Limitorque has revised and reissued Procedure QCP-22 "Reporting of Defects for Safety-Related Equipment." The revised procedure clarifies and strengthens reporting requirements to comply with 10 CFR Part 21. The status of the technical issue cited under this violation was discussed with the QA Administrator. He stated that all instances where unqualified Baldor motors were used on safety-related POs were identified prior to the November 1985 NRC inspection. Limitorque is not aware of any remaining required actions.

2. (Closed) Violation 85-01-02: Limitorque was unable to provide documented evidence that required reviews and evaluations were performed for four identified problems.

   The inspector reviewed Limitorque's corrective action commitments. The revised Procedure QCP-22 clarifies and establishes requirements for maintaining documentation to assure compliance with 10 CFR Part 21. The status of the four technical issues (motor pinion key failures, worm shaft gear failures, H3BC actuator potential field maintenance problem, cracked/broken limit switch rotors) cited under this violation were reviewed with the QA Administrator. He stated that the status of these items has not changed since the November 1985 NRC inspection, and Limitorque is not aware of any remaining required actions.

3. (Closed) Violation 85-01-03: Limitorque did not post Section 206 and a "Notice" in the manufacturing area, and Procedure QCP-22 did not provide for notification of a deviation/defect in a basic component.

   The inspector reviewed Limitorque's corrective action commitments. The revised Procedure No. QCR-22 clarifies the procedure for evaluation of potential defects, notification of a Limitorque officer, and reporting of defects in safety-related equipment to the NRC. Section 206 and a "Notice" are now posted in the manufacturing area as well as outside the lunch room in the office area.

4. (Closed) Nonconformance 85-01-04: Thirteen POs to vendors (Ryerson Steel-two, Copperweld Steel-one, Walker Machine-two, Southern Centrifugal-two, Lynchburg Foundry-two, Foster Electric-three, and WESCO-one) indicated that quality requirements (e.g., QA Program) were not passed on to these suppliers/manufacturers of components destined for actuators purchased by nuclear customers as safety-related items.

   Limitorque said (ref. July 15, 1986 letter) that their POs are consistent with the pertinent requirements of Section 5 of ANSI N45.2 in that they require vendors who are on the Approved Vendors List (AVL) to supply material/components with certified material test reports (CMTR) and/or CofC. In addition, the material/components undergo receipt inspection.

   The inspector reviewed CMTRs/CofCs and Receiving Inspection Reports for items supplied by the seven vendors noted above. The material/components supplied by these vendors were not complex items, and each of the vendors was on the AVL.

5. (Open) Nonconformance 85-01-05: Limitorque did not update the QAM Revision 2, to address training of non QC personnel (e.g., engineers, parts and shipping personnel, purchasing agents, order processors, and field service personnel).

   The inspector reviewed Limitorque's corrective action commitments. Section I of the revised QAM dated July 11, 1986 incorporated "Indoctrination and Training of Non-QC Personnel Performing Activities That May Affect Quality." Training was applicable to personnel in the Engineering Department, Order Processing, Sales Group, Parts Department, Service Department, Purchasing Department, and Shipping Department.

In the July 15, 1986 response, Limitorque stated "Training of above personnel shall commence no later than September 17, 1986." A note dated September 17, 1986 from the QA Administrator to the QC Manager indicated that training for the non QC personnel had been rescheduled to commence on October 15, 1986. Since the training has not been completed, this item is considered open and will be reviewed during a future NRC inspection.

6. (Closed) Nonconformance 85-01-06: Limitorque's written practice, Procedure No. QCP-11, for nondestructive examination (NDE) did not describe the responsibility of each level of certification or rules covering the duration of interrupted service requiring re-examination and recertification, and the qualification records of three NDE personnel did not contain a statement indicating satisfactory completion of training in accordance with Procedure No. QCP-11.

   The inspector reviewed Procedure No. QCP-11, "Nondestructive Testing Procedure," revision dated July 11, 1986. Section II-A.1 included rules covering the duration of interrupted service requiring re-examination and re-certification and described the responsibility of each level of certification. A review of the NDE records of four examiners indicated that each qualification record contained a statement indicating satisfactory completion of training in accordance with Procedure No. QCP-11.

7. (Closed) Nonconformance 85-01-07: The area of "QA Records" (i.e., Section XI of the QAM) was not on the internal audit checklist form No. QC0019 and was not audited in 1983 and 1984.

   The inspector noted that "Section O" has now been identified on the cover page of the checklist form No. QC0019. Section O includes QAM Sections IX "Material Movement" and XI "Records-General and Quality Assurance", and QCP 21 "Nuclear Service Procedure". A review of the 1983 and 1984 Internal Audit of Manufacturing verified that Section O "Safety Related Units" had been audited by the QA Administrator.

8. (Closed) Nonconformance 85-01-08: Limitorque failed to generate semi-annual Vendor Evaluation Reports (VER) for 11 vendors, 11 other vendors having low ratings remained on the AVL, and audits were not performed on Page-Wilson Corporation and Hartzog Granite Company who calibrated the hardness testers and surface plates, respectively.

The inspector reviewed the AVL dated August 13, 1985 and approximately 25 VERs which documented the vendors (including the 11 sited above) performance from June to December 1985. With regards to the other 11 vendors, one vendor was dropped from the AVL, and letters, which the inspector reviewed, were sent to the remaining 10 vendors notifying them of their deficiencies for subsequent corrective action to eliminate the minor nonconformances on future orders. Procedure No. QCP-25 revision dated September 23, 1986 documented the method for qualification of vendors (e.g., Page-Wilson and Hartzog Granite) who perform services at the Limitorque facility. This procedure was reviewed, and it was also verified that a QC personnel had witnessed the calibration of equipment performed onsite by Honeywell on June 18, 1986.

9. (Open) Nonconformance 85-01-09: Limitorque did not effectively implement QA program requirements for reporting defects in safety-related equipment.

   The inspector reviewed Limitorque's corrective action commitments. Limitorque's revised and reissued Procedure QCP-22 became effective July 11, 1986. A memo dated August 6, 1986 in the QC Manager training book stated that the Executive Vice-President, the Chief Engineer, and the Technical Manager had been indoctrinated in the usage of QCP-22. The current revision of Procedure QCP-22 establishes a Part 21 Review Committee and assigns the QA Administrator as the chairperson. The QA Administrator stated that no items have been submitted to the Part 21 Review Committee since this revision became effective, and the Committee has not met as of the date of this inspection. Since documentation does not exist upon which to evaluate the implementation effectiveness of QCP-22, this item is considered open and will be reviewed during a future NRC inspection.

10. (Closed) Nonconformance 85-01-10: Limitorque could not provide any documented evidence that measures were taken to identify the cause of a significant condition adverse to quality, the extent of the problem, and necessary corrective action to preclude repetition; or that the condition was reported to appropriate levels of management.

    The inspector reviewed Limitorque's corrective action commitments. The revised and reissued Procedure QCP-22 clarifies and establishes requirements for maintaining documentation to assure compliance with 10 CFR Part 21. The status of the technical issue cited under this nonconformance (use of unqualified Baldor motors) was

discussed with the QA Administrator. He stated that all instances where unqualified Baldor motors were used on safety-related POs were identified prior to the November 1985 NRC inspection. Limitorque is not aware of any remaining required actions.

D. OTHER FINDINGS AND COMMENTS:

1. Unqualified Buchanan Terminal Strips

Based on information in a 10 CFR Part 21 report dated July 16, 1986, AP&L requested a quote for 15 Buchanan 524 terminal strips for Limitorque actuators while preparing for a refueling outage. The Buchanan 524 terminal strips are environmentally qualified and made of a brown phenolic material. Limitorque stated that Buchanan 524 terminal strips were not available, but that Buchanan 724 terminal strips (Part No. EA3-0724-12) were available and were a correct replacement.

On June 18, 1986 Limitorque received PO No. 51965 from AP&L. Item 9 on the purchase order was for 15 Buchanan 724 terminal strips. The PO invoked 10 CFR Part 21 and requested QA documentation certifying the Buchanan 724 terminal strips to 10 CFR Part 50, Appendix B and ANSI N45.2 and that the Buchanan 724 terminal strips were the correct replacement and environmentally qualified per Limitorque Test Report B0119 Revision 1. (Note: Buchanan 524 terminal strips are environmentally qualified per Limitorque Test Report B0119 Revision 1.)

A PO acknowledgement (Order No. 3T3534) was issued by Limitorque on June 19, 1986, in response to AP&L PO No. 51965, and a nuclear safety-related application product identification code was assigned to this order. On June 19, 1986 the Parts Department initiated a request (Form LP-45D) to the QC Manager to issue a CofC for the Buchanan 724 terminal strips to certify:

a. Parts supplied are compatible and interchangeable with the original equipment.

b. Parts described met the PO requirements.

c. Parts supplied were provided in accordance with Limitorque's QA Program dated January 4, 1984, which meets the applicable requirements of 10 CFR Part 50, Appendix B, and ANSI N45.2.

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d. Parts supplied are qualified per IEEE 323 (1974), IEEE 382 (1972), and IEEE 344 (1975) for use inside/outside containment and are equal to or superior to those parts tested per Test Report B0119, Revision 1, as contained in B0058 (culmination of all Limitorque Test Reports).

e. Proper installation of these parts will not adversely affect existing qualification of the actuator involved.

The Parts Department forwarded the request to issue the CofC to Application Engineering for review and approval. The Manager of Application Engineering approved the request on June 19, 1986. On July 3, 1986, the QC Manager issued a CofC that provided the requested quality documentation for the Buchanan 724 terminal strips.

Based on information in the AP&L 10 CFR Part 21 report of July 16, 1986, AP&L personnel, who were accustomed to seeing brown phenolic terminal strips, questioned the qualification status of the white 724 terminal strips following receipt of the parts. The Limitorque Parts Manager issued a Returned Goods Permit No. 14985 on July 8, 1986, and the strips were returned to Limitorque and receipt inspected (Receiving Report R58144) on July 10, 1986.

A memo dated July 16, 1986, from the QA Administrator summarizes information concerning the cause and corrective measures associated with the Buchanan 724 terminal strips as follows: The AP&L PO for the terminal strips was processed as an urgent delivery item. While processing the PO, Limitorque personnel confirmed that Buchanan 724 and 524 terminal strips were identical in form, fit, and function. However, they did not verify whether the Buchanan 724 terminal strips were environmentally qualified per Limitorque Test Report B0119, Revision 1. As a corrective action measure, cognizant Limitorque personnel were notified by a memo dated July 14, 1986, that Buchanan 724 terminal strips were not qualified. The unqualified Buchanan 724 terminal strips were returned to Limitorque. The QA Administrator believes the issuance of unqualified terminal strips to AP&L is an isolated case. A note dated September 17, 1986, indicated a computer printout had been generated to permit a search for other terminal POs. If an occurrence of a similar item is found, Limitorque will issue a separate 10 CFR Part 21 report.

Limitorque Test Report B0119, Revision 1, dated November 10, 1982, was reviewed. Section 2.0 identifies the terminal strips (e.g., Marathon 300, Marathon 1600, Buchanan 0272, Buchanan 0524, General Electric EB-5, and Curtis Type "L") that were included in the

environmental testing covered by the report. Limitorque PO No. 40958 dated September 9, 1986, for Buchanan 724 terminal strips indicated that the terminal strips were purchased from an electrical supply house as a commercial catalog item. This was confirmed in a discussion with the QC Manager. The QA Administrator stated that Limitorque handled the 10 CFR Part 21 evaluation for the AP&L Buchanan 724 terminal strips in accordance with the version of Procedure QCP-22 "Reporting of Defects for Safety-Related Equipment" in effect prior to July 11, 1986. (See Nonconformance 86-01-05.)

2. Limitorque Spare Parts Order Processing

As a result of unqualified Buchanan 724 terminal strips being supplied to AP&L, an inspection of Limitorque's procedures and practices associated with spare parts processing was performed. The inspection included a review of QAM and QC procedures related to spare parts orders. Actual practices associated with spare parts order processing were discussed with the Manager of Order Entry, the Parts Department Manager, and the QC Manager. Section IV of the QAM and Procedure QCP-16 "Order Entry and Processing Procedure" contain the Limitorque practices for order entry, processing, and specification review. According to the Manager of Order Entry, QCP-16 is applicable to all orders which are divided into either actuator (unit) orders or spare parts orders. All orders go to the Manager of Order Entry who sorts the orders and routes all spare parts orders to the Spare Parts Manager. QCP-16 states the Manager of Order Entry or his designated appointee is responsible for ensuring the correctness and completeness of customer orders. In practice, the Spare Parts Manager is the designed appointee of the Manager of Order Entry for spare parts orders.

The following Spare Parts Department practices, which are not currently covered by written procedures, were identified during conversations with the Parts Department Manager:

a. Any individual in the Parts Department designated by the Parts Department Manager can initiate and review an order acknowledgement for a customer spare parts order.

b. The Spare Parts Department initiates requests to the QC Department to issue quality documentation requested by customer spare parts orders. These requests are made on a standard form (LP-45D). Only the Parts Department Manager or his designated alternate (currently one individual) can initiate these requests. The Parts Department can make a

direct request for the QC Manager to issue CofCs if the certification is for parts that are (1) equal to or better than the original, (2) compatible and interchangeable with the original, (3) in accordance with the customer's own form, (4) in accordance with the customer's PO, or (5) in accordance with the Limitorque QA Program which includes certifying to 10 CFR 50, Appendix B and ANSI N45.2. If the customer PO requests a CofC indicating qualification to a specific test report or certification to some other quality specification, the Parts Department sends the LP-45D form to Applications Engineering for review and approval prior to submittal to the QC Manager.

c. The Parts Department uses parts numbers to determine the correctness and completeness for a customer order. A spare with a part number identical to that supplied on the original order is considered to be the correct spare part for current orders.

Section II of the QAM discusses product design, standards, and engineering document control. Section II.B discusses the function of the Applications Engineering Group as a consulting body for the "Sales Engineering Force" and checkers for the QA Department. A note in this section states the following:

> Order Entry and Processing are a complete and separate entity and have the primary responsibility for the selection of the correct Limitorque "Standard" for the equipment sold (see Order Entry and Processing Procedure). Order Entry and Processing personnel are <u>not</u> permitted to render any judgement or apply any Limitorque "Standard" unless it is strictly within the Engineering Qualification Data without the written approval of the Application Engineering Department.

Procedure QCP-21 "Safety-Related Nuclear Service Procedure" identifies requirements for processing orders for safety-related nuclear service. Some of the requirements in QCP-21 are not specific as to whether they apply to spare parts orders as well as actuator orders. This point was discussed with the QC Manager who indicated he was aware of this concern. The QC Manager further provided a draft revision to QCP-21 that will include some specific requirements applicable to spare parts orders for safety-related nuclear service.

During conversations with various Limitorque personnel, the inspector observed that some confusion exists within Limitorque concerning which requirements in the QAM, QCP-16, and QCP-21 apply to spare parts orders. The confusion arises from requirements that simply refer to "order" and are not specific as to actuator orders or spare parts orders. Some departments consider that certain of these requirements apply both to actuator and spare parts orders while other departments consider that these requirements apply only to actuator orders. The interpretation for a particular requirement is not necessarily consistent among the departments involved. This matter was discussed with the Manager of Order Entry, the Spare Parts Manager, and the QC Manager. (See Nonconformance 86-01-04.)

3. Valve Actuator Weight Discrepancies

Babcock & Wilcox (B&W) was notified in December 1984 of discrepancies in valve actuator weights listed in the valve vendor drawings for seismic analysis. This problem, which involved valves manufactured by Anchor/Darling, Rockwell International, Copes-Vulcan, and ACF industries was identified by TVA at Bellefonte Units Nos. 1 and 2. This item was discussed during the November 1985 NRC inspection of Limitorque and left as an open item.

On July 18, 1986, NRC headquarters was advised that a weight discrepancy was identified with a Limitorque actuator supplied to the Palisades nuclear plant. The actuator in question was a rebuilt SMB-00 that was found to weigh 260 pounds. Consumers Power stated the actuator weight on the valve vendor drawing is 168 pounds. Limitorque's Technical Manager stated that Consumers Power contacted him and advised of the actuator weight discrepancy. He advised that the weight of SMB-00 actuators varies depending on various factors such as motor size and actuator application and that the 168 pound figure seemed low. The Technical Manager stated that Limitorque was unable to provide any additional information on the specific actuator until the serial number or order number for the actuator was identified by Consumers Power.

In an attempt to obtain general information on typical weights of SMB-00 actuators, Limitorque provided two documents for review:

a. A report dated February 25, 1985, with results of calculations performed for B&W (PO 048072LU, Limitorque Order 3M5776). A summary of data in the report for two sizes of SMB-00 actuators having four limit switches, motors (1000 RPM) with no brakes,

and a handwheel size of 12 in. is presented below. Anchor Darling manufactured the valve on Order No. 3A6119 and the rest were manufactured by Rockwell.

Order No.	Motor (Ft-lb)	Enclosure	Total Weight (lbs)
384231	10	Nucl.	290
384673	25	Water-proof	210
3A3659	25	Water-proof	285
3A6119	10	Nucl.	295
3A3658	10	Water-proof	275

b. Limitorque Date Sheet (Form SEL-16 dated October 17, 1977) titled "Approximate Weights*." A summary of data for three sizes of SMB-00 actuators is presented below.

Motor (ft-lb)	Weight*
10	185
15	190
25	200

*CAUTION - Unit weights in this chart are approximate for standard units with standard motor (1800 RPM). Addition of optional equipment can increase standard weight considerably (i.e., add 40 pounds for side mounted handwheel, and 80 pounds for integral cover (min-max 30 pounds)).

In an attempt to gain a more complete understanding of the actuator weight discrepancy item, the inspector considers that the valve vendor drawings for the particular Palisades SMB-00 actuator should be reviewed as well as the technical basis for the information on the drawing.

4. Cracked Limit Switch Rotors

The inspector discussed with various personnel the 10 CFR Part 21 report dated June 10, 1986 from SMUD concerning cracks in limit switch rotors. This problem was previously identified in 1984 (ref. Limitorque letter dated February 15, 1984 to Westinghouse). The cracks have been found only on manufactured white melamine rotors which were discontinued for use by Limitorque after 1979. Most of the cracks reported have been hairline in nature and pose no real safety significance. Rotors currently used by Limitorque are made of a brown fibrite material which has been found to be more suitable for this application. Limitorque has suggested replacing any cracked white melamine rotors with the newer design brown fibrite rotors. NRC Information Notice 86-71 dated August 19, 1986 addresses this problem.

5. H. K. Porter/Peerless Motors

In two 10 CFR Part 21 reports dated May 6 and June 6, 1986, the NRC was informed by PGEC of a defect discovered in the motor lead insulation in replacement motor actuators for auxiliary feedwater flow control valves installed at the Trojan facility. The defect involved cracking of the motor lead insulation causing an insulation breakdown between the motor lead conductors and the grounded operator case. A short circuit developed which blew the power supply fuse to the operator rendering the associated valve inoperable.

Limitorque was advised of this problem and has determined that potentially defective motor lead insulation was installed on DC motors manufactured by H. K. Porter/Peerless between the dates of November 30, 1984 and December 6, 1985. During this time period, Peerless changed the motor lead insulation from a Nomex insulation with a glass braid epoxy impregnated overcoat to a Nomex insulation with a Kapton jacket. This change was approved by Limitorque based upon Peerless recommendations. Peerless has since gone to a Nomex insulation with a non-epoxy impregnated glass braid overcoat. Limitorque has determined this insulation system to be acceptable. Limitorque is currently determining those plants which Limitorque may have received operators containing the potentially defective motors. Limitorque has agreed to inform the NRC in writing of proposed corrective action and a list of all affected plants within 30 days of this inspection.

The motors in question were contained in Model SMB-000 operators purchased from Limitorque on PGEC PO No. N-29175 dated February 27, 1985. PGEC's PO required the operators to be manufactured in accordance with a QA program meeting the requirements of 10 CFR Part 50, Appendix B, and it also imposed the requirements of 10 CFR Part 21 upon Limitorque. Limitorque purchased the motors from H. K. Porter/Peerless under Limitorque PO No. 3M4515 which invoked Limitorque specification EQC-1 Rev. F and Limitorque Drawing No. 21-497-0014-1, Rev. B dated January 23, 1981. These documents contained specific performance characteristics for the motors; however, QA requirements were not imposed on H. K. Porter/Peerless (See Nonconformance 86-01-03).

Item No. 10 of Drawing No. 21-497-0014-1 passed on the requirements of 10 CFR Part 21 to the motor manufacturer. Limitorque revised Drawing No. 21-497-0014-1 to include QA requirements and motor lead insulation requirements in 1985 and 1986, respectively. Revision C dated July 10, 1985 to the Drawing added Item No. 11 which stated "Motors to be supplied in accordance with the H. K. Porter Company's Quality Control Procedure Plant 1 dated April 14, 1982." Item No. 12, which stated, "Lead insulation shall be epoxy impregnated glass braid overcoat over nomex primary insulation" was added in Revision D dated June 15, 1986.

6. Space Heaters

In a March 20, 1986 10 CFR 50.55e report, GPC notified the NRC of a problem concerning burn damage to internal wiring caused by energized space heaters in type SMB-000 Limitorque operators. The wiring was burnt due to its close proximity to or contact with the limit switch compartment space heaters or space heater brackets.

The wiring was not properly routed and was not restrained from contacting the heater or its bracket.

On May 20, 1986 an NRC inspectors along with Limitorque, GPC, and Bechtel personnel performed a random inspection on four type SMB-000 operators installed in Vogtle Unit 2. Three of the four operators were found to contain burnt wiring. The wiring in question was installed by Limitorque and was forced into a small area in the limit switch compartment around the space heater bracket. The wire lengths were excessive making it difficult to fit all the wire into the small compartment.

Limitorque's QA Administrator, who witnessed the inspection at Vogtle, indicated that the problem observed at Vogtle was caused by mishandling in the field and could not have possibly been caused by improper wire routing at Limitorque. GPC personnel disagreed with this observation. On August 19, 1986, the NRC issued Information Notice 86-71 alerting licensees to this potential problem.

The NRC inspector reviewed applicable inspection procedures and wiring drawings at Limitorque and noted that documentation addressing the need to keep wiring away from space heaters or space heater brackets did not exist. The Limitorque Wiring Supervisor indicated that the space heaters used in the operators in question were made of a special chromolox type material which necessitated the use of a special bracket which created a problem due to the bulkiness of the heater bracket resulting in limited space for wire routing. Restraints were not installed at Limitorque to keep the wiring away from the brackets. After viewing pictures of the wiring at Vogtle, the supervisor indicated that the wire routing appeared to be normal except for the fact that the wires were burnt and were in contact with the heater brackets. Limitorque has stated in internal documents that Quality Control Procedure 5 will be revised by December 31, 1986 to include instructions to keep wiring away from heaters and heater brackets.

During the inspection, the QA Administrator stated a 10 CFR Part 21 evaluation was not required because the problem was caused by mishandling in the field. According to Procedure No. QCP-22 "Reporting Defects for Safety Related Equipment-Reference 10 CFR 21" the QA Administrator should have forwarded this problem to the Design Review Committee to permit a full evaluation. This problem constitutes a defect in a basic component, and an adequate evaluation should have been made to assess possible 10 CFR Part 21 reporting. (see Violation 86-01-02)

7. Blocking Valve Actuators

During the NRC inspection in November 1985, Limitorque was requested and was unable to provide Order File No. 3H8897 which contained information pertaining to the blocking valve actuator involved in the "Salem Unit 2 Depressurization Event" discussed in NRC Memo AEOD/E509 dated July 25, 1985. Limitorque was subsequently able to locate and provide the file which contained a bill of materials for the original blocking valve actuators as well as one for the replacement torque switches installed in the actuators at the time of the event.

The torque switches in question were factory set to actuate at 86 ft-lbs, which corresponds to a setting of 1 on the original torque switches and a setting of 1 3/4 on the replacement switches. Specific torque-turn curves are developed at Limitorque for each specific type of torque switch, and these curves must be utilized to properly set the switches. Limitorque noted that licensees should not set new torque switches to old torque switch settings, as settings sometimes vary between original and replacement torque switches as noted by the torque-turn curves.

8. Control of Purchased Material and Services

The inspector reviewed the overall controls that Limitorque imposes upon vendors supplying material and services to assure that applicable specification and QA program requirements were included or referenced in procurement documents, material and services were purchased from qualified vendors, and items meet the technical and quality requirements identified in the POs.

POs and Receipt Inspection Reports (RIR), if applicable, for six vendors of items utilized in a nuclear actuator and one service vendor were reviewed. For all 13 POs, it was noted that QA/QC personnel did not review any of the POs as evidenced by no documented concurrence.

PO No. 3579 dated September 19, 1986 to Barker Jennings for calibration of a set of Brown & Sharp gage blocks specified MIL-STD-45662 and traceability to the National Bureau of Standard that did not specify the requirements of 10 CFR Part 21. WESCO supplied Rockbestos cable under PO No. 1825 dated April 18, 1986. The PO specified the requirements of IEEE 383 and the Rockbestos Quality Manual-Nuclear Safety Related Products but did not document 10 CFR Part 21 requirements (See Violation 86-01-01). A September 17, 1986 memo from the QC Manager to Receiving Inspection documented six activities (e.g., verify wire meets PO requirements, verify that CofC has been reviewed, etc.) to be accomplished prior to releasing Rockbestos wire from the inspection station.

Terminal blocks are ordered from a supplier (e.g., PO No. 40958 dated September 9, 1986 to Foster Electric) to a Buchanan or Marathon catalog number. Receipt inspection only involves verification that the part no. on the box agrees with the part no. on the PO. Gear Limit Switches (GLS) consist of four rotors and a finger base. Two POs to Harvey Hubbell for GLS specified Part 21 requirements and

required a material certification. Rotors and finger bases were ordered on two separate POs from Harvey Hubbell. As noted in RIRs, on a sampling basis both items were visually and dimensionally inspected.

PO Nos. 21287 (August 15, 1986) and 21449 (September 15, 1986) were to Reliance Electric for AC motors. Both POs specified that the motors were to be in accordance with Limitorque specification EQC-10 and Drawing No. 21-497-0001, Revision H. The drawing documented 14 specific items. Item No. 13 which addressed the requirements of 10 CFR Part 21 was added in Revision F dated November 16, 1978. Revision H dated July 9, 1985 added Item No. 14 which stated, "Motors to be supplied in accordance with Reliance Electric Company Quality Assurance Program, Rotating Machinery Group dated September 1984." Receipt inspection is not performed on AC motors.

Two POs (Nos. 21372 dated August 29, 1986 and 21429 dated September 15, 1986) were to H. K. Porter for DC motors. Both POs specified "...Routine MTR Test, CofC, Viton Seals and O-rings, submersible motors...built in accordance with H. K. Porter QC Procedure Plant 1 dated April 14, 1982 and Limitorque specification EQC-1 and Drawing No. 21-497-0014-1, Revision C...CofC required." Receipt inspection is not performed on DC motors.

Shaw Plastics supplied torque switch terminal blocks on two POs. The POs specified Fibrite FM-5064 and a Limitorque drawing along with Part 21 requirements and requested a material certification. On a sample basis, a visual and dimensional inspection was performed on these terminal blocks.

It was noted that Limitorque had approximately 12 QA manuals from vendors on file. The Reliance Electric "Quality Assurance Manual" - Rotating Machinery Group, revised September 1984 was the only controlled copy. There was no documentation available to show that any of the manuals were reviewed and approved by Limitorque. The remaining manuals included:

Webber QCM-1B, "QC Manual" dated October 21, 1983
H. K. Porter "Quality Control Procedures" Plant 1 dated April 12, 1982
Force "Quality Control Manual" dated July 21, 1986
John Fluke "Quality Assurance Manual" dated July 1979
American Spring & Wire Specialty "QC Manual" dated January 26, 1984
Crofoot Gear "QC Manual" (no date)

General Electric - Apparatus Service Division "QA Program" dated July 28, 1983
Ryerson Steel "Quality Control Program" dated February 1980
Walker Machine & Foundry "QC Manual" dated 1982
Southern Centrifugal "QC Manual" dated May 10, 1985
Rockbestos "Quality Manual-Nuclear Safety Related Products" dated September 1982

Vendors are placed on the AVL if the vendor possess the capability to satisfy the requirements of the PO, the quality of the material/item supplied meets the specific requirements of the PO, and they have an acceptable QA/QC program. Limitorque sends Vendor Appraisal Forms (VAC) to a prospective supplier/manufacturer, and the vendor fills in the two page questionnaire and returns it to Limitorque. The NRC inspector reviewed approximately 20 VACs from vendors supplying items such as die castings, gears, springs, wire, and custom molded plastics.

A review of external audits indicated that eight audits were pre-viewed from 1985 to the present. The QC Manager indicated that the eight audits presented approximately 70% of the total of external audits performed by Limitorque. Webber Gage Division (calibrates the linear gage blocks) was audited in April 1985. The QC Manager audited Force, Inc. (electrical wire) in September 1986 with a 'new checklist' which incorporates 13 criteria from Appendix B and is a significant improvement over the old check list. Peerless/Winsmith (DC motors) was audited in March 1985. Three different locations of Reliance Electric (AC motors) were audited in December 1983 and February and March 1984. Fluke Manufacturing (Fluke motors) was audited in April 1985. An internal memo indicated that General Electric-Apparatus Service Division was audited May 31, 1985, but there was no evidence of a checklist or an audit report.

Approximately 14 RIRs and Limitorque drawings for 19 vendors were reviewed at the receipt inspection station to assure that specific characteristics of items were being inspected. The receipt inspector uses the dimensions on the Limitorque drawing to determine the number of checks to be made of an item. The visual and dimensional inspection is based on a sampling basis per MIL-STD-105. The vendors having RIRs included Advance Pressure Casting (gear limit drive cartridge), Belgium Tool & Die (machine drive sleeve), Crofoot Gear (miter gear), Harvey Hubbell (gear limit switch rotor), American

Spring and Wire Specialty (torsion spring), Marion-Heil Bronze (motor clutch gear bushing), Maryland Precision Spring (clutch compression spring), Newcomb Spring (tripper spring), Alling-Lauder (spur gear), Schnorr (belleville spring), WESCO (wire), and Foster (wire). A "receipt inspection stamp" is used in place of a RIR for castings (Souther Centrifugal and Lynchburg Foundry) and steel products (Ryerson and Copperweld Steel).

Limitorque uses a VER on a semi-annual basis to maintain or leave a vendor on the AVL. Based on a review of 15 VERs for 11 vendors whose rating was below Limitorque's quality standard and the vendor still remained on the AVL, it appears the QA program does not take into account the rating of a nonconformance to be a major or minor defect as sufficient reason to remove or leave a vendor on the AVL.

9. Equipment Qualification (EQ) Items

The inspector reviewed supporting data and obtained additional information on the EQ status of relevant features of Limitorque operators. The features addressed were grease relief valves, "T" drains, mounting orientation and grease/lubrication requirements. Other topics included qualification by similarity analysis, tests on degradable parts, quality/maintenance of EQ status on manufactured parts, and control of parts and services.

Testing by Limitorque to support qualification of operators inside containment without grease relief valves was not conducted. All operators tested had grease relief valves. The test manager recalled seeing grease stains on the outside of operators after the tests which indicated that the relief valve had operated. In addition, in test project F600 376A (Report F-C 3441) grease from the operator plugged the test chamber drain. This was interpreted as an indication of a significant discharge of grease from the relief valve.

Two tests were run on operators for outside containment applications without grease relief valves. While no deleterious effects were observed in these tests, they did not duplicate the Design Basis Event (DBE) temperature/pressure excursions which are postulated to occur inside containment.

All inside containment tests on RH class motors used "T" drains. One was used on the lowest part of each of two end bells of the Reliance motors, and the other ports were plugged. The Porter/

Peerless motors were equipped with a single "T" drain in the only drain port which was oriented to be at the low part of the motor. The class B motors tested for outside containment as well as the high temperature class B RAD H Reliance motors tested in 1969 were not equipped with "T" drains. The current replacement is the class RH motors which do require the "T" drains to conform to the tested configuration. While the temperatures and pressures listed in Project Report 600198 are higher than those expected outside containment they are lower than those used for testing of motors for inside containment.

With regards to orientation, all operators were tested with the limit switch mounted vertically upward and the motor horizontal. Limitorque feels that other orientations are satisfactory but recommends avoiding an orientation which puts the gear case above the motor or limit switch to prevent problems resulting from lubrication accumulation in these electrical components. However, Limitorque feels that undesirable orientations can be tolerated if an enhanced surveillance/ maintenance program is implemented to check for and correct any seal leakage. The recommendations concerning "T" drains in the lowest motor end bells continue to apply in off-normal orientations.

Limitorque currently supplies operators with Exxon Nebula EP0 in the main gear box and Beacon 325 in the geared limit switch. Mobil 28 is recommended for the limit switch gears for ambient temperatures above 150°F and is considered an acceptable substitute for the Beacon 325. It is substituted either for convenience during manufacturing or in response to customer specifications. All the qualification tests were run with Nebula EP0 or EP1 in the main gear box, and the instruction and maintenance manual (Bulletin SMB-1-82C) notes that these are the only lubricants approved for nuclear use inside containment.

While deviations from test conditions can usually be supported logically, the analyses are conclusions of the manager of the project group which performed the tests. There are no documented analyses of the effects of the deviations. Limitorque believes that the decision on whether or not an operator is acceptable for use in a nuclear plant is the responsibility of the licensee.

Motor and motor lead insulation, limit switch and torque switch nonmetallic parts, terminal strips, lubricants and gaskets have all been addressed in test reports and other documents. The

only other degradable parts are O-ring and quad ring seals. Limitorque's position is that these rings are maintenance items and aging should not be a consideration. The tests did subject the O-rings and quad rings to DBE temperatures and pressures, and the Viton O-ring and quad rings received thermal aging and radiation in test No. B0212. Since all tests run for inside containment applications were run with "T" drains and it was established that the "T" drains and cable ways allowed the test environment to infiltrate the motor and limit switch compartment, maintaining the integrity of the seals and gaskets was not a concern except to exclude grease from electrical components. Limitorque feels that sufficient grease will stay in place after seal failure such that a significant number of actuator operations can be accomplished without gear or bearing failure.

E. PERSONNEL CONTACTED:

P. McQuillan* - QA Administrator
W. Miluszusky* - QC Manager
R. del Campo - Part Department Manager
D. Martin - Manager of Order Entry
D. Williams - Parts Department
D. Warsing - Technical Manager
F. Denham* - Executive Vice President
D. Eperson - Wiring Supervisor
D. Pillow - Inspector
J. Puryear - Inspector

*Attended Exit Meeting.

ORGANIZATION: PEEBLES ELECTRICAL MACHINES
EDINBURGH EH5 2XT SCOTLAND

REPORT NO.: 99901065/86-01	INSPECTION DATES: 10/06-08/86	INSPECTION ON-SITE HOURS: 15

CORRESPONDENCE ADDRESS: Peebles Electric Machines
ATTN: Mr. A. S. Fisher
General Manager
East Pilton
Edinburgh EH5 2XT Scotland

ORGANIZATIONAL CONTACT: Mr. G. Campbell, Quality Assurance Manager
TELEPHONE NUMBER:

NUCLEAR INDUSTRY ACTIVITY: Manufactures electrical rotating machines.

ASSIGNED INSPECTOR: Naidu 11/18/86
K. R. Naidu, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S):

APPROVED BY: E W Merschoff 11/21/86
E. W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Implementation of the quality assurance program in selected areas.

PLANT SITE APPLICABILITY: Diablo Canyon (50-275).

ORGANIZATION: PEEBLES ELECTRICAL MACHINES
EDINBURGH EH5 2XT SCOTLAND

REPORT NO.: 99901065/86-01	INSPECTION RESULTS:	PAGE 2 of 5

A. VIOLATIONS:

No violations were identified during this inspection.

B. NONCONFORMANCES:

No nonconformances were identified during this inspection.

C. UNRESOLVED ITEMS:

No unresolved items were identified during this inspection.

D. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Background Information

The electrical generators for the emergency diesel engines installed at the Diablo Canyon Nuclear Power Plant (Diablo Canyon) were originally designed and manufactured by Electrical Products Incorporated (EPI) located in Cleveland, Ohio. Northern Engineering Industries Parson Peebles (NEI Parson Peebles) of Great Britian bought EPI in 1979 and renamed it Parson Peebles EPI. The manufacturing operations of Parson Peebles EPI ceased in the fall of 1984. Peebles Electrical Machines, a subsidiary of NEI Parson Peebles is currently manufacturing a spare electrical generator for the Diablo Canyon Nuclear Power Plant. Parson Peebles EPI is furnishing the design drawings and all the components available from their storage to Peebles Electrical Machines for the manufacture of an identical spare electrical generator.

2. Review of Purchase Order

Parson Peebles EPI, Cleveland, issued purchase order (PO) No. 14673 to Peebles Electrical Machines (PEM) for the supply of a 3 phase electrical generator rated for 2600 kw/3250 kva, 900 revolutions per minute, 4160 volt, 60 hertz. The PO attached the technical specifications issued by Pacific Gas and Electric Company, which included the electrical tests to be performed on the completed generator such as open circuit saturation curves and losses, short circuit saturation curves and stray losses, temperature test by simulated loading at zero power factor, temperature rise by increase in resistance, high voltage tests on the rotor and stator, overspeed and efficiency by summation of losses.

PEM translated the PO requirements into their General Order (GO) listing the various components required to manufacture the electrical generator. The GO was circulated to the various departments including mechanical and electrical design offices, and quality control for review and acknowledgement by signature and date. The suppliers for each component were identified against each item on the GO to enable procurement. PEM procured many items such as winding insulation material, sliprings, brushes, etc. from NEI EPI, who supplied the items from their inventory. POs were issued for items which were not available from NEI EPI stock.

3. Review of the Inspection Program

The inspector accompanied by a quality control engineer toured the plant and reviewed the inspections being performed in the areas of receipt inspection, storage of material and in-process manufacturing operations. The objective was to verify the implementation of the inspection program during the manufacture of electrical rotating machines. Individuals assigned to the specific inspections who were interviewed during the process were knowledegable in the activities they performed.

a. All incoming material is received at a designated area, inspected to verify that the material meets the PO requirements and transported to a designated storage area. The storage area was selected to preserve the integrity of the material. For instance, insulation material was stored in a refrigerated room.

b. The components received from NEI EPI were stored in a designated area. The insulation tapes for the windings were stored in a refrigerated enclosure. Certificates of Conformance (CoCs) were available. However, the individual items lacked unique identification and could not be traced to the item numbers of the PO and subsequently the CoCs. The quality control engineer took immediate corrective action by writing deficiency reports and affixing nonconformance tags to the components, as required by the quality control procedure. PEM contacted NEI EPI and requested they identify each item supplied with the relevant GO item number for future shipments and provide a cross reference in the CoC. It was not difficult to establish the identity of the items supplied so far, since only a few items supplied by NEI EPI were in storage. Subsequent to the inspection, on October 14, 1986, the inspector contacted the NEI EPI representative in Cleveland and determined that all items being shipped to PEM would be adequately identified by GO item and cross referenced in the CoCs.

c. The inspector toured the storage area and determined that purchased material was segregated by the GO number. Insulation material which is sensitive to changes in ambient temperature was identified by the GO number and stored in a refrigerated enclosure, the access to which was limited to authorized persons.

d. The inspector witnessed coil bending and forming operations, and insulation of windings being performed in the winding shop. Appropriate drawings were being used. In the core assembly area, core laminations were being assembled. Quality plan QO3-09-16 dated December 20, 1978, Revision 4, titled, "Standard for cores, inspection procedures for stator and core laminations" was available at the work area. Laminations inspected and accepted by the foreman of the stamping shop were being used to assemble the core.

4. Review of Calibration of Measuring and Test Equipment

The inspector randomly selected test equipment and verified that they were currently in calibration. Review of the calibration records indicated that standards traceable to the National Physical Laboratory (Great Britian) were used to calibrate the equipment.

5. Review of Audits Performed on PEM

The PEM program is being revised to conform to a recent reorganization at NEI. The manufacturing facilities are being moved from adjacent buildings and relocated in a central building. Representatives of Diablo Canyon inspected PEM and plan to witness the acceptance tests on the generator which are scheduled for January 1987. Representatives of the Central Electric Generating Board (CEGB) of England, which owns several nuclear power plants, had performed an audit of the revised PEM QA program and corrective action on adverse findings were being implemented. PEM is an approved vendor for CEGB.

E. EXIT INTERVIEW:

The inspector met with PEM representatives mentioned in Section F at the conclusion of the inspection and discussed the scope and findings of the inspection.

ORGANIZATION: PEEBLES ELECTRICAL MACHINES
EDINBURGH EH5 2XT SCOTLAND

REPORT NO.: 99901065/86-01	INSPECTION RESULTS:	PAGE 5 of 5

F. PERSONS CONTACTED:

*A. S. Fischer, General Manager
*G. Campbell, Quality Assurance Manager
*R. B. Smith, Engineering Manager
D. Brunton, Manager, Drawings Office
R. A. White, Quality Control Manager
C. Malecky, Test Engineer
E. McRorie, Purchase Agent
A. Mill, Manager, Sales & Marketing
A. Cowie, Contract Engineer

*Denotes those individuals present at the exit meeting.

ORGANIZATION: ROCKWELL INTERNATIONAL
FLOW CONTROL DIVISION
SULPHUR SPRINGS, TEXAS

REPORT NO.: 99901069/86-01	INSPECTION DATE: 10/20-24/86	INSPECTION ON-SITE HOURS: 28

CORRESPONDENCE ADDRESS: Rockwell International
Flow Control Division
ATTN: Mr. J. J. Osberg
Vice President, Plug Valve Products
1511 Jefferson Street, P. O. Box 501
Sulphur Springs, Texas 75482

ORGANIZATIONAL CONTACT: Thomas W. Schlemm, QA Manager
TELEPHONE NUMBER: (214) 885-3151

NUCLEAR INDUSTRY ACTIVITY: Manufacturer of globe and check valves.

ASSIGNED INSPECTOR: for J. T. Conway, Reactive Inspection Section (RIS) — 11/17/86 Date

OTHER INSPECTOR: T. Tinkle, Consultant

APPROVED BY: Ellis W. Merschoff, Chief, RIS, Vendor Program Branch — 11/17/86 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 an evaluation of the implementation of the QA program in the areas of training, control of special processes, reporting of defects, control of purchased material, nondestructive examination, control of measuring and test equipment, and audits.

PLANT SITE APPLICABILITY: Not identified.

A. VIOLATIONS:

1. Contrary to Section 21.31 of 10 CFR Part 21, a review of documentation packages for Section III nuclear valves revealed that while 10 CFR Part 21 was imposed on Rockwell International (RI) by their customers. RI did not specify that 10 CFR Part 21 requirements would apply on Purchase Orders (PO) L6612 to Sherwin, M6635 to RI-Tulsa, L1119 and P1539 to Weldstar, and L4482 to HyLevel Screw Products. (86-01-01)

B. NONCONFORMANCES:

1. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Subsection 1.1 of the Quality Assurance Manual (QAM), Subsection NCA-4134.6 of the ASME Code, Section 7 of ANSI N45.2-1977 and Subsections 5.1.7 and 6.2.3 of the QAM, it was noted that: (86-01-02)

   a. Revision A dated January 31, 1984 to RMP No. 61176 "Primer Coating of Carbon Steel Valves for Nuclear Service (Mobil zinc)" was not at the coating station.

   b. Revision B dated April 8, 1986 to RMP No. 61360 "Combination GTAW Root Fuse and SMAW Filler Welding of P-1 Materials" was not in manuals No. 11 (Nuclear Assembly and Test Supervisor) and No. 17 (Plant Metallurgist).

   c. RMP No. 61369, Revision C "Procedure for Welding of Permanent Attachments on Stainless Steel Univalves and Hermavalves for Nuclear Service" was in manual No. 11, but a DRF dated November 12, 1985 indicated that RMP No. 61369 was obsolete.

   d. RMP No. 61150, Revision A "Assembling and Testing Rockwell-Edward Hermavalves with Leakoff (for Westinghouse)" was missing from manual No. 4 assigned to the QA Department.

2. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Subsection 10.5.1 of the QAM, and Section 9.6.1 of SNT-TC-1A, a review of qualification records for nondestructive examination personnel revealed that the records for the Level III examiner did not contain a statement indicating satisfactory completion of training in accordance with QA Instruction No. 40-03-12. (86-01-03)

3. Contrary to Criterion V of Appendix B to 10 CFR Part 50, Subsection 12.2.1 of the QAM, and Sections 15.2 and 15.3 of QA Procedure 761-40-04-12, a stationary Ph measuring instrument (RI Asset No. 61-3149) was found to be out of calibration and not identified with a REJECT tag or label. (86-01-04)

4. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and RMP No. 61150, nuclear stainless valves (Assembly Lot H-7510) were hydrostatically tested on October 16, 1986, using Grade B rather than Grade A water. (86-01-05)

5. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Subsection 7.1.4 of the QAM, PO M4454 was issued to Page Wilson for hardness test samples on August 6, 1986, but Page Wilson was not on RI's Approved Vendor List. (86-01-06)

6. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Subsection 9.3.1.5 of the QAM, hardness test results on stem and disc material for nuclear valves on assembly lot numbers H-, 4175, 6866, 7011, 7191, 5824, and 5726 were not recorded on an Inspection Card. (86-01-07)

7. Contrary to Criterion V of Appendix B to 10 CFR Part 50 and Section 6 of ANSI N45.2, documented instructions/procedures did not exist for the accomplishment of specific inspection requirements (i.e., hardness checks) on the "In-process Heat Treatment" report. (86-01-08)

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

None. This was the first NRC inspection at this facility.

E. OTHER FINDINGS AND COMMENTS:

1. Control of Measuring and Test Equipment (M&TE)

The inspector reviewed Section 12 "Control of Measuring and Test Equipment" of the QAM and Procedure No. 761-40-04-12 relating to the calibration control of M&TE. An inspection was made to determine the calibration status of various measuring instruments

found in the heat treatment area, welding areas, and the nuclear test areas. The measuring instruments on the heat treatment furnaces are calibrated in situ by a Field Service representative from Honeywell. These instruments were recently calibrated during the period June 23-25, 1986. The Rockwell hardness tester (Model SDR-P) in the heat treatment area was serviced by Wilson Instruments on December 3, 1985 as noted by a sticker on the tester.

The Pressure Gage Calibration Cards for gage X612 were located in the nuclear test area. It was noted on the card for Assembly Lot H7087 that the gage was calibrated at 1:30 on November 22, 1985, prior to testing and was calibrated at 12:00 on November 23, 1985, after testing. The calibration sticker for the in-line Ph meter (RI Asset No. 61-3149) at the demineralized water station in the nuclear test area had a calibration due date of April 1985. The instrument did not have a REJECT label affixed (see Nonconformance 86-01-04). This matter was brought to the attention of the QC Supervisor who stated that the instrument was not being used. He further stated that the portable, digital Ph meter (Serial No. AAIU) located on a stand next to the in-line Ph meter was being used for Ph measurements of water for hydrostatic testing of valves.

The following Instrument Calibration Cards were checked during an inspection of the Metrology Laboratory:

Item	S/N	Calibration Due Date
Digital Ph Meter	AAIU	Nov 85
Pressure Gage	X547	Feb 87
DC Amps/Volts	ADCD	Apr 87
DC Amps	ADCF	Apr 87
Gas Regulator	7502B	Oct 87
Dead Weight Tester	13504	Mar 87
Ph Meter	61-3149	Apr 85
Wall Thickness Gage	ACNR (NWT-6)	Jan 87
Wall Thickness Gage	ACNQ (NWT-4)	Jan 87

The calibration status of the in-line Ph meter in the nuclear test area was discussed with the metrologist. He stated manufacturing had advised him that the meter was not being used. A note to this effect was found on the instrument calibration card. The calibration history for wall thickness gage NWT-4 was reviewed. The record shows NWT-4 was in calibration when used on October 1, 1982 for Assembly Lot H4602 (Body Serial No. LND-57, 59, and 77).

The Basic Maintenance Agreement (BM-0109) with Honeywell covering the June 86 calibration services for the heat treatment furnace instrumentation was reviewed. The calibration certification indicates Honeywell used a Model 72-311 Biddle (S/N 78940) which was calibrated on May 16, 1986, by the same Honeywell representative providing the calibration services to RI. The certification stated the standard was traceable to the National Bureau of Standards. RI PO L7969 was issued to obtain the calibration services under BM-0109. The meteorologist stated his normal practice is to observe in situ calibration work. However, documentation to support this statement was not available for the June 1986 Honeywell calibration services.

2. Test Control

A pallet of nuclear stainless valves (S/N CD-356 through CD-362) was located in the nuclear test area. Documentation identified the valves as Assembly Lot H7510. Investigation revealed the valves were being built for Westinghouse and had completed hydrostatic testing within the past few days. The Hydrostatic Test Report for this order was located in the QA Department and reviewed. The valves were hydrostatically tested on October 16, 1986 per RMP 61150, and the chemistry results for water used during the testing indicated a conductivity reading of 4 umho/cm.

A review of RMP 61150 noted that Grade A water (maximum conductivity of 2 umho/cm) is required for hydrostatically testing stainless valves, and Grade B water (maximum conductivity of 20 umho/cm) should be used for carbon steel valves. The test report indicated that the stainless valves on Assembly Lot H7510 were hydrostatically testing used Grade B rather than Grade A water (see Nonconformance 86-01-05). The matter was brought to the attention of the QC Supervisor who took immediate action to reject these valves.

3. 10 CFR Part 21 Compliance

The inspector reviewed QA Procedure 761-40-52-03 "10 CFR 21: Reporting of Defects and Noncompliance" dated June 27, 1986. The procedure indicates the Product Engineering Manager shall designate on engineering control documents those items which have critical functions in terms of Section 21.3 of Part 21, and the Purchasing Manager shall state in POs for critical items that conformance with 10 CFR 21 is required. Furthermore, the QA Manager is to create a file of all plant transactions involving 10 CFR 21 and is to document all transactions of the 10 CFR 21 Evaluation Committee.

A review of QA Procedure 761-71-04-03 noted that this document identifies valve and valve parts which have critical functions in terms of Section 21.3 of 10 CFR 21. The valve parts which have critical functions are the body, bonnet, cover, disk, bonnet bolting, stem and actuator. It was also noted that drawings for parts with critical functions shall have a note added to the drawing which reads 10 CFR 21 applies, and Material Specifications (RMC) for parts which have critical functions shall contain a statement which reads 10 CFR 21 applies.

The QA Manager stated that files did not exist for 10 CFR 21 transactions since there had been no transactions, and the 10 CFR 21 Evaluation Committee had never met. A Bill of Material for Assembly Lot H7510 was reviewed, and it referenced Drawing A-21602682 for stainless steel Hermavalves, and the drawing invoked 10 CFR 21. The following RMCs were selected for review and found to invoke 10 CFR 21:

- RMC 26076, Rev. D, dated July 28, 1986, "Austenitic Type 308 Welding Electrodes for Nuclear Service."

- RMC 26075, Rev. E dated July 28, 1986, "Austenitic Type 312 Welding Electrodes for Nuclear Service."

- RMC 70007, Rev. E, dated July 28, 1986, "Carbon Steel Forgings for Nuclear Service."

A number of RI POs were reviewed to determi if RI was invoking 10 CFR Part 21 on vendors. It was noted that RI failed to impose Part 21 requirements on POs L6612 dated August 22, 1985 to Sherwin for liquid penetrant (PT) materials; M6635 dated October 8, 1986 to RI-Tulsa to calibrate gage blocks; L1119 dated February 1, 1985 and P1539 dated August 29, 1983 to Weldstar for welding electrodes; and L4482 dated March 30, 1981 to HyLevel Screw Products for disks. (see Violation 86-01-01)

Three locations prescribed by Section 1.3 of QA Procedure 76-40-52-03 for 10 CFR Part 21 posting were checked. The required notice was found in the Nuclear Production Lab, the QA Department, and the bulletin board adjacent to the cafeteria. Posting prescribed for the Product Engineering area was not checked.

4. Indoctrination/Training of QA Personnel

QA Instruction 40-71-00 "Indoctrination and Training of Quality Assurance Personnel" was reviewed. This instruction requires the results of indoctrination and training activities to be documented and filed. Topics, dates, duration of meetings, and names and job titles of attendees are to be recorded. The indoctrination and training records for the following three individuals were reviewed with the QC Supervisor:

ID No.	Function	Period
GS121	Wall Thickness Measurements	Dec 83-Mar 84
GS135	Wall Thickness Measurements	Dec 85
LT44	Leak Tester	Nov 85

The indoctrination and training records for GS121 and GS135 contained only two entries and both were dated July 29, 1983. However, a detailed review of QC Department Communication/ Training Meeting Reports in the possession of the QC Supervisor indicated that both individuals had attended numerous training sessions over a period of years.

5. QA Audits

The inspector reviewed QA Procedure 761-40-22-08 "Quality Assurance Audit Programs" and QA Procedure 761-40-64-04 "Qualification of Quality Assurance Program Audit Personnel." A review of the internal audit records indicated that each section of the QAM was audited annually, and the last internal audit was completed in June 1986. Selected portions of the audit findings were reviewed. The qualification records for the individuals who performed the audit were reviewed, and all were found to be qualified as an auditor or lead auditor at the time of the audits.

6. Nondestructive Examination (NDE)

The NRC inspector reviewed QA Instruction 40-03-12 "Training, Examination and Certification of Nondestructive Testing Personnel" and the qualification and certification records of eight NDE personnel (two-Level III, four-Level II, and two-Level I) to determine that individuals performing PT examinations were qualified and certified to SNT-TC-1A. The instruction indicated that training and examination

were to be in accordance with SNT-TC-1A. Copies of tests and annual eye examinations were on file for each individual. With the exception of a statement showing satisfactory completion of training in accordance with QA Instruction 40-03-12 for one Level III examiner, the qualification records of the eight NDE personnel appeared to be in accordance with SNT-TC-1A (see Nonconformance 86-01-03).

7. Control of Purchased Material and Services

The inspector reviewed Section 7 "Control of Purchased Items and Services," QA Instruction 40-64-00 "Review of Purchase Order and Vendor Furnished CMTRs" vendor audit reports, and Approved Vendor Lists (AVL) to assure that items/services were being purchased from qualified vendors. The reports selected for review included audits of Texas Bolt in March 84, Consolidated Castings in August 84, and Sherwin in February 85. A review of past AVLs on file in the QA Department indicated that a number of vendors hold an ASME Quality System Certificate (QSC) for materials. Three QSC vendors were selected to determine if copies of the QSCs were on file. The Supervisor of QA Engineering provided copies of current QSCs for each of the vendors and stated that RI only retains copies of current QSCs.

The inspector reviewed approximately 27 POs issued to 12 vendors to assure that quality requirements (e.g., ANSI N45.2 and 10 CFR Part 21) were passed on to suppliers/manufacturers of items/services pertaining to valves destined for nuclear service. The POs reviewed included: four to Texas Forge and three to Trinity Forge for SA 105/182 bodies, two to Consolidated Castings for disks, three to Coulter Steel and Forge for SA 479/696 bar, four to Crucible Steel for SA 564 bar, one to Sherwin for PT material, one to Teledyne-McKay and two to Weldstar for welding electrodes, two to Page Wilson and one to Honeywell for material and calibration services, two to Texas Bolt for SA 193 fasteners, and two to HyLevel Screw Products for A565 disks.

It was noted that PO M4454 dated August 6, 1986 was issued to Page Wilson for test blocks for the Rockwell hardness tester. A review indicated Page Wilson was not on the AVL when the PO was issued (see Nonconformance 86-01-06). This matter was discussed with the Supervisor of QA Engineering who stated that Page Wilson's QA Program had not been audited.

8. Documentation Packages (DP)

The inspector reviewed 13 documentation packages and 10 POs from six customers for Section III, Class 1 and 2, globe and check valves. The 10 POs from 1981 through 1985 from Georgia Power (two), Carolina Power & Light (two), Texas Utilities Generating Company (two), Portland General Electric (one), Florida Power & Light (two), and Tennessee Valley Authority (one) were for valves two inches smaller and referenced technical specifications, 10 CFR Part 21 and Section III of the ASME Code. The DP consisted of a Quality Verification Documentation List (QVDL); a RI Certification Package which included a Code Data Report, Certificate of Compliance (CofC) for wall thickness measurements, Metallurgical and Valve Test Certification, and a Final Acceptance Plan; Certified Material Test Reports (CMTR); NDE Records; Heat Treatment Reports; Hydrostatic & Functional Test Report; Cleaning Records; Travelers; and a Certificate of Seismic Qualification.

The QVDL certified that the documents for nine categories (e.g., hydrotest, heat treat, etc.) met the quality verification requirements of the specification. The CMTRs were from various manufacturers for body forgings, bonnets, disks, stems, and weld wire. The NDE records showed that body, bonnet and disk material were all PT examined. The record identified the Level I and II examiners, the procedure and chemicals used, and the date of the examination. The heat treatment records included a "In-Process Inspection Heat Treatment" document and time/temperature charts. It was noted that hardness checks were recorded on the heat treatment document (see Nonconformance 86-01-07).

14. Welding

The inspector reviewed Volumes I, II and III of the Nuclear Welding Control Manual assigned to the Plant Metallurgist. Volume I contained Welding Procedure Specifications (WPS), Procedure Qualification Records (PQR), and welder's Qualification Tests. Five welding RMPs selected for review included 61038 "GTAW Canopy Seal Weld," Revision J; 61046 "SMAW Fillet Seal Weld", Revision T; 61093 "Nuclear Projection Resistance Weld," Revision D; 61105 "GTAW Canopy Hermavalve," Revision D; and 61149 "GTAW Canopy Sealweld E308," Revision B. The RMP included the WPS and PQR. The welder's Performance Qualification Records for the five WPSs noted above were also reviewed. It was noted that welder No. 51 was qualified to four of the procedures, welder No. 55 to two procedures and welder Nos. 60 and 70 to one procedure each.

Volume II contained Welder Symbols and the Qualification Status of each welder for a particular process (e.g., SMAW, GTAW, etc.). The Qualification Status records documented the process, date, job No., welder's signature, and supervisor's signature. These records have been maintained on a monthly basis from 1972 to the present.

Volume III contained the visual requirements and welder training program. Each welder is given an eye examination on an annual basis. The welder/operator training program consists of a nuclear department tour and classroom sessions on "welding" and RMPs. A refresher training program is given annually to each qualified welder. A review of Nuclear Welding Training Sessions records for welder Nos. 51, 55, 60, and 70 indicated that each individual was trained on an annual basis.

During an evaluation of the manufacturing facility, it was noted that welding RMPs were retained in individual folers in a metal cabinet. The QC Supervisor told the NRC inspector that the Manufacturing Supervisor was responsible for releasing specific RMPs and weld wire for a particular nuclear job. The inspector reviewed three log books for 308, 316, 7018 weld material used on nuclear valves. Documentation identified the job No., welder, amount of material released and retained, and it was signed by the Manufacturing Supervisor.

15. Plant Tour

The inspector toured the RI manufacturing facility at various times during the inspection in the company of RI officials. Items witnessed included: receiving area and inspection; heat treatment area which uses batch and vacuum furnaces; cleaning area which included electrolytic and diaphragm cleaning processes and electroless nickel plating process; barstock storage; welding areas which included projection resistance welding, manual surfacing, and thermal upsetting; machining of bodies and parts; finished part stores which included storage of weld material; nuclear inspection area incorporating PT, wall thickness measurements on forged body, and final inspection; assembly area made up of seat polishing, valve assembly, and hydrostatic testing; coating area; gage calibration; metallurgical laboratory; tool crib; and shipping area

16. Document Control

While evaluating the manufacturing facility, a number of RMPs at various stations were selected to assure that current procedures were at work stations. Three RMPs (Nos. 61036, 61048, and 61153)

were reviewed at the heat treatment station. It was noted that the procedures were prepared by the Plant Metallurgist and approved by the QA Manager and the Chief Product Engineer. At the PT station, RMP Nos. 61050 and 61152 had been approved by the Level III examiner, the QA Manager, and the Product Engineering Manager. RMP Nos. 61160 and 61093 were reviewed at the final inspection and projection welding stations, respectively. At the cleaning station, RMP Nos. 61065, 61056 and 61189 were reviewed. The 10 RMPs noted above were consistent with the applicable revision contained in controlled manual No. 4 assigned to the QA Manager.

At the coating station, four RMPs (Nos. 61077, 61083, 61176, and 61191) were reviewed. It was noted that Revision A of 61176 was not at the station. RMP Nos. 61057, 61109, 61146, and 61150 were reviewed at the testing station. Although Revision A of 61150 was at the station, it was missing from the QA department's controlled manual No. 4.

At the laboratory station, six welding RMPs (Nos. 61038, 61046, 61105, 61149, 61360, and 61369) were reviewed. The welding procedures had been approved by the Plant Metallurgist, QA Manager, and Production Engineering Manager. It was noted that Revision B to No. 61360 was not in the manuals assigned to the Nuclear Assembly and Test Supervisor and the Plant Metallurgist, and Revision C to No. 61369 was in manual No. 11 (Nuclear Assembly and Test Supervisor) but RMP No. 61369 was obsoleted in November 1985. (See Nonconformance 86-01-02)

F. PERSONS CONTACTED:

*T. Schlemm, QA Manager
*R. Wolff, Supervisor QA Engineering
*E. Clifton, QC Supervisor
*R. Spears, Principal Nuclear Engineer
*D. Kowalczyk, Manufacturing Operations Manager
F. Jisha, Manager, Product Engineering
R. Johnson, Metallurgist
B. Lewis, QA Documentation Technician
N. Swann, QA Documentation Technician
R. Baxley, Metrologist
*D. Theareau, Engineering Manager, Globe Valves
*H. Beuford, ANI
*M. Bronson, Materials Engineer, QA

*attended exit meeting

ORGANIZATION: ROTORK CONTROLS, INCORPORATED
ROCHESTER, NEW YORK

REPORT NO.: 99900839/86-01	INSPECTION DATES: 09/09-12/86	INSPECTION ON-SITE HOURS: 24

CORRESPONDENCE ADDRESS: Rotork Controls, Incorporated
ATTN: Mr. W. H. Whiteley
President
19 Jet Drive
Rochester, New York 14624

ORGANIZATIONAL CONTACT: D. Matla, QA Manager
TELEPHONE NUMBER: 716-328-1550

NUCLEAR INDUSTRY ACTIVITY: Fabrication of electric and pneumatic actuators for valves.

ASSIGNED INSPECTOR: K. Naidu, Reactive Inspection Section (RIS) — Date: 11/10/86

OTHER INSPECTOR(S):

APPROVED BY: E. W. Merschoff, Chief, RIS, Vendor Program Branch — Date: 11/17/86

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review of implementation of the Quality Assurance Program in selected areas.

PLANT SITE APPLICABILITY: Fort St. Vrain (50-267), McGuire (50-369).

A. VIOLATIONS:

No violations were identified during this inspection.

B. NONCONFORMANCES:

No nonconformances were identified during this inspection.

C. UNRESOLVED ITEMS:

No unresolved items were identified during this inspection.

D. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Background Information

Rotork Controls, Incorporated (Rotork) manufactures electric and pneumatic actuators for valves. The operators were designed and components manufactured by the parent company, Rotork Controls, Limited, Bath, England. Rotork imports selected components supplied by the parent company and assembles the actuators in the U.S. In 1982, Rotork qualified their electric operators to the requirements of IEEE 382-1980, IEEE 323-1974 and IEEE 344-1975.

Rotork utilizes a modular construction with standard components consisting of the motor assembly, double acting thrust base, auxiliary switch contact assembly and terminal block assembly (synchroset) or reversing starter assembly (synchropack). The Rotork actuator is designed with a minimum number of moving components. The drive train consists of a motor specially designed for valve actuators, a single gear reduction through a worm and worm wheel set, a center column, and the drive sleeve. The worm gear is machined directly into the motor shaft. The motor is constructed such that the torque switch spring pack is located at the end of the motor and is easily adjustable. The motor assembly is sealed with an "O" ring. As such, space heaters are not considered to be necessary to prevent moisture condensation. The actuator body construction is such that the thrust is removed in a thrust base prior to the gear case. This arrangement permits the gear case/actuator to be of a lighter construction and allows the actuator to be serviced while the valve is in operation. Lubrication for the drive train is oil. Manual operation is through a direct drive handwheel.

The electrical compartment in the Rotork actuator is completely sealed from the environment through the use of male/female joints and "O" rings. The auxiliary switch contacts and the motor leads are internally connected to a separate terminal compartment for field connections. This terminal compartment is a sealed enclosure separate from the main electrical compartment.

2. Plant Tour

The inspector toured the plant in the areas of assembly, storage of material, painting shop, incoming receipt inspection and test area. The inspector interviewed operator assemblers and test technicians and determined that the individuals were knowledgeable in the work they performed.

3. Review of Purchase Orders

The inspector reviewed four purchase orders (PO) received by Rotork to determine whether the engineering, manufacturing and quality assurance personnel reviewed and accepted the technical requirements. The inspector determined that the managers of engineering, manufacturing and quality assurance review the purchase documents and indicate their concurrence by their signatures on a Certificate of Conformance that the operators will meet the specification requirements including IEEE 323-1974, IEEE 344-1975 and IEEE 382-1980. The following purchase orders were reviewed.

a. PO P29631-77 dated May 29, 1986 issued by Mill Power Supply Company (MPS) for two 16NA1 type actuators rated for 575 volts, 60H Hz and 57 revolutions per minute (rpm). The PO stated that MPS is the agent for Duke Power Company (Duke) and that the actuators are safety related items intended for installation at the McGuire Nuclear Power Plant. The PO required Rotork to comply with 10 CFR Part 21 and the Rotork quality assurance program which was previously approved by Duke. The PO required a quality release for the components prior to shipment. A Certificate of Conformance dated June 6, 1986 was signed by the managers of engineering, contracts, and quality assurance, in accordance with their QA manual requirements, to acknowledge that they reviewed the purchase specification requirements and that the Rotork operators would meet the appropriate technical requirements. The inspector reviewed the corresponding Rotork job number (JO) 09743 and determined that the technical requirements were translated to manufacturing specifications.

b. PO P24352-13 dated August 20, 1986 issued by MPS for the supply of five add-on-packs type NSK-10008. The add-on-packs contain components to increase the number of auxiliary switches on existing operators to provide additional indications or functions. Each add-on-pack contains 12 add-on-switches, 1 wire harness, 1 potentiometer, 1 adaptor, 1 cover and a set of bolts. During the design review of the PO, a discrepancy was identified in the purchase specification relative to the number of individual components. Rotork clarified this discrepancy with MPS and MPS issued a change to the PO resolving the discrepancy. The PO was being processed during the inspection.

c. Public Service Company of Colorado issued PO N7103 dated July 16, 1986, for the supply of one type 30NA1, 5 rpm actuator, and one type 70 NA1, 29 rpm actuator, both rated for 3 phase, 460 volt, 60 herz. Both operators were required to have water tight enclosures classified as safety related Class 1E. An attachment to the PO titled standard PO Document and Clause requirements required Rotork to implement a quality assurance program meeting 10 CFR 50 Appendix B requirements during the manufacture of the actuators and required Rotork to comply with 10 CFR Part 21 defect reporting requirements. Rotork was to provide a Certificate of Conformance (CoC) stating that the actuators meet IEEE 323-1974 and IEEE 344-1975 requirements. The inspector reviewed a CoC prepared and signed by the managers of contract, engineering and quality assurance to acknowledge that they reviewed the specification requirements and the Rotork actuators would meet all the requirements except paragraph 1.2 of the specification. Paragraph 1.2 required Rotork to supply assembly drawings to facilitate assembly of components. Rotork stated that they would provide outline drawings, nuclear maintenance manual (AE 5/6) and A-Range maintenance manual (AE 5). The inspector reviewed Rotork JO B 09820 dated July 30, 1986 and determined that the JO reflected the PO requirements.

d. Anchor Darling Valve Company (AD) issued purchase order (PO) E-1709 dated June 19, 1986 to Rotork to supply five 7NA1 type actuators which met Duke Power Company specification for nuclear safety related gate, globe and check valves CNS-1205-01-00-0003 Revision 5. Amendment #1 dated June 27, 1986 to the PO required five drive busings to be included in the supply. AD specified the technical requirements of the valve such as the seat mean diameter and area, line design pressure, design differential pressure, disc thrust, stuffing box load, total stem thrust, stem

diameter and thread, steam factor stem torque, disc total travel in inches, required drive sleeve rpm, and maximum voltage reduction. In compliance to the Rotork QA Manual, a Certificate of Compliance dated July 10, 1986 was prepared and signed by the managers of contracts, engineering and quality assurance to acknowledge that they reviewed the specification requirements and that the Rotork actuators would meet the PO requirements including the IEEE 382-1972, IEEE 323-1974 and IEEE 344-1975 standards which were specified. Additionally, it was highlighted that these actuators were intended for 3 phase, 208 volts 60 herz (and not the usual three phase 575 volt 60 herz for Duke).

Rotork internal JO B09789 translated the AD specifications for manufacture of the operators.

4. Review of Quality Assurance Records

The inspector reviewed the quality assurance documents relative to Duke PO P2963-77 for two 16NA1 type actuators which had been shipped recently. The records are considered typical for each operator and consisted of the following:

a. A Nuclear Quality Approval Checklist indicating that the following attributes were verified and determined acceptable:

(1) Inspection of various modules prior to assembly to procedure QC 136.

(2) Pressure test was performed to procedure QC 129.

(3) Verification of Grit Blast to procedure QC 130.

(4) Examination of Grit Blast to procedure QC 127.

(5) Elcometer readings (thickness measurement) verified to procedure QC 126.

b. Quality Assurance records on the external coatings applied to the operator which included the following:

(1) Surface preparations.

(2) Batch numbers of Carbozinc II, Phenoline 305 and their expiration dates.

(3) Temperatures at which they were stored.

(4) Details of paint application and measurements of thickness of the paint coating.

(5) Product identity and quality assurance certification records on the metallic gray final coating.

c. Final test certificate indicating that the following tests were performed:

(1) Circuit continuity was verified to meet the wiring diagram.

(2) Insulation resistance was verified by applying 2KV.

(3) The motor operated at 575 volts 3 phase 60 herz.

(4) Verification of the gear ratio (60:1) and speed (57 rpm).

5. Review of Rotork Purchase Orders to Vendors

The inspector reviewed purchase orders issued by Rotork to their vendors to determine whether Rotork specified applicable quality and technical requirements therein. The inspector determined that Rotork procures the majority of components from their Bath, England plant. Items such as primers, paint and special alloy castings are procured domestically. Review of the purchase orders indicated that Rotork specified the applicable quality and technical requirements.

6. Review of Control of Measuring Equipment

The inspector selected several measuring equipment used on the assembly and test areas, verified their respective calibration documents and determined that the calibration of the equipment was current. The records indicated that the equipment was calibrated to standards traceable to the National Bureau of Standards.

7. Review of Audits Performed

a. The inspector reviewed five audits performed by Rotork QA manager on various activities such as drawing control, procurement, assembly and testing. Timely responses were provided to audit findings and all the findings had been closed.

b. The inspector reviewed three audits performed by licensees on Rotork and determined that corrective action had been completed for all adverse findings.

E. EXIT INTERVIEW:

The inspector met with Rotork representatives mentioned in Section F and discussed the scope of the inspection and findings.

F. PERSONS CONTACTED:

W. H. Whitley, President
R. Arnold, Vice President, Engineering
D. R. Brown, Manager, Manufacturing
D. C. Matla, QA Manager
M. R. Saltzer, Manager, Purchasing
C. Warnett, Vice President, Sales

ORGANIZATION: ROTORK CONTROLS LIMITED
BATH, ENGLAND

REPORT NO.: 99901066/86-01	INSPECTION DATES: 09/09-10/86	INSPECTION ON-SITE HOURS: 14

CORRESPONDENCE ADDRESS: Rotork Controls Limited
ATTN: Mr. T. Eassie
Chief Executive
Bath, BA1 3JQ England

ORGANIZATIONAL CONTACT: G. Hawkins
TELEPHONE NUMBER: (0225) 28451

NUCLEAR INDUSTRY ACTIVITY: Fabrication of electric and pneumatic actuators for valves.

ASSIGNED INSPECTOR: _____ 10/30/86
K. R. Naidu, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S):

APPROVED BY: _____ 11/3/86
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 implementation of the Quality Assurance Program in selected areas.

PLANT SITE APPLICABILITY: Fort St. Vrain (50-267); McGuire (50-369).

A. VIOLATIONS:

No violations were identified during this inspection.

B. NONCONFORMANCES:

No nonconformances were identified during this inspection.

C. UNRESOLVED ITEMS:

No unresolved items were identified during this inspection.

D. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Background Information:

Rotork Controls Limited (Rotork) designs and manufactures electric and pneumatic operators for valves. Rotork operators are being used in nuclear power plants located in England and other European countries. Rotork supplies subassemblies to Rotork Incorporated (Rotork Inc.) located in Rochester, New York, who use the sub-assemblies to manufacture electric operators for use in domestic nuclear power plants.

This inspection was performed as a followup to an inspection at Rotork Inc. conducted during September 9-12, 1986. During this inspection, the implementation of the quality assurance program was verified in selected areas.

2. Plant Tour:

The inspector toured the plant in the areas of assembly, storage of purchased material, paint shop, incoming receipt inspection and test area. The inspector interviewed component assemblers, in-process inspectors, incoming receipt inspection supervisor and test technicians, and determined that the individuals interviewed were knowledgeable in the work they performed.

3. Review of Design Control:

The inspector verified the implementation of design control by reviewing the processing of the purchase orders (PO) received by Rotork. The POs are sorted in two categories, standard and non-standard products. The standard product order is processed on

white paper. The nonstandard product order which is used to distinguish operators for nuclear and special applications is processed on pink paper and requires the signatures of individuals from contracts, technical, engineering and quality control to sign the appropriate places to indicate that they reviewed the product request order and concur that the individual components of the operator are correct and that it will meet the relevant technical requirements. All operators intended for nuclear power plants are processed as nonstandard products. A Certificate of Compliance (CoC) is prepared for the signatures of the managers of contracts, engineering and quality assurance certifying that the actuator/assembly/component (as applicable) when used on the valve identified therein, complies with the relevant specification subject to exceptions (if any) documented on the CoC. The inspector reviewed two processed nuclear product orders and determined that the three managers signed them in compliance with the quality assurance program requirements. Changes and modifications to the products undergo the same review cycle.

4. Review of Purchased Material and Services:

The inspector reviewed compliance to established quality assurance procedures relative to the purchase of material and services. Vendors are qualified through surveys, audits or previous history of performance. Successful vendors are listed on an Approved Vendors List. The inspector determined Rotork purchased components and calibration services from approved vendors. Rotork effectively implements a vendor rating system which controls the quality of components received. The system enables Rotork to take appropriate action when nonconforming components are identified during receipt inspections. The inspector reviewed the control of the procurement of electrical motors, auxiliary switches, wiring harnesses, and worm and wheel gears and identified no unacceptable findings.

5. Review of Identification and Control of Materials and components:

All components used in actuators intended for nuclear power plants are identified with a prefix N and stored in an enclosure, the access to which is controlled by quality control. The enclosure is partitioned into bins of appropriate size to accommodate components. Each bin is adequately identified with the component number and contained the necessary documentation to identify the origin of supply. Retrieval of components for use is controlled by computer printout sheets.

The components are used to assemble modules such as switch mechanisms, synchropacks, add-on-packs and motors. Each module is tested for correct operation on specially designed test fixtures after assembly and placed in the nuclear storage area. A CoC is issued for each module listing the components used in the assembly. When an operator is assembled, the individual CoCs for the modules are attached to a single CoC. This CoC certifies that the individual modules were assembled and that the performance of the actuator meets the relevant specification. If a module is subsequently replaced, the current CoC for the module is attached to update the CoC of the actuator.

The inspector witnessed the assembly and testing of selected modules, interviewed the craft performing the work and determined that the individuals were knowledgeable in the work they performed.

6. Review of Inspections:

The inspector reviewed the implementation of the following inspections:

a. Incoming Receipt Inspections:

Quality Control Inspectors performed inspections on purchased material received. Inspection checklists were used to document the results of inspections. Unacceptable material was identified, segregated and a nonconformance report was generated.

The gears received were inspected for surface finish, surface hardness, gear form, backlash, tooth to tooth accumalative errors and bearing pattern. The inspector verified the calibration of three hardness testing machines and determined that the calibrations were current and that the calibration certifications were acceptable.

Rotork supplied selected suppliers with inspection tools to enable them to inspect the material prior to shipment. Rotork reinspects the material during receipt inspections.

b. In-process Inspections:

In-process inspections are conducted on individual components prior to assembly into the respective modules. As indicated in paragraph 3 of this report, the completed modules are tested

after assembly utilizing specially designed fixtures to simulate the intended function.

c. Final Test:

Every actuator is subjected to routine electrical and mechanical tests. The results are documented on a test certificate. Unacceptable performances are documented in a deficiency report which is reviewed by quality control and engineering for suitable disposition.

7. Review of Corrective Action:

The inspector reviewed the implementation of corrective action. Conditions adverse to quality identified in the factory are documented in deficiency reports which require review and disposition by engineering and quality control. Conditions adverse to quality identified after the operator is delivered are identified in Service Engineer's Reports on Site Problems Defect Feedback Reports. These reports are reviewed by quality control to determine the nature of the problem and whether the problem meets the reportability criteria of 10 CFR Part 21. The inspector reviewed twenty-five Service Engineers Reports and determined that the identified problems did not reflect inadequate quality control inspections. Acceptable corrective action was taken on identified problems.

E. EXIT INTERVIE:

The inspector met with Rotork representatives mentioned in Section F and discussed the scope of the inspection and findings.

F. PERSONS CONTACTED:

*T. Eassie, Chief Executive
*G. Hawkins, Quality Assurance Manager
*G. Malcolm, Engineering Director
*N. Bellworthy, Senior Quality Control Engineer
J. Granger, Quality Control Engineer

*Denotes those individuals who were present at the exit interview at the conclusion of the inspection.

ORGANIZATION: STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASSACHUSETTS

REPORT NO.: 99900509/86-01	INSPECTION DATE: 8/25-29/86	INSPECTION ON-SITE HOURS: 180

CORRESPONDENCE ADDRESS: Stone & Webster Engineering Corporation
ATTN: Mr. R. B. Kelly, Vice President
Quality Assurance
Post Office Box 2325
Boston, Massachusetts 02107

ORGANIZATIONAL CONTACT: Mr. F. B. Baldwin, Assistant QA Manager
TELEPHONE NUMBER: (617) 589-6566

NUCLEAR INDUSTRY ACTIVITY: Major active projects include Beaver Valley Unit 2, River Bend Unit 1, Shoreham, Nine Mile Point Unit 2, Millstone Unit 3, and Comanche Peak Units 1 and 2 (piping & pipe support requalification).

ASSIGNED INSPECTOR: R. P. McIntyre, Special Projects Inspection Section (SPIS) 12/24/86 Date

OTHER INSPECTOR(S): P. L. Pettis, SPIS; K. C. Leu, SPIS; P. J. Prescott, SPIS; K. R. Bockman, Consultant

APPROVED BY: John W. Craig, Chief, SPIS, Vendor Program Branch 12-24-86 Date

INSPECTION BASES AND SCOPE:

A. BASES: Stone & Webster Engineering Corporation (SWEC) Topical Report No. SWSQAP 1-74A and 10 CFR Part 50, Appendix B.

B. SCOPE: The purpose of this inspection was to review the implementation of Stone & Webster's QA program concerning activities relating to the Design Requalification of piping and pipe supports on Comanche Peak Units 1 and 2. Specifically, design control, procurement document control, indoctrination and training, and audits were reviewed.

PLANT SITE APPLICABILITY: Comanche Peak 1 & 2 (50-445/446).

BACKGROUND

General

On March 12, 1984, the U.S. Nuclear Regulatory Commission's (NRC) Executive Director for Operations established a program to ensure that technical concerns and allegations related to Comanche Peak Steam Electric Station (CPSES) design and construction activities would be addressed in a coordinated and integrated manner by the NRC staff. As a means of executing its assigned responsibilities, the NRC formed a Technical Review Team (TRT).

The TRT initiated a series of onsite inspections and evaluations in July 1984, and documented the results of these activities in five supplements to the Comanche Peak Safety Evaluation Report (NUREG-0797, "Safety Evaluation Report Related to the Operation of the CPSES, Units 1 and 2," Supplements 7-11).

In response to the concerns and issues identified by the TRT, Texas Utilities Electric Company (TUEC) formed the Comanche Peak Response Team (CPRT) for the purpose of addressing various construction and design issues. The CPRT has been charged with responding to and resolving the issues raised by TRT concerns as well as concerns from sources external to TUEC. A CPRT action plan describing reanalysis on a sampling basis as well as a complete reanalysis of affected designs are included as a portion of the (piping supports) Discipline Specific Action Plan (DSAP). This action plan initiative was developed to resolve the large number of External Source Issues (ESI) associated with these design areas and have resulted in a significant level of design reanalysis. TUEC has retained contractors not involved with the original design to perform the requalification program for piping and pipe supports (Stone & Webster) and cable tray/conduit supports (Ebasco Services, Inc. and Impell).

Piping and Pipe Supports

A number of issues have been raised in the area of piping analysis and pipe support design. As a result, TUEC has initiated a piping and pipe support requalification program which has resulted in a significant level of reanalysis and reevaluation of the CPSES piping and pipe support designs. SWEC, under contract to TUEC, will perform this requalification program under the direction of the CPSES project personnel. In addition, the TERA Corporation (TERA), also under contract to TUEC, will review this effort to provide assurance that the objectives of the Design Adequacy Program in the piping and pipe support area are being achieved. The scope of the SWEC program for the requalification of piping and pipe supports, as described in Supplement 13 to the Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2 (NUREG-0797) includes:

- 100 percent of all ASME Code Class 2 and 3 piping larger than 2 inches (large bore), excluding portions within the boundary of Westinghouse Class 1 auxiliary branch line stress problems
- 100 percent of all ASME Code Class 1, 2, and 3 large bore pipe supports
- small bore piping and pipe supports on a sampling basis (ASME Class 2 and 3)
- all non-ASME piping and pipe supports within ASME Code Class 2 and 3 large bore stress analysis problem boundary
- all non-ASME supports within the ASME Code Class 1 stress problems
- all ASME Class 2 and 3 small bore piping subjected to significant fluid transients (stress and supports)
- all high energy ASME Class 2 and 3 small bore piping (stress and supports)
- hot functional and vibration testing procedures and technical assistance
- site construction support activities related to piping and pipe supports

The SWEC action plan consists of the following elements:

(1) development of CPSES pipe stress and pipe support design criteria

(2) verification of existing as-built information, performed a walkdown of piping for:

- support location
- valve location
- support function
- valve and support orientation

(3) review and verification of system design input, seismic acceleration, and fluid transients

(4) verification of existing pipe support design documents

(5) resolution of special technical concerns

(6) reanalysis of piping systems and evaluation of pipe support designs

(7) review all existing reports, etc. to identify and develop SWEC resolution to all previous identified technical issues, questions, concerns (Generic Technical Issues Report)

(8) identify pipe break/crack postulation requirements.

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

1. Contrary to 10 CFR Part 50, Appendix B, Criterion V and the SWEC Standard Nuclear QA program manual, Section 5, paragraph 2.1.1, SWEC had not written and implemented instructions and procedures for the transfer of design information (from the SWEC New York office to the Comanche Peak site and between SWEC offices), utilized in the final resolution of safety related pipe support design packages. (86-01-01)

2. Contrary to 10 CFR Part 50, Appendix B, Criterion V and the SWEC Standard Nuclear QA program manual, Section 5, paragraph 2.1.1, SWEC had not written and implemented instructions and procedures which describe how proposed design modifications are documented and reviewed by the applicable Project Engineers to determine if they require review by the Option Review Committee. (86-01-02)

C. UNRESOLVED ITEMS:

1. The results of SWEC's evaluation of computer code error reports submitted to them by Power Computing Company were not reviewed during this inspection. These error notices were for the computer code P-DELTA STRUDL. These evaluations will be reviewed during a future inspection at the SWEC Boston office. (86-01-03)

2. The NRC inspectors were unable to review the results of SWEC's evaluation of computer code error reports submitted to them by the Impell Corporation. Error reports were unique to the QUICKPIPE system and used on the River Bend Project. This evaluation will be reviewed during a future inspection at the SWEC Boston office. (86-01-04)

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not reviewed during this inspection.

E. OTHER FINDINGS OR COMMENTS:

1. Pipe Stress Requalification

During this inspection, 16 packages of ASME Class 2 and 3 pipe stress design calculations were selected for detailed review. These packages represented a spectrum of sample packages from the following systems at Comanche Peak.

- Containment Spray System
- Chemical and Volume Control System
- Component Cooling System
- Station Service Water System
- Safety Injection System
- Demineralized Water System
- Auxiliary Feedwater System
- Diesel Generator Fuel Oil System
- Chilled Water System
- Water Processing Liquid System
- Vent and Drain Systems
- Compressed Air (Instrumentation) System

The review included verification of compliance to "Comanche Peak Project Procedures" (CPPP) 6, 7 and 9, "Comanche Peak Project Memoranda" (CPPM) and other applicable codes and criteria. Also reviewed were pipe stress dynamic analysis modeling, NUPIPE-SW program input and output, fluid transient loads, and load combinations for various plant operating conditions. The associated allowable stress limits were reviewed for compliance with design bases and criteria.

Special consideration, such as mass point spacing, valve modeling, stress intensification factor (SIF), pipe thinning and weld shrinkage were examined among other items for technical completeness and performance. In addition, field correspondence and drawings, interfacing documents between stress, support and power groups were reviewed with respect to document control compliance.

Pipe Stress Analysis Review

The NUPIPE-SW computer program was utilized by SWEC to perform linear elastic analysis of three dimensional piping systems subjected to thermal static and dynamic loads. The use of a cutoff frequency at 50 Hz was included in the program so that the contribution from higher modes was reflected in the piping analysis.

The inspector selected 16 stress packages from 12 systems for review and assessment of technical adequacy.

Each package contained a "Detailed Pipe Stress Analysis Checklist" (Attachment 9-9, CPPP-6, Revision 2) which referenced general items such as mass point spacing and lumping, SIF, vent/drain connections, component configuration, etc. for inclusion in the analysis.

The general package format and contents were in conformance to form "Pipe Stress Analysis Calculation Contents" (Attachment 9-11, CPPP-6, Revision 2) with regard to objectives, assumptions, loads, conclusions, and summary of results, etc. The methodologies and calculations were found to be satisfactory.

Three design packages reviewed are discussed below:

(1) Calculation No. DO-1-165C: Analysis of buried pipe was in accordance with "Procedure for ASME Section III Class 2 and 3 Buried Piping" in Attachment 3-9 of CPPP-7, Rev. 2, where closely spaced lateral restraints were provided to simulate the stiffness of soil restraining the pipe and coded accordingly in the computer input.

(2) Calculation No. CC-1-061A: This calculation reanalyzed existing trapeze supports and snubbers and determined that some were either unnecessary or unstable. Corrections were made to install stable supports and maintain their intended functions. The supports were located at various nodal points.

In addition, the consideration of "break and crack exclusion," and "pipe wall thinning evaluation" were reviewed by the inspector and found to be in conformance to requirements of Section 3.8 and Attachment 3-14 of CPPP-7.

(3) Calculation No. SI-1-071B: The design information contained in this calculation was reviewed by the NRC inspector to verify the following items had been considered and included in the reanalysis:

- Detailed computer input coding for trurnion and drain/vent
- Fluid transient consideration
- Code case N-411 application

Based on the review of stress analysis packages, and the interviews with the SWEC engineering staff, the inspection concluded that the design criteria, analytical methodologies and procedures for CPSES Unit 1 pipe stress analyses met the requirements of the Comanche Peak Project Procedures and Project Memoranda.

However, the inspector identified two items as observations:

(1) In reviewing the interface control between pipe stress and pipe support groups, the inspector found that 11 pipe stress calculations contained a memorandum transmitting the pipe loads to the support group for design use, while five calculations did not include the memorandum. Although this memorandum is not required by SWEC procedures, the five calculations did make it to the support group, and the missing memorandums were produced from other files later on. It appears that better document control is needed in order to provide consistency and quality in documents.

(2) In verifying the Options Review Committee's (ORC) review process at SWEC, the inspector found a memorandum (page 2 of Attachment #9, Cal. No. WP-1-N047) from the responsible engineer to the project engineer's office in reference to a Westinghouse valve overstress. The inspector was informed that the job was still ongoing.

Since the memorandums were not dated when issued and received, the inspector could not determine when the action was initiated or how long it had been an open item. During discussions with SWEC project engineer and QA, the inspector was told that SWEC procedures do not require the memorandum to be dated. This is a poor practice and does not constitute standard QA document control.

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

2. Pipe Support Requalification

a. Review of Pipe Support Design Criteria and Procedures

SWEC CPPP-6, 7, 9, and 11 were reviewed as well as the modifications/additions contained in the active Project Memoranda. Procedure CPPP-7, "Design Criteria for Pipe Stress and Pipe Support," contains the reanalysis project design criteria and analysis methods for plant piping supports. Section 5 indicates that the finite element computer program ANSYS is used for the analysis of pipe support base plates in conjunction with the BAP and other SWEC computer programs. SWEC stated that ANSYS, which presently is not listed as an acceptable computer program for use on Comanche Peak, would be identified on the list of acceptable computer programs used in the Comanche Peak re-evaluation program.

b. Review of Pipe Support Design Calculations

Pipe support calculations were reviewed for problems designated 1.15 B (1-1/2" safety injection system piping) and 1.38 B (3" and smaller reactor containment system piping.) These are safety related ASME Code Class 1 lines which attach to the reactor coolant system piping. These support design calculations had been completed and reviewed by SWEC. However, other design problems were still open and the calculations had not been certified by a Professional Engineer (PE), as required for Class I supports.

Except for snubbers, the supports reviewed were welded frames of tubular steel. A typical support would consist of a box or open frame surrounding or partially surrounding the pipe. This would

be supported by a cantilever beam attached to a baseplate. The calculations for the support were largely computerized. Structural analysis including ASME code stress evaluation was done with the STRUDL computer program, required fillet weld dimensions were calculated using the SANDUL computer program and the adequacy of baseplate and anchor bolts was evaluated using the BAP computer program. Hand calculations were required for conditions such as the sizing of the welds which were not fillet welds.

Considerable effort was expended in relating analytical models and calculation results to the support drawings. STRUDL computer program input and output listings for structural analysis and SANDUL computer program output for weld design were checked. The sizing of anchor bolts and baseplates using the computer program BAP were also reviewed. Stiffness calculations were reviewed, including one for support No. SI-1-199-711-C41K. The structure, weld, and baseplate calculation report was reviewed in detail for support No. RC-1-072-C41R. Complete listings of the computer program input were included in the calculations, as well as specific portions of computer output. A calculation checklist was included with each support calculation. The calculations met the requirements of the applicable SWEC procedures.

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

3. Audits

a. External to SWEC

The NRC inspector reviewed seven (7) Comanche Peak audit reports performed by Texas Utilities Generating Company (TUGCO) on the following SWEC offices:

Audit No.	Office	Dates
TSWEC-1	site	10/21-25, 1985
TSWEC-2	New York	11/4-6, 1985
TSWEC-3	Boston	12/3-5, 1985
TSWEC-4	Cherry Hill	1/21-24, 1986
TSWEC-5	site	4/14-18, 1986
TSWEC-6	Toronto	4/23-24, 1986
TSWEC-7	Houston	5/15-16, 1986

The purpose of these TUGCO audits was to assess the adequacy of SWEC's quality program for Pipe Stress and Pipe Support Qualification and Reverification of design analysis functions being performed for CPSES, Unit 2 activities. Audit scopes included verifications of the SWEC quality program for compliance to CPSES requirements and adherence to TUGCO procurement documents, with emphasis placed on program and technical assessment.

SWEC's response and corrective actions, in most cases, adequately addressed the issues raised by TUGCO. The majority of the findings and observations centered around calculation inconsistencies, computer modeling errors, evaluation of computer error notices for NUPIPE and BASEPLATE II, unconservative weld calculations performed by computer program SANDUL not matching hand calculations performed using formulas in CPPP-7, inappropriate and unconservative use of stress intensification factors (SIF) used in piping stress analysis, valve modeling inconsistencies, mass point spacing used in dynamic piping analysis, and incorrect loading assumptions made in conjunction with integral attachment analyses.

SWEC's responses to these items were adequate and resulted in documented evidence of increased technical training in those areas in which weaknesses were identified by TUGCO.

b. Internal to SWEC

The inspector also reviewed 12 internal audits performed by SWEC's Engineering Assurance Audit Division (EA). These audits are identified and governed by SWEC Engineering Assurance Procedure (EAP) 18.1, Revision 5, dated April 7, 1982, and cover audit activities of SWEC Engineering Department activities in Boston and New York. The audits reviewed were as follows:

Location	Dates
site	7/31-8/1, 1985
New York	10/6-11, 1985
Cherry Hill	10/28-11/8, 1985
Houston	12/9-2/13, 1986
Toronto	12/17-19, 1985
Boston	12/16-2/28, 1986
site	12/16-2/13, 1986

Location	Dates
Boston	2/10-3/7, 1986
New York	3/24-28, 1986
Houston	4/28-5/2, 1986
site	5/19-23, 1986
Toronto	6/2-6, 1986

In general, audit findings and observations resembled that of the externally performed TUGCO audits in that calculation discrepancies and inconsistencies were identified.

SWEC responses to these items were timely and in compliance with EAP 18.1.

c. Engineering Services Supplier Audits

The NRC inspectors reviewed the audit/survey reports of two suppliers of computer services, Power Computing Company (PCC) formerly known as UCCEL, and the Atlanta office of Impell Corporation.

(1) PCC - Dallas, Texas

A review of this September 6, 1985 audit demonstrated several weaknesses in the area of verification/validation of the computer code P-Delta STRUDL.

This computer code is provided to SWEC by PCC through its software supplier P-Delta Inc., and is used primarily in the design verification of safety-related pipe supports and general structural applications.

The SWEC audit report identified examples of technical discrepancies noted during a review of test records for P-Delta STRUDL among which were results that differed in the range of 200-600 percent for support reaction forces used in transient analysis. PCC committed to complete its qualification testing and incorporate it into the 0985 release of P-Delta STRUDL.

The inspector also reviewed SWEC's receipt of error reports supplied to them by PCC, as required by contract. To date only two error report letters have been received by SWEC which contained five individual error notices as follows:

Date of Letter	Error Date	Error #	Error Desc.
5/12/86	5/10/80	B 3.0-402	"Torsion" parameter ignored during AISC code check
8/14/86	1/20/86	B 3.0-328	Section properties of Tubing and Angle shapes corrected. Slight differences observed in both Static and Dynamic analysis results.
5/15/86	N/A	B 3.0-503	Incorrect eigenvectors and participation factors for multiple dynamic analyses in same run.
8/8/86	N/A	B 3.0-533	Removed message from stiffness matrix command.

A review of SWEC's evaluation of these error reports was not performed since documentation is kept at SWEC's Boston offices. In addition, SWEC representatives stated that P-Delta STRUDL had not been used nor any other UCCEL/PCC programs on Comanche Peak. However, the TUGCO-PSE STRUDL group on site may have used P-Delta STRUDL for inputs to SWEC modifications (i.e., load distribution on Richmond Inserts). Due to the unavailability of the SWEC evaluation of these error reports in New York, Unresolved Item 86-01-03 was identified in this area of the inspection.

A related item concerns the manner in which error notices are being distributed to customers of PCC. During an earlier NRC inspection (conducted at the New York office of Ebasco Services August 11-15, 1986) several other error report notices received by Ebasco were identified. However none of these had been received by SWEC. Similarly, SWEC has received notices which Ebasco had not received for the same version of P-Delta STRUDL, version 0385.

(2) Impell Corporation (Atlanta office)

A May 3, 1984 SWEC EA audit of Impell was performed to evaluate Impell's compliance with the requirements of the QA provisions in SWEC Purchase Order No. E-CH-3245, and to evaluate the overall implementation of the Impell QA program.

Several findings identified that Impell was sending error notices for the computer program QUICKPIPE directly to User Manual holders and not to the SWEC Computer Library in Boston, as required in the SWEC purchase order. Impell's response, as part of the corrective action, was to transmit to the SWEC Computer Library, on May 15, 1984, all outstanding error notices for QUICKPIPE in addition to one copy of all previous updates and revisions used by SWEC for the River Bend Project. QUICKPIPE (similar to NUPIPE) had been used as an optimization program for 4" and smaller piping analysis on River Bend but was subsequently replaced by NUPIPE. The results of the evaluation performed by SWEC were not reviewed, Unresolved Item (86-01-04) was identified in this area of the inspection.

4. Design Interface Control

a. SWEC New York Communication with Comanche Peak

The NRC inspectors reviewed the process in which design information, utilized in the final resolution of a safety related pipe support design package, was being transferred from the SWEC New York office to the pipe support design group at the Comanche Peak site.

In an interview with SWEC personnel, it was stated, that when additional information is required from the site for the resolution of a final design package, the requests for the

design information are first transmitted through the SWEC New York office for engineering review and approval. The request is then sent from the New York office to the site using one of three forms, depending on the type of information requested, and the engineering group requesting the information (i.e., stress analysis group, pipe support design group). The NRC inspectors reviewed the three forms utilized by SWEC for the transfer process. The three forms reviewed were: pipe support modification form (PSM), pipe support drawing change notice (PSC) and stress analysis request for action form (SA). A review of the transfer process was performed by the NRC inspectors and there were no significant inadequacies found in the system.

When applicable instructions and procedures for this process were requested by the NRC inspectors, SWEC informed the inspectors that there were no procedures written for this process as part of the Comanche Peak effort. SWEC committed to develop instructions and/or procedures which would outline the transfer of design information, from the SWEC New York office to the Comanche Peak site, including the applicable forms utilized in the process.

Nonconformance 86-01-01 was identified as a result of this review.

b. SWEC Internal Communications

The NRC inspector examined the review, transfer and documentation process for safety related pipe support modifications for the Comanche Peak effort, within SWEC.

The document used by the various SWEC engineering departments for this process is the pipe stress and supports proposed modification report (PM). This document is used to communicate proposed modifications between the various engineering departments internally within the SWEC New York office and for external communications between the SWEC New York office and the other SWEC offices involved in the Comanche Peak effort; (Cherry Hill, Houston, Toronto, and Boston offices).

The PM can be initiated by a responsible engineer at any one of the SWEC offices involved in the Comanche Peak effort. The PM and all applicable information (e.g., excerpts from calculation(s), drawing(s)) is then forwarded to the applicable project engineer, who evaluates the proposed modification against the criteria for submittal to the Option Review Committee (ORC). The ORC is responsible for convening meetings at the request of the project engineer to review recommendations of significant hardware changes, assess the impact of those changes on other disciplines and recommend viable alternatives.

During the review of this process the NRC inspectors requested the instructions and procedures which reference the PM document or designates the individual responsible for performing the engineering reviews. SWEC stated that there had been no formal procedure written as part of the Comanche Peak effort. However, the NRC inspectors were informed that a draft project memorandum was in process, and would be issued as a supplement to the formal Comanche Peak procedures, once it had received final concurrence from the applicable SWEC engineering departments.

Nonconformance item 86-01-02 was identified as a result of these findings.

5. Procurement Control

The NRC inspectors reviewed two procurement packages for the contracting of engineering services for the Comanche Peak effort. EAP 4.1, "Procurement System," is the procedure which establishes the process used to procure power plant equipment, material and services including engineering services. EAP 4.15, "Preparation and Control of Engineering Service Scopes of Work (ESSOW)," describes the procedure for preparation, review, approval, change and control of ESSOW for procurement of engineering services.

An ESSOW is a procurement document used to describe, in detail, an engineering service to be performed by an engineering service supplier. This ESSOW accompanies the purchase order to the service supplier, and includes the overall scope of work as well as the applicable QA requirements per 10 CFR Part 50, Appendix B as well as the requirements of 10 CFR Part 21.

SWEC has issued only two (2) purchase orders to date for services on the Comanche Peak effort. The first purchase order was issued to Stone & Webster Canada Ltd. for engineering and support services for pipe support and pipe stress on Comanche Peak. The second purchase order was issued to UCCELL/Power Computing (PCC) for computer services, although no computer services have been used to date through PCC (refer to number 3.c of Section E).

The inspector reviewed the procurement packages including the purchase order and the ESSOW for both suppliers. The original purchase order with SWEC Canada Ltd. did not include an ESSOW or any provisions for QA requirements and compliance to the 10 CFR Part 21. Change 1 to the purchase order, written three months later, corrected this item and included the documentation which was originally missing. This deficiency was also identified by TUGCO in an audit performed in November 1985.

The ESSOW, included in Change 1 to the purchase order, referenced a procedure for calculation control which was not applicable to the Comanche Peak project, EMTP 8.26 was referenced instead of EAP 5.3. SWEC stated they were going to issue a revision to the ESSOW which corrected the error.

The purchase order and ESSOW to UCCEC/PCC includes the appropriate QA requirements as well as instructions for the receipt of computer program error reports. The procurement package met the requirements of the applicable procedures.

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

6. Indoctrination and Training

The NRC inspector reviewed EAP 2.4, "Indoctrination, Continuing Education and Certification Requirements." This procedure contains the requirements for indoctrination, and continuing education and training in the use of procedures and systems developed to govern and support nuclear projects such as Comanche Peak.

The inspector reviewed the Engineering Department Training Matrix for the Engineering Mechanics Division. This matrix lists the indoctrination and continuing education requirements for personnel performing quality related or quality assurance engineering functions.

SWEC has developed Comanche Peak Project Procedures (CPPP), which describe requirements of many activities performed as part of the SWEC Comanche Peak effort. CPPP-6, 7, 9 are the procedures for requalification activities for ASME III Class 2/3 Pipe Stress Analysis and Pipe Support Verification. This includes the analysis and Design Criteria. These CPPPs are used in conjunction with applicable EAP's, Engineering Department Policies and Procedures (EDPP) and Quality Standards (QS).

The inspector reviewed the training records of five engineers performing pipe stress analysis and five engineers performing pipe support requalification to verify they had received the required engineering division training as well as the required project specific training for the applicable CPPP's.

This was verified through the review of attendance forms and the Computerized Training Records System, which lists SWEC employees by office location and all formal QA, design, and project specific training they have received up to the computer printout date in the records systems. The inspector also reviewed the training outline for CPPP-6, 7, and 9. Training records for the ten employees reviewed indicated that they had received the training required for nuclear engineering activities and the Comanche Peak specific training.

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

7. Temperature Effects on ASME III NF Code Check Equations - USNRC Regulatory Guide 1.124

In the review of SWEC's pipe support design efforts, the NRC inspector reviewed paragraph 2.3 of Comanche Peak Procedure CPPP-7, Revision 2, dated April 25, 1986, "Applicable USNRC Regulatory Guides," and the CPSES/FSAR page 1A(N)-64, Item 2 which references the overall commitment to the use of this Regulatory Guide.

Regulatory Guide, Section B.5, page 1.124-3, cautions the designer to recognize that ASME III, Appendix XVII compression and compression due to bending equations were derived based upon mechanical property constants of steel, Young's Modulus (E) and Yield Strength (YS), at room temeprature (i.e., 70°F). Further, page 1/124-4 of Section C.3b

that materials with values of E at working temperatures substantially different from 29,000 ksi, these constants should be rederived with the appropriate value of E unless the conservatism of using these constants as specified can be demonstrated.

The NRC inspectors reviewed SWEC's primary computer code for pipe support design and structural analysis, STRUDL-SW, in an attempt to verify what treatment was afforded by SWEC to address these concerns expressed in Regulatory Guide 1.124.

Discussions with SWEC personnel in both the New York and Boston offices indicated no special treatment is given to rederive the code equation constants as written into the STRUDL-SW program source. However, to account for the overall temperature effect, it is SWEC's practice to adjust the input parameters of Sy and E accordingly where it explicitly occurs in the NF code check equations.

Additional information provided the NRC by SWEC after the inspection indicated that at a temperature of 300°F, which is slightly higher than the Comanche Peak containment LOCA temperature, the 1974 edition of ASME III lists an E value for SA-36 steel of 27,400 ksi as compared to 27,900 ksi at an ambient temperature of 70°F. This represents a total variation from the nominal value of 5.5% and only a 1.7% difference from ambient.

It was also noted by SWEC that more recent editions of ASME III provide higher values of E (29,500 ksi and 28,300 ksi at 70°F and 300°F, respectively), thereby reducing the previous 5.5% variation to that of less than 2.5%.

SWEC further concluded that they believe their current practice, as outlined, is technically adequate and no modifications are required to the code check equations, other than that described herein, due to the slight variation of E over the normal design temperature range of pipe supports included under the SWEC scope of work for the CPSES.

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

ORGANIZATION: TOBAR INC.
TEMPE, ARIZONA

REPORT NO.: 99900837/86-01	INSPECTION DATE: 11/3-5/86	INSPECTION ON-SITE HOURS: 40

CORRESPONDENCE ADDRESS: Tobar, Inc.
ATTN: T. R. Fisher
President
1441 West Alameda
Tempe, Arizona 85282

ORGANIZATIONAL CONTACT: J. H. Murphy
TELEPHONE NUMBER: (602) 968-3171

NUCLEAR INDUSTRY ACTIVITY: Manufactures pressure and differential pressure transmitters.

ASSIGNED INSPECTOR: K. Naidu 12/9/86
K. R. Naidu, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): R. Moist, Program Coordination Section

APPROVED BY: Ellis W. Merschoff 12/15/86
Ellis W. Merschoff, Chief, RIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review corrective action taken on nonconformances identified during a previous inspection, obtain additional information on setpoint drift problem identified at Seabrook, verify the implementation of selected portions of the quality assurance program.

PLANT SITE APPLICABILITY: Seabrook Station (50-443), South Texas Project (50-498).

ORGANIZATION: TOBAR INC.
TEMPE, ARIZONA

REPORT NO.: 99900837/86-01	INSPECTION RESULTS:	PAGE 2 of 10

A. VIOLATIONS:

No violations were identified during this inspection.

B. NONCONFORMANCES:

No nonconformances were identified during this inspection.

C. UNRESOLVED ITEMS:

No unresolved items were identified during this inspection.

D. ACTION TAKEN ON PREVIOUS INSPECTION FINDINGS:

The inspectors reviewed the corrective action on the nonconformances identified during an inspection conducted on April 9-13, 1984 and documented in Inspection Report 99900837/84-01. Tobar's corrective actions were outlined in their letters dated May 24 and June 7, 1984.

1. (Closed) Nonconformance Item A: The nonconformance identified that contrary to the Tobar Nuclear Quality Program Procedures (NQPP) manual PI-2, page 5-6, Revision 0, the form "Request for Engineering Action" (REA) was being used prior to the formal approval of the procedure. Tobar representatives stated that due to change in ownership of the company from Westinghouse to Tobar, they were rewriting the QA manual and procedures with the new owners name. During the transition, they were implementing the program and procedures established by Westinghouse. The inspectors determined that the REA was approved on July 11, 1984. Review of the REA forms initiated during July-August 1986 indicates that in each instance, the reason for action and the action taken were stated including the necessary signatures with dates.

2. (Closed) Nonconformance Item B: The nonconformance identified that contrary to the Tobar NQPP manual PI-2, page 18-7, Revision 0, the Annual Internal Audit Report (AIAR), dated October 1983, was not signed by the Product Integrity Manager (PIM) and did not identify the General Manager as recipient. As stated in Tobar letter dated June 7, 1984, the format has been revised and the AIAR is addressed to the General Manager and signed by the PIM. Review of the AIARs conducted in 1984 and 1985 indicate that the procedure was implemented.

3. Nonconformance Item C: The nonconformance identified that contrary to the NQPP manual PI-2, Section 18 titled Audits, page 31, Revision 1, the Tobar employee who performed the October 1983 internal audit of the areas of "Purchased Material Inspection" and "Products Integrity Program" was not trained in all five steps of training and was not certified. As stated in Tobar letter dated June 7, 1984, Tobar sent two employees in 1984 to GA Technologies in San Diego, California for training in audits. The inspectors reviewed the certificates for these two individuals and determined them acceptable.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Background Information

a. General

Tobar Incorporated (Tobar) was established in 1983 by a former employee of Westinghouse Electric Corporation (Westinghouse). Prior to 1983 Westinghouse manufactured a complete line of panel instruments at this location, including pressure transmitters which were known as Veritrak. The Veritrak line of transmitters was originally designed and manufactured by Motorola until 1971. Westinghouse bought and manufactured the Veritrak line in 1971 and in 1983 qualified the Veritrak transmitters to the requirements of IEEE 323-1974 and IEEE 344-1975. In 1983, Westinghouse transferred the line of panel instruments to a location in Orrville, Ohio, leaving the facilities to Tobar.

Tobar manufactures the same line of transmitters under the trade name Tobar model 32, which include gauge pressure transmitters, absolute pressure transmitters, and differential pressure transmitters for use in nuclear power plants. Tobar also manufactures a similar line of transmitters for non-nuclear applications and the model numbers are distinguished by the prefixes 75 or 56 (instead of 32 for nuclear).

b. Principle of Operation - Pressure Transmitters

The principle components in the Tobar Model 32PA type pressure and absolute pressure transmitters are the measuring head and the AC amplifier. The measuring head consists of a capsule assembly and process connection flange, both welded together as a single unit. The capsule assembly consists of a diaphram on which has been deposited a bridge network of strain sensitive

resistive elements. The diaphram is electron beam welded to the support and header assemblies. The header assembly contains hermetic feed through leads which carry the electrical signal from the diaphram to the amplifier.

The amplifier amplifies the measured signal received and furnishes a proportional 4-20mA output signal. An oscillator/AC amplifier/demodulator set-up is used to provide stable operation with minimal thermal drifts.

c. Principle of Operation - Differential Pressure Transmitters

The principle components in the Tobar model 32DP type pressure differential transmitters are the measuring head and the AC amplifier. The measuring head consists of a differential pressure capsule assembly (capsule) and two process connection flanges. The capsule consists of a flexure on which has been deposited a bridge network of strain sensitive elements. The flanges are bolted together with the capsule between them. A pressure applied to each flange is transferred as a force to the respective diaphram. The capsule converts the algebraic difference between the two pressures into an electrical signal and provides it to the amplifier.

The amplifier amplifies the received signal and furnishes a proportional 4-20 mA output signal. An oscillator/AC amplifier/demodulator set-up is used to provide stable operation with minimal thermal drifts.

2. Review of 50.55(e) Reports

a. South Texas Project 50.55(e)

On August 1, 1986, Houston Lighting & Power (HLP) the owner of South Texas Project Nuclear Power Plant reported that Veritrak pressure transmitters manufactured and supplied by Westinghouse exhibited excessive changes in their accuracy under varying ambient temperatures. In a subsequent letter to the NRC dated August 29, 1986, HLP stated that Westinghouse confirmed that Veritrak transmitters which provide input for various Reactor Protection System trip functions, ESF actuation, and post-

accident monitoring, are subject to excessive change in transmitter accuracy under varying ambient temperatures. Westinghouse provided the confirmation to HLP based on the results of tests performed on Veritrak transmitters which were supplied to Seabrook Nuclear Power Plant and subsequently returned to Tobar for additional tests.

b. Seabrook Nuclear Power Plant 50.55(e) Report

On March 7, 1986, Public Service of New Hampshire (PSNH), the owner of Seabrook Nuclear Power Plant (Seabrook), reported that during testing of pressure transmitters, Veritrak type 76PH2, calibration shifts due to ambient temperature changes in excess of the 1% per 100°F specified maximum were recorded. The transmitters were returned to Westinghouse to analyze the cause and to repair the transmitters.

On June 30, 1986, PSNH reported that they installed Tobar Model 32 type pressure transmitters to replace Veritrak transmitters after performing ambient temperature compensating tests on Tobar transmitters and determining that their ambient temperature compensation shifts were within Seabrook's allowable technical specification limits.

c. Summary of Discussions with Tobar Management

The inspectors discussed the performance of Veritrak pressure and differential pressure transmitters with Tobar management personnel and determined the following.

1) The Veritrak absolute pressure transmitters (APTs) and differential pressure transmitters (DPTs) installed at Seabrook and South Texas Project were manufactured by Westinghouse. During the change of ownership from Westinghouse to Tobar, Tobar completed the assembly of selected Veritrak transmitters which were in various stages of assembly.

2) Tobar performed temperature compensation tests (TCTs) on 5 APTs and 20 DPTs returned by Westinghouse. These were Veritrak transmitters which had been supplied to Seabrook. During these tests, Tobar determined that 3 APTs, with date code 8240 (manufactured during the 40th week of 1982), exhibited a large change in the temperature coefficient

between the time of shipment (as shipped) and the time of tests (as received). The temperature coefficient is the change in calibration setpoint with the change in ambient temperature. The date code refers to the manufacturing date of the sensing capsule. Tobar determined from the records that 3 other Veritrak APTs with date code 8240 had been shipped to Braidwood Nuclear Power Station. Tobar arranged for their return and performed similar tests to ascertain whether they also exhibited a large change in temperature coefficient between the as shipped and the as returned condition. The test results indicated that these APTs did exhibit a shift, but not as large as the ones returned from Seabrook.

3) Tobar performed TCTs on 2 PTs, 5 APTs and 8 DPTs manufactured by Tobar. The 5 APTs were manufactured during the 15th, 16th and 17th week of 1984. The accuracy ranged from -1/2% to +1/8%. The accuracy of all the PTs and DPTs were within the specified accuracy for operating temperatures (80°F to 130°F) and accident conditions (280°F). The accuracy was above the 6% margin at accident condition (320°F).

3. Tests Performed on Transmitters

Tobar performed TCTs on Veritrak transmitters returned to them by Westinghouse and on Tobar transmitters. After the unit was stabilized at the temperature, the reference accuracy was established for each transmitter at 80°F, by adjusting the outputs at 0, 50% and 100% of the input per range. The accuracies were verified subsequently at 130°F, 280°F, and 320°F. At each of these temperatures, the unit was soaked for at least 4 hours to attain thermal stabilization. The temperature cycle used to verify the accuracy for each unit was 80°F-130°F-80°F-130°F-80°F-280°F-320°F-80°F. The deviations from the reference accuracy were recorded. A summary of the results are as follows:

a. The deviations from the reference accuracy in a population of 25 Veritrak transmitters manufactured and stored over 1 year exhibited a standard deviation of approximately 5%.

b. The above test results were compared with the test results recorded during the environmental qualification tests performed in 1983 to qualify Veritrak transmitters to IEEE 323-1974. The final test results obtained after the Veritrak transmitters were artifically aged, irradiated, and subjected to loss of coolant accident (LOCA) also exhibited a 5% standard deviation. It must be noted that the 5% standard deviation is a statistically derived deviation and individual units may exhibit a larger deviation. For instance, the output of one of the units prepared for environmental qualification test exceeded the permissible 6% deviation at 320°F stabilized temperature. However, the outputs of all transmitters were within 6% deviation during the entire LOCA simulation.

c. Three of the Veritrak APTs returned by Westinghouse with manufacturing date code 8240 exhibited a large change in temperature coefficient between the time of shipment and the time of testing.

d. The deviation from the reference accuracy in a population of 13 Tobar transmitters manufactured and stored for more than one year exhibited a 1% standard deviation. At 280°F stabilized temperature, for all transmitters, the outputs were within the 6% deviation limits. At 320°F stabilized temperature, one unit exhibited a 6.5% deviation. The original test records indicate that the deviation was below 6%.

e. Recently manufactured Tobar transmitters exhibited a 1% standard deviation in steady state accuracy at 4 different ambient temperatures.

f. The inspectors observed that there is no correlation between the results obtained during stabilized condition at 320°F elevated temperature and the output results during a simulated LOCA. During LOCA, the 320°F elevated temperature is brief (only a few seconds) after which the temperature drops. Therefore, the output deviation beyond 6% observed during the 320°F steadystate temperature should not affect the performance of the transmitters under LOCA conditions. Furthermore, the calibration of the units in place are not verifiable at 320°F, because 320°F steadystate temperature cannot be generated and maintained inside the containment.

4. Procurement Document Control

The NRC inspectors verified the implementation of the established procurement document control by review of 5 purchase orders relating to transistors, resistors, capacitors, sensing elements, and printed circuit boards used in manufacturing Tobar transmitters. The NRC inspectors determined that Tobar purchased material from manufacturers listed on the Approved Vendor List, that applicable quality requirements including revision levels were referenced, and that the manager of product integrity signed and dated purchase orders after the review cycle.

5. Document Control

The NRC inspectors verified the implementation of document control by reviewing 5 drawings pertaining to transistors, capacitors, sensing elements, resistors, and printed circuit boards used in manufacturing Tobar transmitters. The inspector determined that the drawings were reviewed and approved; that drawing changes were approved and that the master drawing file including status records, revision notices, and the original drawing was being maintained in accordance with established procedures.

The NRC inspectors also reviewed the established procedures for preparing and controlling the Product Integrity Nuclear Quality Program Procedures Manual. The NRC inspectors determined that the manual was being distributed, controlled, reviewed, and approved in accordance with established procedures.

6. Control of Purchased Material, Equipment and Services

The NRC inspectors verified the implementation of the control of purchased material by reviewing records such as purchase orders, drawings, inspection plans, data cards, test reports and certifications relating to transistors, resistors, capacitors, sensing elements and printed circuit boards. The NRC inspector verified that the vendors were on the Approved Vendor List, that the inspection plan listed the attributes to be inspected such as visual, dimensional, testing as determined by the drawing, that the inspector annotated acceptance/rejection of the material on the inspection plan, that a sampling plan was used, the proper revision of the drawing was used, that material certification was on file, and that the test data was in the file when required.

The inspectors determined that the sensors are manufactured in West Germany. Tobar sends the parts for the diaphrams (for PTs) and flexures (DPTs) and the manufacturer returns the material with the deposited strain guage. Specifically, for a DPT, the metal volume filler and header made of stainless steel are sent to Germany. A molybdenium 138 flexure beam is welded to the header and the volume filler is welded to the header. Both welds are performed by electron beam. A printed circuit is applied to the volume filler, lead bonds are attached from the deposited flexure to the printed circuit and connections are made to the header pins. The manufacturer performs electrical tests to determine acceptability. Tobar verifies the operability during receipt inspections.

8. Nonconforming Materials, Parts, or Components

The NRC inspectors verified the implementation of nonconforming material by reviewing Discrepant Material Disposition Reports (DMDR) initiated at receiving inspection. Of the DMDRs reviewed, the NRC inspectors determined that proper disposition of the material was taken by Tobar, that the DMDRs were concurrently signed by the appropriate personnel, that appropriate corrective action was taken, and that control of nonconforming material was being implemented.

9. Inprocess Inspections

The inspectors observed work in progress and partially completed in-process travellers on which in-process inspection results are documented. Selected travellers were verified to have traceability to the partially assembled parts. The inspectors observed the following in-process inspections that were being performed on differential pressure capsules and that the Tobar QC inspector signed-off indicating that the workmanship was acceptable.

a. The formed diaphrams which enclose the capsule were visually inspected for deformities.

b. The wire leads soldered to the header pin and the flexure assembly were inspected for workmanship.

c. A leak check test was performed on the flange/diaphram assembly.

d. The serial number of the assembly was recorded.

e. The solder on the header wiring was visually inspected.

f. A leak check test was performed on the header/body tube and body tube/base.

g. The capsule assembly was to be electrically tested for zero balance and activity.

A traveller was available with similar inspection attributes for inprocess inspections for pressure capsules.

F. EXIT INTERVIEW:

The inspectors met with individuals identified in section G and discussed the scope and findings at the conclusion of the inspection.

G. PERSONS CONTACTED:

T. R. Fisher, President and General Manager
E. J. Miller, Vice President Engineering
J. H. Murphy, Product Integrity Assurance Manager

ORGANIZATION: TRANSAMERICA DELAVAL, INC.
ENGINE AND COMPRESSOR DIVISION
OAKLAND, CALIFORNIA

REPORT NO.: 99900334/86-01	INSPECTION DATE: 9/8-12/86	INSPECTION ON-SITE HOURS: 22

CORRESPONDENCE ADDRESS: Transamerica Delaval, Inc.
Engine and Compressor Division
ATTN: Mr. C. S. Mathews, Vice President
and General Manager
55 85th Avenue
Oakland, California 94261

ORGANIZATIONAL CONTACT: Bruce C. Guntrum, Manager QA
TELEPHONE NUMBER: (415) 577-7422

NUCLEAR INDUSTRY ACTIVITY: No diesel engines are currently being manufactured for the nuclear industry. Activity consists of replacement parts and service.

ASSIGNED INSPECTOR: _____ 12/3/86
E. H. Trottier, Reactive Inspection Section (RIS) Date

OTHER INSPECTOR(S): W. P. Haass, Program Coordination Section (PCS)

APPROVED BY: _____ 12/3/86
Ellis 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 performed to review the circumstances surrounding a recent notification that the material composition of two connecting rods in the San Onofre Nuclear Generating Station (SONGS) standby diesel generator did not conform to specifications. In addition, (continued on next page)

PLANT SITE APPLICABILITY: Grand Gulf 1/2 (50-416/417); San Onofre 1 (50-206); Rancho Seco (50-312); River Bend 1/2 (50-458/459); Shearon Harris 1/2 (50-400/401); Bellefonte 1/2 (50-438/439); WPPSS 1 (50-460); Comanche Peak 1/2 (50-445/446); Midland 1/2 (50-329/330).

B. SCOPE (continued):

this inspection sought to review progress and actions taken by TDI and Woodward Governor in response to the engine overspeed event that occurred at Grand Gulf Nuclear Station in November, 1985.

A. VIOLATIONS:

There were no violations identified as a result of this inspection.

B. NONCONFORMANCES:

There were no nonconformances identified as a result of this inspection.

C. UNRESOLVED ITEMS:

There were no unresolved items resulting from this inspection.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. Inspection 85-02 Unresolved Item)

The inspection was conducted to establish the circumstances and contribution TDI may have had in the engine overspeed event at Grand Gulf Nuclear Station in November, 1985. A contributing factor to the severity of the damage sustained by the overspeeding Grand Gulf engine was found to be the time delay of the mechanical overspeed trip system. This was identified as an unresolved item.

Subsequent to that event, TDI redesigned the trip system and made the modifications available to their customers. The redesigned mechanical overspeed system modifies both the hydraulic actuation circuit and its resulting output. Actuation time is reduced to less than one second, with the output going to the strangulation butterfly valve. The TDI Owners Group has been notified, both as a group and by individual letter, that the modified trip system is available for installation. This closes the identified unresolved item.

E. OTHER FINDINGS AND COMMENTS:

1. Woodward Governor Company

The inspectors reviewed technical manuals recently revised by Woodward Governor. Of the four manuals and specification documents reviewed, only one was found to need further revision for content and clarity. The Woodward staff made editorial changes to Woodward manual

#54066A, "EGB-10CEGB-13C, EGB-35C, and EGB-50C Governor/Actuator Installation." The suggested revision, presented as a note between procedural steps, follows:

> NOTE: If the booster, servo, and cooler require filling, the EGB will need approximately 2-½ quarts of oil. Air must be bled from the booster and cooler oil lines prior to starting the engine. Continue to refill the EGB until all air has been bled from the booster and cooler lines. See Manual numbers 36684 and 36641.
>
> After filling the EGB with oil, actuate the start sequence a few times to further bleed the system. Do not allow the engine to start. Repeat until oil remains at the midpoint of the oil sight gage.

The revised manuals, when returned from printing, will be available to customers who request them. (NRC Information Notice 86-07, dated February 3, 1986, informs all licensees that revised Woodward manuals are available and provides the information necessary to obtain them.)

Woodward Governor has established a computer assisted retrieval (CAR) system for maintaining records on their customers. While the system permits entry of technical information and distribution data, it does not at this time provide identification of customers by governor model purchased. The system is therefore not suitable for targeting Technical Bulletins or revised manuals to licensees having a particular governor. Woodward Governor maintains, and with some justification, that technical information on components of the diesel generator should be routed through the diesel generator manufacturer. Toward that end, Woodward Governor and Transamerica Delaval have established formal links for the exchange of updated technical information. This, combined with the above referenced information notice, and Woodward's long standing practice of providing manuals on request and with governors returned after service, should provide adequate assurance that licensees have sufficient avenues to obtain current technical information on Woodward Governors.

Woodward has identified one governor problem that has arisen in a nuclear plant since the Grand Gulf event of November, 1985. The problem involved a loose friction clutch in a hydro-mechanical speed setting governor motor on a diesel engine at the St. Lucie Nuclear Station. This speed control system is unique to St. Lucie and the event was reported in a safety evaluation report (SER 22-86) published by the Institute for Nuclear Power Operations (INPO) on June 19, 1986.

2. Transamerica Delaval, Inc.

a. Distribution of Updated Manuals

Subsequent to the Grand Gulf diesel overspeed event of November, 1985, Transamerica Delaval requested that Woodward Governor provide a list of the current revision for each manual applicable to every Woodward component previously purchased by TDI for use on their diesel engines at nuclear power plants. TDI has forwarded this information to the appropriate utilities with the advice that copies of the listed manuals applicable to their equipment can be obtained from Woodward Governor. In addition, TDI has revised its Engine Instruction Manual to include instructions for venting air from a governor system being refilled with oil. The inspectors reviewed the revised instructions and found them acceptable. The revised TDI manual has been sent to each licensee owning a TDI engine.

As a result of this inspection, it was concluded that TDI has performed adequate corrective and preventive actions in this matter by revising the portion of their engine manual dealing with governor maintenance.

b. Apparent Nonconforming Material Content of Connecting Rods

By letter dated May 19, 1986, Southern California Edison (SCE) reported that two connecting rods in a standby diesel generator were made of apparently nonconforming material. The material discrepancy was discovered as the utility sought to comply with the recommendations of the TDI owners group contained in the San Onofre Unit 1 Design Review and Quality Revalidation (DR/QR) report of February, 1985. One of the recommendations of the DR/QR report was to ensure that all connecting rods (master and link) were made to ASTM A237F, 668L or AISI 4142 material

specification. Two round link rods were found to be made from ASTM 235E, 668D or AISI 1042 steel, yet were stamped with a number indicating the material was made to AISI 4142 material specification. These connecting rods were replaced with connecting rods having the proper material content.

In reviewing the circumstances surrounding the two mismarked connecting rods, the inspectors noted that TDI changed the material specification for their connecting rods between March and June of 1974. The specification change accomplished several objectives: It permitted a change in connecting rod shape (from round to "I," with an attendant reduction in machining cost); improved the strength of round connecting rods for a given engine rating; and gave a common material specification to TDI suppliers of connecting rod forgings. However, round connecting rods of two different material specifications therefore were available after the specification change and before stock of the "old" specification connecting rods was used up.

TDI received connecting rod forgings from Canton Drop Forging and Manufacturing Co. in June, 1974 (shipping notice 43879 of June 3). The forgings were accompanied by material certification indicating steel of AISI 1042 specification, and were machined into round connecting rods for "vee" or in-line type TDI engines. In September, 1975 TDI received another shipment of connecting rod forgings from Canton Drop Forging and Manufacturing Co. (shipping notice 48450 dated September 12). These forgings used the new AISI 4142 specification steel and were intended for newer "vee" type TDI engines. As the connecting rod forgings were machined at TDI and stamped with identifying numbers traceable to the "heat" of material from which they were produced, two forgings already in stock and having the older AISI 1042 steel were stamped with numbers traceable to the newer AISI 4142 steel. These identifying numbers, 11441 and 11451, were found on the two subject connecting rods in the SONGS-1 diesel engine.

It should also be noted that while these two connecting rods were mismarked as to their material content, they fully satisfied strength requirements for application in TDI "vee" engines at the time the SONGS-1 engine was constructed (1974). The specification change from AISI 1042 to AISI 4142 steel occurred between March and June 1974.

Additional corrective action performed by TDI consisted of research into the history and current location of all round connecting rods in nuclear service. All "vee" engines were found to contain "I" shaped rods, with AISI 4142 material, except for SONGS-1. As examination by SCE showed, 8 of 10 SONGS-1 connecting rod assemblies contained AISI 4142 steel. The "I" rods that replaced the two subject round master rods contain AISI 4142 material. No other TDI "vee" engines contain round connecting rods of either steel specification. Actions taken to prevent recurrence of this incident included improved tracking of heat and serial numbers by entering both numbers on material travelers (route sheets). Operators performing the final machining operation will then transfer the appropriate number to the connecting rod. The route sheet form has been changed to require this method of tracing connecting rods. The method also is currently used by TDI in the production of combustion cylinder heads.

A complete review of all connecting rod material from all heats supplied to nuclear customers was performed by TDI. Of a total of 683 connecting rods, records showed the following four readings out of specification: Two Brinnel hardness test results, one sulfur content and one elongation. Specifically, the elongation specification requires 20%, but the subject sample yielded 18%. The customer accepted the material with this test result. The maximum sulfur content specification is 0.025 percent, with one connecting rod yielding 0.026 percent. The difference was determined to be insignificant. Two connecting rods had Brinnel hardness tests that were out of specification. Upon review, TDI engineering determined that with all other material tests in the acceptable range, the importance of Brinnel hardness diminishes to the role of a confirmatory test or inspection method. It was decided that in this case, the Brinnel hardness results were too far away from the expected value to be considered anything but erroneous.

The formal engineering analysis of these four connecting rods was performed on March 31, 1986. The engineering analysis report was reviewed by the inspectors.

c. Lube Oil Sump Foot Valve Liner Material (Clearflow Valves, Inc.)

On March 10, 1986, TDI notified the NRC that a potential defect existed in the liner (seal) material used in lube oil sump foot type check valves. (The number of affected plants was reduced to five in the TDI letter to the NRC dated May 8, 1986.)

The problem was brought to TDI's attention by a utility that noted reduced lube oil pressure during certain conditions. The problem was investigated and found to be caused by incompatibility between the foot valve liner material (Neoprene) and hot (190 to 200°F) lube oil. The cognizant TDI Test Engineer discovered that when Neoprene rubber was exposed to such an environment and pressurized - as is the case in a TDI engine lube oil sump - the Neoprene would extrude itself between the valve seat and the cast iron follower that is designed to provide rigidity. The result is a hot lube oil bypass path that effectively short circuits the lube oil cooler and engine supply header. Testing witnessed by the inspectors showed that Viton maintained its resiliency much better under such operating conditions. Evidence of improved performance of Viton over Neoprene for such an application was given by Durometer gauge readings of material resiliency. The resiliency of Neoprene decreased 21% after 24 hours of testing (lube oil at 190 to 200°F and 100 psig), while Viton resiliency decreased 2.6%. The TDI test was conducted over two weeks (simulating 1000 engine operating hours), with results and recommendations issued to all TDI nuclear customers and the NRC on November 13, 1986.

d. Lube Oil System Check Valve Liner Seat Material (Centerline Valves, Inc.)

On September 17, 1985, TDI notified the NRC that valve liner material used on the seating surfaces of some 2 and 4-inch lube oil system check valves was found to be deteriorating. Review by TDI found that the liner material, ethylene propylene (EPDM) rubber is not compatible with petroleum-based (natural) lube oils. All eight check valves (four at Rancho Seco and four at River Bend) were replaced by TDI.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR FUEL DIVISION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99900005/86-02	INSPECTION DATES: 10/06-09/86	INSPECTION ON-SITE HOURS: 75

CORRESPONDENCE ADDRESS: Westinghouse Electric Corporation
Nuclear Fuel Division
ATTN: Dr. Richard Slember
General Manager
Post Office Box 355
Pittsburgh, Pennsylvania 15230
ORGANIZATIONAL CONTACT: Mr. R. Cost, Manager of Quality Assurance
TELEPHONE NUMBER: (412) 374-2359

NUCLEAR INDUSTRY ACTIVITY: Nuclear fuel assembly supplier for Westinghouse (W), General Electric, and Combustion Engineering designed reactors.

ASSIGNED INSPECTOR: R. L. Cilimberg, Special Projects Inspection Section (SPIS) — Date 11/14/86

OTHER INSPECTOR(S): C. M. Abbate, SPIS
K. C. Leu, SPIS
D. J. Lynn, Consultant

APPROVED BY: John W. Craig, Chief, SPIS, Vendor Program Branch — Date 11/14/86

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review fabrication and testing activities, follow-up on previous inspection findings and fuel problems reported at Vogtle 1, Braidwood 1, and McGuire 1.

PLANT SITE APPLICABILITY: PWR facilities with fuel supplied by Westinghouse.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR FUEL DIVISION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99900005/86-02	INSPECTION RESULTS:	PAGE 2 of 13

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

1. Contrary to Section 17.1.5 "Instructions, Procedures, and Drawings," of the Westinghouse Electric Corporation Water Reactor Divisions Quality Assurance Plan, WCAP 8370/7800, Revision 10A/6A, dated August, 1984, and Section 7.1 of Mechanical Operating Procedure (MOP)-730502, Revision 13, dated August 21, 1986, the wash/rinse tank temperatures were observed to be lower than those required by the procedure. (86-02-01)

2. Contrary to Section 17.1.5 of WCAP 8370/7800, Revision 10A/6A, dated August, 1984 and Section 7.0 of Chemical Operating Procedure (COP)-814741, Revision 4, dated March 21, 1986, the operator set the revolution counter at 10 instead of 160 as required by the procedure for blending a bulk blending container of uranium dioxide. (86-02-02)

3. Contrary to Section 17.1.5 of WCAP 8370/7800, Revision 10A/6A, dated August, 1984, and Sections 4.2 and 7.1 of COP-814742, Revision 5, dated August 26, 1986, the sample hood ventilation was not being monitored weekly as required by the procedure. (86-02-03)

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Closed) Nonconformance (85-01-01)

Contrary to Section 17.1.5, "Instructions, Procedures, and Drawings," of WCAP 8370, Revision 10A and WCAP 7800, Revision 6A, dated August, 1984, and Quality Control Instruction (QCI) No. 108857, Revision 6, dated October 22, 1984, an autoclave was left uncovered when not in use.

All certified technicians have been reinstructed on procedural requirements for covering autoclaves when not in use. The NRC inspectors determined during this inspection that one autoclave was not in use and had been covered in accordance with QCI No. 108857, Revision 6. This item is considered closed.

2. (Closed) Nonconformance (85-01-02)

Contrary to Section 17.1.5 of WCAP 8370, Revision 10A and WCAP 7800, Revision 6A, and QCI No. 108823, Revision 0, rod pressurization was determined according to SOI R-0618 and not SOI R-0563 as specified by the QCI.

The NRC inspectors determined during this inspection that QCI 108823 was deleted on January 10, 1986. This item is considered closed.

3. (Closed) Nonconformance (85-01-03)

Contrary to Section 17.1.5 of WCAP 8370, Revision 10A and WCAP 7800, Revision 6A, and Product Assurance Procedure (PAP) 2.2.2, Revision 4, dated December 2, 1982, management approval sign-off for qualification of a metallurgical laboratory technician was missing from the evaluation sheet.

The qualification record was corrected on November 14, 1985, by the responsible product assurance engineer. The NRC inspectors determined during this inspection that the evaluation sheets for qualification of metallurgical laboratory technicians contain the required management approval signatures. This item is considered closed.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Entrance and Exit Meetings

Westinghouse management representatives were informed of the scope of the inspection during the entrance meeting. The inspection findings and observations were summarized during the exit meeting on October 9, 1986.

2. Pellet Fabrication

a. Transferring Material From Polypaks to a Bulk Container

(1) Procedure Review

The inspectors reviewed COP-814751, "Transferring Material from Polypaks to a Bulk Container," Revision 7, dated March 21, 1986, related to powder transferring from polypaks into a bulk blend container.

The procedure contains requirements for moisture content, oxygen to uranium ratio limits for powder to be processed, ventilation requirements, and the steps used for the material transfer operation. The procedure was complete and had been properly reviewed and contained the required signature approval.

(2) Observation of Operation

The inspectors observed the production of uranium dioxide powder for Contract TGAF-01, Blend Number 4099-01. One hundred polypaks of powder were transferred to a bulk container in accordance with Section 7.0 of COP-814751. Powder moisture content, oxygen to uranium ratio, and ventilation requirements had been determined to be within allowable limits and were signed off, as required, by engineers. The steps of the procedure observed by the inspector were properly followed.

Step 9 of Section 7 of COP-814751 requires two people to verify the visual inspection of a bulk container and sign form CSR-051, "Bulk Blend Status." The NRC inspectors found only one signature on the form and informed the supervisor about the missing signature. The form was properly completed. This observation was discussed during the exit meeting.

b. Blending Powder

(1) Procedure Review

The NRC inspectors reviewed COP-814741, "Blending Powder," Revision 4, dated March 21, 1986, for bulk blending a container of uranium dioxide. The procedure was found to be technically correct and complete.

(2) Observation of Operations

The NRC inspectors observed the blending of uranium dioxide powder for Contract TGAF-01, Blend Number 4098-01. After the bulk container had been positioned in the tumbler blender, the operator set the revolution counter at 10. This was corrected by the supervisor and the counter was reset at 160 as required by COP-814741.

Nonconformance 86-02-02 was identified in this area.

c. Quality Control Sampling

(1) Procedure Review

The inspectors reviewed COP-814742, "Quality Control Sampling," Revision 5, dated August 2, 1986. The procedure requires that a Quality Control (QC) inspector designate sample locations, and a health physics technician monitor the sample hood ventilation on a weekly basis. The procedure requires that gloves and face respirators be worn while taking samples.

(2) Observation of Operations

The inspectors observed the sampling of Blend Number 3673-01 for Contract NSLT-OU. The QC engineer designated ten locations for sampling at the bottom of the bulk container. Ten samples were taken from the front to the back, and from the right to the left. Four samples were for chemical analysis, and six samples for isotopic analysis. The procedure was properly followed during the operations but the NRC inspector determined that the ventilation sample hood velocity was being monitored monthly intead of the weekly frequency required by COP-814742.

Nonconformance 86-02-03 was identified in this area.

d. Powder Re-Milling

The inspectors reviewed COP-814746, "Remilling Powder from a Bulk Container," Revision 7, dated March 21, 1986. This is done to control the physical properties of the powder. The procedure describes the item control system (ICS), the safeguards aspect of operations, and the step-by-step operational instructions for re-milling the powder.

The NRC inspectors observed the re-milling of Blend Number 4099-01 Contract TGAF-UL and the after re-milling of Blend Number 4098-01 for Contract TGAF-01. The steps of the procedures were properly implemented.

The NRC inspectors observed that the operator wore a respirator although the procedure does not require one to be worn. Procedure COP-814746 was revised during the inspection to document that respirators are being worn by the operators.

No items of nonconformance or unresolved items were identified in this area.

e. Prepare Bulk Blend for Pelletizing

The NRC inspectors reviewed "Prepare Bulk Blend for Pelleting," COP-814748, Revision 7, dated August 24, 1986, and observed blending of Blend Number 5532 for Contract 53E260. The steps of the procedure were properly implemented.

No items of nonconformance or unresolved items were identified in this area.

f. Press Pellets

The NRC inspectors reviewed COP-820201, "Press Pellets," Revision 12, dated June 19, 1986, and COP-820202, "Checking Green Pellet Density," Revision 6, dated March 5, 1986. The steps of both procedures were being properly implemented while pressing the pellets and checking the pellet density for Contract 5AAF-02.

No items of nonconformance or unresolved items were identified in this area.

g. Grinding Diameter Check

The NRC inspectors reviewed COP-820402, "Grinding Diameter Check," Revision 5, dated August 5, 1986, for checking the diameter of pellets after grinding. The procedure describes the test preparation, setup, step-by-step operation and establishes criteria for identifying pellets whose diameters are outside limits.

The steps of the procedure were properly implemented while checking the diameters of pellets for Contract TGAF-01.

No items of nonconformance or unresolved items were identified in this area.

3. Fuel Rod Fabrication

The NRC inspectors observed five steps in the fabrication of fuel rods and compared the procedural requirements to the actions being performed by the operators.

The following Mechanical Operating Procedures (MOP) and manufacturing steps were examined: MOP-750603, "Tube Labeling and Tube Weighing Operation Procedure," Revision 14, dated May 8, 1986; MOP-750604, "Cleaning Fuel Tubes," Revision 6, dated January 18, 1986; MOP-750304, "Fuel Loading of Preplugged Tubes (Bottom Ends) by Vibratory Loader," Revision 18, dated June 16, 1986; MOP-750305, "Manual Girth Welding of Fuel Rod Bottom End Plugs," Revision 11, dated February 28, 1986; and MOP-750294, "Manual Pressurization and Seal Welding of Fuel Rods," Revision 13, dated March 5, 1986.

Each MOP outlined the purpose, references, terms/definitions, responsibilities, regulatory requirements, special precautions, and procedure, which included preparation and operation. The personnel who were observed by the NRC inspectors performing the MOPs listed above were knowledgeable of the procedures, and followed the steps outlined in the procedures.

An additional area examined in the fuel rod manufacturing process was the equipment calibration and control system regarding ring gages. The inspector randomly selected ten ring gages located on the shop floor to assure that they were in the calibration control system and were being calibrated. For the gages selected, the inspector verified that each gage had a record in the tool and gage room and had been properly released to a specific area after being calibrated.

No items of nonconformance or unresolved items were identified in this area.

4. Fuel Assembly Operations

The NRC inspectors observed several steps in the fuel assembly process. The fuel assembly operation begins with the completed fuel rods being loaded into a magazine which hold the rods until they are placed in a fuel assembly skeleton. The procedure is outlined in MOP-730601, "Load Rods Into Magazine," Revision 19, dated August 18, 1986.

One observation made by the inspector during the review of the activities associated with this procedure involved the traceability of the fuel rods. After the fuel rods were loaded into the magazine, the bar codes used for the traceability of each fuel rod were read into the computer to assure that all the rods in the magazine were in fact QC released rods and the magazine did not contain any defective rods. The inspector observed the operator reading the bar codes into the computer after all but two rows of fuel rods were loaded into the fuel assembly skeleton. Although there is not a procedural requirement which covers when the bar codes are to be read into the computer, there exists the potential to have a defective rod inserted into an assembly before being identified.

After the rods are in the magazine, they are placed in the fuel assembly skeleton. The NRC inspectors observed two operators performing this activity as outlined in MOP-730102, "Load Fuel Rods Into Skeleton Assembly - Pull Loaded Assembly," Revision 20, dated August 4, 1986. After a visual inspection of the loaded fuel assembly was performed, the inspector observed MOP-730109, "Fixture Top Nozzle - Pull Loaded Assembly," Revision 28, dated September 9, 1986, being performed by an operator.

The final fuel assembly was inspected utilizing the following QCIs: QCI 938410, "Fuel Assembly Inspection (Envelope) Using LVDT Probes," Revision 24, dated June 16, 1986; QCI 938420, "Fuel Assembly Inspection (Tilt and Length)," Revision 40 , dated June 23, 1986; QCI 938435, "Fuel Assembly Inspection Channel Spacing," Revision 35, dated July 7, 1986; and QCI 938440, "Fuel Assembly T/N Weld and S-Hole Location Inspection," Revision 14, dated June 23, 1986.

The NRC inspectors observed that the QCIs listed above were performed by personnel as required, and no items of nonconformance were identified in this area.

During the observation of the fuel assembly final cleaning process, the NRC inspectors noted that a fuel assembly was in Tank 2 and that the temperatures of the wash/rinse tanks were as follows: Tank 1 - 142°F, Tank 2 - 162°F, Tank 3 - 122°F. Section 17.1 of MOP-730502, "Fuel Assembly Cleaning - General," Revision 13, dated August 21, 1986, requires that the wash/rinse tank operating temperatures be as follows in preparation for cleaning fuel assemblies: Tank 1 - 145°F to 175°F, Tank 2 - 165°F to 180°F, Tank 3 - 145°F to 195°F. The NRC inspectors were informed by the shop floor supervisor that maintenance

work had been performed on the tanks earlier in the day and the tanks were not yet at operating temperatures. However, an assembly was being cleaned in Tank 2. A deviation notice was written for the three fuel assemblies which were cleaned after the last temperature reading was taken which required that the three assemblies be recleaned. The NRC inspectors determined that these three assemblies were cleaned again with the wash/rinse tanks at the proper temperatures.

Nonconformance 86-02-01 was identified in this area.

Two quality packages were also reviewed. The packages were for completed fuel assemblies, 5A01 and 5A02, in the GAAF01 contract for the Vogtle Nuclear Power Plant. The quality packages contained the Quality Release, which certifies that the assembly was fabricated properly, the Folder Acceptance Index, QC Form 253, Revision 17, and the Fuel Assembly Routing Sheets. Each routing sheet covers a specific part of the manufacturing process and lists the sequence of steps, description, operation number, revision being worked, the operator's initials and the bill of materials. The two packages reviewed contained the required information and no items of nonconformance were identified.

5. Fuel Skeleton Assembly

The NRC inspectors observed some of the steps in the fabrication and inspection of the fuel skeleton assembly and determined that work observed was being performed in accordance with MOP-731101, "Assemble Skeleton," Revision 23, dated September 9, 1986. Skeleton components such as thimble tubes, grids, shoulder screws, plates, and nozzles were checked against routing sheets to ensure that the serial and part numbers matched the identification specified for each fuel assembly for a specific contract number. Thimble tubes and shoulder screws were loaded in a grid in accordance with MOP-731214, "Mechanical Assembly," Revision 8, dated August 13, 1986. The thimble tubes were crimped inside the grid sleeves in accordance with MOP-731214. The torque on the shoulder screws was verified in accordance with QCI 936110, "Thimble Tube Probe," Revision 40, dated August 11, 1986. The inside diameter of the thimble tubes were examined for cracks with a boroscope in accordance with QCI 936101, "Expansion Joint Quality," Revision 33, dated August 4, 1986.

No items of nonconformance were identified in this area.

6. Nondestructive Testing (NDT)

The NRC inspectors reviewed Westinghouse procedures for ultrasonic testing (UT), radiographic testing (RT), helium leak testing (LT), gamma scan and the NDT certification program for the NDT of fuel rods.

The procedures were technically correct and complete. The procedures had also been properly reviewed and contained the required signature approval. The NRC inspectors observed operators performing NDT utilizing these procedures. Observations of the NDT testing and review of the NDT certification program are itemized below:

a. A technician performed circumferential and axial ultrasonic scanning of zircaloy fuel rods in accordance with QCI 920103, "Ultrasonic Testing of Fuel Rod Welds," Revision 10, dated October 6, 1986.

The technique used was in accordance with the written procedure and the required documentation was properly completed.

Additionally, the control of the chuck pressure used to hold the rods during UT was reviewed to assure no indentation of the rods occurred after release from the jaws. The procedure listed a maximum pressure to be applied and this requirement was properly implemented.

b. A technician was observed performing radiographic inspections as described in QCI 920102, "Girth and Seal Weld X-Ray, Seifert X-Ray," Revision 34, dated September 29, 1986. The steps of the procedure were properly followed and the results were documented.

c. A technician was observed reviewing radiographs and processing film in accordance with QCI 928105, "Densitometer Operation," Revision 12, dated October 8, 1984, QCI 928106, "X-Ray Film Processing (Automatic)," Revision 6, dated February 17, 1986 and QCI 920101, "Weld Radiographic Inspection-Film Reading," Revision 64, dated October 6, 1986. All steps of these procedures were properly followed and the results were documented.

Additionally, three completed radiographs were reviewed by the NRC inspector to assure that proper disposition and documentation was achieved. In these instances, the proper disposition and documentation was in accordance with the applicable procedure.

d. A technician was observed calibrating and operating the helium leak tester to confirm the integrity of the assembled rods. The procedures used during these operations are QCI 928021, "Helium Leak Test (Equipment Operation)," Revision 12, dated November 25, 1985 and QCI 922101, "Helium Leak Test (Inspection Requirements)," Revision 36B, dated March 31, 1986. The steps in these procedures were properly followed and the results were documented.

e. The NRC inspector observed gamma scanning. The results of this test provide data on enrichment, gaps in the fuel stack, spring insertion and length of fuel in the rods. Observation of scanning verified that the steps of QCI 923103, "Gamma Scan," Revision 32A, dated August 22, 1986 were properly followed and the results were documented.

f. Training and Qualification of Nondestructive Test Personnel

The training and qualification records of four nondestructive test personnel were reviewed to assure compliance with the Quality Control Training Manual, Revision 2, dated October 11, 1983, the Quality Control Manual dated July 14, 1986, and QCI 000134, "Nondestructive Testing Certification Program," Revision 5, dated March 17, 1986. Documentation describing examinations and record of training was complete.

During the review of the actual testing for radiographic film interpretation, it was observed that questions requiring true or false responses were written in such a manner that either the true or false response would be correct. This was determined by reviewing actual tests and confirming that both the true and false response were considered correct. This observation was discussed during the exit meeting on October 9, 1986.

No items of nonconformance were identified in this area.

7. Fuel Incidents at Reactor Sites

The NRC inspectors reviewed selected information concerning fuel incidents which have occurred at reactor sites which may have been caused by steps in the manufacturing process. Information was obtained during a meeting with site personnel which included a conference call to Westinghouse (W) design personnel at Monroeville, Pennsylvania. The results of this review are as follows:

a. Incorrect Burnable Poison Orientation at Braidwood 1

Burnable poison (BP) assembly C54F arrived at the Braidwood site with the BP rods oriented 90 degrees from the correct position in the BP bundle. W's analysis concluded that the BP rods had been incorrectly oriented because the thimble plugs had been attached in the wrong holes in the holddown assembly. Procedure MOP-732220, "Assemble B/P-Source Assembly," Revision 15, dated March 18, 1986, was revised after the Braidwood incident to require that thimble plugs be matched with the specified traceability document to prevent a recurrence of this problem.

b. Incorrect BP Orientation at Vogtle 1

Two BP assemblies arrived at the Vogtle site with the BP rods oriented in locations which did not meet the design requirements for the Georgia Power order. W's analysis concluded that BP rod location symbol was drawn for core component type M10BP on W Drawing Number 1848E27, "A.W. Vogtle 1 Core Loading Plan Cycle 1," Revision 2, dated April 29, 1986. The two BP assemblies were manufactured according to this incorrect symbol. The symbol was drawn by a W engineer. W has instituted a requirement for an independent check of all core loading plans by an engineer who did not draw the location symbols on the plan being checked. This independent check was initiated by W to minimize the possibility for a recurrence of this problem.

c. Dents in Fuel Rods at Vogtle 1

Fuel rods arrived at the Vogtle 1 site which exhibited dents on the outside surface of the cladding. W determined that the dents were caused by the chuck on the ultrasonic testing (UT) machine. The most severe dents exhibited a maximum depth of 0.0007 inches based on metallographic examination of the two most severely dented fuel rods. The allowable W dimensional tolerance on

surface defects such as dents is 0.001 inches in depth. The dented fuel rods were accepted on a deviation notice - disposition request (DNDR) because the dents were less than the allowable 0.001 inches in depth. Dents have been eliminated in subsequent fuel rods by reducing the chuck pressure on the UT machine. This was reviewed as discussed in Section 6(a) above.

d. Bowed Fuel Assemblies at Vogtle 1

Fuel assemblies arrived at the Vogtle 1 site which did not meet the envelope dimensional requirements for bow. W reworked the assemblies to remove the bow, but the bow returned immediately after rework. Transfer waiver requests (TWR) have been issued by W to accept bowing on fuel assemblies for all contracts until drawings can be changed per engineering change notices (ECN).

e. Loose Fuel Pellets in McGuire Unit 1

On June 27, 1986, ten pellets were recovered from McGuire Unit 1. The top six inches of fuel rod number 16 in fuel assembly D-03 was missing and fuel rod number 15 was deflected two inches from vertical at the top. W suspects that the cause is associated with faulty fuel handling and/or baffle jetting. W concludes that these incidents were not caused by fuel fabrication.

F. PERSONS CONTACTED:

C. Alstadt
J. Barker
J. Bond
L. Burton
*J. Bush
D. Campbell
*H. Corey
*R. Cost
F. Culus
T. DeCristofard
*J. Donovan
G. Duke
A. Engel
C. English
*W. Goodwin
*B. Green-Field
E. Griffin
J. Hall
*J. Hart
B. Hartnett
*J. Higginbotham
*J. Hubich
*P. Kays
*E. Keelen
*C. LaBruyere
*E. Loch
H. Menke
C. Mitchell
*S. Patrick
R. Perkins
*R. Pollard
E. Quarterman
F. Shaffer
T. Silvey
G. Sinkler
N. Spears
*N. Storrs
C. Taylor
T. Riley
T. Ryans
B. Schumpert
K. Singleton
E. Sparrow
N. Stevenson
R. Talbert
B. Weaver

*attended exit meeting

ORGANIZATION: WESTINGHOUSE HITTMAN NUCLEAR, INC.
COLUMBIA, MARYLAND

REPORT NO.: 99900768/86-01	INSPECTION DATE: 9/8-10/86	INSPECTION ON-SITE HOURS: 48

CORRESPONDENCE ADDRESS: Westinghouse Hittman Nuclear, Inc.
ATTN: Mr. Bruce Rowe
Quality Assurance Manager
9151 Rumsey Road
Columbia, Maryland 21045

ORGANIZATIONAL CONTACT: Mr. Bruce Rowe
TELEPHONE NUMBER: (301) 964-5000

NUCLEAR INDUSTRY ACTIVITY: Westinghouse Hittman Nuclear, Inc. is a Certificate of Compliance (C of C) holder for a number of transportation packages (casks) and provides services for the shipping of radioactive material.

ASSIGNED INSPECTOR: John W Craig for 10/30/86
C. M. Abbate, Special Projects Inspection Section (SPIS) Date

OTHER INSPECTOR(S): R. P. McIntyre, SPIS

APPROVED BY: John W Craig 10/30/86
John W. Craig, Chief, SPIS, Vendor Program Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 71, 10 CFR Part 21.

B. SCOPE: Review Westinghouse Hittman Nuclear, Inc. (Hittman) activities related to 10 CFR Part 71 and 10 CFR Part 21 with respect to transportation casks, follow-up on previous inspection findings.

PLANT SITE APPLICABILITY: Nuclear power facilities who are registered users of the casks for which Hittman is the C of C holder.

ORGANIZATION: WESTINGHOUSE HITTMAN NUCLEAR, INC.
COLUMBIA, MARYLAND

REPORT NO.: 99900768/86-01	INSPECTION RESULTS:	PAGE 2 of 8

A. VIOLATIONS:

1. Contrary to Section 21.31 of 10 CFR Part 21, Westinghouse Hittman Nuclear, Inc. (Hittman) issued Purchase Orders (PO) for basic components, N43369, dated April 22, 1986 and N43888, dated July 22, 1986, to the George P. Thomas Rubber Company for replacement gaskets and N43492, dated May 22, 1986, to the Alleghany-York Corporation for replacement O-rings, without specifying that the provisions of 10 CFR Part 21 were applicable.

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

2. Contrary to paragraph 71.133 of Subpart H to 10 CFR Part 71 and Sections 16.1.3 and 16.2.15 of C-200, "Quality Assurance Program for the Use of Radwaste Shipping Packages," Revision 3, dated March 1, 1985, the Corrective Action Log indicated that several responses to Corrective Action Memos (CAM) from the Action Designees had not been received for items identified in 1985, several CAMs from 1984 and 1985 remain open and the last documented review of the CAM system by the QA Manager was performed on April 19, 1985.

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

3. Contrary to paragraph 71.137 of Subpart H to 10 CFR Part 71 and Sections 18.1.7 and 18.2.4 of C-200, "Quality Assurance Program for the Use of Radwaste Shipping Packages," Revision 3, dated March 1, 1985, audits scheduled to be performed in July and August, 1986, have not been conducted and the corrective actions for deficient areas identified in the 1985 Internal Audit have not been re-audited to verify implementation.

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

4. Contrary to paragraph 71.137 of Subpart H to 10 CFR Part 71 and Section 18.2.12 of C-200, "Quality Assurance Program for the Use of Radwaste Shipping Packages," Revision 3, dated March 3, 1985, three deficiencies identified in audit number 86-09 were not documented on a CAM as required.

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

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

1. (Closed) Violation 82-01-01:

Contrary to paragraph 21.21(a) of 10 CFR Part 21, procedures had not been adopted to provide for: (1) evaluating deviations or informing the licensee or purchaser; and (2) assuring that a director or responsible officer was informed if the supplied component (a) failed to comply or (b) contained a defect.

During this inspection, the NRC inspector reviewed Procedure HNDC-C-015, Revision 1, dated February 2, 1983, "Reporting of Defects and Noncompliances." The procedure outlines the requirements, responsibilities, and steps to be taken when a defect or noncompliance is identified. The procedure references Westinghouse Advanced Energy Systems Division (WAESD) Procedure DP-906, "Identification and Reporting of Nuclear Safety Related Noncompliances and Deficiencies." Revision 1, dated February 13, 1984, of DP-906 was also reviewed. This procedure provides an interface with the provisions of HNDC-C-015 to ensure that potentially reportable items are reviewed by the WAESD General Manager. Additionally, evaluations performed by Hittman's Operation Review Committee (ORC) of potentially reportable conditions were examined.

Based upon this review, this item is considered closed.

2. (Closed) Violation 82-01-02:

Contrary to paragraph 21.51(b) of 10 CFR Part 21, records were not prepared with respect to failures (in February through June 1981 testing) of Flextron Urethane 66-11 foam to meet the energy absorption characteristics described in the NRC approved Safety Analysis Report (page 71) for the HN-200 cask.

During an investigation conducted on January 28, 29; February 1-3, 16-19; March 1-5, 10, 1982, Report No. 99900768/82-02, the investigator concluded that the foam in the HN-200 impact skirts met the specifications presented in the Safety Analysis Report.

This item is considered closed.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Entrance and Exit Meetings

An entrance meeting was conducted on September 8, 1986 at the Westinghouse Hittman Nuclear, Inc. (Hittman) office in Columbia, Maryland. The purpose and the scope of the inspection were discussed during this meeting. Hittman is a subsidiary of the Westinghouse Electric Corporation. Hittman utilizes two Quality Assurance (QA) programs. C-100, is the QA program for the engineering, design, procurement, fabrication and initial testing of radwaste shipping packages, and C-200 is the QA program for the use of radwaste shipping packages. Both programs have been reviewed and approved by the NRC and encompass the criteria of 10 CFR Part 71, Subpart H. Cask designs are authorized for use through the Certificate of Compliance (C of C) issued by the NRC. Hittman is the C of C holder for several cask designs. During the exit meeting conducted on September 10, 1986, the inspection findings and observations were discussed with Hittman personnel.

2. Maintenance and Repairs

Hittman repairs and performs periodic maintenance on transportation casks. These activities are governed by Procedure HNDC-0-001, "Cask Maintenance and Repair," Revision 1, dated January 4, 1983.

During the inspection, the inspector examined the Routine Scheduled Maintenance Log for 1986. Routine Scheduled Maintenance consists of changing gaskets and O-rings, leak testing and other maintenance as specified in the C of C and by Hittman. The log was divided to assure that each cask received routine scheduled maintenance at least once every four months. The log included the log number, package identification number, dates of maintenance, where the maintenance was performed, who performed the maintenance, the tag numbers of the replacement material which was used, a summary of the work performed and a summary of the package condition.

During the routine inspections, form HNDC-Q-005-A, dated March 1982, is used. This form is a checklist of items to be inspected and includes items such as gaskets, nuts, bolts, and paint. After being completed, the forms are kept in a file which comprises the routine inspection history of the casks for that year.

When an item needs repair, a Corrective Action Memo (CAM) is written. The CAM describes the condition and the recommended action to be performed by the action designee. Replacement parts are identified in the storeroom by a green tag. Only green-tagged items may be used for maintenance activities. When the part is used in repair or replacement activities, the tag number and package on which it is being used are recorded on the Cask Materials Log Sheet, form 10.

The inspector reviewed individual files for three transportation packages and found that the information in the files regarding maintenance corresponded to the information on the Routine Scheduled Maintenance Log and packages were receiving routine inspections and maintenance at least every four months. The inspector also noted that the replacement parts in the storeroom were green-tagged and that the Cask Materials Log Sheet was being utilized and was up to date.

Purchase Orders (PO) N43369, dated April 21, 1986, N43492, dated May 12, 1986, and N43888, dated July 22, 1986, were reviewed. These POs were for replacement gaskets and O-rings. During the review it was noted that these POs for basic components were written without specifying that the provisions of 10 CFR Part 21 were applicable.

Violation 86-01-01 was identified in this area.

Four POs for replacement studs, nuts, bolts, and washers were also examined. All but one PO required a C of C from the supplier. PO 36330, dated December 2, 1980, did not require a C of C from the supplier, but Hittman sent samples to an outside company to perform chemical analysis on the material to assure that it met the specifications. In one case, one sample failed to meet the specifications and the lot from which this sample was obtained was rejected by Hittman.

3. Corrective Action Memo System

The Corrective Action Memo (CAM) System implemented by Hittman is outlined in Section 16 of C-200, "Quality Assurance Program for the Use of Radwaste Shipping Packages," Revision 3, dated March 1, 1985. This section outlines the requirements, responsibilities, and procedure to be followed when a deficiency is identified.

The Corrective Action Log was reviewed. This log lists the CAM identification number, when a response was received from the action designee, and when the CAM was closed out. During the review it was noted that no responses had been received for five items in 1985 and

that six items from 1984 and 1985 remain open. Additionally, Section 16 of C-200, requires that the QA Manager regularly review the CAM system to assure follow-up action is being taken. The inspector examined the CAM Review Sheet and noted that the last documented review of the CAM System by the QA Manager had been performed on April 19, 1985.

Violation 86-01-02 was identified in this area.

The inspector also reviewed CAM 86-05, dated April 22, 1986. The deadline for the response by the action designee was May 20, 1986, however, the response was not received until July 1, 1986. Although the response had been completed, it was not completed in a timely manner nor had the information been recorded in the CAM log.

4. Audits

Section 18 of the Hittman QA Manual, "Audits," as well as internal and external audits performed by Hittman in 1985 and 1986 were reviewed. The inspector also reviewed the 1984 and 1985 annual audit of Westinghouse Hittman by the Westinghouse Technical Services Division (WTSD).

Internal

The 1985 and 1986 Internal Audit Schedules were reviewed during the inspection. The schedule lists the areas to be audited and the status of the audits (response to findings received, follow-up action, item closed out). The eighteen criteria of 10 CFR Part 71 Subpart H are covered in audits performed by Hittman, Westinghouse Advanced Energy Systems Division (WAESD) or customers. It was noted that corrective actions for 1985 Internal Audit findings had not been re-audited to verify implementation and the 1986 Internal Audit Schedule did not reflect current activities in that the audits scheduled to be performed in July and August, 1986, have not been conducted.

The inspector also reviewed several internal audit reports from 1985 and 1986; 85-03, 85-05, 85-07, 85-09, 85-10, 85-11, 86-04, and 86-09 were examined to determine if Hittman is performing audits in accordance with the requirements of Section 18.

Some of the audit reports had not been written and issued in a timely manner. Audit 85-10 which inspected several different sections of the Hittman QA manual was performed in October 1985, with four deficiencies identified, however, the audit report was

not officially distributed until April of 1986. Audit 85-11 was performed in November 1985, the audit report was issued January 1986. Audit 86-09 was conducted in July 1986, with a draft report written. The report had not been officially issued at the time of the NRC inspection. In addition, the three audit findings listed in the report had not been documented on a CAM as required by Section 18 of the QA manual.

Violations 86-01-03 and 86-01-04 were identified in this area.

External

The inspector reviewed the Hittman Evaluated Suppliers List to determine which vendors are required to be audited by Hittman. The list consisted of five vendors including Riteway Machine and Specialty Company (Riteway) and Nuclear Packaging, Inc. (NUPAC), manufacturers of transportation casks for Hittman. Although no QA requirement exists, it was noted that no external audit schedule for 1986 had been developed, and one vendor on the Evaluated Suppliers List had not been audited in 1985.

The audits of the two cask suppliers reviewed by the NRC inspector were conducted with appropriate checklists and met the requirements of Section 18 of the QA manual.

No violations were identified during this part of the inspection.

5. Procurement

The NRC inspector reviewed the Hittman process for the procurement of safety related Radwaste Shipping Packages. Section 4 of the Hittman QA Manual, "Procurement Document Control," delineates the sequence of actions to be accomplished for the preparation, review, approval and control of procurement documents used for the purchase of items and services for Hittman packages.

Section 4 states that the purchase requisition shall include such information as:

1. statement of work to be performed by the supplier,

2. technical requirements,

3. QA requirements of 10 CFR Part 71 Subpart H, as appropriate,

4. documentation (such as drawings, specifications, procedures, material certifications),

5. requirement for access to supplier facilities,

6. extending applicable requirements to subtier vendors when necessary.

The inspector reviewed the Hittman procurement documents concerning Riteway for the fabrication of an HN-100 series 3 cask, and NUPAC for the fabrication of an HN-500 cask. These documents included the POs and the Cask Fabrication Specifications (SPEC). These SPECs were referenced on the PO. The SPECs set forth the technical and QA requirements to be placed on the supplier fabricating the cask. The technical requirements include performance, materials, fabrication, welding, and nondestructive testing procedures, while the QA requirements include 10 CFR Part 71 Subpart H, 10 CFR Part 21, buyers surveillance, audits, and C of Cs.

The inspector also audited the Final Data Package from the supplier to assure that requirements set forth in the PO and the SPEC had been included. The Data Package included documentation such as the C of C, weld certifications, material test reports, NDE reports, gamma scan and UT examination reports, final paint inspection reports, and appropriate travelers.

All procurement documents reviewed met the Hittman QA and procedural requirements and included the necessary documentation. Data Packages reviewed from Riteway and NUPAC included information that was stipulated in the PO and the SPEC.

No violations were identified during this part of the inspection.

F. PERSONS CONTACTED:

*B. Rowe, QA Manager
*J. Pawlik, Administrative Services Manager
T. Johnson, President
*S. McKew, Transportation and Maintenance Manager

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99901043/86-01	INSPECTION DATE: May 12-15, 1986	INSPECTION ON-SITE HOURS: 17

CORRESPONDENCE ADDRESS: Westinghouse Electric Corporation
Nuclear Services Integration Division
ATTN: Mr. T. A. Christopher
General Manager
Monroeville
Pittsburgh, Pennsylvania 15230
ORGANIZATIONAL CONTACT: M.A. Hamrick
TELEPHONE NUMBER: (412) 374-7504

NUCLEAR INDUSTRY ACTIVITY: Dedication of replacement components manufactured and supplied by other divisions of Westinghouse.

ASSIGNED INSPECTOR: K. Naidu — 9/23/86
K. R. Naidu, Reactive Inspection Section (RIS) — Date

OTHER INSPECTOR(S):

APPROVED BY: E. W. Merschoff — 9/23/86
E. W. Merschoff, Chief, RIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review status of previous inspection findings and discuss the implementation of WCAP-1062.

PLANT SITE APPLICABILITY: Limerick Unit 1 (50-352); Peach Bottom 2&3 (50-277; 50-278).

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99901043/86-01	INSPECTION RESULTS:	PAGE 2 of 8

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

The items of noncompliance discussed below were identified when the manufacturing operations were located at East Pittsburgh and are discussed in Inspection Reports 99900330/83-01 and 83-02. Since then, the manufacture of switchgear components and accessories for nuclear power plant applications have been transferred to their SECO Road facility (Docket No. 99901043).

1) (Closed) Violation (83-01): The violation identified that a current copy of 10 CFR Part 21 was not posted. The violation remained open pending review of measures established to prevent recurrence. The inspector observed two copies of Revision 6 of 10 CFR Part 21, dated September 12, 1983, posted at the SECO Road facility, where activities related to the refurbishment of Class 1E circuit breakers are performed. A note attached to the copy stated that although Revision 7, dated May 9, 1984 was issued, it was not posted because the revision is not substantive. The Quality Assurance Group (QA) is designated lead responsibility to ensure that current copies of 10 CFR Part 21 are posted at various locations. A list of locations and names of responsible individuals are maintained by QA.

2) (Closed) Nonconformance (83-01, Item A): The nonconformance identified that from the available inspection records it could not be established that 21 replacement undervoltage trip attachments (UVTAs) received 100 percent inspection of critical items. This condition was contrary to the commitments made in a Westinghouse letter (NS-EPR-2762) dated May 6, 1983, to the NRC (E.P. Rahe to H. Denton) which stated, in part, regarding the replacement DS-416 UVTAs, "After the failures at McGuire, replacement units were required to be manufactured...Since the problem at McGuire was linked to manufacture control

of critical dimensions on selected parts, 100% quality control checks of the...critical parts were instituted in the new procedure. Those parts were labeled 'critical' meaning only that they would henceforth be subjected to 100% dimensional checks...." Quality Control Inspection (QCI) 37, dated June 17, 1983, titled "Inspection requirements for selected critical parts for use in Class 1E equipment," was developed to document the inspection requirements to be applied.

3) (Closed) Nonconformance (83-01, Item B): The shop order information issued for replacement UVTAs did not identify the applicable drawing revisions to be used for the required 100 percent inspection of the "critical items". Document NQD 379 Revision 0, dated October 31, 1983, was developed which lists the critical parts and the respective drawing numbers, including current revisions.

4) (Closed) Nonconformance (83-01, Item C): The nonconformance identified that Certificates of Conformance (C of C) were on file for only five of the eight replacement UVTAs that have been shipped on Purchase Order 54X470254 to the McGuire site.

Discussions with cognizant personnel indicate that during April 1983, the UVTAs were not identified with unique serial numbers during manufacture. As such, the UVTAs were accounted for by the total lot shipped and not by the unique serial numbers. A report, C&Ps/EP-CB (83)-255, dated April 14, 1983, documenting a trip by a Westinghouse Nuclear Services Integration Division (WNSID) engineer to the McGuire site during April 4-7, 1983, provides the following information.

DATE	EVENT	DATE OF C OF C	QUANTITY RECEIVED AT SITE	QUANTITY INSTALLED
April 5, 1983 a.m.	2 UVTAs & 8 trip shafts were hand carried to W site office	April 4, 1983	2	0
	UVTAs had coils with incorrect voltage and were not given to the station	C of C voided	-2	0

DATE	EVENT	DATE OF C OF C	QUANTITY RECEIVED AT SITE	QUANTITY INSTALLED
April 5, 1983, p.m.	A courier hand-carried 2 UVTAs to replenish the ones given in the morning	April 5, 1983	2	0
April 6, 1983	Duke installed 2 UVTAs			2
April 8, 1983	Design error (groove to ring) identified, replacement activities commence	no charge		
April 10-11, 1983	6 UVTAs hand carried to site	April 8-9, 1983	6	
	Rejected 1 due to incorrect interface		-1	5
	Replaced 2 UVTAs installed on April 6		-2	-2
April 13, 1983	Shipped 3 UVTAs	April 13, 1983	3	3
	TOTAL		8	8

E. OTHER INSPECTION FINDINGS AND COMMENTS:

1. Background Information on Beaver Failures

a. Reported Failures

On February 6, 1986, the shunt trip coil in a LBB 22250 MW type molded case circuit breaker manufactured by Westinghouse failed at the Peach Bottom Unit 2 Nuclear Power Station. The breaker was installed in a transfer panel to protect the 120 volt, 60 Herz power supply to the Reactor Protection System (RPS) from undervoltage, overvoltage, and underfrequency conditions. The breaker was equipped with a magnetic overcurrent trip device and a shunt trip coil (STC) in series with an "A" contact that is activated by the breaker's moving contact arms. The failure

occurred when the "A" contact did not open when the breaker tripped; resulting in the overheating of the STC and ultimately an open circuit in the shunt trip circuit. On January 16 and 24, 1986, similar problems were identified in identical breakers at the Peach Bottom Unit 3 plant. On November 9, 1984, Limerick 1 reported an identical failure. It is essential that the STC remains operable to be ready to trip the circuit breaker on receipt of an abnormal voltage condition. The failures described above have occurred in LBB 2225OMW type breakers with a STC. ASCO Electrical Products Company, Incorporated (ASCO) Parsipanny, New Jersey procured these breakers from Westinghouse Beaver facility and installed them in safety related RPS power supply monitoring panels (required for Boiling Water Reactors (BWRs)) for the Limerick and Peach Bottom Nuclear Power Plants.

b. Follow-up Inspection at ASCO:

The NRC inspector conducted an inspection at ASCO on March 17 and 18, 1986 and the results are documented in Inspection Report 99901054/86-01. During the inspection, ASCO's Purchase Orders (POs) for the breakers to Westinghouse were reviewed. The POs were stamped "Q" to indicate that the breakers were intended for safety-related applications and require C of C stating that the breakers were suitable of Class 1E applications including IEEE-344. ASCO received C o Cs (with the breakers) from WB stating that the breakers satisfy Class 1E nuclear safety-related requirements addressed in IEEE-323 (1974) and IEEE-344 (1975).

c. Follow-up Inspection at Limerick and Peach Bottom:

To obtain additional information on the breaker failures, the NRC inspector visited the Limerick and Peach Bottom Nuclear Power Stations on March 24-26, 1986. The licensee disassembled one of the failed breakers at the Peach Bottom plant and explained the malfunctioning of the "A" contact in the breaker which causes the STC to burn out. The inspector observed that the rivet which joins the handle-post to the operating mechanism was not ground off during manufacture to remove excess material. The excess material in the rivet appeared to impede the proper functioning of the "A" contact which opens when the breaker trips. Westinghouse requested that Peach Bottom ship two of the six failed breakers to WB for tests and evaluation. WB completed the tests on April 10, 1986.

2. Meeting Summary

The Renewal Components Services (RCS) group of WNSID met with members of Westinghouse, Control Products Division, Beaver, the NRC and Philadelphia Electric Company (licensee) to discuss the molded case circuit breakers (breakers) which failed at the Limerick and Peach Bottom Nuclear Power Stations. The inspector stated that the purpose of the meeting and inspection was to obtain additional information on the results of tests performed at the WB facility on two of the six molded case circuit breakers which failed at the Limerick and Peach Bottom Nuclear Power Plants. The information is necessary to determine:

- Whether the RPS power supplies at the Limerick and Peach Bottom Nuclear Power Stations are adequately protected.

- Whether other types of breakers (model numbers) were manufactured with identical operating mechanisms which are susceptible to failures.

- Whether tests could be performed on existing breakers (in operation) to verify the correct operation of the STC.

- The necessity to issue an NRC information notice to alert users of these breakers of the potential problems when used with a STC.

Westinghouse would not provide these test results to the NRC inspector. This reluctance to provide information was contrary to Mr. J. S. Moore's policy statement dated June 29, 1983 which states, in part, "Westinghouse has committed per attached policy, to a program for ensuring that equipment, parts and services as provided by Westinghouse meet applicable requirements for nuclear service and that our customers to whom Westinghouse supplies such equipment and parts are provided with accurate and timely information for the proper and safe installation, operations, maintenance, and repair of such Westinghouse supplied equipment and parts...." Consequently, as stated in the cover letter to this inspection report, Westinghouse has been requested to provide this information within 30 days.

3. Westinghouse Test Recommendations

a. WB recommended certain tests detailed below be performed on the breakers at Limerick and Peach Bottom Nuclear Power Plants to verify operability of the STCs. If the breakers failed the tests they were to be replaced. Implementation of these tests would assure that the RPS power supplies were protected from being exposed to abnormal voltage conditions.

b. WB stated that LB 22250, HLB 22250, and DA 22250 type breakers have operating mechanisms similar to LBB 22250 type breakers and could experience failures when required to trip with STCs.

c. WB recommends the following tests to ascertain the operability of the breaker.

1) Perform a continuity check on the STC after each breaker operation via the two leads exiting the breaker.

2) The potential of non-performance of the back-up STC protection can be determined by completing the following tests.

a) Manually close the breaker and continue to hold the operating handle in the "ON" position.

b) Apply the STC rated voltage through the two leads exiting the breaker until the breaker trips or one second elapses.

c) While continuing to hold the handle in the "ON" position, perform a continuity check of the shunt trip circuit. Continuity indicates a malfunction in the "A" contact. An open circuit indicates that the "A" contact performed its intended function.

4. NRC Action

The NRC issued Information Notice 86-62 informing all nuclear power reactor facilities holding an Operating License or Construction Permit to alert them of a potentially significant problem involving the failure of the LBB 22250 type Westinghouse molded case breakers, and that LB 22250, HLB 22250, and DA 22250 type molded case breakers have similar operating mechanisms.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99901043/86-01	INSPECTION RESULTS:	PAGE 8 of 8

F. PERSONS CONTACTED:

Westinghouse Electric Company
Nuclear Energy Systems
Monroeville, PA

*D. N. Alsing, Manager, QA Systems and Compliance
*B. F. Barnett, Quality Engineer, NTSD
*A. K. Deb, Senior Engineer, RCS
J. Epstein, Manager, RCS, NSID
J. J. Evans, Manager, QA, NSID
*E. K. Figenbaum, Principal Engineer, NTSD
*M.A . Hamrick, Manager, QA Systems and Audits
*J. J. Jelovich, Manager, EQ Programs
*W. J. Johnson, Manager, Product Licensing
*S. Long, Senior Quality Engineer, QA Systems and Audits
R. B. Miller, Nuclear Safety Engineer
P. Morris, Manager, Instrumentation and CH Systems Licensing
*D. E. Rygg, Manager, RCS Engineering

Westinghouse Electric Corporation
Control Products Division
Beaver, Pennsylvania
G. Dore, Product Manager

Philadelphia Electric Company
Philadelphia
C. B. Harmon, QA Manager
C. B. Tuttle, Electrical Engineer
J. J. Gyrath, Electrical Engineer

ORGANIZATION: WILMAR ELECTRONICS
TORRANCE, CALIFORNIA

REPORT NO.: 99901068/86-01	INSPECTION DATES: 10/15-16/86	INSPECTION ON-SITE HOURS: 16

CORRESPONDENCE ADDRESS: Wilmar Electronics
2430 Amsler Street
Torrance, California 90505

ORGANIZATIONAL CONTACT: Zev Heifetz
TELEPHONE NUMBER: 213-326-6668

NUCLEAR INDUSTRY ACTIVITY: Wilmar manufactures catalog and custom order electronic relays used in diesel control panels and in various other applications.

ASSIGNED INSPECTOR: Jeffrey B. Jacobson, Reactive Inspection Section (RIS) 11/24/86
Date

OTHER INSPECTOR(S): None.

APPROVED BY: E. W. Merschoff, Chief, RIS, Vendor Program Branch 11-24-86
Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was conducted in response to a 10 CFR Part 21 report received from Sacramento Municipal Utility District (SMUD) concerning Wilmar Electronics supply of underpower relays to the Rancho Seco Nuclear Power Plant.

PLANT SITE APPLICABILITY: Rancho Seco (50-312).

ORGANIZATION: WILMAR ELECTRONICS
TORRANCE, CALIFORNIA

REPORT NO.: 99901068/86-01	INSPECTION RESULTS:	PAGE 2 of 4

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

N/A.

E. OTHER FINDINGS AND COMMENTS:

Quality Control

Wilmar Electronics manufactures a wide range of catalog and custom order electronic relays and sensors. A substantial number of these relays have been supplied to nuclear power plants or their vendors and in most cases were supplied as commercial grade equipment. Of the purchase orders reviewed, none invoked 10 CFR Part 21, however, a number of purchase orders stated the relays were to be able to meet various seismic requirements. Wilmar has not provided any certification of the relay's ability to withstand a seismic event, but similar Wilmar relays have been tested by Wyle Labs as part of seismically qualified control panels.

Two drawings reviewed, Wilmar drawing 21-441, "Close Differential Relay Three Phase Adjustable Model D101-17X" and drawing 21-442, "Frequency Relay Model 20-050-22X" called out 10 CFR 50 Appendix B quality assurance criteria. These relays were supplied to SMUD in June of 1980.

Wilmar's Quality Control Manual dated June 3, 1985, was reviewed and no deficiencies were noted. Wilmar relies on a testing program in which 100% of relays are functionally checked to ensure a quality product. Each relay is inspected by an inspector who applies his individual stamp to a satisfactorily tested component. Upon completion of the functional test, a dielectric test is performed to check for proper insulation integrity. Wilmar's test equipment control was reviewed and found to be

satisfactory. All equipment calibrations on the master calibration list were found to be up to date with a selective check performed on individual pieces of test equipment confirming this observation.

Design changes were found to be handled properly with changes documented and signed off on the appropriated drawings. Selective production department folders were reviewed and were found to contain the required up-to-date Schematic Diagrams and Bills of Material.

Underpower Relays Supplied to SMUD

On July 30, 1986, the NRC received a 10 CFR Part 21 report from SMUD stating that underpower relays Model 21-71 supplied by Wilmar were defective. The letter stated several of the 16 relays that were ordered contained overheated components, damaged circuit boards, bad solder connections, etc. Documentation pertaining to the supply and repair of these relays was reviewed by the NRC inspector.

The relays in question were originally purchased by Wolfe and Mann who installed them in panels which were seismically qualified by Wyle Labs in test report 42401-1 dated December 18, 1972. The purchase order for the supplied relays required them to be Seismic Class 1. After receiving the panels from Wolfe and Mann, SMUD sent the underpower relays back to Wilmar a number of times. The first time was for a change in the relay's control voltage source from an internal to an external source. This change was performed on all relays. Subsequently, the relays were sent back for a number of reasons. In some instances, bad zener diodes were found while in other instances the relays were tested and found to be in proper working order. Wilmar noted that SMUD had attempted repairs on the relays as noted by the observed installation of new, dissimilar components. In 1984, after failure at SMUD, Wilmar made a design change to the relay by replacing one apparently undersized resistor with two resistors whose combined power dissipation was considerably greater. SMUD did not request this change be performed on all 16 relays. Wilmar stated the design change along with the SMUD repairs may have been the reason SMUD has reported the relays no longer conform to the original Wilmar supplied drawings.

Wilmar has indicated the underpower relays were not designed for a 40 year life and that they should now be replaced with relays of a newer design. Wilmar has similarly recommended replacing the Model 21-172 phase balance monitor relays which were supplied to SMUD in 1972.

10 CFR Part 21

None of the purchase orders reviewed invoked 10 CFR Part 21 on Wilmar Electronics.

F. PERSONS CONTACTED:

*Zev Heifetz, General Manager
*Cesar E. Marestaing, Quality Control Manager

*Attended Exit Meeting.

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

REPORT NO.: 99900902/86-01	INSPECTION DATE: 9/2-5/86	INSPECTION ON-SITE HOURS: 50

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

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: for R. H. Lasky, Equipment Qualification Inspection Section (EQIS) — 11-17-86 Date

OTHER INSPECTOR(S): J. Grossman, Consultant

APPROVED BY: Uldis Potapovs, Chief, EQIS, Vendor Program Branch — 11-17-86 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: Nine Mile Point Unit 1 (50-220), LaSalle County Stations 1/2 (50-373/374), Indian Point 2/3 (50-247/287).

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

REPORT NO.: 99900902/86-01	INSPECTION RESULTS:	PAGE 2 of 6

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. OTHER FINDINGS AND COMMENTS:

I. Four (4) test programs relating to the Environmental Qualification (EQ) of equipment were examined. These programs were examined to verify the following:

a. Required test instrumentation with accuracies described, met the requirements of IEEE-STD-323/1974.

b. Equipment interfaces were addressed.

c. Test acceptance criteria was established as described in the test specifications or in the design engineering letters to meet the requirements of IEEE-STD-323/1974.

d. The same equipment was used for all phases of testing and represented a standard production item.

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

f. The test results were adequately reduced and evaluated against acceptance criteria described in the test specifications or purchase orders.

g. All prerequisites for the given test as outlined in the test specifications were met.

h. The description of test equipment included all materials, parts, and subcomponents.

i. Notice of anomaly reports properly documented the observed condition and reports were properly dispositioned.

j. Appropriate margins were applied as required by test specifications.

1. EQ of GE EB-5 and EB-25 Terminal Blocks for use in Niagara Mohawk Power Company's Nine Mile Point, Unit 1.

The final test report 17775-1 on the Terminal Blocks was examined. The test subjected the terminal blocks to two (2) temperature transients, however the second and higher of the temperature transients was for information only.

In examining test report 17775-1 it was noted that in some cases recorded test data was corrected by writing over the originally recorded data. When questioned about Wyle's policy for correcting test data, the inspectors were informed that corrections are to be made by lining-through original data, initialing and writting in new data. Wyle was already in the process of revising Quality Directive (QD) V-1, Reports, Test Procedures and Plans to recognize line-throughs and initials as the only acceptable method of correcting test data.

No nonconformance or deficiencies were found during the examination of this EQ program.

2. EQ of Raychem WCSF-N Nuclear Cable Splices and Okonite Tapes as installed on various wire insulations at Commonwealth Edison Company's LaSalle County Station.

The qualification plan for this program was examined. This program consisted of 20 different configurations of cable splices which could be found at LaSalle County Station. The splice configurations varied in Raychem sleeve or tape overlap and in the types of leads spliced together.

No nonconformances or deficiencies were found during the examination of this EQ program. The final report will be examined upon completion of the testing.

3. EQ of Lewis Engineering Electrical Cables installed at Indian Point 2 and 3 Nuclear Power Plants.

   The EQ test plan and test procedure of this program were examined. The testing had just been completed and the raw test data was examined. This program intends to qualify the Lewis Engineering cable used at Indian Point 2 and 3 Nuclear Power Plants.

   No nonconformances or deficiencies were found in the test plan, test procedure or data examined. The final test report will be examined during a later inspection.

4. EQ of Target Rock SRV Air Operators and Dual Solenoid Valve Assemblies for General Electric.

   Test report 47732-1 was examined for this program. The purpose of this EQ program was generic qualification of Target Rock SRV Air Operators and Dual Solenoid Valve Assemblies. The test consisted of aging and accident simulation of three configurations of air operators and one dual solenoid valve assembly with a seismic test of the completely assembled SRV. The three air operators were as follows:

   (a) Specimen #1 - EPDM O-ring and diaphragm
   (b) Specimen #2 - Silicon O-ring and diaphragm
   (c) Specimen #3 - Viton O-ring and Silicon diaphragm

   The test report defined twelve function tests that were performed on specimen #3 with solenoid valve assembly after each test segment.

   (a) Solenoid coil resistance
   (b) Coil insulation resistance
   (c) Pickup voltage
   (d) Dropout voltage
   (e) Actuator opening time for different voltages and pressures
   (f) Actuator closing time for same voltages and pressures as for actuator opening
   (g) Minimum pressure to stroke actuator
   (h) Pilot rod stroke

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

REPORT NO.: 99900902/86-01	INSPECTION RESULTS:	PAGE 5 of 6

(i) Solenoid valve leakage (valve closed)
(j) Actuator assembly leakage
(k) Solenoid coil inductance
(l) Dropout solenoid transition.

Only function tests (d), (e), (f), (g) and (i) were performed on specimens #1 and 2 after each test segment using the valve assembly.

In examining this test program the inspection team noted some concerns with three of the test anomalies.

(a) During the post aging functional test, specimen #1 exceeded the actuator leakage criteria. Examination revealed that the diaphragm nut was under torqued. Corrective action resulted in torquing the diaphragm nut to a value above the manufacture's specification to stop the actuator leakage. No justification or analysis was given for the final torque value.

(b) During the accident simulation, the diaphragm of specimen #3 failed and was replaced twice. Justification for continuing the test was that an identical diaphragm was installed in specimen #2 and could be used to establish qualification. However specimen #2 had not been mechanically cycled per IEEE STD 382-1980 as had the specimen that failed. Sufficient justification for the diaphragm qualification was not given.

(c) During the later stages of the accident simulation the spring cartridge/air operator seal failed three (3) times, once in specimen #2 and twice in specimen #3. On all occasions, the acceptance criteria for leakage of the actuator was exceeded. The seal was a standard Target Rock part and had not been irradiated or aged. Replacement was justified on the basis that the seals were not part of the specimen but part of the test fixture and therefore did not affect the test. There was no discussion in the test report of why the failure of a part which affected the acceptance criteria was not considered an EQ concern.

ORGANIZATION: WYLE LABORATORIES
HUNTSVILLE, ALABAMA

REPORT NO.: 99900902/86-01	INSPECTION RESULTS:	PAGE 6 of 6

During the examination of this test program no nonconformances or deficiencies were found in the programmatic review Wyle Laboratories testing.

II. The inspection team examined selected areas of Wyle Laboratories Quality Assurance (QA) Program for compliance to Appendix B of 10 CFR Part 50. The inspection team then examined selected internal QA audit reports, QA audit reports on outside suppliers of services and equipment, and purchase orders for outside services and equipment. In examining the audit reports it was suggested that the addition of a section which addressed previous audit findings would be useful. Wyle Laboratories agreed with this suggestion and stated that they would incorporate this suggestion into future audit reports.

During the examination of Wyle Laboratories QA program, no nonconformances or deficiencies were found in the selected areas examined.

III. Wyle Laboratories has been performing on-site inspections and refurbishments of Limitorque motor operated valves for nuclear utilities. The inspection team examined Wyle Laboratories training course outline for Limitorque motor operated valves. Wyle Laboratories training course appeared to cover all of the major Limitorque EQ issues to include wiring, T-drains, lubricants, seals and gaskets.

ORGANIZATION: YANKEE ATOMIC ELECTRIC COMPANY
FRAMINGHAM, MASSACHUSETTS

REPORT NO.: 99901042/86-01	INSPECTION DATES: 09/15-19/86	INSPECTION ON-SITE HOURS: 35

CORRESPONDENCE ADDRESS: Yankee Atomic Electric Company
Nuclear Service Division
ATTN: Mr. Donald Hunter
Vice President
1671 Worster Road
Framingham, Massachusetts
ORGANIZATIONAL CONTACT: Mr. Donald Hunter
TELEPHONE NUMBER: 617-872-8100

NUCLEAR INDUSTRY ACTIVITY: Engineering services for Yankee Rowe, Maine Yankee, Vermont Yankee, and under contract to other utilities.

ASSIGNED INSPECTOR: P D Milano — 10-29-86
P. D. Milano, Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): R. L. Pettis, SPIS; P. J. Prescott, SPIS; L. Cheung, Region I; J. Diperna, Consultant

APPROVED BY: P. D. Milano for — 10-29-86
J. W. Craig, Chief, SPIS, Vendor Program Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR 50 Appendix B.

B. SCOPE: Review the implementation of the YNSD design control program for plant modifications.

PLANT SITE APPLICABILITY: Maine Yankee 50-309

A. VIOLATIONS:

None.

B. NONCONFORMANCES:

None.

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not applicable. This was the first Vendor Program Branch inspection of Yankee Nuclear Service Division.

E. OTHER FINDINGS AND COMMENTS:

This inspection was performed to review Yankee Nuclear Service Division (YNSD) activities in the area of design control. In order to perform this evaluation, selected Engineering Design Change Request (EDCR) packages were reviewed for technical accuracy and compliance with regulatory commitments and industry standards. The EDCRs for the Maine Yankee unit were selected for the period from 1983 to the present.

The process of preparation, review, and approval of the EDCR is detailed in the Engineering Manual Procedure, WE-100. In this process, the plant issues service requests to the YNSD organization describing the scope of the modification or engineering study that is to be accomplished. YNSD utilizes this scope of work to develop the EDCR which includes the applicable calculations and design output documents. In the case of Maine Yankee, however, revised drawings are prepared by the utility from the YNSD sketches. The EDCR receives independent reviews prior to being submitted to the Manager of Operations for approval. Subsequent changes to the EDCR, whether requested by the field or YNSD, are handled by Engineering Change Notices (ECNs).

As previously stated, the modification packages for work on Maine Yankee were selected for review. These facility modifications were reviewed by taking the perspective of an independent design verifier such as design

reviews or review of alternate/simplified calculations. In general, the design output documentation and basis were found to be technically satisfactory and in compliance with regulatory and industry standards. The specific details of these evaluations are outlined below.

1. EDCR 85-04, Installation of 1 inch Size Auxiliary Feedwater System Check Valve

   This EDCR was prepared to allow installation of a 1 inch check valve in line No. 1 inch-WAPD-15-151 to provide a seismic pressure boundary for a 1 inch branch line off the steam driven auxiliary feedwater pump suction line.

   This seismic upgrade was accomplished by installing a 1 inch 600 psi ANSI class lift check valve at location AFW-358 and analyzing the support loads with the NUPIPE code up to the first orthogonal restraint past the check valve. The piping system was to be safety class 3 up to the check valve inlet and non-nuclear safety past the valve. This Maine Yankee system redesign was performed in accordance with ANSI B31.1 1977 and ASME XI thru Winter 1980 Addenda. Their FSAR commitment is ANSI B31.1-1967.

   An ANSI 600 psi pressure class check valve (Stone & Webster I.D. #VCS-60A) was procured and installed in the ANSI 150 psi line being seismically upgraded. This is a conservative valve selection and the valve procurement was performed by Maine Yankee site personnel. Consequently procurement documents were not available for review. However, the required engineering design specifications (YAGEN-39-8) were reviewed and were adequate. NUPIPE system analysis was verified and found to be acceptable. Seismic response spectra used are a conservative application of Regulatory Guide 1.60 ground response spectra, which is twice the values specified in Maine Yankee's FSAR.

   The adequacy of a single valve as a seismic pressure boundary was questioned. Yankee engineering made this determination per ANSI N.18.2 definitions which state that a single valve can define a safety classification break if the valve is a normally closed "passive" valve. The valve installed at location AFW-358 fits this criteria since it is normally closed and the steam driven AFW pump has no emergency feedwater function at Maine Yankee. Although credit for this pump is taken for station blackout and

Appendix R conditions, these are not currently FSAR commitments for Maine Yankee nor are they related to seismic events. Yankee engineering Letter NMY85-101 discusses this issue and the NRC's review and approval of the use of a single valve pressure boundary in this application.

2. EDCR 83-28, New Condenser Surge Tank and Piping

Although this design modification was on a Yankee Design Grade 2, Non-safety Related system, it was reviewed to determine whether or not there were indications of an appreciable change in the design control process between safety-related and nonsafety-related modifications. This design change was implemented to provide a new 100,000 gallon tank (TK-122) to supply demineralized water to the condenser surge line so the existing demineralized water storage tank can be used exclusively for AFW pumps supply. A new 10 inch SCH 40 carbon steel pipe, Stone & Webster class 151 per MYS-442, was also installed as part of this design change. While not safety-related, the condenser tank does provide a volume that may be used in an emergency.

This change was found to have been implemented in a satisfactory manner consistent with the applicable requirements.

3. EDCR 84-38, Check Valve Installation in Water Treatment Makeup Lines

Two 4 inch swing check valves were installed to reduce piping safety class 3 to non-nuclear safety class on the discharge line from the demineralized water storage tank to the auxiliary feedwater pumps suction. This change permits filling the demineralized water storage tank without requiring a dedicated operator to monitor system integrity should a failure occur in non-seismic water treatment piping.

Two Velan 4 inch ANSI 150 psi class swing check valves were installed at locations AFW-345 and 346 to provide the piping safety class break. Two valves were required to meet ANSI N18.2 requirements which require redundant valves when the single failure of an active normally open valve is assumed.

Since a special bimetallic (P1-P8) weld was required and installation was performed at the plant site, YNSD obtained a weld data map confirming use of a qualified Bimetallic Weld Procedure. The welding procedure was in accordance with General Weld Specification YA-WP-6 specified in the EDCR and was found to be satisfactory.

The Piping system load calculations were acceptable based upon a review of calculation MYC-576, Rev. 1, ATT.C. Seismic analysis was performed using a conservative "Worst Case" combination of FSAR and Regulatory Guide 1.60 response spectra. All calculations and specifications reviewed were acceptable.

4. EDCR 83-29, Auxiliary Feedwater System Modifications

This modification adds a second air operated, fail closed valve in series with an existing normally open, fail closed valve on the motor driven auxiliary feedwater pumps discharge line to prevent dumping feedwater inside containment after a main steam line break (MSLB). Previously isolation occurred when the feedwater flow control valves to a disabled steam generator automatically closed. The modification will allow the intact steam generators to continue to receive AFW flow. Both of the closed isolation valves will reopen after restoring steam generator pressure. This design change eliminates automatic trip of AFW pumps and the five minute restart delay in the present control scheme and uses the signal to operate the solenoids for the new isolation valves. The new control logic permits the motor driven AFW pumps to start automatically when low steam generator level occurs; both isolation valves in each feed line automatically close to block AFW flow on a low steam generator pressure signal.

The flow vs. head loss characteristic of the motor driven auxiliary feedwater pumps was documented in calculation MYC-390 and verified to provide in excess of the minimum 475 GPM flow required at the calculated system resistance. This calculation was found to be satisfactory.

The applicable valve design requirements were adequately specified in the valve procurement (Design) specification.

The control Pilot Solenoid Valves for the valve pneumatic actuators are ASCO M/N 206-380-2U which were confirmed by Yankee engineering to be identical to M/N HVU-260-280 which were qualified in excess of specified (4g) input by ASCO Report AQS-21678/TR, Rev. A.

The Valve-Actuator Qualification Report No. 191316-84N did not show compliance with specified minimum natural frequency limit, although the valve did stroke satisfactorily during the static deflection test simulating the worst case (4g) earth quake load.

Yankee Atomic engineering reviewed the Action Test Report and justified the natural frequency in the 26 to 28 Hz range. Using NUPIPE calculations, YNSD documented that the actual installed piping system frequency was conservatively less than the valve extended structure frequency and resulting accelerations were much less than the (4g) test level. Their evaluation was reviewed and found to be acceptable.

5. EDCR 85-14, Siesmic Anchorage of the Ceiling and Light Fixtures in the Control and Battery Rooms

This EDCR was initiated as a result of a report prepared by an independent seismic consultant under contract to Yankee NSD. The report addressed the concern that the suspended ceiling and associated light fixtures may fall during a seismic event, causing possible injury to operations personnel and damage to safety-class equipment.

The EDCR provided a review which was limited to simple hand calculations. These calculations verified that the attachment wires and anchors for the ceiling and fixtures were sufficient to resist the anticipated seismic forces.

Peak response for the control room ceiling was specified at 2.2g. This was translated to a conservative estimate of seismic loading on the attachments of 1.5 (2.2g) (W), where W is the total weight of the ceiling and fixtures supported by the anchors.

The results of the analysis demonstrated that the as-installed anchorage system was adequate and no modification was necessary.

6. EDCR 83-22, IE Bulletin 79-02 Support Modifications

This EDCR was developed to address the Maine Yankee Plant Operating Review Committee (PORC) comments which required adding additional pipe support modifications to those previously performed under IE Bulletin 79-02, Revision 2, dated November 8, 1979. The Bulletin addressed the need to review all safety class pipe support base plates utilizing concrete expansion anchors to take into account flexibility characteristics of the plate. Additionally, the Bulletin discussed existing expansion anchors and their safety factor when compared to anchor loads as calculated using the plate flexibility characteristics.

YNSD performed a review of the associated piping loads and support functions which revealed that a small number of support modifications were necessary. The calculations to support these modifications,

prepared by Yankee and CYGNA Energy Services, were reviewed and found to address the major concerns of the IE Bulletin and appeared to be satisfactory.

7. EDCR 81-26, Installation of Differential Pressure Transmitters, EDCR 85-18, Primary Inventory Trending System

These EDCRs contained design documentation for modifications required to meet NUREG 0737, Item II.F.2 and Regulatory Guide 1.97.

The Primary Inventory Trending System (PITS) consists of three differential pressure transmitters used for level indication and other display services to trend the mass inventory in the reactor coolant system. The transmitters are used for the following levels: a) top of pressurizer to bottom of core, b) reactor vessel, and c) top of reactor vessel to bottom of hot leg. In accordance with Regulatory Guide 1.97, Table 3 "PWR Variables," these are Type A, Category 1 instruments and are required to be environmentally qualified.

The inspector found that two transmitters, PT 3001 and PT 3003, were shown on Figure 1 of EDCR 81-26 to be mounted at elevation 16'-2" in the sump. This would subject them to flooding during a post accident condition. From Environmental Qualification file No. 0138-3, the environmental data was found to be: a) submergence depth up to 17.7 feet per SCEW sheet, b) pressure of 60 psig for 80 seconds per Figure 4.3.2, and c) operating time of 30 days per SCEW sheet. However, the flooding water temperature was not specified. For a worst case condition, it can be assumed to be the saturation temperature at the maximum expected containment pressure.

The manufacturer of the two transmitters was Rosemount, model 1154. The qualification test reports indicated that this model had passed a LOCA test for steam environment followed by submergence testing. The submergence test conditions were: 130° F, 50 psig, at 2 ft. depth for a duration of 2 weeks. These conditions could not envelope those required for Maine Yankee. Also, there were no evaluation of documents in the EQ file to justify this deficiency.

During the exit meeting, YNSD stated that these transmitters needed to operate at a maximum of 5 psig instead of 60 psig. Even with this pressure reduction, the test conditions still do not envelope the required conditions.

The PITS has not yet been placed in an operational status at Maine Yankee. This item must be resolved prior to placing the PITS in an operational status, and will require future followup at the station.

8. EDCR 80-13, Containment Water Level Indication, and EDCR 84-02, Installation of Gems Level Transmitters

The purpose of these EDCRs was to upgrade the existing containment sump level system to meet the requirement of NUREG 0737, Item II.F.1(5), which calls for continuous monitoring of this level in the control room. The original design utilized two identical indicating systems with an indication span of 0 to 10 feet. The primary function was for flood level indication. These systems, however, could not meet the environmental qualification requirements of 10 CFR 50.49 and Regulatory Guide 1.97.

EDCR 80-13 involved the replacement of the original GEMS model XM 29400 transmitters with new GEMS model XM 36495 which are qualified to IEEE 323 and 344-1971. Also, the instrument cables inside containment were to be replaced with qualified cables. These level transmitters fed two multipen recorders shared with the containment pressure instruments.

EDCR 84-02 provided further upgrades to this system. The level transmitters were replaced with GEMS model XM 54854. However, this model had a shorter range resulting in a span of 0.5 to 8.0 feet. The seismic support and the stilling wells were also redesigned.

The limited design information within the scope of the YNSD work included sketches of seismic mounting details, wiring changes, safety evaluations, and independent review records which were found to be satisfactory. Procurement and installation information was not within the scope of these EDCRs and was not available for review.

9. EDCR 85-07, Pressurizer Level

This EDCR modified the pressurizer level indication to meet the criteria of Regulatory Guide 1.97, two independent channels both providing trending capability in the control room. The modification involved the addition of isolators, and changes to power supplies and wiring to achieve circuit independence. One channel, LT-101X, provided input to a recorder on the main control board and the other, LT-101Y, to the main plant computer.

10. EDCR 85-02, Pressurizer Panel Modification

This EDCR provided for the relocation and replacement of several components on the pressurizer panel section of the main control board based on a human factors review. Since the components affected are part of the reactor regulation system, a safety evaluation indicated that the system is Non-nuclear Safety Class (NNS). Seismic requirements were placed on the mounting hardware to ensure that the effects of an event would not cause damage to other Class 1E equipment within the control board.

This seismic analysis was required in particular for the Pressurizer Level Channel Selector Switch, HS-101-1, since the original model was being replaced by a Westinghouse model OT2S1 selector switch with 7 contact decks. This required a redesign of the mounting hardware. Additionally, the operability of this switch in the required application due to torsional deflection of the operating shaft with this number of decks was an issue identified by YNSD. Thus, YNSD requested that the purchase order documents require Westinghouse to prove the operability prior to shipment.

In an ECN written against this EDCR, it was noted that the contact decks supplied by Westinghouse exhibited a problem in the field which required Westinghouse to fabricate two new sets of decks. The ECN did not describe the problem which necessitated this rework or whether the testing done at Westinghouse could have identified this problem.

11. EDCR 83-509, Main Steam Non-Return Valve Opening Vacuum Assist System

This EDCR prepared a modification to allow condenser vacuum to be utilized to aid in maintaining the main steam non-return valves (NRV) open during plant cool down. During normal operations, the volume above the piston of this Y angle lift check valve sees downstream line pressure through an equalizing line. However, the EDCR adds a tee in the equalizing line, a line to the condenser, and valves to allow selecting between the two points.

Rockwell, the valve manufacturer, raised a concern that excessive differential pressure across the operating piston may cause erosion and recommended a limit on the differential pressure. Based on this, an Engineering Change Notice (ECN) added a pressure gage on the equalizing line to aid the operator when throttling open the condenser line valve. In a comment made by a design reviewer, it

was recommended that the valve be disassembled after a refueling cycle to verify that the valve was not being damaged by this operation. Also, the valve in the line to the condenser must be locked closed whenever plant temperature is above 365° F to ensure proper NRV operation as well as to prevent damage.

The design documentation for this EDCR was found to be satisfactory.

12. EDCR 83-20, Maine Yankee Pressurizer Safety and Relief Valve Piping Modifications

This EDCR was prepared by YNSD as a response to Item II D.1.A of NUREG 0737 which discussed test of safety and relief valve functionability for expected operating and accident conditions and the testing should also demonstrate that the valves would open and reclose under the expected flow conditions.

Onsite testing of the pressurizer SRVs during the hot functional test program (performed in 1972) indicated stable operation of the valves. However, because a longer than usual configuration of SRV inlet and outlet piping, Maine Yankee determined that the existing configuration of this piping with the Dresser 31790KA SRVs should be modeled and analyzed using the Couple fluid dynamic computer program. The resulting analysis indicated that unstable operation for that configuration could be expected. Thus, a redesign to achieve a configuration of inlet and outlet piping combination with the Dresser SRVs which would result in stable blowdown under all anticipated conditions was begun by YNSD in September 1982. Results were documented in EDCR 83-20.

During the review of EDCR 83-10, the NRC inspector noted two (2) redesign modifications which were initiated at YNSD and recommendations were made which required further follow-up at the Maine Yankee site. The first modification involved five (5) pressurizer nozzles, in which six-component envelope loads were constructed for thermal, seismic, fluid dynamic and dead load cases. Those nozzles were found to conform to the provisions of (1974) ASME III, paragraph NB-3227.5 under the envelope loading and work. However, the 10 inch quench tank nozzle did not meet the requirements of NB-3227.5, nor could it be shown to conform to the requirement of subsections NC, ND, or ASME Section VIII, Division 1.

Since the maximum calculated stress in the quench tank shell at the nozzle could result in yield, YNSD attempted to demonstrate conformance with some provisions of the ASME III code, paragraph NB-3113.3

(emergency conditions - infrequent incidents) in order to allow higher stress limits. Conformance of the 10 inch quench tank nozzle with the provisions of (1974) ASME III paragraph NB-3313.3 had been accomplished except for the assurance that the number of stress cycles not exceed 25.

In a memorandum from YNSD dated February 10, 1984, it was recommended that Maine Yankee maintain a record of the total number of discharges of the pressurizer SRVs and the power operated relief valves (PORVs) into the quench tank and to ensure that the total number of discharges does not exceed 25 for the remainder of the plant lifetime. The memorandum also stated that, should this limit be exceeded, YNSD would then reevaluate the adequacy of the quench tank nozzle and provide recommended action at that time.

The second modification was a result of an analysis performed on flow characteristics through the Dresser SRVs, accomplished by the Couple fluid dynamic computer program. The modifications required that the SRVs receive a new set of nozzle ring adjustments, referenced in EDCR 83-20 Section 7.0.

Based upon the review of this EDCR, several aspects of this EDCR will require followup at the station:

1. Has a record/log been established at the Maine Yankee site to ensure that the total number of SRV and PORV discharges does not exceed 25 cycles for the life of the plant?

2. Were the new SRV nozzle ring settings verified by engineering quality control at the Maine Yankee site?

3. Did the pressurizer SRVs undergo any functional or operational testing, utilizing the new piping configuration and nozzle ring settings?

In an interview with YNSD personnel, it was stated that, recommendations are made by YNSD, to the site as part of the resolution to modifications. However, the follow-up to those recommendations which take place at the site, are not required to be documented as part of the EDCR package.

NUCLEAR FACILITY INSPECTION REPORTS

U.S. NUCLEAR REGULATORY COMMISSION

Report No. 50-280/85-35

Licensee: Virginia Power Corporation
Richmond, Virginia

Docket Nos. 50-280/281 License Nos. DPR-32/37

Facility Name: Surry Nuclear Power Station

Inspection Conducted: September 30-October 2, 1985 (corporate)
October 21-25, 1985 (station)

Inspectors: P. D. Milano, Special Projects Inspection Section (SPIS) — Date: 1/2/86

R. P. McIntyre, SPIS — Date: 1/3/86

J. J. Petrosino, SPIS — Date: 1/3/86

Approved by: John W. Craig, Chief, SPIS, Vendor Program Branch — Date: 1-3-86

Consultant: P. Farron, Nuclear Engineers and Consultants, Inc.

SUMMARY

Scope: This routine, announced inspection involved 210 inspector hours both in the corporate office and at the Surry Nuclear Station in the areas of the utility-vendor interface, the receipt and evaluation of vendor information including 10 CFR 21 reports, and the implementation of actions determined by the licensee to be appropriate in response to vendor supplied technical information.

Results: Within the areas inspected, several items of concern are addressed in the report for followup during future inspections. One item concerning the administration of vendor provided 10 CFR Part 21 reports will be forwarded to NRC Region II for followup as a potential enforcement finding.

REPORT DETAILS

1. Licensee Employees Contacted

a. Corportate

R. J. Hardwicke, Jr., Mgr., Nuclear Programs & Licensing
N. E. Clark
D. S. Cruden
L. J. Curfman
H. M. Fonticello
D. Hahn

b. Station

*R. F. Saunders, Station Mgr.
*J. Antigas
*D. L. Benson
*J. B. Costello
*E. P. DeWandel
*W. D. Grady
*E. S. Grocheck
*J. Headden
*J. B. Logan
*J. W. Patrick
*D. H. Rickeard
W. Kibbler
M. Haduck
J. DeMarco
E. Brennan
C. Core
M. Holchkiss
J. Kelly
J. Stoltz
L. Brown

NRC Resident Inspectors

D. Burke, Senior Resident Inspector
M. Davis, Resident Inspector

*Attended exit meeting

2. Exit Meeting

The inspection scope and findings were summarized on October 25, 1985 with those persons indicated in Section 1 above. The licensee was informed of and acknowledged the inspection findings listed below.

The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Enforcement Matters

This subject was not addressed in the inspection.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Control of Vendor Information

Information concerning the design, operation, and maintenance of installed equipment may be provided to the utility by various means. In the case of the Surry Station, information is available from the NSSS supplier, Westinghouse (W), through the use of W Nuclear Service Integration Division (NSID) Technical Bulletins, and from equipment vendors via Notifications of Potential Significant Safety Issues (10 CFR Part 21) reports, and other vendor service information letters or bulletins. Additionally, prior to 1982, Westinghouse utilized another method to transmit information to W Service Representatives called a NSID Data Letter. After the Salem ATWS event, W issued an index of Technical Bulletins and provided copies of all the Data Letters to their customers.

The inspectors reviewed the Virginia Power (VP) program that was developed to process vendor technical information. In the VP corporate office, W Technical Bulletins are received by the Nuclear Operations Department, Safety Evaluation and Control (SEC) Group. Each Technical Bulletin is logged and transmitted to the Surry Station. Procedure SEC 2.1, Revision 3.0, dated April 29, 1983, "Processing and Control of Industry-Wide Information and SEC Safety Reviews," is utilized in conjunction with Procedure SEC 2.2, Revision 3.0, dated April 29, 1983, "Guidelines for Review of Industry-Wide Information" to control review and distribution of vendor technical information. However, Procedure SEC 2.1 is used only for those items that are identified as needing an in-depth review. Also, Procedure SEC 2.2 reviews are performed only to determine applicability of the information to Virginia Power. Thus, technical evaluations of the W Technical Bulletins are not performed at the corporate office and the documents are transmitted to the station for technical review. If the information should require a review that is beyond the capability of the station resources, the station will then request the resources from Engineering and Construction (E & C), Licensing, or other corporate groups to perform or assist in the review.

For information provided to Surry Station, other than W Technical Bulletins, the tracking function is not performed as a general rule by SEC or other corporate staff.

It was observed that for 10 CFR Part 21 reports, Surry Station may (1) receive them directly; (2) may receive them from corporate staff without corporate review; or (3) may receive them from E & C with a preliminary review. As with the W Technical Bulletins, no formal reviews are performed by the corporate office on these reports without a request from the Station. The only deviation from this policy is for those reports from vendors that were initially transmitted to the original Architect-Engineer. In this case, E & C performs a preliminary evaluation prior to providing the report to the station. However, in all cases, the station is the responsible organization for the formal evaluation and corrective action determined to be necessary.

Vendor information such as technical manual changes, service bulletins, and vendor correspondence is not processed through the corporate staff. This information is received and handled, along with information sent to the station from the corporate office, in accordance with Station Administrative Procedure ADM-93, dated July 9, 1985, "Vendor Interface - Control of Vendor Documents."

Procedure ADM-93 requires that this vendor information be sent to Station Licensing where it is to be evaluated for applicability and determination/ assignment to the responsible station organization. Additionally, the items are to be included in the Commitment Tracking System under Procedure ADM-48.0, dated April 15, 1985, of the same title. The program was originally designed such that a Commitment Tracking System (CTS) input would be made upon receipt of the information by the Licensing Coordinator. However, due to system loading constraints, the CTS input takes place after the information has been reviewed by the responsible organization and a documentation change is determined to be necessary. By this method, the product corrective actions are not being addressed on the CTS.

The form sent by Licensing to the responsible organization via Records Management requires that a determination of whether a procedure or technical manual needs changing. Equipment changes or modifications required in response to vendor technical information are not included in the system. Thus, a vendor recommendation such as a one-time inspection that does not require a change to a manual or procedure does not go into the CTS.

b. Westinghouse (W) Technical Bulletins and Data Letters

As previously stated, W Technical Bulletins are first recorded (logged) by the Nuclear Operations Division, Safety Evaluation and Control Group (SEC) at the VP corporate office. This group ensures that a response is sent to W acknowledging receipt of the technical bulletin and that Surry Station acknowledges receipt of a copy. However, once it has been determined that the Station has the Bulletin, the SEC group provides

no followup to ensure that an evaluation of the information for applicability and necessary action is performed or implemented. As a secondary function, all correspondence going to or received from any W organization is recorded and a copy filed in the Nuclear Operations Department Technical Library.

During a review of the correspondence file the inspectors determined that not all of the W Technical Bulletins were included, which indicates that the Library is not the holder of all interface information with Westinghouse. Additionally the Library log was developed to indicate whether replies to incoming correspondence had been issued. However, this was not being performed and the log is now a chronological list of information received by the Library from Westinghouse or sent to Westinghouse by a Virginia Power organizational unit.

Since the administration and evaluation of W Technical Bulletins is a station function, the processing of selected bulletins was reviewed during the inspection at the Surry Station. Using this sample, the receipt, evaluation, and the implementation, if appropriate, of the information in the bulletin was reviewed. These bulletins are advisory notices of recent developments and events that pertain to the operation and maintenance and, to a lesser degree, design of the W supplied components. Recipients are advised to evaluate the information and W recommendations and initiate actions as appropriate.

The following is a brief summary of several of the Technical Bulletins (TB) that were reviewed during the inspection.

a. TB 85-08 - Emergency Hydrogen Recombiner

This TB informed all facilities using a W supplied Hydrogen Recombiner of a general workmanship problem that could lead to heater power supply cable insulation damage. Because of this, W recommended that recipients inspect the power supply cable for specific signs of damage and check for evidence of burrs or other sharp edges on metal parts that could cause them. If damage was found, the recommendations for repair were given. Also, it was recommended that the inspection be conducted yearly.

In the Surry Station evaluation done by Electrical Maintenance, the actions determined to be appropriate in response to this TB were to perform a preventive maintenance procedure, HG-CP-E/R1, "Clean and Service Control Panels-Hydrogen Recombiner," and then perform a EHR performance test, PT-37.1. The preventive maintenance procedure was reviewed and found to contain no specific information on the problem identified by W in the TB. Rather, this procedure addresses the normal step of checking for cable deterioration. The electrical maintenance technician who performed the above task was interviewed concerning the specifics of the inspection and his knowledge of the basis of the technical bulletin. He informed the inspectors that he performed his task in the normal manner through an access panel covering the control circuitry. In the area of cable deterioration, he was specifically looking for signs of cable overheating such as discoloration

especially at the crimped-on terminal connections. The technician was unable to state whether the complete cable length in question could be examined from the access panel.

The inspectors were unable to examine the heater power supply cable during this inspection. However, in reviewing the equipment technical manual, the locations shown of the five heater banks lead to questions as to whether or not an adequate visual inspection from the access panel utilized during routine maintenance could identify the potential insulation damage identified in the technical bulletin.

Additionally, Surry Station personnel referenced the last performance test of the recombiner as a basis for accomplishment of the bulletin. This test and the test results were reviewed. This test involves a heatup of the unit to an operating band of 1200 ± 50°F in 4 hours and was not successfully accomplished. A work order was reviewed which indicated that the temperature elements were tightened and adjusted, and the test was performed on April 21, 1985. However, a copy of the test results for this unit showing successful completion on the above or a later date was not available for review during the inspection.

b. TB 83-03 DB/DS Circuit Breaker Shunt and Undervoltage Coils Trip Test

This TB discussed the W recommendation that the station have procedures to independently test the shunt and undervoltage (UV) coil's ability to trip the main reactor trip breakers and the bypass trip breakers. Existing Surry procedures for periodic surveillance of the reactor trip function consisted of operating the main control board reactor trip switches which would simultaneously actuate both coil trips. The bulletin also provided general guidance from which each utility could develop their own plant specific procedure.

The acknowledgement memorandum from Records Management to the Maintenance Library was returned on June 27, 1984 indicating that changes to station procedures were not required. At the time of the NRC inspection, the inspectors were informed and verified that plant procedure PC-DB-E/R1 (later changed to PC-DB-E/A1) was changed to incorporate this information. No formal evaluation of this TB was available.

c. TB 77-02 - Position of Isolation Valves

This W bulletin discussed the potential for formation of nitric acid in the reactor vessel "O" ring leakoff lines following a refueling. W recommended that the outer and inner leakoff isolation valves be opened after installation of the reactor vessel head and lowering of the water level in the refueling canal. After sufficient time to ensure line drainage, W recommended that the outer "O" ring isolation valve should then be closed.

Surry Operating Procedure OP-15 has provisions to ensure that the leakoff isolation valves are verified open and the area between the inner and outer reactor head "O" rings are flushed of any nitric acid solution which could be detrimental to the reactor head "O" rings.

d. TB 83-01 - Reactor Coolant Pump Diffuser Adapter Bolts

This W TB discussed the fracture of diffuser adapter bolts discovered during inspections of reactor coolant pumps at two different W plants. W determined this was not a generic issue since the cause of the bolt fractures was traced to excessive torque and an improper lubricant. W recommended that only W approved thread lubricants be used and actions employed to prevent overtorquing the subject bolts. However, no unscheduled inspections of the pumps was deemed necessary or recommended by W.

Surry Station had no documented evidence that this TB had been evaluated for applicability to the Surry Station. Virginia Power personnel stated that one reactor coolant pump in each unit was inspected for broken bolts during the 10 year ISI inspection. No problems were encountered, and there are no future plans to inspect any other pumps due to the complexity of the inspection.

e. TB-83-06 - Steam Generator and Pressurizer Manway Bolt Torque Valves

This W TB recommended the use of Fel-Pro N-5000 for pressure vessel closure bolts as a replacement for Fel-Pro C5A and other thread compounds. This replacement also necessitated that utilities revise intermediate and final torque valves for steam generator and pressurizer primary manway closure bolts.

A Surry Power Station memorandum addressed this TB and stated that the applicable maintenance procedures MMP-C-RC-010 and MMP-C-RC-099 had been changed. The inspector verified that these maintenance procedures included the torquing requirements listed on the TB.

f. TB-84-06 - Reduced Auxiliary Feedwater Flow Rates

This W TB described a condition at a W PWR in which the surveillance test did not demonstrate that the auxiliary feedwater system (AFW) system could deliver its design flow at a pressure coinciding with that expected during a Loss of Normal Feedwater (LONFW) accident analysis. The recommended action was to verify that the measured AFW flow rates are consistent with the pressure conditions specified in the design criteria.

The Surry evaluation performed by Operations personnel stated that TB 84-06 was not applicable because flow rates are never measured during the performance of the surveillance activities. AFW Pump discharge pressure is the only parameter measured during AFW surveillance, which is done in the recirculation mode at Surry Station.

The issue described in this TB addresses the verification of the ability of a safety-related system to perform as designed. The evaluation of this TB was inadequate in that it did not address the function of the system but only of one component.

The Surry Technical Specification requires measurement of the system pressure while in the recirculation mode. No evaluation was performed to determine whether this surveillance is sufficient to demonstrate that the AFW system will perform its intended function (i.e., deliver rated flow at design pressure to the steam generators).

In addition to the above, this evaluation appears to be an example where the wrong department was assigned the responsibility to respond to the vendor recommendations. While Mechanical Maintenance was initially assigned the responsibility, Operations should have been the responsible department.

g. TB 85-01 - Reactor Coolant Pumps-Swing Check

Reactor coolant pump motor instruction books state that the oil lift pump should be in operation during the swing check of the lower motor bearing. In this TB, W recommends that the swing check operation be performed with the oil lift pump off to ensure that the swing value obtained is representative of the lower motor bearing only, rather than the upper and lower bearings in combination.

The Surry Power Station Electrical Department Procedure for the reactor coolant pump motors has this provision incorporated into the procedure. This evaluation is documented on station commitment assignment/response (SCARF) form 85-0800, closed May 15, 1985.

h. TB 85-03 - Reactor Coolant Pump (RCP) Coupling Advance

On several occasions, plants have experienced a coupling advance which exceeded the W limits of 0.050 to 0.055 inches. When this occurs, the coupling has to be removed and reinstalled. In this TB, W states that a coupling advance of 0.047 to 0.057 inches is consistent with the latest criteria for model 93 A-1/100 RCPs. Coupling advance which exceeds this parameter requires consultation with W and/or the removal and reinstallation of the coupling.

The Surry Power Station evaluation of this TB is documented on SCARF 85-0802, closed July 15, 1985. Mechanical Maintenance Procedure MMP-C-RC-009.1 was revised to reflect the information contained in the TB.

i. TB 85-05 - Reactor Coolant Pumps No. 2 Seal Upgrade

This W TB described operating experience which indicated a tendency for some No. 2 seals to intermittently "hang up" due to excessive friction between the O-ring/channel seal and the retainer (housing) balance diameter. W has developed an upgraded design intended to optimize No. 2 seal performance, while not significantly increasing

the seal face pressure and resulting seal wear rate. The changes involve the No. 2 seal retainer assembly, the No. 2 seal ring and the RCP seal service kit. W recommends that current inventories be depleted prior to use of the new style No. 2 seal.

The Surry Station evaluation of this TB was documented on SCARF Form 85-0804. Procurement documents were revised to reflect the new W part numbers. All replacement parts are now stocked separately including the individual parts in the RCP seal service kit. The Reactor Coolant Pump Technical Manual drawing DWG618J776, is scheduled to be revised to include the new part numbers.

Prior to 1982, W maintained another system by which their Service Representatives were kept informed of information by NSID. This system was called the NSID Data Letter. After the Salem event, W issued Technical Bulletin TB 83-05 which included an index of all currently valid Technical Bulletins and Data Letters, and notification that all future Data Letters would be internal W service documents and not to be used for customer notification.

While the emphasis of the review of the W supplied technical information was from Technical Bulletins, the accomplishment of the recommendations of several Data Letters was also reviewed. It was found that no formal evaluations had been conducted of the Data Letters that W had provided to VP. However, actions recommended in the Data Letters selected for review had been taken.

7. Significant Safety Item Reports, 10 CFR Part 21

The 10 CFR Part 21 Reports can be and are received by a number of groups within Virginia Power Corporation depending upon the developing organization for the commercial and/or technical aspects of the original purchase order for the affected equipment. The corporate contact for correspondence identified on the purchase order varies resulting in the recipient of the 10 CFR Part 21 report varying. In some cases, the reports were sent to the original architect-engineering firm that handled the initial construction prior to the report being forwarded to Virginia Power.

In several cases, the reports were addressed to a specific individual within Virginia Power. However, if the individual was no longer available, the unopened correspondence would be forwarded and the Station would never become aware of the potential safety concern.

The program for the processing and evaluation of reports of potential significant safety concerns from vendors is not specifically defined within the procedures at Surry Station. At best, the reports would be handled in a manner similar to that of other vendor information, such as W Technical Bulletins.

Unlike other vendor information, these reports are more than vendor recommendations and may transfer a responsibility from a vendor to Virginia Power for further determination of significance of a defect and implementation of appropriate corrective action.

During the inspection, selected 10 CFR Part 21 reports known to have the Surry Station as an affected facility were reviewed. A summary of the reports reviewed is projected below.

a. Conval Clampseal Valves

Conval Inc. notified Virginia Power of a deficiency in Conval Clampseal Valves per the requirements of 10 CFR Part 21 on October 11, 1982. A materials problem was identified in A105 carbon steel and A182-F22 alloy steel valves of sizes 1½ inches and larger, manufactured during the period of February 1981 to August 1982. Operation of these valves can result in a galling action between the stem and bonnet which could substantially increase operation torque and ultimately result in the inability to operate the valve.

The licensee was not able to locate any evaluation of this Part 21 notification or determine if these valves were installed at the Surry Station.

b. Terry Turbine/4" Gimpel Trip and Throttle Valve

On February 3, 1984, the Surry Power Station was notified by Terry Corporation, Steam Turbine Division, of a problem with the 4 inch Gimpel trip and throttle valve. This valve may not be able to close under certain conditions of high inlet pressure and low steam flow. The problem was caused by insufficient spring load which is required to overcome the valve stem imbalance at certain pressures. Gimpel made available a new spring which has sufficient force to overcome this stem imbalance with full rated pressure at all flow conditions.

The Surry Station evaluation consisted of a Maintenance Department memorandum dated May 16, 1984, and stated that the letter is not applicable to the turbines/trip and throttle valves installed at Surry. It further stated that the turbine serial numbers listed on the letter do not correspond to the Surry turbine serial numbers.

The memorandum is correct in that the listed serial numbers do not correspond to the auxiliary feedwater pump turbines. However, those turbines listed in the letter were previously in service as part of the containment spray system. While the containment spray pumps are now motor driven, there is at least one turbine listed in the letter still in place at Surry. Although it is not in operation, it has the potential for future use at the Surry Station.

c. Pacific Air Products Co. Linear Converters

On March 26, 1984, Virginia Power was notified by Pacific Air Products Co. of a deficiency in accordance with the requirements of 10 CFR Part 21 regarding the abnormal wear of linear converters manufactured by Pacific Air. This correspondence provided customers with a maintenance program and field lubrication procedures to be employed on linear converters.

The Surry Station evaluation of this report was inconclusive. The documentation stated that it could not be determined whether or not there are any Pacific Air Products linear converters in service at the Surry Station. It also stated that the maintenance program and lubrication procedures would be kept on file for future reference.

d. Foxboro Model N-E11 and N-E13 Transmitters

The Foxboro company notified Virginia Power that a potential deviation in the application of these transmitters with respect to expected performance under DBE conditions when calibrated to narrow spans. The Foxboro letter dated February 4, 1985, was transmitted by a memorandum to the station on February 8, 1985. While the memorandum and Foxboro letter were availible in the Technical Support Library, a documented evaluation was not available. During the inspection, the NRC inspectors found that Surry does not utilize this model in a safety-related application.

Based on these findings and a review of the applicable procedures, the Surry Station receipt, review and evaluation of 10 CFR Part 21 reports were found to be inadequate. The available procedures have not ensured that these deviations are adequately evaluated and appropriate corrective actions taken. This failure to evaluate and take corrective action, as appropriate, is considered a potential enforcement finding.

8. Other Vendor Information

While the majority of the vendor technical information is received by licensees through the previously described means, there are still other mechanisms by which this information is received at Surry Station.

Surry Station Records Management has made an initial effort to provide updates of equipment manuals and corrected technical information from vendors. This was evidenced by copies of letters in the Technical Support Library to vendors asking if the manuals held were still valid. This effort was initiated approximately one year ago. In a sample of the vendor files reviewed, it was noted that replies had not been received in all cases. Also, several vendors that did respond stated that insufficient information on the equipment was provided in the VP letter to prepare an accurate reply.

The information that is received by this effort along with that received from vendor information notices or letters is processed in accordance with procedure ADM-93. Therefore, the comments on the administration of the system that were made in previous sections of this report also apply in this section.

While reviewing the selected vendor files within the Technical Support Library, several vendor letters were reviewed and are listed below.

a. Enhancement of Reliability for EMD Diesel Generator Units

In a letter from Morrison-Knudson Company to Virginia Power, dated January 15, 1985, recommendations to enhance the reliability of the diesels were provided along with the probable cause of a turbocharger failure that occurred in December 1984. During the repair, the field representatives noted that the unit turbocharger still had the old style spring drive gear which EMD had recommended changing in 1976. Also, the viscous crankshaft vibration dampers presently used are no longer manufactured by EMD. This style damper should have been replaced with a gear type damper in 1978.

The letter also discussed tests that EMD conducted which show that repeat rapid starts could cause damage to the turbocharger thrust bearing. Thus, a modification to consistently supply oil to the turbocharger and crankshaft was recommended. Until such time as this modification is performed, Morrison-Knudsen recommended that owners avoid repeat fast starts after 15 minutes and before 3 hours following a "hot shutdown." These and the other Morrison-Knudsen recommendations are still under review by the VP engineering organization.

b. Charging Pump Speed Increaser

Correspondence from W to Virginia Power, dated February 12, 1985, described a potential problem with the lube oil pump coupling on the centrifugal charging pump speed increaser. (This correspondence was neither a Technical Bulletin nor a Data Letter.) In December 1984, two facilities suffered failures of the couplings and the intent of this letter was to notify Virginia Power of the potential problem.

On April 4, 1984, W sent an update letter on the changing pump speed increaser problem. This letter included a recommendation to include the lube oil pump coupling in a periodic inspection program to monitor coupling wear and recommended that the coupling be maintained/inspected during each plant refueling outage.

As of the date of this inspection, Surry had no documented evaluation of this potential problem, nor had they inspected any of the changing pump speed increaser lube oil pump couplings for abnormal wear.

Surry Station received W TB 85-19 on October 11, 1985, which described this problem. A memorandum was sent to the Supervisor of Records Management, on October 14, 1985, stating that this TB was to be distributed to the Maintenance Department to review for applicability and determination of whether procedure changes are required. While the evaluation of the Bulletin is in progress, no action has been taken based on the W letters.

c. Rockwell Univalves in RTD Bypass Lines

The mechanism by which this information was received at Surry Station is unique from those discussed previously, in that it was from another W information system called RADAR. This was a system devised by W to provide timely information of problems experienced in equipment supplied by W. The system is a computerized data base for rapid data retrieved from all utility outages. It is also a means by which potential problems and concerns which could affect availability or safety could be identified. If an event requires further analysis and resolution, it is assigned to a W task group. Resolutions which identify operational, design or procedural modification are then issued on a formal "Radar Response". Recently, in response to utility requests, the Westinghouse Owners Group developed the "Trip Reduction and Assessment Program" (WOG-TRAP). Since this system incorporates the objectives of the RADAR System, the RADAR System is ceasing and will be replaced by the WOG-TRAP program.

While the Technical Support Library did not have the RADAR System data base, several RADAR Responses were filed. Of the two, Response 84-09, "Stem Breakage of 2" Rockwell Univalve in RTD Bypass Line" was reviewed during the inspection. The response was received at Surry on July 10, 1984 and contained a description and an examination method to identify the problem. At the time of the inspection, the licensee could not provide an evaluation of the potential problem although the Maintenance Department had responded to Records Management on September 19, 1984 that the information affected installed equipment. However, when the inspectors asked for further information, the Surry Station personnel stated that the letter was not applicable because the valves installed were manufactured by Rockwell Edwards and not Rockwell.

The NRC inspector performed a separate review and found that Rockwell Edwards and Rockwell are the same and that the valves installed are either Hermavalves or the Univalves; while from the maintenance documents it appears that the valves installed are of the Hermavalve variety.

d. Surry Unit 2 Instrument Air Compressors

In June 1984, Surry Station requested Nash Engineering Company to provide assistance in troubleshooting the Unit 2 Containment Air Compressors. The service representatives found that the A Compressor, individually, could not maintain instrument air system pressure at the desired pressure. The B Compressor had the capacity but was running continuously rather than periodically unloading.

The Nash Engineering report dated July 23, 1984, detailed a number of deficiencies with the compressor systems and contained recommendations. They also recommended that the makeup water supply be secured to a shutdown compressor, or to vent the unload line to the waste gas sytem to allow proper draining.

Unlike other vendor information, this report was not processed as required by Procedure ADM-93. While these components are not classified as safety-related, recent industry problems with instrument air systems which have resulted in plant transients and challenges to safety systems necessitates this discussion.

8. Determination of Actual Installed Components

During the inspection, it was evident that no auditable system existed where components could be referenced to actual manufacturer or model number. An unofficial summary was developed and located in the control room for mechanical and electrical components. However, this summary was not verified, nor did it contain Instrumentation and Control components. Further, this listing was not available in the Technical Support Center.

Report No.: 50-327/86-61 and 50-328/86-61

Licensee: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos. 50-327 and 50-328 License Nos.: DPR-77 and DPR-79

Facility Name: Sequoyah Nuclear Station Units 1 and 2

Inspection Conducted: September 15-19 and September 29-October 3, 1986

Inspectors: E T Baker — 11/5/86
E. Baker, Team Leader — Date

J. J. Petrosino — 11/5/86
J. J. Petrosino, Quality Assurance Specialist — Date

Steven R. Stein — 11/5/86
S. R. Stein, Electrical Engineer — Date

E T Baker for — 11/5/86
J. C. Harper, Metallurgical Engineer — Date

Approved by: E. W. Merschoff — 11/5/86
E. W. Merschoff, Chief, Reactive Inspection Section, Vendor Program Branch — Date

SUMMARY

Scope: This was an announced reactive inspection which was conducted at the Sequoyah Nuclear Station to determine the effectiveness of TVA's corrective action for problems identified in Nuclear Safety Review Staff Reports R-84-17-NPS, I-83-13-NPS, and R-85-07-NPS concerning past procurement and record storage practices. Areas inspected included procurement actions taken for replacement of safety-related electrical and mechanical components of the residual heat removal, control air, and emergency power (diesel generator) systems; a sample of procurement actions identified by TVA's review of Class 1E replacement devices; a sample of procurement actions taken on equipment selected during a plant walk-through; NDE personnel qualifications; and Power Stores receiving inspection practices.

Results: One Potential Enforcement Finding was identified.

REPORT DETAILS

1. Persons Contacted:

Licensee Employees

*H. L. Abercrombie, Site Director
*W. E. Andrews, Site Quality Manager
*J. Blankenship, Information Office Manager
L. D. Brewer, Control Shift Operator
R. Bruce, Program and Standards Section Supervisor
*R. H. Buchholz, Office of Nuclear Projects (ONP) Site Representative
*S. K. Chapman, Electrical Maintenance General Foreman
M. Crane, Site Services Procurement Supervisor
*D. C. Craven, Quality Assurance Manager
*M. R. Harding, Site Licensing Manager
R. Jenkins, Division of Nuclear Engineering (DNE) Supervisor
*Z. M. Kabiri, Site Services Manager
*J. W. Kelly, EA Engineer
G. Kirk, TVA Compliance
T. Kontovich, Electrical Maintenance Engineering Supervisor
*J. E. Law, Chief Quality Systems Branch
*A. S Lehr, Instrument Maintenance Group Engineer
*M. R. Matthews, Restart Task Force
*B. Patterson, Maintenance Superintendent
G. W. Petty, Materials Officer
H. L. Pope, QC Supervisor
*M. A. Purcell, Licensing Engineer
*B. Schofield, Licensing Engineer
*J. R. Staley, Power Stores Supervisor
C. Stutz, QA Engineer
*P. R. Wallace, Plant Manager
*J. R. Watson, QA Engineer
*W. J. Wilburn, Site Maintenance
*L. J. Wheeler, Materials Manager
*C. R. Wherter, Materials and Procurement Services Staff Chief

The inspectors also contacted or interviewed other licensee personnel from the operations, training, maintenance, and technical staff.

Other Organizations

R. V. Matheison, Westinghouse Site Representative

NRC

*P. E. Harmon, Resident Inspector
*K. M. Jenison, Senior Resident Inspector
*D. P. Loveless, Resident Inspector
*E. W. Merschoff, Chief, Reactive Inspection Section

*attended exit meeting

2. Exit Interview

The inspection scope and findings were summarized and discussed with those individuals identified in paragraph 1, above. The team members described the areas inspected and discussed the inspection findings listed below. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Enforcement Matters

This subject was not addressed in this inspection.

4. Unresolved Issues

No unresolved issues were identified during the inspection.

5. Background

TVA's Nuclear Safety Review Staff (NSRS) performed in-depth audits of TVA's procurement system and records retention system. The results of these audits are documented in reports I-83-13-NPS dated June 17, 1983, R-84-17-NPS dated March 8, 1985 and R-85-07-NPS dated July 5, 1985. These reports highlighted programmatic deficiencies without providing specific examples of the problems found. One of the more significant findings related to the use of components purchased "commercial grade" and used in safety-related systems without any additional determination by TVA of the components' qualifications for use in safety-related applications. Another significant finding dealt with the difficulty or inability to retrieve quality assurance records in a timely manner. TVA took corrective action in response to these findings to improve their procurement and record storage programs to prevent similar problems from recurring. However, these corrective actions did not appear to address the effect past procurement practices may have had on the quality of installed equipment.

This inspection was conducted to determine how procurement practices for original and replacement components and parts affected the original qualification or design requirements of safety-related equipment installed in the plant or if record storage practices prevent determining whether or not the qualification of equipment has been adversely affected. The inspection utilized three separate methods for selecting equipment for detailed review:

a. A sample of electrical, mechanical, and instrumentation components was selected during a plant walk-through for a review of the related procurement documents. Equipment was selected based on problems identified during previous inspections of licensees and vendors. Problems noted in paragraph 6.b.7 were identified for an item from this sample.

b. A computer listing of maintenance requests (MRs) initiated during the time period 1980 to 1984 was obtained for selected systems and MRs were selected from these lists for follow-up based on whether or not a safety-related component was replaced. Problems noted in paragraphs 6.b.1, 6.b.6, 6.b.9, 7.a, and 7.b were identified for items drawn from this sample.

c. TVA began an effort in July 1986 to evaluate the seismic qualifications for a sample of Class 1E devices that had been replaced. From a total of approximately 125,000 MRs, they identified 238 MRs under which Class 1E devices had been replaced. At the time of this inspection TVA was in the process of determining the status of seismic qualification for a sample of 50 MRs which represented Class 1E replacement devices supplied directly from stock. This inspection reviewed several MRs from the 238 that were not part of TVA's sample of 50. Problems noted in paragraphs 6.b.2, 6.b.3, 6.b.4, and 6.b.5 were identified for items drawn from this sample.

6. Review of Procurement

a. Program Review

The team reviewed Part III, Section 2.1, "Procurement," of the Operating Quality Assurance Manual; SQA-45, "Quality Control of Materials, Parts, and Services," Sequoyah's site specific requirements for procurement; and Sequoyah's Administrative Instruction AI-11, "Receipt Inspection, Handling, and Storage of QA Materials, Substitutions and QA Level/Descriptions Changes."

A review of the definitions section of SQA-45 shows that TVA accurately reflected the NRC's intent for the terms "basic component," "Class 1E," and "commercial grade item," terms commonly used when discussing procurement. Three other definitions of interest to the team were, "Level I Quality Assurance Material," and "Level II Quality Assurance Material," commonly referred to as QA Level I and QA Level II, and "Critical Structures, Systems, and Components (CSSC)." A significant part of the problems discovered during this inspection stem from the definitions given for QA Level I and QA Level II and their use. TVA's definitions for these terms are as follows:

> Level I Quality Assurance Material--(1) Those CSSC materials, components, and spare parts which are a part of and whose failure would violate the reactor coolant pressure boundary; (2) those other CSSC items that cannot be bought by industrial standards and reference part number; (3) certain items that must be specifically bought as Level I as required by Attachment 21 (reference DP N76A10) of this standard practice, and (4) electrical equipment defined as Class 1E.

Level II Quality Assurance--CSSC materials, components, and spare parts that can be purchased by positive identification, provided that these items are not QA Level I. QA Level II permits in-kind replacement of items when the specified industry code or standard provides positive identification. Positive identification is established for an item by using one of the following methods:

a. Documentation supplied with the original equipment and/or from nameplate/tag information attached to the equipment.

b. The item's description as specified by the engineering drawing bill of material, provided no special process is performed due to drawing notes, etc.

c. The item's description as specified in the procurement technical specification.

d. The item's noun name and part number when supplied by the original supplier, manufacturer, or vendor.

These definitions are atypical of what is normal industry practice in that they describe what equipment falls into each category rather than describing the quality assurance requirements applicable to each category. Appendices to Attachment 20, "Specification of QA Program Requirements for Nuclear Power Requisitions," of SQA-45 contain the actual quality assurance requirements. These appendices are written for particular categories of equipment, e.g., Appendix 1, "ASME Section III Materials;" Appendix 2, "ASME Section III Valves;" Appendix 5, "Class 1E Materials and Equipment;" Appendix 8, "Non-ASME Code Items Which Have Unique Nuclear Requirements."

A note which is pertinent to the findings of this inspection appears under the title of Appendix 8 as listed on page 250 of the 7/12/82 version of SQA-45. The note reads as follows:

> ..."Used for procurement of QA Levels I and II [10 CFR 21 Applicable] basic components which have unique nuclear requirements, including Seismic Category I Application and equipment used in Class 1E application [device as a whole which requires qualification by supplier or manufacturer] which are not ASME Section III Code."

A cursory reading of this note by someone not familiar with TVA's interpretation of the phrases "10 CFR 21 Applicable," "Seismic Category I Application," and "Class 1E application," would lead one to believe that electrical and non-ASME mechanical equipment that was required to operate during and after a seismic event would be subject to the requirements of Appendix 8. However, as the next two paragraphs show, this is not the case.

When TVA procedures discuss requirements for Seismic Category I application and Class 1E application, they are not necessarily referring to the specific pieces of equipment which must function during or after a seismic event. The statement, "device as a whole which requires qualification by the supplier or manufacturer," appears in the note quoted above. As explained by TVA personnel during the inspection and as can be seen from the examples cited in paragraph 6.b. below, TVA personnel interpreted this to mean that seismic and 1E qualification was only applicable to the assembly qualified and supplied by the manufacturer. Therefore, if a "whole device" (e.g., motor control center, diesel generator set) was replaced, seismic qualification of the replacement was required. However, if a subcomponent of the whole device was replaced (e.g., relay, circuit breaker, air starter motor) seismic qualification was not required for the replacement, regardless of whether or not the replacement had to function during or after a seismic event. The reason given for this practice was that TVA was ordering replacements by part number and was receiving an "in-kind" or "like-for-like" part. This practice was not adequate to assure a qualified part, however, because the procurements were made QA Level II, no documentation required, which amounted to commercial grade and provided no assurance that the replacements were in fact "like-for-like."

The procurement of safety-related equipment requiring seismic and environmental qualification as QA Level II occurred because of two problems. The second statement in the definition QA Level I and the definition of QA Level II allow items which can be identified by a unique manufacturer's part number and are produced to industry standards to be procured as QA Level II regardless of the safety function of the item or the industry standard invoked. The other problem is that rather than basing quality requirements solely on the importance of the item being procured, TVA based quality requirements on whether or not 10 CFR 21 was judged applicable to the item. This problem was compounded by the fact that the form for determining Part 21 applicability, Attachment 11 to SQA-45, was biased towards not imposing Part 21 on anything less than a whole device.

Therefore, a piece of equipment which was originally supplied by the manufacturer as seismically qualified could be replaced by a commercial grade item without appropriate engineering analysis because TVA's procedures allowed the equipment to be classified as QA Level II, Part 21 Not Applicable.

Paragraph 10.3 of ANSI N45.2.13 states that equipment may be accepted by source verification, receiving inspection, supplier certificate of conformance, post installation testing, or a combination thereof. As the examples in paragraph 6.b. show, TVA chose to accept equipment based on receiving inspection and post installation testing. The receiving inspection performed by TVA during the time period 1980-1984 consisted only of a check for shipping damage and cleanliness, and a comparison of the part number on the item to the ordered part number. Functional testing is performed after installation and assures that the piece of equipment will function during normal operation. However, receipt inspection and functional testing in-situ cannot

validate the ability of a piece of equipment to function during or after a seismic event or exposure to a harsh environment. The other methods of acceptance available to TVA were source verification and supplier certificate of conformance. The "source verification required" block and the "documentation required" block were marked N/A for all QA Level II procurements reviewed, although some manufacturers submitted certificates of conformance (C of Cs) anyway. However, in order to take credit for C of Cs, paragraph 10.2 of ANSI N45.2.13 requires that the purchaser have a means of verifying the validity of the C of C. The means permitted by paragraph 10.2 are audits of the supplier or independent inspections or tests. While TVA performed independent inspections and tests, these tests could not be used to validate a C of C for seismic or environmental qualification since they only involved functional testing during normal operating conditions. TVA also performed audits of the suppliers of most of the equipment for which documentation packages were reviewed.

However, a review of those audits indicates that the programs reviewed were the nuclear quality assurance programs and not the commercial grade programs. Based on previous NRC vendor inspection experience, if the nuclear quality assurance program is not invoked in the procurement document, the items being procured are usually not manufactured under the nuclear quality assurance program. Therefore, a C of C received based on a contract which invoked the commercial grade quality program can only be considered valid if the commercial grade program has been audited and found acceptable.

b. Procurement Implementation

The following describes the results of the review of MRs, procurement documents, receipt inspection records, and other associated documentation for selected equipment.

1) Steam Generator Level Solenoid Valve (2-LSV-003-174):

MR A-593379 was issued on 12/2/85 to replace the coils in this ASCO solenoid valve (model 206-381-2RVU) which provides a control function for the Main Steam Isolation Valves. The valve being worked on, S/N 49243J-90, was an environmentally and seismically qualified valve, purchased under contract 80KJ3-00827551, which had imposed 10 CFR Part 21 and an Appendix B quality assurance program on the manufacturer. The valve location is the West Valve Room, a harsh environment. Because of this, the maintenance was being controlled under the Qualified Maintenance Data System (QMDS) and the MR was marked "QMDS" and "10 CFR 50.49 Device."

As stated above, the original scope of work was to replace the coils in this valve. However, because the craftsman could not remove the coil subassembly with the tools supplied for the job, the whole valve was replaced with another fully qualified valve, S/N 84146N-8.

If the craftsman had not encountered difficulty installing the replacement coils, the qualification of the original valve would have been indeterminate since the coils were procured as pieceparts, QA Level II, no documentation required, and no review of the effect of the commercial grade coils on the seismic and environmental qualification of the original valve was made.

2) Reactor Trip Bypass Breaker B (2-BKRC-099-KH/319-G):

The undervoltage trip attachment (UVTA) on this Westinghouse DB-50 breaker was replaced under MR A-101112 on June 24, 1985. The UVTA withdrawn from stock was one of 22 UVTAs purchased under contract 84P64-343921 in November 1983. The contract identified the UVTAs as QA level II, and did not impose QA, 10CFR21, seismic or documentation requirements on the supplier. However, the supplier (Westinghouse) modified the original bid and self-imposed QA and 10CFR21 requirements. The microfilmed contract file at Sequoyah includes a Certificate of Qualification from Westinghouse for ten UVTAs, but only four of the ten listed correspond to the UVTAs delivered under contract 84P64-343921 and the UVTA installed on this breaker is not one of them. As there is no documentation to support qualification of the UVTA, the qualification of the breaker assembly is indeterminate.

3) Normal Feed Breaker to Reactor Vent Board 2B-B (2-BDC-201-JQ-B):

The Reactor Vent Board 2B-B was originally purchased with the 480V motor control centers under contract 71C2-54752, dated May 5, 1971. The contract required a quality assurance program and seismic testing and qualification for the safety-related motor control centers listed on the contract. The seismic test report, dated March 1972, also documented successful separate testing of internal components, including a circuit interrupter of the type specified for the motor control center feed breakers.

On December 26, 1978, Purchase Requisition 272924 was issued to procure a replacement for the Reactor Vent Board 2B-B normal feed breaker and two additional overload trip units. The purchase requisition and contract classified the items as QA level II, did not impose QA requirements on the supplier, did not require any quality documentation from the supplier, and determined that 10CFR21 was not applicable to this procurement. The receiving inspection consisted of a comparison of nameplate data to the purchase documents.

One of the two overload trip units was withdrawn from stock on September 8, 1979, and installed on the normal feed breaker for the Reactor Vent Board 2B-B under MR 00226. As there is no documented evidence of qualification for the trip unit, it renders the qualification of the feed breaker and the safety-related motor control center it services indeterminate.

4) Auxiliary Relay in 6.9kV Shutdown Logic Relay Cabinet 2A-A (2-PNLB-202-SC-A):

This safety-related logic cabinet was originally purchased in June 1972 under contract 72C02-83403, which required a full QA program and seismic qualification. Contract 78P15-24576, dated September 29, 1979, purchased as QA level II, 15 relays of various styles without imposing QA, 10CFR21, seismic, or quality documentation requirements on the supplier.

One of the replacement relays was withdrawn from stock and installed in the 6.9kV Shutdown Logic Relay Cabinet 2A-A under MR A-019533 on May 28, 1982. As there is no documented evidence of qualification for the relay, it renders the qualification of the safety-related relay cabinet indeterminate.

During research of the contract files for this panel, the original seismic report and additional correspondence was found to indicate problems with the seismic testing of two types of relays, including the type of relay replaced under MR A-019533. Although a letter from TVA indicates acceptance of additional seismic testing of the relays, the report documenting the additional testing could not be located in the document control system during the three week period over which the inspection was conducted but was later obtained from an engineer who had retained a personal copy.

5) Boric Acid Transfer Pump Motor (2-MTRB-62-230-A):

The original pump and motor assembly was supplied under the NSSS contract with Westinghouse. The applicable Westinghouse specification (677123 Rev.0) imposed seismic requirements for the assembly and required the supplier to provide documented proof of compliance to those requirements. Westinghouse provided Quality Release 11358-1, dated January 9, 1974, as documentation that the boric acid transfer pump and motor assembly met the requirements of Specification 677123, Rev.0.

A replacement pump motor was purchased on January 13, 1982 under contract 82P64-323717. Although TVA classifies the motor as Class 1E, the contract identified the purchase as QA level II and did not impose QA, seismic, 10CFR21, or documentation requirements on the vendor. A motor different than ordered was supplied. A TVA substitution evaluation form (SQNP AI-11 Attachment 10, Appendix A Rev. 18) was completed on August 9, 1982, but the engineering concurrence for Class 1E substitutions was marked as not applicable. On January 13, 1983, the replacement motor was installed. As there is no documentation to support qualification of the pump motor, the qualification of the pump motor assembly is indeterminate.

6) Diesel Generators:

The diesel generators were procured under contract 71C61-92652 from Bruce-GM which required manufacturing under a 10 CFR 50, Appendix B quality assurance program and seismic qualification of all equipment. Under RD-579489, dated 6/6/77, to contract 73Z23-65906, TVA procured replacement air starter motors from Ingersoll Rand as QA Level II, no documentation required. TVA has replaced these air starter motors numerous times either with motors procured from Ingersoll Rand or rebuilt by Sequoyah plant personnel. As there is no documentation to support qualification of the replacement or rebuilt air starter motors, the qualification of the air starter motors and the diesel generators in which they are installed is indeterminate.

7) Shutdown Board 2A-A Logic Relay Panel (2-BDA-202-CO-A):

Items selected by the inspection team during the plant walkdown included two Agastat time delay relays from this safety-related Class 1E panel. These were Agastat model number 7012PD relays, serial numbers (S/Ns) 81031110 and 80362242, and are in the control logic circuits for the essential raw cooling water and the component cooling water systems. The markings on these two relays indicate that they were not manufactured as qualified Class 1E components since the model number is not proceeded by an "E." This was subsequently confirmed with the relay manufacturer, Amerace Corp. When contacted after completion of the inspection, Amerace Corp. indicated that the relays, S/Ns 80362242 and 81031110, were manufactured commercial grade during the thirty-sixth week of 1980 and the third week of 1981, respectively.

Although requested during the first week of the inspection, TVA was unable to identify or produce the procurement documents for these relays. However, since these relays are not uniquely identified by TVA as Class 1E devices, the inspection team assumes that they were procured as piece parts, QA Level II, without the imposition of QA or documentation requirements on the supplier.

As there is no documentation to support qualification of the relays and the vendor's records indicate that they were not supplied as qualified devices, the qualification of the relays and shutdown board in which they are installed is indeterminate.

8) Containment Isolation Valve (2-FCV-32-103):

This air operated globe valve is required to close on a Class B isolation signal and is listed in Table 9.3.1-2 of the FSAR as a seismic category I valve. This valve was chosen during the walk-through by the team to trace the procurement chain from an installed condition back to the documentation, and was also included in the sample of items for which the maintenance history

was reviewed. The valve was ordered as an ASME Class 2 valve, seismically qualified, under contract 73C34-83577, dated 3/15/73, from Masoneilan International Inc. The serial number of the installed valve, N00137-21-4, traces back to a valve data package for valve PCV-32-103. However, a valve data ordering sheet could not be located for either PCV-32-103 or FCV-32-103 under contract 73C34-83577.

A review of the maintenance history for this valve indicates that the valve failed its local leak rate test and was repaired under MR A-081518, dated 8/1/83. The MR indicates that the soft rubber seat had been damaged in service and was replaced. The seat was procured QA Level II, no documentation required, under contract 79P14-278023 along with two other items. Because the seat was classified as QA Level II, the receipt inspection did not include characteristics such as material type, cure date, durometer standard, estimated shelf life and storage conditions, all of which should be verified for this type of material.

9) Control Valve (2-FSV-32-1031):

This ASCO solenoid valve, model 206-380-2RVU, serves a control function for air operated containment isolation valve 2-FCV-32-103 and is listed in Table 9.3.1-2 of the FSAR as a seismic category I valve. This valve was chosen during the walk-through by the team to try to trace from the installed condition back through the procurement chain to the documentation. The valve (serial number 68337K), was purchased under contract 82PK1-330545, dated 4/8/82, from Mills & Lupton Supply Co. as QA Level II, no documentation required. Also purchased as QA Level II under this contract were replacement coils and repair kits for this model valve. Although no documentation was requested and the receipt inspection report for the valve had the "Certifying Document No." block marked "N/A," a Certificate of Compliance for the valve certifying its compliance to IEEE-323 and IEEE-344 was in the contract file.

c. Conclusions Drawn from Inspection Findings

10 CFR 50, Appendix B, Criterion XVI requires that conditions adverse to quality be promptly identified and corrected. However, although NSRS Report R-84-17-NPS, dated March 12, 1985, pointed out that TVA was using commercial grade items as basic components without determining the effect on the safety function of the component or system in which it was installed, TVA has not initiated an effort to identify equipment which may be in nonconformance with seismic or environmental qualification requirements as a result of this practice.

Sequoyah personnel have started a review of Class 1E replacement devices for seismic qualification. However, the effect of using commercial grade piece parts on seismically and environmentally qualified devices had not been considered at the time of the inspection.

The method used by TVA to identify their sample of 50 devices resulted in the majority of the sample being instrumentation items such as level, pressure, or temperature switches. TVA's selection process started with equipment identification numbers that were categorized as Class 1E, 10 CFR 50.49, or had a safety train designation. However, TVA's program only identifies "those devices qualified as a whole" as Class 1E and does not consider components or parts within a device as being Class 1E or affecting the qualification of the device. As a result, many safety-related electrical items such as circuit breakers, relays, and solenoid coils are not classified as 1E and therefore were not included in TVA's original sample.

TVA management stated that they intended to look at the 238 MRs identified as involving Class 1E device replacements rather than only the sample of 50 as originally intended. This would still not account for the piece part replacements for safety-related electrical or mechanical equipment which comprise the majority of the replacements performed.

The concerns identified above constitute examples which support Potential Enforcement Finding 50-327, 328/86-61-01, Inadequate Corrective Action to Previously Identified NSRS Procurement Findings.

7. Record Retrievability

At the start of the inspection approximately 30 components were identified by the inspectors during a plant walk-through to determine if the installed hardware was traceable back to the required procurement documents. This was in addition to the equipment documentation requested as a result of the review of MRs.

In addition to examples 7.a and 7.b below, which involve record retrievability, paragraphs 6.b.2), 6.b.4), 6.b.7), and 6.b.8) above provide examples of record retrievability in addition to procurement problems.

a) Residual Heat Removal Isolation Valve (2-FCV-74-1):

MR A-10737, for the fabrication and installation of sixteen 1-3/4" x 8" bonnet studs, was issued 8/20/83. Although no procurement deficiencies were noted, a record retrieval or accuracy problem was identified. The MR required a visual examination of the studs. Inspection report R-0686, dated 8/20/83, referenced visual inspection procedure N-VT-1, Revision 4. During the course of the inspection this revision of the procedure was requested, but TVA was unable to produce it.

b) Residual Heat Removal (RHR) Pump Motor (2-PMP-74-10):

MR A-080564, dated 3/15/83, requested that the 2A-A RHR pump motor shaft material be verified. The method chosen to identify the shaft material was the application of a copper sulfate solution, which in combination with visual observation, can be used in making a qualitative decision as to whether the material is carbon steel or stainless steel. Sequoyah Nuclear Power personnel performed the test and determined that the material was stainless steel.

When the inspector requested the procedure for performing the test, TVA could not produce it. As a minimum, there should have been written instructions concerning the strength of the copper sulfate solution, the length of time between application and evaluation, temperature of the material prior to application, method of application, and evaluation criteria.

Several completed Power Stores material receipt inspection packages that were ready to be micro-fisched were reviewed. The review revealed that some packages for safety-related components did not contain the required objective evidence of acceptability, i.e., manufacturer's Certificate of Conformances and the component functional test data sheets. Discussions with Division of Nuclear Engineering (DNE) personnel determined that separate organizational files of these documents were kept by DNE. Therefore, several areas would have to be researched before the required objective evidence of acceptability for an item was available.

As an example, package #839407, for eight nuclear grade Agastat time delay relays (Models E7012/E7022) procured by DNE, contained a closed QC inspection report that accepted the relays based on a handwritten note from DNE stating that DNE intended to accept the manufacturer's C of Cs. The required manufacturer's functional test data sheets were not in this package, but TVA personnel stated that the test data sheets were in a DNE file. When asked how the equipment was controlled or tracked until DNE approved the C of Cs, Sequoyah personnel stated that no additional controls were placed on the equipment and no tracking system existed. If DNE eventually decided that the C of Cs were not acceptable, the equipment may have already been installed. Similar examples were found in package 838674 for four nuclear grade Foxboro differential pressure transmitters, and package 836394 for Atwood and Morrill ASME Section III diesel generator plug valves.

The lack of objective evidence of acceptability in the closed procurement and receipt inspection packages appears to have been caused by the following items. (Items a) and b) were addressed in the NSRS reports.)

a) Failure to establish a procedure, including acceptance criteria, for closed package review to ensure that the necessary documents are included in each package.

b) Lack of adequate TVA organizational coordination, communication and/or procedures to provide one central focal point for review and storage of quality related documents.

c) Failure to establish a tracking system to assure receipt inspection packages eventually contain all necessary documentation or a reference to where the documentation is located.

d) Failure to adequately review and approve the completed QC inspection reports by the QC inspectors immediate management.

10 CFR 50, Appendix B, Criterion XVII requires that records be identifiable and retrievable. ANSI N45.2.9, paragraph 6.2, requires that records be retrievable without undue delay. However, records for installed equipment which do not have a unique plant identification number (i.e., piece parts) could not be identified or retrieved during the three week period between the time the inspection started and the exit interview. Additionally, certain records could not be located through the record retrieval system and were located only by contacting an engineer in the Division of Nuclear Engineering who had a personal copy.

This problem was previously identified in NSRS Reports I-83-13-PS and R-85-07-NPS and provides an additional example of TVA's failure to take adequate corrective action to NSRS findings. The concerns identified above contributed to Potential Enforcement Finding 50-327, 328/86-61-01, Inadequate Corrective Action to Previously Identified NSRS Procurement Findings.

8. Receipt Inspection

Receipt inspection packages for both past and present procurements and the associated versions of Administrative Instruction AI-11, "Receipt Inspection, Handling, and Storage of QA Materials, Substitutions, and QA Level/Description Changes," were reviewed. AI-11, Attachment 1, "Materials Receipt Inspection Report," is a listing of types of inspections such as shipping damage, markings, cleanliness, dimensional, electrical insulation, etc., with three check-off columns, "Yes," "No," and "N/A." Paragraph 2.4 of AI-11 states that the inspector is responsible for determining which of the twenty inspection requirements apply. Paragraph 2.4.1 of AI-11 states that if any inspections in addition to the twenty listed are needed, the originator of the purchase request is responsible for preparing appropriate inspection instructions. However, AI-11 does not contain or reference any detailed inspection or test procedures which would tell receipt inspectors what dimensions or any other characteristic to check, what tools or equipment should be used, or what the acceptance criteria are. The inspection report does not provide any space for recording the results of any inspections other than "Yes," "No," and "N/A."

10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality be prescribed by documented instructions or procedures and include appropriate quantitative or qualitative acceptance criteria for determining that activities have been satisfactorily accomplished. Contrary to the above, Sequoyah has not established detailed receipt inspection procedures which provide appropriate guidance to the receipt inspectors. This lack of instructions contributed to the problem discussed in Section 7 of this report concerning closing of incomplete receipt inspection folders without a tracking system to assure that C of Cs are eventually approved.

This problem was previously identified in NSRS Report R-84-1-17-NPS and provides an additional example of TVA's failure to take adequate corrective action to NSRS findings. The concerns identified above contributed to Potential Enforcement Finding 50-327, 328/86-61-01, Inadequate Corrective Action to Previously Identified NSRS Findings.

U.S. NUCLEAR REGULATORY COMMISSION

Report No.: 50-344/85-025

Docket No.: 50-344

License No.: NPF-1

Licensee: Portland General Electric Company
121 S. W. Salmon Street
Portland, Oregon 97204

Facility Name: Trojan

Inspection At: Rainier and Portland, Oregon

Inspection Conducted: August 9 and 12-16, 1985

Inspectors: _____ 9/25/85
E. W. Merschoff, Chief, Reactive Inspection Section (RIS) Date

_____ 9/25/85
E. T. Baker, RIS Date

_____ 9/25/85
R. P. McIntyre, Special Projects Inspection Section Date

_____ 9/25/85
K. R. Naidu, RIS Date

Approved by: _____ 9/25/85
Gary G. Zech, Chief, Vendor Program Branch Date

Consultant: P. Farron, NEC

SUMMARY

Inspection during the period August 9 and 12-16, 1985 (Report No. 50-344/85-025)

Areas Inspected: Five inspectors from the Vendor Program Branch, Office of Inspection and Enforcement conducted an announced inspection of the effectiveness of the information exchange between vendors of safety related equipment and the licensee. The licensee's program for receipt, evaluation, and implementation of actions determined appropriate for vendor technical information was reviewed. Safety related purchase orders were reviewed to ascertain whether they contained appropriate quality and technical requirements.

The inspection involved 183 hours by four (4) NRC inspectors and one (1) consultant.

Results: Of the areas inspected, two potential violations of NRC requirements were identified. Potential deficiencies were identified in the areas of procurement of safety related equipment and components, and implementation of an effective Part 21 reporting system. In addition, weaknesses were noted in the areas of vendor interface and control of post modification testing.

DETAILS

1. PERSONS CONTACTED

Portland General Electric Company (PGE)

*P. Yundt, General Manager - Technical Functions
*R. Snyder, Administration Supervisor
*A. Roller, Manager, Electrical Engineering
*G. A. Zimmerman, Manager, Nuclear Regulation Branch
*M. H. Malmros, Senior Engineer, Nuclear Safety Branch
*J. Schweitzer, Manager, Quality Assurance Engineering
*H. E. Rosenbach, Supervisor, Material Control, Trojan
*J. Lentsch, Manager, Nuclear Safety and Regulation
*R. Schmidt, Manager, Operations and Maintenance, Trojan
*R. Steel, Manager, Nuclear Plant Engineering
*A. Olmstead, Manager, Quality Assurance
J. Piluso, Supervising Electrical Engineer
M. Cooksey, Supervising Electrical Engineer
R. Greninger, Senior Electrical Engineer
F. Krall, Electrical Engineer
K. Larsen, Electrical Engineer
P. Meyer, Electrical Engineer
D. Walters, Plant Electrical Engineer, Trojan
D. Bennett, Supervisor, Electrical and I&C, Trojan
S. Bauer, Licensing Engineer, Trojan

Nuclear Regulatory Commission

*B. Grimes, Director, DQAVT, OIE
*R. Pate, Chief, Reactor Safety Branch, Region
*G. Kellund, Resident Inspector

*Denotes those individuals who attended the exit meeting on August 16, 1985.

2. BACKGROUND INFORMATION

The purpose of this inspection was to ascertain the effectiveness of the Licensee/Vendor Interface by reviewing the implementation of PGE procedures that identify, track, and resolve vendor and NRC identified deficiencies. This review was accomplished by tracking several specific deficiencies through PGE's review and implementation process, and by reviewing safety related procurement actions taken to support modifications performed during the most recent outage. In addition, Part 21 reporting procedures, control of nonconforming material, and post modification testing were reviewed.

3. PROCUREMENT

a. Program Requirements

PGE procurement procedures and practices were reviewed with respect to safety related procurements made to support safety related work that was performed during the last outage. In particular, the following safety related procurement actions were reviewed in detail.

Table 1

Component	System	RDC #	QA Code	Vendor
Agastat	CCW	83-019	D	Control Products
Control Relay	AFW	83-025	D	General Electric
Control Switch	AFW	83-025	D	General Electric
Pressure Recorder	RCS	77-155	D	Branom Instruments
RTB Shunt Trip	RTS	83-042	D	Westinghouse
Isolation Limit Switch	AFW	84-108	D	Conax
Flow Switch	AFW	84-105	D	ITT Barton
Square Root Extractor	AFW	84-044	D	Branom Instruments
Circuit Breaker	RHR	83-009	D	Brown Boveri
MSIV Isolation Solenoid	MS	84-106	D	Steam Supply & Rubber Co.

Revision 10 of the PGE Nuclear Quality Assurance Program, and the specific implementing procedure, NPEP No. 300-1, "Procurement Procedures for Nuclear Plant Modifications and Additions," establish the methods used by PGE to procure safety related equipment and components. These methods must be consistent with PGE's commitments to the NRC in this area. Specifically, PGE has committed to meet the requirements of ANSI 45.2.13-1976 and Regulatory Guide 1.123 (Rev. 1) "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" with the following two exceptions:

(1) Notification points will not be used for all procurement activities.

(2) Quality-related items and services procured from contractors for which a quality assurance program is required, but has not been reviewed or approved may be utilized by PGE with the specific written approval of the Nuclear Quality Assurance Department.

b. Program Implementation

With regard to implementing these requirements, PGE has established three procurement quality level codes, A, C, and D. Quality level code D, as defined in NPEP 300-1, is intended to be used for, in general, items of relatively simple fabrication, such as O-rings, relief valve springs, or piping. For quality level code D items, PGE has elected to use the method of procurement described in ANSI 45.2.13, Section 10. This method allows the use of certificates of conformance (C of C) as the sole basis for accepting an item provided that the item is of simple design, involves standard materials, processes, and tests, and that a means has been provided to verify the validity of the supplier's certificates and the effectiveness of the certification system, such as the performance of audits of the supplier or independent inspection or test of the item.

Although PGE procured each of the safety related items listed in Table 1 as quality level code D items, with acceptance of the item based solely on the C of C, insufficient action had been taken by PGE to verify the validity of the suppliers' C of Cs. In particular, neither audits, surveys, nor testing to verify the validity of the C of Cs had been conducted. Currently, PGE relies on NQAP 114, "Verification of Vendor Certificates of Conformance" to satisfy the requirements of ANSI 45.2.13 and Regulatory Guide 1.123 for verifying the validity of supplier certificates and the effectiveness of the certification system. However, NQAP 114 only requires the QA Department to verify the certification system of a sample of the vendors from which safety related procurements have been made. Further, the QA Department is only required to interview the C of C signer, rather than to perform the required verification of the effectiveness of the certification system by auditing such areas as material control, nonconformance control, and inspection.

Consequently, PGE's safety related procurement activities are inadequate in that they fail to meet their commitments to ANSI 45.2.13 and Regulatory Guide 1.123 with respect to verifying the validity of their suppliers' certificates and the effectiveness of their suppliers' certification program.

c. NRC Audit of PGE Suppliers

In order to determine if any hardware problems may exist as a result of PGE's improper procurement practices, two of the vendors from which PGE had procured safety related components were inspected by the NRC

Vendor Program Branch. These two vendors and the components they have supplied (by way of a middle-man distributor) to PGE are listed below:

Vendor	Component Supplied
Rochester Instrument Systems, Inc.	Square Root Extractor for AFW System
Texas Instruments	Reactor Coolant System Pressure Recorder

The results of these inspections are documented as Inspection Report Numbers 99900222/85-01 (Rochester Instrument Systems) and 99901028/85-01 (Texas Instruments). A summary of the results is provided below.

(1) Rochester Instrument Systems (RIS)

Audit findings at RIS revealed that RIS had an effectively implemented quality assurance program that was consistent with the requirements of 10 CFR 50 Appendix B. Although PGE had neither verified that RIS had an effective quality program nor imposed any specific quality requirements on RIS in the purchase order, RIS was familiar with these requirements based on their dealings with other nuclear utilities and implemented their nuclear quality assurance program on the work performed to execute PGE's order.

(2) Texas Instruments (TI)

The audit findings at TI revealed that TI did not, nor did they ever, have a documented 10 CFR 50 Appendix B quality assurance program. This finding was confirmed previously by Florida Power and Light and the Tennessee Valley Authority, both of which had audited TI and declined to approve them as a source of safety related equipment due to inadequacies in TI's quality assurance program. Nevertheless, PGE accepted a safety related Reactor Coolant System Pressure Recorder which is relied on for monitoring conditions in the Reactor Coolant System during normal operating conditions and anticipated operational occurrences (as stated in section 7.5 of the Trojan FSAR) from TI. This acceptance was based solely on a TI C of C which contained the following disclaimer:

> "However, such products are not sold for, nor is the foregoing certification a representation that the products are suitable for applications essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, or otherwise essential in operation of the

reactor or preventing a significant release of radioactive material to the environment."

On inquiry, representatives from TI were unable to explain the meaning of the disclaimer. They indicated that they were new to the nuclear business and simply did not have a good feel for the requirements and responsibilities of a supplier. Consequently, their lawyers had recommended use of the disclaimer.

This area has been identified as a potential violation of the requirements of 10 CFR 50 Appendix B (344-85-025-01).

4. 10 CFR PART 21

a. Program Requirements

Nuclear Division Procedure (NDP) 700-4 Revision 1 dated May 17, 1985 establishes the policy and procedures for assuring compliance with 10 CFR 21 to evaluate and report defects in basic components that could cause a substantial safety hazard. Review of this procedure showed significant weaknesses in both adequacy and implementation to the extent that defects identified during the dedication process are not subject to the required review to evaluate their potential significance.

Paragraph 3.2 of NDP 700-4 defines a commercial grade item and states that "A commercial grade item is not part of a basic component until after dedication; i.e., until it is drawn from the warehouse or stock and assigned for use or installation." The item is then dedicated for use as a basic component and becomes subject to the requirements of 10 CFR Part 21. Standard industry practice for dedication of commercial grade items may involve one or more quality control measures such as tests, inspections, or nondestructive examinations to determine that the commerical grade item is free of defects and will satisfy the procurement specification. These tests could reveal inherent defects in the operational characteristics of items which are not detected by visual examinations performed during superficial receipt inspections. However, by PGE's procedure, defects or noncompliances discovered during the implementation of quality control measures are not subject to reporting under the requirements of 10 CFR Part 21. Specifically, paragraph 5.1.D of NDP 700-4 states "Defects or noncompliances found through implementing established quality control measures (e.g., reviews, inspections, tests or non-destructive examinations) which are a result of the work process performed at Trojan and prior to final acceptance of a structure, system, or component for operation, shall not be subject to reporting under 10 CFR Part 21." Thus, any defects revealed during the dedication process are corrected without the benefit of the nonconformance process or the evaluation and subsequent reporting pursuant to 10 CFR Part 21.

b. Program Implementation

The NRC inspector reviewed PGE's implementation of 10 CFR 21 reporting requirements by reviewing Request for Design Change (RDC) package 84-025. This RDC, which was approved on January 22, 1985, involved the replacement of two 400 ampere trip units with 225 ampere trip units in 480 volt circuit breakers 48 and 26, located on Motor Control Centers B 23 and B 24, respectively.

The following is a summary of the actions taken to accomplish RDC 84-025.

(1) Purchase Order N-29326 dated March 18, 1985 was placed with Gould Electric, Gresham, Oregon, to supply the following:

(a) 4 Class 1E circuit breaker trip units (ITE Catalog JL3-T225) rated for 600 volts and 225 amperes.

(b) Certification of seismic qualification.

(c) Vendor certification that the trip units meet the PGE specification requirements.

(2) The trip units were received from Telemechanique (formerly Gould Inc.) with the required C of Cs. They were receipt inspected, accepted, and stored at the warehouse.

(3) In June 1985, the trip units were installed in circuit breakers 48 and 26, and tested. Contrary to both the specification and the C of C, both circuit breakers failed to trip when 225 amperes continuous current was applied.

(4) The PGE electrical engineer, in consultation with ITE, determined that both trip units lacked the required thermal overloads.

(5) ITE agreed to replace the four trip units.

(6) The replacement trip units were installed in the circuit breakers and tested satisfactorily.

Although PGE properly tested these components, determined the cause of the malfunction and corrected it, several weaknesses in the overall program are demonstrated by this example. Specifically:

(1) The PGE purchase order did not impose any specific quality assurance or Part 21 reporting requirements on the vendor.

(2) PGE neither initiated a nonconformance on this item nor were they required, by their procedure, to perform a Part 21 evaluation.

(3) Since neither PGE or ITE reported to the NRC that nonconforming Class 1E trip units were supplied to PGE for use at the Trojan Plant nor did PGE enter this problem in the Nuclear Plant Reliability Data System, this potential problem has not been made known to the industry.

This area has been identified as a potential violation of the requirements of 10 CFR Part 21 (344-85-025-02).

5. CONTROL OF NONCONFORMING MATERIAL

While discussing maintenance practices with one of PGE's on-site engineers, the inspector noticed a number of Pacific Scientific snubbers on a table in the engineer's office. When asked why the snubbers were in his office, the engineer replied that the snubbers had failed during inservice testing and were being sent back to the manufacturer for repair. The inspector noted that there were no QC Hold Tags on the snubbers, no paperwork accompanied the snubbers, nor were the snubbers in a nonconformance hold area.

Revision 1 of Periodic Engineering Test, PET-9-6, "Mechanical Pipe Snubber Inspection," paragraphs 9.1.1 and 9.3.1, require that the test engineer issue a Nonconformance Report (NCR) per Nuclear Department Procedure, NDP-600-1, for snubbers that fail either the inspection or the test. Paragraph 5.3.c of NDP-600-1, Revision 0, states an NCR is not required for nonconforming items being returned to the supplier to restore the item to conformance with the technical description. Paragraph 5.5.M.1 of NDP-600-1, Revision 0 states that the Trojan Plant QA staff may prepare and install a QC Hold Tag at the discretion of the QA Supervisor. However, since an NCR is not required, there is no documented method for informing the Plant QA staff that nonconforming material exists.

Section 15 of PGE's Nuclear Quality Assurance Program (NQAP), Revision 10 requires that when feasible, nonconforming items be clearly identified with Quality Control Hold Tags and segregated to indicate their unacceptable status. Section 15 also provides two methods of documenting nonconformances, Maintenance Requests (MRs) and Nonconformance Reports. The principal difference between the use of an MR and an NCR is that an MR is used when the nonconforming item can be restored to the original design requirements using a rework process at the site, and an NCR is used in all other cases. Note that there may be no differences in the actual nonconformance, only in Trojan's ability to perform the rework.

The fact that the method of "reworking" the nonconforming items determines the method of documentation, regardless of the type of nonconformance, brings up an apparent inconsistency. Paragraph 15.2.3 of the NQAP requires that NCRs be evaluated to determine if a significant condition adverse to quality, an unresolved safety question, or defects resulting in substantial safety hazards exist. However, no such requirements exist under paragraph 15.2.2, "Maintenance Requests," even though the actual nonconformance may be the same.

Additionally, paragraph 5.3.c of NDP-600-1 does not require NCRs to be written on nonconforming items being returned to the supplier for rework. Without an NCR or MR, there appears to be no method for tracking nonconforming items returned to suppliers for rework and, therefore, no way for these nonconformances to be evaluated for significant conditions adverse to quality, unreviewed safety questions, or defects resulting in substantial safety hazards.

The fact that 14 nonconforming PSA mechanical snubbers and 11 nonconforming PSA snubbers were not tagged with QC Hold Tags would indicate that the intent of Appendix B and the PGE Nuclear Quality Assurance Program is not being fulfilled by NDP-600-1. This is considered an Unresolved Item (344-85-025-03).

6. EMERGENCY DIESEL GENERATOR (EDG) MAINTENANCE

Trojan's EDGs are General Motors (GM) diesels which were originally supplied by the now defunct Bruce. Morrison-Knudsen has taken over the role of informing Bruce customers of problems with, and maintenance requirements for, GM diesels. The inspectors reviewed Trojan's maintenance procedures for the EDGs, compared the Maintenance Instructions (MI) and Power Pointers (PP) received by PGE with those issued by GM, reviewed PGE's evaluation and implementation of MIs and PPs, and performed a valve line up verification of the cooling system for the number one diesel in the east diesel room.

a. Valve Line Up Verification

An auxiliary operator and a technician from the maintenance department assisted with the valve line up check. The operator provided a copy of the checklist used for operations, OI-5-1, to assist in performing the valve line up. All valves were in the correct position. The only discrepancy observed was that all the valves listed on the checklist had an alpha-numerical designator, but none of the valves were marked or tagged with the appropriate designator, making identification of the valves difficult.

b. Maintenance

Based on a review of the maintenance procedures, a comparison between them and the MIs and PPs issued by GM, and discussions with maintenance personnel, it was apparent that PGE was not receiving MIs and PPs, and that PGE had not established an agreement with Morrison-Knudsen to receive MIs and PPs issued by GM. There was no record to show which MIs and PPs had been received, evaluated, and implemented by PGE. A review of the maintenance manual for the EDGs revealed that PGE was using MI 1750, "Scheduled Maintenance for Stationary Power Units With Turbocharged Engines," dated October 1973, which had been superceded by Revision G of MI 1728, "Scheduled Maintenance Program for Domestic Stationary Power Units With Turbocharged Engines," issued in 1984. The latest MI filed with the maintenance manual was dated 1978, although numerous MIs have been issued since that time. No PPs were found during the review.

Several differences were noted between MI 1750 (the basis for PGE's maintenance schedule) and the actual maintenance procedure, MP-12-7. MI 1750 recommended that the exhaust manifold flange bolts be torqued every six months while MP-12-7 required it only once every three years; MI 1750 recommended that the governor oil be changed every six months while MP-12-7 required it only annually; and MI 1750 recommended a test on the temperature differential between lube oil and cooling water into the engine every two years while MP-12-7 did not address the test. PGE maintenance personnel acknowledged that a written evaluation of the differences between the MI and the MP had not been performed.

The differences between the most recent MI on maintenance (MI 1728, Revision G) and MP-12-7 are: MI 1728 recommends a monthly lube oil analysis while PGE performs the analysis only quarterly, and MI 1728 recommends an annual check of protective devices and circuit breakers, while PGE does not require these checks to be performed periodically.

As a result of PGE's failure to maintain an effective interface with their EDG supplier, the EDG preventive maintenance program at Trojan is predicated on information that is at least 11 years and eight revisions out of date.

7. TERRY TURBINES

Two issues involving Terry Turbines were investigated during the inspection: (1) inspection of shaft bearing anti-rotation pins and (2) whether the problems associated with conformance to the seismic qualification report identified at Palo Verde Nuclear Station exist at the Trojan plant.

On May 19, 1983, PGE's maintenance staff performed maintenance in response to a Part 21 report issued by Terry Corporation that notified licensees of the possibility that anti-rotation pins for locating shaft bearings may have been left out when the turbine was assembled. During disassembly and inspection of the turbine by PGE, it was noted that the inboard bearing anti-rotation pin had been installed during factory assembly, but the anti-rotation pin was missing from the outboard bearing. As advised by the Terry Corporation, PGE installed the missing anti-rotation pin.

During the course of the inspection, PGE received a letter from the Terry Corporation informing them that Terry had recently completed a detailed seismic analysis of the type GS turbine which was performed as a result of deficiencies identified at Palo Verde Nuclear Station. The results of the analysis were that the pedestal to casing bolting, governor end pedestal to base plate bolting, and coupling end pedestal to base plate bolting may require replacement and/or preloading to specific levels. In addition, the bolting material may not have the minimum strength necessary to carry loads imposed by a seismic event. The inspectors and licensee personnel performed an inspection of the Terry Turbine and noted that the required taper pins were missing from the coupling pedestal to base plate joint and that there were no markings on the bolts or studs, as required, to indicate the grade of fastener. The NRC inspectors discussed these

findings with the licensee and, as a result, the licensee performed a detailed review and determined that in addition to the problems listed above, it was not possible to determine if the governor pedestal to base joints had the required shear pins installed.

At the completion of the inspection, PGE was still evaluating the problem. PGE's engineering staff indicated that a reanalysis using Trojan's actual ground acceleration, 0.25g, rather than the 0.67g used by Terry in their analysis, would resolve the concern and alleviate the need for hardware changes. The reanalysis is pending PGE's receipt of specific seismic response data from Terry Corporation. However, this reanalysis does not negate the fact that PGE accepted and installed equipment which did not conform to the design analyzed in the seismic analysis. This is considered an Unresolved Item (344-85-025-04).

8. TECHNOLOGY FOR ENERGY CORP. (TEC) ACOUSTIC MONITORS

TEC supplies acoustic monitors for use in monitoring conditions downstream of relief valves to indicate when the valve may be leaking. The inspectors examined two areas of concern involving these acoustic monitors: (1) whether or not PGE's engineering staff was aware of and evaluated a Part 21 report submitted by TEC concerning the failure of TEC Model 914-1 valve flow monitor modules to reset after valve cycling, and (2) whether or not Trojan's maintenance procedures accounted for the limited qualified life associated with TEC components.

a. TEC Part 21 Report

PGE had received the July 19, 1985 TEC Part 21 report, had reviewed it, had determined it was applicable to Trojan equipment, and was in the process of taking the appropriate actions.

b. Limited Life of TEC Components

A review of correspondence and maintenance procedures revealed that PGE engineering staff had performed an independent calculation of the qualified life of TEC components. TEC had based their calculations on an activation energy of 0.65eV, while PGE's calculation was based on an activation energy of 1.17eV. PGE justified the use of the higher activation energy by referencing EPRI report NP-1558, "A Review of Equipment Aging Theory and Technology," and previous Bechtel calculations which had used the same activation energy for the limiting component, cross linked polyethylene insulation on the Brand Rex cable to the sensor. The use of a higher activation energy resulted in increased qualified life for the cable, from 4 years to 26.8 years. Based on the fact that the acoustic sensor contained no organic components, its qualified life was also increased to 26.8 years.

No deficiencies were noted in this area.

9. VENDOR/UTILITY INTERFACE

The inspectors conducted a review of PGE's vendor information evaluation and tracking system with respect to the receipt and processing of technical information provided by Westinghouse (W), INPO, the NRC, and vendors of safety related equipment and components.

Starting in January 1985, PGE instituted a new system for tracking vendor correspondence. All vendor correspondence is received by the Nuclear Plant Engineering Department and entered into a computerized filing system by categories listed in the Trojan file index. These documents can be viewed on microfiche.

In this same timeframe, a computerized Commitment Tracking List (CTL) was started to track items of potential interest or impact on plant operations and maintenance. These items include Westinghouse Technical Bulletins, INPO Significant Event Reports (SERs) and Significant Operating Experience Reports (SOERs), NRC Information Notices and Bulletins, Vendor 10 CFR 21 Reports, and general vendor correspondence.

PGE Nuclear Safety and Regulation Procedure NSRP No. 330-2, "Operating Experience Review Program Administration," and PGE Trojan Nuclear Plant Administrative Order-AO-9-4, "Operating Experience Review Program," define the requirements for review, evaluation, and implementation of vendor technical information. This review and evaluation is done through an Operational Assessment Review (OAR). An OAR number is assigned and tracked on the CTL by either subject or OAR number.

From 1981 to 1984, PGE had another formal tracking system, the Required Action Tracking System (RATS) which was also a computerized data base maintained by the Trojan Operational Assessment group. This system functioned almost the same as the current system, except it did not track by OAR number for all vendor correspondence.

Prior to 1981, PGE had no formal system for the receipt, evaluation, and implementation of actions initiated in response to vendor technical information. For example, PGE had virtually no documentation to indicate that a review and evaluation were performed either for Westinghouse Technical Bulletins affecting Trojan which were issued prior to 1981, or for Westinghouse Data Letters issued prior to 1982. PGE has been in the process of reviewing these older Technical Bulletins and Data Letters for applicability to Trojan. PGE stated in Memorandum DRIC-037-85 that documentation and completion of the required actions with respect to Westinghouse Bulletins and Data Letters is to be completed by September 30, 1985.

a. Evaluation of Information Contained In Westinghouse Technical Bulletins

A sample of W Technical Bulletins was selected to review the licensee's receipt, evaluation, and implementation of the information in the bulletins. Technical Bulletins are an advisory notice of recent technical developments pertaining to the installation or operation of

W supplied Nuclear Plant Equipment. Recipients are advised to evaluate the information and recommendations, and to initiate action when appropriate. The following is a brief summary of the Technical Bulletins reviewed by the inspectors.

(1) TB-80-11 - High Head Safety Injection (HHSI) Pump Flow Verification

Deficiency - Degradation of Centrifugal Charging Pump flow performance may result in a considerable change in the pump performance curves and exceed the margin allowed for the safety injection system.

Licensee Action - Trojan is monitoring the minimum flow rate on a periodic basis (at least yearly) and is continuing HHSI pump performance monitoring (monthly) per the current inservice testing program. Total runout flow is tested each refueling outage and branch-line balancing is done if a total flow degradation is observed.

(2) TB-83-07 - Containment Pressure and Containment Sump Water Level Systems

Deficiency - During the severe environmental phase (320°F) of qualification testing of the water filled system, extraneous pressure spikes (ringing) was observed on the containment pressure transmitter output. W solved this problem and recommended that any of the subject systems filled with water be drained, dried, and refilled with specially processed DC702 silicone oil.

Licensee Action - Per the W recommendation, all affected level transmitters were filled with DC 702 silicone oil.

(3) TB-04-06 - Reduced Auxiliary Feedwater Flow Rates

Deficiency - A situation occurred where a W plant throttled its auxiliary feedwater (AFW) control valves to the minimum required flow rate (105 GPM) against a steam generator pressure (700 psig) which was considerably lower than the pressure (1082 psig) assumed in the plants loss of normal feedwater analysis. A subsequent hydraulic analysis showed that the AFW flow rate at the higher pressure would have been 27% less than the flow obtained at the lower pressure, and less than that assumed in accident analysis.

Licensee Action - The PGE evaluation documented in OAR-84-121 concluded that Trojan's variable speed AFW pump system will provide sufficient flow under all anticipated conditions. The

AFW pump speed can be increased to maintain 100 psid across the throttle valve. However, the AFW system performance and valve inservice test (POT-5-3) was revised to assure that it is always performed at steam generator pressures greater than 1000 psig in order to be conservative.

(4) TB-84-08 - Inverter Capacitor Connections

Deficiency - During the 7.5 KVA static inverter qualification testing, it was discovered that two out of the three GE ferro-resonant transformers had two fast-on terminals connected to a solder lug of the tuning capacitors. Electrically the inverters will operate properly, but the mechanical connection could separate during a seismic event.

Licensee Action - All transformer units currently installed at Trojan were checked for fast-on terminal connections to solder lugs. All units had the correct connections.

(5) TB-84-11 - Potential Grounding of Ferro-Resonant Transformers

Deficiency - During initial startup testing at a plant in late construction, three W 7.5 KVA static inverters failed during a four month period. After a fault analysis was performed, the nature of the failure was determined to be shorting between the coil and core of one of two inductors connected to the secondary side of the ferro-resonant transformer. The manufacturer determined that all three units failed due to insufficient securing of the center leg of the inductor core.

Licensee Action - All installed transformers have been in operation at Trojan for greater than six months and were not affected by the bulletin. The newer transformers located in the warehouse were given a high potential leakage current test. All test results were satisfactory. This information was documented in OAR-84-154.

b. Centrifugal Charging Pump Speed Increaser

Written correspondence from Westinghouse to PGE, dated February 12, 1985, described a potential problem with the lube oil pump coupling on the centrifugal charging pump speed increaser. This written correspondence was neither a Technical Bulletin nor a Data Letter. In December 1984, two facilities suffered failures of the couplings and the intent of this letter was to notify PGE of a potential problem.

After receipt of the letter, PGE performed an inspection of both charging pump speed increaser lube oil pump couplings for excessive pin wear. After consulting with Nuttall Gear Corp. (the manufacturer)

it was determined that the pin wear at Trojan was satisfactory for continued operation. Maintenance Procedure MOP-5-2 was revised to include an annual inspection of these lube oil pump couplings.

However, during this inspection it was discovered that the inboard coupling half key had almost completely backed out of its keyway. There was no means of capture on the key way. Undetected failure of the speed increaser would render the centrifugal charging pump inoperable. This is a different problem than that identified by W.

Since this speed increaser is the same unit that is used on the Auxiliary Feedwater Pump, PGE inspected the AFW pump coupling as well. This keyway was also machined too long, but the key was tight. A modification was made to stake the key in place as a precaution. This modification was also made on both centrifugal charging pumps to stake the keys in place.

As of the time of the inspection, PGE had not completed their evaluation for reporting per 10 CFR 21 nor had they filed a Licensee Event Report (LER). PGE personnel interviewed indicated an intent to not report this defect pursuant to 10 CFR Part 21. This problem was first identified on April 1, 1985.

On April 4, 1985, W sent out a letter update on the original charging pump speed increaser lube oil pump problem. They stated they had completed an engineering study of the original two plant failures. The results showed that the failure was due to an abnormal wear mechanism, which was most likely the result of a combination of coupling misalignment, excessive lube oil pump torque, and insufficient coupling lubrication. The W Safety Review Committee concluded that this issue is not a safety concern.

Notwithstanding the W determination that the initial problem was not reportable, PGE's failure to perform a timely evaluation of the second potentially serious common mode failure is considered another example of a weakness of PGE's Part 21 program.

10. POST MODIFICATION TESTING

Approximately one month prior to this inspection, an event occurred at the Trojan plant in which both auxiliary feedwater pumps tripped on low pump suction pressure following a reactor trip. The low pump suction trip was installed as part of a plant modification. The post modification testing did not reveal this deficiency because the tests performed did not include a demonstration of two pump operation with the condensate storage tank at the minimum acceptable water level.

As part of this inspection effort, specific post modification tests were reviewed with respect to whether the testing was sufficient to cover all modes of system operation. In addition, a review was performed to determine if the testing would prove or duplicate the testing specified in the C of Cs supplied by the vendors.

The test procedure requirements and acceptance criteria are designated in Detailed Construction Packages (DCPs). Once the DCP testing requirements are received on site, they are transcribed onto Maintenance Requests to be performed by the applicable site department.

A small sampling of As Built Packages (ABPs) and DCPs from the most recent plant outage were reviewed. The ABPs are completed modifications which contain a listing of the testing that was performed and a statement as to the acceptability of the results. Approximately one third of the modifications performed during the outage were closed out and filed as ABPs. The remaining modification packages were in the process of being closed out and the current documentation was contained in the DCPs. The following is a summary of the testing contained in one ABP and two DCPs, and general observations concerning the testing methods and acceptability.

a. RHR System Modification

ABP 84-058 was performed to replace valve motor operators with environmentally qualified units. The valves are safety related and receive isolation signals. The functional test requirements were reviewed in detail and appeared to be acceptable. Maintenance Request (MR) 85-181 properly summarized the package test requirements. The Testing/Installation Results Section of MR 85-181 stated that applicable portions of the Periodic Operating Tests were performed and valves timed within the required specifications. No specific test data was provided in the ABP nor were the revision numbers or applicable sections of procedures identified.

b. MSIV Isolation Modification

DCP 84-106 was performed to replace the solenoid-actuated pilot valves for the main steam isolation valves and bypass valves. The functional testing requirements and installation checks for this modification were scattered throughout the package. Field Change Notice (FCN) No. 8 to DCP 84-106 provided the final test requirements to be performed for the modification. MR 85-2101 was the only maintenance request in the DCP and it instructed the plant personnel to perform only a portion of the testing delineated in FCN No. 8. Since the ABP for this modification was not available at the time of the inspection, it was not possible to determine if all the required testing had been performed. This is considered an Unresolved Item (344-85-025-05)

c. AFW Flow Control Modification

DCP 84-104 was performed to replace the auxiliary feedwater system motor operated flow control valves. The DCP included the functional testing and installation checks to be performed and listed the Periodic Operating Tests (POTs) that should be run. The POTs were reviewed during the inspection and it did not appear that they would satisfy all the testing requirements. Specifically, the hi-flow

condition valve closure function was not addressed. The MR for this DCP stated that these POTs and Temporary Plant Test (TPT)-137 should be performed to satisfy the post modification testing requirements. The TPT was not available during the inspection and it could not be determined if the TPT would satisfy the hi-flow condition valve closure function. The ABP for this modification was not complete at the time of the inspection so it was not possible to determine what tests had actually been performed. This is considered an Unresolved Item (344-85-025-06).

d. Control of Testing

During the inspection, five ABPs and DCPs from the most recent outage were reviewed. In all cases it was difficult to determine what tests and installation checks were required and what was actually performed. It was also unclear that the testing and checks identified in the originating DCPs would accurately test all the system functions or were properly transcribed to the plant Maintenance Requests. Additionally, the detailed test results in most cases could not be reviewed because they were not included in the ABPs. The licensee indicated that the process was being reviewed and it was expected that any deficiencies identified would be corrected.

e. Verification of C of C Data

The records in the modification packages were reviewed to determine whether the parts or components procured, accepted, and installed for the modifications were subjected to additional tests at the site to verify the validity of the C of Cs which formed the sole basis for acceptance of the items. The review determined that the testing and installation checks were functional in nature and not intended to verify the validity of the vendor's certification system.

This issue is addressed in more detail in Section 3 of this report and has been identified as a potential violation (344-85-025-01).

11. EXIT MEETING

The inspectors met with Mr. P. Yundt and other staff members on August 16, 1985, to summarize the scope and findings of the inspection.

SELECTED INFORMATION NOTICES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

October 16, 1986

IE INFORMATION NOTICE NO. 86-05, SUPPLEMENT 1: MAIN STEAM SAFETY VALVE TEST FAILURES AND RING SETTING ADJUSTMENTS

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This notice is being provided as a supplement to Information Notice (IN) 86-05 to inform licensees of additional experience with ring setting adjustments of main steam safety valves (MSSVs). It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem from occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Seabrook, 1985 (Synopsis)

In the fall of 1984, Seabrook's MSSVs (manufactured by Crosby) were sent to Wyle Laboratories for testing. The results of the tests showed that adequate disc travel was not attainable with the factory-set ring settings. Following additional testing, the ring settings were adjusted from the factory settings to obtain the required disc travel.

Vogtle, 1986

On May 14, 1986, Georgia Power Company informed the NRC that 5 of the 20 Crosby MSSVs due to be installed on Vogtle 1 were tested to determine the appropriateness of their as-shipped ring settings (nozzle ring setting of +150 and guide ring setting of -45). The valve settings did not achieve specified relieving capacity. The typical relieving flow was 75% of the specified value. Subsequently all 20 of the valves were shipped to the Crosby high-flow test facility to be tested and adjusted as required to achieve their design relief capacity. Final ring settings are summarized in Attachment 1. Note that these settings in many instances differ considerably from the as-shipped settings. Unit 2 valves are due to be tested in the future.

8610100107

Millstone 3, 1986

On July 29, 1986 while inspecting the A-bank MSSVs, personnel found that the upper and lower adjusting rings were not set to the vendor's (Dresser) recommended settings. After an investigation, it was determined that the vendor had not reset the rings following functional testing because of a procedural inadequacy. Millstone 3 personnel reset all rings to the recommended settings. (Licensee Event Report 50/423-86/043).

Susquehanna 1 and 2

Pennsylvania Power and Light personnel recognized the 6R10 designation used by Seabrook to describe their valves (IN 86-05) as being the same as the Crosby safety/relief valves (SRVs) installed at Susquehanna Units 1 and 2; however, the designation 6R10 is not a model number, but the inlet, orifice, and outlet size of the valve. These SRVs are not the same model as the Seabrook or Vogtle valves, but they do have adjusting rings which determine the capacity and blowdown of the valves. Because of this, the information in IN 86-05 and this supplement may be applicable to boiling-water reactors.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: Mary S. Wegner, IE
(301)492-4511

Attachments:
1. Summary of Final Rings Settings for Vogtle 1
2. List of Recently Issued IE Information Notices

SUMMARY OF FINAL RING SETTINGS FOR VOGTLE 1

Valve No.	Ring Settings	
	Nozzle	Guide
3001	-130	0
3011	-130	-59
3021	-125	-48
3031	-165	-50
3002	-100	-80
3012	-100	-50
3022	-100	-50
3032	-130	-100
3003	-130	-50
3013	-130	-50
3023	-130	-50
3033	-130	-50
3004	-130	-50
3014	-100	-50
3024	-130	-50
3034	-130	-50
3005	-100	-50
3015	-130	-50
3025	-130	-50
3035	-100	-50

Baker

SSINS No.: 6835
IN 86-21

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

March 31, 1986

IE INFORMATION NOTICE NO. 86-21: RECOGNITION OF AMERICAN SOCIETY OF MECHANICAL ENGINEERS ACCREDITATION PROGRAM FOR N STAMP HOLDERS

Addressees:

All nuclear power facilities holding an operating license (OL) or a construction permit (CP) and all recipients of NUREG-0040 (white book).

Purpose:

This information notice is provided to licensees, construction permit holders, and vendors to inform them of the NRC's recognition of the American Society of Mechanical Engineers (ASME) Accreditation Program for holders of N, NPT, NA, and NV stamps and Certificates of Authorization. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Discussion:

In March 1981, representatives of the NRC, ASME, and the National Board of Boiler and Pressure Vessel Inspectors (NB) signed an Exchange of Correspondence that outlined an agreement to work toward establishing a single set of quality requirements and recognition by the NRC of ASME's Accreditation Program for N, NA, NPT, and NV stamps and the associated Certificates of Authorization. Quality System Certificates issued to Material Manufacturers (MM) and Material Suppliers (MS) are excluded from this recognition. The primary reasons for developing the Exchange of Correspondence were to eliminate overlapping requirements and to allow NRC licensees and their subcontractors to use the results of the ASME/NB nuclear accreditation and inspection system as an acceptable method for satisfying NRC requirements for evaluating the programmatic adequacy of their suppliers' documented quality assurance (QA) programs.

In a meeting on December 10, 1985, representatives of the NRC, NB, and ASME agreed that based on the requirements contained in the 1986 Edition of the ASME Boiler and Pressure Vessel Code all of the conditions established in the Exchange of Correspondence as prerequisites for NRC recognition of the Accreditation Program had been met. On this basis, the NRC has sent letters (see attached) to the National Board and ASME recognizing the holding of an

ASME N, NA, NPT or NV Certificate of Authorization as evidence that the holder of the Certificate of Authorization has a documented QA program that meets the requirements of Appendix B to 10 CFR 50. In accordance with the Exchange of Correspondence, the NRC will make the appropriate changes to Regulatory Guides 1.28 and 1.33 in the next revision. Licensees and their subcontractors, when assessing whether a potential supplier has an acceptable documented QA program, may rely on the fact that ASME has surveyed the supplier and issued a Certificate of Authorization of appropriate scope and for the desired location without performing any additional evaluation of the supplier's QA program.

Licensees and construction permit holders that take credit for the ASME Accreditation Program may have to revise their QA Program. They are reminded that 10 CFR 50.54(a)(3) and 50.55(f)(3) set forth the requirements for making changes to QA programs.

It should be noted that NRC's recognition applies only to the programmatic aspects of the ASME Accreditation Program. Licensees, construction permit holders, and their subcontractors are still responsible for ensuring that the supplier is effectively implementing the approved QA program.

No specific action or response is required by this information notice. If you have any questions regarding this notice, please contact the Regional Administrator of the appropriate NRC regional office or the technical contact listed below.

Edward L. Jordan, Director
Division of Emergency Preparedness and
Engineering Response
Office of Inspection and Enforcement

Technical Contacts: E. T. Baker, IE
301-492-4874

G. G. Zech, IE
301-492-9663

Attachments:
1. Letter to ASME
2. List of Recently Issued IE Information Notices

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

March 18, 1986

Mr. Melvin R. Green, Managing Director
Codes and Standards
American Society of Mechanical Engineers
345 East 47th Street
New York, New York 10017

Dear Mr. Green:

SUBJECT: NRC ENDORSEMENT OF ASME'S ACCREDITATION PROGRAM

At the December 10, 1985 meeting of the NRC/ASME/NB Coordinating Committee it was agreed that all conditions established in the Exchange of Correspondence as prerequisites for NRC recognition and endorsement of the ASME Accreditation Program had been met. At that meeting it was also agreed that the NRC would publish an information notice communicating the NRC's recognition and endorsement of the Accreditation Program. Enclosed is a copy of the information notice which is being sent to all licensees and to other parties on the information notice distribution list.

Sincerely,

James M. Taylor, Director
Office of Inspection and Enforcement

Enclosure:
Information Notice

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

March 18, 1986

Mr. Donald J. McDonald, Executive Director
National Board of Boiler & Pressure Vessel
Inspectors
1055 Crapper Avenue
Columbus, Ohio 43229

Dear Mr. McDonald:

SUBJECT: NRC ENDORSEMENT OF ASME'S ACCREDITATION PROGRAM

At the December 10, 1985 meeting of the NRC/ASME/NB Coordinating Committee it was agreed that all conditions established in the Exchange of Correspondence as prerequisites for NRC recognition and endorsement of the ASME Accreditation Program had been met. At that meeting it was also agreed that the NRC would publish an information notice communicating the NRC's recognition and endorsement of the Accreditation Program. Enclosed is a copy of the information notice which is being sent to all licensees and to other parties on the information notice distribution list.

Sincerely,

James M. Taylor, Director
Office of Inspection and Enforcement

Enclosure:
Information Notice

SSINS NO: 6835
IN 86-21, Supplement 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

December 4, 1986

IE INFORMATION NOTICE NO. 86-21, Supplement 1: RECOGNITION OF AMERICAN SOCIETY OF MECHANICAL ENGINEERS ACCREDITATION PROGRAM FOR N STAMP HOLDERS

Addressees:

All nuclear power facilities holding an operating licensee or a construction permit.

Purpose:

This notice is provided to licensees and construction permit holders to clarify the information provided by IN 86-21 on the NRC's recognition of the American Society of Mechanical Engineers (ASME) Accreditation Program for holders of N, NA, NPT, and NV stamps and Certificates of Authorization. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Discussion:

IN 86-21 informed licensees that the NRC recognized the ASME Accreditation Program for N, NA, NPT, and NV stamps and the associated Certificates of Authorization. IN 86-21 also stated that quality system certificates issued to material manufacturers and material suppliers were excluded from this recognition.

The statement concerning quality system certificate holders has been misinterpreted by several licensees resulting in quality system certificate holders being removed from approved vendor lists. This was not the intent of IN 86-21. Quality system certificate holders were excluded from the recognition announced in IN 86-21 because the quality system certificate program was not covered by the exchange of correspondence between ASME, NRC, and the National Board of Boiler and Pressure Vessel Inspectors. If a licensee's NRC approved quality assurance (QA) program allows material manufacturers and material suppliers to be placed on the approved vendor list on the basis of a quality system certificate, then that method of evaluating and selecting vendors is not affected by the issuance of IN 86-21.

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No specific action or response is required by this information notice. If you have any questions regarding this notice, please contact the Regional Administrator of the appropriate NRC regional office or the technical contact listed below.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: E. T. Baker, IE
301-492-9663

Attachment: List of Recently Issued IE Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

October 15, 1986

IE INFORMATION NOTICE NO. 86-25, SUPPLEMENT 1: TRACEABILITY AND MATERIAL CONTROL OF MATERIAL AND EQUIPMENT, PARTICULARLY FASTENERS

Addressees:

All nuclear reactor facilities holding an operating license or construction permit.

Purpose:

This notice is to supplement IN 86-25 with information provided by the Industrial Fastener Institute (IFI) concerning an example of improperly marked fasteners. It is expected that the recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems from occurring at their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Discussion:

The IFI issued a news release on May 2, 1986 expressing its concerns about intentionally mismarked SAE J429K grade 8 bolts. According to IFI, these falsely marked bolts (grade 8.2 for grade 8) could undergo stress relaxation when subjected to temperatures above 500°F, and such relaxation could result in equipment failure and injury.

The IFI news release states that 75% of the foreign imported bolts (mainly from the Far East) have incorrect "headmarkings," indicating that they are SAE J429K grade 8 rather than grade 8.2. Grade 8 is a fine grain, medium carbon alloy steel (quenched and tempered), and grade 8.2 is a low carbon quenched and tempered martensite steel. The mechanical properties of both grades are essentially identical. For both grades, the tensile and yield strengths are 150 ksi and 120 ksi, respectively. However, the two grades of bolts are heat treated differently. Grade 8 bolts are quenched in oil and tempered at 800°F while grade 8.2 bolts are quenched in water or oil and tempered at 650°F. Consequently, using an 8.2-grade bolt in lieu of an 8-grade bolt at temperatures near 650°F could result in stress relaxation of the grade 8.2 bolt resulting from a change in the original tempered microstructure. Although SAE grade 8 and 8.2 bolts are not commonly used for high temperature safety-related applications in nuclear power plants, this situation provides an example of the type of problem which can result from anything less than a rigorous approach to the

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procurement of any safety-related component. Substitutions of this type could be detected by an effective receipt inspection program if the receipt inspections went beyond visual inspection requirements and included, on a sample basis, chemical analyses, nondestructive testing such as eddy current or conductivity testing, or destructive testing such as metallographic analyses.

No specific action or written response to this notice is required. If you have any questions regarding this matter, contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: J. C. Harper, IE
(301) 492-7568

Attachment: List of Recently Issued IE Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

November 4, 1986

IE INFORMATION NOTICE NO. 86-92: PRESSURIZER SAFETY VALVE RELIABILITY

Addressees:

All pressurized-water reactor (PWR) facilities holding an operating license or a construction permit.

Purpose:

This notice is provided to inform recipients of potentially significant problems pertaining to the reliability of pressurizer safety valves. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem from occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

St. Lucie 2:

On April 17, 1986 while shut down for refueling, St. Lucie 2 personnel bench tested three pressurizer safety valves that have a technical specification (TS) set point of 2515 ± 1% psig. Valve V-1200 lifted at 2554 psig (14 psi above the maximum TS set point, valve V-1201 lifted at 2893 psig (353 psi above the maximum TS set point), and valve V-1202 was leaking too much to test. The problem with V-1201 was attributed to a hole in the bellows which allowed boric acid to contaminate and corrode the valve internals causing them to bind. The seat of V-1202 was badly steam cut. The minor set point drift of V-1200 was considered normal for 18 months operation. All valves were repaired before replacing them in service.

McGuire 1:

On September 2, 1986, McGuire 1 reported that a pressurizer safety valve had lifted at 2375 psig, 85 psi below the 2485 ± 1% TS required set point. The primary system depressurized to 1800 psig before the valve fully closed. This event occurred during pressurization for an inservice leak test. During subsequent bench testing of that valve, the as-found set point was determined to be 2320 psig. Analysis showed that the three lifts used to determine the valve set point at the last test were trending downward. Cocking of the spring, probably because of debris under the spring or a burr, was considered to be the cause of the set point "drift." When the spring was properly seated, the set point stabilized.

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The low pressure at which the valve reclosed is attributed by the licensee to the setting of the blowdown adjusting ring, which deviated from the vendor's recommended setting by 137 notches. The remaining pressurizer safety valves on unit 1 were inspected and one other had misadjusted ring settings. The misadjustment was attributed to on-site readjustment by maintenance personnel. The correctly set rings on the third valve were set by the vendor's representative. Unit 1 valves will be repaired and reset.

Unit 2 had two pressurizer safety valves set by the vendor's representative and one set by plant maintenance personnel. The plant technician had recorded the ring settings, which were the same as the vendor's recommendation. Therefore, Unit 2 valves were determined to be operable.

Arkansas Nuclear One, Unit 2:

On September 15, 1986 while at 29% power, ANO 2 experienced a spurious opening and rapid reseating of two pressurizer safety valves, which resulted in an increase in level and pressure in the quench tank and a 4 psig drop in reactor coolant pressure. On September 25, 1986 following a reactor trip, quench tank parameters and tail pipe temperatures indicated that one or both pressurizer safety valves had lifted. Licensee tests showed that the setpoints of the valves were lower than the TS required 2485 psig ± 1%. One valve lifted at 2365 psig and the other between 2430 and 2455 psig. It was determined that the wrong test equipment had been used to set the lift pressure of the valves at the outage prior to these events. The valve manufacturer sent a representative to the site with the correct equipment. The valves have been correctly reset.

A search of licensee event reports from January 1, 1983 to the present shows that 20 other events involving 32 valves occurred. The problems were: 18 valves with set point drift, 12 valves with seat leakage, 1 valve with set pressure high and 1 valve designated as inoperable. Attachment 1 contains the details of this compilation. Other information on the subject of safety valve performance and reliability may be found in information notices 86-05; 86-05, Supplement 1; and 86-56, the subject of which are main steam safety valves.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: Mary S. Wegner, IE
(301) 492-4511

Attachments:
1. Other Pressurizer Safety Valve Problems
2. List of Recently Issued IE Information Notices

OTHER PRESSURIZER SAFETY VALVE PROBLEMS FROM 01/01/83 TO PRESENT

PLANT	NO. VALVES
INOPERABLE	
SEQUOYAH 1	1
LEAKING	
SALEM 2	3
SALEM 1	3
MILLSTONE 2	1
SEQUOYAH 2	1
SEQUOYAH 2	1
BEAVER VALLEY 1	1
SUMMER	1
SEQUOYAH 2	1
SET POINT DRIFT	
CALVERT CLIFFS 2	1
BEAVER VALLEY 1	1
SET POINT DRIFT HIGH	
FT CALHOUN	1
SALEM 2	3
SALEM 1	3
MILLSTONE 2	1
PALISADES	1
SUMMER	1
YANKEE-ROWE	1
SET POINT DRIFT LOW	
ANO 1	1
SEQUOYAH 1	1
SEQUOYAH 2	3
SET PRESSURE HIGH	
YANKEE-ROWE	1

SSINS No.: 6835
IN 86-93

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

November 3, 1986

IE INFORMATION NOTICE NO. 86-93: IEB 85-03 EVALUATION OF MOTOR-OPERATORS IDENTIFIES IMPROPER TORQUE SWITCH SETTINGS

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This notice is provided to alert recipients of a potentially significant safety problem discovered while performing the evaluations requested by IE Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," (IEB 85-03). It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems from occurring at their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

As a result of followup on IEB 85-03, Duke Power Company (DPC) discovered problems with valves operated by Rotork valve actuators at McGuire Nuclear Station. Specifically, the problem involved valves for which the factory-set torque switch settings had been previously changed at the plant site using a generic correlation between actuator torque output and torque switch setting. This could cause valve actuator motors to switch off before the valves complete their travel. Arbitrarily raising the torque switch setting to its maximum may result in damage to the valve and/or motor especially since thermal overload protection has been eliminated in many applications. Based upon this information, DPC has declared safety systems inoperable and shut down McGuire Units 1 and 2.

Discussion:

The vendor states that whenever the factory torque switch setting is changed in the field, an individual calibration curve or a bench test is required to accurately determine torque output.

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According to information recently provided by the vendor, torque switch settings of 1, 2, 3, 4, and 5, do not always correspond to rated torque output values between 40 and 100 percent as was used based on general information available several years ago. Tests and analytical evaluations by the licensee now confirm that the correlation used by the licensee was incorrect for some actuators. For example, evaluation of several valve actuator certificates for the same model revealed that the actual torque output with a switch setting of "1" varied through a range of 11 to 55 percent of maximum actuator torque output. The review did not indicate a variation in maximum output at the number 5 setting.

Analysis of two valves installed in the normal charging path, which would be required to close during safety injection, indicated that they may not be able to do so under differential pressure conditions which could exist following a loss-of-coolant accident. Although the as-found switch setting agreed with the design setting determined by DPC the application of output torque values being linear between 40 and 100 percent was not correct for all actuators.

Preliminary data indicates that up to 41 nuclear units may have some Rotork valve actuators. It is not known whether or not these are used in safety related applications.

The above example specifically deals with improper setting of the torque switches and illustrates the need for exercising extreme care in the setting of motor-operator switches because all types of switches must be set properly to ensure that the valves will function properly when needed. In fact, the specific event that prompted the issuance of IEB 85-03 was caused by improperly set torque bypass switches. Improperly set thermal overload switches recently (October 22, 1986) rendered the high-pressure coolant injection system inoperable at the Hope Creek Nuclear Station Unit 1. Both of these instances involved actuators manufactured by a company other than Rotork.

In addition, care must be taken to insure that all of the ramifications of changes to any of the motor-operator switches are fully understood. For instance, IE Information Notice 86-29, "Effects of Changing Valve Motor-Operator Switch Settings," describes how the changing of the limit switches on certain motor-operated valves resulted in a control room indication that the valves were closed when, in fact, they were partially open. This led to an excessive cooldown rate in the reactor coolant system at San Onofre Nuclear Generating Station Unit 3.

The information herein is being provided as an early notification of a possibly significant matter that is still under consideration by the NRC staff. Recipients should review the information for possible applicability to their facilities. If NRC evaluation so indicates, further licensee actions may be requested.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contacts: George Schnebli, RII
(404) 331-5582

Richard J. Kiessel, IE
(301) 492-8119

Attachment: List of Recently Issued IE Information Notices

SSINS No.: 6835
IN 86-94

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

November 6, 1986

IE INFORMATION NOTICE NO. 86-94: HILTI CONCRETE EXPANSION ANCHOR BOLTS

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This notice is provided to inform recipients of a revision to the published ultimate tensile capacities for 1/2-inch-diameter Hilti Kwik bolts that could potentially affect pipe support designs covered by IE Bulletin 79-02. This revision resulted from a Hilti review of all onsite test data developed at nuclear power plants after a 10 CFR Part 21 report identified that onsite tests at the Susquehanna plant produced results that were significantly below the published values contained in Hilti's "Architects and Engineers Anchor and Fastener Design Manual" for several sizes of anchor bolts.

It is expected that recipients review this information for applicability to their facilities and consider actions, if appropriate. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Testing of various sizes and embedment depths of Hilti Kwik and Super Kwik expansion anchor bolts was performed at the Susquehanna Steam Electric Station during March 1985 and during March/April 1986. The results of these tests established average ultimate tensile capacities of some expansion anchor bolts that were substantially lower than the published values in the Hilti manual. These results were reported to the NRC by Dravo Constructors, Inc. in a Part 21 report in May 1986. The worst-case result from the Susquehanna onsite tests established an average ultimate tensile capacity for the 1/2-inch anchor bolt with a 5-1/2-inch embedment that was 59% of the value published in the Hilti manual. Other significantly lower values reported by Dravo included 5/8", 3/4", 1", 1-1/4-inch Kwik bolts and the 1-inch Super Kwik bolt. All of these lower values were associated with the deeper embedment depths where failure is governed by anchor pullout. In response to the data generated at Susquehanna, Hilti performed a statistical evaluation of its available onsite test results and provided these results to the NRC in September 1986. Hilti's evaluation of the onsite test data supported the values published in its manual for all anchor bolt sizes except for 1/2-inch bolts. Hilti stated it would revise the

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values for the 1/2-inch-diameter Kwik bolt. A copy of the revised ultimate tensile capacity values for the 1/2-inch-diameter Hilti Kwik bolts is provided in Attachment 1.

Discussion:

A substantial amount of data scatter in the testing of concrete expansion anchor bolts is expected and this data scatter was considered in selecting the appropriate safety factors for design purposes. Although Hilti's evaluation of all plant test data confirmed its ultimate tensile capacities for most sizes of anchor bolts published in the manual, it appears that a greater amount of data scatter exists in the results for deeper embedment lengths. Apparently, the higher capacities that can be achieved for deeply embedded bolts are more sensitive to the exact concrete mix properties and/or installation techniques. Licensees using the ultimate capacities from the Hilti manual should be aware that the concrete mix and installation techniques at their facilities could have a significant impact on the results for the deeper embedment capacities. These concerns should not impact licensees with facilities that have established allowable working loads for expansion anchors based on onsite tests or facilities that establish allowable working loads based on the shorter embedment lengths.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: J. R. Fair, IE
492-4509

Attachments:
1. Revised Ultimate Tensile Values for 1/2-Inch-Diameter Kwik Bolt
2. List of Recently Issued IE Information Notices

REVISED ULTIMATE TENSILE VALUES
1/2" Diameter Kwik-Bolt

Diameter	Embedment	Concrete Strength		
		2000 psi	4000 psi	6000 psi
1/2"	2-1/4"	4545	5510	6845
	2-3/4"	5800	7200	7800
	3-1/2"	7000	9000	10000
	4-1/2"	7275	10000	10500
	6"	9000	10000	11000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

December 15, 1986

IE INFORMATION NOTICE NO. 86-102: REPEATED MULTIPLE FAILURES OF STEAM GENERATOR HYDRAULIC SNUBBERS DUE TO CONTROL VALVE SENSITIVITY

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This notice is provided to alert recipients of a potentially significant safety problem pertaining to recent events in which the steam generator hydraulic snubbers failed to meet their bleed and lockup specifications at two consecutive refueling outages. The primary cause appears to be control valve sensitivity to low hydraulic fluid flow velocity. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On January 7, 1986, the Portland General Electric Company reported [Licensee Event Report (LER) 85-13] multiple snubbers which failed to meet their bleed and lockup specifications at its Trojan Nuclear Plant. The report, and its supplement dated April 1, 1986, identified three areas of multiple snubber failures that were discovered during the 1985 refueling outage that began in May, 1985. These snubber failures were discovered as a direct result of the expanded inservice testing program which was instituted in accordance with a recent change to the plant's technical specifications. The prior inservice inspection program had not required the testing of these snubbers.

The 16 steam generator hydraulic snubbers at Trojan are 900-Kip Anker-Holth units. Following the failure of the first 2 steam generator snubbers to meet their bleed and lockup specifications, the remaining 14 were declared inoperable because of uncertainty regarding the time required to rebuild the snubbers following testing. All the snubbers were removed and overhauled. During the overhaul, the snubber seals were found to be degraded and the hydraulic fluid was heavily contaminated with seal material and rust. However, as discussed below, this was not the primary cause of the problem detected during the subsequent 1986 outage.

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An engineering evaluation of the effect of the failed snubbers on the steam generators was not initiated during the 1985 refueling outage because of the belief that they would not have restricted normal thermal growth. This decision was based on the snubber-testing company's judgment that the foreign material in the hydraulic fluid would not have affected the normal operation of the snubbers because of the relatively large channels through which fluid would flow under thermal growth conditions. However, in the case of a seismic or other severe dynamic event, it was determined that the snubbers would have activated (i.e., locked up*) and the foreign material could have blocked the bleed orifice. Because of the manner in which the snubber control valves are hydraulically interconnected, it is the licensee's belief that this would have to occur to all four snubbers on one steam generator before they would become locked in their current position.** They would remain in this position until a load reversal allowed flow through the main valves or possibly cleared the bleed port in at least one snubber.

The revised technical specifications for testing the snubbers required testing of each snubber that had failed its test during the previous testing program. Therefore, the 16 steam generator hydraulic snubbers were again tested during the refueling outage that began in April, 1986. The results of this testing indicated 12 failures--4 with excessive drag, 4 with high bleed rates at faulted load, 2 with no bleed rate at faulted load, 1 with excessive drag and high bleed rate, and 1 with high bleed in compression and no bleed in tension. The snubbers with no bleed rate cleared themselves upon load reversal.

There also was an issue of unusual movements of the pressurizer surge line that was thought for a while to be related to the snubber problems. This is discussed in Attachment 1.

Discussion:

As a part of its corrective actions during the 1985 refueling outage, the licensee had all the steam generator snubbers overhauled. Following overhaul, the snubbers could not meet their safety analysis acceptance criteria of a

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*Note: Common snubber nomenclature uses the term "lock up" to refer to (1) that point where the main flow path is closed and all flow is forced through the bleed orifice and (2) the condition where all flow is stopped and the snubber becomes a rigid strut. To eliminate any possibility for confusion between the two meanings, the term "activated" will be used for the first definition.

** In their safety evaluation report (Steven A. Varga's May 30, 1986, letter to Bart Withers) the Office of Nuclear Reactor Regulation staff concluded that "...the likelihood of full thermal lock-up occurring would require that the various contributing factors would have to affect three or four of the hydraulic snubbers on a single steam generator."

maximum drag force of 1,000 pounds at a minimum displacement rate of 0.025 in./min. This was because the snubber activated each time the velocity approached 0.025 in./min. Following consultation with the reactor vendor, the acceptance criteria was revised and the snubbers tested satisfactorily.

Because of the reoccurring snubber failures identified during the 1986 refueling outage, the licensee contracted for a detailed root cause analysis of the snubber failures. This analysis indicated that the low activation velocity (0.025 in./min) of the steam generator snubbers caused them to activate at very low fluid velocity through the main flow port. Once the snubber had activated, all flow was forced through the bleed port. Because of its extremely small size, this port acted much like a fine sieve. Apparently the first particle of foreign material would block the port causing the snubber to lock up. Thus, although contamination of the hydraulic fluid was a contributor to the problem, it was not the primary cause.

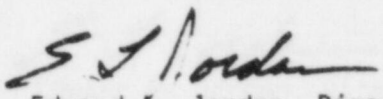
Based on this root cause analysis, the licensee decided to continue with its previously made plan to change out the control valves on the steam generator snubbers. The new snubber control valves have a much higher activation velocity (6 to 9 in./min) which is still acceptably small compared with that expected during any significant seismic event. In addition, the new snubber control valves incorporate a widely used "self-cleaning" poppet valve design as opposed to the original spring-ball check valve design. In the new design, the bleed orifices are grooves on the main poppet valve. In this way, the bleed orifices tend to be self-cleaning whenever there is flow through the main poppet valve.

All of the Anker-Holth steam generator snubbers were initially designed with relatively low activation velocities. Therefore, they are suspected of having the same type of problems as encountered at Trojan. In addition to Trojan, three other utilities have modified their steam generator snubbers so that they have activation velocities in the 6 to 10 in./min range.

However, since the root cause of the problem is the selection of an extremely low activation velocity, as opposed to a design flaw in the snubbers themselves, the problem may not be limited to only the facilities having Anker-Holth snubbers.

Attachment 2 to this information notice describes other multiple snubber failures found at Trojan during the 1985 refueling outage.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.



Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: Richard J. Kiessel, IE
(301) 492-8119

Attachments:
1. Pressurizer Surge Line Movements
2. Other Multiple Snubber Failures
3. List of Recently Issued IE Information Notices

Pressurizer Surge Line Movements

In what had been (until midway through the 1985-86 fuel cycle) a separate issue, the licensee had been monitoring the unusual movements of the pressurizer surge line since 1982. A walk-down of this line at the beginning of the 1985 refueling outage revealed additional movement had occurred during the last fuel cycle. A consultant was hired to evaluate and analyze these movements, and had determined that none of the previously identified potential causes, whether singly or combined, could have produced the observed movement. However, when he was advised of the possible problems with the steam generator snubbers, his worst case analysis (i.e., all snubbers on one steam generator were locked-up) indicated that locked-up snubbers could have produced the observed movement. This discovery delayed the submittal of LER 85-13, which was being prepared at the time.

Testing associated with the root cause analysis demonstrated that the snubbers on a particular steam generator would not restrict growth of that loop unless all four snubbers lock-up because the snubber hydraulic lines were connected in parallel. In addition, based on the results of the thermal expansion monitoring program conducted during the startup from the 1986 refueling outage, the licensee has determined that most, if not all, of the observed movement of the pressurizer surge line is expected due to normal thermal transients experienced by this line during heatups and cooldowns. Based on these findings, the licensee further concluded that the most likely cause of the reactor coolant system thermal restraint was due solely to the inadequate size of the gaps between system components and associated seismic or pipe whip restraints.

Other Multiple Snubber Failures

In addition to the steam generator hydraulic snubber failures, the Trojan LER identified two other areas of multiple snubber failures. Although not the subject of this information notice, they are briefly discussed to assist in identifying all the safety-related failures discussed in the LER.

1. The first additional area of multiple snubber failures was a 25 percent overall failure rate of small mechanical snubbers [Pacific Scientific models PSA-1/4 (36 percent failure rate) and PSA-1/2 (17.6 percent failure rate)].

2. The second additional area of multiple snubber failures involved the four main steam line hydraulic snubbers (two 70-Kip and two 130-Kip Bergen-Paterson units). The snubbers were declared inoperable without testing upon discovery of the steam generator hydraulic snubber failures.

Additional discussions of multiple snubber failures can be found in IE Information Notice 84-67, "Recent Snubber Inservice Testing with High Failure Rates," LER 84-079 for San Onofre Nuclear Generating Station Unit 2 (dated January 25, 1985, and revised March 12, 1985), and LER 85-027 for San Onofre Nuclear Generating Station Unit 3 (dated May 16, 1985).

SSINS NO.: 6835
IN 86-104

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

December 16, 1986

IE INFORMATION NOTICE NO. 86-104: UNQUALIFIED BUTT SPLICE CONNECTORS IDENTIFIED IN QUALIFIED PENETRATIONS

Addressees:

All pressurized and boiling-water reactor facilities holding an operating license or a construction permit.

Purpose:

This notice is to alert recipients to unqualified butt splice connectors supplied by General Electric (GE) in conjunction with qualified F-01 series penetration enclosures. These connectors failed during a recent environmental qualification test performed by Wyle Laboratories. It is suggested that recipients review this information for applicability to their facility and consider actions, if appropriate, to preclude occurrence of similar problems. Suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

During an NRC equipment qualification (EQ) inspection at Dresden Nuclear Power Station, May 19-23, 1986, a deficiency was discovered concerning a lack of similarity between tested and installed nylon insulated butt splices in EQ qualified GE electrical penetrations. Commonwealth Edison sent four sample splices removed from Quad Cities Nuclear Power Station to Wyle Laboratory to further substantiate their qualification for use in a harsh environment. These splices were identical to those installed at Dresden. During the testing performed at Wyle Laboratory December 4-5, 1986, all four samples exhibited excessive leakage currents to ground when exposed to a steam environment. Commonwealth Edison consequently declared the splices unqualified and shut down its Quad Cities Unit 1 to rework the splices by wrapping them with previously qualified tape. Dresden Unit 2 has similarly reworked the splices by wrapping them with tape. Duane Arnold Energy Center also has commenced a shutdown in order to make repairs.

Discussion:

The splices in question were supplied by GE as part of their F-01 penetration assemblies. GE obtained the splices from three different manufacturers: Amp, Thomas and Betts, and Hollinsworth. The splices in question are nylon insulated

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butt splices and most appear to be for 12-to-22 gauge conductors although other sizes may be installed as well. The splices from the three manufacturers are almost identical in design and are all manufactured using either Zytel-42 or Celanese-1200 nylon tubing over a metal crimp connector.

The splices tested at Wyle were manufactured by Amp and had been removed from Quad Cities where they had been exposed to approximately 12 years of aging (thermal and radiation). Wyle also subjected the samples to aging and radiation sufficient to add an additional year. After this aging and irradiation, the samples were inspected and tested for insulation resistance. The samples showed no sign of fatigue and exhibited a high insulation resistence. The samples were then put into a loss-of-coolant-accident (LOCA) chamber and energized for the start of the LOCA test. Two samples were energized with 528 volts and two samples were energized with 132 volts. During the preheat of the LOCA chamber and before the actual LOCA test, one of the 528-volt samples began shorting to ground and blew its fuse. At this point the temperature in the chamber had reached only 150°F. The test was continued with two other specimens blowing fuses at the 20-second point and at the 2-hour, 23-minute point of the test. The fourth sample finished the test, but excessive leakage currents were measured.

The short circuits that occurred appeared to start by condensation entering the splice between the wire insulation and the nylon tubing. The arcing caused insulation degradation that then allowed arcs to pass through the insulation to the enclosure.

The NRC is currently evaluating additional data pertaining to butt splice connectors and may be issuing additional information on this subject in the near future.

No written response to this notice is required. If you have any questions regarding this notice, please contact the Regional Administrator of the appropriate NRC regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness and
Engineering Response
Office of Inspection and Enforcement

Technical Contact: J. Jacobson
(301) 492-8845

Attachment:
List of Recently Issued IE Information Notices

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BABCOCK & WILCOX NUCLEAR POWER DIVISION	313	X						X	X						
EBASCO SERVICES INCORP.						X									
EXXON NUCLEAR CO. NUCLEAR FUELS DEPT.	APPLICABLE TO PWRS WITH FUEL SUPPLIED BY EXXON														
FISHER CONTROLS INT.INC.	NUCLEAR POWER PLANTS THAT USE FISHER MODEL 1073-1076 MANUAL VALVE ACTUATORS														
GENERAL ELECTRIC COMPANY NUCLEAR ENERGY BUSINESS			X	X	461		X				237 249	X	X		
LIMITORQUE CORPORATION	X														
PEEBLES ELECTRICAL MACHINES										275					
ROCKWELL INTERNATIONAL FLOW CONTROL DIVISION	NOT IDENTIFIED														
ROTORK CONTROLS LIMITED															X
ROTORK CONTROLS, INC.															X

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VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

PLANTS / VENDORS	ARKANSAS 1 & 2	BELLEFONTE 1 & 2	BIG ROCK POINT	BROWNS FERRY 1 & 2	CLINTON 1	COMANCHE PEAK 1 & 2	COOPER	CRYSTAL RIVER 3	DAVIS BESSE 1	DIABLO CANYON 1 & 2	DRESDEN 2 & 3	DUANE ARNOLD	FERMI 2	FORT CALHOUN 1	FORT ST VRAIN
STONE & WEBSTER ENGINEERING DIVISION						X									
TOBAR, INCORPORATED															
TRANSAMERICA DELAVAL, INC		X				X									
WESTINGHOUSE ELECTRIC COR.															
WESTINGHOUSE ELECTRIC CORPORATION	ALL PWR FACILITIES WITH FUEL SUPPLIED BY WESTINGHOUSE														
WESTINGHOUSE HITTMAN NUCLEAR, INCORPORATION	NUCLEAR POWER FACILITIES WHO ARE REGISTERED USERS OF CASKS FOR WHICH HITTMAN IS THE C OF C HOLDER														
WILMAR ELECTRONICS															
WYLE LABORATORIES EASTERN TEST AND ENG.															
YANKEE ATOMIC ELECTRIC CO NUCLEAR SERVICE DIVISION															

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VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	GRAND GULF 1	HARRIS 1	HATCH 1 & 2	HOPE CREEK 1	INDIAN POINT 2 & 3	JAFITZPATRICK	LACROSSE	LASALLE 1 & 2	LIMERICK 1 & 2	MAINE YANKEE	MCGUIRE 1 & 2	MIDLAND 1 & 2	MILLSTONE 2 & 3	NINE MILE POINT 1 & 2	OCONEE 1, 2 & 3	PALO VERDE 1, 2 & 3
BABCOCK & WILCOX NUCLEAR POWER DIVISION															X	
EBASCO SERVICES INCORP.																
EXXON NUCLEAR CO. NUCLEAR FUELS DEPT.	APPLICABLE TO PWRS WITH FUEL SUPPLIED BY EXXON															
FISHER CONTROLS INT.INC.	NUCLEAR POWER PLANTS THAT USE FISHER MODEL 1073-1076 MANUAL VALVE ACTUATORS															
GENERAL ELECTRIC COMPANY NUCLEAR ENERGY BUSINESS	X		X	354		X	X	X	X							X
LIMITORQUE CORPORATION																
PEEBLES ELECTRICAL MACHINES																
ROCKWELL INTERNATIONAL FLOW CONTROL DIVISION	NOT IDENTIFIED															
ROTORK CONTROLS LIMITED											369					
ROTORK CONTROLS, INC.											369					

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VENDOR INSPECTION REPORTS RELATED TO REACTOR PLANTS

VENDORS / PLANTS	GRAND GULF 1	HARRIS 1	HATCH 1 & 2	HOPE CREEK 1	INDIAN POINT 2 & 3	JAMES A. FITZPATRICK	LACROSSE	LASALLE 1 & 2	LIMERICK 1 & 2	MAINE YANKEE	MCGUIRE 1 & 2	MIDLAND 1 & 2	MILLSTONE 2 & 3	NINE MILE POINT 1 & 2	DONALD C. COOK 1 & 2	PEACH BOTTOM 2 & 3
STONE & WEBSTER ENGINEERING DIVISION																
TOBAR, INCORPORATED																
TRANSAMERICA DELAVAL, INC	X	X										X				
WESTINGHOUSE ELECTRIC COR.									352							X
WESTINGHOUSE ELECTRIC CORPORATION	ALL PWR FACILITIES WITH FUEL SUPPLIED BY WESTINGHOUSE															
WESTINGHOUSE HITTMAN NUCLEAR, INCORPORATION	NUCLEAR POWER FACILITIES WHO ARE REGISTERED USERS OF CASKS FOR WHICH HITTMAN IS THE C OF C HOLDER															
WILMAR ELECTRONICS																
WYLE LABORATORIES					247			X						220		
EASTERN TEST AND ENG.					287											
YANKEE ATOMIC ELECTRIC CO										X						
NUCLEAR SERVICE DIVISION																

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VENDORS \ PLANTS	PERRY 1 & 2	PILGRIM 1 & 2	QUAD CITIES 1 & 2	RANCHO SECO 1	RIVER BEND 1 & 2	SAN ONOFRE 1	SEABROOK 1 & 2	SHOREHAM	SOUTH TEXAS 1 & 2	SUSQUEHANNA 1 & 2	THREE MILE ISLAND 1	TROJAN	VERMONT YANKEE	VOGTLE 1 & 2	WATTS BAR 1 & 2
BABCOCK & WILCOX NUCLEAR POWER DIVISION				X							289				460
EBASCO SERVICES INCORP.															
EXXON NUCLEAR CO. NUCLEAR FUELS DEPT.	APPLICABLE TO PWRS WITH FUEL SUPPLIED BY EXXON														
FISHER CONTROLS INT.INC.	NUCLEAR POWER PLANTS THAT USE FISHER MODEL 1073-1076 MANUAL VALVE ACTUATORS														
GENERAL ELECTRIC COMPANY NUCLEAR ENERGY BUSINESS	440	293	X		458			X		X			X		397
LIMITORQUE CORPORATION				X								X		X	
PEEBLES ELECTRICAL MACHINES															
ROCKWELL INTERNATIONAL FLOW CONTROL DIVISION	NOT IDENTIFIED														
ROTORK CONTROLS LIMITED															
ROTORK CONTROLS, INC.															

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VENDORS / PLANTS	PER	PIL	QUA	RAN	RIV	SAN	SEA	SHO	SOT	SUS	THR	TRO	VER	VOG	WNP
	PRY	PLG	D	CHO	ER	ON	BRO	ER	EXT	QUE	ER	JAM	MOT	TLE	P
	1 & 2	MI	CIT		BEN	OFF	OK	MA	AS	HAT	MI		NT		PS
		1 & 2	CIE	SEC	ND	RE	1 & 2		PR	NA	LE		AN	2	
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					1	1	2		O		I		K		2
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			2			3			1	2	1				
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									2		2				
STONE & WEBSTER ENGINEERING DIVISION															
TOBAR, INCORPORATED							443		498						
TRANSAMERICA DELAVAL, INC				X	X	206									460
WESTINGHOUSE ELECTRIC COR.															
WESTINGHOUSE ELECTRIC CORPORATION	ALL PWR FACILITIES WITH FUEL SUPPLIED BY WESTINGHOUSE														
WESTINGHOUSE HITTMAN NUCLEAR, INCORPORATION	NUCLEAR POWER FACILITIES WHO ARE REGISTERED USERS OF CASKS FOR WHICH HITTMAN IS THE C OF C HOLDER														
WILMAR ELECTRONICS				X											
WYLE LABORATORIES EASTERN TEST AND ENG.															
YANKEE ATOMIC ELECTRIC CO NUCLEAR SERVICE DIVISION															

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SEE INSTRUCTIONS ON THE REVERSE

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Quarterly Report
October 1986-December 1986

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October 1986-December 1986

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 1986 thru December 1986. Also, included in this issue are the results of certain inspections performed prior to October 1986 that were not included in previous issues of NUREG-0040.

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