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Vol. 20, No. 2

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
April – June 1996

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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

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

ABSTRACT

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from April 1996 through June 1996, and also includes a report, issued in March of 1996, but not included in NUREG-0040, Vol. 20, No. 1.

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INTRODUCTION

A fundamental premise of the U. S. Nuclear Regulatory Commission (NRC) licensing and inspection program is that licensees are responsible for the proper construction and safe and efficient operation of their nuclear power plants. The Federal government and nuclear industry have established a system for the inspection of commercial nuclear facilities to provide for multiple levels of inspection and verification. Each licensee, contractor, and vendor participates in a quality verification process in compliance with requirements prescribed by the NRC's rules and regulations (Title 10 of the *Code of Federal Regulations*). The NRC does inspections to oversee the commercial nuclear industry to determine whether its requirements are being met by licensees and their contractors, while the major inspection effort is performed by the industry within the framework of quality verification programs.

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

The Vendor Inspection Section (VIS) of the Special Inspection Branch reviews and inspects nuclear steam system suppliers (NSSSs), architect engineering (AE) firms, suppliers of products and services, independent testing laboratories performing equipment qualification tests, and holders of NRC construction permits and operating licenses in vendor-related areas. These inspections are done to ensure that the root causes of reported vendor-related problems are determined and appropriate corrective actions are developed. The inspections also review vendors to verify conformance with applicable NRC and industry quality requirements, to verify oversight of their vendors, and coordination between licensees and vendors.

The VIS does inspections to verify the quality and suitability of vendor products, licensee-vendor interface, environmental qualification of equipment, and review of equipment problems found during operation and their corrective action. When nonconformances with NRC requirements and regulations are found, the inspected organization is required to take appropriate corrective action and to institute preventive measures to preclude recurrence. When generic implications are found, NRC ensures that affected licensees are informed through vendor reporting or by NRC generic correspondence such as information notices and bulletins.

This quarterly report contains copies of all vendor inspection reports issued during the calendar quarter for which it is published. Each vendor inspection report lists the nuclear facilities inspected. This information will also alert affected regional offices to any significant problem areas that may require special attention. Appendices list selected bulletins, generic letters, and information notices, and include copies of other pertinent correspondence involving vendor issues.

INSPECTION REPORTS

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

May 3, 1996

Mr. Michael L. Bussler, President
and Chief Executive Officer
Algor, Inc.
150 Beta Drive
Pittsburgh, PA 15238-2932

SUBJECT: NRC INSPECTION REPORT NO. 99901294/96-01

Dear Mr. Bussler:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Algor, Inc., conducted by Robert L. Pettis, Jr. and Billy Rogers, of this office, on February 20 through 22, 1996. The NRC inspection team conducted an evaluation of the Algor quality assurance (QA) program and the implementation of that program as it relates to safety-related software supplied to the nuclear industry.

The NRC inspection team reviewed documentation, procedures, and representative records, conducted interviews and held discussions with members of your staff. On the basis of this inspection, the inspection team determined that the implementation of the Algor QA program failed to meet certain requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations and 10 CFR Part 21. The enclosed inspection report contains a detailed discussion of the areas examined.

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

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

Sincerely,

Gregory C. Cwalina

Gregory C. Cwalina, Acting Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No.: 99901294

Enclosures: 1. Notice of Violation
2. Inspection Report No. 99901294/96-01

NOTICE OF VIOLATION

Algor, Inc.
Pittsburgh, Pennsylvania

Docket No. 99901294

During an NRC inspection conducted at Algor, Inc., on February 20 through 23, 1996, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> Title 10 of the Code of Federal Regulations Part 21.21(b) states if the deviation or failure to comply is discovered by a supplier of basic components, or services associated with basic components, and the supplier determines that it does not have the capability to perform the evaluation to determine if a defect exists, then the supplier must inform the purchasers or affected licensees within five working days of this determination so that the purchasers or affected licensees may evaluate the deviation or failure to comply, pursuant to 21.21(a).
>
> Contrary to the above, Algor, Inc., did not inform NRC licensees of deviations within five working days of discovery when informing licensees of the deviations identified by the Algor, Inc., Quality Assurance Bulletins Nos. 68, 69, 71, 84, 100, 101, and 102.

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

Pursuant to the provisions of 10 CFR 2.201, Algor, Inc., is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Special Inspection Branch, Division of Technical Support, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Rockville, Maryland
this 3rd day of May, 1996

Enclosure 1

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

REPORT NO.:	99901294/96-01
ORGANIZATION:	Michael L. Bussler, President and Chief Executive Officer Algor, Inc. 150 Beta Drive Pittsburgh, PA 15238-2932
ORGANIZATIONAL CONTACT:	Theresa Anania Director of Operations
NUCLEAR INDUSTRY	Suppliers of safety-related finite-element computer programs.
INSPECTION DATES:	February 20 through 22, 1996
LEAD INSPECTOR:	Robert L. Pettis, Jr. Robert L. Pettis, Jr., P.E. Vendor Inspection Section (VIS) Special Inspection Branch (SIB) Division of Inspection and Support Programs (DISP) Office of Nuclear Reactor Regulation (NRR)
	5/2/96 Date
OTHER INSPECTORS:	Billy H. Rogers, VIS/SIB/DISP/NRR
REVIEWED BY:	Gregory C. Cwalina Gregory C. Cwalina, Chief VIS/SIB/DISP/NRR
	5/3/96 Date
APPROVED BY:	Gregory C. Cwalina Gregory C. Cwalina, Acting Chief SIB/DISP/NRR
	5/3/96 Date

Enclosure 2

1 SUMMARY OF INSPECTION FINDINGS

During this inspection, the NRC inspection team evaluated the implementation of the Algor, Inc., quality assurance (QA) program related to the supply of safety-related finite-element computer programs to the nuclear industry. The inspection was conducted to determine Algor's compliance with the requirements of Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B) and the provisions of 10 CFR Part 21 (Part 21). The inspection team reviewed technical information, procedures and representative records, conducted interview and held discussions with members of Algor's staff.

1.1 Violations

1.1.1 Contrary to 10 CFR 21.21(b) which requires that if a deviation is discovered by a supplier of basic components and the supplier determines that it does not have the capability to perform the evaluation to determine if a defect exists, then the supplier must inform the purchasers or affected licensees within five working days of this determination so that the purchasers or affected licensees may evaluate the deviation, Algor, Inc., (Algor) did not inform NRC licensees of deviations within five working days of discovery. (Violation 96-01-01)

1.1.2 Contrary to 10 CFR 21.6 which requires that entities subject to the regulations post 10 CFR Part 21, Section 206 of the Energy Reorganization Act of 1974, and the procedures adopted pursuant to 10 CFR Part 21, Algor had not posted Section 206 or the Algor 10 CFR Part 21 Procedure, QAPM Section 8, and the copy of 10 CFR Part 21 that Algor had posted was dated January 1988 which did not contain the numerous subsequent revisions contained in the later editions. (Non-Cited Violation)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first inspection of Algor.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

Algor, Incorporated, is a designer of various engineering finite-element analysis software programs used both in nuclear and commercial applications. Algor's product line includes both linear and non-linear stress analysis programs designed to analyze various applications including dynamic, vibration, heat transfer and piping analysis.

3.2 Entrance and Exit Meetings

During the entrance meeting, held on February 20, 1996, the NRC inspection team met with members of Algor management and staff, discussed the scope of the inspection, and established organizational contacts. During the exit meeting, held on February 22, 1996, the inspection team summarized its findings with Algor management. Section 4 of this report lists the persons contacted during the inspection.

3.3 10 CFR Part 21 Procedures and Implementation

Section 10 CFR 21.6, "Posting Requirements," requires that entities subject to the regulations post 10 CFR Part 21, Section 206 of the Energy Reorganization Act of 1974, and the procedures adopted pursuant to 10 CFR Part 21. The Quality Assurance Procedures Manual (QAPM) Section 8, "10CFR21 - Error Reporting," Revision 4, dated February 21, 1995, Paragraph 8.6 "Posting," stated that 10 CFR Part 21 shall be prominently displayed on the employee bulletin board but did not address Section 206 or the Algor 10 CFR Part 21 procedure, QAPM Section 8.

The inspectors reviewed Algor's posting and determined that Section 206 was absent and Algor's 10 CFR Part 21 Procedure was absent. In addition, the copy of 10 CFR Part 21 that was posted was dated January 1988 and did not contain the numerous subsequent revisions contained in the later editions. The inspectors provided Algor with current copies of 10 CFR Part 21 and Section 206 and Algor indicated that they would correct their posting. The failure to meet the posting requirements specified in 10 CFR 21.6 constitutes a violation of minor significance and is being treated as a Non-Cited violation, consistent with Section IV of the NRC Enforcement policy (NUREG-1600).

The inspectors reviewed the implementation of Algor's 10 CFR Part 21 program to determine compliance with 10 CFR Part 21. Discussion of Algor products provided to NRC licensees indicated that deviations would occur as errors in the computer software. Algor had defined four error types in QAPM Section 7, "Qalipak Quality Assurance Package," Revision 4, dated February 21, 1995; Class A errors, Class B errors, Class C errors, and Class H errors.

Class C errors, defined as errors which produce results that may appear reasonable and correct, but are in fact erroneous; were difficult for the end user to detect because the program did not provide any clear indication that the results may be invalid; and provided no warning messages, error messages or other abnormalities despite responsible operation of the Algor product, relative to the error itself, which indicated that the results may be invalid. The inspectors concluded that Class C errors would constitute deviations, as defined by 10 CFR Part 21, and would therefore require evaluation by the supplier (Algor) or the user (NRC Licensee). Algor had determined that it would not evaluate Class C errors but would inform the user such that the user could perform the evaluation to determine whether the deviation was a defect or failure to comply.

Algor had notified customers of Class C errors by providing them with Quality Assurance Bulletins (QAB) which described the error and provided customers with applicable information on software revisions which corrected the error or methods that users could follow to work around the error. The inspectors review of QABs included a block of twenty, which were issued during 1995 and early 1996. The review determined that seven of the Class C error reports were provided to NRC licensees with greater than five working days between error classification (identification of the deviation) and providing the QAB to the licensee (informing the customer of the deviation). The length of time between error classification (identification of the deviation) and providing the QAB to the licensee (informing the customer of the deviation) ranged from twenty-three to seventy-one days. The error classification dates and QAB issuance dates are listed as follows:

QAB No. 68 error classified 4/11/95 - QAB issued 5/4/95
QAB No. 69 error classified 4/11/95 - QAB issued 5/5/95
QAB No. 71 error classified 4/11/95 - QAB issued 5/4/95
QAB No. 84 error classified 6/1/95 - QAB issued 8/11/95
QAB No. 100 error classified 6/16/95 - QAB issued 8/14/95
QAB No. 101 error classified 6/16/95 - QAB issued 8/14/95
QAB No. 102 error classified 6/16/95 - QAB issued 8/14/95

The provisions of 10 CFR 21.21(b) require that if a deviation or failure to comply is discovered by a supplier of basic components and the supplier determines that it does not have the capability to perform the evaluation to determine if a defect exists, then the supplier must inform the purchasers or affected licensees within five working days of this determination so that the purchasers of affected licensees may evaluate the deviation or failure to comply, pursuant to 21.21. Contrary to the requirements of 10 CFR 21.21(b), Algor did not notify NRC licensee within five working days of identifying the deviations. This was identified a Violation No. 96-01-01.

3.4 Review of Qalipak Quality Assurance Package

The inspectors reviewed QAPM Section 7, "Qalipak Quality Assurance Package," Revision 4, dated February 21, 1995, which established the service to which Algor customers could subscribe to receive notification of errors which could produce significant, erroneous output data despite responsible operation of an Algor product. Algor offered two levels of the Qalipak Subscription: (1) Basic - a service which was available free, upon request, to all Algor customers. The customer would receive a Qalipak Summary Report (QSR) which would provide a brief description of Class C errors and was to be issued monthly; and (2) Qalifax - a service in which Quality Assurance Bulletins (QAB) are issued to customers by telefax as soon as the QAB is completed.

The Basic Qalipak Quality Assurance Package was discussed in QAPM Section 7, which stated that the Quality Summary Report would be provided to customers monthly, on request. Algor indicated that although QAPM Section 7 required the monthly issuance of the Qalipak Summary Report none had yet been issued. Algor indicated that they were in the process of developing the first issue of the Qalipak Summary Report, to be issued in the near future, and monthly thereafter.

The Qalifax Quality Assurance Package was the method by which Algor notified NRC licensees of deviations (Class C errors) through the issuance of QABs. Algor provided Qalifax as a service sold in conjunction with the purchase of safety-related software. A Qalifax subscription was required to be in place (previously purchased and current) or purchased concurrently with safety-related software purchases. A customer was only required to have one Qalifax subscription in place which could support numerous safety-related software purchases. All QABs related to different software packages were provided to all Qalifax subscribers regardless of the software that the customer had purchased.

The inspectors reviewed Algor's Qalifax subscribers records and Qalifax distribution records to verify that Algor had been providing notification of all Class C errors to all Qalifax subscribers including NRC licensees. Section 7, Paragraph 7.3, of the QAPM, "Qalipak Distribution" states that a Qalipak Transmittal Sheet would be included with each transmittal of a QAB. The Qalipak Transmittal Sheet was provided so that the customer could acknowledge receipt of the QAB. If the first Qalipak Transmittal Sheet was not received Algor would send a Second Notice and, if necessary, a Final Notice in conjunction with telephone calls. The inspectors reviewed the records for QABs Nos. 33 through 47, Class C errors, issued in September of 1993. Algor's records showed that all Qalifax subscribers, current in September 1993, had been notified and the records contained documentation of positive receipt acknowledgement from all Qalifax subscribers (Qalifax Transmittal Sheets signed by the customer and returned to Algor). The inspectors concluded that Algor had taken adequate measures to ensure that licensees had received notification of Class C errors.

The inspectors noted that QAPM Section 7.1 stated that Qalifax subscribers would only receive QABs if the error was in purchased software while Section 7.3 stated that any QAB for Class C errors would be immediately distributed to all current subscribers of the Qalifax subscription program. Algor indicated that although the procedure was contradictory the practice was to distribute QABs for all Class C errors to all Qalifax customers. The inspectors verified that QABs had been distributed to all Qalifax customers, regardless of software purchases. In addition, Algor indicated that, if a customer did not renew the yearly license for a piece of software and also did not renew the Qalifax subscription, all licensees would still be notified of all Class C errors which were applicable to the version of software that they had initially purchased as safety-related with a corresponding Qalifax subscription. The inspectors concluded that although there was an inconsistency in the Algor procedures, Algor had indicated an adequate position on the licensee notification of Class C errors.

3.5 Purchase Order Review

The NRC inspectors reviewed purchase orders (POs) to Algor from nuclear customers to determine the extent to which Appendix B and Part 21 requirements were imposed. The review determined that almost all the POs reviewed imposed such nuclear quality assurance requirements for software and related Qalifax technical support. The following is a list of the POs which were selected for review during the inspection:

Carolina Power & Light Company

- PO No. 597206M-CR, dated December 9, 1993, ordered various finite-element analysis manuals including a Qalifax subscription. The PO was marked "Safety-Related" and imposed the requirements of Appendix B and Part 21. The corresponding Algor invoice for this order was Invoice No. 038558, dated December 23, 1993.

- PO No. 599637M-CR, dated March 31, 1994, ordered numerous unix platform stress and heat transfer software, including technical support. The PO was marked "Safety-Related" and imposed the requirements of Appendix B and Part 21. The corresponding Algor invoice for this order was Invoice No. 040374, dated March 31, 1994.

Florida Power & Light Company

- PO No. B91536 01071, dated December 16, 1991, ordered various finite-element analysis software including a Qalifax subscription. The PO was marked "Nuclear Safety-Related QL-1" and imposed the requirements of Appendix B and Part 21. The corresponding Algor invoice for this order was Invoice No. 033730, dated February 26, 1993.

Entergy Operations

- Contract No. C-6135, dated November 30, 1994, ordered various finite-element analysis software, including a Qalifax subscription, for Arkansas Nuclear One, River Bend Station, Grand Gulf Nuclear Station and the Waterford Steam Electric Station, Unit 3, and imposed the requirements of Appendix B and Part 21. The corresponding Algor invoice for this order was Invoice No. 044404, dated November 30 1994.

Public Service Electric and Gas Company (PSE&G)

- PO P1-431641, dated October 4, 1993; PO P3-0751638-1250-0000, dated October 28, 1994, and PO P3-0826805-1240-0000, dated October 31, 1995. All three POs were for various software modules including a Qalifax subscription. All POs reviewed were for non safety-related material. The above PO review identified an extremely low level of nuclear safety-related activity processed by Algor over the past several years.

Wolf Creek Nuclear Operating Corporation

- Algor Invoice No. 053195, dated December 28, 1995, referenced customer PO No. 566898 for various finite-element software including a Qalifax subscription. The Wolf Creek PO could not be located during the inspection but their quality Program Requirements Document No. AD17, Revision 00, dated December 27, 1995, identified in Section 3.1 that the reporting requirements of Part 21 applied.

A review of revenue reports from Algor's nuclear customers identified only nine nuclear customers out of several thousand total customers purchasing Algor software and related services. However, not all nuclear customers may have purchased safety-related products and services from Algor pursuant to Appendix B and Part 21 requirements, as in the case of PSE&G. Algor's nuclear customers included Bechtel, Carolina Power & Light (CP&L), Crosby Valve, Entergy, EQE Engineering, Florida Power & Light (FP&L), Lockheed Idaho, Northern States Power, Proto-Power, PSE&G and Wolfcreek Nuclear.

A review of all of the available audit reports of Algor from these customers only produced audit reports from FP&L, CP&L and Entergy. The review of the CP&L audit report, performed at Algor in September 1993, identified that it was a Nuclear Procurement Issues Council (NUPIC) based audit and as such, other NUPIC member licensees could use the report for implementation purposes if it adequately covers their respective scope of supply. However, for most of the nuclear customers, purchasing activity preceded the CP&L inspection thereby making it impossible for them to take credit for the CP&L audit of Algor.

The NRC inspectors concluded from its review that not all licensees purchasing Appendix B and Part 21 software and technical services had audited Algor for quality program implementation compliance. This issue will be addressed in a future NRC inspection at these licensees.

4 PERSONS CONTACTED

The following persons were contacted during the inspection and except as noted, attended both the entrance and exit meetings.

Algor, Incorporated:

*	M. Bussler	President and Chief Executive Officer
	T. Anania	Director of Operations
	R. Seebacher	Quality Assurance Manager

U.S. Nuclear Regulatory Commission:

R. Pettis	Team Leader
B. Rogers	Reactor Engineer

* Did not attended exit meeting

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

June 6, 1996

Mr. William Zelner
Quality Manager
AMP Products, Inc.
250 Main Street
Jacobus, PA 17407-4560

SUBJECT: NRC INSPECTION REPORT NUMBER 99901295/96-01

Dear Mr. Zelner:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of your facilities performed April 29 through May 2, 1996, conducted by Mr. Anil S. Gautam of this office. The inspection involved activities authorized by Part 21, "Reporting Defects and Noncompliances," of Title 10 of the <u>Code of Federal Regulations</u>. An exit meeting was held on May 2, 1996, during which we discussed our findings with you.

The inspection was conducted to ascertain whether licensees effectively monitored the control of quality by AMP Products Inc. (AMP) for safety-related electrical terminals and splices purchased by licensees for nuclear power plants. The inspector assessed specific attributes and implementation of AMP's quality control program and the licensees' monitoring of these areas.

We assessed attributes of your quality control program according to the criteria of 10 CFR Part 50, Appendix B, and the guidance in NRC Regulatory Guide 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants." Details of the inspection are discussed in the enclosed copy of our inspection report.

In general, AMP's quality control program was effective and was appropriately monitored by licensees. The NRC inspector evaluated licensee audit findings and AMP corrective actions, commercial-grade item dedication, and monitoring of AMP's sub-vendors.

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

No response is required to this letter. If you have any questions about this inspection, we will be pleased to discuss them with you.

Sincerely,

Robert M. Gallo

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

Docket No. 99901295

Enclosure: Inspection Report 99901295/96-01

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

REPORT NO.:	99901295/96-01
ORGANIZATION:	AMP Products, Inc. 250 Main Street Jacobus, PA 17407-4560
ORGANIZATIONAL CONTACT:	Mr. William Zelner, Manager Quality Assurance
INSPECTION DATES:	April 29 through May 2, 1996
INSPECTOR:	Anil S. Gautam
APPROVED BY:	Robert M. Gallo, Chief Special Inspection Branch Division of Inspection and Support Programs

Enclosure

1 SUMMARY OF INSPECTION FINDINGS

NRC conducted a special inspection to assess whether licensees effectively monitored how well AMP Products, Inc. (AMP) controlled the quality of safety-related electrical terminals and splices purchased by licensees for use in nuclear power plants. The inspection was conducted by an NRC inspector from the Special Inspection Branch of the Office of Nuclear Reactor Regulation at the AMP General Products Business Unit, Jacobus, Pennsylvania. The inspector assessed attributes of AMP's quality control program and its monitoring by licensees for the period of January 1994 through April 1996.

In general, the AMP quality control program and its implementation were in compliance with the requirements of Appendix B to 10 CFR Part 50. Licensees effectively monitored AMP's quality control program through Nuclear Procurement Issues Committee (NUPIC) audit teams, questionnaires, and telephone surveys for safety-related electrical terminals and splices. The inspector had no concerns about AMP's control of quality and licensees' monitoring of AMP.

Status of previous inspection findings are in Section 2. Findings and other comments are in Section 3. The AMP personnel contacted, including those attending the entrance and exit meetings are listed in Section 4.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of AMP Products, Inc.

3 FINDINGS AND OTHER COMMENTS

Licensees are required to monitor how well vendors control quality consistent with the importance, complexity, and quantity of products or services purchased from the vendors. NRC's evaluation of the licensee monitoring process falls, in part, under 10 CFR Part 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," which requires licensees to establish specific measures to ensure that purchased material, equipment and services conform to the procurement documents, and comply with NRC guidance in Regulatory Guide 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants," for auditing quality assurance programs.

The inspection was performed to ascertain whether licensees effectively monitored AMP's control of quality for the manufacture of safety-related pre-insulated electrical terminals and splices purchased by licensees for use in Class 1E applications inside containments of nuclear power plants. The inspector assessed specific attributes of AMP's quality control program and the scope and the licensees' monitoring of these areas.

The inspector examined AMP's quality program, including AMP's quality control organization, conformance to procurement documents, evaluation and corrective actions in response to audit findings, validation of testing and certificates of conformance, commercial-grade item dedication, Part 21, "Reporting Defects and Noncompliances," of Title 10 of the Code of Federal Regulations (10 CFR Part 21) evaluations, monitoring of sub-vendors, and self-assessment of performance.

Licensees monitored AMP through audits conducted by NUPIC audit teams comprised of licensee staff. For example, AMP was monitored by Rochester Gas & Electric based on audits performed by NUPIC on July 24, 1995 (AMP supplied electrical terminals and splices for safety-related applications at the Ginna station). Licensee monitoring was also accomplished through licensee questionnaires and telephone surveys, and through audits conducted by other nuclear organizations.

The inspector evaluated NUPIC audit reports, licensee questionnaires and telephone surveys, audits conducted by Nuclear Logistics, Inc. and ABB Power T&D Company (vendors for nuclear power plants), nonconformance reports for safety-related terminals and splices, licensee purchase orders, AMP purchase orders for sub-vendor, certification of conformance, commercial-grade process, responses to licensees' audits, AMP internal audit report, and AMP followup of NRC information notices.

3.1 AMP's Quality Control Program

In general, the AMP quality control program and its implementation were in compliance with the requirements of Appendix B to 10 CFR Part 50.

The inspector assessed AMP's quality policy, program, and standards described in AMP's corporate quality specification 102-1 Revision K, "AMP Total Quality Management Process," and supplemental specification GPBU 1-1002 Revision B, "Addendum-Total Quality Management Process." Both documents described attributes of AMP's quality program and how it conformed to 10 CFR Part 50 Appendix B and NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities." In catalog No. 82038, AMP described the safety-related preinsulated terminal and splices. (Note: As of April 1, 1996, AMP no longer accepts purchase orders imposing 10 CFR Part 50 Appendix B responsibilities but plans to maintain the existing quality program).

AMP's senior management supported its quality program, assigned responsibilities, and remained involved in the implementation of the process. The program was run by the quality manager and 6 quality auditors. The quality manager reported to the business unit director, who reported to the vice president and general manager of the Automotive/Consumer Business Group, who reported to the president/chief executive officer. The quality manager and auditors had the authority to stop production until nonconforming conditions were corrected.

The inspector asked AMP for documents addressing any design errors and failures during manufacture requiring issuance of 10 CFR Part 21 reports during the past 3 years. AMP stated that it had found no design errors or deviations that required issuing a report in accordance with 10 CFR Part 21.

The inspector requested information on any restrictions imposed by licensees on AMP for the manufacture of electrical splices purchased by licensees during the past 3 years. AMP stated that no restrictions had been imposed on it and no stop-work orders had been issued by licensees.

The inspector assessed AMP's commercial-grade dedication process. AMP purchased commercial-grade items from sub-vendors and qualified them to Appendix B requirements. The commercial-grade process comprised appropriate dedication activities, such as inspections, examinations, and witnessing of tests. Inspection findings were documented on data sheets and kept on file. Testing activities were identified in control plans and performed to test procedures. AMP evaluated test results for compliance to test requirements.

The inspector examined AMP's certificate of conformance for items purchased under licensee Purchase Order (PO) 6J515004. The certificate attested that Kynar insulated ring tongue terminals and environmentally sealed splices were qualified for use in nuclear power plants. The basis of the certificate was AMP Test Report 110-11004, Revision A; and Institute of Electrical and Electronic Engineers standards 323-1974, 344-1975, 383-1974, and Quality Specification 102-1. The inspector determined that Test Report 110-11004 qualified the test specimens mounted on terminal barrier blocks in specific configurations and for specific environmental conditions inside the containment.

The AMP quality program procedures required evaluation of sub-vendors. AMP PO 31606201 for Kynar Grade 460 identified "Ausimont USA" as an AMP sub-vendor. The inspector questioned why AMP had not monitored Ausimont. The inspector also questioned why AMP had not monitored Grant Manufacturing, a sub-vendor for tin and lead anodes, and Hewlett Packard and Quality Technical Laboratories, sub-vendors for calibration services. The 1995 NUPIC audit team also noted that AMP had not audited these sub-vendors and issued audit finding SA-95-029-03 which identified, "Lack of audits/surveys for several sub-tier suppliers, and several performed surveys lacked objective evidence." The AMP quality manager stated that AMP planned to correct the situation by auditing Ausimont, Grant Manufacturing, and the providers of the necessary calibration services in late 1996 (eventhough AMP no longer accepts purchase orders imposing 10 CFR Part 50 Appendix B responsibilities). Since this finding was identified by NUPIC, AMP corrective action is in process, it was not a willful nonconformance, and it was not a repetitive problem, this nonconformance is being treated as a non-cited nonconformance, consistent with Section VII.B.1 of the NRC Enforcement Policy.

The inspector reviewed licensee PO 31606201 for AMP preinsulated splice part No. 53550-1. The PO required AMP to identify and correct/resolve nonconforming conditions in accordance with the vendor's quality requirements and to submit any changes to the licensee for approval. The inspector verified that the safety-related terminals and splices conformed to procurement requirements. Original PO requirements were maintained to prevent inadvertent removal of requirements it was committed to meet. Certifications compared appropriately against the technical and quality requirements contained in the licensees' POs.

AMP stated that it incorporated information on pertinent programmatic and hardware changes into service letters for distribution to appropriate customers. NRC Information Notice 88-81, "Failure of AMP Window Indent Kynar Splices and Thomas and Betts Nylon Wire Caps During Environmental Qualification Testing," alerted licensees to the failure of AMP splices during licensee tests in which the splices touched each other or had a ground path. (Note: Test Report 110-11004 did not test the terminals and splices for configurations where they touched each other or had a ground path.) The inspector questioned if AMP issued any guidance to its customers addressing the failures noted in NRC Information Notice 88-81. The AMP quality manager could not identify any guidance issued to AMP's customers to address these failures.

The inspector reviewed selected AMP nonconformance reports to determine if adequate actions were taken to correct defects or weaknesses identified by AMP quality inspections or NUPIC audits. The inspector reviewed nonconformance reports for AMP terminals and splices (part Nos. 53956-1, 53409-1, and 53550-1) and determined that AMP tracked, handled, and took corrective action in response to the nonconformance reports.

The inspector assessed AMP's internal audit Report No. 032096, dated March 20, 1996. The audit identified 10 nonconformances, involving documentation deficiencies for design reviews, corrective actions, and maintenance procedures. Three corrective actions pertinent to the findings remain to be completed. Eight observations were forwarded to management, among them an observation on effectiveness of corrective actions.

3.2 Licensee Monitoring of AMP

In general, licensees effectively monitored AMP's quality control program for safety-related electrical terminals and splices.

AMP identified two NUPIC audits conducted in October 1993 and July 1995 on licensee monitoring of how well AMP controlled the quality of safety-related electrical terminals and splices purchased by licensees for use in nuclear power plants, and a 1994 NUPIC survey of AMP's process for manufacture of commercial-grade non-nuclear items. AMP also identified licensee questionnaires and telephone surveys conducted to monitor AMP.

AMP identified the following licensees to have purchased nuclear grade AMP terminals and splices qualified to 10 CFR Part 50 Appendix B during 1993 - 1996:

Licensee	Remarks
Arizona Public Service	Participated in 1993 NUPIC audit; conducted 1993 annual questionnaire.
Boston Edison	Participated in 1993 NUPIC audit.
Carolina Power & Light	-
Duquesne Light	-
Houston Lighting & Power	Conducted 1995 telephone survey.
Illinois Power	-

IES Utilities	Conducted 1993, 1994, & 1995 annual questionnaire; conducted 1996 telephone survey.
Northeast Utilities	-
Philadelphia Electric	Participated in 1993 & 1995 NUPIC audits; conducted 1996 telephone survey.
Rochester Gas & Electric	Conducted 1993, 1994, & 1995 telephone survey.
Southern Nuclear Operating	Conducted 1995 telephone survey.
Tennessee Valley Authority	Participated in 1993 NUPIC audit; conducted 1993, 1994 & 1995 annual questionnaire; conducted 1993 telephone survey.
Texas Utilities Electric	-
Washington Public Power Supply	Conducted 1993, 1994, & 1995 annual questionnaire.

(Note: NUPIC sends questionnaires to licensees to plan areas for inspection of the vendor, and makes audit results available to all licensees. Licensee use of NUPIC for auditing vendors is based on 10 CFR Part 50, Appendix B, and Regulatory Guide 1.144, which allow use of outside organizations to reduce the number of external audits as an alternative method for qualifying and monitoring vendors as long as all pertinent information is adequately evaluated.)

The inspector determined that the NUPIC audit reports were, in general, performed in accordance with written procedures and checklists. Audits comprised monitoring, witnessing, and observing activities, such as inspections, examinations, and performance tests.

Arizona Public Service led the 1993 NUPIC audit regarding the control of quality by AMP at its facility in Mechanicsburg, Pennsylvania. The audit was conducted according to the standards of ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities." The audit team uncovered no deficiencies or weaknesses, and the quality assurance program was found to be effectively implemented. However, the audit report gave insufficient detail of activities conducted by the audit team.

IES Utilities led the 1995 NUPIC audit that was performed at five AMP facilities to review the AMP's QA program "from a performance-based aspect." The team identified 12 deficiencies and made three observations. AMP responded to the findings and the licensee accepted AMP's responses. The report contained sufficient details of the review but did not state whether the audit criteria included verifying that AMP quality activities were in conformance with 10 CFR Part 50 Appendix B.

ABB Power T&D Company Inc. (ABB Power) audited its sub-vendor AMP in November 1994 to assure that AMP's quality program complied with their quality assurance programs. The audit focused on commercial-grade (non-nuclear grade) AMP products. The report gave sufficient detail of activities conducted by the audit team. No audit findings were identified.

Nuclear Logistics Inc. (NLI) audited five facilities of its sub-vendor AMP in December 1995. The audit plan addressed 10 CFR Part 50 Appendix B criteria and critical characteristics. NLI monitored compliance to 10 CFR 50 Appendix B activities. The report did not state whether AMP was in compliance with 10 CFR 50 Appendix B; however, when the inspector contacted the NLI lead auditor by phone the auditor verified that AMP was in compliance with Appendix B. No audit findings were identified.

The inspector assessed the questionnaires and telephone surveys conducted by licensees during the period assessed. In general, licensee evaluations were appropriate. Areas evaluated by licensees included potential problems or changes to AMP's product line, quality program, or procedures; purchase orders or procurement specifications; and sub-vendors, facilities, personnel, or quality involvement which could have an effect on product qualifications.

To assess the staff's conclusions on the adequacy of licensees' monitoring AMP's control of quality over the past 3 years, the inspector interviewed the following AMP staff: the quality manager, two quality control auditors, and the manufacturing team leader. In general, the AMP staff considered NUPIC audits a "plus." The quality manager stated that the 1995 NUPIC team was very detailed and did not take credit for areas reviewed during previous NUPIC audits.

4 PERSONS CONTACTED

The NRC inspector and AMP staff contacted during the inspection are listed below.

AMP Products, Inc.

* William Zelner	Quality Manager
Ruth Hershey	Quality Auditor, Team Leader
Carol Presnell	Auditor
Bill Marshall	Manufacturing Team Leader

Nuclear Regulatory Commission

* Anil S. Gautam	Team Leader, NRR

* Attended the entrance and exit meeting

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

May 31, 1996

Ms. Beth A. Barbone, Operations Manager
Cegelec Automation Inc.
2806 Metropolitan Place
Pomona, California 91767

SUBJECT: NRC INSPECTION NO. 99900734/96-01

Dear Ms. Barbone:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Cegelec Automation Inc. at Pomona, California, conducted by Mr. R.C. Wilson of this office on May 7 through 9, 1996. The purpose of the inspection was to review activities conducted under your 10 CFR Part 50, Appendix B, quality assurance program and 10 CFR Part 21 reporting program. The inspection consisted of an examination of procedures and records, interviews with personnel, and observations by the inspectors.

The NRC inspector identified that certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (Enclosure 1). Specifically, your procedure for reporting nuclear safety concerns under 10 CFR Part 21 did not address the evaluation of deviations from purchase order requirements, and the procedure did not reflect significant changes made to 10 CFR Part 21 in 1995. The violation is of concern because deviations are the type of safety concern most likely to be identified by a vendor.

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

In addition, the NRC inspector found that the implementation of your quality assurance program failed to meet certain NRC requirements. Specifically, your procedures and instructions for dedicating commercial grade items for nuclear safety-related use were inadequate to ensure that the dedicated items were suitable for the application. Furthermore, even though your actual dedication practices went beyond the requirements of the procedures and instructions, the dedication of a relay shipped for safety-related use was inadequate in two respects. The specific findings and reference to the pertinent requirements are identified in the enclosures to this letter.

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

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

Sincerely,

for Gregory C. Cwalina

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

Docket No. 99900734

Enclosures:
1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99900734/96-01

NOTICE OF VIOLATION

Cegelec Automation Inc.
Pomona, California

Docket No.: 99900734

During an NRC inspection conducted on May 7 through 9, 1996, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1992), the violation is listed below:

10 CFR 21.21, "Notification of failure to comply or existence of a defect and its evaluation," requires, in part, that each corporation subject to the regulations adopt appropriate procedures to ensure the evaluation and proper reporting of deviations and failures to comply.

Contrary to the above, Procedure 16.1, "Significant Deficiency 'Substantial Safety Hazard,'" Revision 6, of the Cegelec Quality Assurance Manual dated September 19, 1994, failed to address the evaluation of deviations. Instead the procedure, and the posted letter referring employees to the procedure, incorrectly focused on the terms "failure to comply" and "substantial safety hazard." Procedure 16.1 also failed to reflect significant changes that were incorporated into 10 CFR Part 21 in 1995.

This is a Severity Level IV violation (Supplement VII). (99900734/96-1-1)

Pursuant to the provisions of 10 CFR 2.201, Cegelec Automation Inc., is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 31st day of May 1996

Enclosure 1

NOTICE OF NONCONFORMANCE

Cegelec Automation Inc.
Pomona, California

Docket No.: 99900734

Based on the results of an inspection conducted on May 7 through 9, 1996, it appears that certain of your activities were not conducted in accordance with NRC requirements as described below.

The definition of "dedication" in 10 CFR 21.3 specifies that dedication must provide reasonable assurance that a commercial grade item to be used as a basic component will perform its intended safety function, and is deemed equivalent to an item designed and manufactured under a 10 CFR Part 50, Appendix B, quality assurance program. This assurance must be achieved by identifying the critical characteristics of the item and verifying their acceptability.

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

Section 3.2 of Procedure 3.0, "Design Control," Revision 6, of Cegelec Automation Inc. Quality Assurance Manual dated September 19, 1994, requires, in part, that specified design requirements such as regulatory requirements shall be translated into drawings and/or procedures from customer design specifications.

A. Contrary to the above, for nuclear safety-related purchase orders including Florida Power and Light Company order number 00015706 dated April 18, 1996, covering a Potter & Brumfield KRP-11AG-120 relay shipped to the Turkey Point nuclear plant on April 25, 1996, Cegelec performed dedication activities as specified in Engineering Department Work Instruction #4, "Dedication of Commercial Grade Material," Revision 3, dated April 16, 1996. This instruction did not contain sufficient detail to adequately define dedication activities, and it contained definitions that differ considerably from 10 CFR Part 21, including the statement that a critical characteristic provides reasonable assurance that the item received is the item specified. As described in the following nonconformance, dedication activities for this purchase order were in fact deficient. Dedication activities under other purchase orders examined by the NRC inspector were considered acceptable only because they included efforts not specifically required by the work instruction. (99900734/96-01-02)

B. Contrary to the above, Cegelec's dedication activities failed to establish reasonable assurance that the relay delivered to Florida Power and Light Company met the applicable requirements for its safety-related function, because there was no documented basis for the following:

1. assurance that the purchased lot of relays was homogeneous, which was a necessary requirement for the sampling plan used to select relays for verification of critical characteristics.

2. assurance that the purchased lot of relays were similar to the relays that were seismically and environmentally type-tested. (99900734-01-03)

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

Dated at Rockville, Maryland
this 31st day of May, 1996

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

REPORT NO.:	99900734/96-01
ORGANIZATION:	Cegelec Automation Inc. 2806 Metropolitan Place Pomona, California 91767
ORGANIZATIONAL CONTACT:	Eric Morales, Quality Assurance Manager 909/593-8099 ext. 235
NUCLEAR INDUSTRY ACTIVITY:	Hydrogen and oxygen containment atmosphere monitors
INSPECTION DATES:	May 7 through 9, 1996

INSPECTOR: RC Wilson 5/31/96

Richard C. Wilson, Senior Engineer — Date
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

REVIEWED BY: Gregory C Cwalina 5/31/96

Gregory C. Cwalina, Chief — Date
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

APPROVED BY: Greg C Cwalina 5/31/96

for Robert M. Gallo, Chief, — Date
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Enclosure 3

1 SCOPE OF INSPECTION:

During the 1980s Cegelec Automation, then known as Comsip Delphi, supplied containment atmosphere hydrogen analyzers for more than 40 nuclear plants, with oxygen analyzers also supplied for BWRs. Safety-related replacement parts for these systems account for about 25% of current business. Cegelec has 16 employees and occupies about 12,000 square feet.

The hydrogen monitor pulls a sample of containment atmosphere through a heated 3/8-in. line with containment isolation valves into a cabinet where the hydrogen concentration is measured in a thermal conductivity cell at about 280°F. The sample is then returned to containment through another 3/8-in. line. The system is leak tested at 90 psig.

The NRC inspectors reviewed the implementation of selected portions of Cegelec's quality assurance (QA) program, and reviewed Cegelec's 10 CFR Part 21 program. The inspection bases were 10 CFR Part 50, Appendix B, and 10 CFR Part 21.

1.1 Violation 99900734/96-01-01 (Open)

Contrary to 10 CFR 21.21(a), Cegelec did not have a procedure in place for evaluating deviations. Cegelec's procedures also did not reflect significant changes in the 1995 revision of 10 CFR Part 21. (Severity level IV violation; see Section 3.4 of this inspection report.)

1.2 Nonconformance 99900734/96-01-02 (Open)

Contrary to Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, Cegelec's dedication procedures failed to specify appropriate quality standards for the review for suitability of application of commercial grade equipment dedicated and supplied under safety-related purchase orders (POs) from licensees. The procedures lacked necessary detail and used incorrect definitions. (See Section 3.2 of this inspection report.)

1.3 Nonconformance 99900734/96-01-03 (Open)

Contrary to Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, Cegelec performed inadequate sampling of commercial grade items for testing and did not establish similarity between type test specimens and a relay delivered under a safety-related PO from a licensee. (See Section 3.3 of this inspection report.)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

There were no open findings from previous NRC inspections of Comsip Delphi, which were performed in 1981 and 1982.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Entrance and Exit Meetings

In the entrance meeting on May 7, 1996, the NRC inspector discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with Cegelec management. In the exit meeting on May 9, 1996, the inspector discussed his findings and concerns with Cegelec management.

3.2 Quality Assurance Program

The inspector selectively reviewed the Cegelec Quality Assurance (QA) Manual, Revision 7, dated September 19, 1994, the current departmental work instructions for the QA, Engineering, Procurement, and Service Departments, and several Shop Test Procedures (which also cover assembly and other operations). Cegelec's QA program was intended to meet the requirements of Appendix B to 10 CFR Part 50 for nuclear safety-related equipment. The QA manager stated that the QA program and customer PO requirements were applied to all work performed.

QA Manual Procedure 6.0, "Document Control," Revision 6, dated September 19, 1994, appeared to adequately cover control of documents such as procedures and drawings. The inspector noted that QA Work Instruction #3, "Internal Audits," Revision 7, dated November 20, 1995, did not limit the time allowed for an audited department to respond to a finding. Cegelec's QA manager agreed to specify a limit in the next scheduled revision of the QA manual by August 15, 1996.

Most of Cegelec's procurements were commercial grade. Piece parts were dedicated by receipt inspection and testing, and assemblies built from piece parts were further tested. The inspector found that Cegelec's procedures did not define a clearly acceptable commercial grade dedication process. For example, the principal dedication procedure was Engineering Department Work Instruction #4, "Dedication of Commercial Grade Material," Revision 3, dated April 16, 1996, which contained very little detail. It contained definitions that differ considerably from the 1995 revision of 10 CFR Part 21, including the statement that a critical characteristic provides "reasonable assurance that the item received is the item specified."

The definition of "dedication" in 10 CFR 21.3 specifies that dedication must provide reasonable assurance that a commercial grade item to be used as a basic component will perform its intended safety function, and is deemed equivalent to an item designed and manufactured under a 10 CFR Part 50, Appendix B, QA program. This assurance must be achieved by identifying the critical characteristics of the item and verifying their acceptability. Cegelec's dedication procedures failed to specify appropriate quality standards for the review for suitability of application of commercial grade equipment dedicated and supplied under safety-related POs from licensees, as required by Criterion III, "Design Control," of Appendix B to 10 CFR Part 50. This constitutes Nonconformance 99900734/96-01-02.

The inspector found that Cegelec's dedication practices, as reflected in detailed review of licensee purchase order (PO) files, went beyond procedural requirements and were closer to being acceptable. (However, the actual dedication practices also contained deficiencies, as described in Section 3.3 of this inspection report.) Cegelec planned to issue the next QA manual revision by August 15, 1996, and management agreed to consider the results of this inspection in performing the revision

Instructions for handling nuclear safety-related procurements, while apparently adequate, were distributed among four procedures and were difficult to locate. Procurement Department Work Instruction #2, "Control and Issuance of Purchase Orders," Revision 1, dated July 12, 1995, defined three levels of POs. Level II was defined as POs for "materials or services which include specific technical and/or quality requirements." POs for "critical materials or services are only placed with suppliers found on the CEGELEC Automation Inc. Qualified Vendors List." Critical material and service suppliers were defined as suppliers of safety-related or qualified equipment, special process (welding, NDE), or calibration services. QA Work Instruction #14, "Supplier Evaluation and Approval, Revision 3, dated December 23, 1994, required triennial vendor surveys of suppliers of "Critical Materials and Services." Vendor surveys required "direct observation of objective evidence at the suppliers facility which substantiates the documented program." Source inspection on an individual order was permitted in lieu of a survey. QA Manual Procedure 4.0, "Procurement Document Control, Revision 6, dated September 19, 1994, and Procurement Work Instruction #2 both required imposing 10 CFR Part 21 in safety-related orders; Procurement Department Work Instruction #3, "Spare Parts Procurement and Order Entry, Revision 6 dated March 30, 1995, required imposing Part 21 "as applicable." The Cegelec QA manager agreed to consider clarifying the treatment of safety-related POs by the August 15, 1996, revision of the QA manual.

The inspector concluded that the reviewed QA manual, work instructions, and procedures were satisfactory, except for the 10 CFR Part 21 violation discussed in Section 3.4 of this inspection report and the two dedication nonconformances described in Sections 3.2 and 3.3.

3.3 Review of Licensee Purchase Orders

The inspector selected five licensee PO files for review, covering the following items purchased commercial grade and supplied safety grade: a relay purchased as a commercial grade component; a hydrogen sensing cell manufactured from commercial grade piece parts; a pressure regulator purchased as a custom variation of a commercial grade component; and a flow regulator manufactured from commercial grade piece parts. Although no hardware had yet been processed for the fifth licensee PO file reviewed, it provided an example of Cegelec's identifying and correcting errors in licensee POs.

Turkey Point:

Florida Power and Light Company (FPL) PO No. 00015706 dated April 18, 1996, ordered a replacement Potter & Brumfield KRP-11AG-120 relay for the Turkey Point nuclear plant. The PO stated that it was FPL Nuclear Safety Related

(PC-1), and it invoked 10 CFR Part 21, IEEE 323-1974 mild environment, and IEEE 344-1975. A Certificate of Compliance was also required. The Cegelec Contract Review form summarized the PO requirements and recorded Cegelec's sales order numbers for the original system and the replacement part order. The Sales Order form identified that the relay supplied on the replacement sales order 44942 had been purchased on Cegelec PO No. 1230-849. A copy of that PO was in the file, showing that 25 KRP-11AG 120 volt relays had been procured from the Electronic Supply company, a local distributor. Cegelec drawing 31041-17 Rev. 7 specified the following critical characteristics: part/model number, overall dimensions, maximum pull-in and minimum drop-out voltages, and energized and de-energized contact hi-pot testing.

The Receiving Inspection Report addressed simple checks such as physical damage and count. The Material Dedication Inspection/Test Report, also completed by the receiving inspector, stated that 3 of the 25 KRP-11AG relays were tested to verify the critical characteristics specified on the drawing. The sample size was prescribed in QA Work Instruction #1, "Receiving Inspection," Revision 11, dated July 12, 1995, and based on MIL-STD-105D. The 3 relays tested were not specifically identified. For each characteristic, "Acceptable" was recorded. The test equipment was identified by serial number and calibration due date (although the inspector did not pursue calibration traceability, Cegelec's procedures require triennial audits of calibration service suppliers, which the QA manager stated were performed). After receiving testing, the lot of 25 relays was placed in controlled storage with a green tag designating safety-related material.

The invoice from Electronic Supply included a printed "Certificate of Compliance" stating that all materials and/or parts are in accordance with the applicable manufacturer's specifications, and that inspection records and test data are on file and available for review at the manufacturer's plant. This C of C was signed by the Director of Operations. Cegelec did not audit Electronic Supply or Potter & Brumfield. Cegelec provided to FPL a Certificate of Conformance specifying the shelf life of the relay and stating that it was the same in form, fit, and function and interchangeable with items supplied under the original FPL specification, and that it satisfied IEEE 323-1974 and IEEE 344-1975 as documented in the qualification test report for the original analyzer system.

The inspector had two principal concerns with the dedication process for the relay:

(1) There was no assurance that the lot of 25 relays was a homogeneous population, as required for the validity of the sampling method used. Procurement from an unaudited commercial grade distributor, with no attempt to either control or identify characteristics such as date code or serial number, and with no audit of or correspondence with the manufacturer, provides little basis for concluding that the 25 relays were essentially the same. Nuclear industry experience with relays from this particular manufacturer does not provide the necessary confidence, and the certification from the unaudited supplier is of little value for dedication purposes.

(2) there was no assurance that any of the replacement relays were sufficiently similar to those that were environmentally and seismically type-tested in 1980-82; i.e., the dedication process failed to establish traceability of design similarity from the replacement relays to the type-test specimens.

Because of these concerns, dedication of the Potter & Brumfield relay for Turkey Point did not satisfy Criterion III, "Design Control," of Appendix B of 10 CFR Part 50, which requires appropriate quality standards for the review for suitability of application of commercial grade equipment dedicated and supplied under safety-related POs from licensees. These two examples of inadequate design control constitute Nonconformance 99900734/96-01-03.

The sampling concern could be overcome with 100% sample testing of the replacement relays. The problem with similarity to type-test specimens could be addressed by disassembly and comparison if the original test specimens or detailed records were still available; possibly the manufacturer's records could be useful. Procurement of a demonstrably homogeneous lot and performing new type tests on samples from the lot is another possibility.

Palo Verde:

Arizona Public Service Company PO No. 60282071 dated January 4, 1996, ordered (along with other parts) a replacement hydrogen sensing cell, Cegelec type 1427-B5, for the Palo Verde Nuclear Generating Station. The PO specified the qualification report for the original system, invoked 10 CFR Part 21, invoked the current Cegelec QA manual and licensee QA Requirement 003, and stated that all items are safety-related. A Certificate of Conformance to the requirements of the PO was also required.

Applicable Cegelec documents included the "Bill of Materials for Assembly Number 1427-B5," Revision 4, dated January 18, 1994, and Procedure 10.11, "B5 Thermal Conductivity Cell Block Assembly and Test Procedure," Revision 3, dated January 16, 1995. The cell is manufactured by Cegelec from commercial grade piece parts. A Material Requisition form for Sales Order 58265 identified the PO for each of the 22 parts on the Bill of Material. Most parts were stainless steel, and the part drawing typically specified two critical characteristics: dimensions, and material as verified by test procedure 11.1, "Material Verification Test Procedure," Revision 1, dated December 20, 1990. This procedure also specified MIL-STD-105D sampling (typical samples were 32 of 530 diffusion discs and 2 of 8 cell blocks). Chemical and physical mill certifications were typically obtained for steel parts. The procurements were from distributors which Cegelec did not audit.

Procedure 11.1 uses a Koslow 1599 alloy test kit to identify stainless steel when grade is not critical, or when type 316 is specified. In either case, electrical conductivity of the specimen is measured. A test solution color test for high chromium content typical of 300 and 400 series stainless steels is used for all types, and a second solution tests for approximately 3% molybdenum content for type 316 stainless. A Koslow 1900 metal standard set also provides material samples for visual comparison. Procedure 11.1 further specifies a flame test for inorganic materials, typically used for O-rings.

The NRC inspector found that some parts drawings did not specify critical characteristics. An example was drawing 1018-C, "Model B5 O-ring Spacer Catalyst Bed," Revision 0, dated April 2, 1990. In these cases the Spare Parts Inspection Reports nonetheless showed dimensions and material as critical characteristics, and sampling test results were recorded. The QA manager explained that some older drawings had not been updated to define critical characteristics, but the receiving inspector would still perform typical tests. The NRC inspector also found no record of receipt testing for two parts, a hermetic seal and a resistor. The QA manager stated that both of these parts were dedicated during assembly and testing of the cell; the hermetic seal was tested during the pressure test and electrical testing, and the resistor value and temperature response were verified during electrical balancing of the cell. In these cases the NRC inspector again noted that Cegelec's actual performance went beyond the content of procedures, which did not address these variations.

The NRC inspector again believed that the mill certifications and limited testing of sampled parts were not alone sufficient to dedicate the parts. However, the assembly operations and testing performed under procedure 10.11 (including a one minute pressure test at 90 psig), with the other activities performed by Cegelec and the necessity for the customer to calibrate the cell after installation, are considered sufficient for dedication.

Perry:

Cleveland Electric Illuminating Company PO No. S 139804 dated December 13, 1995, ordered two replacement GH41X1-1833-1 pressure regulators for the Perry Nuclear Power Plant. The PO specified safety related procurement level I, required a Certificate of Conformance, invoked 10 CFR 50 Appendix B and 10 CFR Part 21, and referenced the original system PO, specification, and qualification test report. The regulator is a version of a standard Conoflow model, custom engineered for Cegelec, and is procured commercial grade under Cegelec drawing 31041-27, "Downstream Pressure Regulator Set at 3 psig, Conoflow Mdl. # GH41XT--1833-1," Revision 11, dated April 18, 1995. The inspector did not review the Drawing Control Change Notices for this drawing.

The two regulators supplied to Perry had been procured by Cegelec under two different POs, 30-808 and 30-823, from A. Biederman, Inc., a local representative. The drawing specified part/model number and overall dimensions as critical characteristics, which were checked on 5 samples in each lot of 50. For PO 30-808 a damaged thread on one sample caused all 50 to be checked. The drawing also specified 100% testing of all regulators to Procedure 6.10, "Model GH41XT-1833-1 Differential Pressure Regulator Bench Test Procedure," Revision 5, dated February 9, 1995. This procedure includes a ten minute leak test at 90 psig and functional performance testing. The inspector concluded that the dedication activities for this regulator were satisfactory, considering the application.

D.C. Cook:

American Electric Power PO. No. 66069-040-6X dated February 16, 1996, ordered (along with other parts) two replacement model 11727 R-1 downstream flow regulators for the Cook Nuclear Plant. The PO stated that it was nuclear safety-related, it invoked 10 CFR Part 21, and it required certification that the part is equal to or better than the original. The regulator is built according to drawing 11727, "Sample Flow Regulator Rl KIII and KIV," Revision 10, dated May 2, 1995. The inspector did not review the Drawing Control Change Notices for this drawing. The drawing notes specified the applicable procedures for assembly and test: Procedure 10.6, "Cegelec Automation Inc. Model 11727 Series (R_1) Downstream Pressure Regulator," Revision 8 dated May 1, 1995, and Procedure 6.4, "Model 11727 Series (R_1) Downstream Pressure Regulator Bench Test Procedure," Revision 7, dated May 1, 1995. In addition, parts drawings checked by the inspector typically identified critical characteristics as part/model number, dimensions, and material verification per procedure 11.1.

The inspector checked the procurement of the regulator springs, which were ordered to drawing 11768, "Regulator Spring for Rl," Revision 2, dated December 12, 1989. The springs were ordered on PO No. 1270-209 from Superior Spring Company. Although no critical characteristics were specifically identified, the drawing did specify AMS-5673 or AMS-5678, Grade 17-7 stainless steel, 0.035 inch wire diameter, and it specified a spring force (mislabeled "pressure") at a specific working height as measured in the test fixture shown on the part drawing. Superior Spring, an unaudited commercial grade supplier, provided certification of the following: 0.035 17-7SS material, specification AMS 5678, passivated, heat treated 900 deg./1 hr., specifications MIL-S-5002 and CH 900/MIL-H-6875. The Material Verification Test Report for the PO showed that a sample of 8 of 100 springs was tested for dimensions and material (stainless steel per procedure 11.1). The file gave no indication that the spring force was tested, even though the test fixture specified on the drawing was a simple cylinder.

The Procedure 6.4 test reports for the two regulators supplied to Cook showed that both passed the ten minute 90 psig leak test, and both maintained an acceptable vacuum for inlet pressures up to 60 psig, with acceptable repeatability. Even though critical characteristics for the piece parts were not consistently specified or verified, the inspector concluded that final testing adequately dedicated the regulator for its application.

Millstone:

Northeast Nuclear Energy Company PO No. 956726 dated November 1, 1995, ordered two model 11727 and two model 11728 pressure regulators for the Millstone Nuclear Power Station. Cegelec had not yet supplied equipment under this PO, but had sent a letter to the licensee correcting three errors in the PO: the number of the original system PO, the date of Cegelec's QA manual, and the form number of the attachment designating QA requirements including nuclear safety-related. (The inspector noted that, alternately, the wrong PO note and correct form may have been included.) These corrections exemplify proper implementation of QA Manual Procedure 19.0, "Contract Review," Revision 0, dated September 19, 1994. Correction of such errors prior to beginning work on a PO reduces the chance that incorrect safety-related equipment may be supplied.

3.4 10 CFR Part 21 Program

The inspector reviewed Cegelec's procedure for reporting in accordance with 10 CFR Part 21: QA Manual Procedure 16.1, "Significant Deficiency 'Substantial Safety Hazard'," Revision 6 dated September 19, 1994. Both the procedure, and the letter referring to the procedure that was posted on a bulletin board, incorrectly focused on the terms "failure to comply" and "substantial safety hazard." Although the procedure correctly repeated the definition of "deviation" from 10 CFR 21.3, it did not address Cegelec's responsibility to evaluate deviations. Instead, Procedure 16.1 addressed the evaluation by Cegelec of "failures to comply" as possible substantial safety hazards. However, 10 CFR Part 21 uses the term "failure to comply" in two senses, equivalent to either deviations or defects, and Cegelec used the wrong sense, equivalent to defect. Procedure 16.1 (as well as Cegelec work instructions for commercial grade item dedication discussed in Section 3.3 of this inspection report) also failed to reflect significant changes in the 1995 issue of 10 CFR Part 21.

The inspector explained that Cegelec must have procedures for evaluating deviations and failures to comply under 10 CFR 21.21(a). Since Cegelec would rarely be in a position to perform the required evaluation, the inspector suggested that the procedure focus on satisfying this requirement by notifying customers as allowed by Section 21.21(b) if the vendor is unable to perform the evaluation. The inspector also asked that Cegelec's revised procedure encourage employees to report possible deviations, either formally or informally, to their supervisors or QA. The inspector found no evidence that deviations requiring evaluation had ever gone unevaluated or unreported to licensees, and only one Part 21 concern had been reported (that by a licensee, as discussed below) since initial qualification testing in 1980-82.

Failure to have procedures that require the evaluation or reporting of deviations, as required in 10 CFR 21.21(a), and failure to incorporate the significant changes of the 1995 revision of 10 CFR Part 21, constitute Severity Level IV violation 99900734/96-01-01.

The inspector reviewed a 10 CFR Part 21 concern initially reported by the Hope Creek Generating Station of the Public Service Electric and Gas Company by telephone on July 7, 1993, involving a higher-than-analyzed ambient temperature for microswitches used in hydrogen analyzers. The concern was subsequently addressed in written Part 21 reports from Hope Creek on August 5, 1993, and from Cegelec on March 2, 1994. The NRC log numbers for these reports are 93-294, 93-323, and 94-099. The reported concern was that Microswitch model PTW-5300 switches installed at Hope Creek were experiencing unspecified damage because their operating temperature was high. The licensee postulated that humidity could cause a short circuit that would disable a safety-related display, and stated that the qualified life of the switches was shortened. The Cegelec report stated that the Hope Creek switches were being operated more than 10% above their rated voltage, and noted that when the voltage was reduced to the rated 120 vdc, the temperature decreased by 44 F°. During this inspection the inspector verified that the concern is not generic.

The Ametek/Dixson company submitted a 10 CFR Part 21 report dated April 18, 1996, concerning bargraph indicators supplied to Cegelec and other customers. The inspector verified that Cegelec used the indicators only for a foreign customer; none were provided for domestic use.

3.5 Industry Group Audit

The inspector reviewed the reports of an industry group audit of Cegelec in January 1995, and a follow-up surveillance in May 1995. The surveillance report showed that the four audit findings had been corrected. The inspector noted that the industry group audit had thoroughly covered conformance to manufacturing procedures, which the NRC inspector did not have the opportunity to do. The inspector also noted that the industry group audit report stated that Cegelec had developed and implemented appropriate procedures for the evaluation of Part 21 conditions, although that was not the case. The industry group audit otherwise appeared to be satisfactory.

4 PERSONNEL CONTACTED

+ * Beth A. Barbone, Operations Manager
+ * Eric Morales, Quality Assurance Manager
+ * Scott E. Crail, Engineering Manager
Karen Cobb, Spare Parts
Kevin Mullane, Receiving Inspector

+ Attended the entrance meeting on May 7, 1996
* Attended the exit meeting on May 9, 1996

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

May 28, 1996

Mr. Arthur J. Spencer, Manager
ASME Codes and Standards
Factory Mutual Engineering Association
1151 Boston-Providence Tpke.
Norwood, MA 02062

SUBJECT: NRC Inspection No. 99900603/96-01

Dear Mr. Spencer:

This letter addresses the U.S. Nuclear Regulatory Commission (NRC) inspection of your offices in Bala-Cynwyd, PA on April 22 through 24 and in Norwood, MA on April 25 and 26 by Messrs. U. Potapovs, R. McIntyre, and S. Matthews and the discussions of their findings with Mr. S. Rudnickas, you, and members of your staff at the conclusion of the inspection. The inspection was conducted to evaluate the implementation of your quality assurance program in selected areas related to the performance of activities under the scope of American Society of Mechanical Engineers (ASME) Certificate of Accreditation issued to Arkwright Mutual Insurance Company. The scope of this inspection was focused on the Authorized Nuclear Inspector (ANI) and the Authorized Nuclear Inspector Supervisor (ANIS) services that your company provided to Amer Industrial Technologies, Inc. (AIT). Your program for compliance with the requirements of Part 21 of Title 10 of the Code of Federal Regulations (10 CFR Part 21) was also reviewed during this inspection.

The inspection was accomplished through objective evaluation of selected procedures and records and discussions with appropriate staff. The specific areas examined during the NRC inspection and the findings are discussed in the enclosed inspection report.

Based on the results of this inspection, we determined that certain of your activities appeared to be in violation of NRC requirements. Specifically, your procedure CP 110.00, "Title 10 Code of Federal Regulations Part 21", dated January 19, 1996, which defines the policy for 10 CFR Part 21 compliance, did not provide for evaluation of deviations identified as a result of your activities at nuclear facilities and at other covered entities that manufacture basic components.

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

Additionally, during this inspection, we identified instances where the implementation of your quality assurance program failed to fully comply with the ASME Code requirements that are applicable to your activities under the scope of the Certificate of Accreditation issued to Arkwright Insurance Company dba Factory Mutual Engineering Association. The specific findings and references to the pertinent requirements are identified in the enclosures to this letter.

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

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

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

Sincerely,

Gregory C. Cwalina

Gregory C. Cwalina, Acting Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No.: 99900603

Enclosures:
1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99900603/96-01

NOTICE OF VIOLATION

Factory Mutual Engineering Association
Norwood, MA 02062

Docket No.: 99900603
Report No.: 96-01

During an NRC inspection conducted at your Bala-Cynwyd, PA and Norwood, MA facilities on April 22 through 26, 1996, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

10 CFR 21.21(a) states, in part, that each corporation subject to the regulations shall adopt appropriate procedures to ensure the evaluation of deviations within 60 days of discovery, the submittal to NRC of an interim report if the evaluation can not be completed within 60 days, and the reporting to a responsible official of a defect or a failure to comply related to a substantial safety hazard within 5 working days of completing the evaluation.

Contrary to the above, Factory Mutual Engineering (FMEA) Contract Procedure CP 110.00, "Title 10 Code of Federal Regulations Part 21," dated January 19, 1996, which defines FMEA's policy for compliance with 10 CFR Part 21 did not: (1) provide for evaluation of deviations identified as a result of FMEA's activities at nuclear facilities and at other covered entities that manufacture basic components, (2) ensure that, if an evaluation of deviations can not be completed within 60 days, an interim report is submitted to the Commission, or (3) ensure that a director or responsible officer of FMEA is informed within 5 working days of the completion of the evaluation.

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

Pursuant to the provisions of 10 CFR 2.201, FMEA is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Chief, Special Inspection Branch, Division of Technical Support, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Rockville, Maryland
this 28th day of May, 1996

NOTICE OF NONCONFORMANCE

Factory Mutual Engineering Association, Inc.
Norwood, Massachusetts

Docket No.: 99900603
Report No.: 96-01

Based on the results of an NRC inspection conducted on April 22 through 26, 1996, it appears that certain of your activities were not conducted in accordance with The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) requirements that are applicable to your activities under the scope of the Certificate of Accreditation issued to Arkwright Mutual Insurance Company dba Factory Mutual Engineering Association.

A. Paragraph NCA-5220, "Categories of Inspector's Duties," of Subsection NCA, "General Requirements for Division 1 and Division 2," of Section III, "Rules for Construction of Nuclear Power Plant Components," (Section III) of the ASME Code, requires, in part, that the Authorized Nuclear Inspector (ANI) perform all duties specifically required in ASME N626-1990, "Qualifications and Duties for Authorized Nuclear Inspection Agencies and Personnel," and addenda through N626b-1992.

Section 0-3, "The Authorized Nuclear Inspector," Subsection 0-3.2, "Duties," Paragraph 0-3.2.18 of ASME N626b-1992, required, in part, that the ANI shall keep a bound diary of activities and inspections made. The information to be recorded shall include a description of the item inspected, the type of observation made, the requirements that prompted the activity, and the results of inspection.

Subsection O, "Documentation," paragraph 3, "Inspector's Diary," of Factory Mutual Engineering Association (FMEA) Contract Procedures (CP) 106.00, "ASME Code Activity - Authorized Inspector (AI/ANI)," Revision 0, required, in part, that the information recorded include the materials that were checked for identification, documentation verified or reviewed, the fit-ups and dimensional checks made, procedures verified or reviewed, qualifications verified or reviewed, the number of the QA/QC Program Monitoring Report, and a brief description of the monitoring activity and any deficiency.

Contrary to the above requirements, the ANI's diary entries for the inspection activities performed at Amer Industrial Technologies (AIT) during June 1993 through June 1995 did not, for the most part, include the inspection results, the type of observations made, and the requirements that prompted the activity. (96-01-01)

B. Paragraph NCA-5290, "Certification of Data Reports and Construction Reports," of Subsection NCA of Section III of the ASME Code required, in part, that the appropriate Data Reports shall be certified by the ANI after satisfying himself that all requirements of Section III of the ASME Code have been met.

Subsection O, "Documentation," paragraph 2, "Manufacturer's Data Report," of FMEA CP 106.00, "ASME Code Activity - Authorized Inspector (AI/ANI)," Revision 0, required, in part, that the ANI shall verify that all other applicable ASME Code requirements have been satisfied before signing the Data Reports.

Contrary to the above requirements, Section V, "Close Out of QA/QC Program Monitoring Report," (MR) of FMEA CP 106.00 failed to establish adequate measures that ensure that all MRs that document unsatisfactory conditions have been resolved by the Certificate Holder to the ANIs satisfaction before the Data Reports are certified by the ANI.

For example, MR 93-10 written against AIT's job 331 was issued on June 14, 1993, and identified that (1) the route sheet did not identify all the AIT procedures to be used and their revision level, and (2) AIT's purchase order for the tubes did not contain all the requirements of its QA manual. However, MR 93-10 was not closed by the ANI and AIT until May 31, 1995, even though the ANI certified the Data Reports for job 331 in February, 1993.

Additionally, FMEA CP 106.00 failed to establish adequate measures for the use, control, and distribution of the MRs. (96-01-02)

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Special Inspection Branch, Division of Inspection and Support Programs, Office of Nuclear Reactor Regulation, within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include for each nonconformance: (1) the reason for the nonconformance, or if contested, the basis for disputing the nonconformance, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further nonconformances, and (4) the date when your corrective action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 28th day of May, 1996

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

REPORT NO.:	99900603/96-01
ORGANIZATION:	Factory Mutual Engineering Association (FMEA) 1151 Boston-Providence Tpke Norwood, MA 02062
ORGANIZATIONAL CONTACT:	Arthur J. Spencer, Manager ASME Codes and Standards
NUCLEAR INDUSTRY ACTIVITY:	FMEA is a wholly owned and operated subsidiary of Allendale, Arkwright, and Protection Mutual Insurance Companies and provides authorized inspection services under the scope of ASME Certificates of Accreditation held by these insurance companies.
INSPECTION DATES:	April 22 through 26, 1996
LEAD INSPECTOR:	Uldis Potapovs 05-28-96 Uldis Potapovs — Date Vendor Inspection Section (VIS) Special Inspection Branch (PSIB) Division of Inspection and Support Programs Office of Nuclear Reactor Regulation
OTHER INSPECTORS:	Richard P. McIntyre, VIS/PSIB Steven M. Matthews, VIS/PSIB
REVIEWED BY:	Gregory C. Cwalina 5/28/96 Gregory C. Cwalina, Chief — Date Vendor Inspection Section Special Inspection Branch Division of Inspection and Support Programs Office of Nuclear Reactor Regulation
APPROVED BY:	Gregory C. Cwalina 5/28/96 Gregory C. Cwalina, Acting Chief — Date Special Inspection Branch Division of Inspection and Support Programs Office of Nuclear Reactor Regulation

Enclosure 3

1 SUMMARY OF INSPECTION FINDINGS

The scope of this inspection was limited to authorized nuclear inspection services provided by Factory Mutual Engineering Association (FMEA) under the scope of ASME Certificate of Accreditation issued to Arkwright Mutual Insurance Company (Arkwright) doing business as FMEA. More specifically, the inspection focused on the Authorized Nuclear Inspector (ANI) and the Authorized Nuclear Inspector Supervisor (ANIS) services provided to Amer Industrial Technologies, Inc. (AIT) and FMEA's activities related to administering and controlling these services under their Quality Assurance Manual (QAM). The inspection included interviews of the ANI and ANIS assigned to AIT, review of indoctrination and training of personnel, internal and external audits, and the implementation of Title 10 Code of Federal Regulations, Part 21 (Part 21) requirements.

The inspection basis consisted of the following:

- FMEA QAM, Third Edition-1995, Revision 1.
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).
- ASME N626-1990, "Qualifications and Duties of Authorized Nuclear Inspection Agencies and Personnel."
- 10 CFR Part 21.

1.1 Violations

Contrary to 10 CFR 21.21, which states that corporations subject to this regulation must adopt appropriate procedures to ensure evaluation of deviations for reportability under this rule, FMEA procedure CP 110.00, "Title 10 Code of Federal Regulations Part 21," did not provide for evaluation of deviations identified as a result of FMEA's activities at nuclear facilities and at other covered entities that manufacture basic components. See Section 3.1 of the report. (99900603/96-01-01)

1.2 Nonconformances

- Nonconformance 99900603/96-01-02 was identified and is discussed in Section 3.3.1 of this report.
- Nonconformance 99900603/96-01-03 was identified and is discussed in Section 3.3.1 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of FMEA activities performed under an ASME Certificate of Accreditation.

3 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance and Exit Meetings

The entrance meeting was held on April 22, 1996 at FMEA Philadelphia district offices in Bala-Cynwyd, PA. A preliminary exit meeting was held at this location on April 24, 1996. The inspection was continued at FMEA Norwood, MA offices on April 25, and the inspection findings and observations were discussed with FMEA management during the final exit meeting on April 26, 1996.

3.2 10 CFR Part 21 Program

The inspectors reviewed FMEA Contract Procedure (CP) 110.00, Revision 0, dated January 19, 1996, "Title 10 Code of Federal Regulations Part 21," which describes FMEA's policy for achieving compliance with this regulation. The scope of this procedure addresses actions required to be taken by FMEA personnel performing activities at nuclear facilities when a condition "adverse" to quality could qualify as a reportable defect or deviation under 10 CFR Part 21. The procedure is limited to services performed at nuclear plants and does not address the handling of potentially reportable conditions at manufacturing shops (ASME Certificate Holders) serviced by FMEA.

In summary, CP 110.00 requires FMEA personnel to notify appropriate management of potentially reportable conditions unless the FMEA personnel have been advised that the plant management (NRC licensee) is already aware of the condition and documented evidence exists that NRC has been notified, action has or will be implemented to resolve the condition, or the plant's evaluation indicates that the situation is not covered under 10 CFR Part 21. The procedure also notes that "Evaluations as to whether or not a defect or deviation will result in a substantial safety hazard, is beyond the scope of responsibility of Factory Mutual personnel."

The inspectors determined that CP 110.00 did not adequately provide for the implementation 10 CFR Part 21.

(a) The scope of CP 110.00 was limited to situations involving FMEA personnel performing activities at nuclear facilities. It did not recognize that deviations, as defined in the regulation, may also result from FMEA activities related to service contracts at manufacturing shops (Certificate Holder facilities)

(b) The scope of CP 110.00 did not require FMEA to evaluate deviations or failures to comply or to inform the affected purchasers/licensees when such deviations or failures to comply are the direct result of services provided by FMEA to Certificate Holders or licensees. For example, the procedure did not address situations where the ANI inspection activities are found to be in noncompliance with the technical requirements of the purchase documents after the performance of the contracted services. An example would be an after-the-fact identification that an ANI providing services to a nuclear plant or Certificate Holder lacked the required qualifications.

Failure to adopt appropriate procedures to evaluate deviations was identified as a violation of 10 CFR Part 21.21. (99900603/96-01-01)

3.3 Quality Assurance Program Implementation

The FMEA quality assurance program is described in their QAM for Contract Inspection Services, Third Edition-1995, Revision No. 1. The manual contains endorsements and authorization statements by senior officers of the three insurance companies that make up the Factory Mutual System and affirms compliance with ASME QAI-1, "Qualifications for Authorized Inspection" and with ASME Code Section I, III, IV, VIII, X, and XI activities. The program implementation was examined by reviewing selected records and interviewing appropriate FMEA staff. The implementation review was focused on ANI and ANIS services provided to AIT during the 1993-1995 timeframe.

3.3.1 FMEA Activities at AIT

During the NRC's inspection of AIT, conducted on January 29 through February 2, 1996 (NRC Inspection Report 99901292/96-01, dated March 21, 1996), the inspection team reviewed documentation for four AIT jobs (331, 392, 442, and 4102) that had been recently completed and shipped to NRC licensees. These four jobs had been inspected by ANIs employed by FMEA.

Although the ASME Data Reports for these jobs were certified by ANIs indicating that all ASME Code Section III requirements had been met by AIT, the NRC inspection determined that none of the four jobs completely met all ASME Code Section III requirements. Therefore, during the course of this inspection, the team reviewed FMEA's documentation supporting the determination that the certified items complied with the applicable ASME Code requirements. A significant portion of this review was focused on FMEA ANI bound diaries that were maintained for all inspections conducted at AIT.

Based on an evaluation of the bound diaries, the team concluded that the ANIs had performed extensive inspection and verification activities from pre-fabrication through final testing and Data Report certification. A significant amount of inspection activity was documented for all four jobs. However, the team concluded that the individual entries in the bound diary lacked sufficient documentation to substantiate the various and numerous ASME Code compliance issues identified and communicated to FMEA management by the ANIs regarding AIT's performance. The diaries were typically annotated in short, cryptic notes that were not always clear about the specific inspections or verifications performed nor documented the results of these inspections or verifications. The team's evaluation of the bound diary entries and its review of specific issues with the ANIs during an interview, determined that the entries failed to document the inspection results, the type of observations made, and the requirements that prompted these activities.

Requirements to record the above information are imposed by Subsection O, "Documentation," paragraph 3, "Inspector's Diary," of FMEA CP 106.00, "ASME Code Activity - Authorized Inspector (AI/ANI)." These requirements were also imposed on FMEA through its ASME accreditation as an Authorized Inspection Agency (AIA) in accordance ASME N626-1990, "Qualifications and Duties for

Authorized Nuclear Inspection Agencies and Personnel," and paragraph NCA-5220, "Categories of Inspector's Duties," of Subsection NCA, "General Requirements for Division 1 and Division 2," of Section III of the ASME Code.

The team concluded that FMEA's ANI bound diary entries did not meet the requirements of its CP 106.00, ASME Code NCA-5220, and N626b-1992. Failure to demonstrate conformance with the applicable requirements and duties for ANIs was identified as Nonconformance 99900603/96-01-02.

Additionally, during the team's evaluation of the bound diaries, the team noted that the ANIs had issued several QA/QC Program Monitoring Reports (MRs) in the course of performing inspection activities at AIT. As provided for in CP 106.00, the MRs should be completed by the ANIs for (1) any finding not completed before the ANI left the facility, or (2) on completing monitoring of one or more sections of the QA/QC manual.

The bound diary contained an entry on June 14, 1993, that MR 93-10 was issued on AIT job 331 which noted that (1) the route sheet did not identify all the AIT procedures used to meet ASME Code requirements and their revision level, and (2) AIT's purchase order for the tubes did not contain all the requirements of its QA manual.

The Data Reports for job 331 were certified by the ANI on February 4 and 8, 1993, indicating that all ASME Code requirements were met by AIT. However, MR 93-10 was not resolved by AIT or accepted by the ANI until May 31, 1995.

The team's evaluation of CP 106.00 determined that it failed to require MRs to be satisfactorily closed before the ANI certifies the Data Report. Therefore, open deficiencies may exist at the time the ANI certifies the Data Report. The team also concluded that CP 106.00 failed to establish measures for the use, control, and distribution of the MRs, although the team did note that distribution was described on the backside of the MR.

The team concluded that FMEA failed to establish measures to assure that all deficiencies identified on MRs were satisfactorily closed before the ANI certified the Data Reports. The team also concluded that FMEA had not adequately addressed the use, control, and distribution of MRs. Failure to establish these measures was identified as Nonconformance 99900603/96-01-03.

3.3.2 Interfaces with Arkwright Mutual Insurance Company

The team determined that the inspection services contract for AIT was actually signed with Arkwright Mutual Insurance Company (Arkwright). The ASME Certificate of Accreditation for Arkwright to perform these inspection services as an accredited AIA states the following:

Arkwright Mutual Insurance Company
Mutual Boiler Division
dba Factory Mutual Engineering Association
225 Wyman Street
Waltham, Massachusetts 02254-9198

Review of correspondence files related to the AIT contract indicated that FMEA's ANIs assigned to AIT, its District Office in Bala-Cynwyd, Pennsylvania, and the FMEA Home Office in Norwood, Massachusetts, had, since 1993, repeatedly advised Arkwright that the inspection services contract with AIT be dropped. According to FMEA, its request to drop AIT's inspection services was based, in part, on AIT's consistently poor performance in achieving ASME Code compliance. However, Arkwright was not receptive to FMEA's repeated request and did not cancel the contract with AIT until 1995.

The ANI's responsibilities are considered by the ASME Code to be third party oversight of a Certificate Holder that is assumed to diligently comply with the ASME Code requirements. FMEA's correspondence files reviewed during the inspection, however, identified that the normal assumptions and principles of third party oversight did not apply at the AIT fabrication facility. The documentation and interviews with the ANI and ANIS indicated that during the period from 1993 through 1995, the ANIs had to exercise extreme diligence in the performance of inspections and verifications because the lack of ASME Code compliance was pervasive throughout AIT's fabrication activities. In September 2, 1995, correspondence to Arkwright, FMEA's management noted that the services contract with AIT was to provide qualified inspectors to verify compliance rather than to provide shop quality control personnel and suggested that, if quality control services were to be provided, a separate agreement should be developed to cover such activity.

After reviewing FMEA's reports, records, ANI bound diaries and interviewing the ANIs the team concluded that the ANIs had very little confidence in AIT's ability to comply with ASME Code requirements. Therefore, the basic assumption that their oversight activities were adequate to verify with reasonable assurance that every aspect of ASME Code compliance had been achieved by AIT was no longer valid. Although FMEA's management reiterated to the ANIs not to certify Data Reports where they knew that ASME Code compliance was not met, the ANI's could not be expected to verify every detail of the manufacturing process and, under the existing conditions, it was not reasonable for the ANI's to assume that all details that were not verified complied with ASME Code.

The previously discussed NRC findings during its inspection of AIT, described in NRC Inspection Report 99901292/96-01, fully support the teams conclusion that the ANIs could neither verify every detail of the manufacturing process nor assume that those details not verified complied with the ASME Code.

During an interview with Arkwright's management at FMEA's Norwood, MA offices, the team briefly discussed Arkwright's basis for maintaining the contract with AIT, and how its control of the AIT services contract met the ASME Certificate of Accreditation provision that Arkwright was accredited doing business as Factory Mutual Engineering Association when, in fact, FMEA did not have the authority to cancel the contract. Because of unavailability of key Arkwright personnel to adequately address these issues, the team determined that it would be appropriate to develop more complete information during a future follow up inspection of Arkwright.

3.3.3 Audits and Inspector Performance Evaluations

The inspectors reviewed QAM Section 9, "Audits," and CP 108.00, "ASME National Board Audits and Evaluations," Revision 0, dated January 19, 1996, which define the audit and evaluation responsibilities for the ANIS and Authorized Nuclear Inservice Inspector Supervisor (ANIIS) at the FMEA District Offices and as well as the audit responsibilities for Home Office Supervisor. The FMEA audits program is written to meet the requirements of ASME QAI-1. This program includes audit methods, ASME audit reports of nuclear Section III and XI work, authorized inspector performance evaluations and the audit schedule for each. The ANIS also conducts the ASME Pre-Survey audit at the ASME Certificate holders facilities prior to the actual ASME survey to ensure that the Certificate Holder is prepared for the ASME survey team review.

The inspectors reviewed the most recent Philadelphia and New Jersey District Office listing of ASME Section III shops and Section XI sites that also included information such as the ANI and ANII assignments, the ANIS semiannual shop and site audit schedule, and the ANIS schedule for ANI performance evaluations at the Section III nuclear shops and ANII evaluations at the Section XI nuclear reactor sites in each district. The ASME Section III shop audits and Section XI site audits are performed to review ASME Code compliance and implementation of the various sections of the QA manual utilizing standard FMEA check lists (Form 2048 for Section III shops and Form 1453 for Section XI sites). The shop and site audits and inspector's performance evaluations are both performed semiannually.

The inspectors also reviewed the results of the three most recent years of ANIS shop audits conducted at three ASME Section III Code shops; Amer Industrial Technology (AIT), Valcor Engineering Corporation and Yarway Corporation. The audit reports included the appropriate documentation required by FMEA procedures including audit findings when applicable. However the inspectors noted that the ANIS audits of AIT for 1993 to 1995 continually identified AIT's failure to effectively implement their QA program and to meet ASME code requirements, which resulted in unacceptable results in many of the areas reviewed. In fact, in the Comments and Recommendations section of the audit report checklist (Form 2048), the ANIS usually recommended to the home office supervisor that the contract with AIT be terminated.

The inspectors also reviewed the semiannual inspector's performance evaluations conducted by the ANIS during his shop audits at the above mentioned Section III shops. The ANI and ANII inspector's performance evaluations are also conducted utilizing standard FMEA checklists (Form 2051 for ANIs at Section III shops and Form 2050 for ANIIs at Section XI sites). All inspector evaluations were performed and documented as per FMEA QA program requirements.

3.3.4 Personnel Qualification and Training

The inspectors reviewed QAM Section 8, "Indoctrination and Training," which describes the general requirements for the indoctrination and training of personnel and the more specific training requirements and responsibilities for the Home Office Supervisor and the District Supervisors. CP 105.00, "ASME Code Activity - Authorized Supervisor (AIS/ANIS/ANIIS)," CP 106.00, "ASME Code Activity - Authorized Inspector (AI/ANI)," and CP 107.00, "ASME Code Activity - Authorized Inspector (ANII)," all Revision 0, dated January 19, 1996, describe the specific training responsibilities and required training courses for each of the identified positions.

The Philadelphia district ANIS maintains a training matrix of job functions versus training requirements. The inspectors reviewed the indoctrination and training records for several employees currently working out of the Philadelphia district office as ANIs and ANIIs. Training records are maintained separately for ASME training and all other training. The training records reviewed by the inspectors included the appropriate documentation for the qualification and training that required for ANIs and ANIIs per ASME QAI-1 and the FMEA quality program. The inspectors also verified through review of records that the Philadelphia district office ANIS had the appropriate qualifications and training as required by ASME QAI-1 and the FMEA quality program.

3.3.4.1 Lead Auditor Qualifications

The inspectors reviewed QAM Section 9.4, "Selection of Auditors," and CP 109.00, "Lead Auditor - Authorized Nuclear supervisors (ANIS/ANIIS)," Revision 0, dated January 19, 1996, which outlines the minimum qualifications to certify and designate lead auditors for ASME and district audits. The lead auditor qualifications are maintained at the FMEA home office in Norwood, Massachusetts. The inspectors reviewed the qualification and certification record files at the Norwood office for two ANIS/ANIISs, the Assistant Vice President, ATS, and the Manager of ASME Code and Standards, FMEA and verified that these individuals had attained and were maintaining the appropriate qualifications to be certified as a lead auditor. The Assistant Vice President, ATS, conducts the annual evaluation of the QA program at the FMEA home office and the Manager of ASME Code and Standards, FMEA, conducts the annual evaluation of the district ANIS/ANIIS's performance.

4 PERSONS CONTACTED

The persons contacted during this inspection are listed below:

Factory Mutual Engineering Association

* S. Rudnickas, Vice President & Manager B&M Engineering
R.E. Montague, Asst. Vice President-District Manager
* + A.J. Spencer, Manager, ASME Codes and Standards
\+ W.R. Rogers III, District Supervisor
\+ R.G. Edl, District Chief Inspector
D. Kinley, Authorized Nuclear Inspector

Arkwright Technical Services, Inc.

* C.M. D'Esopo, President

+ Attended entrance meeting
* Attended exit meeting

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

May 10, 1996

Mr. James E. Quinn, Projects Manager
LMR and SBWR Programs
GE Nuclear Energy
175 Curtner Avenue, M/C 165
San Jose, CA 95125

SUBJECT: NRC INSPECTION REPORT NO. 99900403/96-01

Dear Mr. Quinn:

This letter addresses the inspection conducted March 5 through 8, 1996, at the Paul Scherrer Institut (PSI) PANDA Test Facility in Würenlingen, Switzerland, by Richard P. McIntyre of the Nuclear Regulatory Commission's (NRC's) Special Inspection Branch, Juan D. Peralta of the Quality Assurance and Maintenance Branch, John A. Kudrick and John D. Monninger of the Containment Systems and Severe Accident Branch, Alan E. Levin of the Reactor Systems Branch, and Dino C. Scaletti of the Standardization Project Directorate. The details of the inspection were discussed with you and the members of your staff during the inspection and at the exit meeting on March 8, 1996.

The purpose of the inspection was to determine if testing activities performed at the PANDA test facility to support design certification of the GE Nuclear Energy (GE) simplified boiling water reactor (SBWR) design were conducted under the appropriate provisions of the May 1990, GE NEDG-31831, "SBWR Design and Certification Program Quality Assurance Plan," as implemented by GE document PPCP-QA-01, "PANDA Project Control Plan" (PPCP) and the GE PANDA Quality Assurance Procedures (PQAPs).

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

The results of the inspection indicate that GE, in general, was adequately implementing the Project Control Plan and the Quality Assurance Procedures for testing activities performed at PANDA with the exception of two nonconformances. Also, the team identified three unresolved items related to PANDA and SBWR design certification that will require response by GE and follow-up by the NRC during a future inspection at San Jose. Specifically, the inspection team identified Nonconformances with program implementation with respect to (1) the preparation and issue of Apparent Test Results Reports and Data Transmittal Reports as required by the PANDA Test Specification and PANDA Test Plans, and (2) the failure to document abnormal occurrences detected during testing (subsequently causing matrix testing to be suspended and re-evaluated) using the existing nonconformance report process.

The unresolved items concerned (1) the appropriateness of GE's acceptance of engineering services activities performed by Elektrowatt Ingenieurunternehmung AG (Elektrowatt) in October 1993 for the PANDA test facility as-built measurements, (2) the level of GE QA oversight for the engineering services work performed by the international technical associates (KEMA and Instituto de Investigaciones (IIE) of Mexico) for PANDA data analysis, and (3) the disposition for the recommendations and specific action items identified during the October 1991 PANDA Design Review regarding facility design, quality assurance programmatic aspects, and technical issues.

The response requested by this letter is not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Public Law No. 96-511.

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

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

Sincerely,

for Gregory C Cwalina

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

Docket No.: 52-004

Enclosures: 1. Notice of Nonconformance
2. Inspection Report No. 99900403/96-01

cc w/encls: See Next Page

cc: Mr. Rob Wallace
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Pittsburgh, PA 15222

NOTICE OF NONCONFORMANCE

GE Nuclear Energy
San Jose, CA 95125

Docket No.: 52-004
99900403/96-01

Based on the results of a Nuclear Regulatory Commission (NRC) inspection, conducted from March 5 through March 8, 1996, of the GE's PANDA test program at the Paul Scherrer Institut (PSI) in Würenlingen, Switzerland, related to the SBWR design certification activities, it appears that certain activities were not conducted in accordance with NRC requirements.

A. Criterion XI, "Test Control," of Appendix B to 10 CFR 50, requires, in part, that test results be documented and evaluated to assure that test requirements have been satisfied.

Chapter II, "Basic Requirements," Section 11, "Test Control," of ANSI/ASME NQA-1-1983, "Quality Assurance Program Requirements for Nuclear Facilities," requires, in part, that test results be documented and that their conformance with acceptance criteria be evaluated.

Paragraph 5.3.14 of PQAP-TC, "Test Control," Revision 3, dated September 18, 1995, requires that the PSI PANDA Project Manager (P-PM) prepare test reports per the Test Specification and Test Plan requirements.

Section 11, "Reporting," of GE Document 25A5587, "PANDA Test Specification," Revision 1, dated January 26, 1995, requires (1) preparation of an Apparent Test Results report within approximately one week following performance of the test, and (2) preparation of Final Test Reports per the schedule specified in the Test Plan and Procedures Document.

Section 10, "Reports," of GE Document 25A5764, "PANDA Test Plan - Tests M3, M3A, M3B, M4, M7," Revisions 1, 2, and 3 dated September 18, 1995, October 16, 1995, and November 15, 1995, respectively, requires (1) preparation of an Apparent Test Results report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a Data Transmittal Report approximately two months after the last test is performed.

Section 10, "Reports," of GE Document 25A5785, "PANDA Test Plan - Tests M2, M10A, M10B," dated November 21, 1995, requires (1) preparation of an Apparent Test Results report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a Data Transmittal Report approximately two months after the last test is performed.

Section 10, "Reports," of GE Document 25A5788, "PANDA Test Plan - Tests M6/8," dated December 7, 1995, requires (1) preparation of an Apparent

Enclosure 1

Test Results report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a Data Transmittal Report approximately two months after the last test is performed.

Section 10, "Reports," of GE Document 25A5824, "PANDA Test Plan - Test M9," dated December 12, 1995, requires (1) preparation of an Apparent Test Results report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a Data Transmittal Report approximately two months after the last test is performed.

Contrary to the above, Apparent Test Results reports and Data Transmittal Reports were not prepared and issued in accordance with the Test Specification or Test Plan and Procedures. *(96-01-01)*

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

Criterion XV, "Nonconforming Materials, Parts, or Components," of Appendix B to 10 CFR 50, requires, in part, that nonconforming items be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures.

Criterion XVI, "Corrective Action," of Appendix B to 10 CFR 50, requires, in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Chapter II, "Basic Requirements," Section 11, "Test Control," of ANSI/ASME NQA-1-1983, "Quality Assurance Program Requirements for Nuclear Facilities," requires, in part, that test results be documented and that their conformance with acceptance criteria be evaluated.

Paragraph 5.3.11 of PQAP-TC, "Test Control," Revision 3, dated September 18, 1995, requires that the PSI PANDA Project Manager (P-PM) (1) review and resolve all test anomalies identified during the test, and (2) document resolutions, conditions requiring correction, and corrective actions per PQAP-NC.

Section 4, "Requirements," of PQAP-NC, "Nonconformance Control and Corrective Action," Revision 0, dated January 31, 1995, provides that any nonconforming item which can affect PANDA test results, or deviations from the test specification/procedure, or test conditions and results showing abnormal occurrences shall be identified, treated as a nonconformance, and documented and reported for resolution (disposition) prior to continuation of subsequent phase testing.

Contrary to the above, (1) when abnormal occurrences (subsequently causing matrix testing to be suspended and re-evaluated) were detected during testing, no nonconformance reports were generated to document these events; (2) PSI Procedure "Data Base Modification" (issued in March 1996) was being used by PSI testing personnel to perform activities that introduced deviations from the test control process already specified by PQAP-TC, and from the nonconformance identification process established in PQAP-NC; and (3) PSI Procedure "Data Base Modification" had not been identified or described as a Quality Assurance Procedure governed by PPCP-QA-01, i.e., as a procedure comprising the bases of the QA system implemented by PSI and GE in meeting the requirements of NEDG-31831, even though it was being used to perform quality related activities affecting PANDA test results. *(96-01-02)*

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

Dated at Rockville, Maryland
This 10 th day of May 1996

ORGANIZATION:	GE Nuclear Energy San Jose, California
REPORT NO.:	99900403/96-01
CORRESPONDENCE ADDRESS:	Mr. James E. Quinn, Projects Manager LMR and SBWR Programs GE Nuclear Energy 175 Curtner Avenue San Jose, California 95125
ORGANIZATIONAL CONTACT:	Mr. Kenneth W. Brayman, Manager Quality Assurance Systems (408) 925-6587
NUCLEAR INDUSTRY ACTIVITY:	GE Nuclear Energy (GE) is engaged in the supply of advanced boiling water reactor designs to utilities. GE also furnishes engineering services, nuclear replacement parts, and dedication services for commercial grade electrical and mechanical equipment.
INSPECTION CONDUCTED:	March 5 through 8, 1996
TEAM LEADER:	Richard P. McIntyre 5/7/96 Richard P. McIntyre Date Vendor Inspection Section (VIS) Special Inspection Branch (PSIB)
OTHER INSPECTORS:	Juan D. Peralta, HQMB John A. Kudrick, SCSB John D. Monninger, SCSB Alan E. Levin, SRXB Dino C. Scaletti, PDST
REVIEWED:	Gregory C. Cwalina 5/8/96 Gregory C. Cwalina, Section Chief, VIS Date
APPROVED:	for Gregory C. Cwalina 5/8/96 Robert M. Gallo, Chief, PSIB Date
INSPECTION BASES:	10 CFR Part 50, Appendix B and 10 CFR Part 21
INSPECTION SCOPE:	To determine if activities performed to support the design of the SBWR and, specifically, testing activities performed at the PANDA Test Facility at the Paul Scherrer Institut in Würenlingen, Switzerland were conducted under the appropriate provisions of the May 1990, GE NEDG-31831, "SBWR Design and Certification Program Quality Assurance Plan."
PLANT SITE APPLICABILITY:	None

Enclosure 2

1 INSPECTION SUMMARY

1.2 Nonconformance

- Nonconformance 99900403/96-01-01 was identified and is discussed in Section 3.4.1 of this report.
- Nonconformance 99900404/96-01-02 was identified and is discussed in Section 3.8 of this report.

1.2 Unresolved Item

- Unresolved Item 99900403/96-01-03 was identified and is discussed in Section 3.2 of this report.
- Unresolved Item 99900403/96-01-04 was identified and is discussed in Section 3.3.2 of this report.
- Unresolved Item 99900403/96-01-05 was identified and is discussed in Section 3.4.2 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous inspections have been conducted at this test facility.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 GE SBWR Quality Assurance Program

Chapter 17 of the SBWR standard safety analysis report (SSAR) describes the GE quality assurance (QA) program for the design phase of the SBWR program. The QA program is identified as "Nuclear Energy Business Operations Quality Assurance Program Description," NEDO-11209-04A, Revision 8, the latest revision approved by the NRC. NEDO-11209-04A applies to all GE activities affecting quality of items and services supplied to nuclear power plants and establishes GE's compliance with the provisions of Appendix B to 10 CFR 50.

NEDG-31831, "SBWR Design and Certification Program Quality Assurance Plan," dated May 1990, was developed by GE to fulfill the QA requirements of the SBWR reactor design and certification program. NEDG-31831 meets the requirements of ANSI/ASME NQA-1-1983 and its NQA-1a-1983 addenda as endorsed by the NRC in Regulatory Guide 1.28, Revision 3. Additionally, NEDG-31831 provides that design and testing work performed by international technical associates will be performed to their internal QA programs acceptable to the regulatory authorities of their respective countries as evaluated by GE for compliance with the provisions of ANSI/ASME NQA-1-1983.

3.2 PANDA QA Program for SBWR Design Certification Testing

Under an agreement between the Swiss Confederation (represented by the Paul Scherrer Institut [PSI]), the Electric Power Research Institute (EPRI), and GE, PSI performed passive decay heat removal and fission product retention tests in the PANDA test facility. The purpose of these tests was to evaluate the performance and behavior of the SBWR passive containment cooling system (PCCS) operating in typical post-LOCA containment environments. These tests were primarily focused on simulating the response of the SBWR containment cooling systems in order to (1) obtain additional data to support the adequacy of TRACG in predicting the quasi-steady heat rejection rate of a PCC heat exchanger and identify the effects of scale on PCC performance, (2) provide a sufficient database to confirm the capability of TRACG to predict SBWR containment system performance, encompassing systems interaction effects, and (3) demonstrate startup and long-term operation of a passive containment cooling system (Concept Demonstration).

A GE readiness review conducted at PSI during October 19 through 21, 1994, concluded that PSI had not adequately implemented a QA program meeting the appropriate requirements of ANSI/ASME NQA-1-1983, and that the PANDA facility was not ready to initiate testing. In a letter to GE, dated December 19, 1994, the NRC staff requested that GE provide a discussion of the corrective actions taken by GE as a result of the readiness review findings, including the area of QA.

In its response letter to the NRC, dated March 7, 1995, GE stated that as a result of a GE readiness review at PSI during October 1994, the PANDA quality assurance program would be restructured so that it would be conducted under direct GE supervision and governed by the provisions in NEDG-31831. To this effect, GE developed the PANDA Project Control Plan, PPCP-QA-01, Revision 1, dated May 1, 1995. PPCP-QA-01, in conjunction with nine other QA procedures, describe the organization, quality related activities, events and procedures necessary to ensure and verify that the PANDA project at PSI is conducted in accordance with the provisions of NEDG-31831. All documentation related to the PANDA test facility and test results is contained in the PANDA Test File (PTF) and organized accordingly.

In accordance with the provisions of PPCP-QA-01 and PQAP-TC, "Test Control," Revision 3, dated September 18, 1995, a test specification, GE Document No. 25A5587, "PANDA Test Specification," Revision 1, was prepared and issued by GE as required by the provisions in Engineering Operating Procedure (EOP) 35-3.00, "Engineering Tests." GE Document No. 25A5587 required that the PANDA tests be performed in conformance with PPCP-QA-01, which is based on the requirements of Appendix B to 10 CFR 50, NEDG-31831, and ANSI/ASME NQA-1-1983.

During the inspection, the team reviewed all relevant documentation and available test data found in the PTF. Based on these reviews, the team concluded that, in general, GE had adequately restructured the PANDA QA program in accordance with the provisions in NEDG-31831. However, GE failed to adequately implement certain provisions of NEDG-31831, prior to January 1995, related to the appropriateness of GE's acceptance of engineering services provided by a subcontractor to PSI. Specifically, the team determined that GE had not yet adequately addressed an issue related to activities performed at PSI by Elektrowatt Ingenieurunternehmung AG (Elektrowatt) in October 1993 and which had been identified during the October 19 through 21, 1994, readiness review at PANDA. Elektrowatt was hired by PSI to perform the facility as-built measurements which was an activity having substantial impact on the quality of test results generated at PANDA.

The team was concerned that GE concluded in the October 1994 Readiness Review Report (without providing any justification or taking any compensatory or corrective actions) that the PANDA facility as-built measurement activities performed by Elektrowatt were satisfactory, while at the same time acknowledged that Elektrowatt had not been audited by either GE or PSI as a supplier of services affecting quality. This issue was identified as *Unresolved Item 99900403/96-01-03*.

3.3 Design Control

The purpose of the review of design control was: (1) to assure that applicable regulatory requirements, design bases, codes and standards, and GE test specification requirements were correctly translated into design drawings, procedures, and instructions per PQAP-DC, "Document Control," (2) to assure that changes or deviations from specified design requirements and quality standards were identified, documented, and controlled, (3) to verify final PANDA test facility as-built drawings and overall control of test facility configuration as described in PQAP-V, "Verification," and (4) to assure that computer data acquisition software and documentation was controlled as described in PQAP-DA, "Data Acquisition System Control."

The team reviewed the following material related to design control for the PANDA test program:

- Design and as-built drawings
- PANDA scaling analyses
- PANDA line loss calculations based on estimated SBWR line losses
- Record of GE Design Review (San Jose, October 1991)

The PANDA test facility at PSI was designed to evaluate the performance of the SBWR passive containment cooling system operating in post-LOCA containment environments. The PANDA tests were to demonstrate the SBWR thermal-hydraulic performance, heat removal capability, and systems interactions and to provide data for confirmation of the TRACG computer models used to analyze the SBWR performance.

GE prepared and issued document 25A5587, "PANDA Test Specification," Revision 1, on January 26, 1995, for PANDA tests. The PANDA test specification specifies the top-level requirements for tests related to post-LOCA decay heat removal from the containment of the SBWR to be performed at the PANDA test facility at PSI. The test specification provides general criteria for the PANDA test program including: purpose, objectives, facility description, test instrumentation, data acquisition system and recording, data processing and analysis, shakedown and plant characterization tests, steady-state performance tests, transient integral systems tests, pretest predictions, acceptance criteria, reporting requirements, record retention, and quality assurance.

The actual experimental testing at PSI was to be performed in accordance with the Test Specification (25A5587) through the development of specific Test Plans and Procedures in accordance with GE PANDA Quality Assurance Procedure PQAP-DC, Revision 1, "Document Control." PQAP-DC defines the requirements and process for issuing, revising, modifying, and distributing the Test Plans and Procedures.

GE and PSI prepared the Test Plans and Procedures, which define the detailed or specific test requirements. The test plans describe how the test is to be set up and performed to meet the quality assurance requirements, any special conditions associated with the test, and the test requirements specified in the Test Specification. The test procedures describe the specific procedures required to perform the test in accordance with test and quality assurance requirements. The specific test plans and procedures reviewed by the team in the course of the inspection are listed in the table below for the plant characterization, shakedown, steady-state, and integral systems tests.

TEST	TEST PLAN	TEST PROCEDURE
VESSEL HEAT LOSS	ALPHA-510	ALPHA-510
LINE PRESSURE DROP	ALPHA-510	ALPHA-510
S1-S6	ALPHA-410-1	ALPHA-410-1
S7-S9	ALPHA-410-2	ALPHA-410-2
S10-S13	ALPHA-410-2	ALPHA-410-2
M3	25A5764 R1	ALPHA-520-0
M3A	25A5764 R2	ALPHA-520-2
M3B	25A5764 R2	ALPHA-520-2
M7	25A5764 R3	ALPHA-521-0
M2	25A5785 R0	ALPHA-527-0
M10A	25A5785 R0	ALPHA-527-0
M10B	25A5785 R0	ALPHA-527-0
M6/8	25A5788 R0	ALPHA-529-0
M9	25A5824 R0	ALPHA-528-0

These Test Plan and Procedures were issued and controlled in accordance with PQAP-DC. A PANDA Engineering Review Memorandum (P-ERM) was required for review and approval of all Test Plans and Procedures by PQAP-DC. In the course of the inspection, the team reviewed several completed P-ERMs relating to revisions to the following Test Plan and Procedures:

- ALPHA-410 "PANDA Steady-State PCC Performance Tests Test Plan and Test Procedures"
- ALPHA-520 "PANDA Transient Tests M3A, M3B, & M4 Integral System Test Procedure"

Based on a review of these P-ERMs, the team concluded that issues and comments identified by GE and PSI personnel as a result of the review and approval process of the Test Plan and Procedures were adequately identified, documented, resolved, and controlled.

GE PANDA Quality Assurance Procedure PQAP-V, Revision 1, "Verification," was developed to control the process for verification of the PANDA test facility configuration and testing activities. Verifications were to be performed for activities such as: calculations affecting test results, measurements appearing on as-built drawings, and test initial conditions. PQAP-V provided a "Verification Cover Sheet" to control and document the verification process. Extensive documentation was contained in the PTF on scaling of the PANDA facility and determination of line losses for the facility, based on design information for the SBWR. During the course of the inspection, the team reviewed an independent verification that required an alternate calculation to be performed to verify the correctness of the original calculations. This verification related to the establishment of the PANDA system line loss coefficient measurements. The line loss calculations were performed by several engineers using different methodologies; the results were then cross-checked and independently design verified and documentation of the results of this design study is extensive. The team noted that the verification was performed in accordance with PQAP-V and utilized the Verification Cover Sheet for control and approval of the verification.

3.3.1 Data Acquisition System (DAS)

The team evaluated the information relating to the Data Acquisition System (DAS) contained in the PTF. The DAS information is contained in four separate volumes of the PTF. However, due to either incomplete or missing information or the use of German documentation within the PTF, this information needed to be supplemented by discussions with PSI personnel so that a thorough understanding of the scope of the DAS could be gained. Based on the information contained in the PTF and discussions with PSI personnel, the team concluded that the DAS was sufficient for meeting the instrumentation requirements of the PANDA test program and that it was controlled in accordance with PQAP-DA,"Data Acquisition System Control."

3.3.2 1991 PANDA Design Review

The original PANDA design was developed from the SBWR conceptual design as it existed in the late 1980's; the volumetric scale was derived from representation of the passive containment cooling system (PCCS) heat exchanger (HX). The PCCS HX design was changed about 1991, which necessitated a slight change in the volumetric scale of the facility (from about 1:18 to 1:25). A design review was convened in San Jose in October 1991 to assess the facility design and to determine technical issues requiring GE or PSI to follow-up. Based on a review of documentation contained in the PTF, it appeared that the design review was independent and comprehensive. Numerous recommendations were made regarding facility design, quality assurance programmatic aspects, and technical issues, and specific action items were assigned to the participating organizations (primarily, GE and PSI). However, there was no record in the PTF of whether the action items and recommendations were ever dispositioned. When GE and PSI were asked about follow-up to the design review, the NRC team was informed that, since the design review memorandum was originated by GE/San Jose, the written record of disposition of the recommendations of the review group should be located in the design record file (DRF) in San Jose. Therefore, verification of the disposition of the design review action items was not possible at PANDA. The GE disposition of the recommendations and specific action items identified during the October 1991 PANDA design review regarding facility design, quality assurance programmatic aspects, and technical issues was identified as *Unresolved Item 99900403/96-01-04*.

3.4 Test Control

The purpose of the review of test control was: (1) to determine whether a suitable test program was developed to assure that all testing required to demonstrate that systems and components would perform satisfactorily in service, (2) to determine that such a test program was identified and performed in accordance with written test procedures which incorporate or reference the requirements and acceptance limits contained in the applicable design documents, (3) to assure that test procedures include provisions for assuring that all prerequisites for the given test have been met, that adequate instrumentation is available and used, and that testing is performed under suitable environmental conditions, and (4) to assure that test results are documented and evaluated to assure that test requirements have been satisfied.

To assess the level of control over the testing program, the team examined the adequacy, implementation, and documentation resulting from the development and performance of facility characterization tests, shakedown tests, steady state tests, and integral system tests. The PANDA matrix tests (steady state and integral system) were performed in accordance with GE Panda Quality Assurance Procedure "Test Control," PQAP-TC. The purpose of PQAP-TC is to define the process for specifying, performing, evaluating, and documenting the PANDA tests. The specific Test Plans and Procedure reviewed by the team along with resulting test file documentation are specified below.

Section 8, "Shakedown and Plant Characterization," of the GE PANDA Test Specification (25A5587, Revision 1) required facility shakedown and plant characterization tests to be performed. The shakedown tests where to be run in a manner which would expose the facility components and systems to conditions similar to those expected during the matrix tests. The characterization tests were to consist of tests that quantify specific characteristics of the facility such as vessel heat loss and line pressure drop tests.

The facility characterization tests were completed in July 1995. The team reviewed Section 9 of the PTF, "Facility Characteristics," to assess whether adequate quality assurance measures had been followed in the preparation, conduct, and documentation of these tests. The facility characterization tests were performed in accordance with ALPHA-510 "PANDA Facility Characterization Heat Loss and Selected System Lines Pressure Loss Test Plan and Procedures." The heat loss test is needed for calculation of energy balances which in turn would be used to assess system performance and to reliably model heat losses from the PANDA test facility in computer code analyses.

With respect to the heat loss test, Section 9 of the PTF only included the test plan and procedure. Results of the test, apparent test result and final test result reports were not available. GE and PSI stated that the test reports were still under development and provided the team with a draft report, ALPHA-519-A, "PANDA Facility Characterization Vessel Heat Loss Measurements," dated August 11, 1995, for review. The draft report indicated that the heat loss calculations were preliminary and were intended to provide a first look at the vessel heat losses. The draft report indicated that the calculations had not included all potential heat losses nor the vessel leakage rates.

The system line pressure loss test was performed to assure that system line pressure drop characteristics measured in the PANDA facility adequately simulated the pressure loss characteristics of the full scale SBWR system. This test was performed for loss measurements in the isolation condenser and primary containment cooling (PCC) system feed line, PCC vent line, gravity driven cooling system lines, equalization lines, and main steam lines. Section 9 of the PTF included a report, ALPHA-517-0, "PANDA Facility Characterization System Line Loss Coefficient Measurements," dated February 14, 1996, which provided the results and an evaluation of the system line pressure loss tests.

The team inquired as to whether additional facility characterization tests had been performed, in addition to the heat loss and system line pressure loss test. GE and PSI indicated that a leak test had been performed at the PANDA facility in accordance with ALPHA-511 "PANDA Facility Characterization Vessel Cold Leak Test Plan and Procedure." This test is important because the leakage rate from each PANDA vessel must be known to permit calculation of vessel heat losses from the heat loss test data. The leak rate is used to separate the components of pressure drop due to condensation and mass lost from the system. Furthermore, an estimate of the overall leakage rate is necessary to characterize the system for the transient tests. The mass loss from the system must be quantified to properly interpret data from the transient tests. From the review of ALPHA-511, the team concluded that it had been developed, reviewed, and controlled through the use of the P-ERM and Verification Cover Sheet in accordance with PQAP-DC.

Section 9, "Test Matrix," of the GE PANDA Test Specification (25A5587, Revision 1) required a series of steady-state tests to be conducted using one of the PANDA PCC condensers. The objectives of the steady-state tests was to provide additional data to support the adequacy of TRACG to predict the quasi-steady heat rejection rate of a PCC heat exchanger and identify the effects of scaling on PCC performance by using one of the PANDA PCC condensers connected directly to the steam supply. The steady-state tests were conducted in accordance with ALPHA-410, "PANDA Steady-State PCC Performance Tests Test Plan and Procedures."

The team reviewed Section 8 of the PTF "Steady-State Tests," including the test specification, test plan and procedures, shakedown test results, data reduction/reduced data records, apparent test results report, and analytical work. In addition, the PTF included: a copy of the control room procedures used, excerpts of the PANDA journal, instrumentation list, DAS channel allocation table, instrument checks, checklists per the test plan and procedures, valve status reports, re-zeroing charts, DAS monitor printout, and any non-conformance reports. The test procedures specified the prerequisites for the test, instrumentation requirements, and test acceptance criteria.

Prior to performing the actual steady-state tests, PSI performed shakedown tests to expose the PANDA facility components to conditions similar to those expected during the matrix tests. During the first series of shakedown tests for the S1-S6 steady-state tests, steady-state conditions could not be achieved as required by ALPHA-410. PSI documented the failure to meet the test acceptance criteria through use of a Nonconformance Report in accordance with GE PANDA Quality Assurance Procedure PQAP-NC, "Nonconformance Control and Corrective Action," Revision 0, dated January 31, 1995. PQAP-NC establishes the requirements and procedures for the identification, documentation, resolution and control of nonconforming items. With respect to tests S1-S6, three nonconformances were identified and documented in accordance with PQAP-NC and one nonconformance resulted from the S7-S9 tests.

Section 9, "Test Matrix," of the GE PANDA Test Specification (25A5587, Revision 1) required a series of transient integral systems tests to be conducted. These tests were to provide an integral systems database for PCC system performance with conditions representative of the long-term post-LOCA SBWR containment. The objectives of the transient integral systems tests was to provide a sufficient database to confirm the capability of TRACG to predict SBWR containment system performance, including potential systems interaction effects. The transient integral systems tests were conducted in accordance with the various test plans and procedures identified in the table in Section 3.3 above.

The team reviewed the available documentation within Section 10 of the PTF relating to the transient integral systems tests. The PTF contained comparable information to that included for the steady-state tests such as copies of the control room procedures used, excerpts of the PANDA journal, instrumentation list, DAS channel allocation table, instrument checks, checklists per the test plan and procedures, valve status reports, re-zeroing charts, DAS monitor printout, trending charts, and any nonconformance reports.

The team concluded that a suitable test program was developed by GE and PSI with applicable test plans and procedures in accordance with the Test Specification. The test procedures included provisions for assuring that prerequisites were met and that adequate instrumentation was available.

3.4.1 Reporting of PANDA Test Results

Section 11, "Reporting," of the GE PANDA Test Specification (25A5587, Revision 1) specified preparation of Apparent Test Results (ATR) reports and Final Test Reports (FTR). The ATR reports are considered to be unverified reports of preliminary results for each test or each test series that were to be issued within approximately one week following performance of the tests. The FTRs are considered to be verified reports which contain the data, analysis, and results of all tests and transmitted to GE per the schedule specified in the Test Plan and Procedures documents. The FTRs are identified as Data Transmittal Reports (DTRs) within the "Reports" section of the various test plan and Procedures.

For the facility characterization tests, Section 4.3, "Post-test/Apparent Test Results Report Inputs" of ALPHA-510 specifies that following completion of the tests, data reduction will be performed to support preparation of the Test Results reports (TR). This data reduction will include time history plots of all the required measurements covering the full test duration. These results will be reviewed and reported in the TR. Section 4.4, "Post-test/Final Test Report," specifies that the Final Test Report (FTR) will transmit all the data for the system line pressure loss and the heat loss tests. It will provide detailed information on the test instrumentation, test conditions, and the format for the data. In addition, samples of key data will be presented in plots along with simplified sketches of the test facility configurations during testing.

GE and PSI provided the team with "draft" copies of ALPHA-519-A, "PANDA Facility Characterization Vessel Heat Loss Measurements," dated August 11, 1995, and ALPHA-517, "PANDA Facility Characterization System Line Loss Coefficient Measurements," dated February 14, 1996. The team concluded that these draft reports do not meet the timeliness requirements of the Test Specification or Test Plan and Procedure for providing ATR reports and FTRs after completion of the tests.

For the steady-state tests, Section 10, "Reports," of ALPHA-410, "PANDA Steady-State PCC Performance Tests Test Plan and Test Procedures," specifies preparation of apparent test results reports within approximately 1 week of completion of the steady-state tests. In addition, ALPHA-410 specifies preparation of the DTR approximately 2 months after completion of the steady-state tests. GE and PSI provided the team with various versions of ALPHA-509, "PANDA Steady-State Tests S1 through S6 PCC Performance Apparent Test Results," however, a DTR or FTR had not been issued. The team concluded that a DTR or FTR had not been prepared in accordance with the Test Specification or the Test Plan and Procedure.

For the transient integral systems tests, Section 10, "Reports," of GE Document 25A5764, "PANDA Test Plan - Tests M3, M3A, M3B, M4, M7," Revisions 1, 2, and 3 dated September 18, 1995, October 16, 1995, and November 15, 1995, respectively, requires (1) preparation of an ATR report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a DTR approximately two months after the last test is performed. GE and PSI had not prepared any ATR reports or DTRs for the M3, M3A, M3B, M4, or M7 transient integral system tests.

Section 10, "Reports," of GE Document 25A5785, "PANDA Test Plan - Tests M2, M10A, M10B," dated November 21, 1995, requires (1) preparation of an ATR report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a DTR approximately two months after the last test is performed. GE and PSI had not prepared any ATR reports or DTRs for the M2, M10A, or M10B transient integral system tests.

Section 10, "Reports," of GE Document 25A5788, "PANDA Test Plan - Tests M6/8," dated December 7, 1995, requires (1) preparation of an ATR report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a DTR approximately two months after the last test is performed. GE and PSI had not prepared an ATR report or DTR for the M6/8 transient integral system test.

Section 10, "Reports," of GE Document 25A5824, "PANDA Test Plan - Test M9," dated December 12, 1995, requires (1) preparation of an ATR report within approximately two weeks of completion of each transient integral system test, and (2) preparation of a DTR approximately two months after the last test is performed. GE and PSI had not prepared an ATR report or DTR for the M9 transient integral system test.

The failure to prepare ATR reports, FTRs and/or DTRs, on a time schedule consist with the applicable Test Specification and Test Plan and Procedures requirements is identified as *Nonconformance 99900403/96-01-01*.

3.4.2 PANDA Test Analyses

The team also examined the analytical efforts that support the PANDA testing program. Section 10, "Pretest Predictions/Acceptance Criteria," of GE PANDA Test Specification 25A5587, Revision 1, specifies that pretest calculations be performed for some of the matrix tests planned for SBWR certification. This activity was to include development of a TRACG input model for the PANDA facility, verification of the input model against as-built test facility data, design review of the input model, calibration of the input model using heat loss and pressure drop data from test facility characterization testing, selection of the test conditions for simulation, performance of the calculations, and documentation of the results.

GE provided the NRC with the SBWR-Pretest Report for PANDA Test M9 in a letter dated December 12, 1995. This report was to support the validation efforts for the TRACG code for application to the SBWR program. These calculations include both pre and post-test calculations for tests in the PANDA test program. The analyses of the tests were being performed by an SBWR PANDA analysis team, with participation from PSI, KEMA and ECN in the Netherlands, Instituto de Investigaciones (IIE) of Mexico, and GE.

GE indicated that the TRACG modeling of the PANDA test facility was developed by PSI. GE then had an individual with IIE of Mexico verify the TRACG modeling and nodalization of the PANDA facility. The team inquired to GE as to the level of quality assurance oversight that GE had performed of IIE or the individual performing the work. GE presented conflicting stories with respect to whether the agreement for review of the TRACG modeling was with IIE as a subcontractor or with an individual of IIE as a subcontractor. The extent and level of effectiveness of quality assurance oversight by GE over its SBWR program international technical associates was identified as *Unresolved Item 99900403/96-01-05*.

3.5 As-Built Drawings and Configuration Control

PQAP-V, "Verification," Revision 1, dated May 1, 1995, implements the applicable requirements of GE EOP 42-6.00, "Independent Design Verification," and EOP 40-7.00, "Design Reviews," for verification of the PANDA test facility configuration and testing activities. Records of PANDA's facility and as-built drawings are stored in Section 2.1, "Facility Drawings," and in Section 3.1, "As-Built Drawings," of the PTF.

In October 1993, PSI contracted with Elektrowatt (see Section 3.2) to generate as-built drawings for PANDA. The facility configuration was originally depicted in a Giovanola (the facility builder) design drawing No. 164-A3526-1c (PSI Drawing No. 1-290111c). This drawing was used by Elektrowatt to develop an as-built of the main configuration and was subsequently given the designation of PSI Drawing No. 1-290300. All subsequent as-built measurements taken by PSI, including instrument and valve locations, were based on the Elektrowatt measurements. The Passive Containment Cooling System (PCC) and Isolation Containment System (IC) units (manufactured by Jaggi, AG) were measured by PSI personnel after their arrival on-site to establish their as-built dimensions.

As-built tolerances for the PANDA facility were established by GE in Document No. 25A5764, "PANDA Test Plan - Tests M3, M4, M7," Revision 1, dated September 18, 1995. The team found evidence that PSI had performed a review to verify that all as-built dimensions identified in drawings generated by Elektrowatt met the tolerance criteria specified in GE Document No. 25A5764.

Except for the unresolved item identified in Section 3.2, above, and based on the reviews of pertinent documents in Sections 2.1 and 3.1 of the PANDA PTF, the team concluded that activities performed by PSI after January 1995 were consistent with the provisions in PQAP-V.

3.6 Procurement Control

PQAP-PC, "Procurement Control," Revision 0, dated January 31, 1995, defines the requirements for procurement initiated by PSI in support of the PANDA test program, after test facility commissioning, for equipment and services. This procedure implements the applicable requirements of GE Engineering Operating Procedures, EOP 45-1.00, "Procurement Initiation and Control," EOP 45-2.00, "Procurement of Engineering Services," and in part, EOP 35-3.20, "Calibration Control."

In this area, the team was primarily interested in examining the implementation of PSI procurement provisions with respect to calibration services. As discussed below in Section 3.7, the team found objective evidence that after January 1995, PSI had adequately implemented the applicable provisions in PQAP-PC.

3.7 Control of Measuring and Test Equipment

PQAP-CC, "Control of Measuring and Test Equipment," Revision 0, dated January 31, 1995 defines and establishes all requirements related to the processes and procedures used for calibration of PANDA instrumentation. Section 4, "Instrumentation," of the PTF contained all documentation related to the procurement and calibration of PANDA instrumentation, including calibration certificates furnished by companies accredited by the Swiss Federal Office of Metrology (Eidgenössisches Amt für Messwesen) in Bern.

In Section 5.4, "Instrument Calibration," of PSI Document No. ALPHA-410, "PANDA Steady-State Tests - PCC Performance Test Plan and Procedures," Revision 2, dated May 16, 1995, PSI describes in detail its approach for ensuring that calibration of the various PANDA instruments was adequately performed and documented. Except for pressure and differential pressure sensors, all instruments were individually, or on a sampling basis, sent to the Swiss Federal Office of Metrology in Bern for calibration.

All pressure and differential pressure sensors used in PANDA were manufactured by Rosemount, Inc. Except for the Model 2088 and SMART, all Rosemount pressure sensors were calibrated by PSI prior to installation in the facility using a reference or standard traceable to the Swiss Federal Office of Metrology and in accordance with the requirements in PSI Document No. ALPHA-408, "PANDA Instrumentation and Control - PANDA Pressure Transmitter Calibration," Revision 1. For the Model 2088 and SMART sensors, the Rosemount factory calibration data was used.

The team inquired as to why PSI was relying solely on the manufacturer's calibration data for the Model 2088 and SMART sensors. PSI stated that these instruments were software-controlled and PSI lacked the necessary hardware and/or software to test them properly. PSI also stated that Rosemount of Switzerland, where these instruments would be re-calibrated after completion of testing, is a metrology laboratory accredited by the Swiss Federal Office of Metrology.

Based on the above information, the team concluded that PSI had adequately implemented the provisions in PQAP-CC and PQAP-PC and this area was identified as a strength in the PANDA QA program.

3.8 Nonconformance Control and Corrective Action

PQAP-NC, "Nonconformance Control and Corrective Action," Revision 0, dated January 31, 1995, establishes the requirements and describe the procedure for the identification, documentation, resolution and control of nonconforming items for the PANDA program. This procedure applies to all PANDA quality related activities that can affect PANDA test results and it implements the applicable requirements of GE EOP 75-4.10, "Control of Nonconforming Material," and EOP 75-3.00, "Corrective Action and Audits."

Based on reviews of nonconformance reports found in the PTF, and based on conversations with PSI test personnel, the team learned that although abnormal occurrences, subsequently causing matrix testing to be suspended and re-evaluated, had been detected during testing, no nonconformance reports had been generated by PSI to document these events. The team also learned of the existence of a new PSI procedure ("Data Base Modification," issued in early March 1996) used extensively by PSI testing personnel to evaluate, and when necessary, modify, i.e., revise or delete, actual test results data. Although this "Data Base Modification" procedure was clearly being used to perform an activity affecting quality as well as an activity that introduced deviations from the test control process specified by PQAP-TC, and from the nonconformance identification process identified in PQAP-NC, the "Data Base Modification" procedure had not been identified or described as a Quality Assurance Procedure governed by PPCP-QA-01. This issue was identified as *Nonconformance 99900403/96-01-02*.

3.9 Personnel Training and Qualification

PQAP-PT, "Personnel Training and Qualification," Revision 0, dated January 31, 1995, establishes the personnel training and qualification requirements to be implemented on the PANDA Project for test facility personnel. PQAP-PT implements the applicable requirements of the appropriate GE EOPs and states that individuals who perform activities affecting the quality of the PANDA project must be proficient in the appropriate technical discipline and the procedural systems.

Technical qualifications specify a minimum education, experience, and/or special technical training requirements. Procedurally, each individual shall be indoctrinated or instructed in the applicable quality assurance procedures. Indoctrination and training shall be attained and maintained by methods such as procedure reading, class training and/or on the job training.

The team verified through review of PTF training records that all PANDA personnel had received training in all sections of the PQAP and test procedures.

Personnel qualifications were done in accordance to level of education and years of experience in the desired fields. All personnel running and supervising the tests were appropriately trained and qualified in accordance with PQAP-PT. The PSI ALPHA Project Manager and the and the PSI PANDA Project Manager both qualified to the highest qualification level required.

There were no records of subcontractor training and qualification in the PTF. However, GE stated that such documentation existed in the PANDA DRF maintained in San Jose for the training of certain international technical associates performing PANDA data analysis.

Based on the above review of personnel training and qualification records in the PANDA PTF, the team concluded that GE/PSI had adequately implemented the provisions and requirements of PQAP-PT for PSI personnel.

3.10 Quality Assurance Records

PQAP-R, "Quality Assurance Records," Revision 0, dated January 31, 1995, defines the requirements for identification, accumulation, review, maintenance, and retention of the quality assurance records in the PTF. PQAP-R implements the applicable requirements of GE EOP 42-10.00, "Design Record File," and EOP 75-6.00, "Quality Assurance Records."

PQAP-R requires that a central file of legible, accurate and complete QA records, the PANDA Test File, shall be established with an index and table of contents. PQAP-R also requires that the PTF be stored in an archive for the duration of the testing and at completion of testing, the PTF will be transferred to GE for inclusion in the PANDA Design Record File.

The NRC team was informed of two pertinent facts about the test files at PSI: first, since the program is still active, the PSI files have not been closed and, in fact, detailed information on most of the tests had not yet been included in the PTF; and, second, the PSI files do not, and are not intended to, comprise the complete DRF for the PANDA program. Important supporting information is contained in the DRF at GE's offices in San Jose. When the PSI test files have been completed and closed, the PTF will be provided to GE, and the combined set of files will comprise the complete PANDA DRF.

Test result records were identifiable and retrievable to the extent they were included in the PTF. However, the team was told that test data is not included in the PTF until it has completed the PSI Project Manager's review process. This process resulted in the team having to request completed test data that was not yet stored in the PTF.

Overall, the documentation in the PTF reflected evidence of appropriate implementation of the PPCP and the QA procedures. Based on the results of these reviews, the team concluded that the QA records control process was adequately established and implemented for the PANDA test program.

3.11 Audits

Sections 4.0, "Project Assessment," and 4.1, "Audits," of PPCP-QA-01, define the project implementation requirements for internal audit activities at PANDA though GE established procedures P&P 70-11, GE Quality System Requirements, and Administrative Guide AG-017. All internal audits and oversight of the PANDA test program were conducted by GE certified auditor(s). The team confirmed that audits were performed by appropriately trained QA personnel with GE-certified lead auditors. Audits included the following:

- Audit plan and schedule
- Audit check list
- Preaudit orientation and training (if required)
- Assessment
- Audit report issued to management
- Corrective action requests (CARs)
- Resolution of audit findings
- Follow-up to closure of CARs

The team reviewed results of a GE audit of the PANDA Test Facility conducted January 31 through February 2, 1995. The team also reviewed the results of the readiness assessment conducted in October 1994, including the open items and recommendations that were documented in the Readiness Assessment Report.

During the January 1995 GE audit, it was determined that 11 of the 15 open items and 5 of the 9 recommendations identified during the readiness review were still unresolved. GE performed a second readiness review in September 1995 and the results were documented in a report identifying several findings. Corrective action requests (CARs) were issued and the appropriate follow-up to assure closure of the CARs was documented.

The team determined that no external supplier audits were performed by either PSI or GE for PANDA suppliers. Based on the results of these reviews, the team concluded that GE was implementing an appropriate internal audit program at the PANDA test facility.

4 PERSONNEL CONTACTED

GE Nuclear Energy

- James E. Quinn, Projects Manager, LMR and SBWR Programs
- John Torbeck, Project Manager, SBWR Test Operations
- Norman Barclay, Manager, Audit Programs
- G. Wingate
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- Richard P. McIntyre, Team Leader, Special Inspection Branch
- Juan D. Peralta, Quality Assurance and Maintenance Branch
- John A. Kudrick, Containment Systems and Severe Accident Branch (SCSB)
- John Monninger, SCSB
- Alan E. Levin, Reactor Systems Branch
- Dino C. Scaletti, Standardization Project Directorate

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

March 5, 1996

Mr. Craig P. Kipp
Plant Manager
General Electric Nuclear Energy
Nuclear Energy Production
P.O. Box 780, Mail Code A20
Wilmington, NC 28402-0780

SUBJECT: NONPROPRIETARY VERSION OF NRC INSPECTION REPORT NO. 99900003/95-01

Dear Mr. Kipp:

This letter transmits the nonproprietary version of the U.S. Nuclear Regulatory Commission's (NRC's) Inspection Report 999000003/95-01. Our letter to you dated January 4, 1996, transmitted the original (proprietary) version of the report. On the basis of our discussions on February 12, 1996, and review of the information in your letters of January 24, 1996, (RJR-96-013), and its enclosures (Affidavit, dated January 23, 1996, and Proprietary Information Summary Sheet) and February 27, 1996, and its enclosures (Affidavit, dated February 27, 1996, and Proprietary Information Summary Sheet), we have concluded that certain specific information identified in your letters could be regarded as proprietary and, as such, were removed from the inspection report. In the revised nonproprietary (public) version of the report, the NRC has briefly summarized the deleted text.

Your response to either this letter or our letter dated January 4, 1996, and their enclosure are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Public Law No. 96-511.

In accordance with 10 CFR 2.790 (a) of the NRC "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room. If you have any questions concerning this matter, please contact Steven M. Matthews at (301) 415-3191.

Sincerely,

for Gregory C Cwalina

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

Docket No.: 99900003

Enclosure: Report No. 99900003/95-01

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

JAN 0 4 1995

Mr. Craig P. Kipp
Plant Manager
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P.O. Box 780, Mail Code A20
Wilmington, NC 28402-0780

SUBJECT: NRC INSPECTION REPORT NO. 99900003/95-01

Dear Mr. Kipp:

From August 14 through September 1, 1995, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection of the General Electric Nuclear Energy (GENE) activities at the Nuclear Energy Production (NEP) facilities in Wilmington, North Carolina. This letter transmits the report of that inspection.

Steven M. Matthews of NRC's Special Inspection Branch led the inspection team, which included the other inspectors named in the report. The team conducted a performance-based evaluation of the NEP management, staff, and quality programs. They also assessed NEP's implementation of those programs related to boiling-water reactor reload core design, safety analysis and licensing processes, fuel assemblies, fuel-related core components, and fuel-related inspection services supplied to the U.S. nuclear industry. In the course of this evaluation, the team examined technical documentation, procedures, and representative records. They also interviewed and held discussions with NEP personnel. In addition, the team listened to presentations by NEP personnel and observed work activities in progress.

On the basis of this inspection, the team determined that, for the most part, the "GE Nuclear Energy, Quality Assurance Program Description," NEDO-11209-04A, Revision 8, Class 1 (approved by the NRC on March 31, 1989), was implemented in an appropriate manner for the areas evaluated. Therefore, the team did not cite any violations or nonconformances. The enclosed inspection report identifies the areas examined and presents a detailed discussion of the team's conclusions.

The report describes the team's conclusion that NEP's nonsafety-related classification of the constituent fuel bundle parts was not based on a functional evaluation of the parts. NEP asserted that, because the channeled fuel bundle operates in a unique BWR environment, a basis for its nonsafety-related classification of parts could be developed. Therefore, NEP is requested to submit its functional evaluation and technical basis for the nonsafety-related parts classification.

During this inspection, the team identified areas and activities within NEP that it considered strengths of your organization, as described in the inspection report. However, the team also observed weaknesses in certain activities that affect quality. The most significant of these concerns the accuracy of design methods used to ensure that the reactor can safely be operated. The team concluded that the errors in calculated hot and cold core reactivity for the recent reloads with the newer fuel designs and long cycle lengths was a weakness of the NEP reload design process.

The team noted, however, that NEP was addressing this weakness by implementing improvements in its steady-state nuclear methods. These improvements include a formal review process, increasing benchmarking of certain calculations for an extensive number of bundle design conditions, implementing revised computer codes, and developing a new lattice code. However, because the issue of accuracy in eigenvalue calculation is related to the issue of the accuracy of design methods used to provide assurance that the reactor can safely be operated, it is appropriate that the proposed design improvements be thoroughly documented, peer reviewed, and monitored over a period of time to ensure that the new design methods are indeed meeting the requirements that have been placed on them.

The team concluded that the near-term use of the revised computer codes in combination with the eigenvalue selection process should help reduce uncertainties in the cold critical and shutdown margins. The team also concluded that the assurance of adequate shutdown margin can be strengthened by joint GENE/licensee actions consistent with the plant startup safety analysis.

During this inspection, the team also reviewed the new GE12 and GE13 fuel designs as well as followed up on issues resulting from an earlier audit of the GE11 fuel design. The team raised concerns regarding the upper limit subcooling range and the altered definition for the bundle R-factor distribution. GENE responded in timely manner to these concerns with submittals described in the inspection report. GENE's submittals, in response to the team's concerns, are under evaluation by the NRC and will be addressed separately.

The team's review of the new fuel designs described in Section 3.5 of the enclosed report identified instances where fuel design commitments to the NRC were not fully met, as prescribed in Amendment 22 of the "General Electric Standard Application for Reload (GESTAR) II" topical report documented in NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel" (approved by the NRC on July 23, 1990).

Neither the weaknesses nor the observations summarized in Appendix A of the enclosed inspection report require a written response from NEP. The report also includes several open items that could not be resolved during the inspection, or for which the team needs additional information to reach its conclusions. NEP is requested to submit a written response to these open items, as listed in Appendix B of the enclosed report.

In accordance with Section 2.790(a) of Tit'- 10 of the Code of Federal Regulations, a copy of this letter and its enclosure will be placed in the NRC Public Document Room and made available to the public unless you notify this office by telephone within 10 days of the date of this letter and submit a written application to withhold the information contained therein. Such application must be consistent with the requirements of 10 CFR 2.790(b)(1). Your response to this letter and its enclosure is not subject to the clearance procedures of the Office of Management and Budget, as required by the Paperwork Reduction Act of 1980, Public Law No. 96-511.

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

Sincerely,

Michael F. [illegible] for

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

Docket No.: 99900003

Enclosure: Report No. 99900003/95-01

REVISED NONPROPRIETARY VERSION

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

REPORT NO.:	99900003/95-01
ORGANIZATION:	General Electric Nuclear Energy Nuclear Energy Production Wilmington, North Carolina
ORGANIZATIONAL CONTACT:	James F. Klapproth, Manager Fuel and Facility Licensing Environmental Health & Safety and Nuclear Quality Assurance
NUCLEAR INDUSTRY ACTIVITY:	General Electric Nuclear Energy (GENE), Nuclear Energy Production (NEP) provides boiling-water reactor reload core designs, safety analysis, and licensing, fuel assemblies, fuel-related core components, and fuel-related inspection services to the U.S. nuclear industry.
INSPECTION DATES:	August 14 through September 1, 1995
LEAD INSPECTOR:	Steven M. Matthews 3/5/96 Steven M. Matthews Vendor Inspection Section (VIS) Special Inspection Branch (PSIB)
OTHER INSPECTORS:	David H. Brewer, PSIB/VIS Daniel W. Brite, Battelle Pacific Northwest Laboratory Carl E. Beyer, Battelle Pacific Northwest Laboratory Dr. John F. Carew, Brookhaven National Laboratory Carl J. Czajkowski, Brookhaven National Laboratory Rodney L. Grow, Parámeter, Inc. Edward D. Kendrick, DSSA/SRXB Patrick S. Lacy, Parámeter, Inc. Dr. Partha Neogy, Brookhaven National Laboratory Larry E. Phillips, Chief, DSSA/SRXB Alexander C. Schafer, Parámeter, Inc. John W. Thompson, AEOD/DEIIB Dr. Shih-Liang Wu, DSSA/SRXB
REVIEWED BY:	Gregory C. Cwalina 3/5/96 Gregory C. Cwalina, Chief PSIB/VIS Date
APPROVED BY:	for Gregory C. Cwalina 3/5/96 Robert M. Gallo, Chief, NRR/DISP Date

ENCLOSURE

CONTENTS

ABBREVIATIONS AND ACRONYMS

ABAM	Automated Bundle Assembly Machine
ADU	Ammonium Diuranate
AOO	Anticipated Operational Occurrence
ANSI	American National Standards Institute
APRM	Average Power Range Monitor
ASME	The American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ASNT	American Society for Nondestructive Testing
ATLAS	GENE's 8.6-megawatt heat-transfer loop
AVL	Approved Vendors List
BOC	Beginning Of Cycle
BWR	Boiling-Water Reactor
BWREDB	Boiling-water Reactor Engineering Data Bank
CAR	Corrective Action Request
CCI	Control Component Individual
CEC	Commonwealth Edison Company
CFR	Code of Federal Regulation
CILC	Crud-Induced-Localized Corrosion
CMDB	Core Monitoring Data Bank
CM&S	Chemical, Metallurgical and Spectral
COLR	Core Operating Limits Report
CP&L	Carolina Power & Light Company
CPR	Critical Power Ratio
CSR	Contractile Strain Ratio
DB	Design Bases
DBE	Design-Basis Event
Δ	Delta (Differential)
DIP	Design Interface Procedure
DISP	Division of Inspection and Support Programs (NRR)
DNBR	Departure from Nucleate Boiling Ratio
DRF	Design Record File
DSSA	Division of Systems Safety and Analysis (NRR)
DWS	Daily Weld Sample
ECCS	Emergency Core Cooling System
ECP	Engineering Computer Program
EOC	End Of Cycle
EOP	Engineering Operating Procedure
ET	Eddy Current Examination
EUP	Energy Utilization Plan
FCO	Fuel Components Operations
FFWTR	Final Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FMO	Fuel Manufacturing Operations
FRED	Fuel Release Engineering Data
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
GENE	General Electric Nuclear Energy
GESTAR	GENE Standard Application for Reload
GESTR	GENE Stress and Thermal Analysis of Fuel Rods

ABBREVIATIONS and ACRONYMS Continued

GRF	Grid-To-Rod Fretting
GWd/MTU	Gigawatt-Days Per Metric Tonne of Initial Uranium Metal
HFP	Hot Full-Power
ICF	Increased Core Flow
IPHT	In-Process Heat Treatment
IMRS	In-Process Material Release System
LFWH	Loss-of-Feedwater Heating
LHGR	Linear Heat Generation Rate
LMCS	Laboratory Material Control System
LOCA	Loss-of-Coolant Accident
LUA	Lead Use Assembly
MAPS	Magnetic and Passive Scanners
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCNP	Monte-Carlo Neutron Particle computer code
MCPR	Minimum Critical Power Ratio
MFB	Mislocated Fuel Bundle
MICS	Material Inventory Computer System
MSG	Mill-Slug-Granulate
M&TE	Measuring and Test Equipment
MSIV	Main Steam Isolation Valve
NEP	Nuclear Energy Production
NDE	Nondestructive Examination
NDR	Nuclear Design Report
NFUF	Nuclear Fuel Users Forum
NIST	National Institute of Standards and Technology
NPI	New Product Introduction
NQA	Nuclear Quality Assurance
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
OD	Outside Diameter
1D	One-Dimensional
OLMCPR	Operating Limit Minimum Critical Power Ratio
OPL-3	Operating Plant Licensing Parameters
OPS	Operation Parameter Sheet
O/U	Oxygen/Uranium Ratio
PCMI	Pellet/Cladding Mechanical Interaction
PLR	Part-Length Rod
PMQC	Purchased Material Quality Control
PO	Purchase Order
P&P	Policy and Procedure
PRC	Potentially Reportable Condition
PSC	Potential Safety Concern
PSIB	Special Inspection Branch (NRR/DISP)
PWR	Pressurized-Water Reactor
QA	Quality Assurance
QATS	Quality At The Source
QC	Quality Control
QCII	Quality Control Inspector Instructions
RBM	Rod Block Monitor
RIP	Rod Internal Pressure

ABBREVIATIONS and ACRONYMS Continued

RLQR	Reload Licensing Quality Review
RMS	Root Mean Squared
RWE	Rod Withdrawal Error
SCO	Service Components Operations
SER	Safety Evaluation Report
SIL	Service Information Letter
SLMCPR	Safety Limit Minimum Critical Power Ratio
SRLR	Supplemental Reload Licensing Report
SRP	Standard Review Plan
SRXB	Reactor Systems Branch (NRR/DSSA)
TDP	Technical Design Procedure
T/H	Thermal-Hydraulic(s)
3D	Three-Dimensional
TIP	Traversing In-Core Probe
TPE	Technical Program Engineer
TPM	Technical Project Manager
TREX	Tube Reduced Extrusion
UT	Ultrasonic Testing
WA	Work Authorization

1 SCOPE AND SUMMARY OF INSPECTION FINDINGS:

From August 14 through September 1, 1995, the U.S. Nuclear Regulatory Commission (NRC) conducted an inspection of the General Electric Nuclear Energy (GENE), activities at the Nuclear Energy Production (NEP) facilities in Wilmington, North Carolina. During this inspection, the NRC inspection team conducted a performance-based evaluation of the NEP management, staff, and quality programs. They also assessed the NEP implementation of those programs related to boiling-water reactor (BWR) reload core design, reload safety analysis and licensing processes, fuel assemblies, fuel-related core components, and fuel-related inspection services supplied to the U.S. nuclear industry.

This inspection was conducted to establish a basis for confidence that NEP products and services supplied to the U.S. nuclear industry would perform their safety function. The following guidelines, standards, and regulations constitute the basis for this inspection:

- General Design Criterion (GDC) 10, "Reactor Design," and GDC 12, "Suppression of Reactor Power Oscillations," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50, "Licensing of Production and Utilization Facilities," of Title 10 of the Code of Federal Regulations (10 CFR Part 50).

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

- 10 CFR Part 21, "Notification of Failure to Comply or Existence of a Defect."

- Section 4.2, "Fuel System Design," of NRC NUREG-0800, "Standard Review Plan," Revision 2, dated July 1981, and its Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Revision 0.

- "GE Nuclear Energy, Quality Assurance Program Description," NEDO-11209-04A, Revision 8, Class 1 (approved by the NRC on March 31, 1989, as meeting the requirements of Appendix B to 10 CFR Part 50), hereafter referred to as the "QA topical report."

- Amendment 22 of the "General Electric Standard Application for Reload (GESTAR) II" topical report documented in NEDE-24011-P-A, "General Electric Standard Application For Reactor Fuel" (approved by the NRC on July 23, 1990).

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

No nonconformances were identified during this inspection.

1.3 Weaknesses, Strengths, and Observations

During this inspection, the team noted weaknesses and observations concerning NEP activities that affect quality. The more significant weaknesses, strengths, and observations are summarized in Appendix A to this report. Neither the weaknesses nor the observations described in this report require any specific action by or written response from NEP.

1.4 Open Items

A written response is requested for the open items described in this report and summarized in Appendix B.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous fuel inspection findings required follow-up during this inspection.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

GENE serves the majority of its fuel and component customers from the NEP manufacturing facility in Wilmington, North Carolina, where production of BWR nuclear fuel and reactor equipment products began in 1968. The NEP facility has been recognized for its manufacturing technology, advanced automation, and integrated information systems. GENE recently moved its core engineering staff to the NEP facility from its facility in San Jose, California.

GENE has amassed an extensive fuel performance experience base. According to GENE, as of December 31, 1993, over 4.8 million GENE 8x8 fuel-type production Zircaloy-clad UO_2 fuel rods were in, or had completed, operation in commercial BWRs. Of the total, nearly 1.5 million GENE fuel rods were in operation, and GENE had loaded more than 2 million pellet/cladding mechanical interaction (PCMI) resistant barrier fuel rods in commercial BWRs.

3.2 Entrance Meetings and Final Exit Meeting

The entrance meetings took place on August 14, 1995, for the reload core design, safety analysis, and licensing process portions of the inspection, and on August 21, 1995, for the fuel design and fuel production portions of the inspection. During these meetings, the team met with members of NEP management and staff, and discussed the scope of the inspection. The team also reviewed its responsibilities for handling proprietary information, as well as those of NEP. In addition the team established contact persons within the management and staff of the applicable NEP organizations.

During the inspection, the team conducted a performance-based evaluation of NEP through technically directed observations and assessments of processes, activities, and documentation. The team examined technical documentation, procedures, and representative records. They also interviewed and held discussions with NEP personnel. In addition, the team listened to presentations by NEP personnel, and observed work activities in progress. This report describes the specific areas examined, the documentation reviewed, and the team's findings. Appendix C to this report, list the persons who participated in and were contacted during this inspection.

During its final exit meeting with NEP management and staff, on September 1, 1995, the team summarized the open items, as well as NEP's weaknesses and strengths, and the team's observations.

3.3 Nuclear Quality Assurance

The team found that NEP's QA process was governed by the QA topical report (NEDO-11209-04A). GENE's stated policy was to obtain quality leadership, and to achieve and maintain high quality in products and services through timely and effective compliance with all quality requirements. The QA topical report described the QA program used by NEP to fulfill the regulatory aspects of this policy. The team observed that all managers within GENE with quality-related responsibility had full authority to implement the applicable elements of the program within their respective areas of responsibility. Implementation of the GENE QA program, a basic responsibility of each organization within GENE, had the unqualified endorsement and support of GENE's Vice President and General Manager.

The team noted that Nuclear Quality Assurance (NQA) reports to the General Manager of Nuclear Operations, who reported to GENE's Vice President and General Manager. NQA was a staff organization assigned responsibility for establishing the GENE level quality-related policies and procedures (P&Ps). The GENE line organizations, including NEP, had the responsibility for ensuring conformance with applicable design and QA requirements. Effective August 25, 1994, the NEP Manager, NQA, began reporting to a newly established Manager of Environmental Health & Safety and Nuclear Quality Assurance with direct access to the NEP General Manager. The team found the following NQA subsections were in effect as of August 16, 1994:

- Fuel QA, located in Fuel Manufacturing Operations (FMO) building
- Components QA, located in Fuel Components Operations (FCO) building
- Services QA, located in Service Components Operations (SCO) building
- Purchased Material Quality Control (QC)
- Quality Audits & Programs
- Fuel Engineering QA
- GENE Systems
- GENE Audits
- Quality at the Source and New Product Introduction (NPI)
- QA Programs Japan
- QA Programs Europe
- Software QA & NPI

The team concluded that the QA topical report and the NQA organization were, for the most part, implemented in an acceptable manner for the areas evaluated during this inspection.

3.4 Reload Core Design, Safety Analysis, and Licensing Processes

To evaluate NEP's reload core design, safety analysis, and licensing processes, the team reviewed the performance, interfaces, and documentation of the reload analysis processes. The reload analysis activities were found to consist of determining the licensee requirements, bundle design and core loading pattern; performing the steady-state and transient neutronic and thermal hydraulic analysis; and updating the cycle-specific reload licensing analysis and the process computer databank. The team observed that NEP performed these processes based on the technical design procedures (TDPs), and documents the results in a series of design record files (DRFs). Specifically, NEP prepared a DRF for the bundle design, reload licensing, core management, and process computer update. The team also noted that the Technical Project Manager (TPM) has the overall responsibility for the reload analysis, and was assisted by other engineers as needed. The following paragraphs describe the areas evaluated by the team and summarizes the team's findings.

3.4.1 Licensee Requirements

The licensee requirements concerning a fuel reload for a given reactor cycle were specified and documented in the fuel release engineering data (FRED) and the operating plant licensing parameters (OPL-3). The OPL-3 database theoretically contained all of the basic information to perform the analyses for the reload licensing submittal. This database had a very structured methodology for verifying (between NEP and the licensee) that each OPL-3 information entry was correct and agreed to by both parties. There were three signoffs for each information entry in the OPL-3 database: one each for NEP and the licensee, and one for resolution of any differences between NEP's data entries and those of the licensee. When the OPL-3 database was agreed to by both parties (when all differences are resolved), the licensee sent an agreement letter to NEP and NEP entered the OPL-3 database into the Boiling-Water Reactor Engineering Data Bank (BWREDB) for specific design analyses.

FRED consists of the licensee's energy utilization plan (EUP) requirements and the domain of the operating cycle, including a list of equipment out of service during the cycle. The team determined that NEP does not verify FRED with the licensee in the same manner as the OPL-3 data. Rather, the licensee simply provides FRED to NEP to enter into the BWREDB. During the review of the reload core design, safety analysis, and licensing processes for Georgia Power Company, Edwin I. Hatch Nuclear Plant Unit 1 (Hatch 1) Cycle 16, the team questioned why NEP and the licensee had not formally reviewed the FRED database as they had for the OPL-3 data, particularly since some of the data from each document was the same and shared by some calculations. The team concluded that not verifying FRED with the licensee in the same manner as the OPL-3 data was a weakness that is further discussed in Section 3.4.4.2(2) of this report.

3.4.2 Reload Core Design Process

The team reviewed the reload core design process as follows and identified that the bundle design process for the next fuel cycle (cycle n) began with the cycle n-1 core inventory and the licensee's EUP for cycle n. NEP performed the bundle physics calculations using the TGBLA[1] computer program and three-dimensional (3D) core simulation calculations using PANACEA[2]. NEP evaluated the preliminary bundle designs and core loading patterns via an iterative design process to determine an acceptable bundle performance. NEP then produced the bundle design report and completed and closed the bundle design DRF.

The determination of an acceptable reload design began with the work authorization (WA), which (via the Product and Performance Specification) identified items such as the applicable documents, procedures, design bases, and inputs from two design interface documents (FRED and OPL-3).

NEP reviewed the nuclear design bases for the reload, with specific emphasis on the choice of hot and cold eigenvalues used for the 3D simulations. NEP then determined the reload core hydraulics data as well as a reference loading pattern that satisfied the EUP, any special licensee requirements, and the nuclear design bases (e.g., hot excess reactivity, thermal limits, shutdown margins). The team noted that the reference loading pattern was the basis for the reload core licensing.

3.4.2.1 Eigenvalue Selection and Uncertainties

The team was aware of NEP's recent problems with the accuracy of the PANACEA hot and cold reactivity (eigenvalue) predictions. The team observed that the recent trend toward longer fuel cycles and new bundle designs, which utilize higher U_{235} enrichments and gadolinium (Gd) loadings, had contributed to increased scatter in the predicted hot and cold eigenvalues.

NEP told the team that the reload core design process was modified in 1994 to add a technical review of the design-basis eigenvalues for a specific reload core. "Determination of Critical Eigenvalues," TDP-0012, Revision 2, M.E. Harding, May 1995, Volume 4: Nuclear 1, was revised to reflect this requirement.

The team observed that for the Philadelphia Electric Company, Limerick Generating Station Unit 1 (Limerick 1) Cycle 6, the cold eigenvalue was mispredicted in 1994 by 0.8 percent delta K[3] (%ΔK). Similarly, in 1995, for

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[1]GENE Fuel Bundle Lattice Nuclear Design Model (TGBLA), a lattice physics code that provides lattice-averaged diffusion cross sections and relative rod power peaking for the BWR simulator core calculations

[2]A three-dimensional (3D), coupled nuclear/thermal/hydraulic computer program representing the BWR core, exclusive of the external flow loop

[3]Percent of Reactivity Addition (%ΔK)

the Boston Edison Company, Pilgrim Nuclear Power Station Unit 1 (Pilgrim) Cycle 11, the cold eigenvalue was mispredicted by 0.9%ΔK. NEP presented to the team data from other plants showing significant variation and trends in the cold and hot eigenvalues within a cycle and from cycle to cycle. The team pursued an in-depth review of the TGBLA and PANACEA benchmarking, the eigenvalue selection process, and NEP nuclear methods development activities. The following paragraphs summarize the team's findings.

NEP told the team that the computer code versions currently in use were TGBLA04 and PANAC09, and that the new versions which will be put into production by February 1996 are TGBLA06 and PANAC11. The team reviewed the benchmarking documentation for TGBLA04, and compared it with the earlier version, TGBLA03. This comparison revealed that both versions produced nearly identical results for the older 8x8 lattice designs. The TGBLA04 Software Test Report documented comparisons between MERIT, NEP's reference Monte-Carlo benchmark code at that time, and both TGBLA04 and TGBLA03. These results showed that TGBLA04 agreed with MERIT better than TGBLA03 for the newer 9x9 and 10x10 designs; however, the team noted that the eigenvalues calculated using TGBLA04 ranged from 0.015 ΔK smaller to 0.003 ΔK larger than MERIT.

In Section 2.1(1), "NRC-Approved Models," of NEDE-31917P, "GE11 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II)," dated April 1991, GENE stated that "extensive comparisons have been made with Monte-Carlo results to ensure that the 9x9 simulations are as accurate as the previous 8x8 experience." However, the team concluded that this statement was not well supported. The team observed that a very limited number of benchmark comparisons were made (three lattices at 40% and 70% void) for the GE11 design. In addition, the team expressed the concern that the comparisons were not consistent between the 8x8 and 9x9 designs, and that the relative accuracy of TGBLA04 for the GE11 design had not been established.

In discussions concerning the development history and benchmark program for the new code versions (TGBLA06 and PANAC11) NEP personnel told the team that the lattice physics code, TGBLA06, contains various enhancements. These enhancements reportedly include features that will improve the accuracy for new designs with higher U_{235} enrichments and Gd loadings by improving predictions for Gd and water rods, as well as various types of fuel and Gd rods near water rods. NEP also told the team that development of a transport theory[4] code is in progress to replace the current diffusion theory[5] code. In addition, NEP told the team that the 3D core simulation code, PANAC11, contains enhancements to its diffusion theory model, spectral and control rod history models, and rod power model.

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[4]A theory for the treatment of diffusion in a medium of neutrons or gamma rays, based on the Boltzmann transport equation

[5]An approximate theory for the diffusion of particles, especially neutrons, based on the assumption that in a homogeneous medium, the current density is proportional to the gradient of the particle flux density

The team examined the DRF documenting the TGBLA06 benchmark, software testing, and design review. The team observed that Monte-Carlo Neutron Particle (MCNP) code, with an NEP-enhanced nuclear data library, was used for the new benchmark, and that a comprehensive test matrix was run. The DRF contained comparisons between MCNP, TGBLA04, and TGBLA06 for 8x8, 9x9, and 10x10 lattices with various enrichment, Gd loading, and water rod configurations. Calculations were performed at 0%, 40%, and 70% void, cold, controlled-uncontrolled, and borated conditions. The team observed that the eigenvalue root mean squared (RMS), **[deleted pursuant to 10 CFR 2.790 - document described specific values]**, for **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** cases was **[deleted pursuant to 10 CFR 2.790 - document described a specific value]**%ΔK. The team also observed that the TGBLA06 results were significantly better than the TGBLA04 results. In addition, the scatter in K-infinity (K∞) differences compared with MCNP was significantly less for TGBLA06 than for TGBLA04.

The team also reviewed the PANAC11 benchmarking, testing, and design review documentation, and examined TGBLA06/PANAC11 benchmark comparisons to operating data from nine different reactors with three to five cycles per reactor. These results showed that, compared to the earlier versions, TGBLA06/PANAC11 significantly reduced the cycle-to-cycle discontinuity in K-effective (*Keff*) (beginning of cycle (BOC) n - end of cycle (EOC) n-1 = **[deleted pursuant to 10 CFR 2.790 - document described specific values]**), as well as the downward drift of *Keff* within the cycle. The team also noted small improvement in power distribution accuracy, as indicated by traversing in-core probe (TIP) comparisons (**[deleted pursuant to 10 CFR 2.790 - document described a specific value]**% RMS vs **[deleted pursuant to 10 CFR 2.790 - document described a specific value]**%).

The team reviewed the hot and cold eigenvalue selection and review processes, and discussed its findings with NEP management and the TPMs. In addition, the team reviewed TDP-0012 for determining hot and cold eigenvalues, and compared the procedure to recent plant DRFs. The team observed that the empirical methodology in TDP-0012 for adjusting the hot and cold eigenvalues did not necessarily improve predictions, as indicated by the Limerick 1 Cycle 6 and Pilgrim Cycle 11 results described below. The team also observed that the TPMs were considering the prior cycle eigenvalue errors, critical value vs prediction, for both the hot and cold eigenvalues. In addition, the team noted that TDP-0012 required a formal review of the eigenvalues to be used before each reload safety evaluation was initiated. The team evaluated documentation from this process for several reloads.

The team concluded that the 9x9 lattice benchmark did not support GENE's statement in NEDE-31917P pertaining to lattice simulation accuracy. Specifically, the team expressed the following concerns:

- Limited comparisons were made between the 9x9 lattices and the Monte-Carlo benchmark code.

- Very limited inter-comparisons of these 9x9 benchmarks were made against the comparable 8x8 lattices compared to the Monte-Carlo benchmark code.

- The comparisons that were made did not show consistent results.

- The relative accuracy was not established.

- A considerable number of comparisons between TGBLA04 and TGBLA03 results were made for 9x9 lattice conditions; however, the team did not consider these to be benchmarks in conformance with the statement in Amendment 22 of GESTAR II.

The team concluded that the errors in calculated hot and cold core reactivity for the recent reloads with the newer fuel designs and long cycle lengths was a weakness of the NEP reload design process. The team noted, however, that NEP has addressed this weakness by implementing steady-state nuclear methods improvements (discussed in Section 3.4.2.3 of this report).

3.4.2.2 Cold Shutdown Margin

At the start of each new operating cycle, each licensee preforms a cold critical shutdown margin demonstration test. The team noted that licensee technical specifications require a cold shutdown margin ranging from 0.25 to 0.38%ΔK. The team was told that NEP typically designs a reload core to have a shutdown margin of at least 1%ΔK. The team observed, however, that should the recent NEP mispredictions of cold eigenvalues continue, the potential exists to have less shutdown margin than required. For example, several plants have missed the cold shutdown K by approximately 0.9%ΔK. The team's evaluation determined that had the cores been designed to a 1% shutdown margin, the misprediction would have resulted in a shutdown margin of 0.1 to 0.2%ΔK compared to the required 0.25 to 0.38%ΔK. The team postulated that the differences between the predicted shutdown margin and the true shutdown margin could be attributed to inaccuracies in the TGBLA/PANACEA model, inaccuracies in modeling the true reactor conditions during the test, or the test procedure itself.

To evaluate these concerns, the team reviewed indepth three recent cold critical shutdown tests (Duane Arnold Cycle 14, Pilgrim Cycle 2, and Limerick 1 Cycle 6), as discussed in the following paragraphs.

In each case, the calculated cold shutdown margin for each reload core is reported in the supplemental reload licensing report (SRLR). The cold shutdown margin is reported in terms of the cold, BOC K_{eff} for the core condition of all inserted control rods, except the strongest worth rod. In addition, the reload R-value, maximum increase in cold core reactivity with exposure expressed in ΔK, is reported. The following paragraphs summarize the team's findings:

(1) Duane Arnold Cycle 14

During the Cycle 14 start-up of the Duane Arnold Energy Center plant, the calculated limiting eigenvalue was found to have a margin of only .07% in excess of the technical specification requirement. Preliminary assessment had indicated that the calculated shutdown margin in reactivity was actually less than the technical specification value, causing a ten-hour delay in plant start-up. NEP characterized the incident as a "near-miss," and prompted a root cause evaluation by GENE (memorandum from G.E. Dix to J.S. Armijo on "Root Cause Evaluation of Duane Arnold Shutdown Margin," dated May 16, 1995). The calculated shutdown margin was found to be less than the nominal design value by about a factor of 2. The team determined that such large differences between the calculated shutdown margin and the nominal design value raise concerns regarding the potential for reduced reactivity margins in the safe shutdown capability of the core, as well as the accuracy of the design methods (and/or their implementation) used to ensure that the reactor core can safely be operated.

The relatively large deviation of the calculated core shutdown margin from the nominal design value for Duane Arnold Cycle 14 is not an isolated incident, and GENE had been aware of the potential issues mentioned above. For example, an 0.8% deviation of the Limerick 1 Cycle 6 calculated cold critical eigenvalue from the design target value prompted a memorandum from J.E. Wood, Manager, Design Process Improvement to the TPM, regarding "Cold Critical Eigenvalue," dated April 11, 1994. The memorandum addressed the regulatory and operational implications of the inconsistency between the calculated and selected target eigenvalues. It also provided the following specific directions, among others:

- Select cold critical eigenvalues on a best estimate basis.

- Provide sufficient cold shutdown margin in the design to meet technical specification requirements.

- Provide a special review of the cold critical eigenvalue selection process.

A later memorandum regarding "Critical Eigenvalue Review," dated August 10, 1994, addressed the issue at greater length. The memorandum characterized the difference between the calculated critical eigenvalue and 1.0000 as separable into two components:

(a) a predictable bias that can be specified at the time of core design for a specific operating cycle

(b) an uncertainty that may cause the calculated critical eigenvalue to move up or down, with respect to the bias, during the operating cycle

The team noted that the objective of the design-basis eigenvalue is to negate the impact of the predictable bias. The memorandum of August 10, 1994, also specified target RMS differences between the design-basis and calculated critical eigenvalues for the BWRs in operation. These target RMS differences had been specified for BOC cold and EOC hot conditions, as well as for the hot eigenvalue trajectory difference. The memorandum noted that the BOC cold critical eigenvalue predictive capability did not meet the accuracy target values. However, the memorandum noted that the EOC hot eigenvalue predictive capability met the accuracy target values, while the hot critical trajectory predictive capability met the accuracy target values only for the low cycle energy cores. The memorandum also recommended design process improvements involving both eigenvalue selection and the nuclear models used in the design calculations.

The team's review of the root cause evaluation concerning the erroneous Duane Arnold Cycle 14 eigenvalue calculation revealed that a little less than one half of the difference between the calculated and design-basis eigenvalues was attributable to an error in applying the design process. Specifically, the results of an improved method for eigenvalue temperature correction had not been properly accounted for. This human error aspect of the incident was discussed extensively in the root cause analysis. The analysis also identified a number of remedial actions, including improved training, more explicitly written procedures, and better implementation of the design verification process.

Even after accounting for the contribution of the error in temperature correction, the root cause evaluation revealed that a recent trend in the shift in eigenvalues in the BWRs in operation had not been factored into the design process. (This last observation is consistent with the observations reported in the memoranda written by J.E. Wood.) Specifically, the team noted that the eigenvalue shift noted in the root cause evaluation occurred at Duane Arnold despite one of the recommendations made in Wood's memorandum of August 10, 1994. (That recommendation involved an improvement in the eigenvalue selection process, being incorporated in the TDP-0012, and implemented in the Duane Arnold Cycle 14 design calculations.) The second recommendation in the Wood's memorandum, improvements in the nuclear models, had not been addressed in the root cause evaluation; however, such improvements were discussed in some detail at a presentation to the team by NEP during the course of this inspection. Significant aspects of the improvements in nuclear models are discussed below in Section 3.4.2.3 of this report.

(2) Pilgrim Cycle 11

The team's review noted that for Pilgrim Cycle 11, the SRLR specified a cold *Keff* of 0.986 to yield a BOC cold shutdown margin of 1.4%ΔK. The specified R-value was 0.003 to yield a minimum shutdown margin of 1.1% sometime later in the cycle. The calculations of these values including the determination of the cold and hot target eigenvalues were performed according to the required process and relevant TDPs. Methods prescribed by TDP-0012, Revision 1, were used to adjust the hot critical eigenvalues for the amount and location of the Gd rods and cycle exposure, and to adjust the cold eigenvalues for exposure. The team noted that NEP compared predicted and measured results from Pilgrim Cycle 10 to other units.

The NEP evaluation team then conducted and documented a technical review of the hot and cold eigenvalue selections report, in accordance with TDP-0012. The team's review noted that the results of the technical review were all based on the TGBLA04 and PANACEA09/10 models; TGBLA06 and PANACEA11 checks were not made.

The actual measurements showed a BOC cold shutdown margin of approximately 0.4%ΔK. Thus, in the BOC cold shutdown with the most reactive rod out, the core was approximately 1.0%ΔK more reactive than predicted by the accepted procedures and computer codes. Pilgrim conducted the actual measurements using local critical measurements (i.e., withdrawing the most reactive rod (object rod) with an adjacent rod (margin rod) set at the required shutdown margin) rather than in-sequence critical measurements (i.e., control rods are removed by bank in-sequence so that the core remains approximately symmetrical over the rods). Pilgrim terminated the initial measurement when it appeared that the core was approaching criticality. Subsequently, with revised data from NEP, Pilgrim repeated the measurement, and made two additional shutdown margin measurements using two other strong rods with adjacent rods as the margin rod.

NEP performed a substantial analysis to understand and isolate possible contributors to the 1.0%ΔK misprediction. These calculations identified two such contributors. The more significant contributor was the lack of burnup mesh points in the nuclear data available to PANACEA. (The mesh points in burnup space did not closely match the bundle peak K_∞ values.) The less significant contributor resulted from control rod depletion. These two effects contributed to a total of 0.4%ΔK, leaving 0.6%ΔK unexplained quantitatively.

Pilgrim also performed in-sequence measurements, and observed some misprediction in the cold BOC critical K. However, this misprediction at 0.3%ΔK was not large. The significance of this value is the comparison against the local critical misprediction of 1.0%ΔK. That is there is a 0.7%ΔK prediction bias between the two types of cold measurements. The team noted that the NEP shutdown margin measurement using in-sequence critical values under-estimated the Pilgrim Cycle 11 BOC cold shutdown margin compared to a direct measurement using local critical values as conducted by Pilgrim. At the time of the inspection, the Pilgrim- specific cause or more likely causes of this 0.7%ΔK were not identified for the team.

The team determined that with a BOC value of 0.4%ΔK, and a SRLR R-value of 0.3%ΔK, it is *possible* that the required shutdown margin at the most reactive burnup point will not be met. "Possible" is used here since the stated values are rounded, and the R-value may have decreased. Therefore, the team identified the need to determine the current shutdown margin value at the most reactive time in Cycle 11. The team considers this an open item, and requested that NEP notify the NRC upon identifying this value. (Open Item 95-01-01)

(3) **Limerick 1 Cycle 6**

The team's review indicated that for Limerick 1 Cycle 6, the SRLR specified a cold *Keff* of 0.980, thus implying a BOC cold shutdown margin of 2%ΔK [(1.00 - 0.980) x 100]. The specified R-value was 0.004, indicating that the shutdown margin was at a minimum sometime later in the cycle, and that the margin was 1.6%ΔK. The calculations of these values including the determination of the target cold eigenvalue was reviewed by the team and compared to the actual results for Cycle 6 operation.

The Limerick reload licensing DRF documented the process of selecting the eigenvalues. The Limerick 1 Cycle 5 BOC *Keff* was 1.004, and the Cycle 4 value was 0.998. An earlier version of the TDP-0012 empirical ΔK correction methodology produced a correction of +0.0047ΔK. Thus, both the prior cycle results and the correction methodology indicated that the BOC cold eigenvalue should be about 1.004 to 1.005. The target cold eigenvalue for BOC Cycle 6 was set at 1.004 via the eigenvalue selection process.

The calculated PANACEA eigenvalue at the actual plant cold critical condition was 0.996 which resulted in a reactivity misprediction of 0.8%ΔK. The its review, the team determined that this implies that the BOC cold shutdown margin may have been 1.2%, instead of the calculated 2%. The team concluded that because this reload was designed with a large shutdown margin, the technical specification shutdown margin was satisfied.

From its evaluation of these cold critical shutdown tests as well as the evidence from Pilgrim Cycle 11 predictions and measurements led the team to conclude that NEP's current methods are not adequate to predict the technical specification requirements for the most reactive rod-out shutdown margin based on in-sequence critical measurements. The team determined that NEP's present errors in target RMS deviations in eigenvalues, and to adequately quantify biases and uncertainties in eigenvalue calculations especially for high-energy cycle, heterogeneous cores was a weakness. The issue of accuracy in eigenvalue calculations relates to the issue concerning the accuracy of design methods used to ensure that the reactor can be safely operated, as discussed in Section 3.4.2.3 below.

3.4.2.3 Steady-State Nuclear Methods Improvements

NEP explained to the team that recent trends in BWR operation and fuel management are characterized by longer cycles, higher fuel exposures, fuel enrichments, burnable poison loading, and lower design margins to reduce fuel cycle costs. NEP stated that these trends have posed a challenge to traditional GENE nuclear predictive capabilities.

To enhance their predictive capability, NEP has initiated a phased program designed to improve the lattice physics and core physics models used in the design calculations. The team found that the lattice physics changes included improved cross-sections, better neutron moderation and resonance shielding treatment, more detailed modeling of burnable poison depletion, improved spectral weighting of diffusion coefficients, and an improved subchannel void distribution model. These efforts were based on a significant number of lattice conditions, and thus also provide a basis for statistical quantification of reactivity differences for the design process.

Core physics enhancements included improved neutron diffusion models, improved cold temperature and rod power models, and better accounting of spectral history and control rod history. NEP reported that the updated nuclear models and codes were being benchmarked, and will be used as design tools after the first quarter of 1996. NEP also reported that application of the new design models shows that the new models significantly improve performance for approximately 30% of the plants with high-energy cycles and highly heterogeneous cores, where traditional design methods do not provide the target predictive accuracy. The new models also provide slightly improved performance for approximately 70% of the plants with lower-energy cycles and more homogeneous cores, where traditional methods already provide the desired predictive accuracy.

The team concluded that the new design methods, therefore, have the potential to resolve at least some of the presently observed shortcomings in eigenvalue predictions. Additionally, NEP had undertaken measures to reduce the incidence of human errors by improving procedures and training engineers in the proper use of the procedures.

NEP management recognized the team's concerns. NEP presented its evaluation and plans, and participated in several discussions with the team on the subject of shutdown margin. In addition, NEP informed the team of the following plans to address these concerns:

- GENE promised to release a service information letter (SIL) to its customers by February 1996, addressing the eigenvalue and shutdown margin concerns.

- NEP will rely on its review process to ensure safe shutdown margins.

- Parallel predictions will be performed using both the current (TGBLA04/PANAC9) methodologies and the new (TGBLA06/PANAC11) methodologies and shutdown margins will be checked with the new methodologies to ensure the 1% design-basis margin.

- Future cold PANACEA models will have multi-temperature neutronic data from TGBLA.

- GENE will revise the appropriate TDPs to reflect the new methodology.

- NEP will review the cold critical test procedure with licensees to determine if a more appropriate test for shutdown margin should be performed.

- GENE will accelerate the development of new methods, including the transport theory lattice code.

The team concluded that the near-term use of TGBLA06/PANAC11 in combination with the eigenvalue selection process should help reduce uncertainties in the cold critical and shutdown margins. The team also concluded that the assurance of adequate shutdown margin can be strengthened by joint GENE/licensee actions consistent with the plant startup safety analysis. The team also discussed with NEP other near-term actions that may include analytical changes, such as increasing the design shutdown margin from 1 to 2%, or implementing test procedure changes (such as performing a strongest rod out cold critical measure instead of an in-sequence critical measure).

In reaching its conclusions concerning this issue, the team considered the information provided in Wood's memoranda, as well as the root cause evaluation of the Duane Arnold Cycle 14 eigenvalue calculation, and the presentations made on the improved steady-state nuclear methods. For shutdown margin calculations, specifically, these requirements are the ability to quantify the predictable biases and uncertainties, and to demonstrate that the target RMS deviations in eigenvalues are being met. GENE's predictive capabilities in eigenvalue calculation have fallen short of its own expectations in the recent past. However, the team also concluded that GENE has a process in place, involving improved nuclear models and better procedures and training, that has the potential to resolve this problem. However, the team observed that to ensure that NEP's effort is meeting its intent, NQA review and monitoring would be appropriate.

The team concluded that, given that the introduction of the new nuclear models and codes constitutes a major upgrade of GENE's nuclear design methods, the use of the new models as design tools after the first quarter of 1996 is not an unreasonable schedule.

3.4.3 Reload Safety Analysis Process

During its review of the reload safety analysis process, the team found that NEP performs the reload safety analysis for approximately 20 reload cores each year. The team noted that the reload safety analysis process was highly automated and computerized, with minimum user involvement in the code-to-code and code-to-database interfaces. The reload safety analysis is initiated by the final release design basis work authorization (WA), the fuel application design basis WA, and/or the Licensing WA. The reload licensing activities with coordinated with the licensee and within NEP by the Fuel Project Manager. The responsible engineer performs and documents the reload safety analyses and the TPM oversees the technical details of the analysis.

During its review of the process, the team found that the OPL-3 and the FRED documents, together with the WAs, provide the plant operating and design data required for the reload analysis. The OPL-3 is an extensive document, which includes initial operating conditions (e.g., power level, flows, and temperatures), scram parameters (e.g., setpoints, response times and delays), and plant equipment parameters (e.g., volumes, lengths, and capacities). After resolving all comments and finalizing the data, the licensee provides NEP with a final signed and verified version of the OPL-3.

The FRED document provides fuel bundle identification and exposure data, cycle energy requirements, selected cycle margin and operational flexibility improvement options, limiting transient analysis statepoints, technical specification requirements, and the plant equipment status.

The team found that the NEP reload safety analysis process is documented and controlled using an extensive set of TDPs, nuclear engineering technical procedures documents (e.g., the GETAB design procedure document), design bases (DBs) documents, and engineering operating procedures (EOPs) given in NEDE-21109. The TDPs and EOPs describe the overall methodology, as well as the task-specific methods, assumptions, and interfaces. The EOPs provide the general requirements concerning computer codes, design review/verification, DRFs, WAs, engineering records, and reload licensing. In addition, NEP is currently establishing a system of "analysis block guides" that will provide additional detail for performing and verifying the reload analyses.

The found that the results of each reload analysis are documented in detail in a system of DRFs, according to the requirements of EOP 42-10.00, "Design Record File," dated March 27, 1995. The detailed results of each of the major reload tasks (e.g., reload licensing, core monitoring, and bundle design) are documented in a separate DRF. The responsible engineer compiles and maintains the DRF.

In order to ensure the quality and traceability of the engineering data, NEP employs the computerized BWREDB system. The BWREDB is used to store and retrieve engineering input/output data, and to facilitate automation of the reload design and safety analysis process. The BWREDB includes data from previous cycles, as well as the data necessary for the current reload cycle analysis. The team's review of the BWREDB system is described in Section 3.4.6.1 of this report.

The team determined that the NEP's reload safety analysis provides the technical basis to support cycle operation and ensure adequate margin to the safety limits during normal operation and anticipated operational occurrences (AOOs). The analysis consisted of several major evaluations, including nuclear design-basis analysis, core hydraulics analysis, determination of the reference core loading pattern, transient analysis, loss-of-feedwater heating (LFWH) analysis, rod withdrawal error (RWE) analysis, mislocated fuel bundle (MFB) analysis, loss-of-coolant accident (LOCA) analysis, and (i) the stability analysis (where required).

The team noted that the reload analysis also includes a fuel-dependent evaluation of the LOCA and a determination of the peak clad temperature, clad oxidation fraction, and maximum average planar linear heat generation rate (MAPLHGR) limit versus fuel exposure. This analysis is performed using the SAFE/REFLOOD/CHASTE code, using gap conductance and fission gas release data determined by GEGAP/GESTR[6], or SAFER/GESTR-LOCA.

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[6]GE-NE Stress and Thermal Analysis of Fuel Rods (GESTR), a computer program to provide best-estimate predictions of the thermal-mechanical performance of nuclear fuel rods experiencing variable power histories

Currently, U.S. BWR nuclear plants are not required to submit a reload stability analysis. Some licensees, however, perform a cycle-specific stability analysis to evaluate the core and channel hydrodynamic stability margins for selected points on the power-flow operating map. The core and channel decay ratios are determined with the ODYSY (frequency-domain) stability code using ISCOR thermal-hydraulic input and cycle-specific dynamic parameters (e.g., reactivity coefficients and neutron lifetime)

The team also noted that in addition to the safety analysis, the cycle reload evaluation includes the fuel bundle design, fuel cycle analysis, core design and management, and core monitoring analysis. The fuel bundle design analysis is performed to determine the bundle neutronics characteristics including reactivity coefficients, pin-wise power distribution, and nuclear data for input to PANACEA. The fuel cycle analysis evaluates the reload cycle core performance, and the core management analysis provides the utility information concerning reactor startup, reactivity curves, and operating strategy. The core monitoring analysis is performed to define the site process computer data bank.

The team evaluated NEP's task-wise verification of the individual analyses (e.g., verification of the rod withdrawal accident analysis), and found that it consisted of a committee review of critical steps in the reload process (e.g., the eigenvalue selection and transient selection reviews), and an overall reload licensing quality review (RLQR), as described in Section 3.4.4.2 of this report. Many of these verifications were performed with detailed checklists that identified the critical issues in the reload evaluation. The team also noted that utility staff frequently participated as members of the NEP reload review committees. Documentation of NEP's verification was included in the DRF.

After NEP verified the analyses and results, NEP entered the reload data to the BWREDB. NEP provided the results of the reload evaluation to the utility in the SRLR and the NDR.

The team concluded that these evaluations were generally performed using the methods described in Amendment 22 of GESTAR II and that NEP's verification of the reload evaluation was extensive.

3.4.3.1 Cycle-Dependent Safety Limit Minimum Critical Power Ratio

In discussions with the team, and also at a meeting with the NRC staff on March 16, 1995, NEP indicated that it intends to submit a cycle-dependent SLMCPR methodology for application to reload cores. According to NEP, the new cycle-dependent methodology will use the plant cycle-specific power distribution rather than the bounding power distribution used in current SLMCPR determination. The team determined that this change will reduce the SLMCPR, as well as the conservatism in the CPR thermal margin for some plants. While the final documentation of the methodology has not been completed, the cycle-specific SLMCPR was considered to be of substantial importance and was included in the team's discussions with NEP during this inspection. The following paragraphs summarize the team's findings.

NEP reported that the present SLMCPR method was developed several years ago. Since that time, the method had become overly conservative for many reload cores because of the improved thermal margin of recent fuel bundle designs, and because of the increased number of core designs that must be bounded. The present method calculated the SLMCPR necessary to ensure that 99.9% of the fuel rods avoid boiling transition. The expected number of rods experiencing boiling transition was determined with GESAM using a Monte-Carlo approach accounting for uncertainties in the GEXL correlation and the bundle power. The core statepoint was selected under the following conservative assumptions:

- a bounding equilibrium core

- a core radial power distribution selected to maximize the number of fuel bundles at or near thermal limits

- a local power distribution selected to maximize the number of rods near boiling transition

The team determined that the cycle-specific methodology relaxes the conservatism inherent in each of these assumptions. On the basis of the team's discussions with NEP about the planned cycle-dependent SLMCPR methodology, the team made the following observations:

- Before eliminating the present conservatism in the SLMCPR determination, the team questioned whether this conservatism was required in the original methodology to accommodate other methods approximations or non-conservatism (e.g., lack of CPR data, differences between the ATLAS[7] tests and actual core operating conditions, statistical assumptions, and rod bowing).

- The present SLMCPR statistical method was a bounding core statepoint, and no allowance was required to account for uncertainty in the assumed statepoint. However, the cycle-specific approach involves, an analysis to determine the limiting core statepoint, and an allowance for uncertainty may be required in this determination.

- The MCPR determination currently incorporates channel bow effects by modifying the fuel bundle R-factor to account for the increased local peaking caused by channel bow. NEP indicated that the new methodology may include the channel bow effects by increasing the R-factor uncertainty in the Monte-Carlo process. However, the Monte-Carlo analysis only accounts for the random component of the channel bow effect. Consequently, the systematic (or bias) component of the local power increase resulting from channel bow should be determined and included separately as an adjustment to the bundle R-factor.

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[7]GENE's 8.6-megawatt heat-transfer loop

3.4.3.2 Rod Withdrawal Error Analysis

The rod withdrawal error analysis determines the rod block monitor (RBM) setpoints, which ensure the necessary critical power and thermal mechanical linear heat generation rate (LHGR) margin. The team's review found that the core statepoint analysis and response to the control rod withdrawal are determined with the PANACEA 3D simulator. In the analysis, the highest worth rod[8], with a fresh bundle immediately adjacent, is selected and withdrawn from the statepoint of peak hot excess reactivity, determined in the reference loading pattern analysis. A statistical approach was used to determine the RBM setpoints for plants equipped with the average power range monitor (APRM), RBM, and Technical Specification upgrade.

3.4.3.3 Reload Licensing Transient Analysis

The reload licensing transient analysis determines the transient overpressurization and reduction in margin to fuel thermal limits and was performed using ODYN or REDY. The team's review found that the reload transient analyses included the limiting pressure and power increase events, feedwater controller failure event, and the main steam isolation valve (MSIV) closure (flux scram) overpressurization event. The ODYN[9] one-dimensional (1D) computer code and the REDY point-kinetics transient code are used to analyze these events and evaluate the reduction in critical power ratio (CPR) and/or system overpressurization. The PANACEA code is used to analyze the LFWH, RWE, and MFB events. (The NRC has approved the steady-state PANACEA analysis of the LFWH transient). The GETAB code is used to analyze the fuel type-dependent operating limit minimum CPR (OLMCPR) for these events, using the ODYN or PANACEA calculated heat flux, the ISCOR thermal-hydraulics data, and the fuel-dependent GEXL[10] correlation.

The team found that the standard reload transient analysis uses precalculated core nuclear characteristics (neutron lifetime; delayed neutron fraction; and doppler, void, and scram reactivity). NEP's transient analysis acceptance criteria were based on the requirements that (a) the number of fuel rods in boiling transition is limited to a maximum of 0.1%, (b) the cladding plastic strain is less than 1% and fuel centerline melt is precluded, and (c) both NRC and operational system and vessel pressure limits were not violated.

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[8]For plants equipped with the rod worth limiter, a four-rod gang is withdrawn.

[9]Computer code that simulates the dynamic behavior of BWRs; ODYNM is applicable to plant analysis beginning with BWR/1 through BWR/4; ODYNV simulates dynamic behavior of valve flow control in BWRs, and is applicable to plant analysis beginning with BWR/5.

[10]GENE Critical Quality Boiling Length Correlation (GEXL) computer code

The team determined that since many licensee final safety analysis report (FSAR) transients are not limiting and/or have minimal sensitivity to reload dependent parameters, typical NEP reload analyses only included a selected set of potentially limiting transients. Typical limiting transients include the turbine trip, load rejection, pressure regulator failure, MSIV closure, LFWH, feedwater controller failure, and feedwater/high-pressure coolant injection events. The team observed that transient-specific TDPs and EOPs specify various assumptions concerning plant equipment performance in these analyses.

3.4.3.4 Quadrant-Symmetric Fuel Failures

During the team's review, NEP provided a discussion concerning an incident in which fuel rod failures occurred in quadrant-symmetric fuel bundles. The bundles containing the failed rods were located in quadrant-symmetric peripheral locations with the reactor during the previous two cycles, and were moved into the central high-powered region of the core during the last cycle. When the first fuel rod failed, the control rods in the failed bundle location were inserted, as were the control rods in the symmetric locations. Although the power was reduced in the symmetric bundles with no fuel failures, fuel rods in these bundles failed later in the cycle (presumably because of operation before control rod insertion).

While NEP reported that the fuel failure mechanism is presently unknown, the team determined that the incident suggests a quadrant-symmetric mechanism that occurs at symmetric locations. The team noted that the failure could be the result of an analytical error (associated with 3D-MONICORE, PANACEA, rod block setpoints, or RWE analysis, for example), since these tend to be quadrant-symmetric. Therefore, the team identified this issue as an open item, and requested that NEP notify the NRC upon identifying the failure mechanism. (Open Item 95-01-02)

3.4.4 Reload Licensing Process

To evaluate the reload licensing process, the team began its evaluation with the documents supplied to the licensee as the end result of the reload licensing process. The following paragraphs summarize the team's findings.

The nuclear design report (NDR) is one of two NEP deliverables to the licensee resulting from the reload licensing process. This report documents the reload core loading pattern and fuel bundle design, as well as the key assumptions and methodology for the analysis, and the anticipated nuclear and T/H core performance. In addition, the report documents the following topics:

- analytical methods used in the analysis
- design objectives for the reload
- description of the core and fuel
- the loading pattern
- the thermal performance and reactivity behavior
- results from steady state neutronic safety calculations

The team reviewed the NDRs as part of the reload design evaluation, and concluded that the required information was adequately documented with traceable references.

The supplemental reload licensing report (SRLR) is the second of the two NEP deliverables to the licensee resulting from the reload licensing process. This report supplements GESTAR II Amendment 22 (the GENE licensing document) for the specific reload being evaluated. The SRLR summarizes the fuel in the reload core, the steady state and transient safety results, and the resulting cycle-specific SLMCPR. In addition, the SRLR documents the following topics:

- the shutdown margins
- the anticipated operational occurrences (AOOs) analysis
- cycle specific margin improvements and operating flexibility
- the cycle SLMCPR and OLMCPR
- the results from the pressurization and non-pressurization events
- discussions concerning events or accidents not analyzed on a cycle-specific basis because they were bounded by the GESTAR II analysis.

The team reviewed the SRLR reports as part of the reload design evaluation, and concluded that the required information was adequately documented with traceable references.

3.4.4.1 Reload Core Monitoring Data Bank

The site computer performs an online core performance evaluation, using the Core Monitoring Data Bank (CMDB) together with the reactor instrumentation (e.g., neutron flux, core flow, and temperatures), to determine the core power distribution and evaluate the fuel thermal limits. The team found that the input used to generate the CMDB includes a set of full-core PANACEA calculations at selected cycle statepoints from the current and previous cycles. The input also includes cycle-specific operating options, thermal limits, and core loading data taken from the BWREDB. Based on that input, the resulting CMDB includes the following engineering data:

- channel-type core maps
- nuclear cross-sections
- T/H constants
- pressure-loss coefficients
- reactivity coefficients
- heat balance data
- gamma and thermal TIP factors
- ARTS coefficients
- R-factor data.

The team concluded that generation of the CMDB is highly automated. The CMDB undergoes an extensive verification including comparisons of the site computer predictions (using PANACEA "measurement" data) with the corresponding PANACEA predictions. These predictions must satisfy specified design acceptance criteria concerning power distribution and thermal limits. When the generation of the CMDB is complete, the data is stored in the BWREDB and transmitted to the licensee by the Fuel Project Manager.

3.4.4.2 Reload Licensing Quality Review

Reload Licensing Quality Review (RLQR) was the last step in the quality assurance and verification process to which the SRLR and the NDR are subjected before being released to the licensee. The team determined that the RLQR primarily concerns the integration and consistency of the SRLR. It therefore constituted a higher-level verification of the engineering analysis and data utilized in generating the SRLR and the NDR. A summary of the RLQR was maintained in a DRF by the chairman of the Review Board. The team concluded that together with the Eigenvalue Selection Review Committee and the Transient Selection Review Committee (which have a narrower focus), the RLQR brings to bear a broader range of expertise on the engineering analysis process.

The team found that the RLQR was formatted into two parts. The first part of the review involves an overview of the scope and key results of the reload analysis. For this part of the review, the Fuel Projects Manager and the Licensing Engineer were required to be present. The second part of the RLQR involved a more detailed review of the technical content of the reload analysis. This part of the review required the participation of an engineering review team with expertise in the five process areas (bundle design, fuel cycle, core design and core management, T/H and transient analysis, and core monitoring). The technical package contained the following key information:

- the nuclear design bases
- the reference loading pattern
- RWE, LFWH, and MFB analyses
- nuclear design parameters
- reload transient analyses
- the GENE thermal analysis bases
- reload core hydraulics
- the emergency core cooling system analysis
- the stability analysis

The team determined that the RLQR process is generally serving its intended function of providing a fairly broad range of QA and design verification. Nonetheless, the team concluded that the errors in eigenvalue calculations for Duane Arnold 1 Cycle 14 indicate that the RLQR process is not a substitute for meticulous verification of each element of the design calculation.

To evaluate NEP's reload licensing practices, the team conducted an indepth review of four recent reload packages (La Salle 1 Cycle 7, Hatch 1 Cycle 16, Limerick 1 Cycle 6, and Pilgrim Cycle 11). The following paragraphs summarize the team's findings:

(1) La Salle 1 Cycle 7

During this inspection, the team reviewed the RLQR for Commonwealth Edison Company (CEC), La Salle County Nuclear Power Station Unit 1 (La Salle 1) Cycle 7. The RLQR contained the signatures of the Fuel Project Manager, the Licensing Engineer, and the Engineering Review Team, indicating their acceptance of the reasonable accuracy and adequacy of the reload licensing analysis. The Engineering Review Team had identified six open items related to details of the analyses performed. Within 7 days of the RLQR meeting, the responsible engineer had resolved the concerns that had been raised to the satisfaction of the Engineering Review Team.

La Salle 1 is one of four CEC plants for which CEC is currently performing all or some of the fuel management, safety and licensing, and plant support services. The team found that GENE is currently performing the balance of these services as well as fuel rod assembly, design, and fabrication services. However, the responsibilities for fuel management, safety and licensing and plant support are currently being transferred from GENE to CEC in a phased manner, distinct for each plant, over several fuel cycles.

The team noted that the input engineering data sources (such as the final release design basis WA, licensing WA, FRED, and OPL-3) were ordinarily sufficient to define the scope of the fuel management and safety and licensing work at NEP. In this case, however, the complex, phased transfer of responsibilities from GENE to CEC has necessitated the development of a design interfacing procedure (DIP). CEC and NEP jointly developed the DIP in compliance with a specific fuel contract provision, which required establishment of interfacing procedures required by the purchaser's assumption of fuel management services. The team's review determined that the DIP provides guidelines for all nuclear design-related technical interactions between CEC and GENE during the transfer of responsibilities from GENE to CEC.

In accordance with the split scope responsibility, the licensing WA only specified the calculation of stability, emergency core cooling system (ECCS), LOCA, and MAPLHGRs by GENE for La Salle 1 Cycle 7 core operation. The DIP specified the detailed scope of the analyses to be performed to meet the WA specifications, including responsibilities for primary and backup analyses.

As part of this inspection, the team reviewed the Nuclear Fuel Independent Design Verification Checklist, the Reload Transient Analysis Design Verification Guide, the Reload Licensing Transient Analysis Selection Review Summary, LOCA Analysis Results, Stability Analysis Results, and the RLQR Report for La Salle 1 Cycle 7. On the basis of this review, the team found that each document had the requisite signatures of the responsible engineer, verifier, and review members (when appropriate). Because of GENE's limited scope in this reload licensing analysis, the DRFs were much less extensive than for a full-scope reload licensing analysis. None the less, the team found that the results of all analyses required by the DIP for La Salle 1 Cycle 7 were documented. The inspection indicated NEP had performed the analyses required for reload licensing of La Salle 1 Cycle 7, in accordance with existing procedures and guides. The team also found that the DIP had served its intended function of clearly specifying the division of responsibilities for primary and backup analyses between GENE and CEC.

The team concluded that the development of a comprehensive and workable DIP, in cooperation with the licensee, to respond to a unique situation (e.g., phased transfer of fuel management and licensing analysis responsibilities) was a strength of NEP.

(2) Hatch 1 Cycle 16

The team found that NEP conducted the Hatch 1 Cycle 16 reload evaluation using the methodology described in Amendment 22 of GESTAR II. The Cycle 16 reload consisted of four GE13 Lead Use Assemblies (LUAs), 180 fresh GE9B bundles, and 376 irradiated fuel bundles. To provide improved operating flexibility and cycle extension, NEP performed an expanded operating domain analyses for increased core flow (ICF) and final feedwater temperature reduction (FFWTR).

The team determined that the Cycle 16 reload was a split-scope analysis. That is, the licensee performed the initial fuel bundle design, fuel cycle analysis, and application fuel cycle analyses, while NEP performed and verified the final fuel bundle design, licensing, core management[11], and core monitoring analyses. Each of these analyses were performed and documented by a different responsible engineer.

Each of the NEP analyses is documented in a separate DRF, which details the formal record of the engineering analysis supporting the reload evaluation. The team reviewed each of the Cycle 16 DRFs; however, the primary focus was on the Reload Licensing DRF (J11-02346) and Core Monitoring DRF (J11-02415). In addition, since the Reload Licensing DRF consisted of 19 volumes, only portions were reviewed in detail (specifically, Volume 1, "OPL-3/FRED"; Volume 6, "PANACEA Collapse"; Volume 7, "ODYN Transient Calculation"; and Volume 8, "GETAB ΔCPR Calculation").

The team's review of DRF J11-02346, Volumes 1 and 6–8, indicated that the core and operating conditions were as specified in the EWA, and that the domains, options, and exposure were consistent with the specification. In addition, the review indicated that the data transfers to the BWREDB were documented and verified. The review of DRF J11-02415, Volumes 1 and 2, indicated that the Cycle 16 core monitoring data generation was performed using the extensive 3D-Monicore QA procedure. However, during the review of this verification procedure, the team identified a concern that the specified 3D-MONICORE/PANACEA accuracy acceptance criteria (i.e., standard deviations) did not appear as stringent as assumed in the SLMCPR determination.

In reviewing the Reload Licensing DRF (J11-02346), the team noted that the scheduled DRF closure date had expired and an extension had not been obtained as required by EOP 42-10.00. During the review of the DRFs, the team noted that, for several volumes of DRFs J11-02346 and DRF

[11]The licensee performed certain sections of the core management analysis.

J11-02415, the table of contents was missing or the forms required by EOP 42-10.00 were not used. The team concluded that these omissions in the DRF documentation did not comply with EOP 42-10.00. As a result, the team identified a potential nonconformance during this part of the inspection.

In response to the team's determination that the documentation of DRF J11-02346 did not conform to EOP 42-10.00, NEP explained that the DRF had not yet been completed. In addition, NEP explained that these items would be documented as required by EOP 42-10.00, and that the responsible engineer would be further trained on the documentation requirements of EOP 42-10.00. As a result of the corrective actions taken by NEP, the team determined that its concern regarding compliance with EOP 42-10.00 had been satisfied, and the potential nonconformance was closed.

During the review of Volume 1 of DRF J11-02346, the team noted that the FRED document provided safety-related input to the reload licensing safety analysis. EOP 42-6.00, "Independent Design Verification," dated March 27, 1995, required that this input data be verified before use in the licensing analysis. However, certain FRED data provided by NEP is not covered by an established verification process. After expressing this concern to the NEP staff, the team was provided with a GENE internal Corrective Action Request (CAR) 95-3 (CAR No. 8), which identified the same concern. The team concluded that the lack of verification of the reload licensing FRED data was a weakness in the NEP verification process.

The team's review determined that selection of the limiting transients for a particular reload depends on the plant configuration, selected margin-reduction and operating flexibility options, and the reload core design. NEP has recognized the importance and complexity of this step in the reload analysis, and has established the Transient Selection Review Committee to review this selection process. The team considered the formation of the Transient Selection Review Committee a strength in the NEP approach to reload licensing. The team noted, however, that the reload process does not require that the Transient Selection Review Committee document the basis of its selection of limiting transients. The team also noted the recent loss of several highly experienced senior engineers (in the transfer from San Jose) and the commensurate loss in corporate knowledge. Consequently, the team concluded that it was a weakness not to require that the Transient Selection Review Committee document the basis of its findings in order to allow new staff members to receive training in the transient selection in the context of an actual licensing analysis.

(3) Limerick 1 Cycle 6

NEP documented the reload licensing for Limerick 1 Cycle 6 in DRF J11-02141. Bundle design, core management, and core monitoring were documented in additional DRFs. Split engineering responsibility exists for Limerick 1, and the division of responsibility between NEP and the licensee was reviewed by the team with the TPM and Fuel Project Manager. The team determined that in general, the licensee has responsibility for the bundle design, reload loading pattern, and core management, while NEP has responsibility for the reload licensing analysis and the review of licensee activities.

The team discussed the Limerick 1 Cycle 6 core design and plant operations with NEP to determine the changes from Cycle 5 and the potential reload licensing implications. In addition, the team inspected the reload licensing process for Limerick by reviewing the DRF relative to the requirements of TDP-0023, "Reload Licensing Analysis Procedure," Revision 0, G.N. Marrotte, dated April 1994, and the supporting TDPs. The team's review included the key documents and interfaces for the initial steps of the reload licensing process (WA, product & performance specification, FRED, and the OPL-3).

The team found that all steps in the reload licensing process were documented in the DRF. In addition, the analysis methods, rational, and results for various analyses were discussed with the TPMs. The team selected two items for indepth review and discussion. The first involved the cold shutdown margin and corresponding eigenvalue selection as discussed in Sections 3.4.2.2 and 3.4.2.1, respectively. The second item was the reload transient analysis for the turbine trip without bypass. The team reviewed this analysis relative to the requirements in TDP-0039, Automated Transient Analysis," Revision 0, D.C. Serell, dated April 1995, and TDP-0042, "Turbine Trip," Revision 0, F.T. Bolger, dated April 1995. The team found that the documentation and traceability were of a good quality. The team also reviewed the end product of the reload process, the SRLR, and found that the report was traceable to the DRF and consistent with the RLQR.

The team concluded that the overall quality of the Limerick 1 Cycle 6 DRFs was very good, and that a strength of the process was the technical discussions in the DRF by the TPM or the responsible engineer. The team also found that the Limerick 1 Cycle 6 reload team members were experienced engineers. (The team noted, however, that the reload effort was largely completed in GENE's facility in San Jose, California before the move to Wilmington, North Carolina.) The team also concluded that the high level of automation in the analysis process contributed to the overall quality.

(4) Pilgrim Cycle 11

NEP documented the reload licensing for Pilgrim Cycle 11 in DRF J11-02368. Bundle design and core management were documented in DRF J11-02335 and DRF J11-02528, respectively. The team's review of these DRFs determined that Pilgrim Reload 10 (Cycle 11) involved 136 GE11 bundles of a total core loading of 580 bundles. Cycle 10 included 140 GE10 bundles. Cycle 11 bundle average enrichment increased by 0.23 weight percent (w/o) U_{235} relative to Cycle 10. The review also noted that other than the potential significance of introducing GE11 bundles, there were no cycle-specific changes from Cycle 10 to Cycle 11 of significance to the reload licensing process. The reload licensing DRF contained the WA, EUP, FRED, and OPL-3 documents.

The team began its evaluation with an overview discussion with the Cycle 11 TPM and a process manager. This discussion included a review of the purpose and use of the EUP, FRED, and OPL-3, and the applicable EOPs and TDPs. The Pilgrim Cycle 11 design did not present unusual requirements. Except for cold shutdown margin measurements (discussed in Section 3.4.2.2(2)), at the time of this inspection, there had been no unusual events in the startup and operation of Cycle 11.

The team determined that WA, FRED, and OPL-3 documents were exchanged between NEP and the licensee with NEP signatures per interface requirements of the applicable TDP, and verification per the requirements of EOP 42-6.00. The team concluded that the documents were clear as to cycle-specific changes (such as improved scram time, use of a power coastdown, and a preliminary design change from 140 to 136 bundles to meet the EUP requirements). The FRED and OPL-3 documents also addressed the need for additional testing for Cycle 11 because of the introduction of GE11 fuel.

Volumes 3, 5, and 6 of DRF J11-02368 contained the RWE analysis, reload transient analysis, and GETAB mini review, respectively. The team examined these volumes and found that they complied with the relevant TDPs. The team primarily used this effort to investigate the GETAB mini-review, comparing the analysis process to the relevant TDPs and confirming that the Pilgrim Cycle 11 SRLR contained the ΔCPR data. From this investigation, the team concluded that the Pilgrim Cycle 11 reload transient analysis was satisfactorily completed per EOP 42-6.00 and the relevant TDPs. The team also found that the end product results in the SRLR were traceable to the DRF. Furthermore, although much of the analysis is automated, the DRF volumes contained discussion of the analysis results, comparisons to other results, and verification discussion. Both the responsible engineers and the verification engineers exhibited knowledge of the subject matter. This discussion of results, combined with computer code automation as demonstrated in the Pilgrim Cycle 11 reload licensing analysis, is considered a strength for the organization.

Based on the team's examination of the appropriate EOPs, TDPs, computer code documentation, topical reports, and selected DRFs, as well as discussions with the engineers involved, the team determined that, with the exception of the weaknesses noted, the reload core design, reload safety analysis, and reload licensing analysis activities were excellently performed.

3.4.5 Fuel Assembly Mechanical Design

To evaluate NEP's fuel assembly mechanical design process, the team conducted an indepth review of the design process with certain emphasis on fuel failures mechanisms. The following paragraphs summarize the team's findings.

The fuel assembly mechanical design process consisted of the design analysis and supporting mechanical, seismic, and flow-induced vibration (FIV) testing of the fuel assembly components, channels, and control blades. The process also includes preparation of the licensing documentation and following the design through the manufacturing process. Typically, the fuel assembly mechanical design is performed generically as new fuel designs are introduced, and the cycle-specific design analysis is minimal. NEP's generic analysis typically includes the following assessments:

- evaluation of fuel rod stress with the FURST code
- seismic spacer tests
- mechanical analysis for transient overpower conditions using GESTR-M
- stress analysis of the various assembly components (e.g., tie plates and water rods) using the ANSYS code
- FIV testing for the evaluation of fuel rod fretting
- a finite-element cladding collapse analysis.

As part of the evaluation of the fuel mechanical design analysis, the team selected and reviewed DRF J11-01652 (the GE12 lower tie plate), DRF J11-02363 (the evaluation of a GE10 lower tie plate manufacturing deviation), and DRF J11-02223 (the Hatch 1 Cycle 16 reload design). The team concluded that these DRFs included the proper WAs, analysis documentation, and verification, and conformed to the requirements of EOP 42-10.00. The team also reviewed several component analyses in these DRFs, and found them to be acceptable.

3.4.5.1 Rod Bowing

The NRC's acceptance of the Amendment 22 of GESTAR II rod bowing evaluation was based, in part, on the observation that the effect of fuel rod bowing for GENE BWR fuel was small, as well as the requirement that the rod bowing will be reported when the rod-to-rod gap closure is greater than 50%. In a letter from GENE to the NRC, dated January 30, 1995, "Fuel Rod Bow in Excess of 50% Gap Closure," GENE reported gap closure greater than 50% for two GE6B fuel assemblies.

In order to eliminate the requirement in Amendment 22 of GESTAR II to report future fuel rod bowing, GENE performed a set of full-scale fuel bundle ATLAS tests to determine the CPR effect of bowing the most limiting rod to contact with the adjacent rods. GENE documented the results of these tests in "GE11 Critical Power Test with Rod Bow to Contact," NEDE-31829-P, dated April 1990. In its letter to the NRC dated January 30, 1995, GENE submitted the results of the tests (NEDE-31829-P), which indicated no CPR margin reduction as a result of rod bowing, and requested elimination of the reporting requirement from Amendment 22 of GESTAR II.

During its review of the tests, the team noted that the ATLAS measurements differed from the tests originally proposed to the NRC for assessing rod bow (letter from GENE to NRC dated March 14, 1995, "Proposed Rod Bow Test") in the following respects:

- The case of 85% gap closure was not included; this prevented the measurement of a CPR penalty at intermediate gap closures.
- The bowed rod was between spacers 2 and 3, rather than between spacers 1 and 2, as proposed.

Consequently, the observed boiling transition occurred above the PLRs, and the team questioned the applicability of the measurements to elevations below the top of the PLRs. Therefore, the team determined that the correlation of the measurements between spacers 2 and 3 to the evaluation of rod bow below the top of the PLRs was an open item, and requested that NEP notify the NRC upon developing the correlation. (Open Item 95-01-03)

3.4.5.2 Fuel Failures

NEP categorizes fuel failures observed in GENE BWR fuel into the following five failure mechanisms (as discussed in the paragraphs that follow):

(1) crud-induced-localized corrosion (CILC)
(2) debris fretting
(3) undetected manufacturing defects
(4) pellet/cladding mechanical interaction (PCMI)
(5) unknown causes.

NEP explained that the unknown category is somewhat misleading, because these failures most likely result from one of the other four listed causes; however, they have not been examined in enough detail to determine which of the four caused the failure. NEP reported that for fuel manufactured since 1989, there have been 13 failures that have been dispositioned, with 7 resulting from debris fretting, 5 from undetected manufacturing defects, and 1 from unknown causes.

(1) **Crud-Induced-Localized Corrosion**

According to NEP, CILC failures have not been observed in the last 6 years. NEP believes that these failures have been eliminated with the change from copper condensers and the use of deep bed dimineralizer cleanup systems along with better coolant chemistry controls.

(2) **Debris Fretting**

Debris fretting results when foreign material becomes entrapped adjacent to the cladding, and subsequent FIV causes the foreign material to wear through the cladding. The foreign material is either inadvertently introduced by the licensee into the core during a maintenance outage, or it is introduced into the fuel assembly during the manufacturing process (in the form of metal turnings or other debris created during the manufacturing process). NEP has recently implemented several steps to eliminate the introduction of debris in the fuel assembly manufacturing process. These improvements are discussed in detail in Section 3.6.5.1 of this report.

(3) **Undetected Manufacturing Defects**

NEP reports that manufacturing defects usually involve end plug welding defects or defects in the cladding or end plugs. To eliminate these defects NEP has introduced various changes in their manufacturing process, including performing 100% inspections of the various fabricated components and steps when possible. These improvements are discussed in detail in Section 3.6.5.1 of this report.

(4) **Pellet/Cladding Mechanical Interaction**

PCMI failures result from significant changes in the local power level of a fuel rod, usually caused by control blade movements adjacent to the rod. The power changes result in fuel pellet expansion and the release of fission products leading to stress-corrosion assisted cracking of the cladding. According to NEP, PCMI failures were significantly reduced in the mid-1970s with the introduction of preconditioning interim operating management recommendations that reduced the magnitude of power changes during short time periods. In the late-1970s, NEP introduced the barrier cladding that eliminated PCMI failures until recently.

NEP examined a recent fuel failure, and found that it was caused by the combination of a chip missing in the fuel pellet surface, and a power increase resulting in PCMI failure in a barrier clad rod. As a result of this discovery, NEP tightened the acceptance requirements for fuel pellet chip sizes in their fuel manufacturing process to eliminate the possibility of similar failures in the future.

In addition, a licensee recently reported three fuel failures in barrier fuel. Of these, two occurred in rods with initial low-power operation for greater than two cycles of operation, and the failures occurred when control blades were withdrawn adjacent to these rods. (These are classic operational symptoms of PCMI failures.) NEP indicated that the failures may have resulted from existing cladding defects assisted by the PCMI mechanism, or there may be other reasons for these failures (rather than PCMI being the principle cause for failure). However, the cause is currently listed as unknown because these rods have not been examined.

As NEP noted, PCMI has nearly been eliminated as a primary failure cause, but it can result in extensive secondary damage to barrier clad rods and, to a lesser extent, to non-barrier clad rods once a primary rod failure occurs as a result of other causes. The extensive secondary damage results from cladding hydride assisted embrittlement caused by water ingression. If a significant local power increase occurs in this failed and embrittled rod, a long longitudinal crack usually results, extending from several inches to several feet, and often results in fuel loss and very high coolant activities. To prevent the extensive secondary damage, NEP introduced a solution that involves leaving a control blade in near the failed rod to keep its power level low until it can be discharged from the core. This has been shown to significantly reduce the likelihood of secondary failure damage.

On the basis of its review, the team concluded that NEP has an aggressive program to monitor and review fuel performance. However, the team noted that post-irradiation examinations of failed fuel were not routinely performed to determine actual root cause of the fuel's failure.

3.4.6 BWR Engineering Database and Engineering Computer Programs

To evaluate NEP's BWR Engineering Database (BWREDB) and engineering computer programs (ECPs), the team conducted an indepth review of the data collection processes, and the configuration, testing, and verification of the computer programs.

NEP uses the BWREDB as the QA-qualified input and output files for individual reload core design and safety analyses. In addition to the reload specifications, the BWREDB also includes generic design information for each fuel design, as well as design and operating conditions for each plant/reactor in which NEP designs are applied. The BWREDB was applied in the reload core design and safety analyses and in the manufacturing of the fuel bundle. The engineering computer programs (ECPs) use the input data and generate output data for the reload analyses that demonstrate that the specific reload meets safety and design operating limits. The following paragraphs summarize the team's findings.

3.4.6.1 BWR Engineering Database

The team evaluated the BWREDB system in two areas of NEP's operations (reload core design and safety analyses, and fuel production). The following paragraphs summarize the team's findings.

(1) **Reload Core Design and Safety Analyses**

The team determined that once the FRED and OPL-3 data were entered into the BWREDB, the reload core design process began. The process used the BWREDB (includes input and output data from previous cycles) and the ECPs to perform the analyses necessary to demonstrate that safety and design limits and EUP requirements met for each reload cycle.

Before data can be transferred to the BWREDB, the independent design verification and responsible manager who must indicate that all issues have been resolved and that the verifier is qualified to perform the verification. When the reload design analyses are completed and transferred to the BWREDB and the DRF is closed, a bundle design report and a fuel management summary report are created from the BWREDB. The licensee provides an authorization for release and NEP issues a WA to fabricate the fuel.

(2) **Fuel Fabrication**

The team determined that the manufacturing release activity began with advancing inputs to manufacturing, which include enriched uranium and zircaloy hardware requirements. The BWREDB system is then used to define the fuel design dimensions and material, including enrichment and gadolinia (Gd_2O_3) levels, in each fuel rod and bundle for fabrication. The BWREDB provides a paperless system that automatically prepares engineering instructions and bundle drawings, as well as digitized information used to drive the Automated Bundle Assembly Machine (ABAM). This BWREDB information is automatically transferred to the appropriate fabrication areas, where nearly all of the fabrication is machine automated (including bundle loading of the fuel rods) so that the data does not need to be input.

The team concluded that The BWREDB is a strength in the NEP fuel design and fabrication process, because it helps to reduce the probability of human error. The team also noted, however, that as with any automated system, human error can be introduced early in the process and will then be propagated through the design and fabrication process. Therefore, it is essential to ensure that the inputs to the BWREDB are adequately QA qualified and without error. After reviewing the BWREDB, the team determined that the data input into the BWREDB have adequate review procedures defined in DB-0002.

3.4.6.2 Engineering Computer Programs

The team's review of the ECPs determined that once the ECP development is completed, validated, and verified, a review team (level 2) performed an independent review to determine that specifications have been met. The level 2 review team also ensured that the code has been validated against proper data or other codes, and that the code has been properly tested, and gives appropriate calculational results. The ECP is not transferred to level 2 until the Level 2 review is completed and ECP problems resolved.

NEP explained that completion of the level 2 review does not mean the code is approved for production applications for a particular plant and fuel design. Approval is not provided until an application design review is completed to determine whether the following criteria have been met:

- Implementation will measurably improve the process.
- Implementation problems have been identified and resolved.
- The new ECP is adequately controlled to ensure that its implementation is consistent across NEP.

The team noted an exception to the level 2 requirements. These ECPs are designated as Level 2R. The NEP level 2R definition in "Engineering Computer Programs," EOP 40-3.00, Revision 17, dated March 27, 1995, explicitly states that restricted approved production programs are ECPs that do not satisfy all requirements for level 2, but may be applied to design tasks for a limited time with control component individual (CCI) approval. The team concluded that no other requirements are imposed for these ECPs. Therefore, the team questioned why EOP 40-3.00 did not include minimum requirements for level 2R ECP. The CCI advised the team that, according to EOP 40-3.00, any software components of the ECP that do not comply with level 2 requirements and that can affect the calculational results should be excluded in a level 2R ECP. (That is, an unresolved issue on the independent design verification of a software component of the ECP would exclude the use of this software component in level 2R applications of the ECP.) Even though the CCI's interpretation of the level 2R requirements in EOP 40-3.00 were satisfactory, the team noted that other interpretations of the EOP requirements could be significantly less restrictive. Therefore, the team concluded that the minimum requirements for level 2R designated ECPs were not adequately prescribed by EOP 40-3.00.

The team identified concerns described below that level 2 ECPs were used outside of the NRC-approved range of applications, and that NEP changed the ECPs without notifying the NRC of the changes. The two ECPs examined were the GESTRM versions (GESTRM06V, GESTRM07V and GESTRX01V) and the ODYN code versions (ODYNM09V, ODYNV09V and ODYNM10V), as discussed in the following paragraphs.

(1) **GESTRM**

The GESTRM ECP versions are codes used to provide best-estimate predictions of the thermal-mechanical performance of nuclear fuel rods. These codes set the maximum LHGRs versus burnup for different fuel designs. The team examined GESTRM ECP versions along with their stated range of applications, as well as the ECP software coding changes between versions. The NEP documentation for GESTRM gave the application range for Gd_2O_3 additions was **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o maximum. The **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o Gd_2O_3 limit allowed by GESTRM was contrary to the NRC agreement with GENE and the subsequent NRC approval that GENE would not exceed **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o Gd_2O_3 until NEP obtained confirmatory fission gas release data from four segmented rods with **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o Gd_2O_3-UO_2 (per letter MFN-193-83, dated October 18, 1983, from GENE to NRC). Once this confirmatory data demonstrated satisfactory results with their ECP predictions, NEP would use up to **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o Gd_2O_3.

This agreement was implemented because of NRC concerns about fission gas release and thermal differences between Gd_2O_3-UO_2 fuel and UO_2 fuel. When the team questioned NEP about the status of the confirmatory fission gas release data from **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o Gd_2O_3-UO_2 segmented rods, NEP stated that the irradiations are complete, but the rods have not been punctured yet to measure fission gas release. NEP further noted that Gd_2O_3 content in the fuel is controlled and limited to **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o for all of their current fuel designs, and not by the GESTRM ECP application. NEP's DB-0009.02, "Design Basis Document - Standard Pellet List (GE11)," Revision 0, dated May 1995, and DB-0010.02, "Design Basis Document - Standard Fuel Rod List (GE11)," Revision 2, dated May 1995, were reviewed by the team and found to limit Gd_2O_3 to **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** w/o for the GE11 designs. The team therefore considered this issue closed.

The team's review found that the application range for fuel burnup in GESTRM06V was **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** GWd/MTU (peak rod average), while in the GESTRM07V version it was increased to **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** GWd/MTU (peak rod average). NRC NUREG-1503, "Final Safety Analysis Report (FSAR) ABWR," Volume 1, documenting NRC's approval of the ABWR reactor design, specified a burnup limit of 60 GWd/MTU (peak rod average), and further stated that any extensions of this burnup limit would be submitted to the NRC for review and approval. Therefore, the team concluded that the application limit of **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** GWd/MTU in GESTRM07V exceeds the NRC staff burnup limit of 60 GWd/MTU for BWR fuel designs.

NEP stated that it disagreed with the NRC staff's position on the burnup limit of 60 GWd/MTU, and asserted that burnup extension does not constitute a safety issue. However, for the reasons stated in NUREG-1503, the NRC staff considers fuel burnup extensions a safety issue and, therefore, disagrees with NEP's position. The issue of a burnup extension beyond 60 GWd/MTU was not resolved during this inspection, and, therefore, the team requested that GENE respond to this issue as an open item from this inspection. (Open Item 95-01-04)

The team's review found that the GESTRX01V version was created for use by external users with only minor changes to the originally approved GESTRM06 version. The application range for fuel burnup for this version was limited by NRC to 60 GWd/MTU rod average.

(2) **ODYN**

The ODYN code versions simulate the dynamic behavior of BWRs, and are used by NEP to perform 1D pressurization transient analyses. The ODYNM09V version is approved by the NRC for plants with motor generator regulated flow pumps, and the ODYNV09V version is approved for plants with control valves on recirculation flow. The team determined that a new version ODYNM10V has recently been given Level 2 status, but had not received application design review approval by the NRC. The ODYNM10V version includes the modeling for motor generator plants from ODYNM09V, as well as the modeling for control valve plants from ODYNV09V. A third model had been added to simulate variable speed flow pump motors, and was being offered to licensees as an option for GENE BWRs. The team determined that the variable speed motor simulation required several lines of software coding additions to ODYN, and was needed for plants that intend to use the new variable speed motors in order to calculate the correct timing of the pump trip during pressurization transients.

The team asked NEP whether the ODYNM10V version will be submitted to the NRC for review because the team concluded that the simulation of the variable speed motor was not a trivial modeling task. The NEP ECP responsible engineer for ODYN versions stated that ODYNM10V was not to be sent to the NRC for review and approval because it did not result in a significant change to the calculational results for those plants using ODYNM09V and ODYNV09V. The team reviewed NEP's analysis results, and confirmed that the changes in the results were indeed small and considered this issue closed.

3.5 Fuel Designs

As part of its inspection, the team evaluated NEP's new fuel designs, GE12 and GE13 (as described in NEDE-32417-P and NEDE-32198-P, respectively) for compliance with Amendment 22 of GESTAR II (NEDE-24011-P-A). The design acceptance criteria described in Amendment 22 of GESTAR II were approved by the NRC staff in a safety evaluation report (SER) dated July 23, 1990. Amendment 22 of GESTAR II also established a set of fuel licensing acceptance criteria for evaluating new fuel designs, and established the critical power correlation bases and SLMCPR criteria for new designs. In addition, Amendment 22 of GESTAR II established the applicability of previous GENE generic analyses to new fuel designs.

The NRC staff previously evaluated, and generally approved, the GE11 fuel design, as documented in an SER dated March 25, 1992, with regard to the approved design acceptance criteria in Amendment 22 of GESTAR II. However, the NRC staff also identified deficiencies in NEP's application of the design acceptance criteria to the GE11 fuel design. The NRC staff discussed certain improvements for future fuel design submittals to NRC. The team's evaluation of the GE12 and GE13 fuel designs was therefore, in part, a followup of the previous evaluation of the GE11 fuel design. The following paragraphs summarize the team's findings.

The GE13 fuel design is essentially a variation of the GE11 fuel design, in which the GE13 has one more grid spacer (8 spacers in total) and longer part-length rods (PLRs) than the GE11 design. The variation improves critical power performance for GE13. The team considered that GE13 is basically a product evolved from GE11, and that there are no fundamental differences between these two fuel designs. The team's review of the GE13 fuel design for reload licensing applications resulted however in two open issues relating to MCPR and R-factor approved methodologies (as discussed later in this section).

Compliance of the GE12 fuel design with Amendment 22 of GESTAR II was therefore the main focus of this portion of the inspection. The team's evaluation also included a detailed review of the NEP evaluation for compliance with each of the Amendment 22 of GESTAR II design criteria addressed in NEDE-32417-P. The team also reviewed NEP's corrective actions taken to close out the deficiencies identified by the NRC SRXB during its audit of the GE11 fuel design (as reported in an SER dated March 25, 1992).

From its review, the team determined that the GE12 fuel has some design features that differ significantly from those of GE11/13. For example, the GE12 fuel design is a 10x10 array, while the GE11/13 fuel designs are in a 9x9 array. The GE12 fuel consists of 92 fuel rods, 2 central water rods, with 8 Inconel grid spacers encased in an interactive fuel channel. There are more PLRs in GE12 than in GE11/13. The fuel rods can have UO_2 rods or UO_2-Gd_2O_3 rods with a natural uranium blanket at the top and bottom ends.

The following topics are organized and numbered to correspond to NEP's documentation of the GE12 (NEDE-32417-P) and GE13 (NEDE-32198-P) evaluations for compliance to Amendment 22 of GESTAR II. The following paragraphs summarizes the team's findings:

3.5.1 Lead Use Assemblies, 2.1(2)

For GE12 fuel design, NEP has four demonstration lead use assemblies (LUAs) currently irradiated in Pennsylvania Power and Light Company, Susquehanna Steam Electric Station Unit 2 core. The team noted that NEP plans to have one or more domestic reactors to receive GE12 fuel for irradiation program. In addition, NEP reported loaded another 12 fuel assemblies of various GE12 design features in German and Swedish reactors in 1993 and 1994, respectively. NEP will use these irradiated fuel assemblies to verify the GE12 fuel design features. Based on these LUA programs and NEP's commitment to continue collecting corrosion data (Section 2.3 of NEDE-32417-P), the team concluded that NEP has satisfied the intent of the LUA requirement in the acceptance criteria.

3.5.2 Thermal-Mechanical, 2.2

(1) Stress, Strain, and Fatigue, 2.2(1)

The stress, strain, and fatigue criteria for NEP BWR designs have been previously reviewed and approved and applied to previous designs (NEDE-22148-P-A and NEDE-31917-P). NEP maintains that these same criteria remain applicable to the GE12 fuel design. Of particular concern in the

GE12 design is the high burnup levels of **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** Gwd/MTU (peak rod averages) intended for this design. This concern is due to the accelerated corrosion at high burnups (≈ 60 Gwd/MTU) observed in pressurized-water reactor (PWR) fuel designs. The high corrosion in PWR cladding has reduced the strain capability to below the 1% uniform strain limit specified in the NRC Standard Review Plan (SRP) Section 4.2, "Fuel System Design." NEP uses a total strain criterion of 1% that is more conservative than the SRP Section 4.2 strain criterion. Based on the 1% total strain criterion, NEP estimated the effective uniform strain for their high burnup cladding to be below the SRP strain criterion.

In addition, NEP presented both rod-average and maximum corrosion values up to a rod-average burnup of **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** Gwd/MTU, demonstrating that the maximum corrosion in NEP designs is less than those observed in PWR fuel designs. This level of corrosion was found to be acceptable for the strain and fatigue criteria used for the GE12 design.

NEP has performed cladding stress, strain, and strain fatigue analyses for the GE12 design using the GESTR-MECHANICAL code NEDE-24011-P-A-6, Amendment 10, and approved methods that include consideration of normal operation, AOOs, and uncertainties in operation and fabrication. These analyses demonstrate that the GE12 design meets the previously approved design criteria for those plant applications defined. Therefore, the team concluded that stress, strain, and strain fatigue are acceptable for the GE12 design up to a rod average burnup limit by NRC of 60 Gwd/MTU.

(2) Fretting, 2.2(2)

The GE12 fuel has a unit cell spacer design made of heat-treated Inconel alloy. All eight low-pressure drop Inconel spacers are identical. The GE12 fuel pressure drop is controlled by part length rods, low-pressure drop Inconel spacers, a low-pressure drop upper tie plate, and a high-pressure drop lower tie plate.

The team asked NEP whether the GE12 fuel design including new spacers will affect assembly rod vibration and fretting wear characteristics. NEP responded that the GE12 fuel was tested to ensure that the design features do not significantly increase FIV response, and thereby do not increase the potential for fretting wear. The GE12 fuel was tested for vibration response in a flow device against the P8x8R flow data. The results showed that the two vibration responses of GE12 and P8x8R fuel designs were very similar. Since the P8x8R fuel presented no significant vibration problem in the past experience, GENE concluded that the GE12 fuel also should have no vibration problem.

Another type of fretting wear is caused by foreign material (debris) entrapped and vibrating adjacent to the fuel rods. The failure is usually located near the vicinity of lower tie plate and lower end plugs. The debris fretting is characterized by a smooth abraded area with secondary hydriding away from the debris perforation. NEP has a new design lower tie plate with smaller flow holes that significantly reduce the size of passable debris.

Based on the similarity of the two vibration responses of GE12 and P8x8R fuel and the new design of the lower tie plate, the team concluded that the GE12 fuel design has minimized the tendency for fretting wear caused by flow-induced vibration or debris-induced fretting. Thus, the team concluded that the fretting wear design performance is acceptable and meets the acceptance criteria.

(3) Metal Thinning, 2.2(3)

NEP does not have a cladding corrosion or crud limit on the cladding design, other than the criterion that the effects of cladding corrosion and crud buildup are to be included in its thermal and mechanical analyses. In the past, the BWR fuel rods have been troubled by a particular type of nodular corrosion called crud-induced localized corrosion (CILC). The CILC failure mechanism was attributed to environmental condition, operational history, and material susceptibility. The CILC failures were usually limited to plants with copper alloy condenser tubes and filter demineralizer condensate cleanup systems.

NEP and BWR owners have taken mitigating actions, including replacement of copper bearing condensers with titanium or stainless steel, water chemistry control, and improved tubing fabrication and testing methods. With these improvements, GENE has effectively reduced the fuel failure rate associated with CILC.

NEP has various LUA programs to continue collecting corrosion data for high burnup fuel. In general, visual inspection of the LUAs revealed excellent corrosion performance along the full length of the fuel rods. NEP will continue to irradiate LUAs for the purpose of extended burnup regime to confirm acceptable corrosion performance.

Based on the improvement of plant operation and cladding fabrication to eliminate the nodular corrosion, and based on NEP's commitment to collect corrosion data during high burnup, the team concluded that the GE12 fuel design has adequately analyzed the corrosion performance, and thus meets the acceptance criteria.

(4) Fuel Rod Internal Hydrogen Content, 2.2(4)

Fuel rod internal hydrogen content is controlled during the manufacture of the fuel rod. To limit the maximum amount of hydrogen, NEP has specified standards (C776-83 and C934-85) defined by the American Society for Testing and Materials (ASTM). The manufacturing process for the GE12 design includes steps to remove hydrogen and verify that these ASTM standards are met. The team concluded that the design criterion to prevent internal cladding hydriding has been met.

(5) Fuel Rod/Channel Bow, 2.2(5)

Fuel rod and channel bow are included in the T/H analyses to prevent boiling transition, as discussed in Section on 3.5.3 of this report. The decrease in rod diameter of the GE12 design compared to earlier designs has the potential to increase the effects of rod bow. However, to help offset the effect of a decrease in rod diameter, NEP increased the number of spacers in the GE12 assembly as compared to the GE11 design, and decreased the spacer spacing where rod bow is calculated to occur. The team concluded, therefore, rod bow is not expected to differ significantly for the GE12 design. The effects of rod and channel bow on MCPR will not be discussed in the thermal-mechanical section, but are discussed in Section 3.5.3 of this report.

(6) Cladding Pressure Loading, 2.2(6)

NEP has performed rod pressure analysis for the GE12 design using conservative upper bound rod powers and analysis methods described in NEDE-22148-P-A and NEDE-24011-p-A-6, Amendment 10. The fuel swelling and cladding creep equations remain the same as those evaluated for the GE11 design NEDE-31917. These equations are used to determine the maximum critical pressure at which the cladding creep out rate will not exceed the fuel swelling rate (NEP criterion for rod pressure). The team examined the GE12 rod pressure analyses, and verified that the fuel and Gd_2O_3 rod pressures remain below the critical pressure threshold. The team also confirmed that the rod pressure criterion has been met for the GE12 design.

(7) Control Rod Insertion, 2.2(7)

Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," to SRP Section 4.2 described the requirements of control rod insertion during the combined seismic and LOCA loadings. For BWRs, the fuel assembly (including fuel rods and grid spacers) must maintain its structural integrity and resist fuel liftoff from the core plate during the combined seismic and LOCA loading. The liftoff of fuel assemblies would interfere with control blade insertion because of the fuel assembly lateral movement. To ensure GE12 structural integrity, NEP performed a fuel liftoff calculation based on the approved methodology described in NEDE-21175-3-P-A. A referenced plant was chosen for this calculation based on the previous staff evaluation of the GE11 fuel design, which recommended that NEP select a referenced plant with significant liftoff. The result showed that the GE12 fuel liftoff in the referenced plant is substantially below the fuel liftoff limit. Therefore, GE12 fuel assemblies remain seated in the core plate during the combined seismic and LOCA loading for the referenced plant.

Based on the approved methodology and the acceptable referenced plant, the team concluded that NEP has demonstrated that the GE12 fuel conforms to the acceptance criteria of no fuel liftoff during the combined seismic and LOCA loading.

(8) **Cladding Collapse, 2.2(8)**

The team questioned NEP with regard to whether their analytical models or methods have changed. NEP indicated that the creep model remains the same as that defined in 1985 for GESTR-MECHANICAL, and the methods are the same as those approved in NEDE-20606-P-A.

The NEP cladding collapse analysis demonstrated that cladding collapse will not occur in the GE12 design at the maximum in-reactor times for a rod average burnup of 60 GWd/MTU, as required by the GENE design criterion. Therefore, the team concluded that the GE12 design is acceptable with regard to cladding collapse.

(9) **Fuel Melting, 2.2(9)**

The NEP criterion is to prevent fuel melting for normal operation and AOOs. NEP has performed fuel melting analyses for the GE12 design using NRC-approved analysis methods and models that demonstrate that fuel center melting will not occur as a result of normal operation or AOOs. The team concluded that the GE12 fuel design is acceptable with regard to fuel melting.

3.5.3 Safety Limit MCPR, 2.6

The SLMCPR is influenced by the critical power correlation and by bundle design parameters which affect the bundle R-factor distribution and the core radial power distribution. These parameters include the spacer design, assembly dimensional geometry, enrichment level and distribution, and fuel discharge exposure. The SLMCPR provides the margin required to account for uncertainties in the core monitoring system and the GEXL critical power correlation. The determination of the SLMCPR for the GE12 and GE13 fuel designs employed the standard limiting statepoint assumptions and analysis methods. However, because of the design improvements made in the GE12 and GE13 bundle designs, a series of ATLAS tests was performed to determine design-specific GEXL correlations for both the GE12 and GE13 fuel design.

The critical power analyses for the GE11, GE12, and GE13 fuel designs were performed with the GEXL-07, GEXL-10 and GEXL-09 CPR correlations, respectively. The GEXL-07 and GEXL-09 correlations were defined over a range of inlet subcoolings from zero to 70 British thermal units per pound (Btu/lb). For the GEXL-10 correlation, the upper limit of the subcooling range has been extended from 70 Btu/lb to 100 Btu/lb. However, the GEXL-10 critical power database does not include data above 60 Btu/lb subcooling. This prevents the determination of the correlation statistics (i.e., the mean and standard deviation) in the extended range for use in the SLMCPR calculation. The team identified this concern to NEP and NEP responded that additional data (which is presently archived) can be used to resolve this concern. The team noted that this issue should be resolved during the NRC's GE12 and GE13 fuel design reviews. Therefore, to resolve this concern relative to the GE12 and GE13 design reviews, the team requested that NEP provide the data used to extend the upper limit subcooling range from 70 Btu/lb to 100 Btu/lb, and considered this matter an open item.

NEP responded to the team's concerns in its letter JFK-95-073, "GEXL10 Subcooling Data Extrapolation," dated October 17, 1995. The data submitted by GENE in response to the team's concern is currently under evaluation by SRXB and will be addressed separately by that branch.

In order to properly account for the axial dependence of the fuel rod power in the case of PLRs, NEP changed the definition of the R-factor. The R-factor change applies to the GE11, GE12, and GE13 PLRs fuel designs, as well as the corresponding GEXL07, GEXL10, and GEXL09 correlations. For previous (full-length rod) fuel designs the R-factor is defined using an axial integral of the "rod power." For the GEXL07, GEXL09, and GEXL10 correlations, the R-factor is defined using an axial integral of the "local R-factor." Although Amendment 22 to GESTAR II requires that changes in R-factor definition be reported to the NRC, this change was not reported. The team concluded that this indicated a weakness in NEP's adherence to the Amendment 22 to GESTAR-II reporting requirements.

NEP responded to the team's concerns in its letters JFK-95-092, "GEXL09 Auditing Information," dated October 20, 1995; JFK-95-093, "R-Factor Calculation Method," dated October 20, 1995; and JFK-95-113, "GE13 MCPR Safety Limit," dated November 30, 1995. The data submitted by GENE in response to the team's concern is currently under evaluation by SRXB and will be addressed separately by that branch.

3.5.4 Stability Licensing Acceptance Criteria, 2.9

(1) Comparison With Previously Approved Designs, 2.9(1)

During its audit of the GE11 fuel design[12], the team noted that the stability analyses procedures used by GE had been demonstrated to be non-conservative, and should be replaced by the BWROG stability analysis procedures that were then under review. NRC approval of the BWROG procedures was subsequently documented in the SER[13] dated July 12, 1993. The team was concerned with the acceptability of the GE12 stability design because the smaller diameter fuel rods tend to reduce the margin to instability, and require compensating design measures to maintain acceptable stability decay ratio.

In its stability compliance evaluation, NEP compared the GE12 decay ratio to values for the previously approved R8x8R design. The analyses described to the team were performed for an appropriate range of operating conditions, and included both the regional and core-wide modes of instability. The results indicated that the GE12 fuel provides additional margin to the onset of regional oscillations. However, the GE12 fuel shows a slightly greater propensity for core-wide instability than cores operating with the previously approved P8x8R fuel design.

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[12]documented in SER dated March 25, 1992

[13]NEDO-31960 and Supplement 1, dated July 12, 1993

The team concluded, therefore, that the GE12 fuel design does not comply with the first stability design acceptance criterion (i.e., 2.9(1), "Comparison With Previously Approved Designs") of Amendment 22 of GESTAR II.

(2) Exclusion Zone Evaluation, 2.9(2)

The second stability acceptance criterion of Amendment 22 of GESTAR II permits a new fuel design to be more limiting in core and channel decay ratios if it can be demonstrated that there is no change to the calculated exclusion zone to ensure stable operating conditions. GENE performed sensitivity studies to compare the calculated boundaries for stable behavior with the comparative fuel designs. The comparison indicated mixed results, with the GE12 stability boundary producing a slightly more restrictive operating condition for some regions of the power/flow map and a less restrictive condition for others. GENE concluded that the differences between the two designs were very small and produce negligible differences (1% - 2% power) in the exclusion zone. The team also found the differences too small to be significant, and that the GE12 stability design appears to be acceptable with regard to this criterion.

The team determined, however, that the design does not strictly satisfy the condition of "no change" to the stability boundary expressed by this criterion. Therefore, in order to avoid a non-conservative creep in the exclusion boundary for successive fuel designs in accordance with Amendment 22 to GESTAR II, the team concluded that the GE12 fuel design not be used as a reference design for future comparisons with Amendment 22 of GESTAR II.

3.5.5 Refueling Accident, 2.13

The NEP criterion is that the radiological consequences of this accident must either bound previous analyses or perform a new analysis that demonstrate that the consequences are within the country-specific limits on radiological dose. In a simple analysis in NEDE-32417P, NEP demonstrated that the radiological consequences of a GE12 bundle refueling accident are less than those for the previous 7x7 and 8x8 array fuel designs for plants equipped with a standard triangle refueling mast.

However, for the application of the GE12 design to plants with the heavier NF500 cylindrical mast, the NEP analyses in NEDE-32417-P predict higher radiological consequences than the previous result for the 8x8 array fuel design. This is because there is an increase in the number of failed rods ... GE12 fuel as compared to the 8X8 fuel design. Therefore, plants with the NF500 mast and an FSAR refueling accident analysis based on the 8x8 fuel design will require a re-analysis of this accident for GE12 fuel application. On the other hand, the application of GE12 fuel design to plants with the NF500 mast predicts less radiological consequences than the previous result for the 7X7 fuel design. Thus, plants with the NF500 mast and an FSAR analysis based on the 7X7 fuel design do not need a re-analysis of this accident for GE12 fuel application.

The team concluded that the radiological consequences of the refueling accident of the GE12 application to plants with a triangular refueling mast are acceptable; however, a re-analysis will be required for plants with both the NF500 mast and an FSAR result based on the 8x8 fuel design for GE12 fuel application.

3.6 Nuclear Fuel Fabrication

To evaluate NEP's nuclear fuel and core component fabrication activities, the team evaluated NEP's activities in the fuel manufacturing operations (FMO) building, the fuel component operations (FCO) building, and the service components operations (SCO) building. This part of the team's inspection of NEP emphasized the manufacturing processes that relate to fuel rod failure mechanisms (e.g., hydriding, fretting, pellet/cladding mechanical interaction (PCMI), overheating, cladding collapse, bursting, and mechanical fracturing). The following paragraphs summarize the team's findings.

3.6.1 Procurement

The team found that NEP distinguished the procurement of materials and products delivered to their customers from materials and products used in their own manufacturing operations. Procurement of materials and products eventually shipped to their customers was referred to as "direct material sourcing," while procurement of materials and products used in the manufacturing operations was referred to as "indirect material sourcing."

Vendor approval activities were governed by "Vendor Approval and Survey," P&P 60-11, Revision 14, dated April 1, 1994, that described the methods for conducting vendor surveys and approving vendors for various types of procurement (e.g., the American Society of Mechanical Engineers (ASME) Code materials, safety-related materials and services, and radioactive material shipping containers). All production materials and products were procured from vendors that were on the approved vendors list (AVL).

Production procurement activity was initiated by generating a purchase requisition. Before placing of the purchase order (PO), the requisition had to be reviewed and approved by the purchased material quality control (PMQC) and purchasing organizations. The team reviewed at least one PO for each of the following product forms:

- barrier tube-reduced extrusion (TREX)
- nonbarrier TREX
- endplug bar
- channel strip
- upper and lower tie plates

The team's review of POs for the fuel channel strip procurement will be discussed here. The team noted that three suppliers had provided zirconium (Zr) alloy (zircaloy) channel strip to NEP in accordance with GENE specification 23A7239, "Thermal Size Anneal Channel," Revision 1, and NEP's QC plans for zirconium and zircaloy. In accordance with the specification, channel strip was produced from double-width material that was slit into two pieces and shipped to NEP as a matched pair. Each channel was manufactured with matched pairs of strip and identified with a heat, anneal, and strip number when received from the supplier.

A fuel channel is fit over the fuel bundle to direct in-reactor coolant flow. Made of zircaloy, each channel is approximately 14 feet in length. Newer fuel designs use a fuel channel that is interactive with the bundle itself, and is intended to enhance critical power performance. NEP classified the fuel channel and channel shell as a safety-related basic component (as described in Section 3.6.2 of this report).

NEP's requirements and the POs reviewed during this review required the channel strip suppliers to maintain a quality control system and provide or have available lot samples in a manner similar, although not in accordance with, the quality requirements of Appendix B to 10 CFR Part 50. In addition, NEP also imposed its own standards and requirements. One of these suppliers, Western Zirconium, supplied fuel channel strips to NEP specifications in accordance with QC plans A-208, "Zirconium and Zircaloy," as well as A-232, "General Documentation Requirements for Material, Services or Products." From its review of NEP's POs to Western Zirconium, the team determined that NEP did not procure the channel strip as a basic component (as defined in 10 CFR Part 21). The team concluded that NEP's procurement of the channel strip did not meet the definition in 10 CFR Part 21.3 for a commercial grade item because, in part, it was subject to NEP design and specification requirements that are unique to NRC licensees.

With the exception of the fuel channel fastener (discussed in Section 3.6.2 of this report), the team determined that NEP's procurement activities reviewed during this inspection did not impose the reporting requirements of 10 CFR Part 21 on its suppliers of materials, items, or services that NEP supplied to the nuclear industry as safety-related basic components. During the team's discussions with NEP regarding its procurement practices, NEP stated that it did not dedicate (as defined in 10 CFR Part 21) the materials, items or services. Consequently, the team determined that NEP's procurement practices for basic components may not comply with the requirements of 10 CFR Part 21. NEP's procurement practices are further discussed in Section 3.6.2.

The team advised NEP that the requirements of 10 CFR Part 21 were being revised (affective October 19, 1995) and that it was expected that NEP will review the information for applicability to their procurement practices (specifically, the expanded definition for commercial grade items) and consider corrective actions, as appropriate.

On the basis of the issues identified above, the team raised concerns regarding NEP's safety classification of the fuel assembly and its component parts, as discussed below.

3.6.2 Safety Classification

The team requested GENE to explain its safety classification of the fuel assembly and its component parts. In response to the team's concerns, NEP presented GENE's Position Statement entitled, "Safety Classification of GE Fuel Assembly and Related Components," dated November 17, 1994. The position statement noted that, for the purpose of this position statement, the term "basic component" as defined in 10 CFR Part 21.3(a)(I) is equivalent to the term "safety-related" used by NEP for the purpose of functional classification. The GENE procedure for the functional classification of structures, systems, components or parts thereof (as used in 10 CFR Part 21) was given in EOP 65-2.10, "Safety-Related Classification," Revision 8, dated March 27, 1995.

Paragraph 2.6, "Classification Determination," of EOP 65-2.10 stated, in part, that the safety-related classification starts with the design-basis events[14] (DBEs) and the functions required to prevent or mitigate these events, and extends to the system, component, and eventually to the part level. The paragraph continued by stating that the determination of a specific classification to be assigned to a system, component, or part was divided into two categories, functional and procurement. Additionally, the paragraph specified that the procurement category was then established either as safety-related or commercial grade, and concluded by adding that a commercial grade item was subject to dedication before it was used as a safety-related item (in accordance with EOP 65-2.20, "Dedication of Commercial-Grade Items," Revision 5, dated July 27, 1994).

On the basis of its review of paragraph 2.6 and the POs reviewed and described above, the team concluded that NEP's procurement and safety classification practices for certain "direct material sourcing" did not comply with the requirements of paragraph 2.6 of EOP 65-2.10 because NEP did not procure these materials as either safety-related basic components or commercial grade items. For the same reasons stated in Section 3.6.1 of this report, the team identified this weakness in NEP's procedural compliance with paragraph 2.6 of EOP 65-2.10 as a non-cited nonconformance.

According to GENE's Position Statement dated November 17, 1994, a fuel assembly consists of only three safety-related components and gives the basis for that classification, as follows:

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[14]Design basis events are defined in EOP 65-2.10 as conditions of normal operation, including Anticipated Operational Occurrences (AOOs), design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure three functions described in 10 CFR Part 21.3(a)(I), i.e., (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to those referred to in 10 CFR Part 100.11.

(1) Fuel Channel - The fuel channel and channel shell provide a guiding surface for control rod insertion and direct coolant flow within the fuel bundle during DBEs. GENE concluded that the fuel channel and channel shell help to ensure functions 2 and 3 described in footnote 14 and are therefore both safety-related.

(2) Fuel Channel Fastener - The fuel channel fastener, through the channel fastener bolt, connects the fuel channel to the fuel bundle, thereby ensuring sufficient space for proper insertion of the control rods and proper flow paths for coolant flow to the fuel bundle during and following DBEs. Therefore, GENE concluded that the fuel channel fastener and channel fastener bolt support functions 2 and 3 described in footnote 14 and are therefore both safety-related.

Note that the channel fastener bolt screws into a post that is one of four integral-cast posts of the upper tie plate which is not considered by NEP to be safety-related.

(3) Fuel Bundle - The fuel bundle configuration provides reactivity characteristics (i.e., Gd_2O_3 neutron absorber and enriched UO_2 distribution) which ensure the capability to shutdown the reactor and maintain it in a safe shutdown condition during and following DBEs. The fuel bundle must also retain it general mechanical and geometric integrities to maintain a coolable geometry during and following DBEs. GENE concluded that the fuel bundle, therefore, helps to ensure functions 2 and 3 described in footnote 14 and is therefore safety-related.

However, the Position Statement adds that GENE does not classify as safety-related fuel clad tubing, end plugs, upper tie plates, lower tie plates, spacers, or miscellaneous hardware typified by finger springs, coil springs, screws, nuts, lock tab washers, etc., used in fuel bundles (i.e., none of the fuel bundle's constituent parts are safety-related).

The team therefore determined that GENE claimed that no bundle part contributed to the fuel bundle's ability to perform its safety function as described in (3) above. However, paragraph 2.6.1(e) of EOP 65-2.10 further supports a functional evaluation of each part as follows:

> *"It will often be the case that a safety-related component contains items which are both safety-related and nonsafety-related. This is strictly due to the function of the items within the component. If the item provides support for or is integral to the performance of the safety-related function of the component, or its failure can prevent the component's satisfactory performance of the safety-related function, the item must be considered safety-related. If not, the item is nonsafety-related."*

However, contrary to the functional evaluation of each part prescribed in paragraph 2.6.1 (e) of EOP 65-2.10, GENE classified all of the fuel bundle's constituent parts as nonsafety-related on the basis that these component parts, in and of themselves, could not contain defects which would credibly create a substantial safety hazard as set forth in 10 CFR 21.3. The team determined that this position was not based on a functional evaluation of the parts.

The team held several discussions with NEP staff about the issues relating to its nonsafety-related classification of the constituent parts of the fuel bundle. The team then concluded that NEP had not established, during the course of this inspection, an acceptable basis for its position that none of the fuel bundle's parts support or are integral to the performance of the safety-related function of the fuel bundle. NEP asserted that, on the basis of the safety-related function of the channeled fuel bundle in the unique BWR environment, an acceptable basis for its safety classification position could be developed. Therefore, the team agreed to provide GENE the opportunity to submit such an evaluation for the NRC staff's review. The team therefore considered this matter an open item and requested that NEP develop a functional evaluation for the fuel bundle parts, as prescribed in paragraph 2.6.1(e) of EOP 65-2.10. (Open Item 95-01-05)

3.6.3 Chemical and Ceramic Operations

The team reviewed the chemical and ceramic operations from the receipt of uranium hexafluoride (UF_6) through the production of sintered and ground pellets. The recycle of scrap material was included in this review. All processes are housed in the FMO building. The team inspected all operations as they were performed. Items checked included the presence of written procedures at work stations, the calibration of instruments and gages, sampling points, analytical equipment, and analytical procedures. Interfaces with QC and QA oversight were reviewed. The following paragraphs summarize the team's findings.

3.6.3.1 Chemical Conversion

The UF_6 is converted to ammonium diuranate (ADU) by a conventional wet process. The UF_6 cylinders are heated to vaporize the UF_6, where it reacts with water forming a uranyl fluoride solution. This solution, together with ammonium hydroxide solution, form a precipitate of ADU. The ADU in paste form is heated to where it decomposes thermally releasing NH_3 and then is reduced to UO_2. The UO_2 is loaded into cans which have a bar-coded traveler card. The cans weight is automatically entered into the Material Inventory Computer System (MICS). The operator also enters the weight of the can on the traveler card and also into the In-process Material Release System (IMRS) computer. The powder is transported to the hammermill in the mill-slug-granulate (MSG) area.

There are four lines for the MSG production of UO_2 powder. When the cans of UO_2 powder enter the MSG area, the transaction is entered into the MICS. The cans of UO_2 powder are dumped into a milling machine and the resulting milled powder is discharged to a slugging machine. The slugging machine produces a pressed compact which is pushed into a granulating machine. The granulated product is collected in cans which are weighed and receive new bar-coded traveler cards. Data on weights and composition are entered into the IMRS. The identity and weight of the cans are entered into the MICS system and the cans are placed in storage for later blending operations.

Samples of the granulated powder are sent to the Chemet Lab for analyses (i.e., isotopics, metal impurities, carbon, chloride, fluoride, nitrogen, moisture, and O/U ratio). Bar-coded sample labels received with the traveler cards are attached to the sample containers. The results of analysis are entered into the Laboratory Material Control System (LMCS).

The team concluded that the process controls and sampling process were well defined and controlled and ensured expected-quality UF_6 powder.

3.6.3.2 Uranium Recycle

A fifth line of powder production originates in the uranium recycle unit where scrap material was divided into two categories, dirty and clean scrap. Dirty scrap contained gadolinia (Gd_2O_3) and/or scrap which did not meet the specifications for other impurities. Clean scrap contained less than 1 part per million (ppm) of Gd_2O_3 and meets specification for other impurities. The resulting powders are MSG processed, sampled and weighed, and then are available for the blending operation.

The team found the MSG powder was identifiable through the entries in the three computer systems (MICS, IMRS, and LMCS). Written procedures were found at workstations. Calibration records were checked for several of the instruments and were performed on schedule. The team concluded that the analytical equipment and analytical procedures for the MSG powders were satisfactory.

3.6.3.3 Powder Blending

The team found that powder blending following, the MSG operation, was performed in both the UO_2 shop and the Gd shop to produce press feed powder having uniform physical and chemical properties.

(1) UO_2 Blending

The team's review determined that the blend may be all of a single enrichment that was blended solely to provide uniform pressability characteristics and chemical properties; or a blend may include two or more materials of different enrichments. However, the maximum difference between enrichments can be no greater than 1.4% U_{235}.

The team observed the production control operator check the IMRS for the results of the MSG powder analysis. If IMRS releases the powder, it can be used for blending. The operator specified the identity of cans to be used for the blend. Blended powder was placed in cans that were weighed and given new card travelers. The identity and weight of the cans and the enrichment were entered into the MICS system and then the cans were placed in storage awaiting pellet pressing operations.

The team found that blend samples were pressed in a sample press. The green density was measured and the result entered in the IMRS. The samples were sent to the production area where they were sintered and ground in a production grinder. The sintered and ground pellets were examined for defects and sent to the laboratory for analysis. The results of analysis were entered into the Chemet Lab LMCS. If the results were satisfactory the powder was released by the IMRS to pellet press.

The team observed that if any sample analysis results were out of specification, the IMRS system printout instructed the shift production advisor regarding resampling, as described in the quality control inspection instructions (QCII). The IMRS system automatically released the blend if resampling results met specifications. The team concluded that the blending operations observed were satifactory.

(2) Gd_2O_3 Blending

The team noted that there was a separate line for fuel rods which contain Gd_2O_3. Pre-blended UO_2 powder was blended with Gd_2O_3 powder. UO_2 entered the Gd shop in containers specified for either low enriched ($\leq$4.013% U_{235}) or high enriched (>4.013% U_{235}) powder. A process parameter sheet identified for the operator the batches of blended UO_2 powder to be used, their weights, the w/o Gd_2O_3 desired in the completed blend, the pellet press pressure, the green density, and the sintering temperature for the batch.

The team observed one sample taken at random from one can of blended powder for fluoride analysis, and one from each of five randomly selected cans for Gd determination. The vibromill operator performed a Gd analysis on the samples. This is a process control measurement. Final assessment of Gd content was performed on finished Gd-containing fuel rods using the magnetic and passive scanners (MAPS) system. Completed blends were placed in controlled storage until released for pressing by IMRS. The team concluded that the blending operations observed were satifactory.

3.6.3.4 Pelleting

The team observed that the IMRS system released the material for pellet pressing. The blend operator recorded the transfer of product on the MICS system and dumped the powder into the hopper for the pellet press.

A sinter test was performed on a sample taken from every 20th can of calciner powder, and every 20th can of MSG powder. A sinter test was also made on a sample from each powder blend. Sinter test pellets were evaluated based on density, cracks, surface pits, etc. Results were factored into the selection of specific material used in each powder blend.

(1) UO_2 **Pelleting**

The team observed the UO_2 pelleting on three rotary presses. The press operator monitored the weight and green density and made adjustments as necessary to maintain the green density within specifications. The pellet punch had a three digit code on the dish-forming surface which formed identifying marks on each pellet for enrichment and project identity.

After a pellet was ejected from the die, it was guided into a molybdenum furnace boat and then transferred to the sintering furnace area. A traveler card was generated for each boat at the pressing station that followed it through sintering, storage, and release for grinding by the IMRS.

The team noted that the GENE fuel designs required several different enrichments. At the time of this inspection, 13 different enrichments were being processed at the same time. As there were only three rotary presses, to change enrichments it is necessary to frequently clean out the pressing system. The team determined that the frequent enrichment changes in the pressing operations introduced the possibility of cross-contamination of enrichments due to hold-up and release of pockets of powder in the equipment. This condition can not easily be detected during processing, but it did show up during active gamma scanning in the "Fat Albert" scanner. The phenomenon was referred to as "fuel spiking" (as discussed in Section 3.6.6.2 of this report).

(2) Gd_2O_3-UO_2 **Pelleting**

The team noted that the press feed for Gd_2O_3-UO_2 pellet pressing has poor-flowability unsuitable for rotary press feed, therefore, special presses were used in the Gd shop for pelleting. The press tonnage used for pressing was taken from the process parameter sheet which was based on sinter test results for the UO_2 blend used in Gd_2O_3-UO_2 blending. Pellets were marked with an identifying 3-digit code as described for UO_2 pellets. The team concluded that powder was supplied to the process as described previously for UO_2 pellet pressing.

3.6.3.5 Sintering

The sintering furnace contained six independently controlled heating zones. The team determined that a uniform temperature exists over almost all of the high heat zones. The operator sampled five pellets from each boat for density determination, using gamma source densitometer and associated laser micrometer. Verification standards were run following calibration, every four hours during densitometer operation, and immediately before and after any sintering test runs. The results give the average density of 5 pellets, the high density, the low density, and the deviation. The results were entered into the MICS system. The team observed a measurement of 10 samples; all samples were within specification.

Boats of pellets meeting density specifications were sampled for enrichment, hydrogen, densification, metal impurities, various chemical elements, total rare earths, total impurities, boron equivalent, and one archive sample is taken per every 15 blends. Pellets meeting specifications for the foregoing analyses were released by IMRS for grinding.

From its review, the team determined that Gd_2O_3-UO_2 pellet sintering was similar to UO_2 sintering, with few exceptions. Pellets were approximately the same as for UO_2 pellets with some minor exceptions. As with UO_2 pellets, Gd_2O_3-UO_2 pellets were released for grinding by the IMRS based on the analysis results. The team concluded that the sintering operations observed during this inspection were satisfactory.

3.6.3.6 Grinding

Pellets were dry-ground using a centerless grinder. There were two operators at each grinding station. One operator monitored operation of the grinder while the other operator transferred the ground pellets to trays.

The team noted that the pellet grinding operation was a "quality at the source" (QATS) process. The operators also performed the first QC inspection. Grinder operators received formal and on-the-job training. The training included operation of the machine, use of micrometers and other measuring devices and visual inspection of pellets.

The team observed that as each pellet emerges from the grinder, a laser micrometer measured the diameter of each pellet several times. The inspecting operator visually evaluated the pellets and rejected pellets with excessive pits, chips, or cracks into a scrap container. A set of visual standards was located at the inspection station for this purpose.

The team determined that, although every pellet was measured by the laser mike, there was no provision for automatically scrapping pellets that were out of diameter specifications. The QATS operator checked the diameter of one pellet from every other tray row using a hand held micrometer. The team observed that while checking diameter, the operator also checked for out-of-roundness. If a reject occurred, several adjacent pellets were checked to determine if there was a trend.

The results of operator inspections were entered into the IMRS. The team found that the inspector put a hold on a tray (Blue Tag), this action was entered via a MICS transaction. The team reviewed the records for a day shift grinder and observed that the tray was satisfactory for operator inspection and over-inspection.

The team determined that only pellet inspection results based on ground pellets used by the IMRS for releasing pellets for rod loading were surface finish and open porosity. Surface finish was measured on one pellet per tray. Open porosity is determined on one pellet per five UO_2 powder blends and one pellet per each Gd_2O_3-UO_2 blend. All other pellet release criteria were based on analyses of unground pellets.

Because pellets are ground dry, they have visually observable grinding dust on their surfaces, and being ground, they have significantly greater surface area. The team determined that both factors may contribute to a greater amount of absorbed moisture than was present on unground pellets. The team concluded that NEP's practice of using the hydrogen analysis from unground pellets as a basis for estimating the hydrogen content of the fuel was not as conservative a basis for hydrogen content as using the hydrogen analysis from pellets after grinding.

3.6.4 Fuel Clad Tubing

Fuel clad tubing was manufactured by NEP either with and without a pure Zr barrier clad on the inside surface of the tubing. The team determined that in either case, the GENE designs incorporated a heat treated outer surface for corrosion resistance. The zircaloy tubing was initially received in the form of tube-reduced extrusions (TREX). After shop release, the TREXs were prepared for reduction (reduction of both diameter and thickness dimensions). The tubes were then saw cut to length, cleaned and annealed. After annealing, a sample was sent to the chemet lab for evaluation of the heat treatment. If acceptable the tubes were straightened, cleaned again and subjected to an in-process heat treatment (IPHT) for increased corrosion resistance.

3.6.4.1 Alpha-Beta Quench

From its review, the team determined that the alpha-beta quench heat treatment process for zircaloy tubing dated back to the early 1970s. This particular heat treatment produced a metallurgical structure of intermetallic precipitates which provided good corrosion resistance with no change in mechanical properties and characteristics (grain growth). In the 1980s (after an extensive corrosion test program by GENE), a solution quench heat treatment was developed to assure uniform resistance to nodular corrosion of the tubing. This process was accomplished by induction heat treating the tubing while simultaneously running cold water through the tube inside surface. This method of heat treating the tubing resulted in a "skin effect" protection of the outside surface to corrosion processes. In the late 1980's, the IPHT was developed for fuel tubing. This process used the solution quench heat treatment on the tubes but at a later step in the process. This had the effect of providing enhanced corrosion resistance with a faster quenching time and no change in the final tubes' mechanical properties or texture.

The team observed the IPHT also referred to as the alpha-beta quench. Quality Control Inspector Instructions (QCII) No. 15.2.1, "In-Process Heat Treatment," Revision 16, June 9, 1995, was used to assure that proper heat treatment results had been attained. A sample metallurgical cross section was prepared to observe the depth of penetration of the heat affected zone. The thickness of the heat affected zone was measured and compared to the specified limits. Product heat treatment was accepted when the heat affected zone was within the specified limits. Other visual criteria were checked and the product accepted for continued processing when the tube shell had passed the criteria of QCII 15.2.1.

The team determined that manufacturing and quality control operations were performed according to the required operation parameter sheets (OPS) and QCIIs in the order specified by the shop traveler.

3.6.5 Fuel Bundle Components

The team determined that, at the time of this inspection, NEP produced fuel bundles for six fuel designs, GE8 through GE13. The rod matrix design was 8x8 for GE8, GE9 and GE10, 9x9 for GE11 and GE13, and 10x10 for GE12. All fuel rods offered the GENE patented barrier cladding option and all fuel designs offered axial gadolinia zones. All fuel designs except GE8 offered axial enrichment zones, the large central water rod and ferrule spacers. Designs GE11, GE12 and GE13 offered an anti-debris lower tie plate and part-length rods. NEP's manufacturing of the following fuel bundle components was reviewed by the team.

3.6.5.1 Upper and Lower Tie Plates

The team reviewed manufacturing and inspection operations for upper and lower tie plates, including debris-filter lower tie plates. Both the upper and lower tie plates were machined from a single stainless steel investment casting. The debris-filter lower tie plates were machined from two separate investment castings welded together. Manufacturing and inspection operations were specified on shop travelers for all tie plates. The team noted that two QCII's reviewed, QCII 501, "Upper Tie Plate (GE9)," Revision 5, February 10, 1994, and QCII 501, "Lower Tie Plate (GE9)," Revision 4, August 28, 1992, had the same QCII number which had the potential for creating some confusion.

The team observed that activities affecting quality were being performed in accordance with written instructions and suitable tests and verifications had been satisfactorily completed for activities affecting quality.

(1) Lower Tie Plate Debris

The team evaluated NEP's response to a recent issue regarding manufacturing debris in lower tie plates. The following paragraphs summarize the team's findings.

During an audit of NEP by Carolina Power and Light Company (CP&L) in January 1995, its auditor discovered a piece of metallic debris lodged in a GE10 lower tie plate between a tie rod and an adjacent peripheral rod boss. The debris was ≈ 1/2-inch long and 1/8-inch wide. The team noted that if such debris was allowed to remain in the lower tie plate, it can potentially lead to a failure of fuel rods in reactor operations due to a phenomenon known as debris fretting.

After an additional piece (≈ 1/8-inch long and 1/8-inch wide) of debris was identified that had not been detected by NEP quality inspectors, NEP instituted an augmented inspection program in both SCO, where the tie plates were manufactured, and in FMO, where the tie plates were assembled into the completed fuel bundle. The augmented inspection consisted of an improved visual inspection and a 100% mechanical sweep of the peripheral locations in which the debris had been discovered. NEP also formed an investigation team to determine the source of the entrapped debris and to assure that the necessary steps were taken to prevent recurrence.

In addition, NEP decided to reinspect the GE10 lower tie plates for CP&L's Brunswick Steam Electric Plant Unit 1 reload which had already been assembled and which were awaiting shipment in shipping containers at NEP (Quality Notice F-P-1299, "Reinspection of Bundles for Lower Tie Plate Debris, Revision 0, dated January 18, 1995). This reinspection of the assembled lower tie plates resulted in the detection of one additional piece of debris ($\approx$ 1/10-inch long and 1/5-inch wide).

The team reviewed NEP's response to these findings and found that, according to NEP, the difficulty with which the debris was observed indicated that inspection personnel needed a greater awareness of the condition to observe it. As a result of its investigation, NEP performed debris awareness training of inspection personnel, improved lighting to aid in debris observation, and revised procedures and training documents to reflect these changes. NEP also changed its manufacturing activities to prevent debris from being caught in the tie plate. Cutting tools were retained in a sharper condition to minimize the number of burrs generated. Improvements in deburring were established. Hollow cutting tools providing coolant flushing action were incorporated in addition the standard external coolant flush as well as high pressure washing of the tie plates.

The team observed that foreign material exclusion zones were established and general improvements in housekeeping requirements were instituted in the foreign material exclusion zones. Debris generating activities were not permitted in the foreign material exclusion zones. Part storage areas were removed from proximity to the manufacturing areas and placed in the foreign material exclusion zones. Wrapping for storage was changed to improve debris shedding abilities and various other improvements in material handling have been incorporated to minimize the possibility of debris contamination.

From this review, the team determined that NEP also planned the following future improvements:

- The fuel bundle assembly area will become a foreign material exclusion zone with limited access by personnel.

- NEP will establish cleaning lines for the exclusive use of tie plate cleaning operations.

- Air-tight containers will be used for tie plate storage.

NEP also developed two analyses of the debris problem. A Kepner Tregoe type problem analysis and decision analysis were performed in February 1995, and a Tap Root type analysis was performed and documented in "GENE Energy Production Root Cause Analysis Report RCA-SCOQA9501," dated March

17, 1995. The team evaluated both reports and found them both to be an exhaustive analysis of the debris problem. As evidence of NEP's thoroughness in its analysis of the debris problem, the team noted that NEP determined that an outside auditor had first identified the lower tie plate debris in March 1994. NEP noted that the NEP system at that time did not react vigorously to the earliest signal of debris in its lower tie plates.

Additionally, on January 20, 1995, NEP requested a potential safety concern (PSC) review in accordance with its 10 CFR Part 21 procedures: EOP 65-4.00, "Potentially Reportable Conditions," Revision 2, dated October 23, 1992; NEP P&P 30-07, "Reporting Defects and Noncompliance," Revision 10, dated February 15, 1995; and NEDE-31746, GENE P&P 70-42, "Reporting of Defects and Noncompliance Under 10 CFR Part 21," issued August 1994. On February 6, 1995, GENE concluded that PSC 95-01, "Metallic Debris in Fuel Bundle Lower Tie Plate," was not a potentially reportable condition (PRC) nor a safety concern for the following reasons:

- There was no safety limit impact. There was no reason to believe the debris would reduce the margin to the MCPR safety limit, nor that it would affect the reactor water level or the coolant system pressure.

- As the debris was inside the channel, there was no concern with interference with control rod motion. Any debris small enough to find its way into the bypass region would be too small to pose a problem.

- There was a concern for potential localized fuel rod failure resulting from debris fretting. This would be readily detectable through the offgas system and poses an operational concern, not a safety concern.

Additionally, GENE first called all of its customers to inform them of the lower tie plate metallic debris and followed up with at least three written communications to its customers in February, March, and April, 1995. The NRC has tracked the lower tie plate debris as a 10 CFR Part 21 issue. On the basis of the its review of GENE's corrective actions taken, problem analyses, and PSC evaluation, the team concluded that the 10 CFR Part 21 issue can be closed on the basis that the GENE's actions were adequate.

Additionally, GENE provided assistance to those licensees that decided to inspect the peripheral flow region in the lower tie plates in recently delivered fuel that was fabricated prior to implementation of the augmented inspection procedures described above. For the inspection and removal of lower tie plate debris from unirradiated fuel bundles at licensee's sites, GENE issued Fuel Examination Technology procedure 246-GP-18, Revision 0, dated February 1995.

[deleted pursuant to 10 CFR 2.790 - document described specific values]

The team's investigation concluded that NEP management, in conjunction with and in full agreement with the licensee, did make a conscious decision to ship ce.tain fuel bundles to the licensee without reinspecting the bundles at NEP's facility. However, in these instances NEP implemented its augmented inspection of the fuel bundles at the licensee's site, as describe above. NEP's rational for inspecting the fuel at the licensee's site as opposed to inspecting in the NEP facility after the bundles were packaged in shipping containers awaiting shipment was to avoid possible damage in NEP's facility by utilizing the licensee's new fuel inspection equipment for up-ending and inspecting fuel bundles.

The team concluded that with the exception of not reacting vigorously to the first indication of a debris problem in lower tie plates that occurred in March 1994, the actions taken by GENE to the January 1995 Brunswick event were very proactive and beneficial to other licensees concerning the problem.

3.6.5.2 Spacers

The team observed the fabrication of GE9 spacers. Spacer fabrication was performed under the control of an computer record-keeping system referred to as the Traceway System. Documentation regarding all the components was entered into the system to provide traceability from the spacer assembly to the raw material. All the components were assembled into a fixture that accommodated welding and was loaded into a computer controlled 5-axis positioning system and laser welder. Laser marking for assembly identification was performed in the same setup.

Weld penetration samples were taken as required and submitted to the Chemet Laboratory to be evaluated for weld penetration of one joint of each of the following types: cell to cell, cell to band, cell to support plate and band to band. Evaluation of the weld penetration information was completed and determined to be satisfactory before the lot represented by the sample was released for assembly into fuel bundles. The team confirmed that spacer assemblies were constructed and inspected in accordance with approved drawings, procedures and specifications using materials that met design requirements.

3.6.5.3 Fuel Channels

Fuel channel strips were procured from NEP's suppliers in matched pairs with significant refinement and machining prior to the strip being welded into a fuel channel box. NEP assumed credit for supplier provided machined channel strip devices in the safety analyses. Other than the issue of fuel channel procurement and safety classification described in Section 3.6.2 of this report, the team did not identify any issues of concern during the fabrication or inspection processes for fuel channels.

3.6.5.4 Control Rods

GENE control rods are of two designs, DuraLife™ and Marathon™ blades. Both designs were manufactured from high purity stainless steels and the designs had been approved by the NRC. The DuraLife™ blades utilize stainless steel tubes encased in a stainless steel wrapper which protects the individual tubes and provides stiffness to the design. The Marathon™ design reduces both the volume and surface area of stainless steel in the irradiation zone. The design utilizes square tubing which is laser welded. The tubes are 100% eddy current (ET) inspected before welding. These tubes are not wrapped in a stainless shell, but are open to the reactor environment.

While observing the start of mechanical measurement of a thrust limiter, the team inquired into the availability of the dye penetrant procedure which had been performed on the part. The team was informed that the procedure could be at another location, and was not available at the work station. Further discussion with the employee disclosed that the dye penetrant examination had been performed from memory. The team concluded that this lack of procedures in the work area was contrary to Section 5, "Instructions, Procedures, and Drawings," of the QA topical report.

From its review, the team made a second observation from this incident that concerned a shop traveler which had two different inspection steps in sequence. Since the traveler was unsigned and unstamped for both of the inspections (the dye penetrant examination had been previously performed) it was not possible for the team to determine the status of the inspections performed. The QCII for the first operation was also not available at the work station. The team concluded that this practice was also contrary to Section 5 of the QA topical. The team found that this traveler also served as evidence of completion of the operation and certification of the acceptance status of the item(s) inspected.

In response to the team's observations, NEP management instructed floor personnel (the same afternoon as the observation) on the importance of having procedures available for use at the work station. These instructions were documented (by signature attendance) and were shown to the team. The team noted that the attention to corrective action coupled with no additional areas noted without procedures (during the inspection) demonstrated prompt and adequate short term response by NEP management. The larger issue of addressing the culture of having procedures available for use at the work station will be addressed in the training plan improvements in Section 3.8.2 of this report.

3.5.5.5 Finger Springs

During the inspection, NEP identified a problem in a shipment of finger springs. Of the 503 finger springs received, 9 – 11 of the finger springs may have oversized thickness. NEP stated that the spring's function is to prevent bypass flow; the loss of a spring leaf will allow some bypass flow. NEP determined that a broken leaf cannot travel to an area where it could cause problems as either a loose part or a contributor to unanticipated flow restriction. Although the discussions with NEP disclosed that the oversize finger springs did not present a safety issue but could cause a degradation in fuel bundle performance, the team monitored NEP's response to this event.

NEP initiated corrective actions to perform 100% inspection of the received 503 finger springs, including those installed in completed fuel bundles. In addition, NEP initiated Supplier Corrective Action Request W3102, dated August 15, 1995. NEP's supplier initiated corrective actions to review all previous coils of material supplied by their sub-contractor, to review the continuous thickness chart of the master coil in addition to the material supplier charts.

The team concluded that the actions taken by NEP were adequate and prudent for the complexity and severity of the oversize finger spring material.

3.6.6 Fuel Rod Fabrication

From its review, the team determined that the fuel rod fabrication process starts with the welding of the lower end plug. The lower end plugs were welded into the fuel rods using a butt weld between the end plug and tubing. The butt weld presented a smooth surface for the examination of the weld using an ultrasonic (UT) examination. During the UT inspection, the bar code was read and related to the number stamped on the lower end plug. The results of the UT inspection and the identity of the fuel tube were entered into the fuel business system computer.

(1) UO_2 Fuel Rods

The team observed the operator of the rod loading machine draw from storage a specified batch of empty fuel rods with lower end cap welded. The operator also drew trays of released pellets from storage to meet the loading pattern for the tubes. The loading machine was computer linked to the MICS as well as reads the bar code and provided approval to proceed.

The team also observed the operator load pellets from the tray up to the push rod. The push rod checked the stack length; the stack was weighed and the weight and length were entered into the computer. When all zones were loaded and weighed, the entire stack of pellets was inserted into the tube. As the fuel was pushed into the tube, a laser micrometer measured the diameter on each pellet. The weight measurements for each enrichment were reported as the net weight in fuel certification reports.

The team determined that the upper end plug weld was made with bead welders or flush butt welders. As each tube enters the welding chamber, the bar code was read by the machine and the rod type must compare with the specification for rods being welded, otherwise the welding operation would not be allowed to proceed.

(2) Gd_2O_3-UO_2 Fuel Rods

The team found that the operation of the gadolinia line was similar to the UO_2 line.

3.6.6.1 Rod Inspection

The team determined that in the case of bead welded end plugs, a sample was taken for X-ray inspection. In the case of butt (flush) welded end plugs, each weld was subjected to a UT examination. There was a visual inspection of each completed rod while rotating. Each weld must be acceptable based on visual standards.

A sample of the rods, including the first rod welded, was checked for end plug parallelism. In the case of bead welded end plugs, a sample of the rods, including the first acceptable rod, and all reworked rods were segregated and tagged for X-ray analysis. A sample of production rods was taken weekly for analysis of rod atmosphere. Sample welds were made for a weekly 72 hour corrosion test and for a daily visual examination and weld parameter printout. Individual rods were not subjected to a helium test. A helium test was performed on the completed fuel bundles.

3.6.6.2 Rod Scanner

Each UO_2 rod was sent through an active scanner used to detect enrichment blending. The scanner, nicknamed "Fat Albert," had a **[deleted pursuant to 10 CFR 2.790 - document described a specific value]** percent probability of detecting a single pellet that was greater than **[deleted pursuant to 10 CFR 2.790 - document described a specific value]**% difference in relative enrichment than the adjacent pellets. The detector for enrichment provided a map which was compared to the specification trace. If the comparison was within limits, the rod was accepted and the computer sent its acceptance to the MICS computer system.

The team reviewed NEP's report of a 4.2% enriched, 0.376-inch diameter pellet found in a zone of 3.95% enriched 0.411-inch diameter pellets in a fuel rod that was down-loaded for other reasons. According to NEP, the active scanner failed to detect this error because the difference in diameters compensated for the difference in enrichments. Since that event, the laser micrometer inspection (described in Section 3.6.6.1 of this report) was added to prevent a recurrence. NEP concluded that there was no definitive explanation for the presence of the rogue pellet; although it must have occurred somewhere between pellet grinding and fuel loading.

The team noted that if the laser micrometer inspection had been a part of the rod loading process when the defective rod was loaded, the loading machine would have stopped pushing pellets into the rod when the misplaced pellet was encountered. The team concluded that the probability of this type of defect occurring again was small, as long as the laser micrometer inspection was continued during rod loading.

From its review, the team determined that the Gd_2O_3 bearing rods cannot use a neutron source for scanning, therefore, a magnetic and passive scanner (MAPS) was used. The results obtained from Fat Albert and MAPS systems was used as certification for the content of U_{235} in each rod.

3.6.6.3 Storage and Release

Fuel rods that have successfully passed the Fat Albert and MAPS scanners were placed in storage. These rods were then available for accumulation of rods for bundle assembly. The team verified that adequate measures were established to control the release of rods to bundle assembly.

3.6.7 Fuel Bundle Assembly

From its review, the team determined that production travelers for fuel assembly operations were available on computer terminals at the workstations and acknowledgment of the completion of each production and inspection operation was also made at the computer terminals. The team observed the assembly of a GE9 fuel bundle. Skeleton assembly began with the assembly personnel drawing from storage the appropriate number of spacers and a center water rod, all of which had been released for assembly. Identifying numbers on these components were entered into the computer system which confirmed their released status. When all spacers were properly positioned, the skeleton was clamped into the automated bundle assembly machine (ABAM). The lower tie plate was also properly oriented and clamped into position in the ABAM. This completed the skeleton assembly process.

ABAM examined the preestablished bundle matrix pattern and loaded the rods into the bundle. Any discrepancy between rod identification information and the preestablished bundle matrix pattern caused ABAM to stop until the discrepancy was resolved. After all the rods were installed, the upper tie plate was placed in position and lock tab washers were installed.

3.6.8 Fuel Certification

The team reviewed NEP's basis for reporting the enrichment of fuel delivered to licensees. NEP sends three reports to licensees, as follows:

- Final Certification Bundle Verification for Bundle - provided a detailed report for each fuel bundle. It referenced the drawing number for the bundle and the drawing numbers for all components as well as a fuel bundle matrix which identified the rod type and number for each location in the matrix. This report also listed the composition of each fuel rod in the bundle by zone; include the zone number, the design enrichment of the zone, the load date, the actual enrichment of the zone, the net weight of the zone, the uranium weight in the zone, and the U_{235} weight in the zone. The uranium weight and U_{235} weight are determined from measurements by the active or passive scanner devices (Fat Albert or MAPS).

• Bundle Shipping Document - listed each bundle in the shipment and gives the net weight, uranium weight, and U_{235} weight for each enrichment in the bundle. The totals for each bundle and the total for the shipment are also reported.

• Product Quality Certification Document - provided an itemized list of fuel bundles in the shipment, the product configuration for each bundle model number, and the number of bundles of this configuration in the shipment.

3.6.9 Laboratories

The team evaluated NEP's Chemet Laboratory and the Gage Laboratory. The following paragraphs summarize the team's findings.

3.6.9.1 Chemet Laboratory

The team found the Chemet Laboratory an integral part of the in-process release of raw materials, process control, and final release of finished products. The Chemet Laboratory consisted of a metallurgical laboratory, wet laboratory, and spectrographic/uranium recycle laboratory.

(1) Metallurgical Laboratory

During the team's review of lab examination techniques, using some weld samples chosen at random from a Marathon™ control rod final package for Philadelphia Electric Company, Peach Bottom Atomic Station Unit 3 (Peach Bottom 3) (WA HE3958058), the team determined that daily weld samples (DWSs) weld penetration examinations were performed at the lab without procedures. While verifying the metallurgical examination of weld samples from Drawing 107E61226002, Lot 76MRO for welds on the Marathon™ control rod, the team determined that the weld penetration examination was performed in accordance with a memorandum and not an approved procedure. (The memorandum, dated April 7, 1992, was a request for the lab to issue these instructions as a procedure.) The applicability of the memorandum was directed to the following Marathon™ DWSs:

- weld-9 annulus seal (S1,S2)
- weld-12 square tube to end plug (A1,B1,A2,B2)
- weld-14 square tube to end plug (D,E,F)

As a result of its investigation into the DWS evaluations in response to the team's concerns, NEP determined that other Marathon™ design weldments were also evaluated for weld penetration without benefit of procedures (e.g., weld-5 tube to tie rod, and welds-6 and -7 tube to tube).

Therefore, the team concluded that since there was no other procedure for the determination of weld penetration on the Marathon™ configuration, either the lab personnel have been using this non-approved procedure or they have been evaluating this manufacturing parameter with no procedure. For the Marathon™ design this issue became a concern for the team because there was no NDE performed on these welds and the destructive evaluations of the DWSs was the only criteria with which to judge the quality of the laser welder.

The team raised the additional issues of how and who evaluates metallurgical tests since many requests come to the lab with specific instructions while others simply have a miscellaneous weld designation assigned to them. The team concluded that the only reason the system currently works was that the lab had a staff comprised of personnel with many years of metallurgical laboratory experience and could have resulted in a worse condition if there was a large turnover of personnel. As a result, the team identified this issue as a potential nonconformance during this part of the inspection.

In response to the team's determination that DWSs weld penetration examinations were performed by the lab without procedures as required by P&P 30-10, "QA Program - GENE," the following corrective actions were either planed or taken by NEP during the inspection. Corrective Action Request (CAR) 95-12, dated August 28, 1995 included the following recommended actions to correct the procedural oversight:

- Issue Chemical, Metallurgical and Spectral (CM&S) procedure for the evaluation of the DWSs for the welds identified above. Provide training in usage thereof to lab analysts. Conduct training of lab personnel to identify to management any requested analysis for which written instruction was not provided.

The team reviewed CM&S Analytical Method 2.2.1.0, Revision 0, "Metallographic Evaluation of Laser Welds," and concluded that it adequately covered the appropriate welds to be evaluated and the necessary procedural information to inspect for weld penetration as well as other weld attributes on the Marathon™ blades.

- Review reported weld penetration results for Marathon™ DWSs, characterize and document the margin of acceptance criteria. Document margin of acceptance criteria to satisfy Marathon™ production to date.

The team reviewed Quality Notice E-G-064, Revision 0, dated August 31, 1995, "Review of Postulated Variability in Marathon™ Laser Weld Penetration." NEP reviewed a random sample of 603 weld penetration evaluations from the Marathon™ production period of 1989 to the present. This review indicated that all of the welds could be accepted after considering such factors as visual inspection or bonus tolerance. According to NEP, the end result of this exercise indicated that the design requirements provided adequate assurance that any postulated variability in the measurement technique for the weld penetration results would not affect product safety or performance. The team concluded that from this review that NEP's review provided the necessary actions.

- Review lab weld analyses and notify affected parties if there are any evaluations performed by the lab analysts without instructions.

The team reviewed Chemet Lab memorandum CL-95-056, "Marathon™ Weld Evaluations," dated August 30, 1995. The memorandum documented that training had been provided to the analysts and that three additional evaluations were being performed without benefit of procedures. These included the bar code depth heat-affected zone, water rod weld, and channel clip adapter weld. This element of NEP's corrective actions remains open, pending evaluation by NEP to determine whether any safety concerns were raised for products which had been evaluated for these attributes. The team requested that NEP notify the NRC when the evaluation was complete. (See Open Item 95-01-06 below.)

- Review lab activities and controls. Issue instructions to ensure that approved written procedures were used to perform any lab analysis. Conduct training of personnel on control of written instructions for lab analysis.

This action was to be completed by NEP by October 31, 1995. This element of NEP's corrective actions remains open. The team requested that NEP notify the NRC when the review was complete.
(Open Item 95-01-06)

As a result of its review of CAR 95-12 and the corrective actions taken by NEP, the team determined that its concern regarding DWSs weld penetration examinations that were performed by the lab without procedures had been satisfied and that the potential nonconformance was closed. Additionally, in its letter dated September 11, 1995, NEP submitted its documentation for CAR 95-12.

(2) Wet Laboratory

The wet chemical laboratory performs a variety of chemical and physical analyses on UO_2 powder and pellets. Several fuel rod and assembly components are also analyzed in the wet lab. In addition, archive samples of all of these materials are catalogued and placed in storage through the wet lab.

The team determined that analyses performed on UO_2 powder samples include isotopic, carbon (C), Chloride (Cl), fluoride (F), nitrogen (N_2), water (H_2O), and Oxygen/Uranium ratio (O/U). Analyses performed on pellets include O/M (Gd pellets), isotopic, C, Cl, F, N_2, hydrogen (H_2), total outgas, densification, O/U, and porosity. The following observations were made.

A LECO hydrogen analyzer was used for the analysis to measure H_2 release from UO_2 pellets and zirconium and stainless steel.QN F-Q-2006, "Qualification of LECO RH404 Hydrogen Analyzer," Revision 1, dated July 30, 1993, documents the basis for using this equipment for hydrogen analysis.

The team determined that there was no direct measurement of the actual temperature reached by the sample during measurement. The specified temperature was > 1800°C. A power setting of 2600 watts was found to be sufficient to melt chromium metal using the heating schedule used for UO_2 pellet analysis. The procedure specified a chrome melt temperature check once each month. Pure chromium melts at 1857°C.

Upon further investigation, the team determined that a Fisher Chemical Company jar of chromium granules (stock number C-317) was used for this test. There were no certificates or purchasing papers for the chromium metal and the same jar of material had been used for several years. Fisher Chemical Company was contacted by NEP for information regarding the purity of the chromium. NEP found that item C-317 was a discontinued item that had not been offered for several years and that the purity claimed by Fisher Chemical Company for this material was typically 99% pure.

NEP stated that given the 99% purity of the chrome metal, the chrome melt temperature check, together with the unchanging induction heating operating parameters through the years, had given adequate confidence that the samples were indeed being heated to > 1800°C.

From the "Handbook of Binary Phase Diagrams," Update Number 46, December 1994, by William C. Moffat, Genium Publishing Co., Schenectady, NY., the team examined binary phase diagrams for chromium with 38 different metallic elements. From this review, the team determined that the greatest lowering of the melting point by any of these at low concentrations were by platinum, which lowered the melting point to ≈ 1700°C at about 3 atom per cent, and lead, which required about 4 atom per cent for the same effect. Even though NEP had not verified the alleged purity of chromium samples, the team concluded that the chrome melt test, along with the record of unchanging power parameters, were serving the intended purposes of verifying temperatures reached in the H_2 analysis.

The team observed the chemical analyst perform a verification of the system using a H_2 standard. The results for four tests were 5.89, 12.93, 4.74 and 4.67 ppm H_2. The operating procedure gave the analyst the option of rejecting a result for sufficient cause; the 12.93 ppm was rejected by the analyst based on "common sense and logic." The team requested the analyst to explain how far a value had to be from the standard's stated value to be discarded. The analyst stated that any number that differed from the accepted value by more than three times the ± value for the standard would be rejected (as provided for in procedure COI 003, "Chemet Laboratory Rerun Outlier Criteria," Revision 8, dated May 13, 1994).

The team observed the analyst analyze three of the 1.4 ppm H_2 standards which obtained an average result of 2.42 ppm. This value represented a recovery of 172.86% of the stated value. Whenever the instrument was calibrated, the results were automatically entered into the LMCS. When results fall between the lower and upper alarm limits, the system is considered operating satisfactorily. The team observed that NEP did not determine whether these values were statistically significant in determining the control of the calibration process for the analyzer.

The team determined that in this case the value of 172.86% is above the alarm limit but within the control limits. The analyst was required to make an investigation but it was not clear to the team how extensive an investigation was required. During the team's review, the analyst received a standards results report covering the last 25 calibrations, which covered a period of 14 days. Four of the last six results were between the upper alarm limit and the upper control limit, one occurring on each of four consecutive days. All of the results for the preceding 10 days were inside the alarm limits. The team concluded that, based on NEP's failure to correct this situation after four days, the investigative procedure invoked when the alarm limit was exceeded was not effective.

The team determined that the specification maximum limit for H_2 in UO_2 fuel pellets is 1 ppm. Typical results for production material is about 0.1 ppm. In the qualification document (F-Q-2006) the average found for 15 pellets from the same tray was 0.09 ppm (± 0.025 ppm). Pellet weights are six to eight grams. Thus the amount of H_2 measured is about 0.7 μg. The "blank" value for empty crucibles is about 0.5 μg, a value almost equal to the sample value. The team concluded, therefore, that the hydrogen content of UO_2 fuel pellets was so low that the analytical technique was pushed to its lower limit for detecting hydrogen.

The team concluded that this system was highly automated in handling data when results are all nominal, but it left considerable leeway to the operator for discarding poor results. The team also observed that in some instances there seemed to be a lack of in-depth understanding of the equipment operators regarding the equipment they were running and the reason they were performing certain steps in the operation.

(3) Spectrographic/Uranium Recycle Laboratory

The team observed that the spectrographic lab equipment in operation included a grating spectrometer, two ion coupled plasma mass spectrometers, and a conventional mass spectrometer. The grating spectrometer was used to determine the concentration of 26 metallic impurities in UO_2 powders and pellets, and also analyzed other materials used in the plant.

With the exceptions noted above, the team concluded that the overall procedures and practices in the Chemet Laboratory were well organized and well run. Operating personnel were generally well qualified for their positions, or were monitored by experienced personnel. Calibrations of instruments used for operations and analysis were carried out in a timely fashion.

3.6.9.2 Gage Laboratory

The team reviewed the calibration, tracking, and control of measuring and test equipment (M&TE), such as micrometers and calipers used by QC inspectors and workers throughout NEP. The inspectors noted that the NEP gage lab was responsible for all of the calibrations and repair of M&TE. The NEP gage lab was staffed by 5 individuals who maintained and updated the QC Gage Control System, a computerized central mainframe database that was used to track and control M&TE. The database contained an active equipment list of approximately 3200 gages and other instruments with an archive file of over 30,000 M&TE records that have accumulated since the late 1970s. This computer system, created over 20 years ago, had been updated several times, although the team determined that the system still had only limited sort and no trend capability. The team identified the following observations that could have an impact on the use of the QC Gage Control System.

• The team determined that there did not exist a user's manual for the database, either at the terminal workstations, or at the computer mainframe programmer interface, even though this system had been designated by NEP as a "QA-controlled software record file." This was a concern to the team because NEP procedure 120-13, "Development, Control and Maintenance of NF&CM Computer Software," required that user manuals be available for QA-controlled record files unless a valid exemption could be established. The team noted that although the lab personnel were intimately familiar with the gage control system, only their many years of experience in use of this database has enabled them to avoid potential gage control problems and prevent situations which could allow out-of-calibration equipment to go unnoticed. In response to the team's observation, NEP stated that a user's manual was in development and would be completed in the near future (as discussed in Section 3.7 of this report).

• The team observed the lack of procedural control concerning new gages that had been calibrated but were found to be out-of-tolerance or in a nonconforming condition. Although these gages had been tagged as nonconforming, they were stored in the same location with new, calibrated gages and NEP's existing procedures had not required these gages to be tracked or accounted for in a log. After discussions of this observation with NEP, NEP revised procedure A13, "Entering Gages Into the Gage Control System," to include additional requirements on disposition of new gages that were found to be nonconforming.

• The team found that each of NEP's supervisors (who were also the designated "owners" of gages), received a weekly status report of gages that were past due, delinquent, or lost. NEP's system for controlling these gages was based on a 1% delinquent or overdue threshold which was maintained and tracked by the Gage Laboratory. M&TE owners who had 1% or more of assigned gages overdue or delinquent could not receive additional gages. The team found that during the week of August 21, 1995, 1.7% of 3233 gages were overdue or delinquent (approximately 56 gages). Of the 46 NEP supervisors within 22 functional departments, 14 of these supervisors were at or above the 1% threshold. In three of the 22 department areas, between 9% - 33% of the gages assigned to these supervisors were overdue or delinquent. The team determined that, although these numbers change on a weekly basis, various individuals who needed additional gages were routinely denied gages at the beginning of each work week.

• The team observed that, in the past, the many years of experience and low turnover rate of NEP's Gage Laboratory personnel had minimized problems associated with the control of M&TE. However, the team concluded that the lack or unavailability of gage trending data could allow an increase in gage control problems to occur.

3.7 Corrective Actions

The team evaluated NEP's corrective action program. The following paragraphs summarize the team's findings.

A corrective action request (CAR) form was implemented and tracked to correct a problem in the design/analysis process that did not fit in the design change requests or in the engineering computer program (ECP) problem report. A CAR can be initiated from within NEP or from a customer. Once the CAR is initiated there was a 30 day time period for a corrective action plan to be identified although there was no time limit to implement the corrective action, or to close the CAR.

A ECP Problem Report was generated for a software problem that was categorized as either a Type A or Type B problem. (Type A was defined as changing the calculated design value and a Type B as having no impact on calculated design values.) If a Type A problem was identified it was up to the ECP responsible engineer to report the problem to the ECP users within 30 days of the problem by memo (per EOP 40-3.00, "Engineering Computer Programs," issued March 27, 1995). The memo requests each ECP user to determine the impact of the problem on analyses, identification of the specific analyses, and to recommend a corrective action. It was the responsibility of the ECP responsible engineer to implement the corrective action, notify users that corrective action has been implemented and the corrective action taken (software changes implemented). The team reviewed several Type A software problem report to determine the types of problems, assessed the adequacy of the corrective actions and evaluate the time taken to close out the CARs.

Two ECP problem reports (94018 and 95004) were greater than 90 days old and not closed out. The team observed that neither of these ECP problems were of great significance in terms of calculational errors; but concluded that a maximum time limit to close previously identified problems would eliminate Type A problems from being left open for a year or longer, which could have a significant impact.

Type A problem report 95049, issued on July 7, 1995, identified in the RELSB02V program was also reviewed by the team and found to be significant because it would always list the rotated bundle as having the most limiting MCPR when both the mis-located and rotated bundle analyses were performed. The team concluded that this problem was significant because the mis-located bundle analysis could be more limiting. At the time of the team's exit, NEP was continuing its evaluation of this problem report to address the team's concerns.

The Nuclear Fuel Users Forum (NFUF) performed an audit at the NEP facility during the week of May 1, 1995. The NEP response to the NFUF findings was included in the team's review, as part of its evaluation of NEP's corrective action program.

- For CAR 95-003-OBS-2, omission of a user's manual for the rod load pellet diameter laser micrometer computer software, NEP stated that the reason for omission was "operator action was not required with the software" and has since determined that this statement was incorrect. The corrective action taken by NEP included changing the reason why the manual was omitted. However, the team observed that NEP did not consider the broader corrective action to re-review the "grandfathering and baseline" efforts that were undertaken in 1992 to bring up-to-date approximately 175 pre-existing computer-controlled software and databases, which was the basis for the original incorrect statement.

During the team's review of this CAR, the team identified another database where the QC Gage Control Database System user's manual did not exist. NEP's justification for not requiring a user's manual was again its review during the 1992 baseline effort. Although this baseline review identified that no user manual was required, the team questioned this rationale because this system was a QA-controlled database listed in the software records file; users frequently need to manipulate and change data within the database and perform other manipulations in order to obtain database printouts.

As a result of interviews with several of the users and the designated QC Gage system engineer, the team concluded that without a users manual present, competent operation and use of the database system was not feasible without significant on-the-job training by other competent users. NEP agreed with the team's conclusion and stated that the QC Gage Control Database manual would be completed in September 1995.

The team noted that the adequacy of the 1992 baseline effort was also questioned by a GENE NQA internal audit of the information management systems being conducted concurrently during this inspection. Preliminary findings from this audit, as presented to the team, appear to address the same observations of the team.

- For CARs 95-003-OBS-1, and -3, the team concluded that NEP provided an acceptable response. However, in CAR 95-003-OBS 9, NFUF noted that the basis for allowing the rod to exceed the Rod Internal Pressure (RIP) guideline was not provided in the DRF (as required by EOP 42-10.00). The fuel rod thermal-mechanical evaluation of the GE11 BWR2 high Gd fuel rod was given in DRF J11-01663. After reviewing the NEP response CAR 95-003-OBS-9, the team concluded that the response also did not provide an adequate basis for accepting the RIP guideline violation. When this concern was brought to the attention of NEP, the necessary DRF updates were made immediately and the CAR report was reissued. NEP's corrective actions taken in response to the team's concern was acceptable.

3.8 Training

The team evaluated NEP's personnel training program for those activities that affect quality in reload core engineering and fuel production. The following paragraphs summarize the team's findings.

3.8.1 Reload Core Engineering

The team noted that the methodology for core design and licensing at NEP had clearly evolved over a number of years, as demonstrated by 77 technical design procedures (TDPs) and a similar number of engineering operating procedures (EOPs). There was a set of 7 separate TDPs that relate to critical eigenvalue and reactivity calculations alone indicating the complexity of the core design and licensing process. Given this complex design process, the team determined that training for the design engineers was needed to acquire and maintain suitable proficiency.

The team concluded that the recent relocation of the reload core engineering group from San Jose, California, to Wilmington, North Carolina, resulted in a need to hire new staff members to make up for lost personnel as well as emphasize the critical role played by training in core design and licensing activities. NEP appears to have come to this conclusion as well, and has recently issued a TDP on training (TDP-0077, "Training," Revision 0, issued August 1995).

Training at NEP is provided to the responsible engineer within a loosely hierarchical matrix that includes the responsible manager, the process development leaders (who are assigned to five technology areas, and are responsible for continual improvements of the work flow process within these areas), mentors (who have demonstrated in-depth knowledge of a specific technology area, and are utilized as training and consultation resources for responsible engineers), and verifiers (who have demonstrated sufficient proficiency in a specific technology area to allow detailed verification and review of the work performed by the responsible engineer in that area). The responsible engineer works closely with an assigned mentor till the responsible engineer demonstrates proficiency in independent analyses and verification. Each stage of proficiency was evaluated and documented by the responsible manager in concurrence with a process leader and a mentor, as appropriate. In addition to the proficiency demonstration by the responsible engineer discussed above, TDP-0077 required development of a process-based training plan to be maintained by each process leader in his area. This training was required for all responsible engineers on a continuing basis, and was to be conducted either in a formal class environment, or as one-on-one training. The elements of process training were to include a review of new technologies and procedures, a review of document or procedural changes, and discussion of CAR findings and other reported errors and preventive measures.

Recently, the reload process has been subdivided into 22 subprocesses, each one of which is scheduled to be covered approximately once a year in a training course. The team noted that cold shutdown margin calculation, an area that had recently experienced several instances of calculational errors and procedural shortcomings, had been covered in a training session in June 1995.

In order to evaluate whether the training programs discussed above are responsive to the specific needs of the design and licensing efforts at NEP, the root cause evaluation of the errors made in Duane Arnold 1 Cycle 14 eigenvalue calculations was reviewed by the team as a representative example. The root cause evaluation identified two causal factors, and ten associated root causes that led to the errors in the calculation. All ten root causes identified were human performance difficulty on the part of either the technical program engineer (TPE) or Management. The performance difficulties on the part of the TPE were related to missing knowledge or skill in procedures used, lack of training, incorrect use of procedures, incorrect or inadequate procedures, and inadequate training or controls provided to interpret limitations of analyses used. Preventive actions recommended (related to TPE performance) included training in relevant areas and modification of procedures. Most importantly, the recommended preventive actions also included a management plan to obtain consistency between procedural detail and training, an adequacy review of the procedures to ensure that they are structured to facilitate understanding and use by the newer engineers, and an increased emphasis on automation options. The recommended preventive actions related to management performance included a focused Fuel Engineering training on the verification process and the responsibilities for this process.

The level and extent of training available until recently at NEP, viewed in the context of the extensive and complex set of procedures used and the recent hiring of engineers after the move to Wilmington, North Carolina, was considered a weakness by the team. The team noted, however, that the training process had recently been formalized in TDP-0077, and a program of training in the reload subprocesses had been initiated. The team also noted that the root cause evaluation of the errors in Duane Arnold 1 Cycle 14 eigenvalue calculations had identified specific areas for focused training that should benefit the newer engineers and enhance the verification process. The potential benefits from the new training program need to be monitored to determine whether the weakness had been removed.

3.8.2 Fuel Production

From its review the team found that NEP had lacked a formal, integrated training program to address common technical training needs, plan and schedule site-wide training, track and manage records, and feedback audit findings and lessons learned.

At the time of this inspection, training was conducted and handled separately for each of the three production areas (FMO, FCO, and SCO). Managers in these areas were administering and managing their own training needs without benefit of common training needs or resources to provide the same level and quality of training to all NEP activities affecting quality. The team found that isolationism and resistance to change still existed at some levels within the NEP organization and that this had hampered past efforts to focus attention on site-wide training issues. Although NEP management appeared to recently acknowledge training as a general issue, the team observed that weaknesses in administrating and organizing technical training across the site may not have been fully recognized by management.

For example, the team's interviews with several managers indicated that the highly specific training requirements, such as on-the-job training for many of the fuel manufacturing activities would not have benefited from an integrated training program. The team found that many of the highly-experienced workers were now area coordinators or supervisors, and were also relied upon to conduct and administer training in their areas on an as-needed basis. This team effort had assisted management in maintaining the high skill level and experience necessary to maintain product quality. The team also concluded that for the majority of workers within NEP, a historical low turnover rate coupled with an average experience level of 10 to 15 years per worker were principle reasons why NEP had been able to minimize any detrimental effects from manpower shortages and the lack of an integrated training program. However, the team determined that an aggressive training program would facilitate NEP's future ability to maintain product quality.

During the inspection, the team found several examples in NEP's QA audit findings and other documents where a lack of training was identified as root causes and could have potentially impacted product quality if allowed to go uncorrected. For example, a 1994 internal QA audit in the SCO building identified several areas where over 40 CARs were written that had root causes traceable to inadequate training. Many of the identified deficiencies observed were personnel not following instructions and documentation errors in the SCO processes. This audit finding and the subsequent management briefing appeared to be the first indication to the team that at some level within NEP, training was fully recognized as a potential site-wide issue.

In another example, the team found that in 1994, two SCO managers identified that their staff lacked adequate micrometer training. SCO management subsequently organized formal training on micrometers and administered this training for all individuals who use micrometers in the SCO area. Although micrometer training was also a critical issue in the FMO building and FMO managers had experience problems in the past with individuals receiving inadequate micrometer training, the team found that FMO managers had not been aware that special micrometer training was planned or administered for workers in the SCO building. This example was typical of how training had been conducted in the past.

The team found that training conducted through out the NEP facilities was a complex and evolving area that did not have a central focus. The team noted that in 1994, several NEP QA auditors, training personnel, and others, presented to NEP management the need for improving the training programs currently being conducted at NEP. A training program manager was created in order to centrally focus and track training for all of NEP. Although these actions were only recently initiated, the team concluded that NEP management was taking appropriate steps in order to improve the weaknesses identified by the team and the NEP staff.

3.8.2.1 Quality At The Source

In 1992, NEP initiated a program whereby operator/worker job tasks and in-process inspection activities were combined into a single function performed by one individual. NEP called this program "Implementing Quality At The Source," or QATS. The program was in varying stages of implementation across the site. Although combined production/inspection tasks were not novel to fuel fabrication vendors, the team decided to perform an in-depth review of the QATS program in order to assess how the program was functioning, since it had not been used by NEP before. Since the team was told that QATS was only fully implemented in the FMO production facility, the team limited its in-depth review to the FMO production facility, although general aspects of the NEP QATS program were included in the assessment.

The team found that overall, the QATS program appeared to be a sound and practical approach for NEP to control product quality. However, in several instances, weaknesses in QATS training has not enabled NEP to achieve the level of effectiveness that was expected from such a quality program. Specifically, a lack of training for certain FMO QATS-qualified operators/inspectors on how to use inspection gages necessary for individuals in inspector-qualified positions.

For example, the team observed pellet grinding and inspection. During the observation, a QA inspector identified 5 trays of fuel pellets which contained unacceptable amounts of pellet defects and were rejected. The QA inspector told the team that it appeared the QATS-operator/inspector apparently failed to follow established procedures for the proper use of micrometers. The team found the that these individuals had completed QATS training and had been certified to the qualifications stated in the American National Standards Institute (ANSI) standard N45.2.6, "Qualifications for Inspection, Examination, and Testing Personnel." The team determined that this level of certification should have provided an acceptable level of performance to ensure product quality was maintained.

The team identified other examples which support these concerns, including for example, certain CARs issued by QA that contained the following findings:

- QATS-qualified operators/inspectors have left uncertified trainees unattended at the workstations who were performing pellet grinding/inspection activities. Although final product quality was unaffected, poor product quality was passed through the first inspection station.

- Unqualified trainee operators have used the QATS-qualified operator identification number in order to sign-on or login-in in order to operate computer-controlled equipment and some QATS-qualified operators condoned this practices and had repetitively allowed poor product quality to proceed unidentified, after performing the first quality inspection.

Root causes identified in some of these CARs indicated a lack of appropriate training for new employees. This root cause may include such identified causes as failure to follow established procedures, failure to follow instructions and properly document work processes.

3.9 10 CFR Part 21

During this inspection, the team evaluated the GENE procedure and the NEP procedures that address the requirements of 10 CFR Part 21. While the evaluation determined that the GENE and NEP procedures met the requirements of 10 CFR Part 21, the evaluation determined that revisions to the procedures and NEP's procurement practices (as described in Sections 3.6.1 and 3.6.2 of this report) will be necessary when the revised 10 CFR Part 21 is effective (October 19, 1995).

APPENDIX A

WEAKNESSES, STRENGTHS, AND OBSERVATIONS

Neither the weaknesses nor the observations described in the inspection report require any specific action by or written response from NEP. The more significant weaknesses, strengths, and observations identified by the team are given below.

- Sections 3.4.2.1, "Eigenvalue Selection and Uncertainties," and 3.4.2.2, "Cold Shutdown Margin," describe a recognized weakness in core reactivity calculations.

- Section 3.4.2.3, "Steady-State Nuclear Methods Improvements," describes new design methods as a potential strength.

- Section 3.4.3.1, "Cycle-Dependant Safety Limit Minimum Critical Power Ratio," describes observations regarding a cycle-dependant SLMCPR methodology.

- Section 3.4.4.2(1), "La Salle 1 Cycle 7," describes the development of a comprehensive and workable design interfacing procedure (DIP) as a strength.

- Section 3.4.4.2(2), "Hatch 1 Cycle 16," describes the lack of a complete verification of the FRED data as a weakness.

- Section 3.4.4.2(2), "Hatch 1 Cycle 16," describes the implementation of the Transient Selection Review Committee as a strength and noted, however, that it was a weakness to not require documenting the committee's selection basis.

- Section 3.4.4.2(3), "Limerick 1 Cycle 16," describes the high level of automation in the design and analysis process as contributing to the overall quality of the analysis process.

- Section 3.5.5, "2.6 Safety Limit MCPR," describes a weakness in adherence to the reporting requirements in Amendment 22 of GESTAR II.

- Section 3.6.3.6, "Grinding," describes an observation regarding the hydrogen analysis of unground pellets.

- Section 3.8.1, "Training," describes a weakness in previous level and extent of training and noted that the new training program will need to be monitored to determine whether the weakness has been removed.

APPENDIX B

OPEN ITEMS

The report includes several open items that could not be resolved during the inspection, or for which the team needs additional information to reach its conclusions. NEP is requested to submit a written response, as described in the enclosed report, for the open items given below.

- Open Item 95-01-01, Section 3.4.2.2(2), "Pilgrim Cycle 11," requested the current shutdown margin value at the most reactive time in Cycle 11.

- Open Item 95-01-02, Section 3.4.3.4, "Quadrant-Symmetric Fuel Failures," requested that the NRC be notified when the failure mechanism was identified.

- Open Item 95-01-03, Section 3.4.5.1, "Rod Bowing," requested the correlation of the measurements of rod bow.

- Open Item 95-01-04, Section 3.4.6.2(1), "GESTRM," requested a response to the burnup extension issue.

- Open Item 95-01-05, Section 3.6.2, "Safety Classification," requested a functional evaluation for fuel bundle parts.

- Open Item 95-01-06, Section 3.6.9.1(1), "Metallurgical Laboratory," requested that the NRC be notified when all corrective actions for CAR 95-12 are complete.

APPENDIX C

PERSONS CONTACTED

The U.S. Nuclear Regulatory Commission staff participating in the inspection of General Electric Nuclear Energy activities conducted at the Nuclear Energy Production facilities in Wilmington, North Carolina and the Nuclear Energy Production personnel contacted during this inspection are listed below and designated as follows: a bullet (•) indicates that person attended the entrance meetings; a dagger (†) indicates that person attended the interim exit meetings.

Reload Core Design, Safety Analysis, and Licensing Processes:

August 14, 1995, Entrance Meeting
August 24, 1995, Interim Exit Meeting

Nuclear Energy Production:

•		Akerlund, S.O.	Principal Engineer, Operating Fuel Performance/Support
	†	Alzaben, A.F.	Technical Program Manager, Nuclear Fuel Americas
•		Armijo, J.S.	General Manager, Nuclear Fuel
•	†	Babb, S.J.	Manager, Components Quality Assurance, Nuclear Quality Assurance (NQA)
•		Baka, G.M.	Technical Program Manager, Nuclear Fuel Americas
•	†	Batchlor, C.D.	Manager, Channels Product Line
•	†	Baumgartner, J.A.	Fuel Project Manager, Nuclear Fuel Americas
•		Bolger, F.T.	Transient Analysis Process Leader, Design Process Improvement (DPI)
	†	Bowman, J.S.	Team Member, DPI
•		Brayman, K.W.	Manager, GENE Systems, NQA
	†	Brechtlein, T.D.	Acting Manager, Nuclear Fuel Asia
•	†	Brohaugh, T.R.	Fuel Project Manager, Nuclear Fuel Americas
•	†	Butrovich, R.M.	Fuel Project Manager, Nuclear Fuel Americas
•	†	Congdon, S.P.	Manager, DPI
	†	Currier, J.W.	Manager, Customer Service
•	†	Dix, G.E.	Manager, Development Program, Nuclear Fuel
•	†	Elkins, R.B.	Manager, Mechanical Design
•		Embley, J.L.	Licensing Program Manager, Fuels & Facility Licensing
•	†	Fawks, R.A.	Sr. Engineer, DPI
•	†	Galloway, G.D.	Process Team Leader, DPI
•		Gardner, K.E.	Process Team Leader, DPI
•	†	Gibbs, E.W.	Technical Program Manager, Nuclear Fuel Americas
•		Halls, J.L.	Program Manager, Technology Licensing
	†	Harmon, J.L.	Manager, New Product Introduction & Value Engineering
•		Hoffman, P.K.	Manager, GENE Audits
•		Hauser, T.M.	Manager, Environmental Health & Safety and Nuclear Quality Assurance (EH&S/NQA)
	†	Hull, G.R.	Fuel Project Manager, Nuclear Fuel Americas
•		Jackson, R.O.	Process Team Leader, DPI
•	†	Kingston, R.E.	Fuel Project Manager, Nuclear Fuel Americas
•	†	Klapproth, J.F.	Manager, Fuels and Facility Licensing, EH&S/NQA
•		Marlowe, M.O.	Manager, Fuel Materials Programs, Nuclear Fuel

•	McCaughey, D.A.	Manager, Fuel Quality
†	Mills, V.W.	Design Automation Project Manager, DPI
•	Milmoe, C.J.	Legal Counsel
•	Nichols, K.W.	Acting Manager, Manufacturing Facilities
•	Pogosian, A.	Production Manager, Control Rods
†	Potts, G.A.	Manager, Operating Fuel Performance/Support
•	Rand, R.A.	Principal Engineer, Operating Fuel Performance/Support
• †	Rash, J.L.	Licensing Program Manager, Fuels & Facility Licensing
•	Reda, C.A.	Manager, Information Management Systems
†	Russell, W.E.	Technical Program Manager, Nuclear Fuel Americas
•	Sependa, W.J.	Manager, Chemical Product Line
†	Serell, D.C.	Technical Program Manager, Nuclear Fuel Americas
•	Sick, P.W.	Manager, NQA
• †	Smith, C.W.	Sr. Licensing Engineer, Fuels & Facility Licensing
•	Soulis, R.E.	Acting Manager, Nuclear Fuel Asia
†	Stachowski, R.E.	Nuclear Methods Project Manager, DPI
• †	Stepp, M.R.	Manager, Nuclear Fuel Europe
• †	Stier, D.P.	Technical Program Manager, Nuclear Fuel Americas
• †	Stirn, R.C.	Manager, New Product Introduction
†	Tuttle, J.L.	Project Manager, Fuel Engineering QA
•	Wei, P.	Principal Engineer, DPI
•	Wileman, J.T.	Technical Program Manager, Nuclear Fuel Americas
†	Williams, R.D.	Fuel Project Manager, Nuclear Fuel Americas

U.S. Nuclear Regulatory Commission:

†	Beyer, C.E.	Neutronics Specialist, Battelle-Pacific Northwest Laboratory
†	Brewer, D.H.	Metallurgical Engineer, Division of Inspection and Support Programs (DISP), Special Inspection Branch (PSIB), Vendor Inspection Section (VIS)
•	Carew, Dr. J.F.	Physicist, Brookhaven National Laboratory
•	Grow, R.L.	Neutronics Specialist, Parámeter, Inc.
• †	Kendrick, E.D.	Reactor Engineer, Division of Systems Safety and Analysis (DSSA), Reactor Systems Branch (SRXB)
•	Lacy, P.S.	Neutronics Specialist, Parámeter, Inc.
• †	Matthews, S.M.	Quality Assurance Specialist, DISP/PSIB/VIS
•	Neogy, Dr. P.	Physicist, Brookhaven National Laboratory
†	Phillips, L.E.	Reactor Engineer, DSSA/SRXB
†	Wu, Dr. S.L.	Reactor Engineer, DSSA/SRXB

APPENDIX C Continued

Fuel Designs and Fuel Production:

August 21, 1995, Entrance Meeting
August 25, 1995, Interim Exit Meeting

Nuclear Energy Production:

•	Armstrong, R.P.	QA Engineer, Fuel Quality, NQA
•	Babb, S.J.	Manager, Components Quality Assurance, NQA
•	Baldwin, J.W.	Quality Auditor, Quality Audits & Programs, NQA
• †	Baumgartner, J.A.	Fuel Project Manager, Nuclear Fuel Americas
•	Bianchi, R.R.	Quality Engineer, Fuel Quality, NQA
•	Brayman, K.W.	Manager, GENE Systems, NQA
•	Brechtlein, T.D.	Acting Manager, Nuclear Fuel Asia
•	Bradberry, J.H.	Leader, FMO Regulatory Team, Chemical Product Line
•	Brown, D.W.	Program Manager, Environmental Programs Team
†	Butrovich, R.M.	Fuel Project Manager, Nuclear Fuel Americas
•	Calcaterra, R.F.	Sr. Engineer, Purchased Material QC, NQA
†	Congdon, S.P.	Manager, DPI
•	Currier, J.W.	Manager, Customer Service
•	Davis, T.C.	Specialist, Organization Effectiveness
•	Downs, J.M.	Quality Auditor, Quality Audits & Programs, NQA
•	Dowker, D.K.	Fuel Support Team Leader
•	Elkins, R.B.	Manager, Mechanical Design
†	Embley, J.L.	Licensing Program Manager, Fuels & Facility Licensing
•	English, D.G.	Quality Auditor, Quality Audits & Programs, NQA
•	Farella, M.	Sr. Manufacturing Engineer, Channels Product Line
•	Fawks, R.A.	Sr. Engineer, DPI
†	Frederickson, C.D.	Project Manager-GE12, New Product Introduction
•	Gallerani, J.J.	Manager, Sourcing/Purchased Material QC & Shipping/Traffic
•	Haughton, R.A.	Team Leader, Rod Fabrication
• †	Hauser, T.M.	Manager, EH&S/NQA
•	Henderson, M.L.	Sr. Specialist, Fabrication Product Line
†	Hull, G.R.	Fuel Project Manager, Nuclear Fuel Americas
†	Kaiser, B.J.	Manager, Fabrication Product Line
•	Keenan, R.J.	Program Manager, Compliance Auditing
•	Kipp, C.P.	General Manager, Nuclear Energy Production
• †	Klapproth, J.F.	Manager, Fuels and Facility Licensing, EH&S/NQA
•	Lamb, M.C.	Team Leader, Powder Prep and Pack
•	Laufer, S.C.	Manager, Software QA & New Product Introduction
•	Landry, J.D.	Leader, Technology Team, New Product Introduction and Value Engineering
•	Lion, E.A.	Quality Engineer, Fuel Quality, NQA
•	McCaughey, D.A.	Manager, Fuel Quality
•	McGowan, R.R.	Leader, URLS/Waste Treatment Team
•	Milmoe, C.J.	Legal Counsel
•	Mobley, E.R.	Quality Control Specialist, Fuel Quality, NQA
•	Nunalee, J.J.	Quality Auditor, Quality Audits & Programs, NQA
•	Ogden, W.	Manager, Manufacturing Facilities
•	Palmer, E.E.	Project Engineer, Channels Product Line
• †	Pensinger, D.L.	Manager, QA Programs, Europe, NQA

•	Plotycia, G.D.	Manager, Materials Services Product Line
•	Pope, D.D.	Quality Auditor, Quality Audits and Programs, NQA
• †	Potts, G.A.	Manager, Operating Fuel Performance/Support
•	Quintana, L.M.	Manager, Services Components QA, NQA
†	Rash, J.L.	Licensing Program Manager, Fuels & Facility Licensing
•	Reda, C.A.	Manager, Information Management Systems
•	Reda, R.J.	Manufacturing Technologist, Fabrication Product Line
•	Rochelle, D.L.	Quality Auditor, Quality Audits & Programs, NQA
•	Rochelle, S.J.	Specialist, Fuel Release, Mechanical Design
• †	Schardt, J.F.	Manager, Tubing Product Line
•	Scott, A.W.	Quality Auditor, Quality Audits & Programs, NQA
•	Selby, S.E.	Leader, UO_2 Production Line
•	Sependa, W.J.	Manager, Chemical Product Line
†	Serell, D.C.	Technical Project Manager, Nuclear Fuel Americas
•	Sick, P.W.	Manager, NQA
• †	Smith, C.W.	Sr. Licensing Engineer, Fuels & Facility Licensing
•	Smith, G.H.	Leader, FMO Maintenance Support Team
•	Smith, M.W.	Nuclear Engineer, DPI
•	Snell, D.H.	Manager, Control Rods Product Line
†	Stepp, M.R.	Manager, Nuclear Fuel Europe
• †	Stirn, R.C.	Manager, New Product Introduction
•	Strickler, H.R.	Manager, Environmental Protection & Industrial Safety
•	Sullivan, A.E.	Sr. Program Manager, Order To Remittance
•	Sweet, F.W.	Quality Auditor, Quality Audits & Programs, NQA
•	Tarrer, C.C.	Leader, Configuration Management
†	Tuttle, J.L.	Project Manager, Fuel Engineering QA
•	Theriault, K.M.	Leader, Uranium Recovery/Process Team, Chemical Product Line
•	Williams, W.K.	Quality Control Specialist, Fuel Quality, NQA

U.S. Nuclear Regulatory Commission:

•	Brewer, D.H.	Metallurgical Engineer, DISP/PSIB/VIS
• †	Beyer, C.E.	Neutronics Specialist, Battelle-Pacific Northwest Laboratory
•	Brite, D.W.	Sr. Research Engineer, Battelle-Pacific Northwest Laboratory
•	Czajkowski, C.J.	Sr. Research Engineer, Brookhaven National Laboratory
• †	Kendrick, E.D.	Reactor Engineer, DSSA/SRXB
• †	Matthews, S.M.	Quality Assurance Specialist, DISP/PSIB/VIS
†	Phillips, L.E.	Chief, DSSA/SRXB
•	Schafer, A.C.	Chemical Engineer, Parámeter, Inc.
•	Thompson, J.W.	Reactor Engineer, Office for Analysis and Evaluation of Operational Data (AEOD), Diagnostic Evaluation and Incident Investigation Branch (DEIIB)
• †	Wu, Dr. S.-L.	Reactor Engineer, DSSA/SRXB

Exit Meeting, Nuclear Energy Production - September 1, 1995

Nuclear Energy Production:

Armijo, J.S.	General Manager, Nuclear Fuel
Babb, S.J.	Manager, Fuel Components QA
Baker, W.L.	Manager, Quality Audits & Programs
Brechtlein, T.D.	Acting Manager, Nuclear Fuel Asia
Congdon, S.P.	Manager, DPI
Creech, M.G.	Specialist, Shipping & Traffic
Currier, J.W.	Manager, Customer Service
Dowker, D.K.	Team Leader, Environmental Process Team
Davis, T.C.	Specialist, Organization Effectiveness
Diersing, M.L.	Team Leader, Ceramic Production, Fabrication Product Line
Downs, J.M.	Quality Auditor, Quality Audits & Programs, NQA
Elkins, R.B.	Manager, Mechanical Design
English, D.G.	Quality Auditor, Quality Audits & Programs, NQA
Fawks, R.A.	Sr. Engineer, DPI
Guilliams, B.S.	Sr. Program Manager, Information Management Systems
Hanvelt, R.A.	Manager, Nuclear Fuel Americas
Hauser, T.M.	Manager, EH&S/NQA
Haughton, R.A.	Team Leader, Rod Fabrication
Hoffman, P.K.	Manager, GENE Audits
Howard, D.C.	Acting Manager, Channels Product Line
Hull, G.R.	Fuel Project Manager, Nuclear Fuel Americas
Kaiser, B.J.	Manager, Fabrication Product Line
Kingston, R.E.	Fuel Project Manager, Nuclear Fuel Americas
Kipp, C.P.	General Manager, Nuclear Energy Production
Klapproth, J.F.	Manager, Fuel and Facility Licensing, EH&S/NQA
Laing, C.F.	Manager, Purchased Material QC
Lamb, M.C.	Team Leader, Powder Prep and Process
McCaughey, D.A.	Manager, Fuel Quality
McGowan, R.R.	Leader, URLS/Waste Treatment Team
Mobley, E.R.	Quality Control Specialist, Fuel Quality, NQA
Nixon, C.L.	Material Control and Accountability (MC&A) Operator, Emergency Preparedness and MC&A, Fuel and Facility Licensing
Nunalee, J.T.	Quality Auditor, Quality Audits & Programs, NQA
Ogden, W.	Manager, Manufacturing Facilities
Pace, R.C.	Manager, Chemet Lab
Parnell, R.I.	Principal Engineer, Metallurgy, Chemet Lab
Pinckney, C.C.	Quality Support Coordinator
Plotycia, G.D.	Manager, Materials Services Product Line
Pope, D.D.	Quality Auditor, Quality Audits & Programs, NQA
Quintana, L.M.	Manager, Services Components QA, NQA
Reda, C.A.	Manager, Information Management Systems
Reda, R.J.	Manufacturing Technologist, Fabrication Product Line
Scott, A.W.	Quality Auditor, Quality Audits & Programs, NQA
Schardt, J.F.	Manager, Tubing Product Line
Selby, S.E.	Leader, UO_2 Production Line
Sependa, W.J.	Manager, Chemical Product Line

APPENDIX C Continued

Sick, P.W.	Manager, NQA
Smith, C.W.	Sr. Licensing Engineer, Fuels and Facility Licensing
Snell, D.H.	Manager, Control Rods Product Line
Stepp, M.R.	Manager, Nuclear Fuel Europe
Stirn, R.C.	Manager, New Product Introduction
Sweet, F.W.	Quality Auditor, Quality Audits & Programs, NQA
Theriault, K.M.	Leader, Uranium Recovery/Process Team, Chemical Product Line
Torres, R.L.	Auditor, Compliance Auditing
Tuttle, J.L.	Project Manager, Fuel Engineering QA
Walker, H.F.	Leader, Shipping & Traffic Team
Williams, W.K.	Quality Control Specialist, Fuel Quality, NQA

U.S. Nuclear Regulatory Commission:

Cwalina, G.C.	Chief, DISP/PSIB/VIS
Brewer, D.H.	Metallurgical Engineer, DISP/PSIB/VIS
Matthews, S.M.	Quality Assurance Specialist, DISP/PSIB/VIS
Schafer, A.C.	Chemical Engineer, Parámeter, Inc.
Thompson, J.W.	Reactor Engineer, AEOD/DEIIB

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

April 8, 1996

Mr. L. Charles Spriggs
Vice President, Product Assurance
PROMATEC, Inc.
P.O. Box 309
Cypress, TX 77429

SUBJECT: NRC INSPECTION NO. 99901292/95-01

Dear Mr. Spriggs:

This letter transmits the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Promatec Incorporated facilities in Cypress, Texas by this office on December 6-8, 1995.

The inspection was conducted to evaluate Promatec's quality assurance program as implemented in the process for providing fire barriers penetration seals to the nuclear industry. Specific areas of review included, product design and design change controls, product testing and installation, procurement of materials, and document controls. In addition, the inspectors reviewed Promatec's program for implementing Part 21, "Reporting of Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR Part 21).

During the inspection the inspectors examined procedures and representative records, for activities such as receipt inspections and product dedication, and conducted discussions and interviews with personnel. The inspectors' observations were discussed with the staff at the conclusion of the inspection, on December 8, 1995.

Based upon its observations and evaluation, the team concluded that the silicone-based nuclear power plant fire barrier applications designed, tested, and installed by Promatec are commercial grade items, as defined in 10 CFR 21.3. Although the silicone products supplied for nuclear plant fire barrier applications are not subject to the NRC's Part 21 regulation, it appears that Promatec established and implemented a system which meets the intent of certain 10 CFR Part 21 requirements. Areas examined during the inspection and our conclusions are discussed in the enclosed inspection report.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practices," a copy of this letter and the enclosed inspection report will be placed in the NRC Public Document Room. If there are any questions concerning this inspection we will be pleased to discuss them with you. No response to this letter is required.

Sincerely,

Donald P. Norkin, Acting Chief
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Docket No. 99901292

Enclosure: Inspection Report No. 99901292/95-01

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

ORGANIZATION:	Promatec, Inc. P.O. Box 309 Cypress, TX 77429
REPORT NO:	99901292/95-01
ORGANIZATIONAL CONTACT:	Mr. L. Charles Spriggs, Vice President
NUCLEAR INDUSTRY ACTIVITY:	Designed, tested and installed rated fire barrier penetration seal assemblies in nuclear power electrical generating stations. The assemblies combined commercial grade silicone products and various proprietary additives developed to meet specific fire barriers penetration seal fire test criteria.
INSPECTION CONDUCTED:	December 6-8, 1995

LEAD INSPECTOR: _________________________ 4/5/96
Peter S. Koltay, Team Leader — Date
Special Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

OTHER INSPECTORS: Armajit Singh, NRR
Christofer Bajwa, NRR
T.L. Tinkel, Brookhaven National Laboratory

REVIEWED: Gregory C. Cwalina _________________________ 4/8/96
Gregory C. Cwalina, Chief — Date
Vendor Inspection Section
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

APPROVED: _________________________ 4-9-96
Donald P. Norkin, Acting Chief — Date
Special Inspection Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

Enclosure

1.0 INSPECTION SUMMARY

Based on the scope of this inspection, the team determined that Promatec, Inc. does not manufacture and supply any "basic components" as defined in Part 21, "Reporting of Defects and Noncompliance," of Title 10 of the Code of Federal Regulations (10 CFR Part 21). While some customers may contractually impose the requirements of 10 CFR Part 21 on Promatec, Inc. for their fire barrier penetration seal assemblies supplied for use in nuclear power plants, the inspectors verified that the products did not meet the definition of basic components as defined in 10 CFR 21.3. However, the inspectors found that Promatec has established and implemented a program which meets the intent of certain sections of 10 CFR Part 21, such as 10 CFR 21.21(b). Promatec, Inc. personnel stated to the team that they will continue to implement the existing program for meeting the intent of 10 CFR Part 21.

In response to NRC concerns regarding the fire endurance qualification testing of cable slot type fire barrier penetration seals at the Watts Bar Nuclear Plant, additional testing on site specific penetration configurations were conducted. The test assemblies were constructed by the penetration contractor, Peak Seals (Promatec). The tests were performed on October 22, 1995, at Omega Point Laboratories, Elmenford, Texas. All test assemblies met the acceptance criteria (section 3.5). Details of the test are described in Appendix A to this report.

2.0 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC staff inspection of this vendor; thus, there were no previous findings.

3.0 INSPECTION FINDINGS AND OBSERVATIONS

3.1 Entrance Meeting

The entrance meeting was held on Wednesday, December 6, 1995. At this meeting, Mr. Koltay, the NRC inspection team leader, discussed the scope of the inspection with the Promatec staff.

3.2 Background and Description of Facilities

Following the Browns Ferry nuclear power plant fire, increased emphasis was placed on fire protection at commercial nuclear power plants. A new line of fire barrier penetration seals, using silicone based foam and elastomer materials, were developed to seal openings created in safety related structures designed to accommodate piping, cables, and ventilation duct work.

Promatec was formed in September 1983 from the fire safe division of B&B Insulation company. Subsequently, Promatec also obtained the rights to the results of fire tests of fire barrier penetration seals of several companies including TechSil, ICM and Peak Seals.

Promatec provides a full line of fire barrier penetration seal services including engineering, supply of materials, training, and inspections. The company's quality assurance program satisfies the requirements of 10 CFR Part 50, Appendix B and ANSI N45.2. Promatec has performed fire barrier penetration seal work at over 40 domestic nuclear utilities.

3.3 Penetration Seal Configurations and Sealing Materials

The Promatec corporate test procedure index, dated July 10, 1995, indicates that Promatec has tested and qualified dozens of fire barrier penetration seal design configurations between 1976 and 1994.

The penetration seal design configurations vary depending on many factors including the type of fire barrier construction, the type of penetration, and the size of the opening being sealed. Examples of the various kinds of penetration seal configurations include:

- Penetrations for electrical cables
- Penetrations for electrical cable pipe conduits
- Penetrations for electrical cable trays
- Penetrations for piping and ducts

Raw materials, silicone foam, silicone elastomers, and silicone gel, are purchased from the Dow Corning and the General Electric corporations. Generally, Promatec reformulates the raw materials using proprietary ingredients and sells them under various names such as PROMAFLEX (FIREFLEX), LDSE, HDSE, RADFLEX, and others.

3.3.1 Material Receipt Inspection

The inspectors reviewed the receipt inspection and Quality Control (QC) procedures. Raw materials are receipt inspected in accordance with detailed procedures and are clearly marked with QC tags which define the status of the materials. Materials that are used in products supplied to domestic commercial nuclear facilities are tested by Promatec or by an independent laboratory to verify manufacturing specifications, prior to use. For example, elastomer and silicone seal materials are tested by Promatec, while filler materials are tested by independent laboratories to assure that the materials are acceptable according to Promatec's specifications. These independent verifications ensure that the fillers are of the correct purity and are free of contaminants. QC involvement is required in all repacking operations and batching processes so that inadvertent use of untested material will be prevented.

Overall the inspectors found that Promatec has a well maintained material control program. All stored material is clearly marked with its inspection status. Items are identified with a sticker which contains the receiving information for the material. Generally, all material is identified with the receiving report number, date of receipt, purchase order (PO) number, lot number when applicable, and shelflife or expiration date.

Proportioning records and procedures for the mixing and formulation of Promatec products, were reviewed by the inspectors. The inspectors noted that complete and detailed batching records are established and the identification of materials is maintained throughout the operation. Additionally, the procedures are established and very well implemented for maintaining shelf life of stored materials and assignment of shelf life to repackaged and blended products.

The inspectors also reviewed the dedication activities which are defined in procedure QAP-0001, "Procurement Control," Issue J, dated March 28, 1994. The inspection and test requirements are prescribed for material specifications. A review of the material specifications and inspection instruction against test results received for a sample of items found that appropriate test results were provided. The inspectors concluded that the vendor has established an adequate test and inspection program. The inspectors did not identify any deficiencies in this area of the inspection.

3.3.2 Procurement Document Control

The inspectors reviewed a sample of Promatec customer procurement documents including some recent customer Purchase Orders (POs). Some of Promatec's customers impose the provisions of 10 CFR Part 21 on individual POs, while other customers do not invoke 10 CFR Part 21 at all. Requisitions and POs are reviewed by QA prior to issuance. A Customer Traveler form is prepared which is used to record all vital information, from the receipt inspection number, to the classification of the order as safety or non-safety related, to the final shipping information. This sheet is attached to the PO. Also attached to the POs is a purchase specification, which provides information required to assure the quality of the materials to be provided. All POs reviewed by the inspectors were in compliance with the established procedures. Where required, POs are placed with suppliers listed on the approved vendor list (AVL). Promatec's AVL contains both Appendix B and commercial grade suppliers. The list clearly distinguishes how the supplier was qualified and will be used and whether there is additional testing required to complete dedication, special instructions, etc. The inspectors concluded that the vendor has implemented a good program in the area of procurement.

3.3.3 Fabrication/Assembly and Special Processes

The procedures for producing Promatec products address the batching of foam and elastomeric materials. The inspectors also noted that there are no special processes performed by Promatec on the materials they supply. The batching of formulations is conducted by specified individuals and is overseen by a QC inspector. The batching is performed in accordance with the instructions provided in proportioning procedures and is recorded on the batching record. Scales used to weigh the materials are confirmed to be accurate by calibration by an outside supplier. The inspectors did not identify any discrepancies in this area of inspection.

3.3.4 Calibration

The inspectors reviewed the calibration program at Promatec. The weight scale calibrations are subcontracted to a survey qualified supplier. The vendor provided adequate evidence of traceability, accuracy and certification of its supplier. The weight scales were found to be uniquely identified and had a calibration sticker indicating the calibration date, due date for next calibration and the identification of the person who did the calibration. The calibration records provided the status, location, calibration frequency, and as-found and as-left data for each scale. The inspectors concluded that the vendor has an adequate calibration program.

3.3.5 Document Control

The vendor has implemented a documentation control program which meets its procedure requirements and QA program commitments. The procedures were found to provide sufficient detail and direction. Personnel are assigned controlled copies of applicable procedures; the controlled procedures issued were the correct revision and stamped with a control number. Drawings are well maintained and drawing files contained previous revisions to drawings, where applicable. The inspectors did not identify any deficiencies in this area of the inspection.

3.4 Quality Assurance and Design Control

The inspectors reviewed the vendor's Quality Assurance Program (QAP) document QAM 20188, Issue D, dated June 28, 1995, which establishes those planned and systematic actions necessary to assure that products provided by the company will perform satisfactorily in accordance with their intended purpose. In addition, that such products, including services associated therewith, comply with customer specifications and all corresponding regulatory requirements. The Quality Assurance Manual incorporates the applicable requirements of ANSI/ASME NQA-1-1989 EDITION (ANSI N45.2). The vendor's QA program is applied to the following; (1) penetration seals and protective coatings for fire, pressure, smoke and hot gas, hydrostatic, and/or thermal protection; (2) fire protection systems for separation of electrical trains, boundary separations including flexible blankets and rigid barriers; (3) thermal insulation specified for use with safety-related equipment; and (4) activities and services affecting quality, including design, testing, materials, manufacturing, installation, inspection, and training. The inspector concluded that the vendor's QA program and manual establishes the organizational structure, the responsibilities of individuals, and departments performing activities governed by this program. The inspectors did not identify any discrepancies in this area of inspection.

3.4.1 Methodology for Qualifying Penetration Seal Configurations

Promatec qualifies the design of a penetration seal configuration using one of two methods.

a. The first method consists of subjecting the penetration seal design configuration to a fire endurance test. The results of successful testing are used to qualify the particular parameters of the design. This method was used extensively during the development of the fire barriers penetration seals. Promatec maintains a library of fire penetration test results on file.

b. The second method consists of performing an engineering analysis for the penetration seal design configuration requiring qualification in the field and establishing a correlation between the field penetration and tested penetration configurations on file.

The engineering analysis method for qualifying penetration seal configurations permits Promatec to use a combination of engineering judgment and results from multiple prior tests to qualify new or modified design configurations without performing additional testing.

Engineering analysis for qualification consists of analyzing reports of pertinent past tests performed by or available to Promatec, and identifying qualitative and quantitative information for selected design parameters that represent the candidate design configuration and are covered by prior testing.

The inspectors noted that the Promatec methodology for qualifying configurations using engineering analysis is not documented except for the notes on drawings. Based on a review of drawings and discussions with the Operation's Manager and the Product Assurance Manager, Promatec's past practice and methodology were reviewed. The team identified and verified the valid range of the following design parameters that can be used by Promatec for qualifying candidate penetration seal configurations:

a. Primary seal material type. The candidate seal material must be the same as the tested seal material. In the event the particular seal material is available from more than one vendor, a comparative test of the two seal materials must be performed.

b. Primary seal material depth. The candidate seal material depth must be equal to or greater than the tested seal material depth.

c. Fire barrier orientation. Tested floor orientations can be used to qualify a candidate floor or wall orientation. Tested wall orientations can only be used to qualify candidate wall orientations.

d. Service temperature. The candidate design service temperatures must be equal to or less than the service temperature of the seal material.

e. Radiation resistance. The required radiation exposure of the candidate configuration must be equal to or less than the radiation exposure of the seal material.

f. Maximum penetrant movement (lateral and/or axial). The allowable motion of the candidate configuration must be equal to or less than the allowable motion of the seal material.

g. Maximum opening size. The maximum opening size of the candidate configuration must be equal to or less than the maximum opening size of the tested configuration.

h. Maximum free area (also referred to as maximum unsupported area). The maximum free area of the candidate configuration must be equal to or less than the maximum free area of the tested configuration.

i. Minimum clearance of penetrants. The minimum clearance between penetrants or between penetrants and the substrate (fire barrier boundary) in the candidate configuration must be equal to or greater than the minimum clearances in the tested configuration.

j. Maximum annular area (for pipe/sleeve designs). The maximum annular area of the candidate configuration must be equal to or less than the maximum annular area of the tested configuration.

k. Penetrating object material type. The penetrating object material type of the candidate configuration must be the same as the tested configuration.

l. Fire barrier rating. The fire barrier rating of the candidate penetration configuration shall be equal to or less than the fire barrier rating of the tested configuration.

m. Cable way fill. The percent fill of the candidate penetration configuration must be less than or equal to the percent fill of the tested configuration. The ignitability of the insulation for the candidate configuration must be equal to or less than that in the tested configuration.

n. Damming material. The same type of damming material and thickness used on the tested configuration must be used on the candidate penetration if the tested configuration did not use a dam.)

o. Penetrant item wall thickness. The wall thickness of the candidate configuration must be equal to or less than the wall thickness of the tested configuration.

p. Penetrant item size. The size of the candidate configuration must be equal to or less than the size of the tested configuration.

Since engineering analysis is an important method for qualifying Promatec penetration seal configurations, the inspection team recommended that consideration should be given to formally documenting and controlling the engineering analysis method in a Promatec engineering standard or procedure. The acceptance criteria and limitations for each of the selected design parameters along with the technical basis or technical justification for the analysis method should be included.

The inspectors verified that parameters are limited by tested configurations; however, there is no limitation on the length and width for the parameter that establishes the acceptable range in the maximum free area parameter assessment. Promatec management agreed to review this weakness and establish appropriate limitations as needed.

The inspector assessed the quality of completed engineering analyses by reviewing the following resultant Fire Barrier Seal Design Test Report:

> Test report CTP 1063 dated 21 January 1985 indicated that the unexposed side temperature limits defined by ASTM E-814 of 325 °F plus initial temperature (325 °F+ 84 °F = 409 °F) were exceeded during the 3 hour test for a number of penetrations included in the test. The high temperatures occurred on thermocouples located at the unexposed side at the interface of the sealant material and the penetrating items.
>
> The inspectors noted that portions of this test were used as the qualification basis for certain site specific penetration configurations at a power plant. Additionally, the power plant penetration seal specification (Rev. 1 dated 3 April 1992) invokes a maximum temperature of 325 °F on the unexposed side surface and penetrating items. All the above noted penetrations, as well as a number of others in this test, failed to meet this requirement. This matter was discussed with the Promatec Product Assurance Manager. The inspectors verified that the Promatec fire qualification test were conducted in accordance with ANI/AMERP for which the limiting end point temperatures on the unexposed side of the penetrations are less restrictive than those of ASTM E-184 in that the ANI test does not specify the temperature of the penetrating item on the unexposed side. The ANI acceptance criteria, that meets regulatory expectations, were satisfied by the test. The requirements for the power plant installation were appropriately adjusted.

In conclusion, the review of actual design configurations did not identify any design issues. However, the implementation of the engineering analyses to qualify tested fire barrier penetration seals on file, for installation into fire barrier penetrations of various configuration, partially relies on informal engineering judgment. This could be improved by formally documenting and controlling such aspects of the engineering analysis methods in a engineering standard or procedure.

3.5 Fire Endurance Test of Cable Slot Penetration Seals

On October 22, 1995, the concrete test slab, containing the 14 cable slot penetration seal test assemblies was subjected to a 3 hour fire endurance test which followed the ASTM E-119 standard time-temperature curve and a fog nozzle hose stream test. The acceptance criteria of IEEE 634-1978, "Standard Cable Penetration Fire Stop Qualification Test," was used to evaluate the thermal/fire resistive performance of the test assemblies. This criteria requires the test assemblies to: 1) withstand the fire endurance test without the passage of flame or hot gases hot enough to ignite cables on its unexposed side; 2) heat transmission through the penetration seal shall not raise the temperature on its unexposed surface above 700 °F; and 3) not allow water to be projected through the penetration seal during the hose stream test.

All test assemblies, met the IEEE 634 acceptance criteria. The maximum unexposed cable/seal interface temperatures ranged from 323 °F for instrument and control cables and 601 °F for power cables.

The staff, based on the results of the applicant's supplemental cable slot fire endurance tests, finds those "as-built" penetrations which have been installed in accordance with WBN cable slot penetration seal design details A4, H1, L1, and M4 and that are bounded by the tested cable fill (thermal mass of copper conductors) conditions will provide an equivalent level of fire safety to those which were tested and, therefore, they are acceptable.

3.6 Exit Meeting

The exit meeting was conducted on December 8, 1995, by the NRC team leader prior to the team's departure from the Promatec facility.

APPENDIX A

TRIP REPORT

TRIP DATE: October 15-22, 1995

REVIEWER: Patrick Madden, Senior Fire Protection Engineer
Fire Protection Section
Plant Systems Branch
Division of Systems Safety and Analysis

LOCATION: Omega Point Laboratory
San Antonio, Texas

APPLICANT: Tennessee Valley Authority (TVA)

SUBJECT: CABLE SLOT 3-HOUR PENETRATION SEAL TEST SPECIMEN CONSTRUCTION AND FIRE ENDURANCE TEST

1.0 APPLICANT/CONTRACTOR PERSONNEL CONTACTED:

Joseph A. Lisa,	TVA (QA)
J. J. Pierce,	TVA (ENGINEERING)
Charles Spriggs,	PROMATEC/Peak Seals
Michael Jordan,	PROMATEC/Peak Seals
DeDe Smithwick,	PROMATEC/Peak Seals
James Grancio,	PROMATEC/Peak Seals
Mike Murphy,	PROMATEC/Peak Seals

2.0 DESCRIPTION OF TEST ASSEMBLY

The test assembly consists of a 8' x 13' by 12-inch thick concrete test slab with 14 5" x 20" cable slot blockouts. On one half of the slab, 8 cable tray slots (A1 through A8) are arranged in two parallel columns with 4 cable slots in each column. The slots in each column are separated by a 7" wide concrete mullion and a 6" concrete mullion exists between the cable slot ends between the columns. The two cable tray slot columns were constructed so that they were maintained at least 24" away from the edge of the test slab. The remaining 6 cable tray slots (B1 through B6) are located on the second half of the slab and are arranged in two parallel columns with 3 slots in each column. The columns are separated by a 6" wide concrete mullion and each cable slot within each column is separated by a 7" wide concrete mullion. These two cable slot columns were constructed so that the edges of cable slots columns were maintained at least 24" away from the edge of the test slab. The following summarizes the cable fill of each cable slot test specimen:

- Penetration Seal (PS) Test Specimen A1 - single layer of 4/c-#16 (43 cables)

- PS Test Specimen A2 - 100% visual fill of 4/c-#16 (230 cables)

- PS Test Specimen A3 - 100% visual fill of 4/c-#16 (230 cables)

- PS Test Specimen A4 - 50% visual fill of 4/c-#16 (150 cables)

- PS Test Specimen A5 - 50% visual fill of 4/c-#16 (150 cables)

- PS Test Specimen A6 - single layer of 4/c-#16 (43 cables)

- PS Test Specimen A7 - blank spare (no cable fill)

- PS Test Specimen A8 - blank spare (no cable fill)

- PS Test Specimen B1 - contains 300MCM (14 cables), 2/0-600v (3 cables), 2/0-8Kv (9 cables), 4/0 (2 cables), #2 (2 cables), #6 (4 cables), 3/c-#10 PXMJ (8 cables), and 3/c-#10 CPJJ (3-cables)

- PS Test Specimen B2 - contains 300MCM (9 cables)

- PS Test Specimen B3 - contains 300MCM (7 cables), 2/0-8Kv (3 cables), 4/0 (4 cables), #2 (1 cables), #6 (2 cables), 3/c-#10 PXMJ (10 cables), and 3/c-#10 CPJJ (6-cables)

- PS Test Specimen B4 - contains 300MCM (20 cables), 2/0 (12 cables), 4/0 (4 cables), #2 (2 cables), #6 (4 cables), 3/c-#10 PXMJ (14 cables), and 3/c-#10 CPJJ (6-cables)

- PS Test Specimen B5 - contains 300MCM (9 cables)

- PS Test Specimen B6 - contains 300MCM (7 cables), 2/0-8Kv (3 cables), 4/0 (4 cables), #2 (1 cables), #6 (2 cables), 3/c-#10 PXMJ (10 cables), and 3/c-#10 CPJJ (6-cables)

3.0 PENETRATION SEAL

General Description of Foam Installation

For PS test specimens A1 through A6, each seal was constructed by installing a damming board (Carborundum Fiberfax 1-in thick low density board) on the exposed (fire) surface of the specimen and filling the blockout void with 12-inches of Dow Corning RTV 3-6548 silicone foam and flush with the surface of the concrete. Once the foam has been injected into the blockout void a damming board is installed on the unexposed surface of the slab.

PS test specimens A7 and A8 are spare penetrations with a sleeve that extends 4-inches out away from the wall on each side. On the exposed (fire) side of PS test specimen A7 the damming board is attached to the end of the sleeve and 11-inch foam fill is injected into the cable slot blockout penetrations thus creating a 9" air gap between the damming board and the foam on the unexposed side of the seal. PS test specimen A8 was constructed in the same manner except that the 9" air gap was on the exposed (fire) side of the seal.

On the exposed (fire) side of PS test specimens B1 through B6 a damming board is installed and 11-inch thickness of foam was injected into the blockout to fill the void. On the unexposed side 1-inch thickness of ceramic fiber (Carborundum Durablanket) is installed between the foam seal and the outer damming board on the unexposed surface.

Procedure Review

The following procedures related to the installation of the silicone foam seal test specimens were reviewed:

1. Promatec Procedure OCP-0067, 7/1/91, "Density Verification" (procedure sets a minimum density of 14 lbs/ft^3 as compared to TVA specified density range of 15-30 lbs/ft^3)

2. PCI/ICMS - Tennessee Valley Authority Procedure QC-101, Revision 9 (7/15/95) "QC Inspection Silicone Foam" (procedure establishes the inspection criteria for the inspection of foam samples and installed seals)

3. PCI/ICMS - Tennessee Valley Authority Procedure QC-103, Revision 4 (7/17/95), "Density Measurement Multi-Component Silicone Material" (procedure sets a minimum density range of 15-30 lbs/ft^3. This is consistent with TVA's penetration seal engineering report)

4. PCI/ICMS - Tennessee Valley Authority Production Work Instruction PWI-051, Revision 9 (9/22/94), "Installation of Damming Material" (procedure provides guidance to the craft regarding the installation of damming materials prior to pouring silicone materials)

5. PCI/ICMS - Tennessee Valley Authority Production Work Instruction PWI-052, Revision 7 (4/14/94) "Installation of Silicone Foam" (procedure sets the criteria for inspecting blockouts and damming prior to silicone installation)

6. TVA Maintenance/Addition Instruction MAI-3.6, "Cable Tray and Sleeve Seals," Revision 7 (9/19/95), Unit 0, Appendix E, Sealing Cables in Vertical Sleeves.

Penetration Seal Materials

Damming Materials Used:

Carborundum Fiberfax 1-in thick low density board (density - 15 to 18 lbs/ft^3), Lot Nos. 3053 and 4311, date of manufacturer 3/30/95

Carborundum Durablanket, 1-in thick, density - 6 lbs/ft^3, Lot No. 5117, date of manufacturer 5/19/95

Silicone Foam Material: Dow Corning 3-6548 Silicone Foam

Part A - Lot No. ET55392A (shelf life 10/31/96)

Part B - Lot No. ET065158B (shelf life 10/31/96)

General Methods of Installation

The applicant used MAI-3.6, "Cable Tray and Sleeve Seals," Revision 7 (9/19/95), to install the penetration seal test specimens B-2, B-4, and B-6. The following summarizes the general installation methods used by the craft to install the foam in these penetration seals:

1. Distribute the cables evenly, lift cables and firmly pack a two-inch depth of ceramic fiber (Kaowool or equivalent) around and between each cable at both ends of the sleeve to provide approximately 1/4-inch separation between cables and metal surfaces. (Note: the ceramic fiber extends at least one-inch on each side of the minimum depth of seal material.)

2. Cut a temporary form and install temporary form over bottom end of sleeve.

3. Apply silicone foam around cables and between the cables from the top side of the sleeve. (Note: An initial sealant application is made and allowed to cure. This acts as a base which eliminates excessive leakage of uncured material through the bottom side of the sleeve.)

4. Cut a temporary form and cut three one-inch holes in the form for foam applications. Install the temporary form over the top side of the cable sleeve.

5. Apply silicone foam inside the sleeve through the application holes.

6. Remove the temporary forms and any excess sealing material. (Note: Random surface irregularities or voids less than 1/2-inch in their major dimension are acceptable.)

7. Apply ceramic fiber board cement to the surface of foam and edges of the sleeve and allow to set 3-5 minutes.

8. Install pieces of one-inch ceramic fiber board over each cable sleeve face with a 1/2 inch overlap of the sleeve edges. Press ceramic fiber board firmly onto previously prepared surfaces.

For the remaining penetration seal test specimens (B-2, B-4, B-6, and A-1 through A-8), the applicant used QC-101, QC-103, PWI-051, and PWI-052 to install these penetration seal test specimens. The following summarizes the major quality verification attributes and general installation methods used by the craft to install and QC personnel to inspect these foam penetration seals:

QC-101 - Quality Control Inspection Silicone Foam

1. Permanent forms (dams) shall fit snug of shall be mechanically fastened to the face of the barrier on both openings.

2. Electrical cable trays in floor/wall configurations are to be formed to allow a minimum depth of 12 inches of foam.

3. Electrical cables are spread where possible to allow foam to flow freely around them.

4. Silicone foam samples are prepared in accordance with PWI-052. One of the two samples shall be cut open and examined for the following: (a) foam color shall be dark grey to black; (b) cell structure shall be uniform; (c) foam texture shall be set and firm, and (d) material tear shall be crisp and firm. The second sample shall be trimmed flush with the top of the cup and weighed to the nearest tenth of a gram. (Note: Acceptable density range for silicone foam is 15 to 30 pounds per cubic foot.

5. Key inspection criteria are: (a) seals are neatly trimmed where needed; (b) foam color is dark grey to black; (c) foam cell structure uniform; (d) foam texture is set and firm; (e) penetration fill sufficient; (f) seal conforms with detail shown on customer supplied field data sheet, and (g) permanent forms are properly fitted; and material is acceptable.

QC-103 - Density Measurement Multicomponent Silicone Material

1. Determine the weight and volume of sample cup.

2. Determine weight of sample cup filled with silicone material and calculate density.

PWI-051 - Installation of Damming Material

1. Forming materials be cut and fit snug into or against the barrier to prevent leakage.

2. Where cables protrude through penetration, care should be taken to tightly pack damming materials to prevent excessive leakage.

3. In electrical cable trays, cables will be spread where possible to allow material to flow around and between them. Blanket or bulk fiber material may be used for this purpose.

PWI-052 - Installation of Silicone Foam

1. Distribute a light layer of silicone foam on the form to seal off any potential leaks.

2. Electrical cable trays in floor/wall configurations are to be formed in accordance with a minimum twelve inches of silicone foam.

4.0 CONSTRUCTION ACTIVITIES WITNESSED:

OCTOBER 17, 1995 - OBSERVATIONS

The following construction activities were witnessed:

1. To facilitate the installation of damming materials the test slab (assembly) was configured in a vertical position. Cutting and fitting of the damming boards for the exposed side of the individual penetration test specimens was started and welding of damming board studs to cable slot steel liners was completed on the unexposed side of the test assembly.

2. Filling void spaces between the cables, cables and the cable tray, and the cable tray and the sides of the cable slot blockout (test specimen B-2, B-4, and B-6) with Durablanket per TVA MAI-3.6 and installation of Carborundum Fiberfax damming board.

OCTOBER 18, 1995 - OBSERVATIONS

The following construction activities were witnessed:

1. Cutting and fitting of the damming boards and welding of damming board studs to cable slot steel liners.

2. Installation Carborundum Fiberfax damming board and filling void spaces between the cables in the damming board cable cutouts with Durablanket per PWI-051.

3. Setup of foam gun and mixing air pump and the pre-mixing of silicone foam Part A and B prior to filling the Part A and B injection pump reservoirs.

4. Calibration of the foam density sample cup (verified that the scale used to determined weights was calibrated - calibration date 10/2/95, re-calibration due 3/31/96)

5. Test assembly was placed in a horizontal configuration to facilitate filling the penetration seal test specimens with foam.

OCTOBER 19, 1995 - OBSERVATIONS

The following construction activities were witnessed:

1. The cables in the individual penetration seal test specimens were spread in order to facilitate the flow of foam between the cables.
2. Foam samples were taken to determine its density. Using the PCI/ICMS density method the initial density was 24.1 lbs/ft^3 (average of three

samples). Using the PROMATEC density method a density of 19.64 lbs/ft^3 was achieved.

3. When the slab was placed in the horizontal condition the cables slid from their original position. The cables had to be re-pulled to meet the test plan specification. This resulted in some of the Durablanket damming material to be dislodged. In order to prevent the cables from sliding during seal installation, the cables were secured on the unexposed side of the test assembly to the cable trays with stainless steel tie wire. This was done prior to re-working the dams and filling voids around the cables and the damming boards with Durablanket ceramic fiber material.

4. During the initial pouring/injection of the silicone foam into penetration seal test specimen A-4 (first seal to be filled), the silicone gun experienced a problem with the silicone foam Part B injection nozzle and shutoff valve. This resulted in improper mixing of Part A and B in the mixing chamber and poor foam color quality. Work was stopped to repair the gun.

5. As result the foam gun problems, Penetration Seal A-4 was completely re-worked. The damming and foam was removed from the penetration seal opening. The damming was re-cut and re-installed and the seal re-sealed with silicone foam.

6. As result of the foam gun problems a reverification of foam density was performed three additional times. The final average foam density was 24.5 lbs/ft^3.

<u>OCTOBER 20, 1995 - OBSERVATIONS</u>

The following construction activities were witnessed:

1. Final installation of foam completed, penetration seal test specimens were trimmed to meet specification requirements established by TVA's test plan. At the start of work a foam density measurement was made. The average foam density was 25.6 lbs/ft^3.

2. Installation of engineering thermocouples on unexposed side silicone foam surfaces by laboratory personnel.

3. Installation of Carborundum Fiberfax 1-in thick low density damming board on unexposed side of each penetration seal test specimens.

4. Installation of qualification thermocouples on the individual penetration seal test specimens by the Laboratory personnel. The test assembly was instrumented with a total of 286 thermocouples.

5.0 <u>FIRE ENDURANCE AND HOSE STREAM TEST</u>

DATE OF TEST: October 22, 1995

TEST DURATION: 3-hour (ASTM E119 - standard time-temperature curve)

TEST ACCEPTANCE CRITERIA: IEEE 634-1978, "Standard Cable Penetration Fire Stop Qualification Test"

This criteria requires that: (1) the test specimen withstand the fire endurance test without the passage of flame or gases hot enough to ignite cables on its unexposed side; (2) heat transmission through the penetration seal shall not raise the temperature on its unexposed surface above 700 °F, and (3) not allow water to be projected through the penetration seal during the hose stream test.

FURNACE PRESSURE: 0.01 inches of water

TEST ASSEMBLY ORIENTATION: Horizontal

TYPE OF HOSE STREAM TEST: Fog (30 °F pattern, 75 gpm @ 75 psi); Duration of hose stream application is 2-1/2 minutes and was applied within 10 minutes after the fire endurance test.

INITIAL TEMPERATURE CONDITIONS: 69 °F

TEST OBSERVATIONS:

0:02 Cables on the exposed side fully ignited

1:10 Penetration seal test specimens B3, B5, and B6 emitting smoke through the unexposed side of the seal. Slight smoke streaming discoloration is occurring on the ceramic fiber damming board on the unexposed side of these penetration seals.

1:40 Penetration seal test specimen B4 emitting smoke fairly steady. Smoke streaming discoloration is occurring on certain areas of the unexposed side ceramic fiber damming board.

2:17 All "B" penetration seal test specimens emitting steady smoke through the penetration seals and cable bundles.

RESULTS: (maximum temperatures noted during the test)

a. Test Specimen: A1
Cable Fill: single layer of 4/c-#16 (43 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 112 °F
- Interface between the seal and cable: 205 °F
- Interface between the seal and tray side rail: 160 °F
- On the steel liner of the cable slot: 144 °F
- Under damming board on top of silicone foam: 150 °F

b. Test Specimen: A2
Cable Fill: 100% visual fill of 4/c-#16 (230 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 150 °F
- Interface between the seal and cable: 280 °F
- Interface between the seal and tray side rail: 196 °F
- On the steel liner of the cable slot: 179 °F
- Under damming board on top of silicone foam: 229 °F

c. Test Specimen: A3
Cable Fill: 100% visual fill of 4/c-#16 (230 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 156 °F
- Interface between the seal and cable: 323 °F
- Interface between the seal and tray side rail: 160 °F
- On the steel liner of the cable slot: 151 °F
- Under damming board on top of silicone foam: 263 °F

d. Test Specimen: A4
Cable Fill: 50% visual fill of 4/c-#16 (150 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 150 °F
- Interface between the seal and cable: 264 °F
- Interface between the seal and tray side rail: 137 °F
- On the steel liner of the cable slot: 159 °F
- Under damming board on top of silicone foam: 210 °F

e. Test Specimen: A5
Cable Fill: 50% visual fill of 4/c-#16 (150 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 156 °F
- Interface between the seal and cable: 318 °F
- Interface between the seal and tray side rail: 151 °F
- On the steel liner of the cable slot: 180 °F
- Under damming board on top of silicone foam: 227 °F

f. Test Specimen: A6
Cable Fill: single layer of 4/c-#16 (43 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 123 °F
- Interface between the seal and cable: 243 °F
- Interface between the seal and tray side rail: 135 °F
- On the steel liner of the cable slot: 187 °F
- Under damming board on top of silicone foam: 161 °F

g. Test Specimen: A7
Cable Fill: SPARE
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 155 °F
- On the steel liner of the cable slot: 295 °F
- Under damming board on top of silicone foam material: 311 °F

h. Test Specimen: A8
Cable Fill: SPARE
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 110 °F
- On the steel liner of the cable slot: 328 °F
- Under damming board over the air space: 141 °F

i. Test Specimen: B1
Cable Fill: 300MCM (14 cables), 2/0-600v (3 cables), 2/0-8Kv (9 cables), 4/0 (2 cables), #2 (2 cables), #6 (4 cables), 3/c-#10 PXMJ (8 cables), and 3/c-#10 CPJJ (3-cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 192 °F
- Interface between the seal and cable: 574 °F
- Interface between the seal and tray side rail: 225 °F
- On the steel liner of the cable slot: 194 °F
- Under damming board on top of ceramic fiber blanket: 385 °F

j. Test Specimen: B2
Cable Fill: 300MCM (9 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 158 °F
- Interface between the seal and cable: 527 °F
- Interface between the seal and tray side rail: 139 °F
- On the steel liner of the cable slot: 162 °F
- Under damming board on top of ceramic fiber blanket: 138 °F

k. Test Specimen: B3
Cable Fill: 300MCM (7 cables), 2/0-8Kv (3 cables), 4/0 (4 cables), #2 (1 cables), #6 (2 cables), 3/c-#10 PXMJ (10 cables), and 3/c-#10 CPJJ (6-cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 208 °F
- Interface between the seal and cable: 595 °F
- Interface between the seal and tray side rail: 233 °F
- On the steel liner of the cable slot: 168 °F
- Under damming board on top of ceramic fiber blanket: 264 °F

l. Test Specimen: B4
Cable Fill: 300MCM (20 cables), 2/0 (12 cables), 4/0 (4 cables), #2 (2 cables), #6 (4 cables), 3/c-#10 PXMJ (14 cables), and 3/c-#10 CPJJ (6-cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 263 °F
- Interface between the seal and cable: 562 °F
- Interface between the seal and tray side rail: 298 °F
- On the steel liner of the cable slot: 195 °F
- Under damming board on top of ceramic fiber blanket: 434 °F

m. Test Specimen: B5
Cable Fill: 300MCM (9 cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 236 °F
- Interface between the seal and cable: 564 °F
- Interface between the seal and tray side rail: 148 °F
- On the steel liner of the cable slot: 165 °F
- Under damming board on top of ceramic fiber blanket: 381 °F

n. Test Specimen: B6
Cable Fill: 300MCM (7 cables), 2/0-8Kv (3 cables), 4/0 (4 cables), #2 (1 cables), #6 (2 cables), 3/c-#10 PXMJ (10 cables), and 3/c-#10 CPJJ (6-cables)
Unexposed side temperatures:

- On top of the ceramic fiber damming board: 200 °F
- Interface between the seal and cable: 601 °F
- Interface between the seal and tray side rail: 180 °F
- On the steel liner of the cable slot: 141 °F
- Under damming board on top of ceramic fiber blanket: 287 °F

All test specimens, met the IEEE 634 acceptance criteria. The maximum unexposed cable/seal interface temperatures ranged from 323 °F for instrument and control cables and 601 °F for power cables.

6.0 POST-FIRE TEST SPECIMEN EXAMINATION

Post-fire tear down and examination of the test specimens was performed on October 23, 1995. Due to the reviewer's travel schedule, only a portion of these examination was witnessed. The following summarizes the reviewer's observations made during the examination process:

Specimen A1 - Bottom side of cable bundle closest to cable tray side rail had no foam remaining. The penetration space between the sleeve and the top of the cable bundle had 6" of foam remaining. Cable jacket and insulation damage was present through the full thickness of the seal.

Specimen A3 - The ceramic fiber damming board on the unexposed side sustained smoke damage. Bottom side of cable bundle closest to cable tray side rail had no foam remaining. The penetration space between the sleeve and the top of the cable bundle had 3" of foam remaining. Cable jacket and insulation damage was present through the full thickness of the seal.

Specimen A7 - 3" of undamaged foam was present at the sleeve interface and approximately 1" at the center of the penetration seal. The lower ceramic damming board was heavily fire damaged but, remained intact and the upper damming board was undamaged.

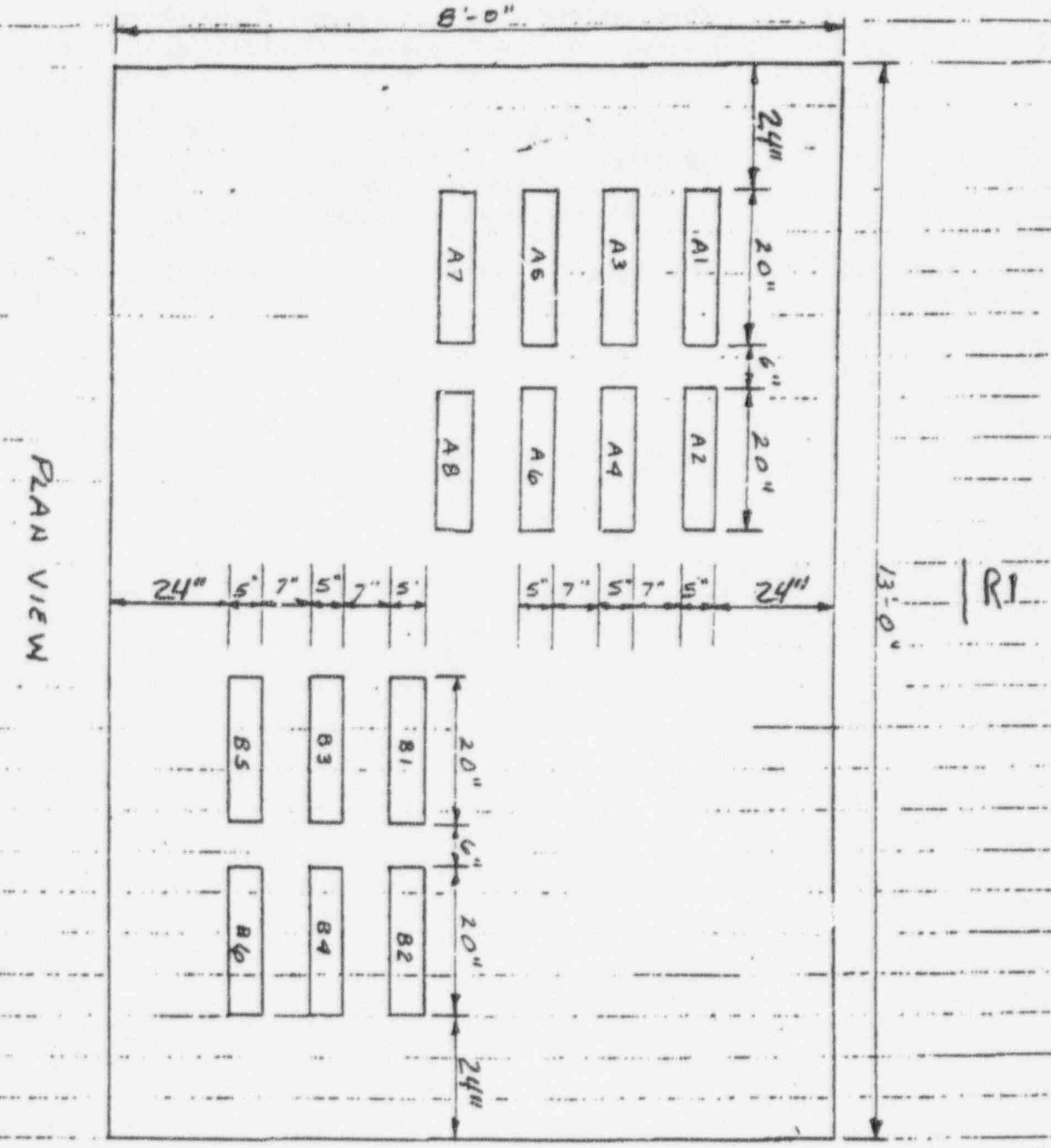
Specimen A8 - 9" of undamaged foam was present at the sleeve interface and approximately 7" at the center of the penetration seal. The lower ceramic damming board was heavily fire damaged but, remained intact and the upper damming board was undamaged.

Specimen B1 - Bare copper and sever jacket and insulation fire damage up to the unexposed side ceramic fiber damming board.

SUBJECT Penetration Seal Test - Detail 1 - PROJECT WBN

J. J. PIERCE / b[illegible] COMPUTED BY 10/16/95 DATE [illegible signature] CHECKED BY 10/16/95 DATE | R1

Detail 1
Tray Slot Location



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SUBJECT PENETRATION SEAL TEST PROJECT WBN

J J PIERCE 10/2/95 [signature] 10/2/95
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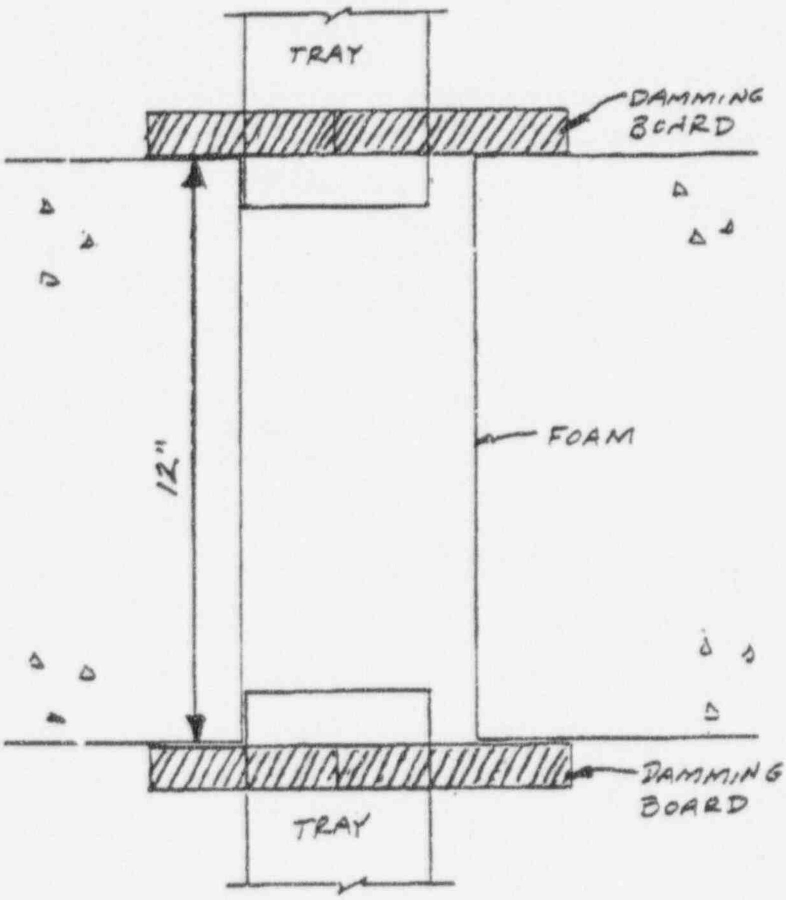
Detail 2



TYPICAL DETAIL A1-A6
(SCALE 1/4" = 1")

SUBJECT PENETRATION SEAL TEST PROJECT WBN

J.J PIERCE JJP 10/20/95 COMPUTED BY DATE

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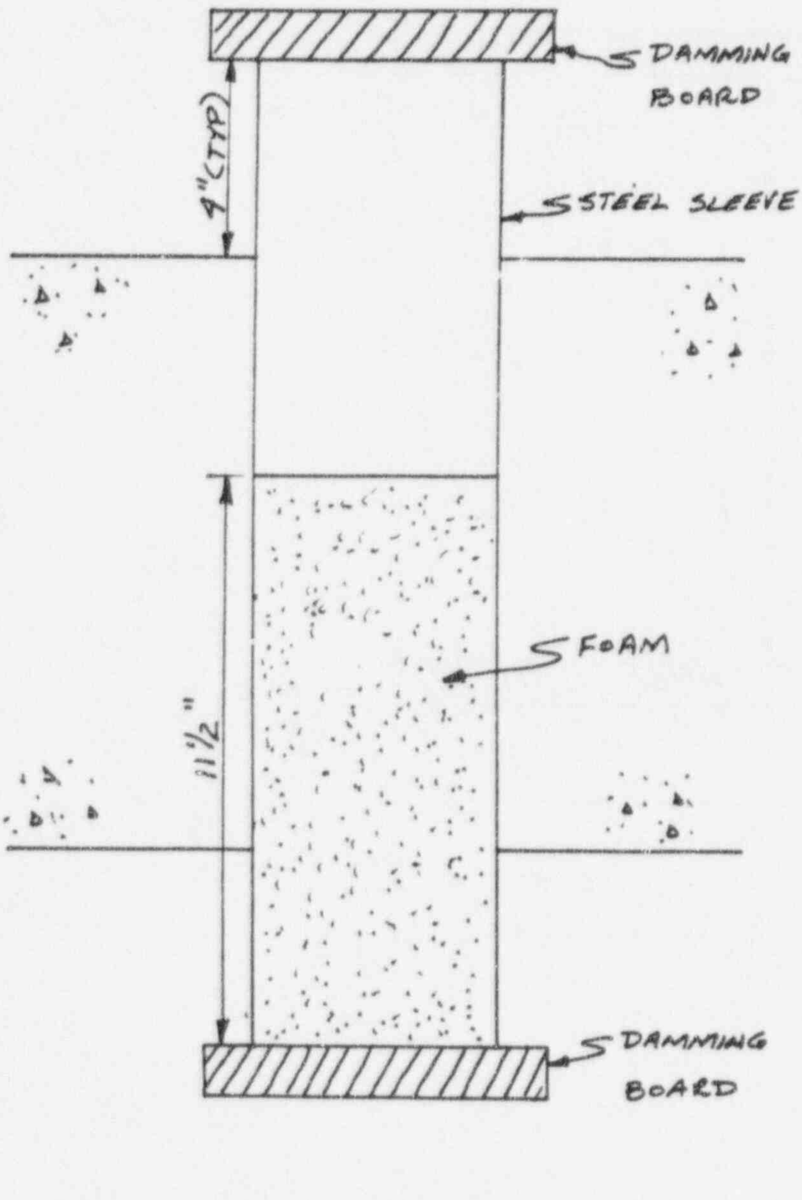
R2 ~~R3~~ JJP 10/20

DETAIL 3



TYPICAL DETAIL A7 & A8
(SCALE 1/4" = 1")
A7 - FIRE SIDE ON BOTTOM
A8 - FIRE SIDE ON TOP

SUBJECT PENETRATION SEAL TEST PROJECT WBN

J J PIERCE 10/2/95 [signature] 10/2/95 | R2

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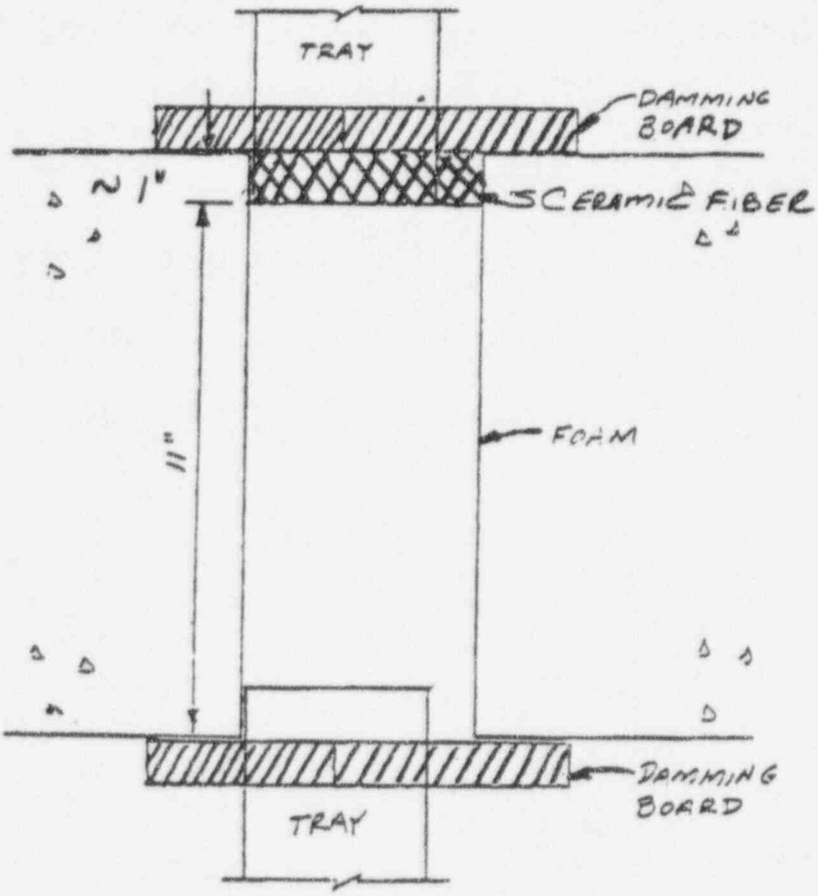
R2 JJ Pierce 10/20/95

Detail 4

R2



TYPICAL DETAIL "B"
(SCALE 1/4" = 1")

SUBJECT Cable Tray Frame & Tray Connection PROJECT

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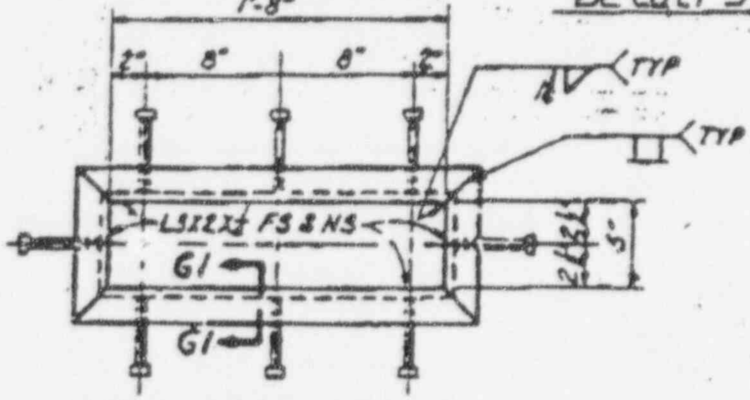
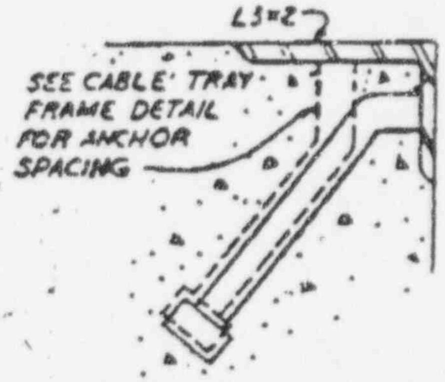
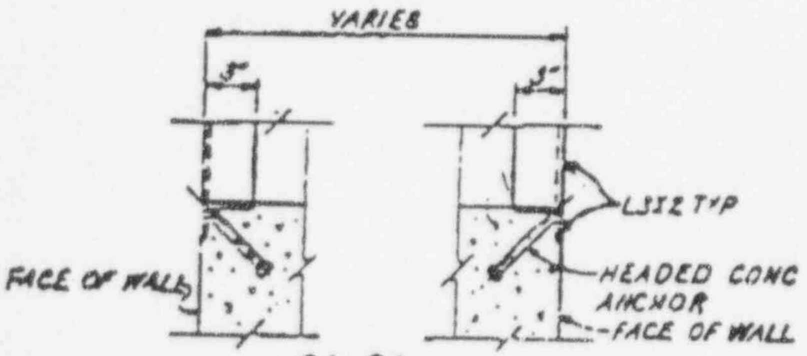
Detail 5



CABLE TRAY FRAME
ELEVATION
SCALE: 1½" = 1'-0"

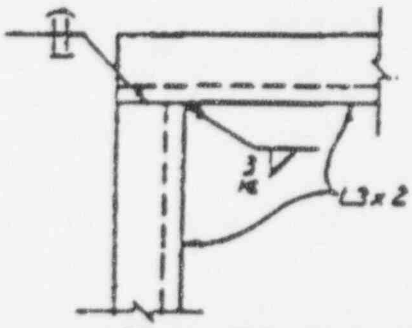


ALTERNATE CONCRETE ANCHOR DETAIL
SCALE: 6" = 1'-0"



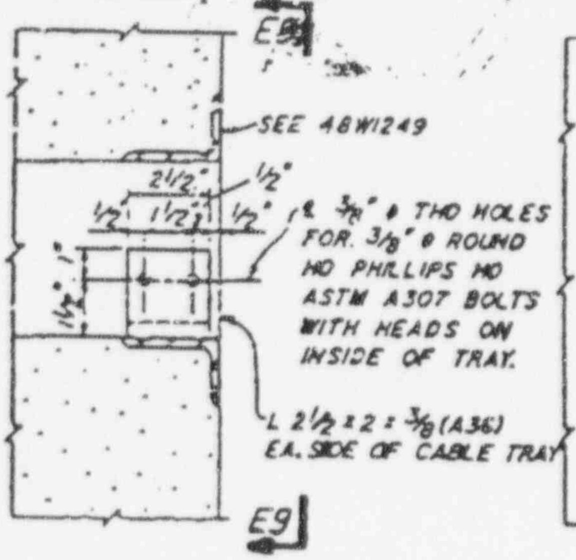
G1-G1

SCALE: 1½" = 1'-0"

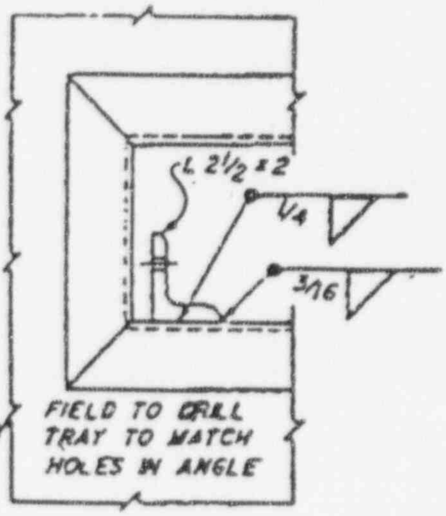


ALTERNATE CORNER DETAIL

SCALE: 3" = 1'-0"



ELEVATION



E9-E9

TYP HOLD DOWN CLIP DETAIL
FOR WALL FRAME CONNECTIONS

TENNESSEE VALLEY AUTHORITY SHEET

SUBJECT PENETRATION SEAL TEST PROJECT WBN

J.J. PIERCE 10/2/95 [signature] 10/2/95
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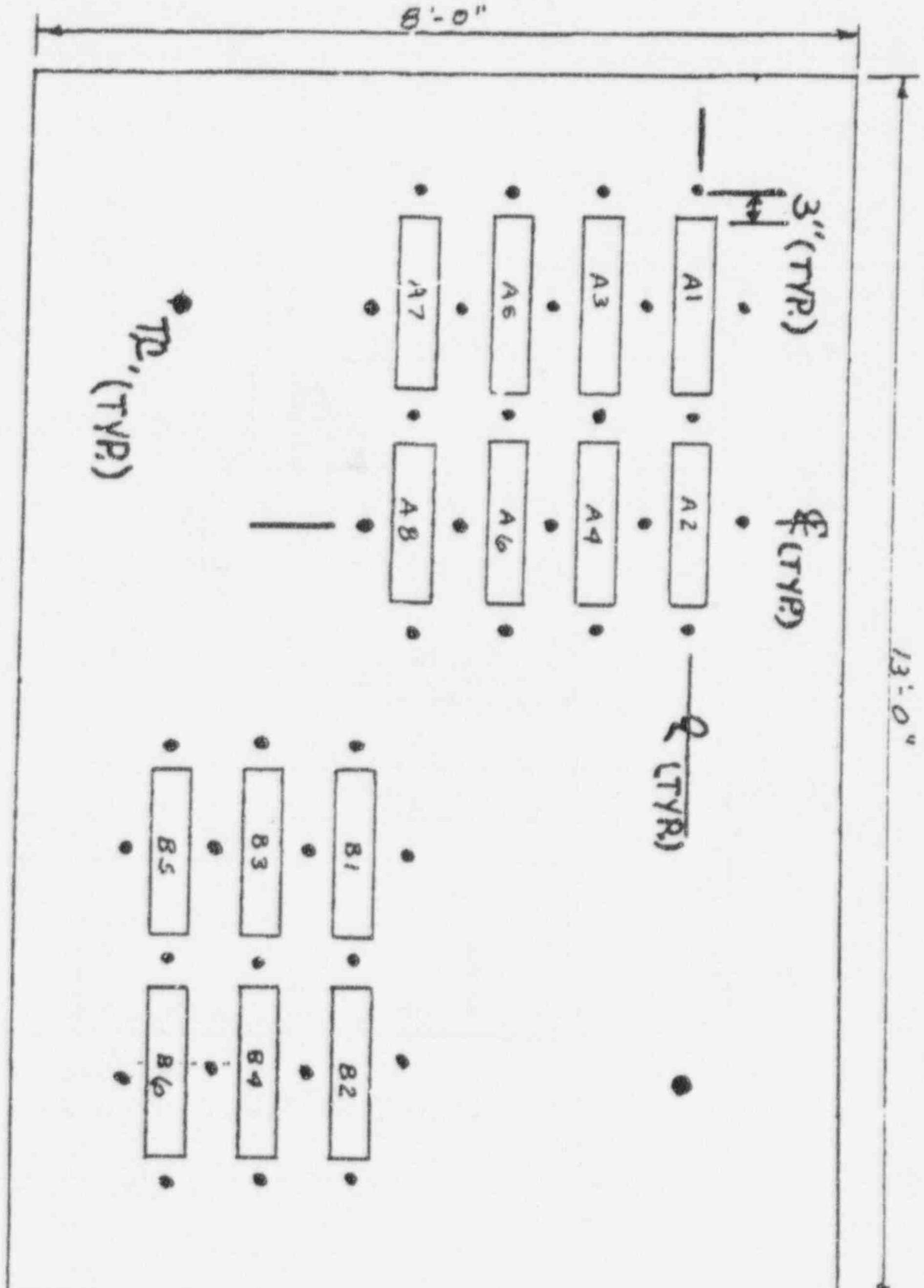
Detail 6

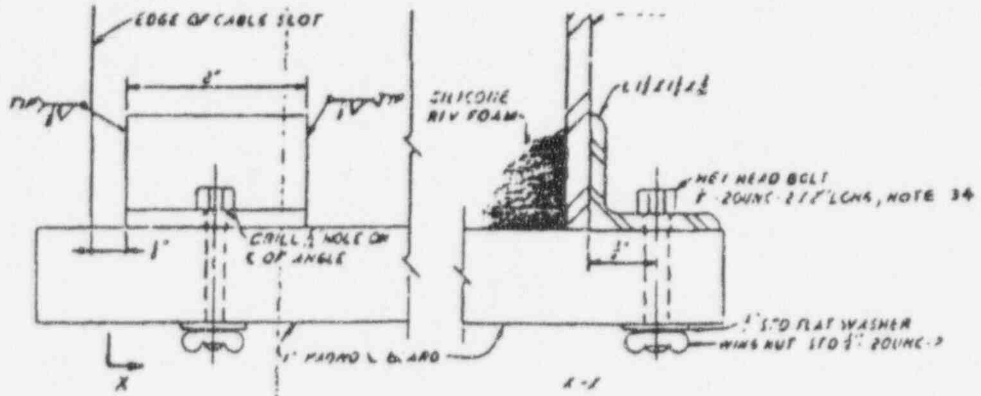
Engineering Thermocouples - Slab



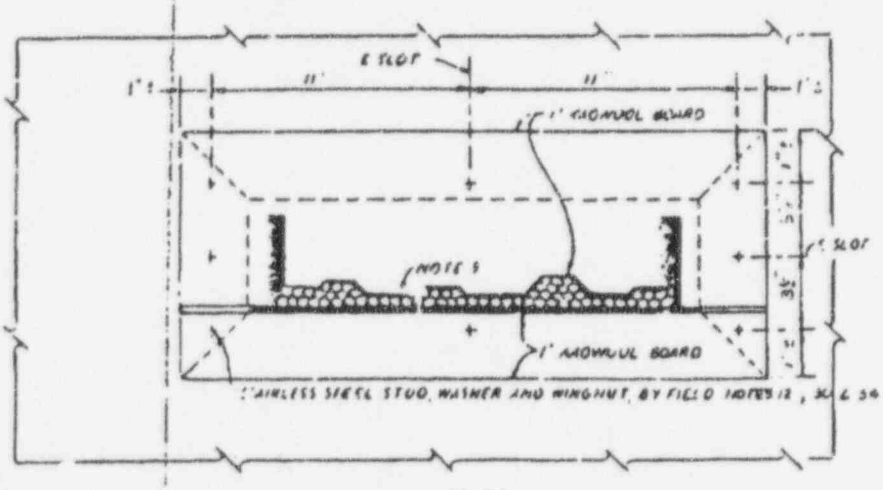
SUBJECT Penetration Seal Test PROJECT WBN

J. J. Pierce JJP 10/20/95 — COMPUTED BY, DATE
[signature] 10/22/95 — CHECKED BY, DATE

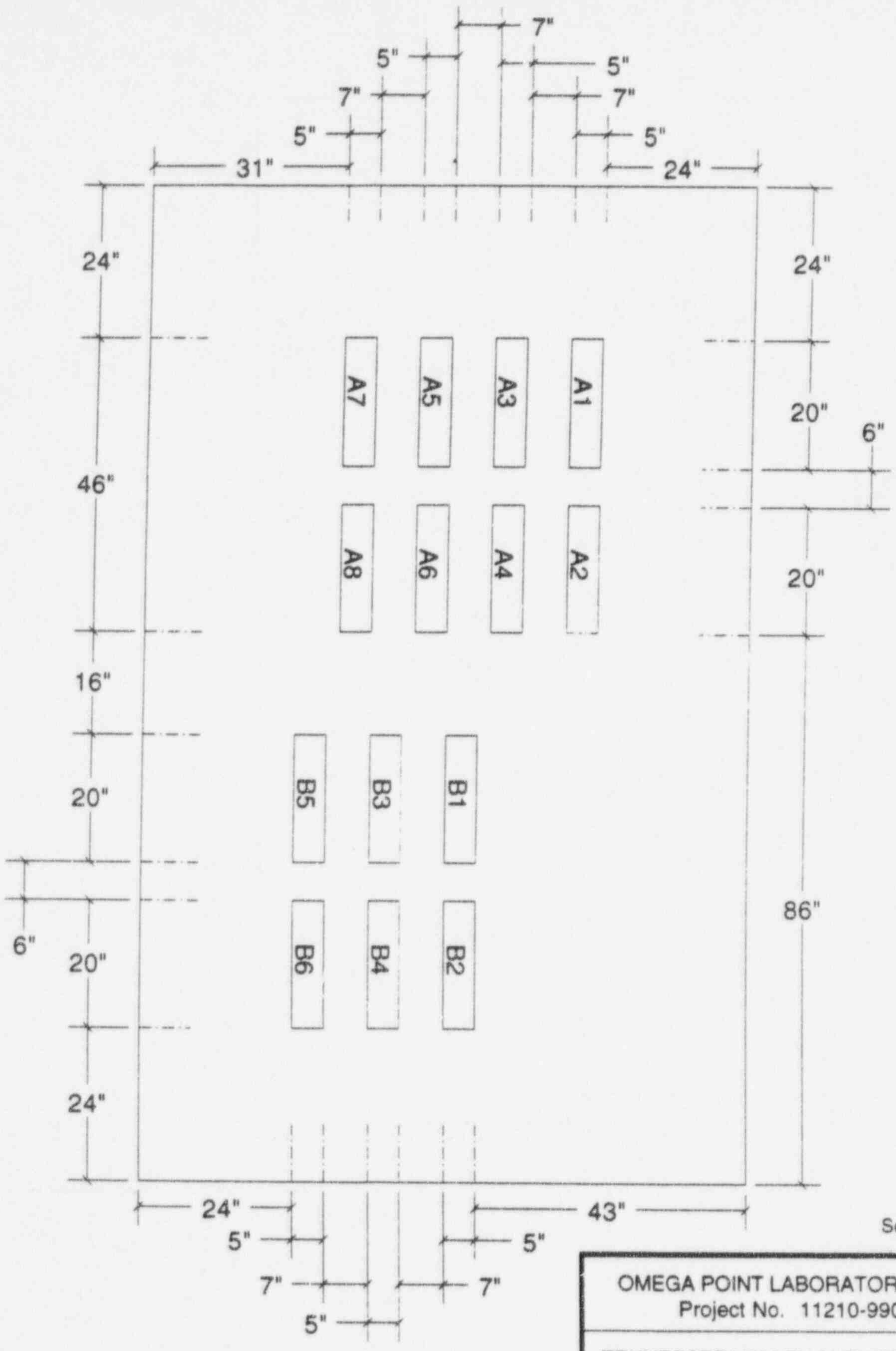
DETAIL 7



DETAIL 01
ANGLE ASSEMBLY USED TO SECURE KAOWOOL BOARD NOTE 10



A1-A1



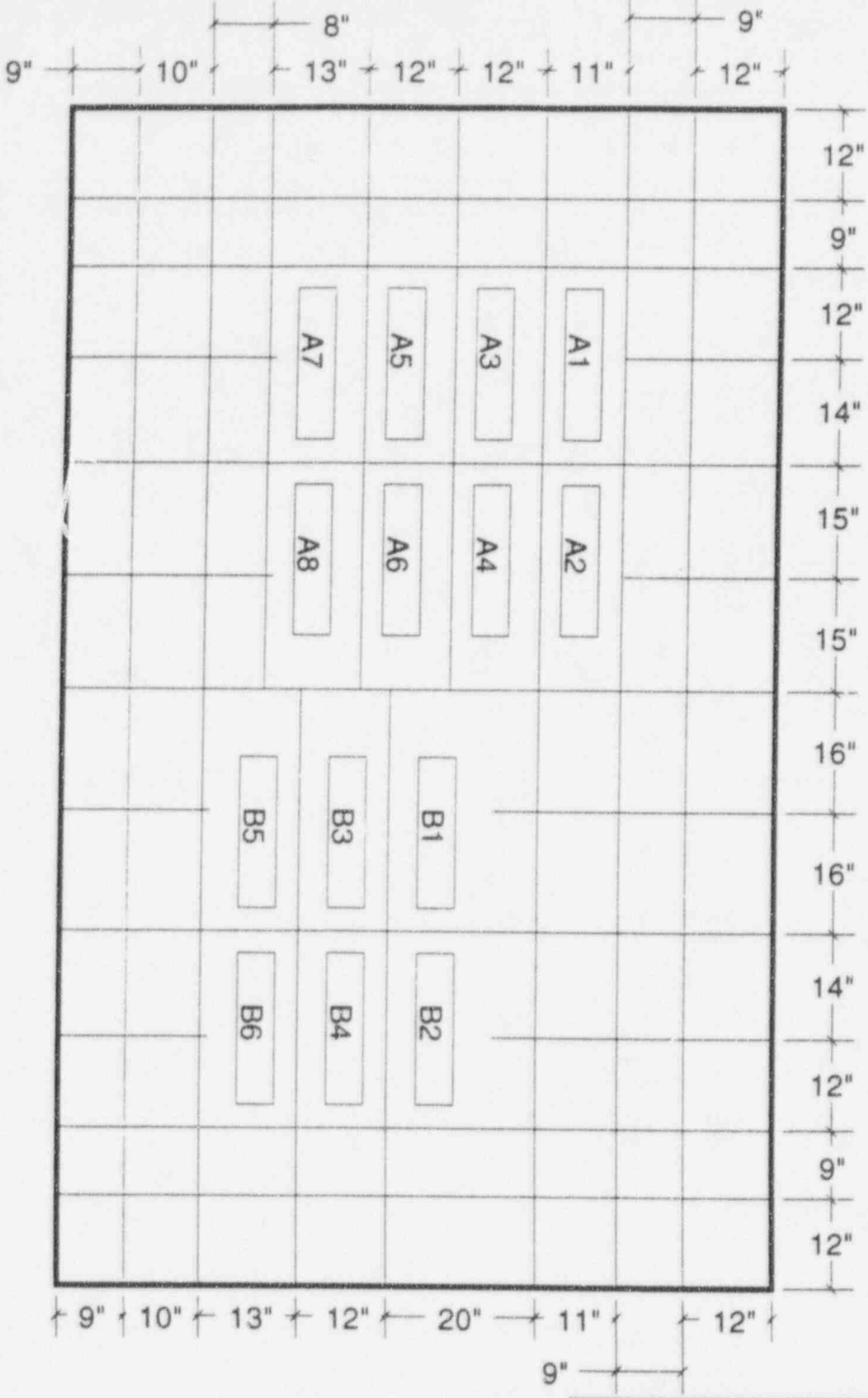
Scale:1/2"=1'

PLAN VIEW

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY - WBN
Fig. 1 Plan View of Specimen
Drawn By: [illegible] 10/7/95 Approved By: C Humphrey 10/7/95 Client Approval: JJ Pierce 10/21/95

OMEGA POINT LABORATORIES

ATTACHMENT 3



PLAN VIEW

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY - WBN
Fig. 2 Slab Rebars
Drawn By: H. Stacks 10/7/95 Approved By: C Humphrey 10/7/95 Client Approval: JJ Pierce 10/12/95

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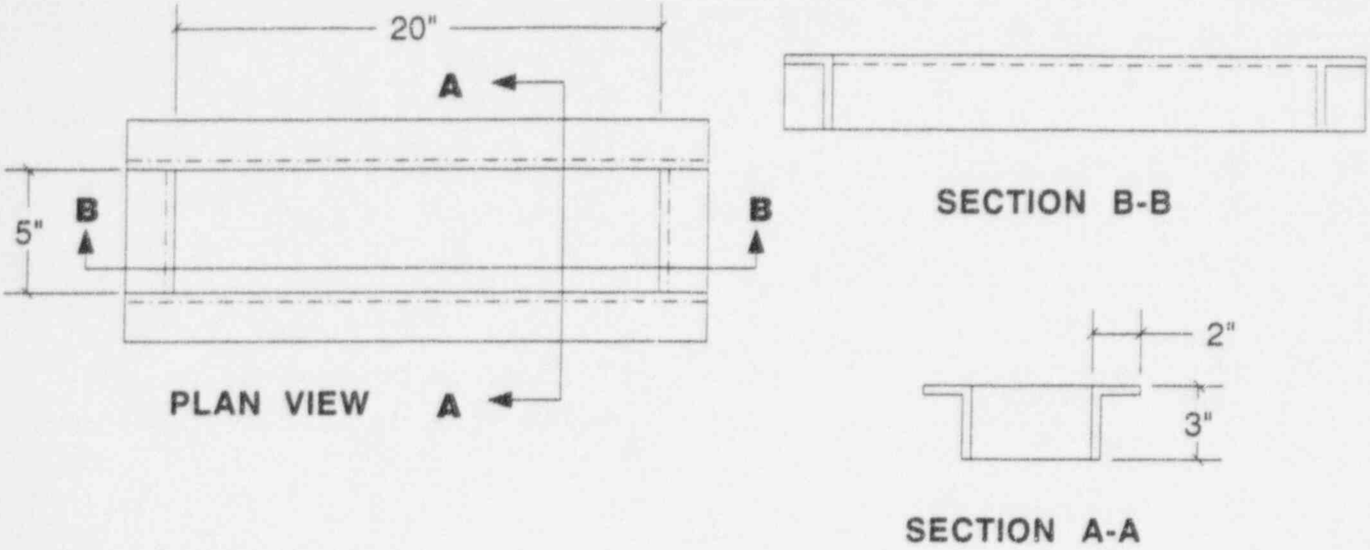

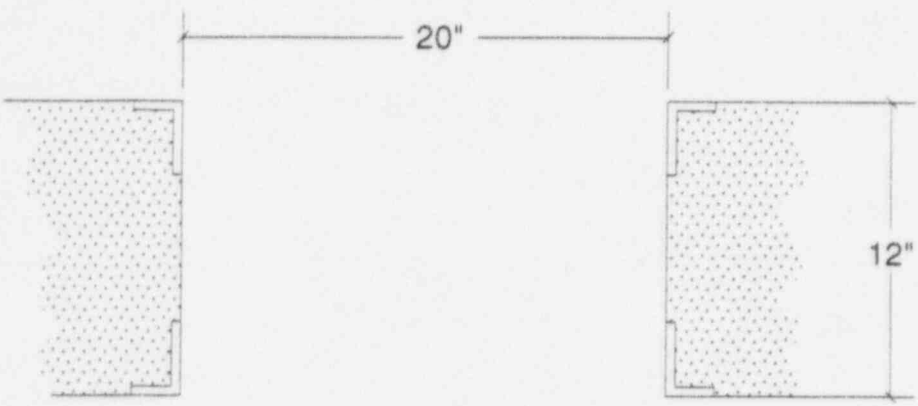

NOTE:
The upper and lower edges of the blockouts containing penetrations, included "picture frame" surrounds, fabricated of 3" x 2" x 3/8" steel (A36) angle, welded together at the ends and welded to 1/2"ø steel round stock, which was in turn welded to the concrete rebar as necessary to ensure structural stability. This blockout edging method was utilized for Penetrations A1-A6 and B1-B6.

Scale: 1-1/2"=1'

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
Tennessee Valley Authority - WBN
Fig. 3 Blockout Edge Details
Drawn By: [illegible] 10/7/95 Approved By: C Humphrey 10/7/95 Client Approval: JJ Pierce 10/12/95

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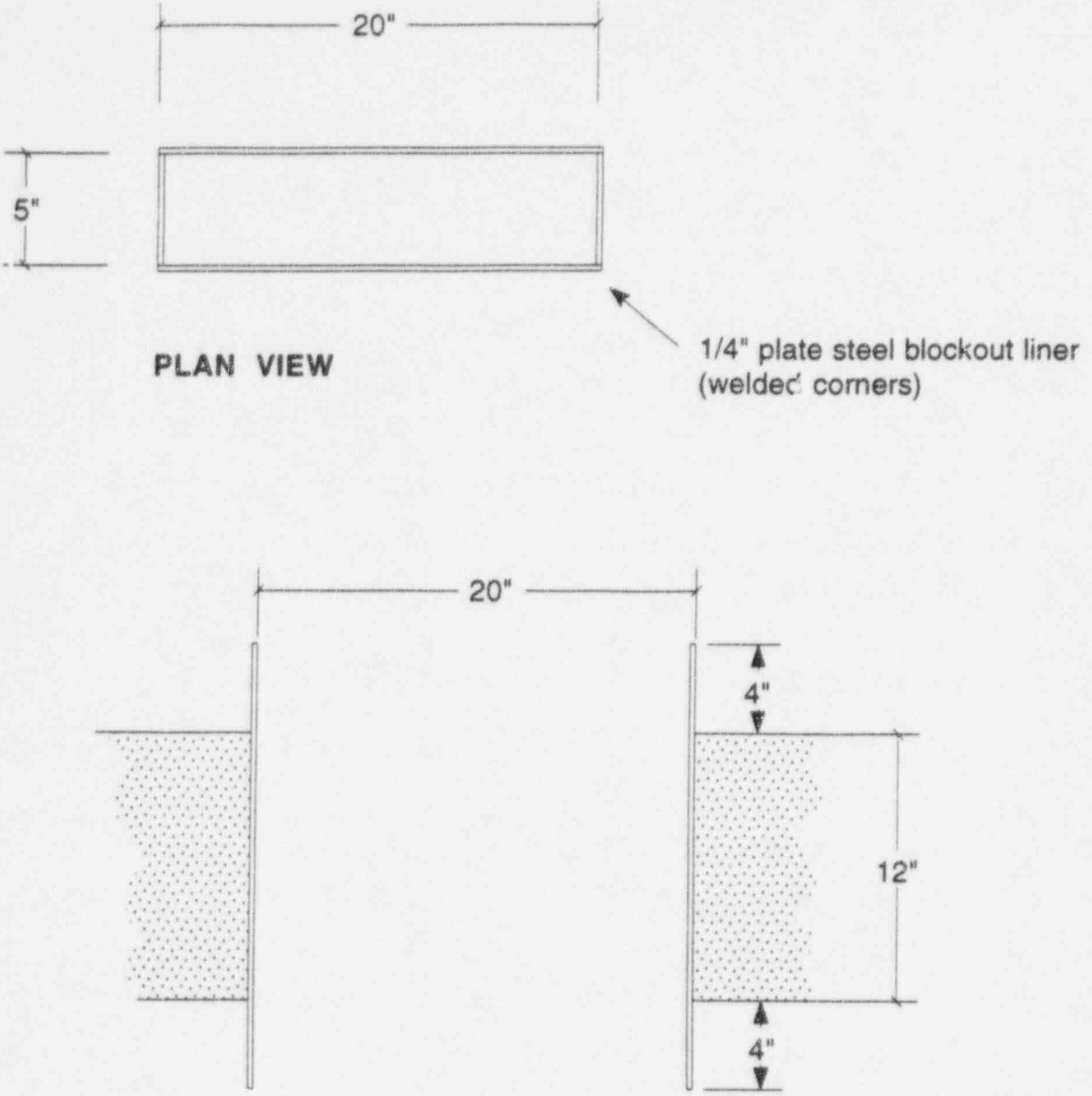
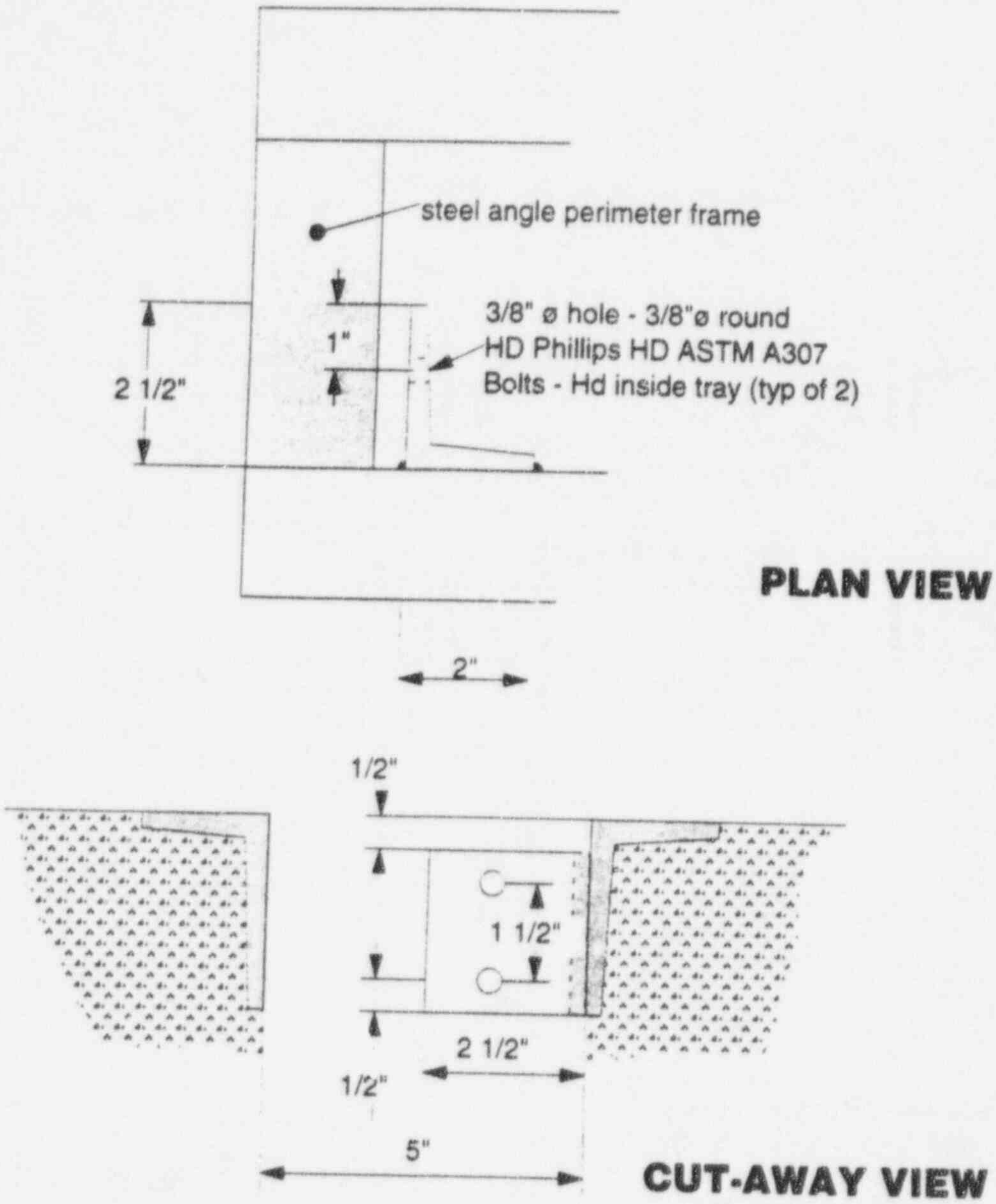


NOTE:
The blockouts not containing penetrations included sheet steel sleeve surrounds, fabricated of 1/4" thick plate steel (A36), welded together at the corners and welded to 1/2"ø steel round stock, which was in turn welded to the concrete rebar as necessary to ensure structural stability. Additional studs were welded to the faces of the sleeve in contact with concrete to insure a solid anchoring. This blockout edging method was utilized for Penetrations A7 and A8

Scale: 1-1/2"=1'

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
Tennessee Valley Authority - WBN
Fig. 4 Blockout Edge Details - A7, A8
Drawn By: H. Stanley II 10/7/95 Approved By: C. Humphrey 10/7/95 Client Approval: JJ Pierce 10/20/95



NOTE:
The above figure shows details of the cable tray mounting clips used at both side rails of each penetrating cable tray, on both the top and bottom metal blockout frames.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 5 Cable Tray Mounting Clip Details
Drawn By: [signature] 10/7/95 Approved By: C Humphrey 10/7/95 Client Approval: JJ Pierce 10/12/95

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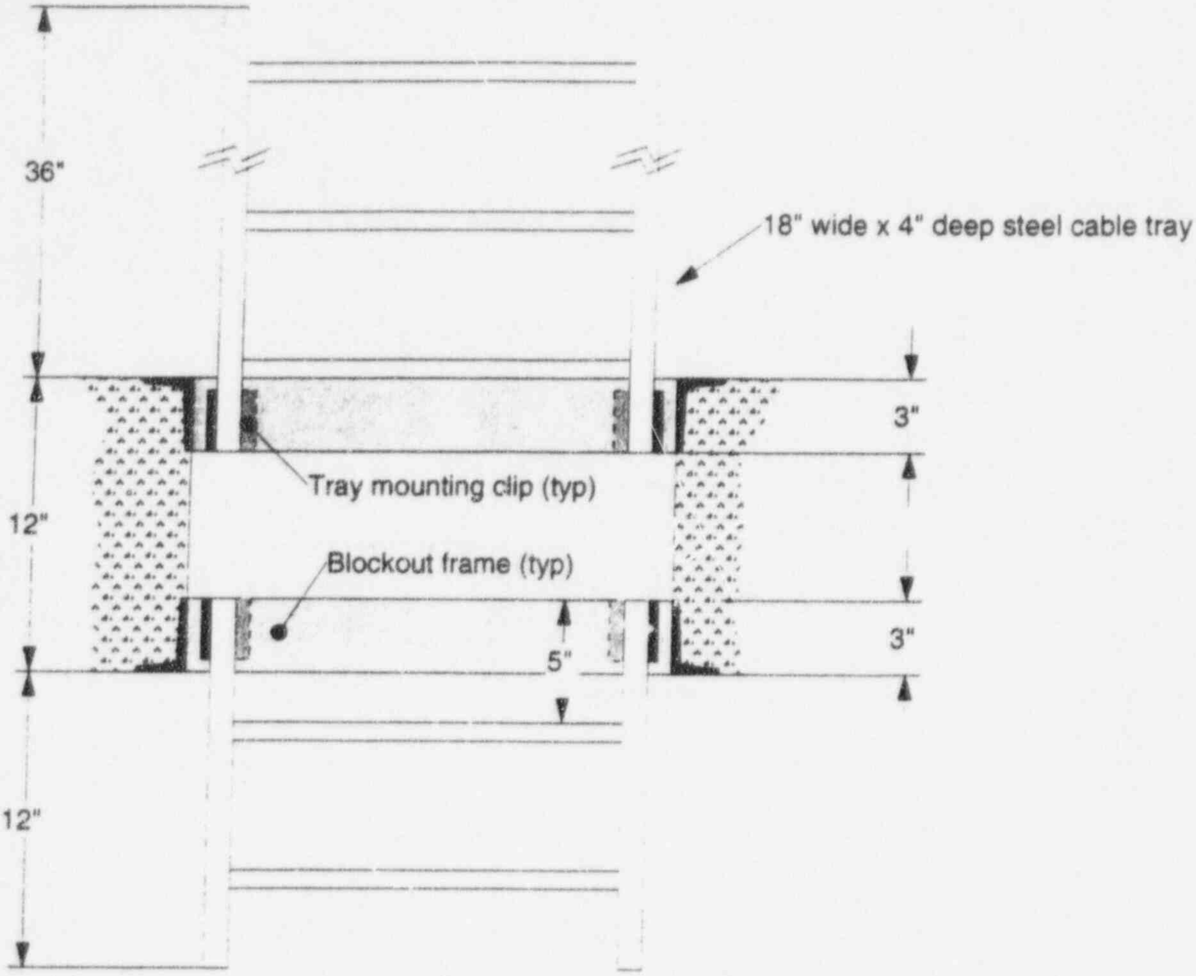

NOTE:
The 18" wide x 4" deep steel cable trays extended 36" above and 12" below the concrete slab. The trays did not pass through the penetration seal - each tray section extended 3" into the seal material. The trays were secured to the steel blockout frames with the tray mounting clips detailed in Fig. 5. Due to the location of the bolts securing the tray sections to the tray mountiong clips, a minimum of 3" was allowed between the end of the tray embedded in the seal and the first tray rung (5" for the lower tray sections).

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 6 Typical Cable Tray Elevation Details
Drawn by: H. Stanley 10/7/95 Approved by: C Humphrey 10/7/95 Client Approval: JJ Pierce 10/12/95

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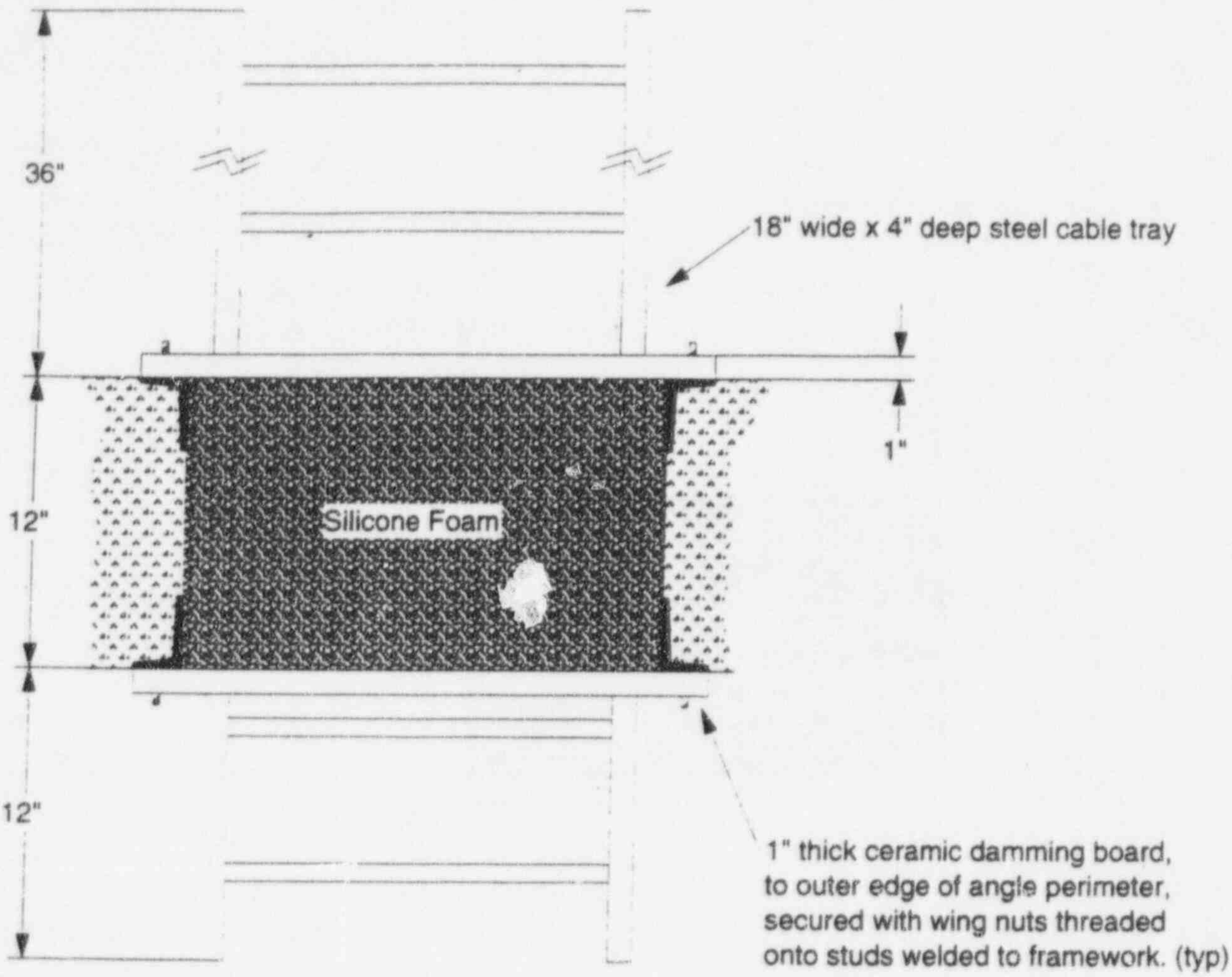
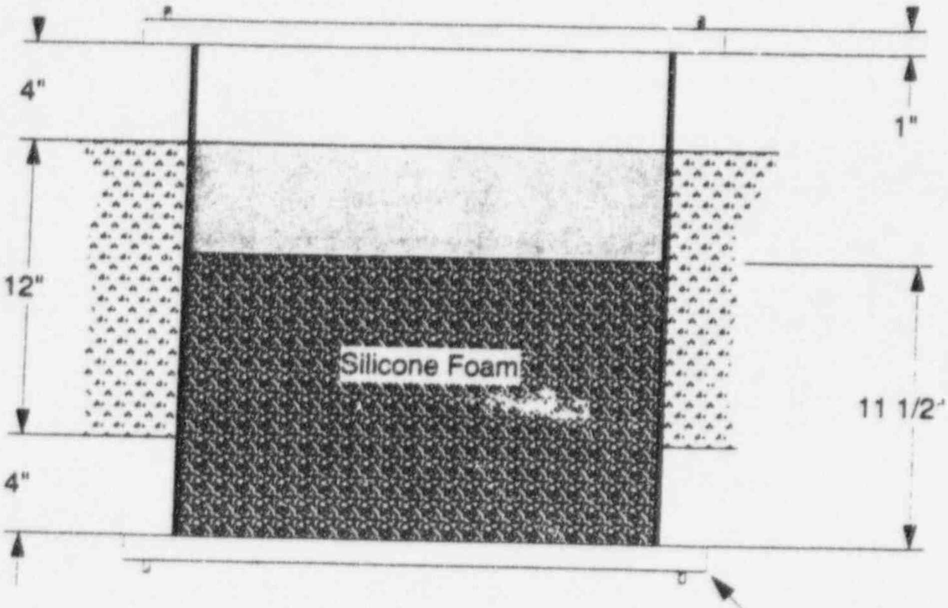


NOTE:
The penetration seal consisted of : 1) a 1" thick layer of ceramic damming board, installed flush with the bottom surface of the slab, secured to 1/4" studs welded to the metal blockout frames with washers and wing nuts, 2) a 12" depth of silicone foam, 3) a 1" thick layer of ceramic damming board, installed flush with the top surface of the slab, secured to 1/4" studs welded to the metal blockout frames with washers and wing nuts. The gaps between cables and damming board were filled with bulk ceramic fiber.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 7 Typical Penetration Seal Detail Penetration Set A (A1-A6)
Drawn by: H. S[illegible] 10/20/95 Approved by: C Humphrey 10/20/95 Client Approval: JJ Pierce 10/20/95

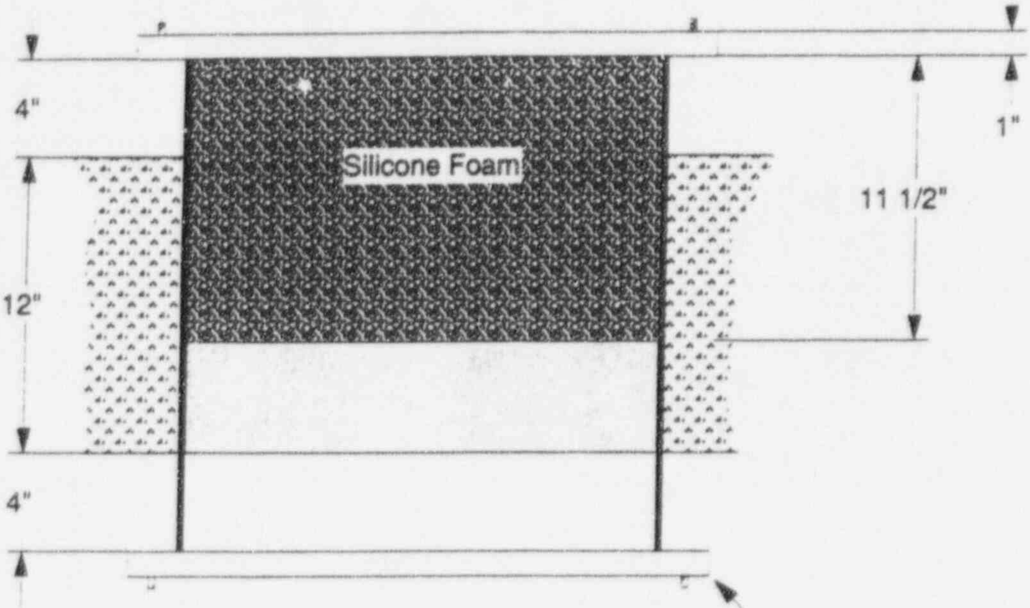
OMEGA POINT LABORATORIES



NOTE:
The penetration seal consisted of : 1) a 1" thick layer of ceramic damming board, installed flush with the bottom of the steel penetrating sleeve, secured to 1/4" studs welded to clip angles welded to the outside of the steel sleeve with washers and wing nuts, 2) an 11-1/2" depth of silicone foam, 3) an air gap of 8-1/2", and 4) a 1" thick layer of ceramic damming board, installed flush with the top of the steel penetrating sleeve, secured to 1/4" studs welded to clip angles welded to the outside of the steel sleeve with washers and wing nuts.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 8 Penetration Seal Detail Penetration A7
Drawn by: [signature] 10/20/95 Approved by: C Humphrey 10/20/95 Client Approval: J J Pierce 10/20/95

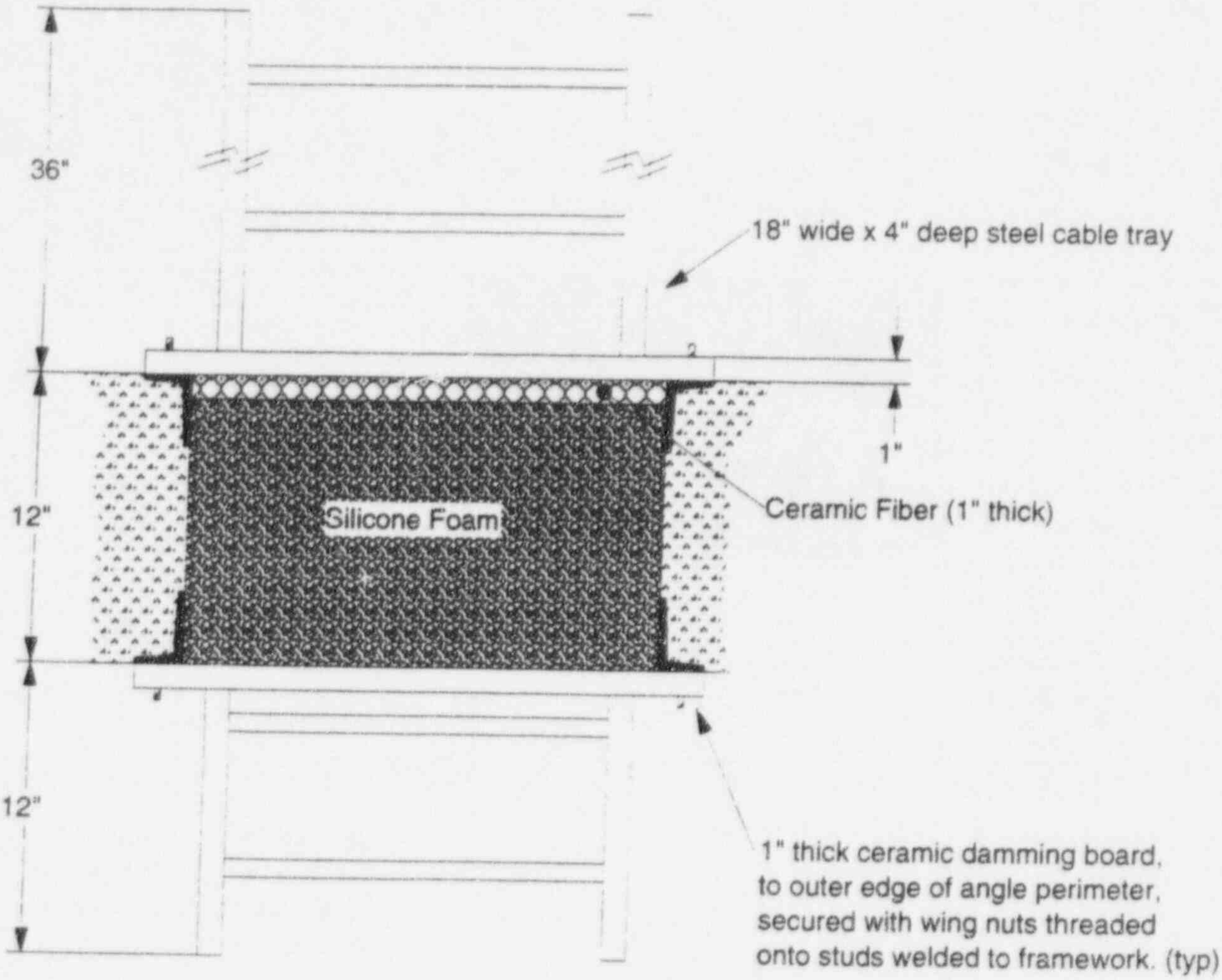
OMEGA POINT LABORATORIES



NOTE:
The penetration seal consisted of : 1) a 1" thick layer of ceramic damming board, installed flush with the bottom of the steel penetrating sleeve, secured to 1/4" studs welded to clip angles welded to the outside of the steel sleeve with washers and wing nuts, 2) an air gap of 8-1/2", 3) an 11-1/2" depth of silicone foam, and 4) a 1" thick layer of ceramic damming board, installed flush with the top of the steel panetrating sleeve, secured to 1/4" studs welded to clip angles welded to the outside of the steel sleeve with washers and wing nuts.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 9 Penetration Seal Detail Penetration A8
Drawn by: H. Stanley 10/20/95 Approved by: C Humphrey 10/20/95 Client Approval: JJ Pierce 10/20/95

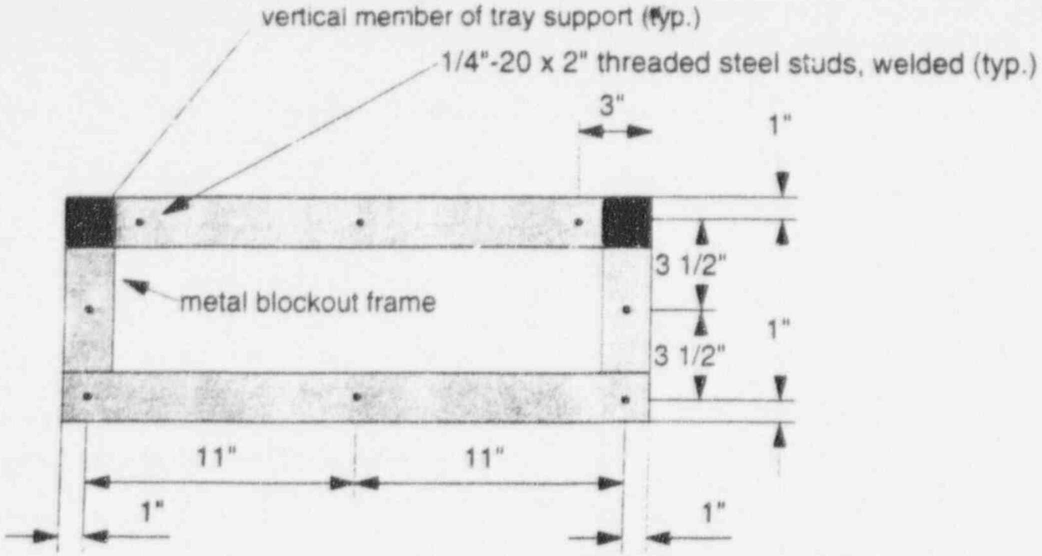
OMEGA POINT LABORATORIES



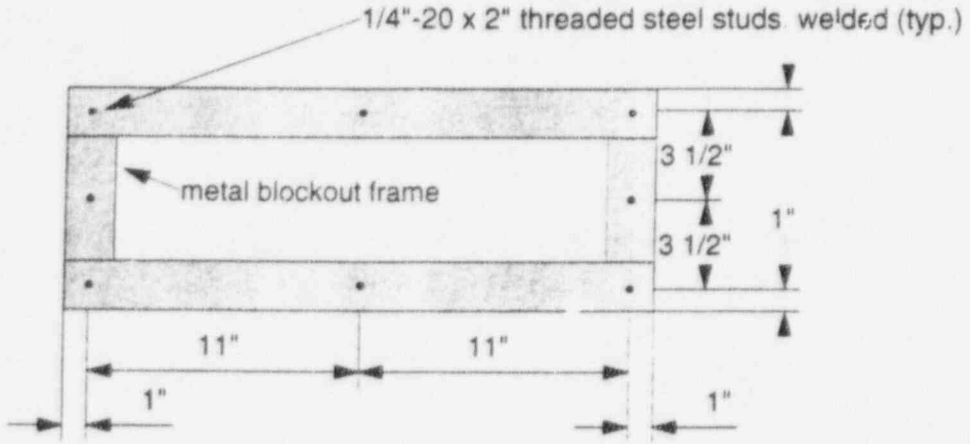
NOTE:
The penetration seal consisted of : 1) a 1" thick layer of ceramic damming board, installed flush with the bottom surface of the slab, secured to 1/4" studs welded to the metal blockout frames with washers and wing nuts, 2) an 11" depth of silicone foam, 3) a 1" thick layer of ceramic fiber blanket material, and 4) a 1" thick layer of ceramic damming board, installed flush with the top surface of the slab, secured to 1/4" studs welded to the metal blockout frames with washers and wing nuts. The gaps between cables and damming board were filled with bulk ceramic fiber.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 10 Typical Penetration Seal Detail Penetration Set B (B1-B6)
Drawn by: [signature] 10/20/95 Approved by: C Humphrey 10/20/95 Client Approval: J J Pierce 10/20/95

OMEGA POINT LABORATORIES



PLAN VIEW - TOP OF SLAB

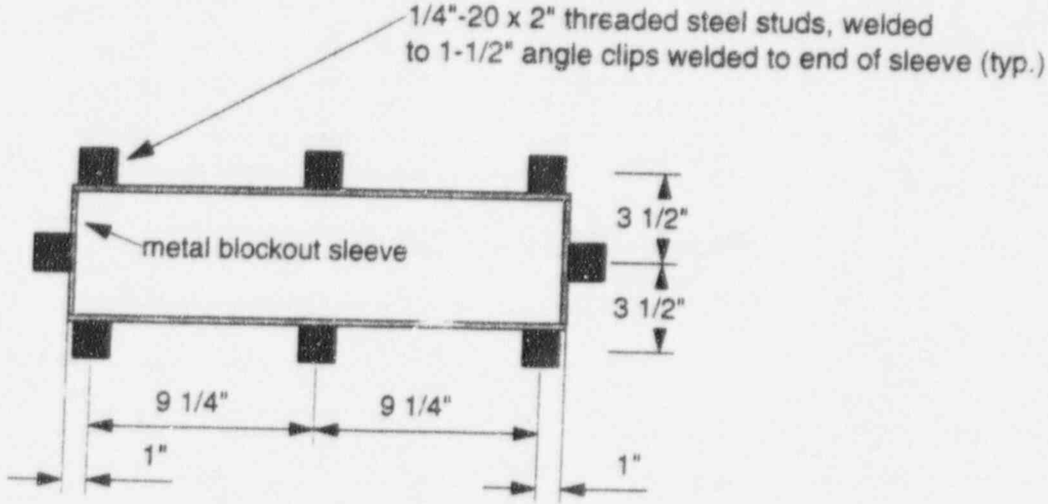


PLAN VIEW - BOTTOM OF SLAB

NOTE:
The 1/4"-20 x 2" long steel studs were welded to the blockout perimeters as shown above. The ceramic damming board was then installed over the studs, with the board covering to the outer edges of the blockout frames, and was secured with washers and wing nuts.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 11 Stud Placements For Damming Board Installation (A1-A6, B1-B6)
Drawn by: H Stanbery II 10/20/95 Approved by: C Humphrey 10/20/95 Client Approval: JJ Pierce 10/20/95

OMEGA POINT LABORATORIES

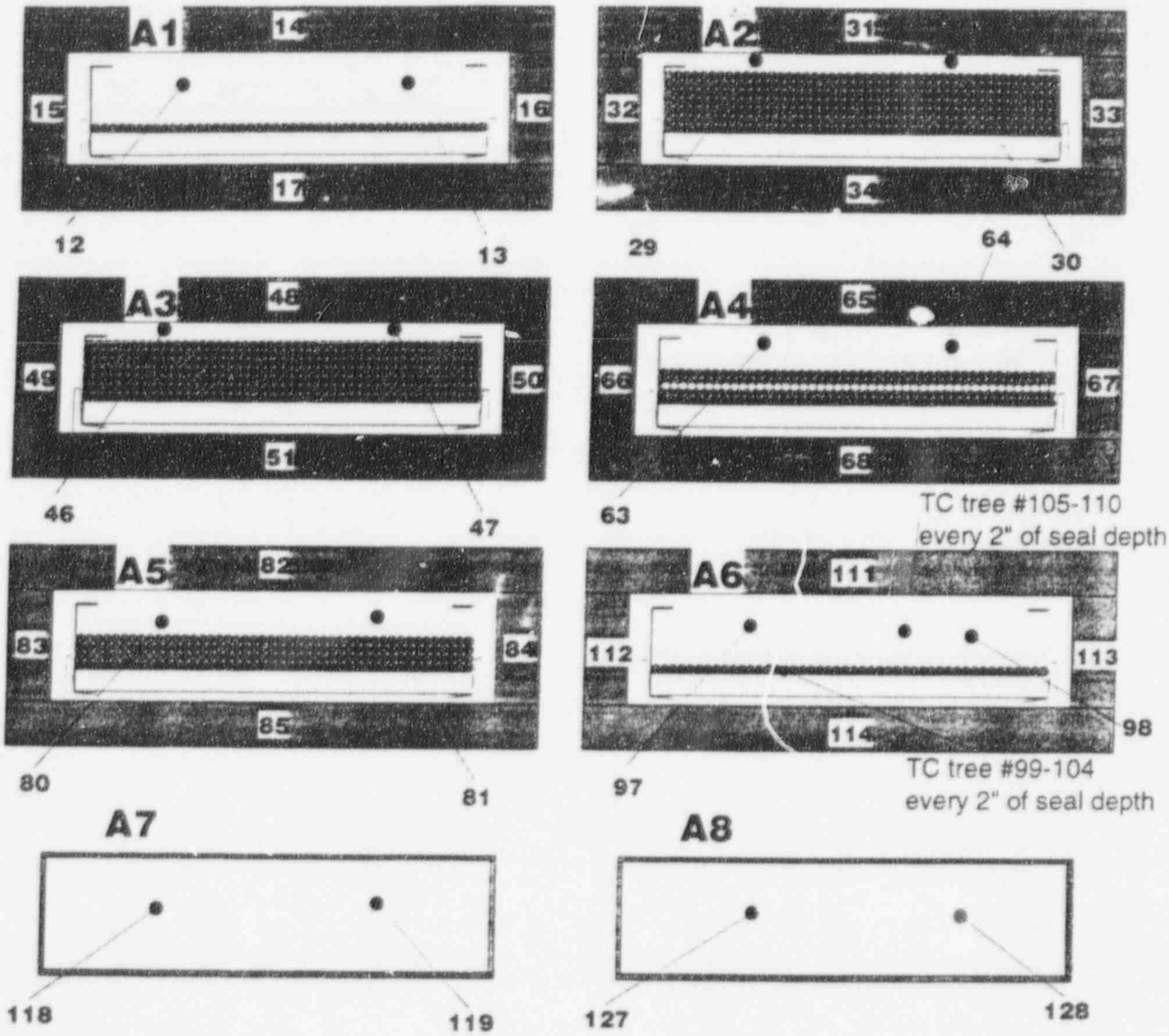


PLAN VIEW - TOP OF SLAB

NOTE:
The 1/4"-20 x 2" long steel studs were welded to 1-1/2" long sections of 1-1/2" x 1-1/2" x 3/16" steel angle, welded to the outside of the penetrating sleeve. The angles were placed such that the flat of each angle was flush with the end of the steel sleeve. The ceramic damming board was then installed over the studs, and was secured with washers and wing nuts.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Fig. 12 Stud and Clip Angle Placements For Damming Board Installation (A7-A8)
Drawn by: H. Stanley 10/21/95 Approved by: C Humphrey 10/21/95 Client Approval: JJ Pierce 10/21/95

OMEGA POINT LABORATORIES



NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. All TCs shown above were placed prior to installation of the top damming board material on the seals. TCs were placed on the top surface of the silicone foam (TC Nos. 12, 13, 29, 30, 46, 47, 63, 64, 80, 81, 97, 98, 118, 119, 127, 128), on the surface of one side of the blockout perimeter frames (TC Nos. 14-16, 31-34, 48-51, 65-68, 82-85, 111-114) 1" from the 1/4"-20 x 2" damming board stud. Additional TC trees were installed on one 4C/#16 cable and in the field of the seal in Pen. A6.

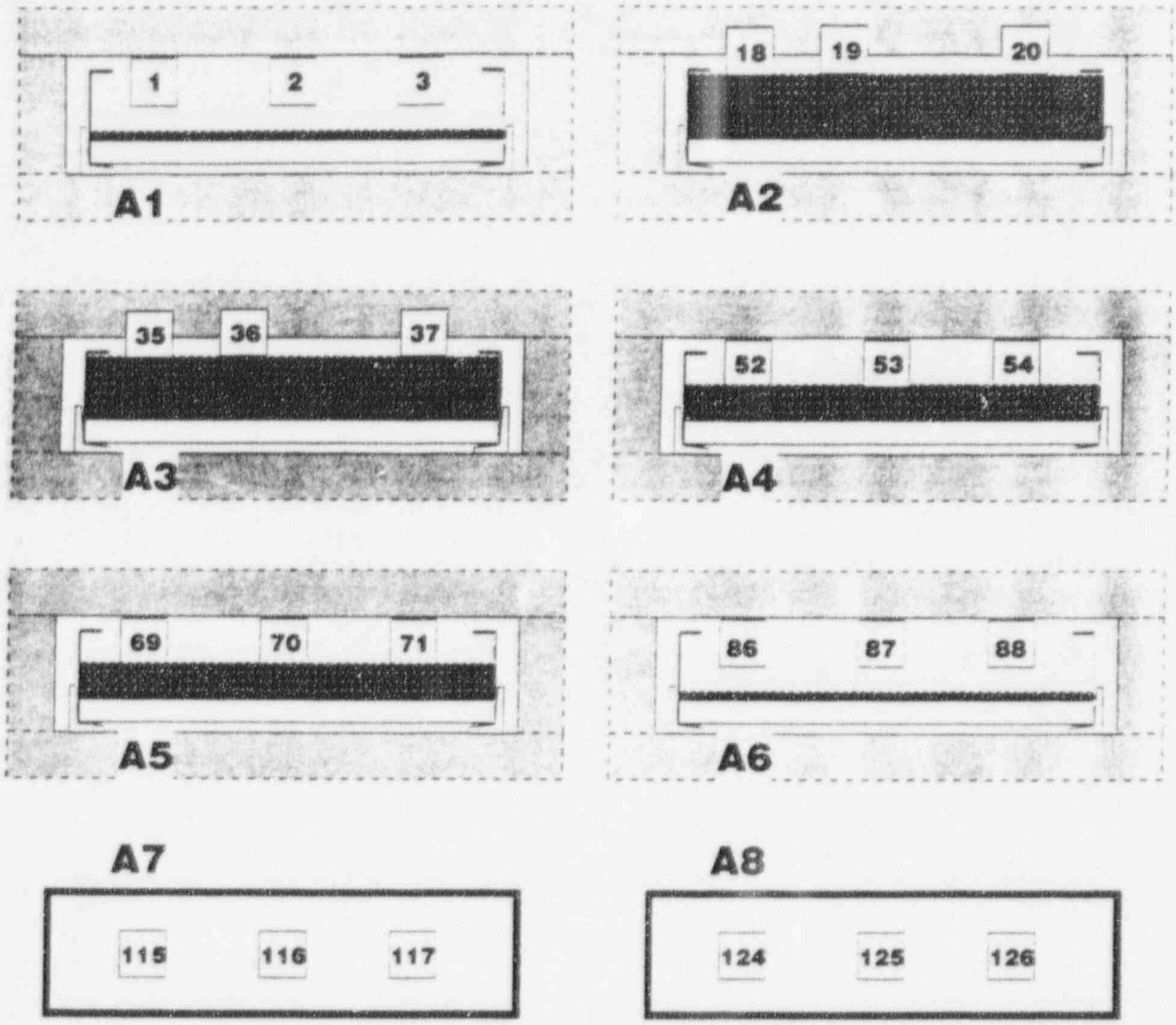
OMEGA POINT LABORATORIES, INC.
Project No. 11210-99050

TENNESSEE VALLEY AUTHORITY

Thermocouple Locations - Penetration Set A
Under Upper Damming Board Material

Drawn by: H. [signature] 10/21/95
Approved by: C Humphries 10/21/95
Client Approval: JJ Pierce 10/21/95

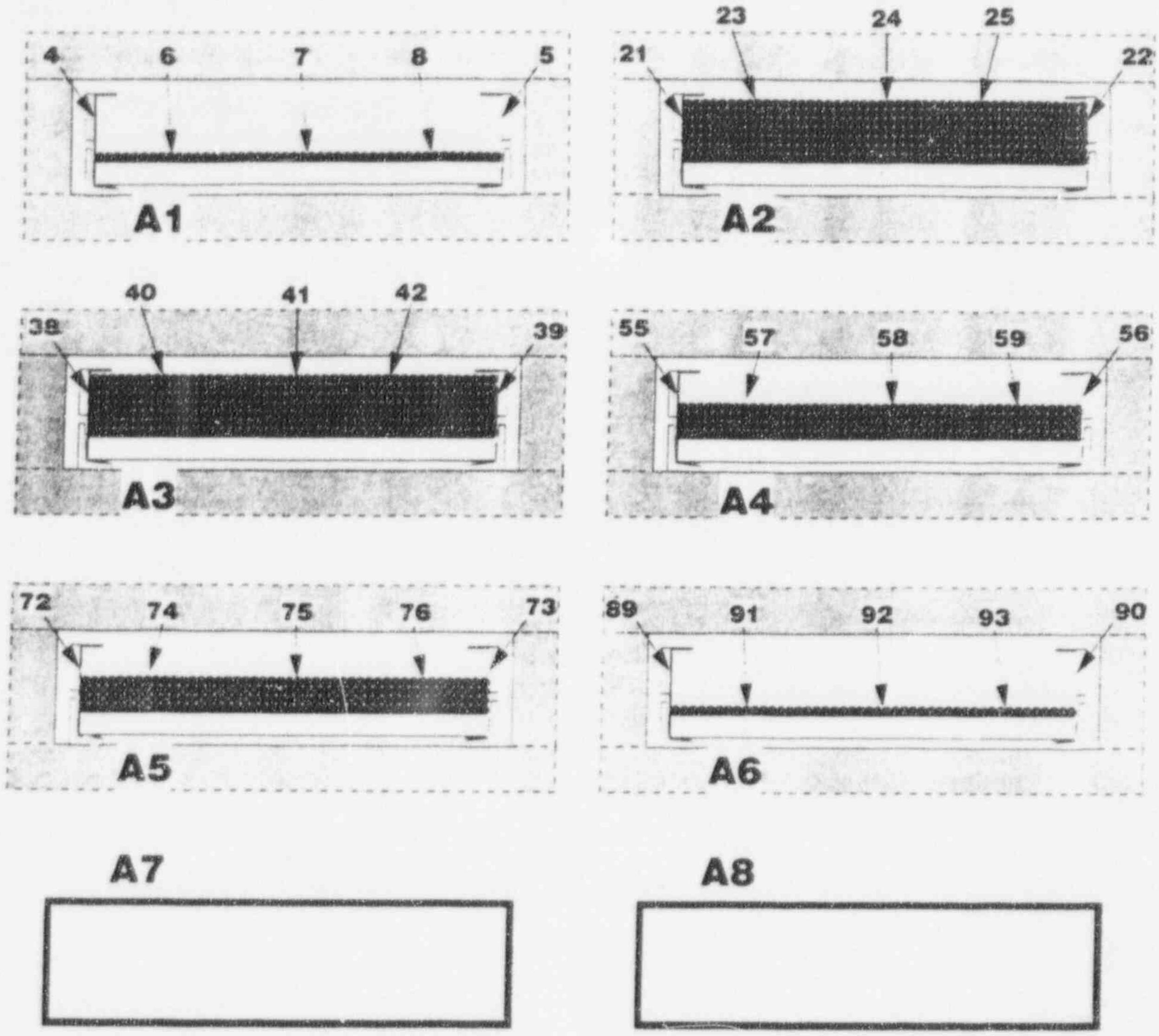
OMEGA POINT LABORATORIES



NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on the surface of the ceramic fiber damming board on the unexposed face of the seals. The TCs wer covered with 2" x 2" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Thermocouple Locations - Penetration Set A Surface of Unexposed Seal Face
Drawn by: H. [illegible] 10/20/95 Approved by: E. Humphrey 10/20/95 Client Approval: J.J. Price 10/~~25~~ 20/95 [illegible] 10/20/95

OMEGA POINT LABORATORIES



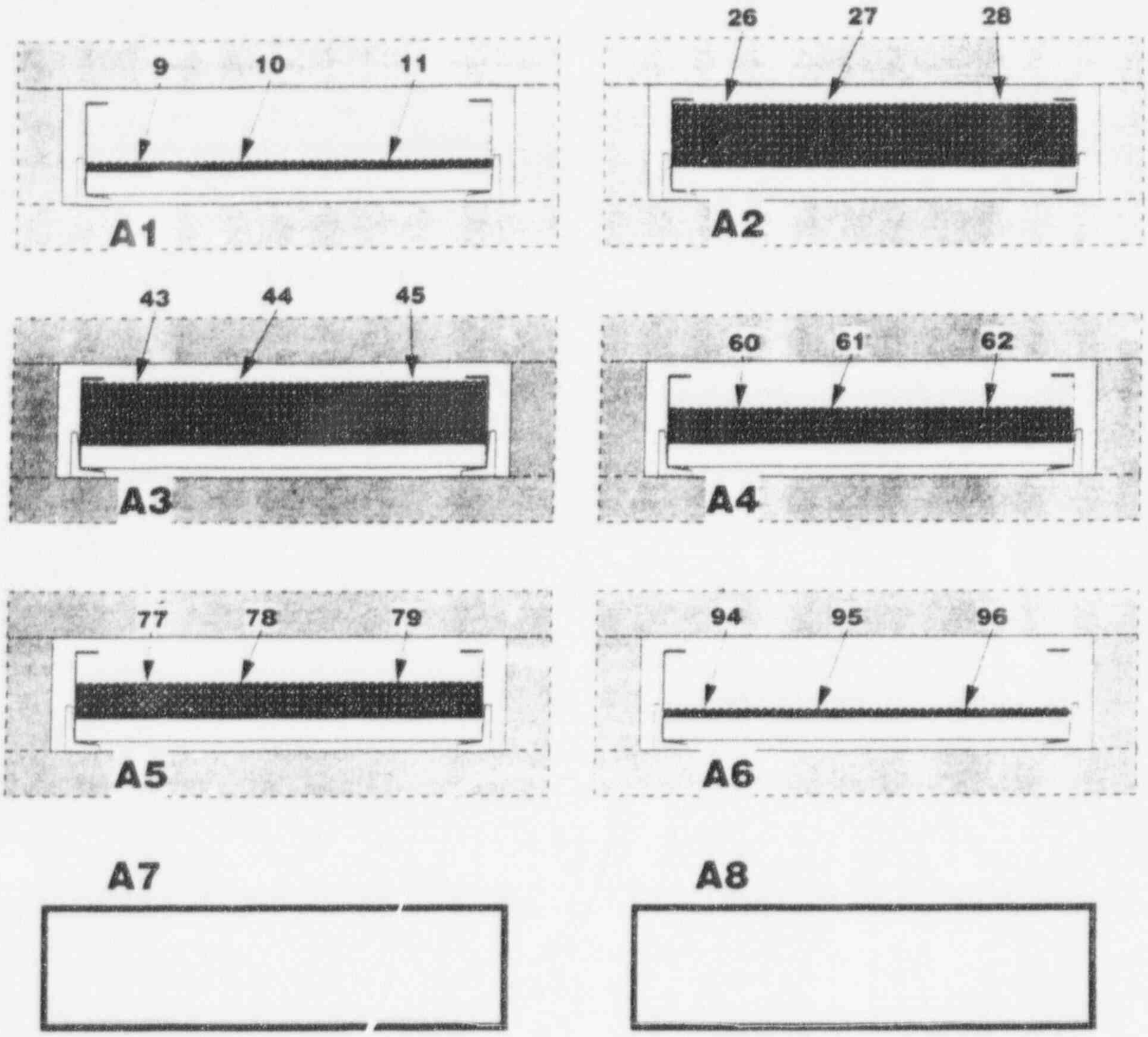
NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on cables (TC Nos. 6-8, 23-25, 40-42, 57-59, 74-76, 91-93) and on tray side rails (TC Nos. 4-5, 21-22, 38-39, 55-56, 72-73, 89-90), at the interface between the penetrants and the upper damming board. The TCs were covered with 3/4" x 3/4" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC.
Project No. 11210-99050

TENNESSEE VALLEY AUTHORITY

Thermocouple Locations - Penetration Set A
Interface Between Seal and Penetrants

Drawn by: H [signature] 10/20/95
Approved by: C Humphrey 10/20/95
Client Approval: JJ Pierce 10/20/95



NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on cables, 1" to 3" above the interface between the penetrants and the upper damming board. The TCs were covered with 3/4" x 3/4" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Thermocouple Locations - Penetration Set A On Cables Above Unexposed Seal Face
Drawn by: H. [illegible] 10/20/95 Approved by: C Humphrey 10/20/95 Client Approval: JJ Pierce 10/20/95

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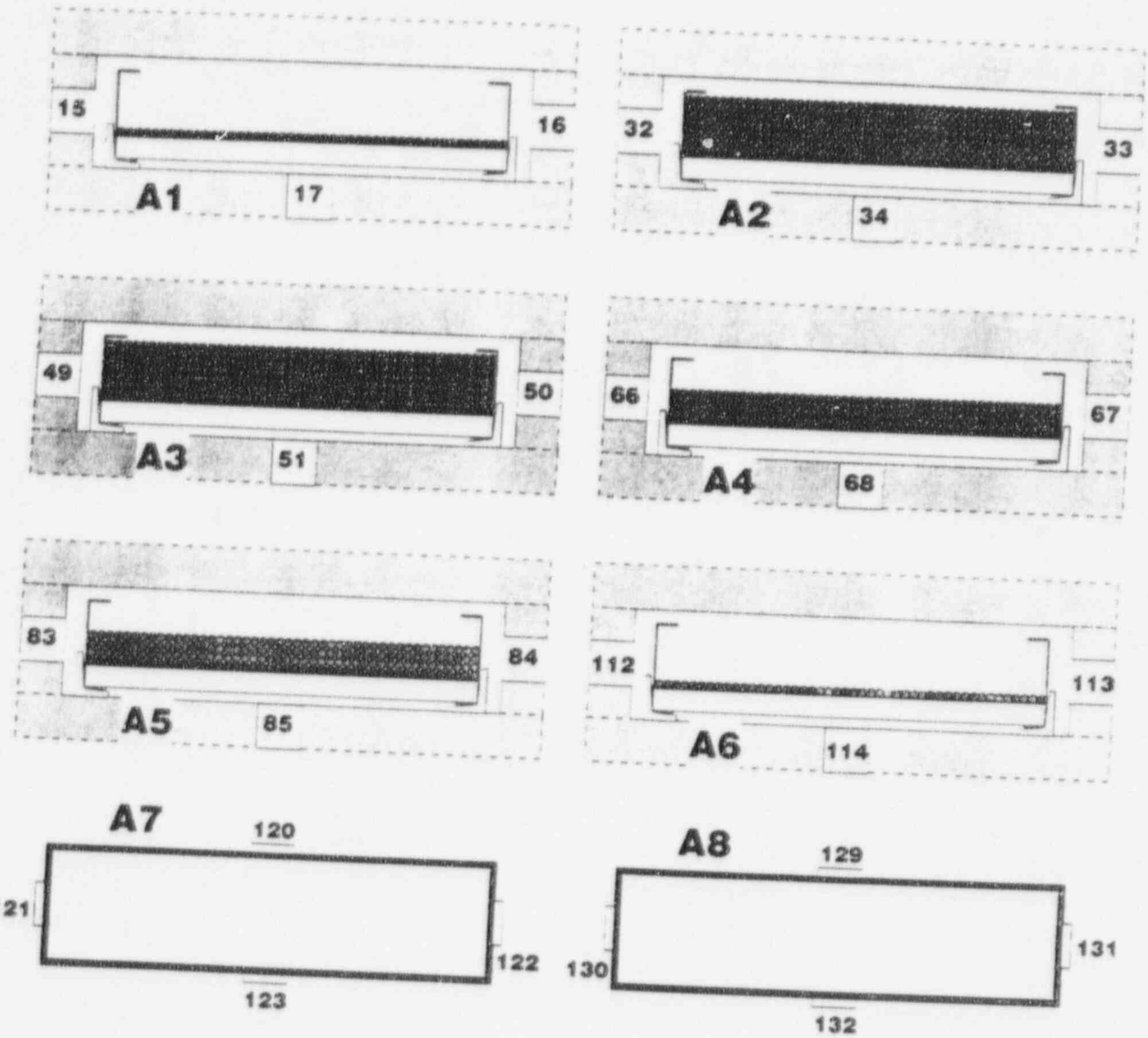


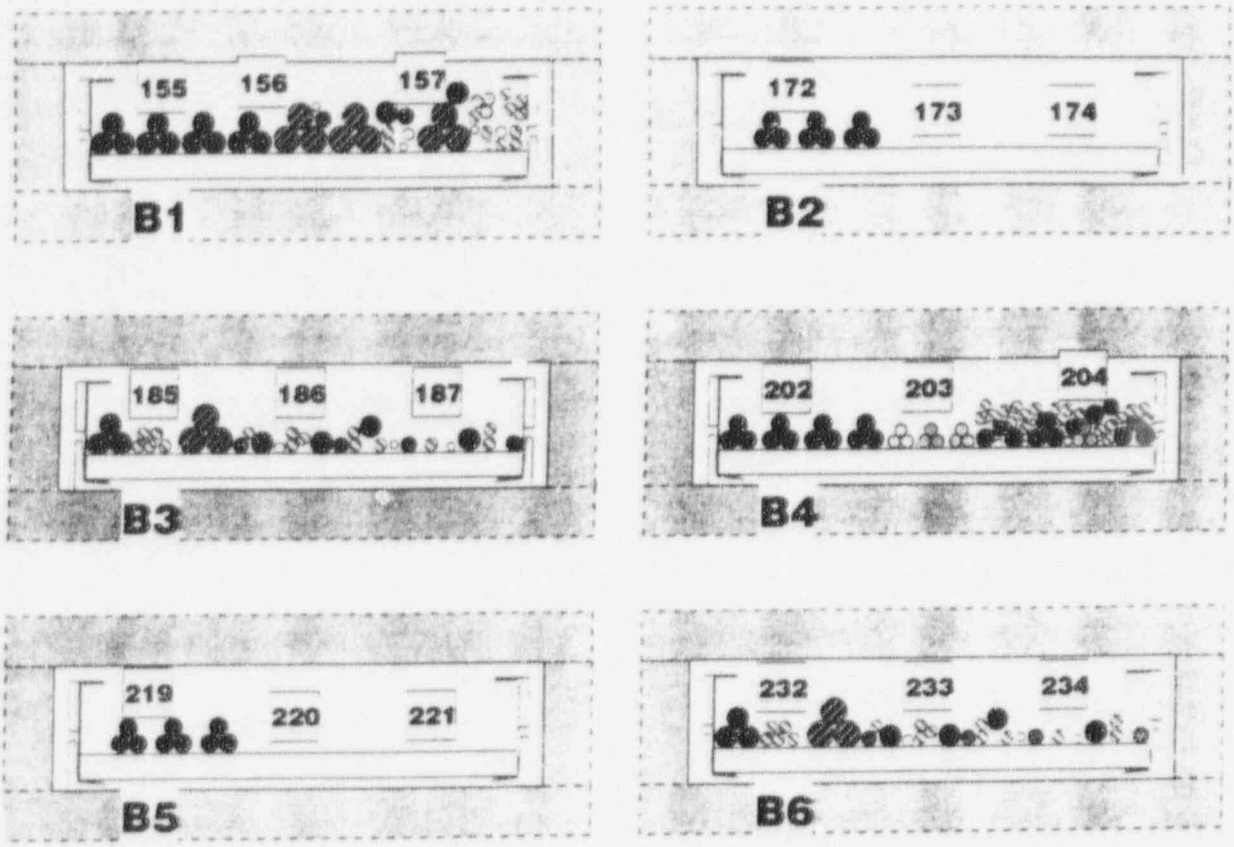
NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs on A1-A6 were placed on the unexposed face of the seal system, directly above the steel perimeter framework (on ceramic damming), at least 1" from mounting studs. TCs on A7 & A8 were placed on the outer surface of the steel penetrating sleeve, 1" above the slab surface. TCs were covered with 2" x 2" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC.
Project No. 11210-99050

TENNESSEE VALLEY AUTHORITY

Thermocouple Locations - Penetration Set A
On Unexposed Seal Face Above Frames

Drawn by: [signature] 10/20/95
Approved by: C Humphrey 10/30/95
Client Approval: JJ Pierce 10/20/95



● 300 MCM ● 4/0 ○ #2 ● 2/0 (8KV)
○ 2/0 ○ 3/C#10 ◦ #6

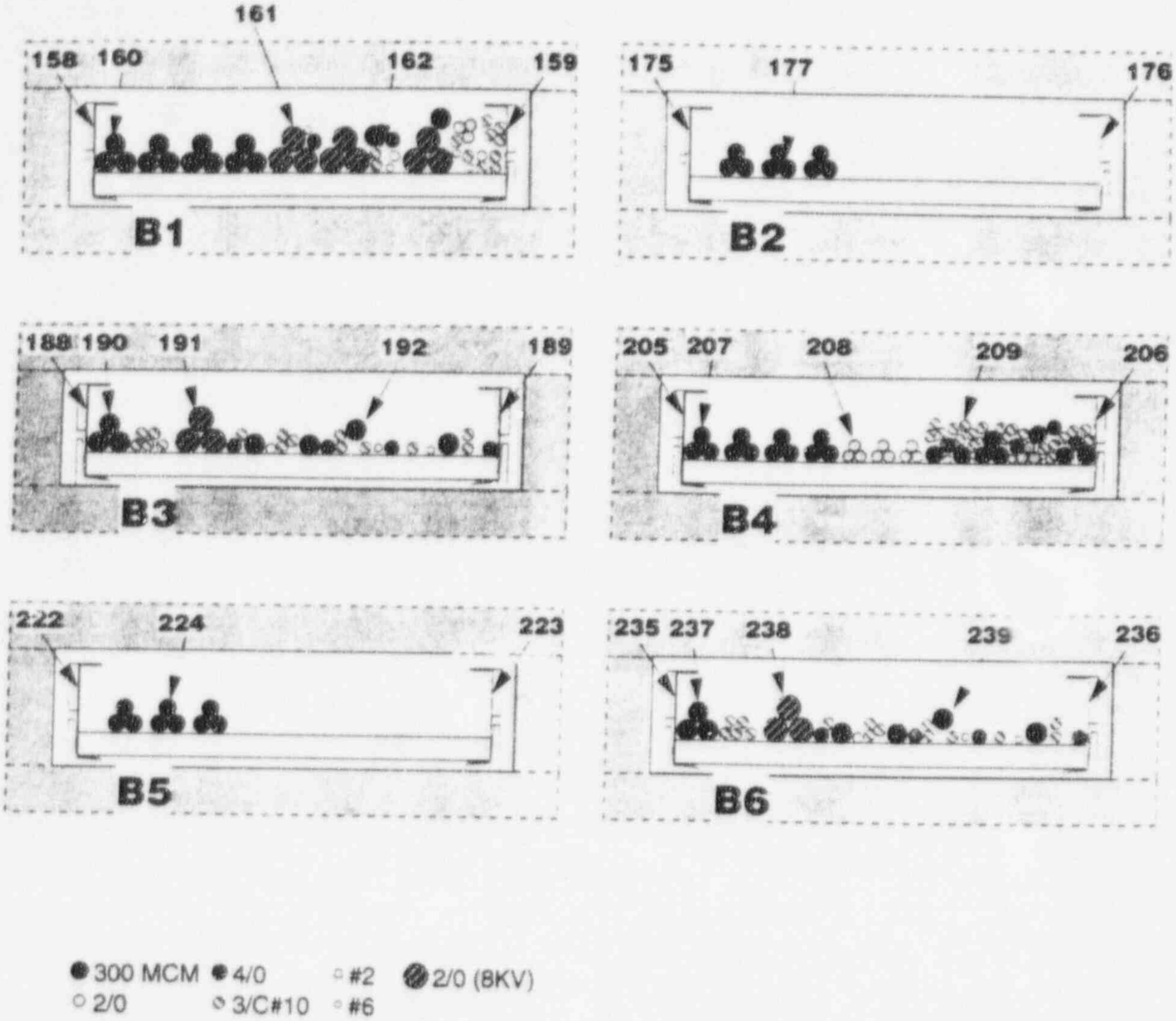
NOTE:
24 gauge, Type K, Chromel-Alumel electrically-walded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on the surface of the ceramic fiber damming board on the unexposed face of the seals. The TCs wer covered with 2" x 2" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC.
Project No. 11210-99050

TENNESSEE VALLEY AUTHORITY

Thermocouple Locations - Penetration Set B
Surface of Unexposed Seal Face

Drawn by: H. Schultz II 10/21/95
Approved by: C Humphrey 10/31/95
Client Approval: JJ Pierce 10/21/95



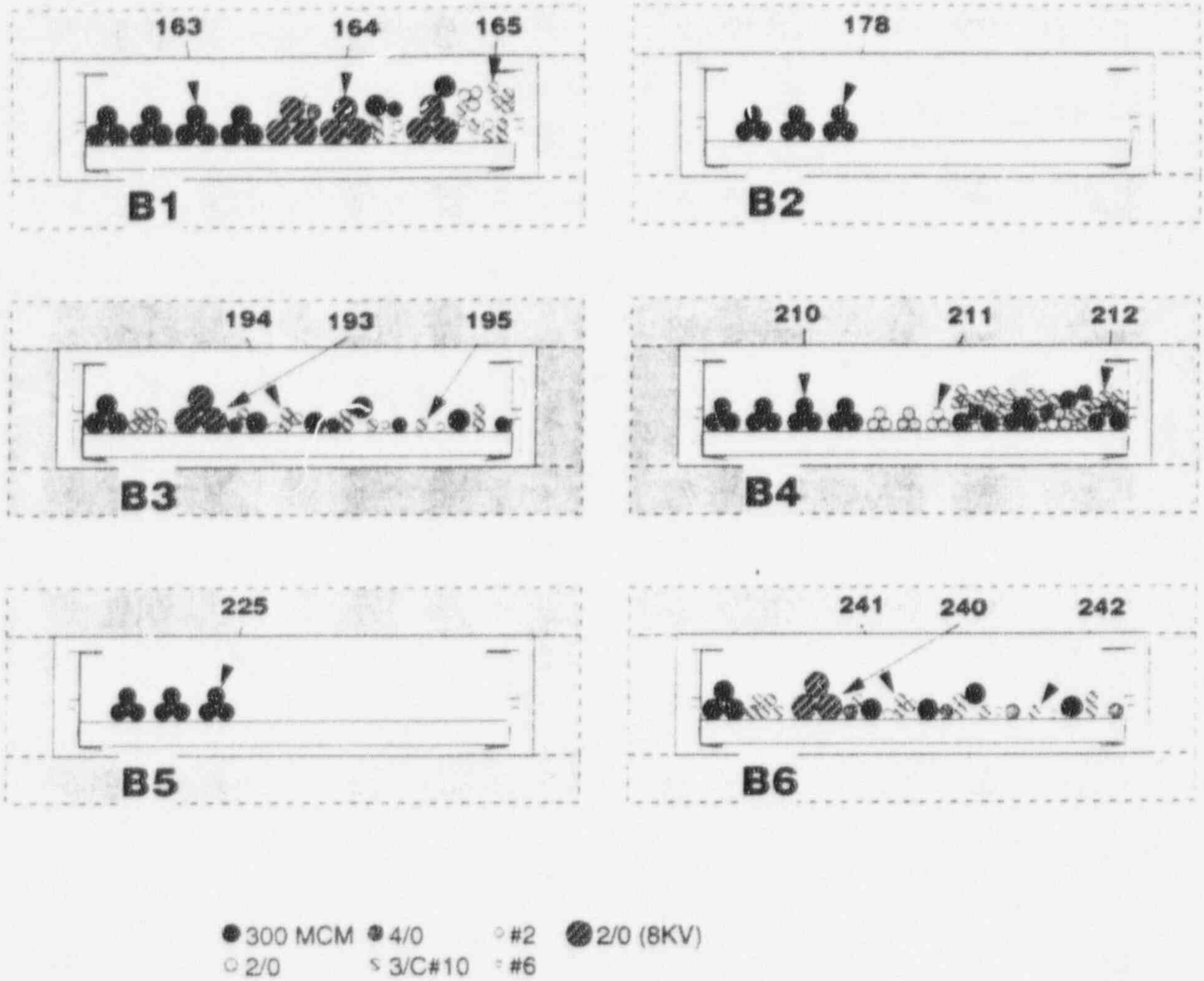
NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on cables (TC Nos. 160-162, 177, 190-192, 207-209, 224, 237-239) and on tray side rails (TC Nos. 158-159, 175-176, 188-189, 205-206, 222-223, 235-236), at the interface between the penetrants and the upper damming board. The TCs were covered with 3/4" x 3/4" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC.
Project No. 11210-99050

TENNESSEE VALLEY AUTHORITY

Thermocouple Locations - Penetration Set B
Interface Between Seal and Penetrants

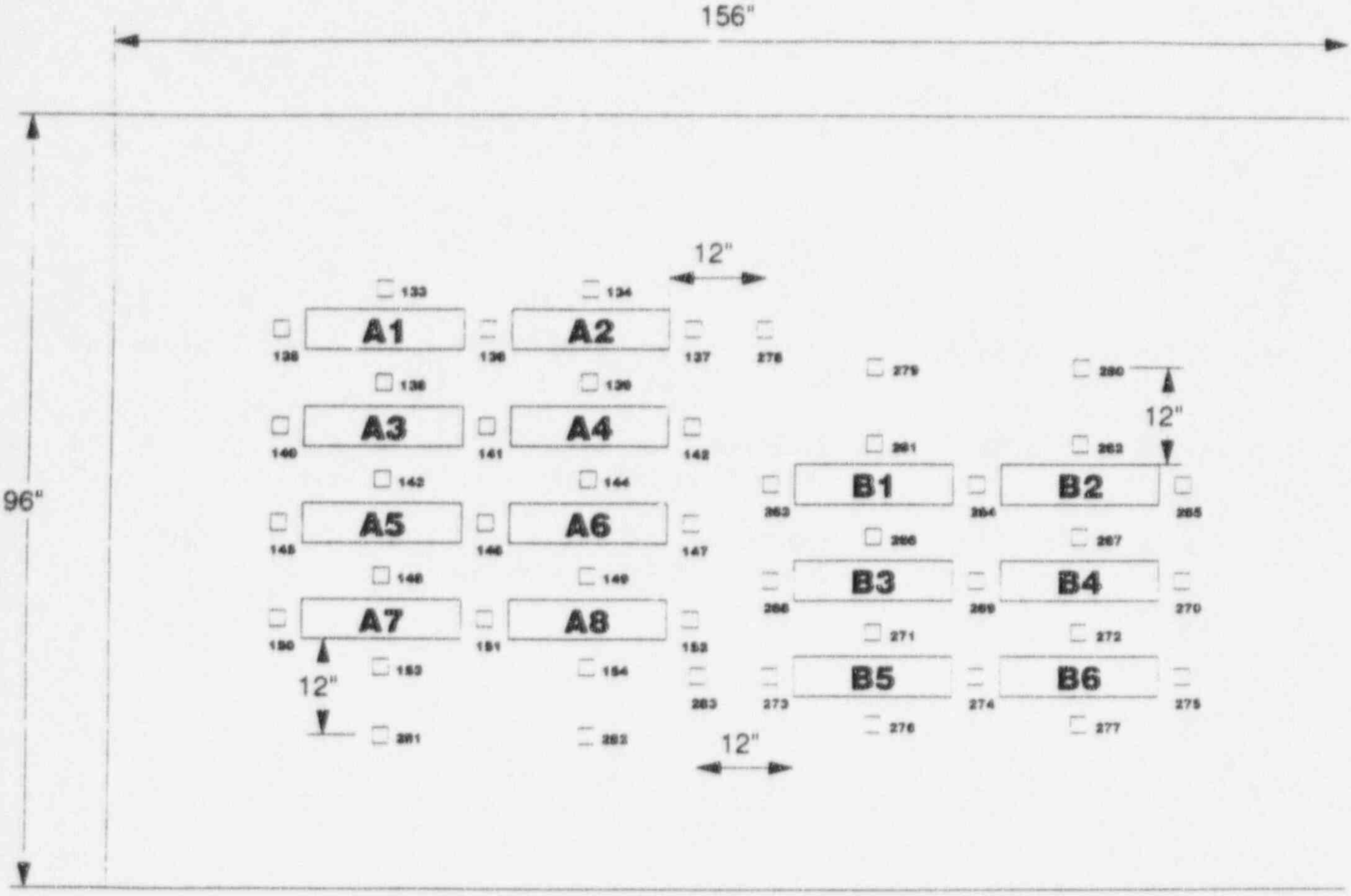
Drawn by: H. Stanton 10/21/95
Approved by: E Humphrey 10/21/95
Client Approval: JJ Pierce 10/21/95



NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on cables, 1" to 3" above the interface between the penetrants and the upper damming board. The TCs were covered with 3/4" x 3/4" x 0.4" felted fiber pads.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Thermocouple Locations - Penetration Set B On Cables Above Unexposed Seal Face
Drawn by: H. Stanton 10/21/95 Approved by: C Humphrey 10/21/95 Client Approval: JJ Pierce 10/21/95

OMEGA POINT LABORATORIES



NOTE:
24 gauge, Type K, Chromel-Alumel electrically-welded thermocouples (Special Limits of Error: ±1.1°C, purchased with lot traceability and calibration certifications) were placed in the locations illustrated above. TCs were placed on the slab surface adjacent to the blockouts, and in several locations 12" from the penetration blockouts. All thermocouples were covered with 2" x 2" x 0.4" felted fiber pads in accordance with ASTM E814.

OMEGA POINT LABORATORIES, INC. Project No. 11210-99050
TENNESSEE VALLEY AUTHORITY
Thermocouple Locations - Slab Surface
Drawn By: [signature] 10/21/95 Approved By: C Humphrey 10/21/95 Client Approval: J Pierce 10/21/95

Selected Generic Correspondence on the Adequacy of Vendor Audits and the Quality of Vendor Products

Identifier	Title
Information Notice 96-24	Preconditioning of Molded-Case Circuit Breakers Before Surveillance Testing
Information Notice 96-29	Requirements in 10 CFR Part 21 for Reporting and Evaluating Software Errors
Information Notice 96-30	Inaccuracy of Diagnostic Equipment for Motor-Operated Butterfly Valves

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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(See instructions on the reverse)

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Vol. 20, No. 2

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Licensee Contractor and Vendor Inspection Status Report
Quarterly Report
April - June 1996

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1996

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

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April - June 1996

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Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Same as above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This periodical covers the results of inspections performed by the NRC's Special Inspection Branch, Vendor Inspection Section, that have been distributed to the inspected organizations during the period from April through June 1996.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Vendor Inspection

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