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Vol. 22, No. 1

Licensee Contractor and Vendor Inspection Status Report

Quarterly Report
January – March 1998

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 Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This periodical covers the results of inspections performed between January 1993 and March 1998 by the NRC's Quality Assurance, Vendor Inspection and Maintenance Branch that have been distributed to the inspected organizations.

CONTENTS

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 NRC 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 NRC 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. This report lists selected bulletins, generic letters, and information notices, and include copies of other pertinent correspondence involving vendor issues.

INSPECTION REPORTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

January 2, 1998

Mats Tynell, General Manager
Product Center-Steam Generator Tubing
AB Sandvik Steel
SE-811 81 Sandviken
Sweden

SUBJECT: NRC INSPECTION REPORT 99901326/97-01 AND NOTICE OF NONCONFORMANCE

Dear Mr. Tynell:

On October 27-31, 1997, the U.S. Nuclear Regulatory Commission (NRC) performed an inspection at the AB Sandvik Steel (ABSS) steam generator tube manufacturing facility. The enclosed report presents the findings of that inspection.

The inspection was conducted to assess: (a) attributes and implementation of the ABSS quality assurance program in the areas of manufacturing process control, control of special processes and material identification and traceability, to ascertain whether they met NRC requirements; and (b) ABSS conformance to customer procurement requirements. In addition, the inspection reviewed the method of compliance with Part 21 of Title 10 of the Code of Federal Regulations.

Overall, the results of the inspection indicate that you have established appropriate program criteria for control of tube manufacturing and examination activities, with implementation noted generally to be good. During the inspection, the inspectors determined, however, that ABSS did not adequately implement its quality assurance program criteria for control of special processes to comply with NRC and customer requirements. Specifically, ABSS did not fully conform to its defined requirements for thermocouple locations to be used during thermal treatment of South Texas Projects, Unit 1, steam generator tubing. As a result, the specified thermal treatment hold period was commenced without the necessary assurance that the full length of steam generator tubing had reached the prescribed temperature range.

This issue is cited in the enclosed Notice of Nonconformance (NON), and the circumstances surrounding it are described in detail in the enclosed report. You are requested to respond to the nonconformance and should follow the instructions specified in the enclosed NON when preparing your response.

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,

Suzanne C. Black

Suzanne C. Black, Chief
Quality Assurance, Vendor Inspection,
and Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket No. 99901326

Enclosures: 1. Notice of Nonconformance
2. Inspection Report 99901326/97-01

cc:
Mr. R. Rehkugler
Director, Quality
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

Mr. E. Renaud
Manager, Quality Assurance
Westinghouse Electric Corporation
8301Scenic Highway
Pensacola, FL 32514

NOTICE OF NONCONFORMANCE

AB Sandvik Steel
SE-811 81 Sandviken, Sweden

Docket No.: 99901326

Based on the results of an NRC inspection conducted on October 27-31, 1997, it appears that certain of your activities were not conducted in accordance with NRC requirements:

Criterion IX of Appendix B to 10 CFR Part 50, "Control of Special Processes," states, in part, "Measures shall be established to assure that special processes, including . . . heat treating . . . are controlled and accomplished . . . using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements."

Paragraph 2.3 of Control Procedure 4636, "Long time thermal treatment," Revision 1, states, in part, "Three load thermocouples are inserted from each furnace end. The thermocouples have three different lengths (16.4, 32.8 and 52.5 Feet = 5, 10 and 16 meters) to survey the longitudinal heat distribution. Some are located at the centre and some are attached to the tube surfaces by stainless steel wire at the surface of the tube bundle in order to survey the heat distribution in cross section . . . ." Paragraph 4 of Control Procedure CP 4636, Revision 1, states, in part, ". . . When the coldest load thermocouple has reached 1319°F (715°C) the hold time timer, set at 10 hours, is started. During the hold time, the load thermocouple readings must be within 1319°-1350°F (715°C-732°C) . . . ."

Contrary to these requirements, the inspectors identified on October 29, 1997, that a 5-meter long load thermocouple was not being inserted from the front end of the furnace (as part of the survey of longitudinal heat distribution) during thermal treatment of South Texas Project, Unit 1 steam generator tubing (Sandvik Order 381-00197/201). Following the identification, AB Sandvik Steel included a 5-meter long thermocouple inserted from the front of the furnace in the next furnace charge (i.e., Charge 2056). This thermocouple proved to be the coldest load thermocouple, reaching 1319°F (715°C) 30 minutes after the load thermocouple (5-meters long from the back end of the furnace) that had previously controlled the start of the 10-hour hold period (Nonconformance 99901326/97-01-01).

Please send a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with a copy to the Chief, Quality Assurance, Vendor Inspection and Maintenance Branch, Division of Reactor Controls and Human Factors, 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

"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 noncompliance, 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 2nd day of January 1998

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report No:	99901326/97-01
Organization:	AB Sandvik Steel Steam Generator Tubing SE-811 81 Sandviken Sweden
Contact:	Mats Tynell, General Manager 46/26 263873
Nuclear Industry Activity:	Manufacture of steam generator tubing
Dates:	October 27-31, 1997
Inspectors:	Ian Barnes, Technical Assistant, Division of Reactor Safety Region IV Phillip J. Rush, Materials Engineer Materials and Chemical Engineering Branch Division of Engineering
Approved by:	Robert A. Gramm, Chief Quality Assurance and Safety Assessment Section Quality Assurance, Vendor Inspection and Maintenance Branch Division of Reactor Controls and Human Factors

Enclosure 2

1 INSPECTION SUMMARY

AB Sandvik Steel (ABSS) holds a current ASME Quality System Certificate for manufacture of ferrous and non-ferrous bars, seamless tubular products, rounds, hollows, billets and ingots, bare electrodes, strip electrodes, bare wire, hot rolled wire, and hot rolled rod. This inspection was performed at the ABSS Tube Mill 68 and supporting facilities and was focused on manufacture of Inconel 690 tubing for the South Texas Project, Unit 1 replacement steam generators.

During this inspection, the inspectors assessed conformance of manufacturing and examination activities to NRC, ASME Code, and customer requirements. Specific subject areas reviewed during the inspection were manufacturing process control, control of special processes, and material traceability and identification.

The inspection bases were as follows:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50),

- 10 CFR Part 21, "Reporting of Defects and Noncompliance," and

- ABSS's Quality Assurance Manual, Revision 24, dated November 25, 1996.

Overall, the results of the inspection indicated that ABSS had established appropriate program criteria for control of manufacturing and examination activities, with implementation noted generally to be good. The inspectors noted during the inspection, as discussed in Section 3.1 below, that Westinghouse Pensacola Plant (WPP) had not contractually imposed the requirements of 10 CFR Part 21 on ABSS. WPP had, however, required that all deviations regardless of the time of disclosure or nature of the deviation be reported to WPP for evaluation and disposition. In addition, the inspection identified that ABSS did not conform to certain NRC and customer requirements pertaining to thermal treatment of South Texas Project steam generator tubing. This nonconformance is discussed in Section 3.3.1.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of ABSS.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 10 CFR Part 21 Program

a. Inspection Scope

The inspectors reviewed the technical requirements (for South Texas Project, Unit 1 steam generator tubing) contained in Westinghouse Purchase Order 4500022778 and

Change Notices 01 through 05, in order to ascertain whether requirements had been imposed on ABSS with respect to implementing the provisions of 10 CFR Part 21, "Reporting of Defects and Noncompliance."

b. Observations and Findings

The inspectors noted that Change Notice 01, dated February 21, 1997, to Purchase Order 4500022778 added the statement, "Westinghouse assumes responsibility for 10 CFR Part 21 as applicable," to the purchase order. It was additionally noted during the review that an attachment to the purchase order, Quality Note HA0051 for Inspection Code 51, stipulated that non-U.S. vendors report all deviations regardless of the time of disclosure or nature of the deviation to WPP for evaluation and disposition. The inspectors evaluated the five deviation disposition requests that had been submitted to WPP as of the inspection and noted no conditions that would be reportable to the NRC under 10 CFR Part 21.

c. Conclusions

The responsibility for 10 CFR Part 21 has been assumed by WPP, with ABSS required to report all deviations for WPP evaluation and disposition.

3.2 Manufacturing Process Control

a. Inspection Scope

The inspectors reviewed Section 9, "Process Control," of the ABSS Quality Assurance Manual, Revision 24; and the technical requirements of Westinghouse Material Specification B163C23, Revision D, pertaining to chemistry, mechanical properties, microstructure, hydrostatic testing, cleaning, prohibited materials, and packing. The inspectors also reviewed the ABSS control procedures that had been developed to satisfy the Westinghouse technical requirements, and observed their implementation in the production process. The reviewed procedures were: Control Procedure 4601, "Contamination surveillance program," Revision 1; Control Procedure 4611, "Melting and metal working," Revision 0; Control Procedure 4612, "Cleaning processes prior to final pilgering," Revision 0; Control Procedure 4622, "Control of micro inclusions on bars," Revision 0; Control Procedure 4631, "Cleaning after final cold pilgering," Revision 4; Control Procedure 4644, "Packing," Revision 2; Control Procedure 4657, "Hydrostatic pressure test," Revision 0; Control Procedure 4668, "Micro tests," Revision 0; Control Procedure 4669, "General and intergranular attack test," Revision 0; and Control Procedure 4671, "Intergranular corrosion test," Revision 0.

b. Observations and Findings

The inspectors verified that the content of the ABSS control procedures was consistent with the technical requirements of Westinghouse Material Specification B163C23, Revision D. Performance of cleaning processes prior to and after final cold pilgering was found to conform to procedural requirements. Production personnel were observed, without exception, to wear gloves during handling of final dimension tubing and exhibited commendable rigor in their efforts to preclude contamination of the tubing surfaces. The inspectors noted from review of a sample of chemistry and mechanical property data that the data conformed to the requirements of Westinghouse Material Specification B163C23, Revision D. The limited heat-to-heat variation in tubing composition and mechanical properties were considered by the inspectors to be indicators of good melting and manufacturing process controls. The inspectors determined from review of metallographic and laboratory data that the carbide morphology, grain size, and resistance to intergranular corrosion conformed to Westinghouse Material Specification B163C23, Revision D, requirements. During this review, the inspectors independently reviewed available specimens to confirm the accuracy of the ABSS photomicrographs. The inspectors were, however, unable to preselect specific specimens for review due to the ABSS practice of discarding specimens after completion of examination.

Observation of hydrostatic testing indicated testing conformed to the test pressure and holding time requirements of Control Procedure 4657, Revision 0, with the operator noted to comply with the procedural criteria for removing residual water. The inspectors also verified that the demineralized water used for the testing satisfied the chemistry and conductivity requirements of the procedure.

The inspectors noted from review of Control Procedure 4601, Revision 0, that ABSS had attempted to comprehensively define the materials that come into physical contact with final size tubes. The procedure was verified by the inspectors to appropriately reflect the material prohibitions of Westinghouse Material Specification B163C23, Revision D. The inspectors toured Tube Mill 68 to confirm that the procedure accurately reflected contact materials. With the exception of one minor discrepancy, the inspectors found that the procedure appropriately described the materials that come into contact with final size tubing. The discrepancy pertained to the omission of the contact of plywood in intermediate storage racks. ABSS immediately issued Revision 1 to Control Procedure 4601 to incorporate the information regarding the intermediate storage racks.

c. Conclusions

The procedural controls used for manufacture of the South Texas Project, Unit 1, tubing were consistent with the technical requirements of Westinghouse Material Specification B163C23, Revision D. Overall procedural implementation was good, with the limited heat to heat variation in tubing composition and mechanical properties considered to be indicators of good melting and manufacturing process controls.

3.3 Control of Special Processes

3.3.1 Material Heat Treatment

a. Inspection Scope

The inspectors reviewed the tubing heat treatment requirements contained in: Westinghouse Material Specification B163C23, "Thermally Treated Alloy UNS N06690

(Alloy 690) Tubing for South Texas Unit No. 1 Replacement Steam Generators (Section III-NB, SB-163, Code Case N-20-3)," Revision D; and Section 9.3, "Process Control," of the ABSS Quality Assurance Manual, Revision 24. The inspectors also reviewed the ABSS heat treatment procedures that had been developed to implement the requirements of Westinghouse Material Specification B163C23, Revision D, and observed their implementation during the production process. The reviewed heat treatment procedures were: Control Procedure 4632, "Final bright annealing," Revision 0; Control Procedure 4636, "Long time thermal treatment," Revision 1; and Control Procedure 4642, "Stress relieving," Revision 1.

b. Observations and Findings

The inspectors verified that the technical requirements of the ABSS heat treatment procedures were consistent with Westinghouse Material Specification B163C23, Revision D. One discrepancy had been previously identified by ABSS in a Deviation Disposition Request dated September 25, 1997. The deviation, which had been dispositioned "use-as-is" by WPP, identified that the Westinghouse material specification requirement for monitoring the hottest and coldest tubes during bright annealing was not valid for the type of furnace used by ABSS. The inspectors reviewed the furnace design and concurred with the WPP disposition.

The inspectors observed final annealing operations that were being performed on South Texas Project, Unit 1, tubing and noted no departures from procedural requirements with respect to temperature, travel speed, and dew point of the protective hydrogen gas.

During review of a sample of thermal treatment charts for South Texas Project, Unit 1, tubing, the inspectors noted that the temperature printout for the No. 1 position consistently showed values (i.e., 690-695 °C) during the hold period that were below the minimum specified 715 °C. The inspectors ascertained that the No. 1 position corresponded with a thermocouple that was the closest to the front end of the furnace. This thermocouple was located closer to the end of the furnace than tubing in furnace charges. While assessing the potential significance of the 690-695 °C temperatures, with respect to the temperatures reached by tubing at locations closest to the front end of the furnace, the inspectors found that ABSS was not fully complying with the load thermocouple requirements of Control Procedure 4636, Revision 1.

Section 2.3 of Control Procedure 4636, Revision 1, states, in part, ". . . Three load thermocouples are inserted from each furnace end. The thermocouples have three different lengths (16.4, 32.8 and 52.5 feet=5, 10 and 16 meters) to survey the longitudinal heat distribution. Some are located at the centre and some are attached to the tube surfaces by stainless steel wire at the surface of the tube bundle in order to survey the heat distribution in cross section . . . ." The inspectors noted that ABSS was inserting two 10-meter long thermocouples from the front end of the furnace, rather than the required 5-meter and 10-meter long thermocouples. The inspectors determined that the longest tubes in the tube bundle, if uniformly positioned on the furnace car, would extend to approximately 3 meters from each end of the furnace. In response to the notification by the inspectors of the procedural noncompliance, ABSS included a thermocouple at

5-meters from the front end of the furnace in the next thermal treatment furnace charge (i.e., Charge 2056). The inspectors noted from review of the furnace chart for Charge 2056 that this thermocouple was the slowest to reach the hold range, taking approximately 30 minutes longer than the previous coldest load thermocouple (i.e., 5 meters from back end of the furnace). Section 4 of Control Procedure 4636, Revision 1, states, in part, ". . . When the coldest load thermocouple has reached 1319°F (715°C) the hold time timer, set at 10 hours, is started . . . ." The inspectors concluded from review of the temperature data that the potential consequence of not inserting a 5-meters long thermocouple from the front end of the furnace was a portion of the tubing length could receive a 9.5 hour soak in the thermal treatment range, rather than the 10 hours specified by Control Procedure 4636, Revision 1. A deviation disposition request was prepared by ABSS, in response to the inspectors' observations, in order to initiate an evaluation of the effects of the nonconforming heat treatment practice on previously thermally treated tubes. The failure to appropriately use the thermal treatment procedure, as required by Criterion IX, "Control of Special Processes," of Appendix B to 10 CFR Part 50, was identified as Nonconformance 99901326/97-01-01.

c. Conclusions

The ABSS procedural requirements for performance of bright annealing, thermal treatment, and stress relief of South Texas Project, Unit 1, tubing were consistent with Westinghouse Material Specification B163C23, Revision D. Observed bright annealing operations conformed to the requirements of Control Procedure 4632, Revision 0, with respect to temperature, travel speed, and dew point of the hydrogen protective gas. A nonconformance was identified in regard to the failure to conform to the thermocouple location requirements of Control Procedure 4636, Revision 1.

3.3.2 Nondestructive Examination

a. Inspection Scope

The inspectors reviewed the ABSS nondestructive examination procedures that had been developed to satisfy the requirements of Westinghouse Material Specification B163C23, Revision D, and observed their implementation during the production process. Control Procedure 4651, "In-service inspectability," Revision 1, specified the procedure to be used for assessing the signal-to-noise ratio of the tubing. Control Procedure 4651, "Ultrasonic test," Revision 1, and Control Procedure 4652, "Eddy-Current test," Revision 2, defined the respective ultrasonic and eddy current examination requirements for detection of discontinuities in straight tubes prior to thermal treatment. Dimensional requirements for straight tubes were contained in Control Procedure 4653, "Continuous wall thickness measurement," Revision 0, and Control Procedure 4654, "Continuous outside diameter measurement," Revision 0. process. Eddy current examination requirements for the final inspection after bending of U-tubes were specified in Control Procedure 4659, "Multi Frequency EC-examination of U-bent tubes (MIZ-18)," Revision 3.

b. <u>Observations and Findings</u>

The inspectors verified that the eddy current data analysts observed conducting Control Procedure 4659, Revision 3, were appropriately qualified. Records for the qualification of these and other analysts were available and observed to be appropriately approved by inspection oversight authorities. Inspection parameters (e.g., probe type, acquisition equipment, test frequencies) were documented in the ABSS control procedures in accordance with Westinghouse Material Specification B163C23, Revision D; and Section V, Article 8, and Section XI, Appendix IV, of the ASME Boiler and Pressure Vessel Code. The inspectors observed that the tube signal-to-noise ratios for a number of straight tubes examined during the inspection were in excess of the Westinghouse specified minimum value of 15:1. In addition, ABSS made permanent records of these measurements in accordance with Section 6.7.5 of Westinghouse Material Specification B163C23, Revision D.

The inspectors noted strengths in the practices employed by the data analysts in evaluating the eddy current data from the inspection of U-bent tubes. Specifically, the analysts undertook additional conservative measures not required by the analysis guidelines to ensure that tubes did not contain unacceptable discontinuities. These actions included visually confirming the presence of manufacturing burnish marks (MBMs) during the initial stages of tube production, assessing the root cause of absolute drift response signals, and locating indications on the low frequency channels using signals from the tube support table structure.

The inspectors noted that two different diameter bobbin coil probes, 0.520-inches and 0.560-inches, had been used to inspect U-tubes in accordance with Control Procedure 4659, Revision 3. In addition, the assessment of tube signal-to-noise utilized a bobbin coil probe that was a different type to the two sizes used in this control procedure. The inspectors further noted that Westinghouse Material Specification B163C23, Revision D, did not preclude the use of different probes in the production process, and also provided no criteria on allowable probe wear. ABSS had also not restricted the amount of probe wear. The inspectors considered the absence of probe wear criteria and the use of various probe sizes as potential contributors to inconsistencies in the eddy current data obtained during tube production. Such inconsistencies could complicate the comparison of tube production data with that obtained during preservice inspection of the completed steam generators.

ABSS, in conjunction with the licensee for the South Texas Project, Unit 1, established a 6 volt criterion to identify tubes with MBM indications that locally reduced wall thickness in excess of the 5 percent limit imposed by Westinghouse Material Specification B163C23, Revision D. The inspectors reviewed the technical bases for the 6 volt criterion and identified that the voltage level for 5 percent through-wall MBM indications was slightly nonconservative (5.91 volts versus 6 volts), did not contain any margin for data scatter due to the uncertainty in the analysis and acquisition of data, and did not take into account the use of bobbin coil probes with different diameters, as discussed previously. Because of these deficiencies, the inspectors concluded that this criterion by itself would not ensure that all MBMs identified during the inspection would have acceptable wall thicknesses in the area of the indication. However, the inspectors observed that, based on the extremely limited metal removal that would be anticipated to occur during buffing operations following thermal treatment, the eddy current methodology used for examining MBMs did not appear to represent other than a minor technical concern.

The inspectors observed scratches on the surface of the finished U-bend tubes. Section 7.4 of Westinghouse Material Specification B163C23, Revision D, states that, "tubing shall be . . . free from seams, cracks, tears, laminations, laps, pitting and other injurious imperfections." The ABSS control procedures did not include any specific guidance for assessing when imperfections, such as the noted scratches, could be considered to be injurious imperfections. During the tube inspections in the final stages of production, workers were observed attempting to remove these scratches through a manual buffing process. Despite smoothing the scratches slightly during this stage of production, some visible axial imperfections remained in the tubes that were not detected in the final eddy current inspections. The inspectors noted that degradation history for mill annealed Inconel 600 tubing indicates that surface scratches can often be the initiation sites for service induced tube degradation. However, the inspectors concluded that the absence of specific criteria in the ABSS control procedures and in the Westinghouse specification for determining when imperfections were injurious was a technical deficiency rather than a procedural nonconformance.

c. Conclusions

The ABSS nondestructive examination procedures and practices were found to be consistent with the requirements specified in Westinghouse Material Specification B163C23, Revision D. The inspectors identified minor technical concerns regarding MBM inspection criterion, maintaining consistency in the eddy current inspection data, and the absence of criteria for classifying what surface imperfections were considered injurious.

3.4 Material Traceability and Identification

a. Inspection Scope

The inspectors reviewed the material traceability and identification requirements contained in: Westinghouse Material Specification B163C23, Revision D; Section 8, "Product Identification and Traceability," of the ABSS Quality Assurance Manual, Revision 24; and related ABSS control procedures. Personnel practices for maintaining material traceability and identification were also observed at various stages of manufacture production to verify consistency with program requirements. With the exception of activities ongoing after final pilgering, the observed ABSS practices for maintaining material traceability and identification were conducted on material not destined for the production of tubing for the South Texas Project replacement steam generators. However, ABSS personnel stated that the material traceability and inspection practices observed were the same as those used for the production of steam generator tubing.

b. Observations and Findings

Control Procedure 4603, "Identification and traceability," Revision 2, included the general requirements for ensuring that material identification is maintained throughout the production process. The procedure, however, did not specifically indicate how identification and traceability are maintained at each stage of the process. The inspectors assessed the practices utilized by personnel at several stages of production through direct observation and verified the identification of materials between process steps. The inspectors observed that the production practices were consistent with the requirements of Section 8 of the ABSS Quality Assurance Manual, Revision 24, and that all materials were appropriately labeled when possible. During stages of production where actual labeling on the material was not possible (e.g., hot rolling, extrusion, pilgering), operators employed practices for tracing the material through the stage of process that should minimize the potential for mislabeling or a loss of material identification.

ABSS used a labeling system where material was tracked by a lot identification number. Lots of material are from a single heat and assigned a unique lot number after cutting round bars into extrusion billets as specified in Control Procedure 4602, "Lot identification," Revision 0. After final pilgering, the tubes within a single lot were worked, in general, separately from tubes in other lots during each stage of production. The inspectors observed discernible breaks between processes from one lot to the next. In instances where lots were continuously processed, the inspectors noted that production personnel were tracking tubes within each lot and identified the break between tubes in separate lots.

c. Conclusions

The procedures and practices utilized during the production of tubing appeared adequate for ensuring proper material identification and traceability throughout the process, and were consistent with the requirements of Section 8 of the ABSS Quality Assurance Manual, Revision 24.

3.5 Entrance and Exit Meetings

At the entrance meeting on October 27, 1997, the inspectors discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with ABSS management. In the exit meeting on October 31, 1997, the inspectors discussed their findings and observations. Westinghouse Material Specification B163C23, Revision D, was identified as containing proprietary information. No information was included from this document in the inspection report that was considered proprietary. No information was identified by ABSS during the inspection as being considered proprietary.

4 PARTIAL LIST OF PERSONNEL CONTACTED

ABSS Corporate

Bertil Larsson, Quality Assurance Manager

ABSS Product Center Steam Generator Tubing

Mats Tynell, General Manager
Göran Björkman, Production Manager
Per-Olof Lund, Quality Assurance Manager
Hans Törnblom, Manager, Technique and Development
Benny Pettersson, Manager, Marketing
Jan-Erik Bohman, Manager, Nondestructive Examination

STP Nuclear Operating Company

H. Gunther Domschke, Quality Assurance Staff Specialist

Westinghouse Pensacola Plant

James Allen, Senior Quality Assurance Engineer
Richard Fremgen, Senior Engineer
Magnus Larsson, Resident Inspector (SAQ)

ITEMS OPENED

Opened

99901326/97-01-01	Para. 3.3.1	NON	Inadequate control of thermal treatment

March 27, 1998

Mr. Jerome I. Rosenstock, Chief Executive Officer
Allied Group
520 Hertzog Blvd
P.O. Box 60670
King of Prussia, PA 19406

Dear Mr. Rosenstock:

On March 19, 1998, the U.S. Nuclear Regulatory Commission (NRC) performed an inspection at the Allied Group, Allied Nut & Bolt Company, Inc. (Allied) facility in King of Prussia, Pennsylvania. The enclosed report presents the findings of that inspection. The inspection was conducted to review selected portions of your quality assurance program, and its implementation, as it relates to the supply of safety-related fasteners to the nuclear industry. This inspection specifically focused on activities related to the supply of phenolic graphite coated fasteners to the Public Service Electric and Gas Company. The inspectors assessed Allied's conformance to their customer's procurement requirements and compliance with NRC regulations. Within the scope of this inspection, we found no instance in which Allied failed to meet NRC requirements.

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,
Original signed by
Suzannne C. Black, Chief
Quality Assurance, Vendor Inspection
and Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket No. 99901093

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report no:	99901093/98-01
Organization:	Allied Group
Contact:	David Perkins, Quality Assurance Manager (610) 275 2200
Nuclear Activity:	Manufacturer and supplier of threaded fasteners, ferrous and nonferrous bars, fittings, flanges, and other products used in nuclear applications.
Dates:	March 19, 1998
Inspectors:	Gregory C. Cwalina, Senior Operations Engineer James A. Davis, Materials Engineer
Approved by:	Robert A. Gramm, Chief Quality Assurance Section Quality Assurance, Vendor Inspection and Maintenance Branch Division of Reactor Controls and Human Factors

Enclosure

1 INSPECTION SUMMARY

The NRC inspectors examined the circumstances surrounding the supply of nuclear safety-related phenolic graphite coated fasteners to the Public Service Electric and Gas Company (PSE&G), the licensee for the Salem nuclear plant. Specifically pertaining to the above, the inspectors reviewed the implementation of selected portions of Allied Group's (Allied) quality assurance (QA) program for compliance with the requirements of 10 CFR 50, Appendix B. The inspectors also reviewed the implementation of Allied's program for identifying and evaluating deviations and reporting of defects and failures to comply under the requirements of 10 CFR Part 21.

The inspection bases were:

- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
- 10 CFR Part 21, "Reporting of Defects and Noncompliance."

During this inspection, no violations or nonconformances were identified.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

Violation 99901093/87-01-01(Closed)

During a June 1987 inspection of Allied Nut & Bolt Company, the inspectors found that Allied failed to post copies of Section 206 of the Energy Reorganization Act of 1974.

During this inspection, the inspectors observed Allied's postings and determined that they met the requirements of 10 CFR Part 21, including posting of Section 206.

Violation 99901093/87-01-02 (Closed)

During a June 1987 inspection of Allied Nut & Bolt Company, the inspectors found that Allied did not pass the requirements of 10 CFR Part 21 on to certain sub-tier vendors.

During this inspection, the inspectors observed procurement documents to sub-tier vendors and noted that the requirements of Part 21 had been appropriately passed on.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Description of Facilities and Activities

The Allied Group consists of two organizations, the Allied Nut & Bolt Company, Inc. and the Allied Precision Machine Company, Inc. Allied has been granted a Quality System Certificate (QSC-528, expires May 3, 2000) by the American Society of Mechanical Engineers (ASME) as a Material Organization (MO), for manufacturing and supplying ferrous and nonferrous bars, threaded fasteners, seamless fittings, flanges, and other products. Inspection and test capability includes complete dimensional and visual inspection, tensile and hardness testing, and chemical analysis capability for carbon and low alloy steels. Nondestructive examination and heat treating are subcontracted to approved suppliers.

3.2 Review of Allied's 10 CFR Part 21 Program and its Implementation

The inspectors reviewed Allied Procedure QAI:1, "10 CFR Part 21 Procedure For Reporting of Defects and Noncompliance," Revision 3, April 8, 1997. The practices for identifying and reporting deviations and the time frames for evaluation and notification are appropriately described in QAI:1. The procedure requires any Allied employee who identifies a deviation to document the deviation in Section I of the Allied Part 21 Report form (included in QAI:1). The responsibility for documenting the results of deviation evaluations and the need to inform customers and/or the NRC are also described. The inspectors noted some minor weaknesses in procedure QAI:1. The weaknesses were identified to the Allied QA manager who stated that appropriate revisions would be incorporated.

The inspectors also observed 10 CFR Part 21 postings at the manufacturing facility and found them to be consistent with the current requirements.

3.3 Review of Supply of Phenolic Graphite Coated Fasteners to PSE&G

a. Inspection Scope

The NRC inspectors reviewed the document files related to the supply of phenolic graphite coated fasteners to PSE&G for use at the Salem nuclear generating station.

b. Observations and Findings

PSE&G Purchase Order On June 15, 1990, PSE&G issued purchase order (PO) P2-361636 to Allied Nut & Bolt Company, Inc. The PO called for the supply of numerous fasteners (nuts, bolts and screws) coated with a "phenolic graphite." The PO specified that the coating was to be supplied by the G* Chemical Corporation (G*) c/o the King Finishing Company (King). The original PO specified that the order was nuclear safety related and that 10 CFR Part 21 applied. The PO also specified QA requirements in accordance with QAF-19 No. QC-3847. QC-3847 listed QA program requirements to include PSE&G's QA program; 10 CFR Part 50, Appendix B; 10 CFR Part 21; and ASME NCA-3800.

The PO was filled by Allied in the following manner. Allied supplied the specified fasteners to King. King applied the coating material, supplied by G*, to the fasteners in accordance with G*'s procedures and subject to G* QA oversight. The coated fasteners were returned to Allied prior to shipment to PSE&G.

Allied Survey of G* In anticipation of the PO, Allied performed a survey of G* at the coating subcontractor, King Finishing Company on June 1, 1990. Results of the survey were included in an Allied internal memorandum dated June 11, 1990. The survey results include the following statements:

> G*Chemical Corp. is the manufacturer of a proprietary bonded solid film lubricant coating system that is applied by subcontracted (job shop) finishing companies.
>
> Prior to this survey G*Chemical was qualified by specific reference to their product on Allied's customer purchase orders.
>
> The purpose of this survey was twofold; to check the control of Code material during the coating process and to review G*Chemical's quality program to assure that the coating materials being applied comply with the original coating materials tested and accepted as described in G*Chemical's report, "PEPCOAT" A Study in protection and performance.
>
> ...discussions with Mr. King [QA Manager - G*] and Mr. Heisel [Corrosion Engineer - G*] indicate that each batch of PEPCOAT is analyzed for elements or compounds considered to be deleterious in the Nuclear Power Industry, but the resistance and lubricity testing, originally performed in 1982 through 1984 has not been checked since then.

The survey report raised some questions regarding the coating material. First, Allied lacked assurance that current batches of the coating "perform in a manner similar to the performance obtained when the lubricity and corrosion resistance of the original product was tested." Second, Allied was concerned that the coefficient of friction may vary from batch to batch, which may have an effect on calculated torque values.

<u>PSE&G Purchase Order Revisions</u> Based upon the survey, Allied communicated their concerns to PSE&G on June 13, 1990. In that letter, Allied stated that they were unable to evaluate the consistency or quality of the product because it is a proprietary product. Alternatively, periodic testing could be performed, however, G* indicated they had not performed any tests since the original product was tested and they would not perform any until they mixed their next batch. Allied informed PSE&G that Allied found that response unacceptable and halted coating of Allied material.

In order to resolve the issue, Allied requested PSE&G to inform them, in writing, that the coating materials were not safety-related and that Allied would not be held responsible for the quality and performance of the PEPCOAT coating.

PSE&G responded in a telefax from J. Harper dated June 14, 1990. That fax supplied QAF-19 No. QC3847A which would be applicable to all POs in relation to PEPCOAT. The fax noted that PSE&G did not require traceability of the coating and that Allied was only acting as PSE&G's agent in obtaining PEPCOAT and had no liability with regard to

PEPCOAT. Subsequently, QC3847A, which removed specific QA requirements from the coating material, was amended to the PO.

Allied Purchase Order to G* Allied obtained the coating service from G* through several POs. The inspectors reviewed POs Q19622 (June 20, 1990), Q19803 (July 13, 1990), and Q20046 (August 8, 1990). The inspectors noted that all 3 POs required G* to certify that the corrosion resistance and lubricity of the applied lot was equivalent to that described in the report, "PEPCOAT A Study in Protection and Performance." In addition, the POs required a Certified Material Test Report and a Certificate of Compliance (CoC). Finally, the PO invoked the requirements of Part 21. In filling the order, G* supplied a CoC for all 3 POs (dated June 29, 1990, July 23, 1990, and August 10, 1990) attesting that PEPCOAT "meets the corrosion resistance and lubricity as described in the Pepcoat Technical Manual."

c. Conclusions

PSE&G issued a safety-related PO to Allied which originally included the phenolic graphite coating. Based upon a survey conducted by Allied, PSE&G removed QA requirements relating to the coating from the PO. Therefore, the inspectors concluded that PSE&G relieved Allied from any material or regulatory responsibility relating to the coating.

3.4 Qualification of PEPCOAT

a. Inspection Scope

During this inspection, the inspectors reviewed documentation regarding the qualification of PEPCOAT as a nuclear grade lubricant.

b. Observations and Findings

Based upon discussions with Allied personnel, the inspectors determined that, in 1994, Allied was trying to fill a PO for phenolic graphite coated fasteners on an expedited time frame. Their contacts with G* indicated that material was not readily available to complete the PO requirements in a timely manner. Allied contacted another lubricant manufacturer, E/M Corporation (E/M), to see if E/M supplied a graphite coating that met the requirements of the nuclear industry. E/M supplied Allied with literature regarding two of their products, Everlube 6122, a bonded solid film lubricant designed to meet the needs of the nuclear industry (i.e., formulated without elements considered deleterious to nuclear power plant components), and Everlube 6120, a similar material designed for non-nuclear applications.

Allied noted that the technical data sheet provided by E/M for Everlube 6120 was identical to the data sheet provided by G* for PEPCOAT. However, the technical data sheet for Everlube 6122, the nuclear grade lubricant, was different in some key aspects. The inspectors reviewed the subject data sheets and confirmed Allied's findings. Allied then noted that the qualification report provided by G* (PEPCOAT, A Study in Protection and Performance) was similar to one provided by E/M for Everlube 6122. Allied was informed by E/M that Everlube 6122 had been supplied to Precision Engineered Products Company (subsequently acquired by G*) who marketed the product as PEPCOAT 6122. Allied was later informed that E/M had not been selling or supplying Everlube 6122 for several years.

After receiving this information, Allied informed PSE&G by memorandum of August 24, 1994, that the coating PSE&G had been receiving from G* was not the same coating that was tested for nuclear use. Allied also informed PSE&G that Allied would no longer supply parts coated with PEPCOAT as supplied by G*. The memorandum also noted that Allied could not perform an evaluation in accordance with Part 21.

c. Conclusions

Based upon documentation reviewed, the inspectors determined that there is some doubt as to whether PEPCOAT, as marketed by G*, is a nuclear grade coating.

The inspectors determined that Allied's memorandum to PSE&G constituted a courtesy notification with respect to Part 21. (Note: the inspectors determined that a notification was not required because PSE&G had relieved Allied of any responsibility for the coating material.)

PARTIAL LIST OF PERSONS CONTACTED

Allied Group

J. Rosenstock, Chief Executive Officer
M. Rosenstock, Executive Vice President
D. Perkins, Quality Assurance Manager

Conrad O'Brien Gellman & Rohn, P.C.

J. Guernsey

ITEMS OPENED, CLOSED AND DISCUSSED

Closed

99901093/87-01-01	VIO	Failure to post Section 206
99901093/87-01-02	VIO	Failure to pass Part 21 to subtier vendors

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

January 28, 1998

Mr. E.R. Kane, Vice President
Engineering Services
Framatome Technologies, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

SUBJECT: NRC INSPECTION OF FRAMATOME TECHNOLOGIES, INC. (INSPECTION REPORT NO.: 99901300/97-01)

Dear Mr. Kane:

This letter forwards the report of the U.S. Nuclear Regulatory Commission (NRC) inspection of Framatome Technologies, Inc. (FTI), conducted May 19-21, 1997, at your Lynchburg, Virginia, Engineering Facility, and with further reviews conducted through September 1997, by Mr. Stephen Alexander of this office and by Mr. Barry Elliot, Ms. Meena Khanna, Ms. Andrea Lee, and Mr. Ken Karwoski of the Materials and Chemical Engineering Branch; Division of Engineering; Office of Nuclear Reactor Regulation.

The primary purposes of the inspection were to (1) determine the availability and traceability of all of the data at FTI regarding the copper and nickel content of submerged-arc welds involving certain heats of weld material used in the fabrication of reactor pressure vessels (RPVs) by Babcock & Wilcox, Inc.(B&W) for domestic nuclear power plants, (2) evaluate the consistency between FTI's B&W vessel fabrication records and topical reports BAW-1500 and BAW-2121P submitted to the NRC, and (3) establish whether the weld material chemistry data in licensees' responses to NRC Generic Letter 92-01, Revision 1, and its Supplement 1, bound all available data. The NRC needed this information in order to validate the chemistry factor values in the NRC's Reactor Vessel Integrity Database (RVID) and ensure that those values, reported by NRC licensees, based on information provided to them originally by B&W, collected in reactor vessel material surveillance programs and used in the development of plant operating limits, are still conservative when all valid, traceable data are considered. The inspectors also evaluated the effect on chemistry factors of copper and nickel content data not included in the information originally published by B&W (Owners Group) in its reports BAW-1500 and BAW-2121P because B&W either was not able to establish its heat and/or weld traceability or considered the data suspect for one or more technical reasons.

With respect to the copper and nickel data, the inspectors identified some discrepancies, but ultimately determined that the RVID data were still conservative in most cases because either the licensee's copper and nickel data were conservative relative to FTI's raw data or that when included in the calculations, the formerly unconsidered FTI data had a negligible effect. The few exceptions, including, especially, one low-copper weld wire heat, have been reported to licensees so that they can assess the impact of the new information. The details of the inspectors' findings are in the enclosed inspection report.

A secondary purpose of the inspection was to conduct a routine review of FTI's program for complying with 10 CFR Part 21. The lead inspector reviewed FTI's 10 CFR Part 21 procedures against the version of the regulation that became effective in November 1995 and FTI's records of Part 21 evaluations back through 1992. The inspector identified some deficiencies in FTI's Part 21 procedures. In addition, the inspector identified one instance in which the time requirement for notification of affected licensees or purchasers pursuant to 10 CFR 21.21(b) was not met. These deficiencies constituted minor violations of the provisions of 10 CFR Part 21. In accordance with the NRC's enforcement policy as promulgated in NUREG 1600, the minor violations are discussed in the report, but no notice of violation will be issued.

In accordance with Section 2.790 of the NRC's Rules of Practice in 10 CFR Part 2, a copy of this letter and its enclosure will be placed in the NRC's Public Document Room.

We appreciate the cooperation of your organization in our conduct of this inspection. Should you have any questions regarding this inspection, please contact Mr. Stephen Alexander at 301-415-2995 or sda@nrc.gov.

Sincerely,

Suzanne C. Black

Suzanne C. Black, Chief
Quality Assurance, Vendor Inspection,
and Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket Number 99901300

Enclosure: Inspection Report 99901300/97-01

cc w/encl:
K.E. Moore, Manager, Materials & Structural Analysis Unit
M.J. DeVan, Materials Engineer

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report No:	99901300/97-01
Organization:	Framatome Technologies, Inc.
Location:	3315 Old Forest Road Post Office Box 10935 Lynchburg, Virginia 24506-0935
Contact:	K.E. Moore, Manager Materials & Structural Analysis Unit (804) 832-3277
Nuclear Industry Activity:	Engineering and material support services to owners/operators of nuclear power plants with nuclear steam supply systems or reactor pressure vessels designed and supplied by portions of the former Babcock & Wilcox Company purchased by Framatome.
Dates:	Onsite: May 19-21, 1997, continuing review through September 1997
Team Leader:	Stephen D. Alexander, Reactor Engineer, HQMB
Technical Lead:	Barry J. Elliot, Senior Materials Engineer, EMCB
Inspectors:	Meena K. Khanna, Materials Engineer, EMCB Andrea D. Lee, Materials Engineer, EMCB Kenneth J. Karwoski, Materials Engineer, EMCB
Approved by:	Richard P. Correia, Chief Reliability and Maintenance Section Quality Assurance, Vendor Inspection and Maintenance Branch Division of Reactor Controls and Human Factors

Enclosure

1 INSPECTION SUMMARY

On May 19-21, 1997, the NRC conducted an inspection at the Lynchburg, Virginia, engineering facilities of Framatome Technologies, Inc. (FTI), to review information related to the fabrication by Babcock & Wilcox Company (B&W) of reactor pressure vessels (RPVs) for domestic nuclear power plants. B&W Nuclear Technologies, who maintained this information, was purchased by the Framatome Technologies Group (FTG) and became FTG's affiliated company, FTI. The inspectors found some discrepancies and learned that there were some data that had not been reported in B&W topical reports. The inspectors identified the need for additional information from FTI that was provided in separate letters to the NRC subsequent to the onsite portion of the inspection. After completing their review of the inspection data and subsequent vendor submittals, the inspectors were able to determine that even when taking previously unreported data into account, most of the chemistry factors in the NRC's Reactor Vessel Integrity Database are still conservative. Licensees have been notified of the few cases involving low-copper weld wire heats in which inclusion of additional data could affect chemistry factors. They are expected to assess the impact of the additional data and take appropriate action in accordance with applicable regulations.

With regard to FTI's program for compliance with 10 CFR Part 21, the inspectors found that FTI's procedures adopted pursuant to 10 CFR 21.21(a) had some deficiencies that amounted to a minor violation. The inspector also reviewed FTI records relating to 10 CFR Part 21 evaluations and found one instance in which FTI did not meet the time requirements for notification of affected licensees or purchasers in accordance with 10 CFR 21.21(b). In accordance with the NRC's enforcement policy as promulgated in NUREG-1600, the procedural deficiencies and the missed time requirement (because of minor safety significance) will be treated as a minor violation of 10 CFR 21.21 and no notice of violation will be issued.

The inspection bases were as follows:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50),
- 10 CFR Part 21, "Reporting of Defects and Noncompliance"
- 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operations"
- 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"
- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements,"
- 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements,"
- Generic Letter 92-01, Revision 1, and its Supplement 1
- Regulatory Guide 1.99, Revision 2 (May 1988)

2 STATUS OF PREVIOUS INSPECTION FINDINGS

No previous findings relating to FTI or B&W were addressed during this inspection.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 10 CFR Part 21 Program

a. <u>Inspection Scope</u>

In conducting a routine review of FTI's program for compliance with 10 CFR Part 21, the inspector reviewed the latest effective revision, Revision 10, dated April 16, 1997) of Framatome Technologies Group (FTG), Inc. (FTI's parent company), Corporate Policy Statement 0401, "Reporting of Defects and Noncompliances Concerning Substantial Safety Hazards," and the latest effective revision (Revision 26, also dated April 16, 1997) of FTI Administrative Procedure (Quality Assurance) 1707-01, "Processing Safety Concerns." In addition, the inspector reviewed FTI's records of safety concern evaluations back through 1992.

b. <u>Observations and Findings</u>

The inspector found that Policy Statement 0401 indicated that it was applicable to Framatome Technologies Group (FTG), Inc., and its affiliated companies, including FTI. Section IV.A of Policy Statement 0401 stated that it was FTG's policy that FTG and its affiliated companies should "adopt appropriate procedures" to accomplish the requirements of 10 CFR 21.21(a), which it listed correctly. Section 21.21(a) of 10 CFR Part 21 sets forth the requirements for the content of procedures that are required to be adopted pursuant to the regulation. Section IV.A of 0401 also required adopting procedures to comply with 10 CFR 21.21(b) regarding informing affected licensees or purchasers of deviation or failures to comply. While 10 CFR Part 21 does not at the present time explicitly require this provision in procedures adopted pursuant to the regulation, it may in the near future, and, more importantly, it does require that Section 21.21(b) be followed when applicable; therefore, the inspector found the inclusion of this provision to be prudent. Section IV.B also prudently specified the requirements for compliance with 10 CFR 21.21(c) governing the NRC reporting process, although this section of Part 21 is also not presently required to be covered by Part 21 procedures. Likewise, Section IV.C of 0401 covered the provisions of 10 CFR 21.31 regarding procurement documents and also 10 CFR 21.6(b), alternative posting requirements, including the prescribed notice as an attachment to the policy statement.

Administrative Procedure 1707-01 contained the detailed provisions for implementing FTG Corporate Policy Statement 0401 and complying with Part 21 requirements. It assigned responsibilities for meeting the various Part 21 requirements by title and delineated the mechanics, including forms, etc., for processing what it called Preliminary Reports of Safety Concerns, or "PSCs." Paragraph VIII.A defined a safety concern in general terms, broad enough to encompass most things that might be considered deviations or failures to comply as defined in Part 21. However, the inspector identified several weaknesses in the procedure. First, the procedure did not relate safety concerns to deviations or failures to comply or explain deviations or failures to comply as defined in Part 21 (or state that included with safety concerns, specifically, are deviations and failures to comply) such that knowledgeable staff could decide if a given safety concern actually constituted a deviation or failure to comply in a "basic component" (also not explained) as defined by Part 21, that has been delivered or offered for use at an NRC-licensed facility, and thus requires evaluation in accordance with 10 CFR 21.21(a)(1). The procedure also referred to time limits in Part 21 for processing safety concerns when the term safety concern is not used in Part 21.

Second, the procedure required an evaluation of safety concerns, but did not explain that the evaluation should determine (a) if the safety concern was a deviation or failure to comply as defined in Part 21, and (b) if the deviation was a "defect" as defined in Part 21 or if the failure to comply was associated with a "substantial safety hazard," also as defined in Part 21. The inspector was concerned that without reference to or explanation of the terms used in Part 21 (and also in Policy Statement 0401), Procedure 1707-01 might allow some deviations or failures to comply to go unevaluated. For example, certain departures from technical requirements in procurement specifications (deviations) might not be recognized as a safety concern as defined in the procedure. Hence, those deviations might not be properly evaluated to determine if they constitute defects, i.e., deviations that could create substantial safety hazards or lead to exceeding a license technical specification safety limit.

Third, the procedure stated that safety concerns that were to be transferred to customers would be transferred after management concurrence with the "final report" instead of within five working days of determining that FTI would not be able to perform the 10 CFR 21.21(a)(1) evaluation as required by 10 CFR 21.21(b). Presuming "the final report" to mean the report of the group performing a 21.21(a)(1) evaluation of a PSC (what should be related to a deviation or failure to comply), the language of the procedure effectively allows information on deviations and failures to comply to remain unreported to affected licensees or purchasers until the 60-day evaluation time limit has expired, or the evaluation time as extended by an interim report when FTI may have known it was not going to perform an evaluation much earlier in the process. If the procedure had made it clear that at any time after the initiation of a PSC, one *disposition* of the PSC could be a transfer of the information to the customer, and at the point when that determination is made (documented, for example, by some kind of memo or entry in the PSC file on that date), the five-working-day time period starts, then compliance with 10 CFR 21.21(b) would be better assured.

The inspectors review of FTI's records of evaluations of safety concerns revealed that Rosemount Instruments had sent a letter to Bailey Meter Company (a B&W/McDermott subsidiary) dated August 31, 1992, informing its customer of a problem with Rosemount differential pressure transmitters (static pressure effect on span). As a result of a Bailey Meter letter to B&W Nuclear Technologies (as FTI was then known), dated October 15, 1992, informing B&W of the problem, B&W initiated PSC 4-92 on October 29, 1992 recommending that B&W's affected customers, the Tennessee Valley Authority (for Bellefont), the Washington Public Power Supply System (for WNP-1) and Consumers Power (for Midland), be notified of the problem (this would have been a 10 CFR 21.21(b) notification). Although the B&W Part 21 procedure corresponding to FTI's 1707-01 in effect at the time, BWNT-1707-01, Revision 23, dated October 1, 1992, generally reflected the revision to 10 CFR Part 21 that became effective on October 29, 1991, the procedure contained essentially the same language as FTI's 1707-01 regarding transferring safety concern information to customers. Apparently as a result, the file for PSC 4-92 contained another memorandum, dated November 19, 1992, indicating that the concern would be transferred to customers and distributing the preliminary evaluation report in house for comment. It requested that comments on the customer notification be returned in one week. Comments were received and the final evaluation report was routed for engineering and QA concurrence on December 1, 1992. The PSC was finally closed out on December 9, 1992, and the affected utilities (as well as those with non-affected, B&W-designed operating plants) were notified of the concern by Letter ESC-1047, dated December 16, 1997. The notification of affected licensees and purchasers occurred within five working days of management concurrence with the final evaluation report as prescribed by the procedure. However, this was not consistent with the requirement of 21.21(b) (or its intent) because affected licensees or purchasers were not notified of the deviation within five working days of the time (November 19, 1992) at which the file clearly indicated that the cognizant staff within BWNT had determined that the PSC would not be evaluated.

c. **Conclusions**

Procedure 1707-01 referenced Policy Statement 0401 as well as Part 21 itself and NUREG-0302 (contains some interpretation guidance). However, Procedure 1707-01 was the only working procedure adopted pursuant to 10 CFR Part 21 and was supposed to stand alone. Therefore the inspector concluded that with the weaknesses identified, the procedure might not always be effective in ensuring compliance with 10 CFR 21.21(a), and, although compliance with 10 CFR 21.21(b) is not currently required to be covered by procedures, the language of the procedure if followed without reference to the regulation could allow violation of 10 CFR 21.21(b). Accordingly, the inspector determined that these weaknesses constituted a minor violation of 10 CFR 21.21(a). In accordance with the NRC Enforcement Policy promulgated in NUREG-1600, the matter was bought to the attention of FTI and is discussed here in the report, but no Notice of Violation is being issued.

With regard to the failure of B&W in 1992 to notify affected purchasers of a deviation (that B&W did not intend to evaluate) within the time limit prescribed by 10 CFR 21.21(b), this occurrence would constitute a Level-IV violation of 10 CFR 21.21(b), except that in this particular case, the deviation was of minor safety significance. This was because (1) the affected plants of BWNT's affected purchasers were not operating, (2) the file indicated that Rosemount had also informed affected utilities, and (3) Bailey Controls had determined that the problem had a minor impact on safety. Therefore, in accordance with the criteria in NUREG-1600, it is considered a minor violation and no notice of violation will be issued.

3.2 Reactor Vessel Weld Chemistry Data

a. **Inspection Scope**

In order to obtain and validate weld chemistry data from available sources for domestic B&W-fabricated, submerged-arc welds in pressurized water reactor (PWR) and boiling water reactor (BWR) vessels, the inspectors reviewed information at FTI to:

1. Establish the traceability for weld chemistry data points by comparing them to the weld metal qualification (WMQ) test reports, chemistry laboratory work requests, or other fabrication records.

2. Review B&W's fabrication records to verify general consistency with topical reports that were submitted to the NRC.

3. Obtain all available data in order to assess the impact of including these data in best-estimate chemistry determinations.

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides a method for calculating the amount of radiation embrittlement. RG 1.99, Revision 2, indicates that the amount of radiation embrittlement is dependent upon the amount of copper, nickel, and neutron fluence. It also provides a method for determining the amount of radiation embrittlement from surveillance data. In topical reports BAW-1500 and BAW-2121P, the B&W Owners Group (B&WOG) reported the chemical composition of the Linde 80, submerged-arc welds used in fabricating the 177-Fuel Assembly OG reactor vessels. BAW-1500 and BAW-2121P identified the chemical composition of welds fabricated using the following weld wire heat numbers: 8T1762, 299L44, 72105, 406L44, 8T1554, 821T44, 61782, 71249, 72102, 72442, 72445, 1P0962, T29744, 8T3914, 1P0661, and 1P0815.

As a result of the NRC Staff review of licensee responses to Generic Letter 92-01, Revision 1, a comprehensive database was developed to compile and record summaries of the materials properties of the reactor vessel beltline materials for each plant. This database is called the Reactor Vessel Integrity Database (RVID). It should be noted that the data in the RVID came from pressurized thermal shock and pressure-temperature limit assessments, from surveillance capsule reports, and in responses to staff requests for additional information (RAIs) as well as in responses to generic letters. All of the heats identified above are included in the RVID. In addition, a few heats that were identified in the RVID, but not reported in BAW-1500 or BAW-2121P, include what are called "low-copper" weld wire heats, comprising heat numbers 442002, 442011, H4498, 31401, and 1084-18. Low-copper welds are, by definition, ones in which the weld wire had no copper coating. There may be trace amounts of copper present from other sources, but none from weld wire coatings.

As part of the process of verification of the RVID, the Materials and Chemical Engineering Branch (EMCB), Division of Engineering, Office of Nuclear Reactor Regulation, identified the need to review the chemistry data for the heats of submerged-arc weld wire material used by the B&W in fabricating RPVs used in domestic nuclear power plants. Accordingly, at FTI, the EMCB inspectors reviewed the recorded data for weld wire heats 299L44, 72105, 71249, 61782, 72445, 406L44, 72442, 821T44, 8T1554, 72102, 8T3914, T29744, 8T1762, 1P0962, 1P0661 1P0815, 442002, 442011, H4498, 31401, and 1084-18.

b. **Findings and Observations**

The chemical composition data used by FTI to determine the best-estimate chemistry for each heat came from four potential sources. These sources included: (1) nozzle beltline dropouts, (2) surveillance weld samples, (3) WMQ tests, and (4) reactor vessel beltline welds (e.g., Midland). The inspectors verified tha' FTI collected and reviewed all available material chemistry data, including results of the previous studies performed on weldments manufactured by B&W. While performing this review, the inspectors noted a weakness regarding the traceability of the recorded test data. As a result of discussions with the inspectors, FTI agreed to provide documentation that contained information to assist the inspectors in establishing/verifying traceability of each of the heats to its respective recorded test data.

The inspectors noted that the data and calculations for the majority of the heats, were recorded in an acceptable manner. However, the inspectors identified some data inconsistencies in FTI's records. The inspectors found that FTI had not transferred some of the data to its "master list," when determining the best-estimate chemistry. In addition, the inspectors noted that FTI had not identified suspect data, where applicable. During the review of "raw" data from original laboratory or weld metal qualification test reports, the inspectors considered some data points to be suspect, but these had not been identified as such by B&W or FTI. For example, data from tests on Zion Unit 2 weld WF 209-1, specimens W-40A and W-40B, had been omitted from the suspect data section in FTI's "master list." The inspectors noted that many weld wire heats had suspect data. Therefore, the inspectors requested detailed information which explained why the data were considered to be suspect and whether the data should have been included in determining the best-estimate chemistry.

Subsequent to the on-site portion of the inspection, FTI submitted the additional information in its letter INS-97-2262, dated June 6, 1997. FTI addressed suspect data, provided traceability documentation, included the additional data, and determined the effect of the additional data on the mean values of copper and nickel.

By letter INS-97-2450, dated June 19, 1997, FTI submitted the available data for the non-copper-coated weld wire heats used with Linde 80 flux in fabricating the low-copper welds. In the enclosure to that letter, FTI included the best-estimate chemistry values for low-copper welds. The inspectors had requested the information because of apparent discrepancies (i.e., values of copper too high for ostensibly low-copper welds) they found in some of the reported values for welds which had been identified as having low-copper content values. Also, in this letter, FTI committed to provide the additional data to the members of the B&WOG.

On July 10, 1997, as stated above, a conference call was held between FTI and the inspectors to discuss: traceability, correction factors, misprints and/or data not matching the "master list," and reasons for not including missing or suspect data in determining the best-estimate chemistry. These issues were resolved during the conference call and by FTI's letter, dated July 10, 1997.

Detailed Observations and Findings for Each Copper-Coated Weld Wire Heat of Interest

Table 3, included after the discussions of copper-coated weld wire heats below, compares the amount of copper and nickel in Linde 80 welds, fabricated with copper-coated weld wire, that were reported (1) by the licensees, (2) in B&WOG Reports BAW-1500 and BAW-2121P, and (3) from FTI's evaluation of all traceable data. FTI's estimates of the chemical composition were derived from the average of all "non-suspect" traceable data and from a coil-weighted average for copper. The coil-weighted average is determined by taking the sum of the products of the average amount of copper from each sample and the number of coils used in the fabrication of the sample, and dividing this sum by the number of coils to fabricate all sample welds. Note that FTI's estimates of percent copper using the coil-weighted averages were less than or equal to the best-estimate values reported by licensees, except for weld wire heat numbers 821T44, 72442, and 72445. For these heats, the coil-weighted average was 0.01 percent greater than the FTI best-estimate values derived from the average of all the data. Where the coil-weighted averages exceed the best-estimate values derived from averaging all data, the NRC will notify affected licensees and request that they assess the impact on a plant-specific basis.

b.1 Heat 8T1762

FTI's best-estimate chemistry for weld wire heat 8T1762, based on averaging the individual data points, was 0.19 weight percent for copper and 0.55 weight percent for nickel. The inspectors independently assessed the data and verified that FTI's best-estimate chemistry calculations were correct. Nevertheless, the copper and nickel values reported by licensees and contained in the RVID (0.20 and 0.55 wt-% respectively) were greater than or equal to the average from all traceable data. Therefore, there were ultimately no concerns with weld wire heat 8T1762.

b.2 Heat 299L44

The inspectors identified additional data for weld wire heat 299L44 that resulted from FTI-sponsored "round-robin" testing. The round-robin testing was conducted by several laboratories on specimens from the same welds to evaluate error sources associated with analysis methods and practices, including equipment, calibration, analyst perception and judgement where applicable, data reduction, and laboratory practices. FTI committed to include the data for these subject welds in its assessment of best-estimate chemistry. FTI submitted this information to the NRC with its June 6, 1997, letter. The inspectors verified that the additional data were added to the existing data for weld wire heat 299L44 and found the values to be acceptable. Although the inspectors noted that the standard deviation for the mean copper remained unchanged, and the standard deviation for the mean nickel for heat 299L44 increased slightly (from 0.03 to 0.04 weight percent), the inspectors confirmed that the copper and nickel values reported by licensees for 299L44 and contained in the RVID were greater than or equal to the average from all traceable data. No further concerns were identified with this heat.

b.3 Heat 72105

The inspectors identified additional data that had not been reported for weld wire heat 72105. Also, review of the supplemental information (original data) for this heat provided by FTI in its July 10, 1997, letter revealed several apparent typographical errors in the B&W topical report (BAW-1500). For example, BAW-1500, Exhibit B-6 listed the copper content for weld WF-209-1 (heat 72105) as 0.40 weight percent. The retest value for copper reported in Table C-2 of BAW-1500 was listed as 0.49 weight percent. The WMQ test report indicated that the correct value is 0.40 weight percent. Among the data not previously reported were suspect/rejected data for weld wire heat number 72105. The chemical analyses of three Zion Unit 2 samples fabricated with weld wire from this heat were rejected since it was believed that the specimen was notched in the base metal resulting in erroneous chemistry values (Zion 2 WF-209-1: 1 specimen with ID of W-40A, and 2 specimens with ID of W-40B). Several WMQ test data points were considered suspect and, therefore, the data were rejected, including: (a) the nickel value for WF-70 WMQ from Mt. Vernon WMQ Test Lab No. 6595; (b) the copper content for WF-113 from Mt. Vernon WMQ Test Lab No. 7277; (c) the nickel content for WF-209-1 from Mt. Vernon WMQ Test Lab No. 10029; and (d) the copper and nickel content for WF-353 from Mt. Vernon WMQ Test Lab No. 14433. A total of 5 copper data points and 6 nickel data points were rejected for weld wire heat number 72105.

However, the data reported for heat 72105 in FTI's letter dated July 10, 1997, accurately reflected the data identified by the licensee and the inspectors. FTI's best-estimate for the chemistry of heat 72105 was based on averaging the individual data points. The inspectors verified that the calculations were correct. This resulted in 0.32 weight percent for copper and 0.57 weight percent for nickel.

The inspectors also assessed the best-estimate chemistry, for heat 72105, by averaging the mean value from the individual sources of data. The 12 sources of data, if the data points discussed above are excluded, include: (1) WMQ test for weld WF-70, (2) WMQ test for weld WF-113, (3) WMQ test for weld WF-209, (4) WMQ test for weld WF-209-1, (5) nozzle belt dropout for Midland 1, (6) beltline weld for Midland 1, (7) reactor vessel surveillance program (RVSP) from Oconee 2, (8) RVSP from Oconee 3, (9) RVSP from Crystal River 3, (10) RVSP from Midland 1, (11) RVSP from Zion 1, and (12) RVSP from Zion 2. The resultant values from this "mean of the means" approach are listed in Table 1. The inspectors also assessed the impact of including the data points excluded from the database (discussed above) on the "mean of the means" chemistry. For this case, another source of data was added (i.e., the WMQ test for weld WF-353). These values are also listed in Table 1. The results indicate that the copper and nickel values reported in RVID for heat 72105 are conservative. Therefore, the inspectors had no further concerns with weld wire heat 72105.

TABLE 1: MEAN of MEANS for HEAT 72105

	Suspect Data Excluded		Suspect Data Included	
	Copper	Nickel	Copper	Nickel
Mean	0.323	0.578	0.301	0.561
Standard Deviation (sample)	0.049	0.018	0.083	0.043

b.4 Heat 821T44

FTI provided a comprehensive listing of chemistry data for weld wire heat 821T44. FTI's estimates of the percent copper, using the coil-weighted averages, were less than or equal to the best-estimate values reported by the licensees, except for weld wire heat numbers 821T44, 72442, and 72445. For these weld wire heats, the coil weighted average was 0.01 weight percent greater than the best-estimate values derived from the average of all the data. However, the inspector's review of the data for weld wire heat 821T44 indicated that the copper and nickel values reported by licensees and contained in the RVID are greater than or equal to the average from all traceable data. Therefore, the slightly higher coil-weighted average for copper notwithstanding, there were ultimately no concerns with weld wire heat 821T44.

b.5 Heat 72442

FTI provi a comprehensive listing of chemistry data for weld wire heat 72442. Although for 72442, t ercent copper by coil-weighted average was 0.01 weight percent greater than the best estin value from averaging all the data, the inspector's review of the data for weld wire heat 72442 indicated that the copper and nickel values reported by licensees and contained in the RVID for 72442 are equal to the average from all traceable data. Therefore, there were no concerns with weld wire heat 72442.

b.6 Heat 72445

FTI provided a comprehensive listing of chemistry data for weld wire heat 72445. FTI's best-estimate copper content values, derived from the average of all data, as well as by coil-weighted average as stated above, was found to be 0.01 weight percent greater than the best-estimate values reported by the licensees, for weld wire heat number 72445. However, for this heat, the average amount of nickel was less than that reported by licensees. The lower value for nickel in this case offset the higher copper values such that the chemistry factor remained the same. Theretore, there were no concerns with heat 72445.

b.7 Heat 71249

The inspectors identified additional data for weld wire heat 71249 resulting from the round-robin testing. FTI committed to include the data in its assessment of best-estimate chemistry. FTI submitted the information to the NRC with its July 10, 1997, letter. Also included with this letter were data points from several sources cited in a letter from Florida Power & Light Company (FP&L), dated February 10, 1984. However, these data were not included in the database for this weld because they could not be verified or traced. These data included: (a) 5 data points from Westinghouse, (b) at least 2 data points from the Heavy Section Steel Technology (HSST) program, (c) 1 data point from Point Beach 1 (WCAP 8743), (d) 1 data point from Zion 2 (SECY 82-465), (e) 1 Oconee data point (SECY 82-465), and (f) 1 data point from Ginna. Of the 5 Westinghouse data points, some may be duplicates of the values listed in the June 30, 1977, and April 11, 1977, letters from FP&L included and referenced in the June 6, 1997, FTI letter. The inspectors noted that the data points from Point Beach 1 and Ginna may have been duplicates of the WMQ test report data from weld SA-1101; although this could not be verified with the available data. However, the possible duplication in this case is inconsequential because the data were not used.

The four copper and nickel values for heat 71249 reported from the National Bureau of Standards (NBS) analysis of specimen 62W (Oconee 1, SA-1101) are the mean of four analyses (i.e., there are four groups of data, each with 4 analyses, for a total of 16 measurements). However, the chemistry values from these four groups may not have been corrected for calibration procedures and for spectral interferences. Nevertheless, the overall average of copper and nickel of the 16 measurements took these "correction" factors into consideration as indicated in a July 24, 1985, letter from John A. Norris of NBS to Evan Morgan of B&W. These corrections to the "raw measurements" appeared to be insignificant, as evidenced by a comparison of the average value of copper from the "raw measurements" (0.172 weight percent) to the "corrected measurements" (0.170 weight percent).

A total of 17 copper and nickel data points were not included in the best-estimate chemistry for heat 71249 because the WMQ test reports were not available These included 7 specimens from analysis of HSST- 62W [62W-309, 62W-359, 62W-202, 62W-223 (2 specimens), 62W-276 (2 specimens)] and 10 specimens from analysis of HSST-61W [61W-232, 61W-276, 61W-246, 61W-225, 61W- 222, 61W-270 (4 specimens), and 61W-234].

The inspectors noted that not all of the available data were listed in the FTI letter dated July 10, 1997, although the letter contained all of the data where traceability could be established. FTI's best-estimate for the chemistry of heat 71249 was based on averaging the individual data points. The inspectors verified that the calculations for these values were correct (0.24 weight percent for copper and 0.61 weight percent for nickel). In addition, the inspectors assessed the impact on the best-estimate chemistry of including the previously rejected data points. The result was the same mean value of copper, a lower mean value for nickel, and a higher sample standard deviation for both.

The inspectors also assessed FTI's best-estimate chemistry for heat 71249 by averaging the mean values from the individual sources of data. The 10 sources of data, when the data points discussed above are excluded, included: (1) WMQ test for weld SA-1094, (2) WMQ test for weld SA-1101, (3) WMQ test for weld SA-1229, (4) WMQ test for weld SA-1344, (5) WMQ test for weld SA-1706, (6) WMQ test for weld SA-1769, (7) RVSP from Westinghouse, (8) RVSP from Turkey Point 3, (9) RVSP from Turkey Point 4, and (10) nozzle belt dropout (NBD) from Oconee 1. The resultant values from this "mean of the means" approach are listed in Table 2. The inspectors also assessed the impact of including the data points excluded from the database (discussed above) on the "mean of the means" chemistry. For this case, two sources of data were added (i.e., RVSP from the HSST program and a NBD from the HSST program). These values are also listed in Table 2.

TABLE 2: MEAN of MEANS for HEAT 71249

	Suspect Data Excluded		Suspect Data Included	
	Copper	Nickel	Copper	Nickel
Mean	0.234	0.591	0.234	0.592
Standard Deviation (sample)	0.049	0.036	0.048	0.036

FTI's best-estimate nickel content values were less than or equal to the values reported by the licensees, except for weld wire heat number 71249. However, for this heat, the average amount of copper was less than that reported by affected licensees, which resulted in a lower chemistry factor. In addition, the mean copper content for heat 71249, determined from the WMQ test data and the nozzle belt dropout data, were significantly lower than the mean determined from the RVSP. This indicated that there was considerable variability in the copper content within this heat of material. However, the results indicated that the copper and nickel values reported in the RVID for heat 71249 are conservative.

The inspectors confirmed that FTI added the data from Florida Power & Light (FP&L), for weld wire heat 71249 to the data tables, as appropriate.

b.8 Heat 72102

For weld wire heat 72102, the WMQ test report for weld SA-1187 indicated that the WMQ test failed. The reason the test failed was not detailed. Although the copper and nickel values reported for this WMQ test were used to estimate the mean value for the total population of welds using weld wire heat 72102, the values were low to medium range and had a negligible effect on the mean. For heat 72102, FTI's best-estimate chemistry, based on averaging the individual data points, was found to be 0.21 weight percent for copper and 0.59 weight percent for nickel. Nevertheless, the inspectors confirmed that the copper and nickel values for weld wire heat 72102 reported by licensees and contained in the RVID are greater than or equal to the average from all traceable data. Therefore, there were no further concerns with this heat.

b.9 Heat T29744

The inspectors identified additional data for weld wire heat T29744 during the inspection. However, the additional information on T29744 in FTI's letters of June 6 and July 10, 1997, indicated that the best-estimate mean copper and nickel values did not change significantly with the additional data. Also, FTI added weld identification 63W to the existing data for heat 299L44, but the inspectors determined that the best-estimate mean copper and nickel values remained unchanged with the addition of the data from weld identification 63W. The inspectors confirmed that the copper and nickel values reported by licensees and contained in the RVID for T29744 were greater than or equal to the average from all traceable data. Therefore, there were no concerns with this weld wire heat.

b.10 Heat 1P0661

The inspectors identified additional data for weld wire heat 1P0661 during the inspection. The inspectors found that the best-estimate chemistry was not conservative when comparing it to the previous data for these weld wire heats. The inspectors reviewed the new information in FTI's letters dated June 6 and July 10, 1997, and concluded that the best-estimate mean copper and nickel values did not change significantly with the additional data for weld wire heat 1P0661. The inspector's review of the data for weld wire heat 1P0661 indicated that the copper and nickel values reported by licensees and contained in the RVID are greater than or equal to the average from all traceable data. Therefore, there were no further concerns with this heat.

b.11 Heats 406L44, 8T1554, 61782, 1P0962, 8T3914 1P0815

FTI provided a comprehensive listing of chemistry data for these weld wire heats. The inspector's review confirmed that the copper and nickel values reported by licensees and contained in the RVID for these heats were greater than or equal to the average from all traceable data. Therefore, the inspectors had no concerns with these weld wire heats.

TABLE 3: CHEMICAL COMPOSITION DATA for LINDE 80 WELDS FABRICATED With COPPER-COATED WELD WIRES

Heat Number	Chemical Composition Reported by Licensees and Compiled in RVID		Chemical Composition from BAW-1500 and BAW-2121P		FTI Best-Estimate of Chemical Composition from Average of All Data		FTI Estimate of Chemical Composition from Coil-Weighted Average
	%Cu	%Ni	%Cu	%Ni	%Cu	%Ni	%Cu
299L44	0.35	0.68	0.35	0.68	0.34	0.68	0.34
72105	0.35	0.59	0.35	0.59	0.32	0.57	0.33
406L44	0.31	0.59	0.31	0.59	0.29	0.58	0.26
821T44	0.24	0.63	0.24	0.63	0 24	0.63	0.25
61782	0.25	0.54	0.25	0.54	0.24	0.54	0.24
71249	0.26	0.60	0.26	0.61	0.24	0.61	0.22
72442	0.24	0.60	0.24	0.60	0.24	0.60	0.25
72445	0.21	0.59	0.21	0.59	0.22	0.58	0.22
8T1554	0.18	0.63	0.18	0.63	0.16	0.61	0.16
72102	0.25	0.63	0.23	0.63	0.21	0.59	0.21
T29744	0.29	0.68	0.29	0.68	0.27	0.68	0.27
8T3914	0.18	0.64	0.18	0.64	0.18	0.64	0.18
8T1762	0.20	0.55	0.20	0.55	0.19	0.55	0.19
1P0815	0.17	0.52	0.17	0.52	0.17	0.52	0.17
1P0962	0.21	0.64	0.21	0.64	0.21	0.64	0.21
1P0661	*	*	0.19	0.63	0.17	0.64	0.17

* No data were reported to the NRC for this heat.

Detailed Observations and Findings for Non-Copper-Coated Weld Wire Heats of Interest

Table 4 compares the amount of copper and nickel in Linde 80 welds, fabricated with non-copper-coated weld wire, that were reported by the licensees and documented by FTI's evaluation of all traceable data. The chemical composition database for non-copper coated weld wires has been distributed by FTI to the licensees with RPV welds fabricated using these heats of weld wire. Licensees are expected to address the effect of the data on their pressure-versus-temperature operating limit curves and pressurized thermal shock (PTS) assessments in accordance with applicable regulations.

TABLE 4: CHEMICAL COMPOSITION DATA for LINDE 80 WELDS FABRICATED with NON-COPPER-COATED WELD WIRES

Heat Number	Chemical Composition Reported by Licensees and Compiled in RVID %Cu	%Ni	FTI Best Estimate of Chemical Composition from Average of All Data %Cu	%Ni	FTI Estimate of Chemical Composition from Coil-Weighted Average %Cu
442002	0.030 0.050	0.460 0.620	0.029	0.680	0.059
442011	0.030 0.030	0.630 0.650	0.033	0.688	0.032
H4498	0.030	0.500	0.042	0.460	0.042
31401	0.180 0.190 0.230	0.540 0.560 0.570	0.193	0.576	0.193
1084-18	0.040	0.600	0.038	0.600	0.038

b.12 Heat 442002

The inspectors noted that FTI's best-estimate copper value for low-copper weld wire heat 442002 (0.029 wt-%) was less than that reported by the licensee (0.030 wt-%), but the best-estimate nickel value (0.680 wt-%) was significantly greater than the 0.460 wt-% reported by the licensee. The reported copper values for heat 442002 were not considered suspect; however, best-estimate chemistry values were obtained for all of the low-copper welds for completeness. To assess this information, the inspectors computed the resultant chemistry factors using the RVID program. The chemistry factor associated with the licensee-reported values was 41.0. However, the chemistry factor associated with the best-estimate values was less than that reported by the licensee, i.e., 39.6 because copper is much more heavily weighted in the calculation. Therefore, there were no further concerns with this heat.

b.13 Heat H4498

The inspectors identified some discrepancies in the reported copper values for low copper weld wire heat H4498; and requested additional information. FTI provided the NRC with a summary of the best-estimate chemistry for the low-copper welds with its June 19, 1997, letter. The copper value for H4498 was slightly higher than previously reported values, but the welds still reportedly contained very low copper. The reported copper values for heat H4498 were not identified as suspect. The inspectors determined the chemistry factor for H4498 associated with the data reported by licensees using the RVID program (41.0) and compared it to the chemistry factor associated with FTI's best-estimate copper and nickel values derived from the average of all data (56.7). However, FTI stated that the affected weld for this heat was in a lower-fluence location in the Braidwood vessel, and was not the controlling (limiting) material/weld for that vessel. Therefore, the apparently significant change in the chemistry factor would have no impact on the operating limits for the Braidwood vessel. There were no further concerns with this heat.

b.14 Heat 442011

The inspectors noted that FTI's best-estimate copper value for heat 442011 (0.033 wt-%) was more than those reported by the licensee (0.030 wt-%), and the best-estimate nickel value at 0.688 wt-% was also greater than the 0.630 and 0.650 wt-% values reported by the licensee. The data for heat 442011 were not considered suspect. To assess this information, the inspectors computed the resultant chemistry factors using the RVID program for comparison. The chemistry factor computed for the licensee-reported values was 41.0 and the chemistry factor computed for the best-estimate values was 44.9. Although the copper values are very low compared to those from copper-coated weld wire, they would be expected to have a calculable effect on operating limits because, according to FTI, heat 442011 was used in the center circular (beltline) weld, the controlling material in the reactor vessels in Braidwood Station. FTI confirmed that the affected licensee, Commonwealth Edison Company, was promptly informed of the information and had reported it was assessing the impact on operating limits as required by regulations. The licensee will be expected to make the appropriate submittals to the NRC associated with any resultant changes.

b.15 Heat 1084-18

The inspectors identified some discrepancies in the reported copper values for the low copper weld wire heat 1084-18 and requested additional information. In a subsequent submittal, FTI provided a summary of the best-estimate chemistry for the low-copper welds. The inspectors noted that only one data point was identified by FTI for low-copper weld wire heat 1084-18. The best-estimate mean values for copper and nickel for this weld wire heat were recorded as 0.038 and 0.600 wt-%, respectively. However, the copper and nickel values reported by licensees and contained in the RVID were greater than or equal to the average from all traceable data. Therefore, there were no concerns with this heat.

b.16 Heat 31401

The inspectors noted that the copper values reported by licensees and listed in the RVID for low-copper weld wire heat 31401 were unusually high for an ostensibly low-copper heat; although still not as high as those typical of copper-coated weld wire. The nickel values were not unusual. The inspectors requested additional information which was provided in a subsequent FTI submittal. FTI's best estimate values for both copper and nickel were consistent with the values in the RVID, falling in the middle of the range. Therefore, there were no further concerns with this heat.

c. **Conclusions**

The inspectors concluded that the data compiled by the B&WOG in topical reports BAW-1500 and BAW-2121P were not complete and not fully traceable. However, the inspectors were provided with sufficient information during and following the inspection to conclude that (1) adequate traceability was eventually established for most of the weld chemistry data, (2) consistency was verified between B&W's fabrication records and topical reports, and (3) most of the information in the RVID (at least chemistry factors) remained conservative even when all available data were included in the best-estimate chemistry determination. The exceptions have been appropriately dispositioned as discussed in the individual weld wire heat sections above.

It should also be noted that FTI informed the B&WOG Reactor Vessel Working Group, Rochester Gas and Electric, and Tennessee Valley Authority, by letter INS-97-2526, dated June 30, 1997, of the results of their efforts prompted by the NRC inspection. This letter included summary tables of the chemical composition assessments for the Linde 80 weld metals, as reported to the NRC. The chemical composition data were presented using three different assessments: (1) currently reported best-estimate values, (2) best-estimate values based on total population of all traceable data, and (3) coil-weighted average for weld copper content. FTI's review of the data assessments indicated that the weld compositions determined prior to the inspection were conservative for most of the weld wire heats. For the copper-coated wire heats, where the estimates were non-conservative, FTI determined the effect to be inconsequential (i.e., greater by only 0.01 weight percent). The inspector's review of these data confirmed that the copper and nickel values for copper-coated weld wire heats reported by licensees and contained in the RVID are greater than or equal to the average from all traceable data, except for weld wire heat numbers 72445 and 71249. However, since the chemistry factor values (used by the licensees in developing operating limits), based on the average of all data for these weld wire heats are less than or equal to the chemistry factor values in the RVID, the chemistry data for copper-coated weld wire heats in the RVID are acceptable.

For low-copper heats, FTI's best-estimate values for copper or nickel or their associated chemistry factors were either conservative with respect to previous reported values or were not significant because of not being in the controlling welds with one exception. For heat 442011, although it contained very low copper, the copper value and resultant chemistry factor was greater than that previously reported and was in the controlling material. FTI confirmed that the affected licensee, Commonwealth Edison Company, was promptly informed of the information and had reported that it was assessing the impact on operating limits as required by regulations. The licensee will be expected to make the appropriate submittals to the NRC associated with any resultant changes.

The staff assessment evaluated chemistry data. It did not evaluate the impact of surveillance data on embrittlement. Licensees that use surveillance data to determine the amount of embrittlement will need to assess the FTI chemistry data. The NRC will notify the affected licensees.

4 ENTRANCE and EXIT MEETINGS

At the entrance meeting on May 19, 1997, the NRC inspectors discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with cognizant FTI management and staff. At the exit meeting on May 21, 1997, the inspectors discussed their findings and observations and obtained a commitment from FTI to submit the remainder of the chemistry data and other information requested by the inspectors that FTI was unable to provide during the onsite portion of the inspection.

5 PARTIAL LIST OF PERSONNEL CONTACTED

FTI

K.E Moore, Manager, Materials and Structural Analysis Unit
L.B. Gross, Advisory Engineer
M.J. DeVan, Metallurgical Engineer
R.J. Schomaker, Project Manager
Richard Rawlings, Quality Assurance Audits Manager
Stephen Fyfitch, Senior Supervisor, Materials Group
Jim Taylor, Licensing Manager

NRC

Jack Strosnider, Chief, Materials and Chemical Engineering Branch (EMCB)
Keith Wichman, Chief, Materials Integrity Section, EMCB

ITEMS OPENED, CLOSED, AND DISCUSSED

None

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

February 5, 1998

Mr. Timothy Rotti, Division Manager
National Technical Systems
Nuclear Services Group
533 Main Street
Acton, Massachusetts 01720

SUBJECT: NRC INSPECTION REPORT 99900912/98-01

Dear Mr. Rotti:

On January 6-8, 1998, the U.S. Nuclear Regulatory Commission (NRC) performed an inspection at the National Technical Systems (NTS) facility in Acton, Massachusetts. The enclosed report presents the findings of that inspection. The inspection was conducted to assess specific attributes and implementation of the NTS quality assurance (QA) program to ascertain whether it met NRC requirements. The inspectors specifically reviewed your activities relating to the procurement, dedication, and supply of replacement 4-kV circuit breakers to the Diablo Canyon Power Plant. The inspectors assessed NTS's conformance to customer procurement requirements, commercial grade dedication activities and compliance with NRC regulations. In addition, the inspectors examined corrective action taken in response to the findings of the previous inspection of NTS documented in Inspection Report 99900912/93-01. Within the scope of this inspection, we found no instance in which NTS failed to meet NRC requirements.

As discussed in detail in the enclosed report, the inspectors found that NTS's commercial-grade dedication of the Yaskawa modular assemblies, the conversion hardware, and final conversion units was acceptable. The inspectors also found that the evaluations of Yaskawa design changes and PDS conversion hardware by NTS and PG&E were adequate to support taking credit for the Yaskawa ANSI interrupting capacity test.

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,

Richard P. Correia for

Suzanne C. Black, Chief
Quality Assurance, Vendor Inspection,
and Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket No. 99900912
Enclosure: Inspection Report 99900912/98-01

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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report No:	99900912/98-01
Organization:	National Technical Systems, Inc. Acton, Massachusetts
Contact:	Christine Briggs , Vice President Director of Quality 508/263-2933
Nuclear Activity:	Provides commercial grade dedication and equipment qualification services and dedicated equipment for the commercial nuclear power industry.
Dates:	January 6-8, 1998
Inspectors:	Stephen D. Alexander, Reactor Engineer Gregory C. Cwalina, Senior Operations Engineer Bill H. Rogers, Reactor Engineer
Approved by:	Robert A. Gramm, Chief Quality Assurance and Safety Assessment Section Quality Assurance, Vendor Inspection and Maintenance Branch Division of Reactor Controls and Human Factors

Enclosure

1 INSPECTION SUMMARY

National Technical Systems (NTS), Inc., provides engineering and test services to industry and government. With a quality assurance (QA) program intended to meet the requirements of 10 CFR Part 50, Appendix B, NTS's Acton, Massachusetts, division has been a major supplier of basic components (both hardware and services) to the commercial nuclear power industry. In recent years NTS has been primarily involved in the procurement and dedication of commercial grade items for safety-related applications, including environmental and/or seismic qualification testing and analysis as required.

On June 9-12, 1997, the NRC conducted an inspection at Pacific Gas & Electric Company's (PG&E's) Diablo Canyon Power Plant (DCPP) to review activities related to the procurement, modification, installation and testing of 4-kV circuit breakers of 350-megavolt-ampere (MVA) interrupting capacity to replace most of the plant's original GE Magne-Blast breakers of 250-MVA capacity (NRC Inspection Report No. 50-275,323/97201). NTS, Acton, was PG&E's prime contractor for the project. The replacement breakers are specially converted Type OGR SF_6 Gas Floupac Series, Rotary Arc, 4.16-kV circuit breakers; rated at 1200 and 2000 A, continuous, with a 41-kA current interrupting rating, manufactured by the Yaskawa Electric Company, LTD. (Yaskawa) of Tokyo, Japan. The inspectors found that Yaskawa had made some modifications to the breaker's operating mechanism to facilitate the conversion, and that these modifications had been made subsequent to design verification testing of the original breaker done in accordance with applicable U.S. industry standards (specifically, American National Standards Institute (ANSI) standards) to which PG&E had committed for DCPP. The inspectors also found that NTS had collaborated with a commercial switchgear services vendor, Power Distribution Services, Inc. (PDS), to accomplish the conversions, with NTS providing engineering support and nuclear QA coverage. NTS had the design and production testing performed on the complete conversion units prescribed by the applicable breaker conversion standard, to test the hardware added to the Yaskawa breaker, but took credit for some of the original design tests (in particular, the interrupting capacity tests) performed by Yaskawa. This was allowed by the conversion standard, provided engineering analyses could establish that the modifications by Yaskawa and PDS would not invalidate any design tests for which credit was being taken.

Yaskawa certified that none of the breaker modifications would impact the results of the testing. NTS reviewed and evaluated the Yaskawa modifications and also concluded that the changes would not invalidate the results of the breaker tests. NTS also evaluated the PDS conversion modifications and concluded that either they also did not invalidate the design tests or were covered by the additional design and production testing prescribed by the conversion standard. In addition, PG&E had performed its own evaluations with the same conclusion. However, based upon the information available at Diablo Canyon, the inspectors could not independently reach the same conclusion during the June 1997 NRC inspection. Therefore, the inspectors identified the need to review the NTS evaluations and other pertinent information during an inspection at NTS's Acton, Massachusetts, facility.

During this inspection of NTS, the inspectors assessed specific attributes and implementation of the NTS 10 CFR Part 50, Appendix B, QA program, specifically as it applied to the procurement modification/conversion and dedication of the replacement 4-kV breakers for DCPP.

The inspection bases were as follows:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50),

- 10 CFR Part 21, "Reporting of Defects and Noncompliance."

During this inspection, the inspectors found no instance in which NTS failed to meet NRC requirements. The inspectors found that NTS's commercial-grade dedication of the Yaskawa modular assemblies, the conversion hardware and final conversion units was acceptable. The inspectors also found that the evaluations by NTS and PG&E of Yaskawa design changes and PDS conversion hardware were adequate to support taking credit for the Yaskawa ANSI interrupting capacity test.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

The NRC last performed an inspection of NTS in July 1993 (NRC Inspection Report No. 99900912/93-01). During that inspection, the NRC identified one violation, one nonconformance, one unresolved item and three open items. Resolution of those items is discussed below.

2.1 Violation 99900912/93-01-01 (Closed)

Contrary to the requirements of 10 CFR 21.21(a), NTS procedures, adopted pursuant to the regulation, would not, as written, ensure that deviations or failures to comply would be properly evaluated. The procedure also lacked certain provisions required by the July 1991 revision of the regulation. In addition, the posted notice prescribed by the procedures lacked certain information required by 10 CFR 21.6(b).

NTS provided a response to Violation 99900912/93-01-01 in a letter to the NRC dated December 23, 1993, which stated that Quality Assurance Procedure (QAP) 1, "Reporting Requirements Per 10CFR21," had been revised and that the NTS 10 CFR Part 21 posting had been corrected to meet the requirements of 10 CFR Part 21. The NRC inspectors reviewed Revision 4 of QAP 1, dated February 16, 1996, and determined that it required employees to notify their supervisor of conditions which did not meet technical procurement specifications (deviations) and required NTS to evaluate identified deviations or notify the customer within the applicable periods of time as specified in 10 CFR Part 21. The inspectors observed the current NTS 10 CFR Part 21 posting and determined that the posting met the requirements of 10 CFR 21.6. The inspectors concluded that NTS's corrective actions had been adequate and that the requirements of 10 CFR Part 21 regarding procedures and posting were being met.

2.2 Nonconformance 99900912/93-01-02 (Closed)

Contrary to the requirements of Criteria III, V and VII of Appendix B to 10 CFR Part 50, NTS-prepared procedures for dedication testing of Klöckner-Moeller (K-M) molded-case circuit breakers for the North Anna Power Station did not properly incorporate design requirements because they did not specify a minimum duration for the full-load hold-in test, and there was evidence that the test was conducted for an inappropriate time.

NTS provided a response to Nonconformance 99900912/93-01-02 in its December 1993 letter to the NRC which stated that NTS had performed a prompt evaluation of the safety significance of the issue and had concluded that there were no immediate safety concerns since the licensees' actual plant loads were less than 40% of the main breaker rated current. However, to provide for the potential occurrence that plant loads could be changed, NTS reported that it had conducted a commercial grade survey of K-M and verified that K-M had performed adequate testing to verify the full-load capacity of the dedicated breakers. The inspectors concluded that NTS's corrective actions were adequate.

2.3 Unresolved Item 99900912/93-01-03 (Closed)

Out-of-tolerance tripping of certain K-M overload relays during NTS testing was attributed to age. The inspectors were not able to determine during the July 1993 inspection what the basis was for the K-M revised trip time tolerances, nor what other installations might be affected by relays with similar age/shelf-life-shifted performance characteristics.

NTS provided a response to Unresolved Item 99900912/93-01-03 in its December 1993 letter which stated that NTS had contacted K-M, Commonwealth Edison Company (ComEd) and its architect-engineer firm, Sargent & Lundy (S&L), to assist in resolving the issue. The S&L response letter to NTS, dated June 22, 1994, stated the trip-time curves used by NTS during the testing had been obtained from incorrect revisions of the drawings and were not applicable to the relays which had been tested. S&L stated that when it had compared the relay test data to the correct trip-time curves, it determined that most relays had met the required specification. S&L identified several relays which had not met specification and recommended that they not be used. It also identified several relays which were to be returned to S&L for adjustment. S&L determined the relay test results which had been previously identified by NTS as early tripping should be considered acceptable (other than those test results which S&L had identified as not meeting specification) and that there was no indication of any age-related failure mechanism. The ComEd letter to NTS, dated June 27, 1994, concurred with the S&L conclusions. The inspectors concluded that NTS had established an acceptable basis for the acceptance of the previously identified trip-time characteristics of the K-M relays and had taken adequate corrective action to resolve the apparent anomaly.

2.4 Open Item 99900912/93-01-04 (Closed)

NTS had properly reported performance anomalies of Z4-100/K-NA overload relays which were identified by NTS during environmental qualification testing to K-M and to NTS's customers. However, there was insufficient information to determine how or if NTS, K-M, or affected customers had evaluated any effects these anomalies might have on existing installations in harsh-environment, Class 1E applications.

NTS provided a response to Open Item 99900912/93-01-4 in its December 1993 letter to the NRC which stated that NTS had contacted ComEd and S&L to assist in resolving the issue. The S&L letter to NTS, dated May 5, 1994, stated that S&L had evaluated the testing occurrence during which a Z4-100/K-NA overload relay tripped at 135°F while ramping down from the peak temperature of 170°F. The unexpected tripping did not occur when the test was repeated for the relay which had previously tripped nor did it occur during the testing of the other relays. S&L reported it had performed an evaluation and analysis and concluded that the unexpected tripping had been a random occurrence that did not affect the use of the relay type in the intended application. The ComEd letter to NTS, dated May 16, 1994, concurred with the S&L conclusions. The inspectors concluded that NTS had established an acceptable basis for the acceptance of the qualification of the K-M relays and had taken adequate corrective action to resolve the identified anomaly.

2.5 Open Item 99900912/93-01-05 (Closed)

NTS had not evaluated the impact of environmental qualification test failures of Continental silicone rubber-insulated electrical cable on installations, if any, of this cable in Class 1E, harsh-environment applications, but it had reported the failures to its only customer for the tests, Spectrum Technologies. There was insufficient information for the inspectors to determine whether Spectrum or Continental had performed such an evaluation either.

Spectrum had provided an evaluation of the test failures in a letter to the NRC dated March 28, 1994. The letter stated that Spectrum had contracted NTS to perform the qualification tests in order to support a Northern States Power Company (NSPC) order to Spectrum for qualification and dedication of certain Continental electrical cable. Spectrum had purchased commercial grade cable from Continental and provided samples to NTS for aging and qualification testing. When the cable failed the qualification tests, Spectrum discontinued further qualification or dedication of this particular cable type, returned the cable to Continental and advised NSPC of the results. NSPC revised its purchase order to substitute Rockbestos cable (previously qualified by Rockbestos) which Spectrum purchased from Rockbestos on the basis of a successful 10 CFR Part 50, Appendix B, audit. In addition, the Spectrum letter emphasized that the Continental cable was never considered qualified nor dedicated and was not delivered to the customer. The inspectors concluded that NTS had taken adequate action in notifying Spectrum of the identified anomaly and that Spectrum had correctly concluded that the cable had failed qualification and had properly notified the potential end user.

2.6 Open Item 99900912/93-01-06 (Closed)

During the July 1993 NRC inspection at NTS, the inspectors identified that, during NTS qualification testing of certain Static-O-Ring (SOR) pressure switches, one sample had developed a pressure leak during a high-energy-line break (HELB) test, and another sample had suffered excessive leakage current during a dielectric withstand test. The inspectors found no documented evaluation by NTS or SOR of the root cause of the test failure or evaluation of the impact, if any, on existing safety-related applications.

Following the NTS inspection, SOR provided the NRC its complete qualification report for the switches which had not been available at NTS. In the report, and with clarification provided during subsequent telephone conversations with the NRC, SOR reported that it had found that the leakage path was provided by unsealed mounting bracket screws for the micro switch (switching element) mounted in the switch housing. SOR had evidence that the screws had not been resealed after the micro switch was readjusted during factory calibration. Failure to reseal the screws allowed the switch diaphragm (seal) to be over pressurized during the test which caused it to leak. SOR stated that since other switches with the same type of housings did not suffer a similar test failure, the test failure was attributable to a random occurrence, not to an inherent weakness in the design. SOR's corrective action consisted of (1) resealing the micro switch mounting screw threads if the micro switch was readjusted during factory calibration, and (2) applying a primer to the micro switch bracket screws to improve the curing of the thread sealant in the stainless steel housings of the potentially affected switch models. SOR confirmed that applicable safety-related pressure switches installed in plants were not compromised because the corrective measures had been instituted before production.

Regarding excessive leakage current in one sample: the inspector observed that the SOR pressure switch test specimens passed the dielectric withstand test at 1500 Vac for 1 minute, except for one sample that experienced about 2 milliamps (mA) of leakage current at 900 Vac. SOR could not find a root cause and believed that the 2-mA leakage current was a random anomaly and not indicative of a common failure mode. SOR's basis was that the other specimens (1) passed the test at 1500 Vac, (2) had adequate insulation resistance at 500 Vdc, and (3) had sufficient margin for service conditions because the switches were rated for 250 Vac and typically used in 120-Vac or 125-Vdc applications. The inspector determined that on the basis of information provided by SOR, and because leakage current from moisture intrusion would typically have been higher than 2 mA, the anomaly was satisfactorily addressed.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 10 CFR Part 21 Program

a. Inspection Scope

The inspectors reviewed the effective NTS 10 CFR Part 21 implementing procedure required by 10 CFR 21.21(a), QAP 1, "Reporting Requirements Per 10CFR21," Revision 4, dated February 16, 1996, and the posting required by 10 CFR 21.6. The purpose of the limited Part 21 review was to verify that NTS had implemented the corrective actions reported and proposed in its response to the previous Notice of Violation (99900912/93-01-01), had updated its Part 21 procedure to reflect the revision to the regulation that became effective in November 1995 (to the extent applicable to NTS scope of activities) and to confirm continued compliance with the requirements of 10 CFR 21.21(a) and 10 CFR 21.6.

b. Observations and Findings

The inspectors found that QAP-1 now appropriately required employees to notify their supervisors of conditions which do not meet technical procurement specifications (deviations) and required NTS to evaluate identified deviations or notify the customer of the deviations within the applicable periods of time as specified in 10 CFR Part 21. The inspectors observed that the current NTS posting required by 10 CFR 21.6. consisted of 10 CFR Part 21, Section 206 of the Energy Reorganization Act of 1974, and a notice stating the location of the NTS 10 CFR Part 21 implementing procedure and the name of the person(s) to whom the report should be made.

The inspectors determined that NTS had not performed any 10 CFR Part 21 evaluations since the 1993 NRC inspection.

c. Conclusions

Within the limited scope of this review, the inspectors concluded that NTS had implemented appropriately revised procedures required by 10 CFR 21.21(a). Further, the posting required by 10 CFR 21.6 was constructively in compliance with the regulation.

3.2 Conversion and Design Verification of 4-kV Circuit Breakers

a. Inspection Scope

The inspectors assessed attributes and implementation of the NTS 10 CFR Part 50, Appendix B, QA program as it applied to the procurement modification/conversion and dedication of the replacement 4-kV breakers for DCPP. Specifically, the inspectors examined 1) NTS's evaluation of design changes made by Yaskawa subsequent to ANSI type testing, 2) design documents for the complete conversion, and 3) NTS/PDS design and design verification activities.

The review included evaluating NTS activities with regard to oversight of manufacturer and PDS design, design verification, commercial quality controls and production testing. In addition the inspectors reviewed NTS activities relating to the manufacturer's modifications to the original breaker, fabrication of adapting hardware by PDS, prototype testing and production of the final breaker assemblies.

b. Observations and Findings

NTS used a "modular assembly" conversion approach in accordance with standard C37.59-1991, "IEEE Standard Requirements for Conversion of Power Switchgear Equipment," of the American National Standards Institute (ANSI) and the Institute of Electrical and Electronic Engineers (IEEE). This approach comprised adapting a modular assembly, defined by the standard as the circuit breaker operating mechanism, the interrupting devices, interconnecting hardware and supporting frame, by fitting the modular assembly (supplied by Yaskawa) into a carriage with the associated equipment for electrical and mechanical interface with the existing (Magne-Blast) breaker cubicles. This composite unit is what the standard calls the "complete conversion."

b.1 Yaskawa Design Changes: During the review of the design documents relating to Yaskawa design changes, the inspectors noted that Yaskawa had made several changes to the modular assembly deisgn subsequent to the ANSI type testing.

Based upon interviews of cognizant NTS staff and examination of representative parts of a Yaskawa circuit breaker mechanism, the inspectors identified those modifications that had some potential for affecting breaker performance in the area for which original design testing had not been repeated and for which credit was being taken. The design changes of interest included (1) lengthening the trip levers, (2) replacing the original single-stage interrupter bottle gas pressure switch with a two-stage unit (for sequential alarms on lowering SF6 pressure, (3) replacing the original trip and close coils with stronger ones, (4) replacement of certain stamped, machined and/or welded parts with investment castings. The inspectors found that the post-design-testing changes made by Yaskawa either would have no effect on interrupting capacity or were validated by the subsequent C37.59 design and production testing performed on the complete conversion as follows:

In order to accommodate the linkage used in the complete conversion for the rackout interlock which trips the breaker (if it is not already open) when either beginning to disconnect the breaker from the bus in racking down or when beginning to rack the breaker up into the connect position, Yaskawa fabricated special trip levers, elongated by about 1 inch. The subsequent timing tests confirmed that this modification had no adverse effect on the tripping time.

With regard to the interrupter bottle gas pressure switches, the inspectors found that the original single-stage or setpoint switches had been replaced by Yaskawa with two-stage or setpoint swirches. However, the new switches had lower contact ratings for continuous load current as well as load current interrupting. By review of the NTS/PDS and PG&E design drawings (circuit diagrams) and technical information for the loads controlled by the pressure switch (the largest of which was a time delay relay), the inspectors determined that the contact ratings of the replacement switches were adequate. In addition, the inspectors' independent assessment of the NTS failure modes and effects analysis for the pressure switches confirmed that the credible failure mode if the contacts were underrated would be possible welding of the contacts. In this case, the low gas pressure alarm (and possibly also the low-low-pressure alarm) would be actuated, allowing the breaker to interrupt at least one more time (even if SF6 pressure were actually as low as 1 atmosphere), but preventing recharging of the closing spring, thus preventing subsequent automatic or remote operations.

With regard to the trip and closing coils, the inspectors determined that Yaskawa had replaced the original 60-ohm coils with coils of 30-ohm DC resistance coils (thus drawing more current), but with sufficient turns to make the ampere-turns product greater than that of the original coils, thus allowing the coils to develop more magnetic force on the armature/plunger. The inspectors noted that this change would tend to improve breaker performance, particularly timing, and thus not invalidate previous testing.

With regard to the originally machined, stamped and/or welded parts that Yaskawa had replaced with investment castings, the inspectors determined that the process and materials used, which had been reviewed by NTS during its commercial grade surveys of Yaskawa, produced parts that were as strong and durable as those in the ANSI-tested version. In addition, the suitability of the investment castings was further confirmed during post-modification design verification and production unit testing.

b.2 Conversion Hardware: The inspectors then reviewed the design documents for the complete conversion, interviewed cognizant NTS staff and examined representative parts of a Yaskawa circuit breaker mechanism. Design documents reviewed included Yaskawa and PDS/NTS drawings, failure modes and effects analyses, component technical information, and testing procedures and records.

During the review of the design documents for the complete conversion, interviews of cognizant NTS staff and examination of representative parts of a Yaskawa circuit breaker mechanism, the inspectors identified the hardware added by NTS/PDS to form the complete conversion. Of particular interest was that which had some potential for affecting breaker performance in the area for which original design testing had not been repeated and for which credit was being taken. The added hardware of interest included (1) the trip and close riser linkages for the cubicle interlocks, (2) the linkages and plunger assemblies that operate the stationary auxiliary switch (SAS) mounted in the cubicle or cell and (3) 10-Ohm, 0.62-Henry chokes (inductors) installed in series with the trip and spring-release coils,.

The inspector noted that the trip riser interlock linkage (of particular interest) acts to raise the trip lever instead of being raised by it and then only during racking up or down, not during breaker opening. The inspector also examined the design and hardware and confirmed that the presence of the interlock linkages did not impede the motion of the trip lever when it is operated by the trip solenoid during a remote electric trip, whether automatic or initiated by an operator.

With regard to the SAS linkage and plunger driven by the mechanical position indication mechanism, which is driven by the main contact linkage, the inspector noted that during the breaker closing cycle, when the breaker mechanism drives the SAS plunger upward, the plunger strikes the operating rod of the SAS in the cubicle and, in moving it upward to actuate the SAS, it causes the SAS reset spring to be compressed. This action stores energy that, when released, acts to help open the breaker. Thus, the inspector noted that when the trip mechanism initiates a tripping operation, the SAS is already pushing downward on the breaker mechanism-operated plunger, helping it open the breaker and overcoming any additional inertia that was added by the small mass of the SAS plunger and associated linkage. Although driving the SAS plunger has the potential to retard the closing operation (but not interrupting capacity), post-modification design verification and production unit testing established that this additional hardware did not have a noticeable effect on timing.

PDS installed approximately 0.25-Henry inductance chokes in series with both the trip and spring release coils. This also added 10 ohms DC resistance to the circuit. The intended effect of the chokes would be to limit inrush current or retard its rise, thus slowing the breaker's response to spurious trip signals that might come from protective relays that seismic testing had indicated would chatter excessively (i.e. > 2msec contact bounce) in a severe earthquake. In addition, measures had been taken to prevent the spurious signals, such that the chokes were technically no longer required. However, the chokes would also limit the maximum steady state current available after about five inductance/resistance time constants (a few milliseconds). Test data in the NTS files indicated that with the chokes, the time required to open the breaker was approximately 59 milliseconds at 70 Vdc, well below the 125-Vdc nominal control voltage at DCPP. This is less than the 5 cycles (about 83.5 milliseconds) allowed by design specifications within which the breaker must be able to clear a fault.

b.3 Design and Design Verification The inspectors confirmed that NTS and PDS completed the design and production testing prescribed by C37.59 on the complete conversion units with satisfactory results. However, not all of the design verification tests prescribed in ANSI/IEEE design standard C37.09, "Test Procedure for AC High Voltage Circuit Breakers Rated on a Symmetrical Current Basis," were repeated. In particular, the short-circuit interrupting test at full rated voltage was not repeated. Therefore, PG&E took credit for certain design verification testing conducted on the SF_6 breaker by Yaskawa, the original breaker manufacturer. The NRC had confirmed through consultation with the ANSI/IEEE Switchgear Committee subcommittee responsible for the conversion standard, C37.59, that it allowed this approach, i.e., not repeating certain design tests (in this case, the interrupting capacity test), as stated above, provided that it is established through engineering analysis that none of the changes to the modular assembly or hardware added to the modular assembly in order to create the complete conversion (done in this case by Yaskawa and PDS respectively) will invalidate the original ANSI design testing

c. Conclusions

The inspectors determined that NTS activities with regard to evaluation of Yaskawa design changes, design and constrcution of the complete conversion, and design verification of the complete conversion was adequate to assure that the replacement breakers were capable of performing their intended safety function.

3.3 Dedication of Yaskawa Modular Assemblies and PDS Conversion Hardware

a. Inspection Scope

To evaluate the dedication process, the inspectors first reviewed NTS Procedure No. 60431-95N, Revision 3, August 3, 1995, "Dedication/Acceptance Basis for Class 1E Retrofit Circuit Breakers 4kV, 350 MVA for Diablo Canyon Power Plant Units 1 & 2, Pacific Gas & Electric Company." Because certain critical characteristics were accepted on the basis of commercial-grade surveys, the inspectors reviewed reports NTS/CGS/94-015, "NTS Quality Department Commercial Grade Survey Report," , Revision 2 (June 15, 1995) and NTS/CGS/94-015A, "NTS Quality Department Commercial Grade Follow-up Survey Report," Revision 1.1 (August 15, 1995). In addition, the inspectors reviewed the NTS dispositions of the survey findings and also a representative sample of Piece Part Verification Data Sheets.

b. Observation and Findings

b.1 Critical Characteristics of Yaskawa Modular Assemblies

NTS Procedure No. 60431-95N prescribed the dedication process and called for use of a variety of dedication methods, including special tests and inspections, commercial grade survey of the suppliers and source verification. NTS dedicated the Yaskawa modular assemblies (purchased as commercial grade) by identifying and verifying the assembly's critical characteristics. NTS performed a failure modes and effects analysis (FMEA) on the critical components of the Yaskawa circuit breakers in order to identify the critical characteristics necessary for the breakers to perform their intended safety function. NTS's evaluation determined that the Yaskawa circuit breaker contained 134 piece parts which have critical safety functions. Further review of the 134 parts identified 36 critical characteristics. NTS then identified 22 specific piece parts/assemblies which collectively encompassed all 36 critical characteristics. NTS conducted a performance-based commercial grade survey of Yaskawa in order to evaluate quality and manufacturing controls for the 22 parts which covered all critical characteristics. In addition, the 22 parts were selected to cover major components of the completed modular assembly, including the operating mechanism (8 items), primary circuit assembly (9 items) and control circuit items (5 items). The survey reports included Piece-Part Verification Data Sheets which listed the part evaluated and identified the critical characteristics, the Yaskawa work operating standard that controlled the manufacturing process for the characteristic, the method used to verify each characteristic, and acceptance criteria. The data sheet also included a remarks section which provided specific information regarding the survey team member's observations. The inspectors' review of the representative sample of the data sheets did not identify any concerns.

The commercial grade survey reports documented the results of the two commercial grade surveys performed on June 20-24, 1994 (94-015) and October 24-28, 1994 (94-015A), respectively. The second survey was performed to complete some activities that NTS was not able to perform during the first survey and to follow-up on findings and open items identified during the first survey. The surveys assessed several Yaskawa design and manufacturing processes, such as design control, procurement control, material control, manufacturing and process controls, inspection and test control, and measuring and test equipment.

The inspectors noted that NTS identified several findings and open items during the initial survey. All of the findings and open items were closed satisfactorily during the second survey. The surveys identified the need to include specific requirements in the NTS purchase order to (1) assure Yaskawa informs NTS of any significant design changes that could affect the breaker's form, fit, function, or materials, and (2) specify

NTS requirements designed to supplement Yaskawa standard quality practices. In addition, NTS identified the need to perform some tests and inspections to supplement the control and verification of some critical characteristics performed by Yaskawa.

As a result of the review of NTS's disposition of the survey findings, the inspectors determined that the basis for accepting the Yaskawa's corrective actions was adequately documented. The inspectors had no concerns regarding the commercial grade surveys of Yaskawa.

b.2 Fasteners

The inspectors reviewed NTS activities related to the fasteners used in the Yaskawa modular assemblies and in the NTS/PDS complete conversion. The fasteners used in the complete conversion had been provided by NTS/PDS and had been specified as SAE Grade 5 (hex head cap screws). The inspectors reviewed Procedure No. 60431-95N-1466-FAS, "Receipt Inspection and Sampling Procedure for Safety-Related Fasteners for the PG&E Retrofit Circuit Breakers," Revision 2, dated January 1, 1996, which provided instructions to verify the acceptability of fasteners by including receipt inspection, thread dimensional verification, magnetic testing, hardness testing, and materials analysis of the safety-related fasteners.

All fasteners used in the Yaskawa modular assembly had been purchased from a Japanese Industrial Standard (JIS) supplier. A JIS supplier is approved by the Japanese Ministry of International Trade and Industry (MITI). Suppliers are approved based on their ability to provide a specific product in accordance with the applicable JIS Standard (the standards specify necessary quality and technical attributes of products and materials and their manufacture). A manufacturer or supplier is only approved after MITI investigates the supplier's quality assurance program and process control for compliance to the standard. Following approval, the supplier is permitted to place a JIS mark on the approved commodity. The JIS mark is not transferable across company or product lines. The inspectors reviewed documentation available at NTS which described the JIS Marking System and the authorities, responsibilities and processes used by MITI. The fasteners provided in the Yaskawa modular assemblies had been supplied as original circuit breaker components and had been identified as "4T." The fasteners had been procured from a MITI approved JIS supplier. In addition, NTS had tested a representative 4T fastener, determined it to be equivalent to SAE Grade 2 and acceptable for the application.

In addition, during a telephone discussion with NTS and the NRC inspectors, PG&E indicated that PG&E had reviewed the existing breaker cubicle, duplicated the cubicle, installed the complete conversion using the provided NTS fasteners, and had performed a seismic test which qualified both the fasteners used in the modular assembly and those used in the complete conversion. PG&E had determined, as verified by testing, that the modular assembly and the complete conversion was capable of performing its safety function under the design commitments. The inspectors concluded that the NTS/PDS and PG&E had taken adequate action to verify the suitability of the fasteners used in the modular assemblies and the complete conversions.

c. Conclusions

The inspectors determined that the NTS commercial grade survey verified that Yaskawa's commercial quality program was documented and effectively implemented. The surveys examined Yaskawa's controls and processes for individual components and specific critical characteristic and provided assurance that the NTS identified critical characteristics were properly controlled.

3.4 Entrance and Exit Meetings

At the entrance meeting on January 6, 1998, the NRC inspector discussed the scope of the inspection, outlined the areas to be inspected, and established interactions with NTS management and staff. In the exit meetings on January 8, 1998, the inspectors discussed their findings and observations.

4 PARTIAL LIST OF PERSONNEL CONTACTED

NTS

Timothy J. Rotti, Division Manager
Christine C. Briggs, QA Manager
Keith Pogarian, Project Engineer
Daniel R. Cannon, Project Engineer

National Quality Assurance, USA

James E. Dozier, Regional Manager

PG&E

Mohsin Kahn, Principal Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

99900912/93-01-01	VIO	Failure to meet 10 CFR 21.21(a) and 21.6 requirements
99900912/93-01-02	NON	Failure to incorporate design requirements in dedication testing
99900912/93-01-03	URI	Determine basis for revised trip time tolerances for K-M relays
99900912/93-01-04	OPEN	Evaluate performance anomalies seen in K-M relay testing
99900912/93-01-05	OPEN	Determine impact electrical cable test failure
99900912/93-01-06	OPEN	Determine impact of SOR pressure switch test failures

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

March 18, 1998

Mr. William C. Whitehead, Plant Manager
Westinghouse Electric Company
Specialty Metals Plant
R D #4, Box 333 Westinghouse Road
Blairsville, Pennsylvania 15717-8904

SUBJECT: NRC INSPECTION NO. 99900005/98-01

Dear Mr. Whitehead:

During the period February 18-20, 1998, the U. S. Nuclear Regulatory Commission (NRC) performed an inspection of the Westinghouse Electric Corporation Specialty Metals Plant (SMP), located in Blairsville, Pennsylvania. The enclosed report presents the results of that inspection.

The purpose of the inspection was to review the manufacturing and testing activities and determine if these activities were conducted in accordance with NRC requirements. The inspectors reviewed documentation associated with the manufacture of zirconium alloy fuel clad tubing used in nuclear power reactors. Within this area, the inspection consisted of an examination of procedures and representative records, interviews with SMP personnel, and observations by the inspectors. The inspectors did not identify any instances where the SMP quality assurance program failed to meet NRC requirements for the areas inspected. Therefore, no response to this letter is required.

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

Sincerely,

Suzanne C. Black

Suzanne C. Black, Chief
Quality Assurance, Vendor Inspection, and
Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket No.: 99900005

Enclosure: Inspection Report No. 99900005/98-01

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report No.:	99900005/98-01
Organization:	Westinghouse Electric Company Specialty Metals Plant Commercial Nuclear Fuel Division Blairsville, Pennsylvania 15717-8904
Contact:	Mr. Joseph H. Ewing, Manager Product Assurance Department
Nuclear Industry Activity:	Westinghouse Specialty Metals Plant manufactures zirconium alloy fuel clad tubing for use in the nuclear power industry.
Inspection Dates:	February 18-20, 1998
Inspectors:	Robert Pettis, Jr.,HQMB/DRCH Gregory Cwalina, HQMB/DRCH Donald Naujock, ECMB/DE
Approved:	Robert A. Gramm, Chief Quality Assurance and Safety Assessment Section Quality Assurance, Vendor Inspection, and Maintenance Branch Division of Reactor Controls and Human Factors

1 INSPECTION SUMMARY

The purpose of the inspection was to review the manufacturing and testing activities for the Specialty Metals Plant (SMP) and determine if these activities were conducted in accordance with NRC requirements. The inspection bases were as follows:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of the Code of Federal Regulations (CFR), 10 CFR Part 50.

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

2 STATUS OF PREVIOUS INSPECTION FINDINGS

The previous inspection, 99900005/95-01, did not identify any findings.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background

The Westinghouse SMP, part of Westinghouse Commercial Nuclear Fuel Division (CNFD), located in Blairsville, Pennsylvania, manufactures zirconium (Zr) alloy (zircaloy) tubing for use in the nuclear power industry. Although production at the plant in Blairsville began in 1955 (e.g., nuclear fuel pellets through 1960, stainless steel turbine blades, and forged bar and strip products), the manufacturing of zircaloy tubing started in 1967, and the manufacturing of inconel steam generator tubing began in 1968. In 1985, the manufacturing of inconel was discontinued and the CNFD SMP was committed completely to zirconium-alloy-based nuclear-grade (a) tubing for fuel rod cladding, (b) tubing for discrete burnable absorber rod cladding, and c) tubing for thimble tubes, instrumentation tubes, sleeves, spacers and connectors. According to CNFD SMP, it has produced over 70 types and sizes of zirconium alloy tubing. Typical products produced at SMP are Zircaloy-2 for boiling-water reactors, Zr4 for PWRs, ZIRLO™ for longer operating cycles and higher burnups, guide thimbles, and burnable absorber tubes.

3.2 Fuel Rod Manufacturing Process

The inspectors evaluated the adequacy of the fuel rod manufacturing process (see Figure 1), including process control, manufacturing operations, and in-process checks and inspections. Specifically, the team observed pilgering, cleaning, pickling, straightening, grit blasting, cutting and finishing of the rods. The inspectors reviewed applicable procedures and interviewed several manufacturing process operators and in-process technicians during the inspection.

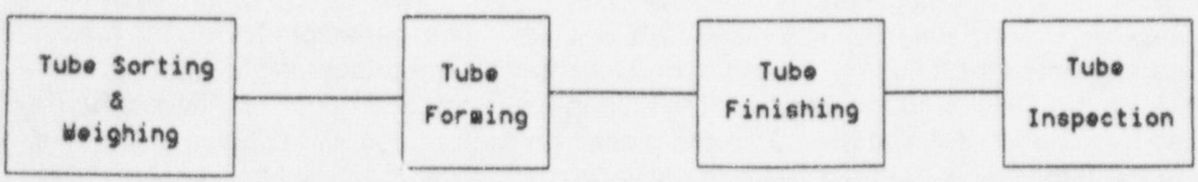

Figure 1 - Zircaloy Tube Manufacturing Process

3.2.1 Follow Cards and Inventory Control

The inspectors reviewed SMP procedure PE (Process Engineering)-200, "General Manufacturing Requirements," and observed SMP processes and procedures for assuring material traceability and process control. Prior to manufacture, each manufacturing lot is given a lot number for process tracking and inventory control. Each production lot is segregated and will be accompanied by a follow card for the entire manufacturing process. The follow card contains the lot number (which is also etched on the stock as it moves through the manufacturing operations in accordance with PE-200), original ingot number (from the Western Zirconium facility), customer identification and applicable drawing number. The follow card also identifies each manufacturing step (operation) and the appropriate PE procedure for performing the step, and any other requirements (e.g., weighing and recording). The lot number and each individual manufacturing step are also identified by a unique bar code, included on the follow card, which is read into the computer tracking system by the operator at the beginning and end of each operation. The operator is required to enter into the computer the bar code of each lot, the process operation number, the stage of the process (e.g., start or stop), and the operator's identification which is read off the operator's badge. The computer tracks the manufacturing progress and prevents a step from being started before the previous step is complete. The follow card also requires the operator to initial for the completion of each step and to record the date and amount of material processed and scrapped in accordance with PE-200.

The NRC inspectors observed operators using the follow cards during several manufacturing operations. Operators logged in all pertinent information and utilized the bar code readers with no difficulty. In response to an inspector questions, an operator logged incorrect information into the system. The inspectors noted that the computer tracking system identified the errors and required the operator to provide corrections before allowing the operator to proceed. The inspectors inquired as to the difficulty of using the system. Operators indicated that the current system worked very well and did not express any concerns. Some operators did identify that a few problems had existed in the past, when pencil style bar code readers were used. However, the switch to pistol grip readers approximately a year ago has significantly reduced any problems in using the bar code system.

The inspectors concluded that the current manufacturing process provides sufficient controls to assure proper material traceability throughout the manufacturing process and to track manufacturing progress of the individual lots.

3.2.2 Sorting

The inspectors reviewed PE-201, "TREX Sorting and Weighing," and observed TREX (Tube Reduced Extrusions) that had been sorted and identified for manufacturing to assure SMP's method for identifying material for manufacture was adequate. SMP receives Zr4 and ZIRLO™ TREX from Westinghouse's Western Zirconium facility located in Ogden, Utah. After receipt, SMP sorts the TREX into production lots, the lot size depending upon the final product. For example, fuel rods are typically manufactured from lots in the 550-650 pound range, while thimble tubes lots usually weigh approximately 250 pounds. The inspectors toured the fuel sorting area and noted the TREX had been sorted into specific manufacturing lots, lot numbers were etched on the TREX and the weight of the lot had been noted. In addition, the lots had been sorted by material, Zr4 and ZIRLO™, and placed on color coded skids. Several lots had been tagged with the follow card and were ready to be moved into manufacture. Lots for which the follow card had not yet been prepared were identified by color coded tracking cards until the follow cards could be attached. Based upon sorting area observations, the inspectors concluded that SMP's method for lot preparation was adequate.

3.2.3 Tube Forming

The inspectors reviewed procedures and observed SMP forming operations, including tube reduction (by pilgering), deburring, cleaning, pickling and annealing. SMP receives hollow TREX approximately 2.5 inches in diameter from Western Zirconium. Fuel clad tubing is produced by reducing the outside diameter (OD) and the TREX wall thickness through three basic reduction steps (described below) to form fuel rods. The production steps include cold pilgering, deburring, cleaning, pickling and annealing, as well as in-process Ultrasonic (UT) and dimensional checks (see Figure 2). The final rods are then sent through the tube finishing and final inspection processes. All activities were performed in accordance with Follower Cards (travelers) and written procedures.

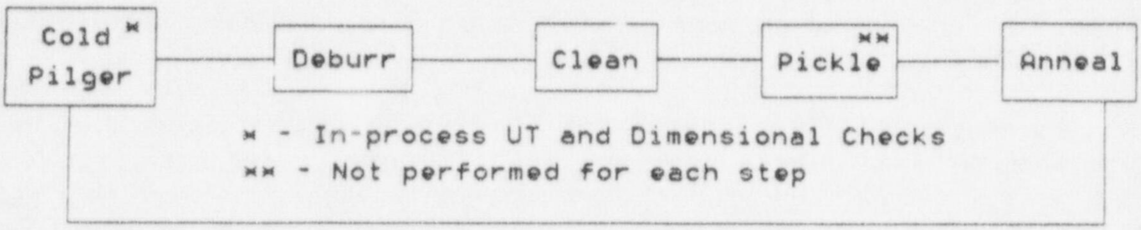

Figure 2 - Zircaloy Tube Forming Process

The inspectors were provided a typical follow card for producing .360" OD ZIRLO™ fuel tubes and followed the process used for that item. In some cases, depending upon the product, the manufacturing process or sequence varied slightly.

Pilgering

Fuel rods are produced by cold reducing TREX in three successive steps through cold pilgering until the final product size is reached. This process, performed in accordance with PE-211, "Cold Pilgering," accomplishes tube elongation and wall reduction by rolling TREX back and forth between two grooved dies. During this process, the tubes are rotated and advanced in small increments over a stationary mandrel. These tube hollows are then cut to a specified length, deburred and engraved with the lot number. The cut tubes are placed in a wet holding tank until the run lot is completed. The holding tank keeps the tube lubricant moist, making it easier to remove in the cleaning process. The inspectors observed the pilgering process and discussed the process with pilger operators. The process was being performed in accordance with the specified procedure. The inspector had no concerns in this area.

In-Process Testing

During the pilgering process, several samples are taken to assure the pilger machines are operating properly and producing tubes within specified acceptance ranges. SMP personnel use the test results as a go no-go test for releasing the material to the next processing step. The number and sizes of the samples are given in Parameter Sheet 5.1 of PE-211. There are two basic sample sizes: 1) the first is a long sample (full rod length) which is sent for UT examination, 2) the second is a short sample consisting of a an approximately 3-inch section and another 1-inch section. The pilger operator cuts and prepares a 3-inch short sample and performs an initial look by measuring the ID with an air gage and the OD with a laser micrometer. The sample is then placed on a tray marked with the pilger machine identifying number. An in-process technician is responsible for measuring and recording the sample information.

The 3-inch sample is checked for ID and OD by using the air gage and laser micrometer and the outside surface is visually examined. The 1-inch sample is cut (by the operator) lengthwise and is visually examined for inside surface defects. The inspector observed an in-process technician reject a 3-inch sample due to outside surface indications. The pilger operator changed pilger dies on the machine in question, before continuing the production run, after being informed of the problem by the technician.

The long sample is checked by UT and the UT results are checked by the in-process technician and plotted on the same control chart as the short samples. The inspectors determined that the UT machine is calibrated for ZIRLO™, although Zr4 is also examined using the same machine. Discussions with the technicians determined that there is a very slight difference between the materials. Therefore, the technicians must be aware of that when making their determinations. However, the technicians informed the inspector that the control charts are plotted using the actual measurements, not adjusted values.

The inspectors observed the pilger operator cut, prepare and measure samples. The inspector also observed the in-process technician measure, examine and plot the results for the short samples and review and plot the UT results on the long samples in accordance with PE-469, "Pilger Control Chart Forms." The inspectors determined that samples were taken, measured and recorded in accordance with applicable procedures. The inspectors reviewed a completed control chart for a .374-inch diameter fuel tube and noted that the ID and OD variance was greater for the long sample than the short sample. The operators explained that the long sample UT results were taken for an entire tube length as opposed to the short sample length of approximately 3 inches. Therefore, it is expected that the variance would be greater. The inspectors examined several other charts and noticed similar results. In all cases examined, the long sample results bounded those taken on the short samples. The inspectors concluded that the in-process testing performs an adequate function for providing the process operators with timely input on process quality. The inspectors had no concerns in this area.

Cleaning

Following the pilgering process, the tube hollows were transferred to the pickle house for cleaning and pickling (if necessary). Cleaning is accomplished in accordance with PE-205, "Alkaline Cleaning Pilgered Tubes." First, the tube hollow bundles are rinsed with service water for several minutes. The bundles are then lifted in slings and immersed in a heated alkaline solution for several minutes, periodically being lifted and drained to remove most of the loose dirt. The bundles are then rinsed again. Final cleaning is accomplished after the tubes are built into a carrier (an array which separates the tubes allowing more thorough cleaning of each tube). Again, the tubes are immersed in an alkaline solution for several minutes, being raised to drain and redipped several times. The tubes are rinsed to remove most of the soap from the tubes, then are rinsed in a second tank. Two more rinses are performed, one with service water and one with deionized water. The tubes are then hung to drain before being placed in a dryer. The cleaning process is performed after each pilger reduction. The inspectors observed several bundles being cleaned, of various size tube hollows. The cleaning was accomplished in accordance with the requirements of procedure PE-205. The inspectors did not identify any concerns in this area.

Pickling

After cleaning, the tube hollows are pickled. The pickling process consists of acid etching the tubes to remove a small amount of material and provide a smooth, uniform finish, both inside and out. SMP utilizes two pickling methods, depending upon the material and process requirements. For the example follow card provided to the inspectors, the tubes are to be ID flush pickled after the first pilger reduction. This process is controlled by procedure PE-210, "Flush Pickling." The tubes are mounted in a plastic-lined tank and attached to a manifold. The tubes are then flushed with water, acid etched and then flushed again. Wall thickness is measured before and after pickling to assure stock removal is within specified limits. The inspectors observed the flush pickling process performed on tubes following their first reduction and discussed the process with the pickle operator. The operator measured and recorded the wall thickness at the same location before and after pickling.

In addition to flush pickling the ID, all tubes are subjected to bright pickling of both the ID and OD after the final pilger reduction and cleaning. PE-209, "Bright Pickling," specifies the requirements and acceptance criteria for this pickling process. In this process, the tubes are dipped into the acid solution to remove stock material from both the inside and outside surfaces and then immediately dipped in a rinse tank. (Note: when performing bright pickling, the tubes are not rinsed with deionized water or dried first). The number of dips is determined by the strength of the acid solution. Measurements are taken and plotted before and after dipping. The inspectors observed the bright pickling process performed on tubes following their final reduction and discussed the process with the pickle operator who performs and records all measurements as required. The inspectors did not identify any concerns in this area.

Annealing

Following each cleaning/pickling operation, the tube bundles are placed in a furnace for annealing. The annealing process heat treats the tubes to relieve stresses caused by the pilger process, permitting the tubes to continue through the process, either the next pilger reduction or tube finishing. Annealing is done in either cold or hot-walled furnaces at a specified temperature and time. The inspectors had no concerns with the annealing process.

The inspectors determined that the tube forming process was accomplished in accordance with approved procedures. The inspectors noted some minor discrepancies with some procedures. Those discrepancies were discussed with SMP personnel who agreed to make necessary changes. No other concerns were identified.

3.2.4 Tube Finishing

The inspectors observed SMP tube finishing operations to assure they were being accomplished in accordance with approved procedures. At SMP, tube finishing includes tube straightening, ID blasting, cutting, polishing and final cleaning (see Figure 3).

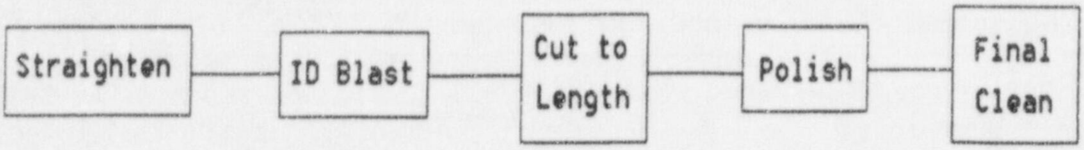

Figure 3 - Zircaloy Tube Finishing Process

Straightening

Tube straightening was performed by machine. The inspectors observed that each tube is run through a series of offset rollers which slightly bend then straighten the tubes. Each tube is monitored by in-process checks. The inspectors did not identify any concerns in the tube straightening area.

Inside Diameter Blast

The inside diameter of the tubes are then blasted to provide a final internal polish and remove any residual manufacturing imperfections. The tubes are blasted using a specified size grit. Blasting can be done in either or both directions. The inspectors observed the grit blasting process and discussed the process with the operator. The machine automatically monitors the grit flow and shuts down when flow is insufficient The inspectors were present during an automatic shutdown and observed the operator adjust the machine and perform some surface cleaning. The machine did not experience any further problems. The inspectors did not identify any concerns with the grit blasting process.

Cutting

The tubes are then cut to a specified length, and the ends were faced, deburred, and checked for squareness. The cutoff machine number was recorded, the weight measured and recorded, and the follow card signed by the operator. The inspectors observed the operators set up and operate the machine to perform automatic cutting and measuring of end squareness. Cutting is accomplished by removing a specified length from one end and the rest of the material from the opposite end. The inspectors did not identify any concerns in the cutting area.

Finishing

At this point tubes are sent to the finishing cell. This cell cleans the tubes using a specified cleaning solution, rinses and dries the tubes, inserts polyurethane end plugs and polishes the outside surface. The inspectors observed an operator load and operate the finishing cell and also observed another operator prepare and operate the surface polisher. The operator adjusted the pressure of the polishing belts (the machine includes three belts of specified grit). The pressure is adjusted to remove a specified amount of material from the OD. A test piece is measured, run through the polisher, and remeasured to determine the amount of material removal. During this inspection, the inspectors observed the first test piece was processed and found to have had too much material removed. The operator readjusted the machine and polished another test piece within acceptable limits. Finally, the tube is passed through an alloy verification system to assure that the proper alloy was being supplied to the customer. ZIRLO™ tubes are marked with their trade name at the end of each tube.

Based upon observations and discussions with operators, the inspectors determined that the manufacturing and inspection activities were performed in accordance with established SMP procedures. The inspectors did not identify any concerns in this area.

3.2.5 Tube Inspection

The inspectors observed SMP final tube inspection operations to assure they were being accomplished in accordance with approved procedures, including Procedure PA-212, "Dimensional Standards for Zirconium Alloy Tubing," Revision 5, dated August 22, 1994 and the UT portions of Procedure QC-301, "Final Inspection - Ultrasonic Dimensional Setup and Calibration," Revision I, dated July 6, 1989, and Procedure QC-302, "Final Inspection - Ultrasonic Flaw Setup," Revision 24, dated October 15, 1997.

Final tube inspection involves UT examinations, and inspections of ID, length, end squareness and OD surface (see Figure 4). The inspectors reviewed nondestructive examinations (NDE) of fuel clad tubing. SMP performs UT for OD, ID, wall thickness, and flaw detection on 100% of the Zr4 and ZIRLO™ finished tubing produced for nuclear application.

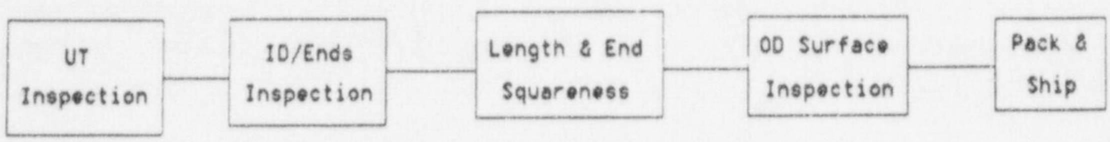

Figure 4 - Zircaloy Tube Inspection Process

UltrasonicTesting (UT)

SMP performs final UT inspection to the recommendations of the 1991 Edition of the American Society for Testing and Materials (ASTM) B 353, "Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service," Annex A3, "Recommended Procedure for Ultrasonic Testing of Zirconium and Zirconium Alloy Tubing for Nuclear Service," and ASTM B 811, "Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding," Annex A3, "Procedure for Ultrasonic Flaw Testing of Zirconium Alloy Nuclear Fuel Cladding Tubes." The inspectors examined UT equipment installed at high speed UT Inspection Station 12. The equipment consisted of a personal computer connected to two test units, a fiber optics positioner, and a strip chart recorder. One test unit examines the tube for flaws using 45 degree shear wave, 10 MHZ, cylindrically focused transducers. These transducers are positioned to examine the tube in two transverse directions and two longitudinal directions. The other test unit examines tube dimensions with 0 degree, 10 MHZ transducers located on both sides of the passing tube; a third transducer measures the effects of water temperature change on the UT signals.

The inspector observed a SMP Level II UT examiner demonstrate setting up UT Inspection Station 12 for a run of 0.360 OD ZIRLO™ fuel tubes. The demonstration started with verifying the calibration of "dimensional standard" 31495Z (0.360 OD ZIRLO™ tube). The examiner used calibrated micrometers and UT to determine wall thickness, a laser micrometer for OD measurements, and a calibrated air gage for ID measurements. After verifying the dimensional standard, the examiner checked the dimensions of a "reference standard" tube. The reference standard is used every two hours or more frequently, as needed, during the run to verify the dimensional accuracy of the UT equipment.

For the demonstration, the Level II examiner used Inspection Station 12. The setup was verified under dynamic conditions by running the dimensional standard, reference standard, and flaw standard through the inspection station. The flaw standard tube has one transverse and one longitudinal electrical discharge machined notches located according to SMP specifications. Following the demonstration, the inspectors observed the operators at Inspection Station 4 inspecting ZIRLO™. The operators inserted the reference standard and flaw standard into a lot of ZIRLO™, a part of the routine dynamic calibration checks required by SMP's procedures. The inspection station automatically culls rejected tubes from the lot and separates them by defect type. After running the lot, the rejected tubes were checked for ID moisture and retested using a slower inspection speed.

Final Inspection

After UT inspection, the inspection station operator performs a visual examination for camber, stains, and pin hole defects (pitting). Pin hole defects are usually too small for the high speed automatic UT to detect. The inspectors observed two tubes that a visual examiner had rejected for surface flaws, one for small pits and the other for spiral stains. Both tubes had successfully passed the UT examination.

The tubes are then passed on for further visual and other examinations. The ID surface was examined by visually examining the tube ID from both ends against a lighted background. The finish on the ends was also examined. Length and end squareness were checked, as well as straightness and ID at the ends. The OD surface was visually examined and a mechanized OD surface examination was performed to check for surface roughness. Throughout the process, the number of pieces accepted, reworked, and scrapped was recorded and the Follow Card was signed by the operator.

Handling, Storage, and Shipping

The inspectors evaluated the packaging and shipping of tubing with respect to the protection of the metal surface condition during shipping. Full sheets of Styrofoam contoured to match the geometry of the tubing were used to separate the full length of each layer of tubing within heavy wooden boxes lined with thick brown paper. The packaging appeared effective and had not resulted in any reported shipping damage. The inspe... s concluded that the 100% final UT and visual inspection process provides an adequate ... ction to insure product quality and reliability.

3.2.6 UT Technician and Pilger Operator Training

The NRC inspectors reviewed SMP Procedure PA-204, "Inspection - NDE Certification," Revision 10, July 18, 1995, which specifies the training, qualifications and certification of personnel required to perform nondestructive examination methods and techniques.

The inspectors observed the training records of one UT Level II examiner. UT training followed the guidelines in American Society for Nondestructive Testing SNT-TC-1A, 1988 Edition. The Level II examiner received his Level I training in-house under a program developed by SMP's Level III examiner. For Level II training, SMP's Level III examiner reviewed and approved a training program administered by US Air, Pittsburgh, PA. The inspectors determined that SMP training records satisfied the criterion contained in the guidelines.

The inspectors reviewed a sample of training records for several pilger operators to determine if they were trained in the various PEs necessary to perform their job responsibilities. For example, a review of the records of one pilger operator indicated he was trained to PE-460, "Instructions-Cold Pilger," dated June 1994; PE-469, "Pilger Control Charts," dated July 1995; PE-469, Revision 051, dated September 1995; PE-469, Revision 055, dated January 1996, and PE-211, "Cold Pilgering," Revision 26, dated January 1997. Based on the sample selected, the inspector determined that the training was satisfactory.

3.3 Entrance and Exit Meetings

An entrance meeting was held on February 18, 1998, in which the scope of the inspection was discussed with SMP management and staff. On February 20, 1998, an exit meeting was held with SMP to discuss the inspection.

4 PERSONNEL CONTACTED

The following represents a partial list of persons contacted during the inspection:

Specialty Metals Plant

Joseph Ewing, Manager, Product Assurance
Brian Jones, Manager, Business Process Improvement
Richard Kaiser, Manager, Production Services
Jerry Leysock, Manager, Product Assurance Engineering
Chris Mitchell, Manager, Human Resources
Ron Weisser, Manager, Product Assurance Operations
Chris Skupien, Manager, Materials Management
Mark Troxell, Audit Coordinator
Carrie Monaco, Customer Projects
William Jacobsen, Process Engineering

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

January 26, 1998

Mr. John R. Heine, President
Westlectric Castings, Incorporated
2040 Camfield Avenue
City of Commerce, California 90040

SUBJECT: NRC INSPECTION REPORT 99901323/97-01 (NOTICE OF VIOLATION AND NOTICE OF NONCONFORMANCE)

Dear Mr. Heine:

On November 17-19, 1997, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Westlectric Castings, Incorporated (Westlectric), facility. The enclosed report presents the results of that inspection.

During this inspection, the NRC inspectors noted that Westlectric personnel were knowledgeable and competent in the performance of their job functions. However, the NRC Inspectors found that certain of your activities appeared to be in violation of NRC requirements. Specifically, although 10 CFR 21.21, "Notification of failure to comply or existence of a defect and its evaluation," requires that each corporation subject to Part 21 regulations adopt procedures to ensure the evaluation and proper reporting of deviations and failures to comply is performed, Westlectric did not establish an adequate procedure. This violation is cited in the enclosed Notice of Violation (NOV), and the circumstances surrounding the violation are described in detail in the enclosed report. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed NOV when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In addition, the NRC inspectors found that the implementation of your quality assurance program failed to meet certain NRC requirements imposed on you by your customers. Specifically, the inspectors determined that compliance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," was contractually imposed on Westlectric in approximately 1993 by Pacific Pump (currently Ingersoll-Dresser Pump, Inc. (IDP) and Bryon Jackson Pump Division (BW/IP). Although Westlectric unconditionally accepted the quality assurance (QA) provisions of 10 CFR Part 50, Appendix B, Westlectric did not establish an adequate QA program to specifically identify or address certain of the Appendix B criteria that were applicable to Westlectric activities affecting the quality of the components supplied for safety-related use.

These nonconformances are cited in the enclosed Notice of Nonconformance (NON), and the circumstances surrounding them are described in detail in the enclosed report. You are requested to respond to the nonconformances and should follow the instructions specified in the enclosed NON when preparing your response.

Given the extent of the concerns identified by the NRC inspectors in the enclosed Inspection Report, the adequacy of the quality oversight functions for the spectrometer operations is questionable. Therefore, you are also requested to respond regarding the review that Westlectric has implemented to identify: (1) whether your chemical analyses of nuclear component castings has been correctly performed, (2) whether the spectrometer operator training was adequate, (3) whether the spectrometer correction factor practice was appropriate, (4) whether any shipped casting's chemical analyses are questionable, and (5) whether appropriate quality verification activities had been performed to ensure the integrity of the spectrometer operations.

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 (PDR).

Sincerely,

Suzanne C. Black

Suzanne C. Black, Chief
Quality Assurance, Vendor Inspection
and Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket No. 99901323

Enclosures: 1. Notice of Violation
2. Notice of Nonconformance
3. Inspection Report 99901323/97-01

NOTICE OF VIOLATION

Westlectric Castings, Incorporated
City of Commerce, California

Docket No.: 99901323

During an NRC inspection conducted on November 17 through 19, 1997, 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, "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, Quality Assurance (QA) Procedure QA104, "Part 21 - Reporting of Defects and Noncompliance," Revision A, dated January 26, 1996 failed to address the evaluation of deviations. (99901323/97-01-01)

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

Pursuant to the provisions of 10 CFR 2.201, Westlectric Castings, incorporated, is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Chief, Quality Assurance, Vendor Inspection, and Maintenance Branch, Division of Reactor Controls and Human Factors, 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. 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.

Dated at Rockville, Maryland
this 26th day of January 1998

Enclosure 1

NOTICE OF NONCONFORMANCE

Westlectric Castings, Incorporated
City of Commerce, California

Docket No.: 99901323

Based on the results of an inspection conducted on November 17 through 19, 1997, it appears that certain of your activities were not conducted in accordance with NRC requirements.

A. Criterion I, "Organization," of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Appendix B), requires in part, that the authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing.

Criterion II, "Quality Assurance Program," of Appendix B, requires that a quality assurance program which complies with the requirements of Appendix B be established.

Westlectric QAM Procedure Section 1.0, "General Policy," states that "all inspectors, including M.T. and P.T. Level II who are under the direct supervision of the Quality Assurance Assistant are responsible for the proper execution of required N.D.T."

Contrary to the above, the NRC found that the authority and duties of all Westlectric Quality Assurance department personnel were not accurately delineated in the Westlectric Quality Assurance Manual (QAM), Revision K, dated October 14, 1996. Specifically: (Nonconformance 99901323/97-01-02)

1) Westlectric has not had a QA Assistant since approximately 1991.

2) Until recently the visual inspector worked for and reported to production department personnel.

3) Westlectric has had only one NDE certified inspector (Level II) since approximately 1990 even though the QAM indicates it has multiple NDE Inspectors.

4) The Level II NDE Inspector also holds the position of "Chief Inspector," the authority and responsibilities of the Chief Inspector were not delineated in the QAM.

5) The only Westlectric NDE Inspector (the Chief Inspector/Level II NDE Inspector) reports to the Operations Manager, who is responsible for the activity being inspected.

6) Although both Westlectric visual inspectors report to the QA Manager, they also take NDE and inspection direction from the Chief Inspector/Level II NDE Inspector.

B. Criterion V, "Instructions, Procedures, and Drawings," of Appendix B requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Paragraph 1.4 of Section 3.0, "Quality Control Department Operation and Duties Procedure," of Westlectric's QAM states: the QA Manager will maintain a file in his office of all inspection stamps issued to workers.

Paragraph 1.2 of Section 19.0, "Casting Traceability," of Westlectric's QAM states heat code numbers shall be affixed to pattern with raised aluminum letters or pressed into molding sand with letters attached to a handle.

Contrary to the above: (Nonconformance 99901323/97-01-03)

1) two inspection signature stamps, used for the stamping of approval signatures on Westlectric CMTRs and NDE records, in the possession of the Chief Inspector and QA Manager were not identified as being issued on the QA manager's inspection stamp file. As a result, no record existed regarding the signature stamps even though they were used for approving quality records.

2) Although Westlectric's manufacturing process can result in situations where heat numbers are occasionally changed, the QAM and implementing procedures did not specify that grinding out portions of cast heat numbers and replacing them with stamped numbers was an allowable practice.

C. Criterion XII, "Control of Measuring and Test Equipment," of Appendix B requires that measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

Criterion XVII, "Quality Assurance Records," of Appendix B requires that sufficient records be maintained to furnish evidence of activities affecting quality. The records will include at least the following: Operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records shall also include closely-related data such as qualifications of personnel, procedures, and equipment. Records shall be identifiable and retrievable.

Westlectric QAM Procedure Section 16.0, "Calibration Procedures," states that "all calibration shall be performed in accordance with MIL-C-45662 by a qualified outside calibration source and traceable to NIST Standards."

Contrary to the above, Westlectric staff was unable to provide calibratio ecords for the spectrometer curve-sets, did not have traceability to NIST for its standaros, and did not have adequate information regarding the accuracy of the curve-sets. Additionally, Westlectric did not establish specific documented procedures or instructions delineating acceptance and rejection limits on the analysis range for each element affected by one-point standardization. (Nonconformance 99901323/97-01-04)

D. Criterion IX, "Control of Special Processes," of Appendix B requires that measures be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Paragraph 1.5.1 of Procedure 11.0, "Metal Control," of Westlectric's QAM stated the chemical analysis testing shall be performed by a trained laboratory technician.

Contrary to the above, Westlectric did not establish adequate procedures or appropriately trained personnel to control its spectrometer chemical analysis' operation. (Nonconformance 99901323/97-01-05)

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, Quality Assurance, Vendor Inspection and Maintenance Branch, Division of Reactor Controls and Human Factors, 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 26th day of January 1998

U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report No:	99901323/97-01
Organization:	Westlectric Castings, Incorporated 2040 Camfield Avenue City of Commerce, California 90040
Contact:	Andrew Arechiga, Quality Assurance Manager (213) 722-8000
Nuclear Industry Activity:	Manufacturer of low carbon and stainless steel sand castings such as impellers, valve bodies and pump replacement parts.
Dates:	November 17 - 19, 1997
Inspectors:	Joseph J. Petrosino, NRR Donald G. Naujock, NRR
Approved by:	Robert A. Gramm, Chief Quality Assurance and Safety Assessment Section Quality Assurance, Vendor Inspection and Maintenance Branch Division of Reactor Controls and Human Factors Office of Nuclear Reactor Regulation

Enclosure 3

1 INSPECTION SUMMARY

During this inspection, the NRC inspectors reviewed the implementation of selected portions of the Westlectric Castings, Incorporated, quality assurance (QA) program, and reviewed activities associated with its manufacture and supply of low carbon and stainless steel sand castings to the nuclear industry.

The inspection bases were:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the Code of Federal Regulations (Appendix B)

- 10 CFR Part 21, "Reporting of Defects and Noncompliance"

During this inspection, a violation of NRC requirements was identified and is discussed in Section 3.1 of this report. Additionally, several instances where Westlectric Castings, Incorporated (Westlectric), failed to conform to NRC requirements contractually imposed upon them by sub-tier suppliers for NRC licensees were identified. These nonconformances are discussed in Sections 3.2, and 3.4 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

This was the first NRC inspection of the Westlectric facility.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 10 CFR Part 21 Program

a. Inspection Scope

The NRC inspectors reviewed Westlectric's procedure for reporting in accordance with 10 CFR Part 21: QA Procedure QA104, "Part 21 - Reporting of Defects and Noncompliance," Revision A, dated, January 26, 1996. The NRC Inspectors also observed and reviewed Westlectric's 10 CFR Part 21 posting.

b. Observations and Findings

The procedure did not address evaluation, notification or associated time constraint requirements of §21.21, "Notification of failure to comply or existence of a defect and its evaluation," of 10 CFR Part 21. Instead, it focused on delineating specific requirements for Westlectric employees regarding internal processes to identify, document and transmit product deviations to a Westlectric Part 21 Committee member. For example, the "reporting" section of the procedure had three paragraphs indicating how an employee was required to fill out the Westlectric deviation documentation form.

The NRC Inspectors conducted discussions with the QA Manager to outline the salient points of the Part 21 regulation that are required to be addressed and included in a Part 21 procedure. Failure to have adequate procedures to require that evaluations or reporting of deviations, is performed as required in 10 CFR 21.21(a), constitutes a violation of NRC requirements. The QA Manager committed to revising and issuing the corrected Part 21 procedure within 120 days of the November 19, 1997, exit meeting. Violation 99901323/97-01-01 was identified in this area.

The NRC Inspectors determined that Westlectric's posting documents were in compliance with the Part 21 regulation and that it was conspicuously displayed in multiple locations at Westlectric's facility. However, it was noted that the inadequate Part 21 procedure was integral with the posting. Therefore, the QA Manager stated that he would also revise the posting document to include Westlectric's new Part 21 procedure when it was revised and issued.

Potential 10 CFR Part 21 Issue

As discussed in Section 3.4 below, a Westlectric practice was identified by the NRC Inspectors regarding the method employed for "correcting" the chemical composition analysis heat results obtained from Westlectric's spectrometer readings. As stated in §21.3 of Part 21, a deviation means a departure from the technical requirements included in a procurement document. Since this matter may represent a deviation of the , procurement documents, as of November 19, 1997, Westlectric was reviewing the circumstances to determine if its customers needed to be informed in accordance with §21.21(b) of Part 21.

c. Conclusions

The NRC Inspectors concluded that Westlectric's procedure adopted to implement the provisions of 10 CFR Part 21 was not adequately established to ensure that evaluations were performed in accordance with §21.21, "Notification of failure to comply or existence of a defect and its evaluation."

3.2 Quality Assurance Program

a. Inspection Scope

The NRC Inspectors reviewed Westlectric's QA Manual (QAM), Revision K, dated October 14, 1996, and associated procedures and records to assess the Westlectric quality program to determine whether it adequately addressed applicable Appendix B requirements.

b. Observations and Findings

Quality Assurance Manual

The NRC Inspectors determined that Appendix B requirements were contractually imposed on Westlectric in approximately 1991 by Pacific Pump (currently Ingersoll-Dresser Pump, Inc. (IDP) and Bryon Jackson Pump Division (BW/IP). It was noted that Westlectric's QA manual (QAM) did not address whether it did or how it would meet the applicable Appendix B criteria necessary to provide adequate confidence that its components supplied for safety-related use would perform satisfactorily in service. That is, the QAM did not adequately address how the applicable requirements of the 18 criteria of Appendix B were to be satisfied by the Westlectric QA program. The introduction of the QAM stated that the manual is a description of the procedures followed to maintain quality standards required to produce castings in accordance with Military Specification-Inspection (MIL-I)-45208, "Inspection System Requirements." It also stated that the manual meets the calibration system requirements of Military Specification-Calibration (MIL-C-STD)-45662A, "Calibration System Requirements." The QAM also stated that all measurements related to product conformance are traceable to the National Bureau of Standards (NBS), and that the manual establishes the quality system and procedures required by the company.

The NRC Inspectors noted that the QAM contains general instructions and procedures such as casting traceability, pattern maintenance control, corrective action, control of non-conforming castings, calibration procedures, processing test bars and test reports, shipping procedures, sand control, heat treat process, and metal control.

Accuracy of QAM Procedures

The NRC Inspectors also selectively assessed the Criterion I aspect of requiring that the authority and duties of persons and organizations performing activities affecting the safety-related functions of nuclear power plant components be clearly established and delineated in writing. After reviewing specific requirements of selected procedures, the NRC Inspectors identified several areas where it determined that the procedural steps and requirements were not clearly or accurately established or delineated in writing. For example, the NRC Inspectors found that the QAM general policy section did not accurately reflect the QA personnel and QA department structure that was in existence for several years. The QAM's general policy section states that all of the Westlectric inspectors, including non-destructive examination (NDE) level II inspectors are under the direct supervision of the "QA Assistant." However, the NRC Inspectors determined that as of November 19, 1997:

- Although Westlectric's QAM states that all Westlectric visual and nondestructive examination (NDE) inspectors are under the direct supervision of the "QA Assistant," Westlectric has not had a QA Assistant since approximately 1991.

- Until recently the visual inspector worked for and reported to production department personnel. Currently, there are two visual inspectors.

- Until recently Westlectric's QA department consisted of only the QA Manager. Currently, the Westlectric QA department consists of the QA Manager and both visual inspectors.

- Westlectric has had only one NDE certified inspector since approximately 1991, and the NDE Inspector does not report to or take direction from the QA Department.

- Although the Level II Inspector also holds the position of "Chief Inspector," the authority and responsibility of the Chief Inspector were not delineated in QAM Section 1.0, "General Policy," 3.0, "Quality Control Department Operation and Duties," 8.0, "In-Process Inspection," or 10.0, "Final Inspection."

- Although both Westlectric visual inspectors report to the QA Manager, they also take inspection activity direction from the Chief Inspector/Level II NDE Inspector. Thus, their independence from production is not assured.

The NRC Inspectors identified that the Level II NDE Inspector is also designated as the Cleaning Room Foreman (discussed in weld rod control procedure), and the Cleaning Room Supervisor (discussed in heat treat process). The NRC Inspectors determined that the employee that holds each of these positions reports to, and is under the responsibility of the Operations Manager, and does not report to or take inspection direction from the QA Manager. Discussions with the Westlectric staff and management indicated that this relationship had not been reviewed or approved by the QA organization to determine whether sufficient independence from cost and schedule when opposed to safety considerations existed.

The NRC Inspectors identified that the QAM description of the reporting relationship of the visual and NDE inspectors is inconsistent with actual practices. Nonconformance 99901323/97-01-02 has been identified in this area.

Inspection Stamps

The NRC Inspectors noted during review of PO packages that Westlectric's original certified material test reports (CMTRs) were not hand signed. It was noted that the approving official's signature was stamped instead of being signed by the approving official, the QA Manager. The QA Manager stated that he had a signature stamp for his use (mostly on CMTRs). During a discussion between the NRC Inspectors and QA Manager, the shipping supervisor brought in a stack of CMTRs to sign (stamp), and the QA manager stamped the records and returned them. The NRC Inspectors noted that the QA manager kept his signature stamp in his desk drawer. When asked whether anyone else was issued signature or inspection stamps, the NRC Inspectors were informed that the Level II NDE Inspector/Chief Inspector had also been issued a signature stamp for stamping NDE quality records. Subsequently, the NRC Inspectors requested the Chief Inspector to retrieve his signature stamp and it was noted that the Chief inspector's signature stamp was stored in a locked cabinet in the Cleaning room office. The Chief Inspector stated that the main reason that he had a signature stamp was because many of the Westlectric documents were triplicates. The NRC Inspectors obtained a blank Westlectric CMTR form and noted that the form was made of carbon-copy type paper, which would allow a signed signature to be reproduced on the second and third pages of the triplicate form.

The NRC Inspectors noted that paragraph 1.4 of Section 3.0, "Quality Control Department Operation and Duties Procedure," of Westlectric's QAM requires the QA Manager to maintain a file in his office of all inspection stamps issued to workers, and Criterion V, "Instructions, Procedures, and Drawings," of Appendix B requires activities affecting quality to be accomplished in accordance with documented procedures. The NRC Inspectors review of this area revealed that the two signature stamps for the QA Manager and Chief Inspector were not indicated as being issued even though they are used for quality record approval. Nonconformance 99901323/97-01-03 was identified in this area.

c. Conclusions

The Inspectors found that Westlectric's QAM did not accurately describe its QA department personnel and associated duties. Further, the NRC Inspectors concluded that the QA Manager's and Level II NDE Inspector's signature stamps were not identified as being issued in the QA Manager's inspection stamp file.

3.3 Welding Control

a. Inspection Scope

The NRC Inspectors conducted discussions with Westlectric staff regarding its welding program including welding processes, observed weld rod control and issuance, and reviewed certification and qualification of personnel. The NRC Inspectors noted that Westlectric's welding program control is outlined in a "Welding Control Procedure," that is part of Section 22.0, "Control of Special Processes," of Westlectric's QAM. There were no orders for nuclear applications being processed at the time of the inspections involving weld maps or other special controls. Therefore, the NRC Inspectors were not able to observe or witness implementation of the welding control process.

b. Observations and Findings

Paragraph 1.1.1 of Section 22.0 of Westlectric's QAM states in part, that the welding program will meet Section IX, "Welding and Brazing Qualifications," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The NRC Inspectors reviewed welding procedure specifications (WPS), procedures qualification reports (PQR), and welder qualifications. Westlectric has 20 different WPSs, each with a unique PQR, and currently has 11 welders that are qualified to at least one weld procedure. The NRC Inspectors determined that the welding records are well maintained and records are kept for each welder to indicate when last a specific procedure was used during each 3 month interval. This satisfies the requirements of American Society for Testing and Materials Standard (ASTM) A 488, "Standard Practice for Steel Castings, Welding, Qualification of Procedures and Personnel," which in turn, satisfies the 6 month interval required by subparagraph QW-322.1, "Expiration of Qualifications," of Section IX, ASME Code.

Paragraph 5.1.5 in the weld control procedure stated that major weld repair shall be documented by generating a weld map if required by specification or customer purchase order (PO) requirements. The NRC Inspectors observed examples of weld maps in shop traveler record package 128674, dated February 11, 1997, for Ingersoll Dresser Pumps (IDP), PO 91797, for an intermediate cover, grade CA6NM, (Heat 976A).

c. Conclusions

The NRC inspectors concluded that Westlectric maintained good records of its welder qualification and certification program and also generated and maintained satisfactory records to allow traceability of individual welders to the applicable welding specifications and procedures to maintain certification.

3.4 Spectrometer - Chemical Analysis

a. Inspection Scope

The NRC Inspectors reviewed the current and historical chemical analysis records generated from Westlectric's Thermo Jarrell Ash spectrometer. The purpose was to assess chemical analysis accuracy (by verifying calibration), ensure that the results were accurately documented on the applicable certified material test reports (CMTRs), and verify compliance to Westlectric and customer requirements.

b. Observations and Findings

Spectrometer Calibration - First Step

Westlectric performs chemical analysis with a Thermo Jarrell Ash (TJA) 181/81 optical emission spectrometer, Model 12621600, Serial Number 41883. The TJA recommended calibration consisted of two steps. The first step required creating a curve-set[1] by developing curves for each element that the machine is capable of assessing. Each curve is a plot of light intensities emitted from electrical arcing against metal standards of different chemical concentrations and the certified chemical concentrations for these standards. Each curve-set is a family of curves for the chemical elements to be analyzed. For example, carbon steel is one unique curve-set, while stainless steel is a different set of curves. The standards are normally purchased from recognized sources including the National Institute of Standards and Technology (NIST).

The NRC Inspectors requested to see the calibration records and traceability back to NIST for Westlectric curve-sets identified as: LOWB (ASTM-A-216), SER3M (ASTM-A-743, CF3M), and SERIES4M (ASTM-A-743, CA-CNM). The Westlectric staff was unable to provide curve-set calibration records for the spectrometer, and did not have curve-set traceability to NIST. The NRC Inspectors and QA Manager phoned the spectrometer service representative and were informed that the curve-sets were retrievable from the computer files along with the identity of the standards.

Spectrometer Standardization - Second Step

The second recommended Spectrometer manufacturer's step in calibration is the standardization of the curve-set. The repositioning of the curves in a particular curve-set is called standardization. Standardization of the curve-set is necessary because the spectrometer is sensitive to atmospheric effects, equipment wear, and equipment cleanliness. In order to maintain a high level of accuracy, repeatability, and reproducibility, the curve-set must be standardized each day before use, and more frequently if necessary. The NRC Inspectors determined that Westlectric used a one-point technique for standardization. The one-point technique locks a particular alloy's curve-set's chemical composition parameters to the associated NIST standard. This technique is a fast and effective technique for analyzing specimens with a chemical composition that is similar to that particular standard.

A common weakness in the one-point technique is that the further the test specimen's analysis is from the locked chemical composition value, the larger the potential error in analyses. For the curve-sets LOWB, SER3M, and SERIE4M, Westlectric used NIST Standards 1261A, 1155, and C-1289 respectively. The NRC Inspectors identified that Westlectric did not establish specific documented acceptance or rejection limits on the analysis range for each element affected by the one-point standardization for each curve-set.

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[1] One curve-set consists of multiple individual or unique curves.

Since Westlectric did not have comprehensive spectrometer operational procedures, as discussed further in this Section, the spectrometer technicians developed the concept of looking at the standard deviation for each element to determine the acceptability of the calibration. One of the technicians (who is the production melter) informed the NRC inspectors that low standard deviations were acceptable and high standard deviations were rejectable. The operator could not place a number on what would be considered the maximum acceptable standard deviation values. The NRC Inspectors consider the lack of written procedures for the acceptance of the standardization calibration and the absence of limits to the chemical range for each element as a deficiency in Westlectric's spectrometer control.

Since the NRC Inspector was not able to obtain evidence of the accuracy of the curve-sets, and since Westlectric was not able to demonstrate traceability of the curve-sets back to NIST standards, the NRC Inspectors were not able to verify that all measurements related to product conformance are traceable to NIST as required. Additionally, the NRC Inspectors were not able to verify that Westlectric's spectrometer meets the calibration system requirements that Westlectric had stated, that is MIL-C-STD-45662A requirements. The NRC Inspectors determined that Westlectric did not establish adequate measures to control its spectrometer operation and maintenance. Nonconformance 99901323/97-01-04 was identified in this area.

As a result of this identified deficiency, a departure from the technical requirements, as defined in §21.3 of 10 CFR Part 21, included in a procurement document may have occurred as discussed within Sections 3.1 and 3.4 of this report.

Spectrometer Procedures

The NRC Inspectors found that production personnel that used the spectrometer for its heat verifications used a written procedure for a standardization calibration but it was not controlled within Westlectric's QA program. The procedure was found to be untitled, was not dated, and was not approved or signed by the person(s) who developed the procedure. Additionally, the procedure for a standardization calibration did not set limits on the analysis range for each element affected by the one-point standardization. Therefore, the NRC Inspectors determined that Westlectric did not assure that the spectrometer was properly controlled, calibrated, and adjusted to maintain its accuracy. This is another example of Nonconformance 99901323/97-01-04.

Chemistry Verification

Metal Control Procedure, Section 11.0, states that heats shall be comprised of select scrap, iron ore, processed alloys, and remelt-material. It also requires that a minimum of three chemical analysis checks be made on the spectrometer for arc furnace heats prior to pouring castings. The three checks for the arc furnace heats are performed at the beginning, middle and end of the process, specifically: (a) melt-down, (b) preliminary, and (c) final. The preliminary analysis checks all chemical elements to allow calculation of the necessary alloy additions to the heat as well as assuring that a vigorous carbon boil is occurring. The final analysis verifies that all alloy additions were properly added and that the analysis meets industry and customer specifications.

The Westlectric person that assures that each heat is comprised of the correct amounts of material is the melter. The melter is also responsible for the outcome of each of the three checks by verifying on the spectrometer that each heat, which he is making and monitoring, meets the specifications.

Criterion I, "Organization," of Appendix B, state that the QA functions are those of assuring that an appropriate program is established and effectively executed, and verifying, such as by checking, auditing, and inspection, that activities affecting safety-related items have been correctly performed.

The NRC Inspectors noted that Westlectric's QAM did not specifically allow or disallow its melter from verifying his own work. It merely stated that the chemical analysis testing shall be performed by a trained laboratory technician. The Inspectors also noted that Westlectric's in process procedure (Section 8.0), and final inspection procedure (Section 11.0) did not address "inspection" of heat chemistry, even though the chemical composition is very important to the safety-related component.

Evaluating Correction of Chemical Composition Analyses Effects

Immediately after standardizing the curve-sets, Westlectric runs the same NIST standard as if it was an ordinary heat specimen. The values from this run are compared to the NIST standard's CMTR values, and the differences for each major element are noted on a daily adjustment work sheet. The operator told the NRC Inspectors that the daily adjustment work sheet is used to change the final chemical analysis of a heat that might otherwise be out of specification. The rationale is that the standardization process is not accurate because the spectrometer does not provide identical results with the values shown on the NIST standard's CMTR (neglecting any tolerances).

Therefore, Westlectric would either add or subtract the daily adjustments to the actual values from the spectrometer printer. The NRC Inspectors observed that the operator adjusted the chemical results while making out the "Daily Chemical Analysis Report."[2] After the operator performed the corrections to the daily chemical analysis report, it was given to the QA Manager. The QA Manager then takes the daily chemical analysis report that has been corrected and manually inputs the data into Westlectric's computer network for the documentation and issuance of applicable CMTRs for each heat. The NRC Inspectors observed an example where the shipping department supervisor called up a specific heat on the network and printed the CMTR in the shipping department. The NRC Inspectors were informed by the QA Manager that he is not typically involved in generating the chemical analysis and has limited knowledge regarding the operation of the spectrometer.

During a subsequent telephone conversation with the TJA spectrometer representative, the representative stated to the NRC Inspectors and Westlectric's QA Manager that there is no need to perform any correction or adjustment of the spectrometer values because all necessary adjustments are automatically performed during the standardization process. This was not known by the Westlectric personnel.

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[2] The daily chemical analysis report was merely a pre-printed form where the corrected values would be documented and handed to the QA Manager.

As a result of Westlectric's adjustments of its spectrometer results, Westlectric was documenting incorrect spectrometer values and consequently, was issuing CMTRs that were not representative of the actual chemical composition of the casting.

Section 11.0, "Metal Control," states that the chemical analysis verifies that all alloy additions were properly added and that the analysis meets customer specifications. To determine the effects from Westlectric using this manual adjustment technique, the NRC inspectors reviewed five heats that were applied on purchase orders identified for nuclear safety-related applications. The five heats were 109B, 125B, U195, U354, and 976A. Table 1 tabulates the chemistry before and after adjustment of the actual spectrometer's chemical analysis and the CMTR chemistry.

Table 1

Heat Chemical Analysis Before & After Manual Correction/Adjustment
(Values are in Weight Percent)

Heat No	C	Mn	Si	P	S	Cr	Ni	Mo	Cu	Adjusted?
976A	0.06	0.71	0.52	0.029	0.011	12.27	4.25	0.62	--	No
976A	0.03	0.67	0.52	0.028	0.011	12.08	4.23	0.63	--	Yes
125A	0.052	1.23	0.81	0.037	0.021	18.71	10.50	2.92	0.26	No
125A	0.03	1.30	0.84	0.038	0.020	18.81	10.25	2.92	–	Yes
109B	0.076	0.67	0.67	0.027	0.022	12.27	3.70	0.56	–	No
109B	0.03	0.67	0.67	0.027	0.022	12.27	3.70	0.56	—	Yes
U195	0.216	0.55	0.50	0.021	0.038	0.34	0.21	0.06	0.05	No
U195	0.21	0.68	0.50	0.020	0.031	0.34	0.21	0.06	0.05	Yes
U354	0.28	0.92	0.56	0.019	0.022	0.18	0.18	0.04	0.04	No
U354	0.28	0.90	0.57	0.023	0.023	0.18	0.18	0.04	0.04	Yes

Table 2 tabulates the heat numbers with customer purchase order information. From Table 1, the carbon in heat 109B is 0.076 before adjustment and 0.03 on the CMTR after adjustment. The 0.076 exceeded the Ingersoll-Dresser Pump Company purchase order number 074528 specification ASTM-A743 CA6NM which has a 0.06 maximum carbon. The NRC Inspectors asked Westlectric to standardize the spectrometer and run a recheck of the initial test specimen which came back as 0.05 carbon. Based on the effects that the manual adjustment can have on the final chemical analysis, the NRC Inspectors informed Westlectric that they were required to review this matter in accordance with §21.21 of 10 CFR Part 21.

Table 2

Heat Number and Customer Purchase Order Information

Heat No	Traveler	Customer	PO	PO Date	Material Spec	Part No
976A	128674	IDP	91797	1/31/97	ASTM A 743 CA6NM	M6535
125A	130255 130256 130257	BW/IP	21V550517	8/29/97	ASTM A 351 CF3M	24296 21997 23968
109B	130120	IDP	95695	7/28/97	ASTM A 743 CA6NM	M6535
U195	128220				ASTM	
U195	126220	BW/IP	W433933	11/13/96	A 216	72579

Westlectric started to review the circumstances surrounding the matter to determine if it was a deviation, and whether it needed to either evaluate the issue or inform the customer. Subsequent to the inspection, Westlectric informed the NRC staff that it had informed an applicable customer pursuant to §21.21(b) of 10 CFR Part 21, so that it could cause an evaluation to be performed, and it was still reviewing the remaining issues to determine if additional deviations existed.

The NRC Inspectors determined that the adjustment of the spectrometer's chemical analysis results without an established program to control the special process and without a program to assure that appropriately trained personnel operated the spectrometer were indicative of an inadequately controlled special process. Nonconformance 99901323/97-01-05 was identified in this area.

Westlectric QA 11.1.5.1 states that chemical analysis testing shall be performed by a trained lab technician. Westlectric said that their melt shop superintendent has been trained by TJA on the operation of the spectrometer. However, the melt shop superintendent was unavailable during the inspection. The NRC Inspectors were told that the melt shop superintendent gave on-the-job training to the individuals operating the spectrometer. The NRC Inspectors observed a melter and a recently assigned lab tech operating the spectrometer. The melter told the NRC Inspector that when problems occur he would check the argon pressure, push the spectrometer reset button, adjust the lens (mirror), clean the arc chamber, and/or restandardize the spectrometer. If none of these actions clear up the problem, he calls the melt shop superintendent. The lab tech said he calls the melter if he has problems. The NRC Inspectors noted that some specimens had multiple cracks across the testing surface and some of the burns overlapped each other. The NRC Inspectors noted that although either of these conditions can cause incorrect or erroneous readings the Westlectric personnel were not aware of the impact of these conditions. Additionally, the NRC Inspectors determined that neither the melter nor laboratory technician knew how to control movement in the mercury meter during the mirror adjustment prior to standardization. It appeared to the NRC Inspectors that the individuals operating the spectrometer received limited guidance on operating the spectrometer and specimen preparation. Except for the standardization procedure, the spectrometer is operated with no other written guidance. Nonconformance 99901323/97-01-05 was identified in this area.

Spectrometer Technicians

The NRC Inspectors conducted discussions with and observed two production personnel that normally test heat[3] samples taken from the Westlectric arc and induction furnaces during the melting/combining process. Both employees typically operate the spectrometer; therefore, the NRC Inspector's discussion encompassed the operation, calibration and principles of the spectrometer's chemical analyses. The QA Manager was involved in all of the discussions. The NRC Inspector determined that the two production personnel were regularly assigned to operate the spectrometer and verify the chemical analyses of the heats. One of the employee's title was "melter," and the other was the production department's "laboratory technician." Both employees work and report to production department management. The melter is a production department employee that is responsible for measuring and putting the correct amount of raw material (alloy additives) into the furnaces to bring the heat within the chemical analysis before the pouring of the specific castings.

The NRC Inspectors were informed by the Westlectric QA Manager that the two Westlectric personnel had received indoctrination on the operation of the spectrometer, but their responses indicated that the training was limited and was not overly comprehensive. The NRC Inspectors noted that the Westlectric personnel were not very familiar with the calibration and operating principles of the spectrometer. Although the NRC Inspectors were informed that both employees had received training for the operation of the spectrometer, records to indicate the training was satisfactorily completed were not provided during the inspection. The NRC Inspectors could not verify that both production department employees were appropriately trained for the job activity. Westlectric provided certification records for the employees subsequent to the inspection; however, those records were not specific enough to determine the appropriateness of training.

Although Appendix B requires that special processes are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements, Westlectric did not establish adequate procedures or appropriately trained personnel to control its spectrometer chemical analyses' operation. This is another example of Nonconformance 99901323/97-01-05.

c. Conclusions

The NRC Inspector was not able to verify that "all measurements related to product conformance regarding its spectrometer are traceable to NIST, and that Westlectric's spectrometer meets the calibration system requirements specified in MIL-C-STD-45662A." Westlectric was not able to produce records to indicate that it had been adequately controlling, calibrating and adjusting its spectrometer within documented

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[3] The contents of one furnace batch that is blended to a predetermined chemical composition is commonly referred to as a "heat." Therefore, each particular furnace batch will be identified with a different heat number.

parameters, and that its operators were appropriately trained. Given the extent of the concerns identified by the NRC Inspectors the adequacy of the quality oversight functions for the spectrometer operation is questionable.

3.5 Traceability of Castings

a. Inspection Scope

The NRC Inspector reviewed procedure 19.0, "Casting Traceability," to identify the Westlectric requirements that have been established and to determine whether manufacturing has been in compliance with the requirements.

b. Observations and Findings

Paragraph 1.2 of Procedure 19.0 states that heat/code numbers shall be affixed to patterns with raised aluminum letters or pressed into molding sand with letters attached to a handle. The NRC Inspectors observed three 8" x 8" valve body castings from heat U818 which showed that the production department was not in strict compliance with its procedure. The NRC Inspectors observed that the first three digits of the heat number cast into the body appeared satisfactory, but the fourth digit was ground off and replaced with a stamped letter. Westlectric stated that the heat number sequence is determined by the molding foreman because he has to insert the heat number into the mold several days before pouring.

Occasionally, process control modifications such as, smaller quantities for a specific heat, or a chemical composition analysis that was incorrect will cause a change in the anticipated production and heat numbers. Instead of making up new molds with the correct heat number, Westlectric casts the existing molds, grinds off the wrong portion of the cast heat number, and stamps the correct heat number in its place.

The NRC Inspectors were informed that another exception to the procedure is that on occasion, the cast heat number is illegible and Westlectric will grind off the illegible portion and stamp it. The NRC Inspectors confirmed that even though these practices are commonly performed, they are not delineated in the Westlectric casting traceability procedure. Although this process is not in conformance with an Appendix B QA program, the NRC Inspector notes that a nuclear safety related component would be handled somewhat different. That is, a Westlectric shop traveler would be used on nuclear orders and the shop traveler, which accompanies the component throughout the manufacturing process, requires that the heat be documented. As a result, if the heat was illegible, the traveler could provide some assurance of the correct heat.

c. Conclusions

The NRC Inspectors concluded that although these manufacturing practices may be necessary to prevent costly rework, they are not in compliance with the documented Westlectric and Appendix B requirements. This practice could be a factor for consideration if the component was destined to be "dedicated," in accordance with 10 CFR Part 21, because traceability to a heat may be indeterminate if numbers are modified from original cast numbers. This issue is an additional example of Nonconformance 99901323/97-01-03.

3.6 Nondestructive Examination

a. Inspection Scope

The NRC Inspectors reviewed Westlectric's Procedure's 1.0, "General Policy," and 22.0, "Control of Special Processes." Procedure 1.0 outlined Westlectric's QA department policies and responsibilities and Procedure 22.0 addressed the control of welding, heat treat and nondestructive examination (NDE) to assess the adequacy of Westlectric's NDE control.

b. Observations and Findings

Paragraph 1.1.1 of Procedure 22.0 stated that the QA manager is responsible for the welding program which will meet Section IX of the ASME Boiler and Pressure Vessel Code. Paragraph 2.6 of Procedure 1.0 stated that all inspectors, including magnetic particle testing (MT) and dye penetrant testing (PT) Level II who are under the direct supervision of the QA Assistant are responsible for the proper execution of required NDE. They must inspect all castings in accordance with customer requirements and standards used in the casting industry.

As discussed above, the NRC Inspectors determined that Westlectric has three inspectors total, two of whom are only visual inspectors certified to MSS-SP-55, and the third is characterized as a Level II NDE. The two visual inspectors are not trained to ASME Code. Instead, they use the visual acceptance and rejection criteria illustrated in MSS-SP-55. The NRC Inspector observed examples of rejected material that had been identified by the visual inspectors such as, coup-drag misalignment, interrupted/cold pour, porosity, and holes.

The NRC Inspectors requested Westlectric's NDE personnel certification and qualification records. The NRC Inspectors were only provided with the Chief Inspector's records, which indicated that he was certified as a Level II MT Inspector, certification dated February 21, 1997. The certification stated it was in accordance with SNT-TC-1A, "American Society for Nondestructive Testing Recommended Practice," but did not list the applicable edition. The Westlectric staff stated that it would review the matter.

The NRC Inspectors were also informed that a Level III NDE Inspector from Sun-Ray Testing International, Incorporated, Downey, California, developed Westlectric's NDE training program, performed training, maintained the personnel certifications and provided certifications to Westlectric personnel as necessary. The NRC Inspector requested to see the employer's written practices covering all phases of certification including training as required by SNT-TC-1A. At the time of the exit meeting, Westlectric had not made this information available for review. Unresolved Item 99901323/97-01-06 was identified in this area.

c. Conclusions

Westlectric was unable to provide the procedural controls and documentation associated with NDE personnel certification practices.

3.7 Entrance and Exit Meetings

In the entrance meeting on November 17, 1997, the NRC Inspectors discussed the scope of the inspection, outlined the areas to be inspected, and established interfaces with Westlectric management. In the exit meeting on November 19, 1997, the NRC Inspectors discussed their findings and concerns.

4. PERSONS CONTACTED

J.R. Heine	President
R.L. Ogden	Operations Manager
A. Arechiga	QA Manager
G. Kusumi	Sales
D. O'Sullivan	Sales Manager
M. Gutierrez	Clean Room Foreman/ Chief Inspector
S. Fericean	Shipping Supervisor
J. Lietzau	Inside Sales/Lab Technician
R. Young	Core/Molding Supervisor

5. ITEMS OPENED, CLOSED, AND DISCUSSED

99901323/97-01-01	VIO	Inadequate Part 21 Procedure
99901323/97-01-02	NON	QA Organization and Program
99901323/97-01-03	NON	Documented Instructions
99901323/97-01-04	NON	Control of M&TE and QA Records
99901323/97-01-05	NON	Control of Special Processes
99901323/97-01-06	URI	Unresolved Item - NDE Program

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

January 28, 1998

Mr. Nicholas J. Liparulo, Manager
Nuclear Safety and Regulatory Activities
Nuclear and Advanced Technology Division
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, Pennsylvania 15230

SUBJECT: NRC INSPECTION NO. 99900404/97-02

Dear Mr. Liparulo:

During the period November 17 through November 21, 1997, the U. S. Nuclear Regulatory Commission (NRC) performed an inspection of AP600 design control quality assurance (QA) activities at the Westinghouse Energy Center in Monroeville, Pennsylvania. The enclosed report presents the results of that inspection.

The purpose of the inspection was to review Westinghouse corrective actions with respect to findings identified previously in several NRC inspection reports and to determine if quality activities performed as part of the design of the AP600 Advanced Light Water Reactor were conducted under the appropriate provisions of the Westinghouse 10 CFR Part 50, Appendix B, QA program of record in the AP600 standard safety analysis report (SSAR) (Westinghouse Electric Corporation-Energy Systems Business Unit, Quality Management System, Revision 1, approved by the NRC on February 23, 1996).

During this inspection, the NRC determined that the implementation of the Westinghouse QA program for AP600 design certification activities failed to meet certain NRC requirements. Specifically, the team identified examples of inadequate program implementation with respect to control of design calculations which contained discrepancies or errors without documentation of an adequate evaluation by Westinghouse. The team also identified examples of errors that were allowed to propagate in design calculations without correction, inadequate documentation of design and analysis conclusions, and errors in both WGOTHIC and WCOBRA/TRAC computer codes. Of particular concern to the NRC was Westinghouse's failure to adequately evaluate the impact of GOTHIC computer code errors reported by Numerical Applications, Inc., in accordance with Westinghouse QA requirements and 10 CFR Part 21, as applicable. Accordingly, Westinghouse needs to evaluate the impact of these findings on the AP600 SSAR Chapters 6 and 15 analyses, and all other SSAR AP600 design information, based on affected computer codes and associated calculation notes.

In light of the number of discrepancies identified by the NRC inspection team in such a small sample of the total population of documents reviewed, Westinghouse needs to establish, via a comprehensive evaluation and/or assessment, the adequacy of the AP600 QA design review process and the integrity of the AP600 design, particularly containment design, and demonstrate that the requirements of 10 CFR 50, Appendix B, 10 CFR Part 21, and the applicable design certification provisions of 10 CFR Part 52 are being satisfied. The NRC requests that Westinghouse provide a schedule outlining the basis and completion of such assessment within 30 days of receipt of this letter.

Detailed technical issues arising from this inspection, which require prompt action by Westinghouse due to the staff's schedule to complete safety evaluations for the AP600 design certification program, were documented in a Request for Additional Information (RAI) and sent to you on December 17, 1997. The substance of these RAIs has been incorporated into this report and identified as either a Notice of Nonconformance or an Unresolved item (URI). The NRC requests that Westinghouse provide the information requested in the URI within 30 days of receipt of this letter.

The responses requested by this letter and the enclosed Notice of Nonconformance 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 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,

Suzanne C Black

Suzanne C. Black, Chief
Quality Assurance, Vendor Inspection, and
Maintenance Branch
Division of Reactor Controls and Human Factors
Office of Nuclear Reactor Regulation

Docket No.: 52-003

Enclosures:
1. Notice of Nonconformance
2. Inspection Report No. 99900404/97-02

cc w/encls: See Next Page

Mr. Nicholas J. Liparulo
Westinghouse Electric Corporation

Docket No. 52-003
AP600

cc:

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NOTICE OF NONCONFORMANCE

Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

Docket Nos.: 52-003
and 99900404

Based on the results of a Nuclear Regulatory Commission (NRC) inspection conducted during the week of November 17-21, 1997, of activities supporting Westinghouse Electric Corporation's AP600 design certification, it appears that certain activities were not conducted in accordance with NRC requirements.

A. Criterion III, "Design Control," of Appendix B to 10 CFR Part 50, states, in part, "The design control measures shall provide for verifying or checking the adequacy of the design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program."

ANSI/ASME NQA-1-1989 "Quality Assurance Program Requirements for Nuclear Facilities," Supplement 3S-1, "Supplementary Requirements for Design Control," Section 4.2.1, "Design Reviews," states that design reviews "are critical reviews to provide assurance that the final design is correct and satisfactory. Section, 4.2.1, also states, in part, that where applicable, the following shall be addressed during such reviews:

- Were the design inputs correctly selected?
- Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent reverifications when the detailed design activities are completed?
- Were the design inputs correctly incorporated into the design?
- Is the design output reasonable compared to design inputs?

ANSI/ASME NQA-1-1989, Supplement 3S-1, Section 3.1, "Design Analyses," states, in part, that design analyses documents "shall be sufficiently detailed as to purpose, method, assumptions, design input, references, and units such that a person technically qualified in the subject can review and understand the analyses and verify the adequacy of the results without recourse to the originator."

WCAP-8370, Westinghouse Electric Corporation - Energy Systems Business Unit/Power Generation Business Unit - Quality Assurance Plan," Revision 2 (April 1992), Section 4.4.1, "Design Reviews," states, in part, that independent reviews address the following, as applicable, "design input selection, described and reasonable design output compared to design input, design input and verification requirements from interfacing organizations, appropriate design method used, design inputs correctly incorporated into the design, and adequately described, reasonable, and identified assumptions."

Westinghouse Electric Corporation - Energy Systems Business Unit Policy/Procedure WP-4.17, "Design Verification by Independent Review or Alternate Calculations," Revision 0 (8/31/96), states, in Section 8.3, that the assigned *verifier* "**Verifies the adequacy of the design** [emphasis added] or changes thereto by independent review or alternate calculations."

The following examples demonstrate failure to comply with the above requirements and constitute Nonconformance 99900404/97-02-01.

Contrary to the above, the inspection team found that for the following AP600 calculations Westinghouse failed to perform an adequate design review to establish the acceptability of the corresponding AP600 design bases analyses:

1. SSAR-GSC-189, "AP600 SSAR Inadvertent ECCS Analysis," Revision 2, assumed the need of operator actions in the analyses of increased RCS inventory events, but failed to address:

   - The availability of (1) unambiguous alarms or indications for increased RCS inventory events, and (2) clear procedural instructions to operators to take appropriate actions within the time-frame assumed in the analyses.

   - 10 CFR 50.36, "Technical Specifications," §50.36(c)(2)(ii)(C).

   - Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) to verify the capacity of the system including the RV head vent valves that would be used by operators to prevent pressurizer overfill from occurring as assumed in the analyses.

   - Standard Safety Analysis Report (SSAR) text providing detail of the analyses that credited the requisite operator actions.

2. SSAR-GSC-188, "AP600 Boron Dilution Analysis," Revision 0, relied on the boron mixing testing data documented in EGG-LOFT-5867 (Project No. P 394) to establish the required RCS circulation flow rate of 1000 gpm in TS 3.4.9 and to support its complete boron mixing model assumed in the boron dilution analyses (SSAR 15.4.6). However, despite significant differences between the AP600 design and the test facility configuration and testing conditions discussed in EGG-LOFT-5867, SSAR-GCS-188 failed to reconcile the applicability of the boron mixing testing data to the AP600 design or to validate the complete boron mixing model assumed in the boron dilution analysis.

3. SEC-APS-4838-CO, "Software Design Specifications of AP600 NOTRUMP User Externals Cycle 2," Revision 0, dated September 9, 1995, contained a statement that numerous errors in code parameters were reviewed as insignificant and would be corrected in a later code version however, no basis was given to support this conclusion.

4. SEC-APS-4837-CO, "Software Change Specification of NOTRUMP Cycle 32," Revision 0, dated September 9, 1995, contained a statement that "...the author doesn't know enough about the subject (void propagation) to determine the impact of the reviewer's comments." No evidence existed to support resolution of the reviewer's comment.

5. SEC-APS-4746-C0, "WCOBRA/TRAC Long-Term Cooling," provided no basis for concluding that "...variations in the initial conditions are expected to have relatively unimportant effects on the analysis results," and "the results of changing ICHP is noticeable but not large..."

6. LTCT-T2C-417, "WCOBRA/TRAC Geometrical Input Data for the OSU Testing," Revision 0, and LTCT-T2C-418, "OSU LTC Comparisons with WCOBRA/TRAC," Revision 1.
   - LTCT-T2C-417. Pages 180-181 (Figures 6 and 7) acknowledged the failure to fit DP vs $(\text{flow})^2$, however, the basis provided was that "...despite the failure to match, overall agreement is reasonable." This unquantified anomaly was used as input to calculation LTCT-T2C-418.

   - LTCT-T2C-418. On page 16, a bias of 0.2 psia was applied to the atmospheric pressure to compensate for the disparity in DP vs $(\text{flow})^2$ in calculation LTCT-T2C-417. However, the calculation did not provide an explanation for the use of this bias.

7. SSAR-GSC-356, "Two-Inch Break LOCA, LTC," Revision 0, presented a solution of DP vs. flow in which the author observed that a harmonic oscillation was built-in to the solution, and therefore, he proposed to take the average value. However, the inspection team could not determine if the average value was equal to the asymptotic solution had the oscillation not been present. The calculation also did not address the impact of oscillation in the asymptotic solution, the impact of the oscillation on the flow resistance, and the presence of the oscillation in the vessel flow, DP, and vessel collapsed liquid level solutions.

8. SSAR-GSC-377, "SBLOCA Long-Term Cooling," identified discrepancies which included a calculation for negative (reverse) DVI flow with no corresponding physical explanation provided, a two-sided open break which did not agree with a two-inch pipe break assumed in the calculation, and discrepancies related to initial conditions assumed for leakage through ADS 1-3.

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

   Westinghouse Procedure WP-4.19.3, "Software Error Reporting and Resolution, "requires that the impact of errors be reviewed on all work activities where the program was used during the time period the error existed and documented on an Error Impact Review sheet within 60 days of receiving the error report.

   The following examples demonstrate failure to comply with the above requirements and constitute Nonconformance 99900404/97-02-02.

   Contrary to the above, Westinghouse did not provide documentation to support the review and evaluation of computer code errors for the following examples:

   1. WCOBRA/TRAC code error report for MOD 7A, Revision 1, listed an error affecting timestep control which was not evaluated for the specific case of the AP600 design. In addition, code failures identified in AP600 calculations were not reported and tracked in Westinghouse's error tracking system.

2. Over 100 code errors associated with GOTHIC (after Version 4.0) were identified to Westinghouse by the developer, Numerical Applications, Inc. (NAI). NAI stated to Westinghouse that some of the errors could affect safety determinations and may be reportable under 10 CFR Part 21. Westinghouse could not provide documentation to support the review and disposition of these code errors.

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, Quality Assurance, Vendor Inspection, and Maintenance Branch, Division of Reactor Controls and Human Factors, 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 28th day of January, 1998

U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Report No.:	99900404/97-02
Organization:	Westinghouse Electric Corporation Nuclear and Advanced Technology Division Pittsburgh, Pennsylvania 15230
Contact:	Mr. Nicholas J. Liparulo, Manager Nuclear Safety and Regulatory Activities
Nuclear Industry Activity:	Nuclear steam supply system design, components and services
Date:	November 17-21, 1997
Inspectors:	Robert Pettis, Jr.,HQMB/DRCH Juan Peralta, HQMB/DRCH Edward Throm, SCSB/DSSA Ralph Landry, SRXB/DSSA Summer Sun, SRXB/DSSA Joseph Staudenmeier, SRXB/DSSA Lambros Lois, SRXB/DSSA Alan Levin, SRXB/DSSA William Huffman, PDST/DRPM
Approved:	Robert A. Gramm, Chief Quality Assurance and Safety Assessment Section Quality Assurance, Vendor Inspection, and Maintenance Branch Division of Reactor Controls and Human Factors

1 INSPECTION SUMMARY

The purpose of the inspection was to determine if quality activities performed as part of the design of the AP600 Advanced Light Water Reactor (ALWR) were conducted under the appropriate provisions of the Westinghouse 10 CFR Part 50, Appendix B, quality assurance program of record in the AP600 SSAR.

The inspection bases were as follows:

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of the Code of Federal Regulations (10 CFR Part 50).

- ANSI/ASME NQA-1-1989, "Quality Assurance Program Requiremenst for Nuclear Facilities," Edition through NQA-1b-1991 Addenda.

- WCAP-8370, Westinghouse Electric Corporation - Energy Systems Business Unit/Power Generation Business Unit - Quality Assurance Plan," Revision 2 (April 1992).

- AP600 SSAR, Revision 11, Section 17.3, "Quality Assurance."

- WCAP-12600, Revision 2, dated December 1993, "AP600 Quality Assurance Program Plan."

During the inspection, the NRC inspection team identified the following instances where Westinghouse failed to conform to NRC requirements:

1.1 Nonconformances

- Nonconformance 99900404/97-02-01 was identified and is discussed in Section 3.2, 3.3 and 3.5 of this report.

- Nonconformance 99900404/97-02-02 was identified and is discussed in Section 3.4 and 3.6 of this report.

1.2 Unresolved Item

- Unresolved Item 99900404/97-02-03 was identified and is discussed in Section 3.4 and 3.6 of this report.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 Nonconformance 99900404/95-01-01 (closed)

Contrary to the provisions of Engineering Procedure AP-3.11, "AP600 Testing," of Westinghouse Electric Corporation AP600 Program Operating Procedures Manual (WCAP-12601), Revision 1, dated April 1, 1995, Section 5, "Instrumentation Requirements," of

Test Specification: Large Scale Passive Containment Cooling Test, AP600 Document Number PCS-T1P-002 (WCAP-13267), Revision 1, dated December 1991, did not accurately reflect the instrumentation that was procured and installed in the PCCS Large Scale Test Facility.

The team reviewed Section 5 of WCAP-13267 and determined that the instrumentation listed was not an accurate reflection of what was procured and installed in the test facility. The test specification had not been revised since December 1991. Westinghouse revised PCS-T1P-002 to reflect the final configuration of the Large Scale Test Facility. As preventive actions, Westinghouse revised test specifications for subsequent tests (including the Core Make-Up, Long Term Cooling, and Full-Height/Full Power tests) to reflect the final configuration of the respective facilities. The team verified that these actions had been completed and documented.

2.2 Nonconformance 99900404/95-01-02 (closed)

Contrary to procedure DP-5.0, Revision 3, "Instructions, Procedures and Drawings," of Westinghouse Electric Corporation Nuclear and Advanced Technology Division (NATD) Quality Assurance Program Manual (WCAP-9565), Revision 34, no procedure or instruction was available or utilized for determining the as-built elevations and dimensions of the PCCS Large-Scale Tests. The critical dimensions of the Large-Scale Test Facility were to be identified by the Containment and Radiological Analysis group. No records of identification of the critical dimensions by this group were available.

Although critical dimensions for the CMT test were verified using AP600 Document No. MT01-T1P-003, "AP600 CMT Test Facility Piping Dimensional Characterization Procedure," Revision 0, dated April 20, 1993, no such procedure existed for the dimensional characterization of the Large-Scale Test facility. The Large-Scale Test critical dimensions were taken prior to April 20, 1993. Therefore, there was no record of how the dimensions were taken or if the Containment and Radiological Analysis group had approved the critical dimensions.

Specified critical dimensions were re-measured by Westinghouse in accordance with written procedures. As preventive actions, AP600 Procedure AP-3.11, "AP600 Testing," was revised to require that a written procedure be used to obtain the specified critical dimensions of safety-related AP600 test facilities in use subsequent to the PCCS Large Scale Tests. The team verified that these actions had been completed and documented.

2.3 Nonconformance 99900404/95-01-03 (closed)

Contrary to Program Operating Procedure AP-7.1, "Supplier Evaluation, Audit and Approval," Revision 2, of WCAP-12601 and Procedure DP-7.0, "Control of Purchased Items and Services," Revision 7, of WCAP-9565, the audit conducted by Westinghouse at Alden Research Laboratory, Inc. (Alden Research) on March 4, 1992, and later confirmed as acceptable for AP600 design certification activities by the Energy System Business Unit (ESBU) Projects Quality Assurance organization on October 20, 1994 (Report/File No. PQA-94-0032, dated October 14, 1994) did not provide adequate objective evidence that Alden Research was a supplier of calibration services as a Basic Component (as defined and used in Part 21 to Title 10 of the Code of Federal Regulations (CFR)) nor did it demonstrate the acceptability of Alden Research's technical and quality program capabilities with respect to 10 CFR 21 requirements.

Westinghouse performed an audit of Alden Research on May 25, 1995. The audit found that while implementation of Alden's quality assurance program needed improvement in several areas, no findings were identified that would impact the calibration activities performed to support the AP600 testing program. Based on internal and customer audits, Westinghouse considered this issue to be an isolated occurrence for which continued implementation of the ESBU QA program is relied upon to prevent recurrence. Findings from the audit of Alden were followed to resolution through the existing ESBU supplier audit process. The team verified that these actions had been completed and documented. Based on the limited scope of calibration services as a basic component performed by Alden Research, and based on Westinghouse's conclusions that no audit findings identified impacted the calibration activities performed to support AP600 testing, the team agreed that corrective actions taken by Westinghouse were appropriate.

2.4 Nonconformance 99900404/95-02-01 (closed)

Contrary to Section 9.0, "Quality Assurance Requirements" of WCAP-14112, "Automatic Depressurization System Test Specification (Phase B1)," Revision 2, and Section 7.0, "As-Built Records" of ENEA document AP600-GQ9402, "Quality Assurance Plan Description: AP600 Test Program Conducted at the VAPORE Plant in ENEA Cassacia (Phase B)," Revision 2, as-built drawings, pertaining to the ADS Phase B tests that characterize the features which influence thermal-hydraulic and structural parameters [illegible] validation and calculation methodology verification efforts, had not been generated for AP600 ADS Phase B testing at VAPORE.

Modifications to the VAPORE test facility, necessary to support AP600 ADS design certification testing, were performed by ANSALDO S.p.A. under contract to Westinghouse. On November 29, 1994, Westinghouse placed a contract with ANSALDO to provide as-built documentation of the ADS test loop at the ENEA's VAPORE test facility. Westinghouse stipulated that ANSALDO provide one full set of as-built drawings (comprising P&ID, line list of principal flow paths, valve list, ADS loop layout drawings, ADS loop isometric drawings, ADS loop platform, and ADS loop support drawings) covering both ADS Phases B1 and B2 configurations. Westinghouse intended to include these drawings as part of the as-built records package for AP600 VAPORE Phase B testing.

During the inspection, however, the team found that as-built drawings, as defined and stipulated in WCAP-14112, and in AP600-GQ9402, had not been generated for AP600 ADS Phase B testing at VAPORE.

Westinghouse had identified this issue during a June 6-9, 1995 audit of ENEA. However, since ANSALDO was responsible for generating the as-built documentation for the facility, Westinghouse initiated additional actions to resolve this issue accordingly. A Westinghouse review of all documentation at ANSALDO offices in Genoa, Italy, on July 19 and 20, 1995, revealed that ANSALDO had used a combination of shop drawings and field measurements to create their as-built drawings. Based on an assessment of the elements used to define the as-built configuration of the ADS test facility as well as the supporting documentation on the procurement and fabrication of the piping sections, Westinghouse concluded that the as-built documentation was in compliance with AP600 project requirements and that the requirements of AP600-GQ9402, Revision 2, had been satisfied. Westinghouse added that the process of establishing the final test configuration dimensional characteristics had been performed in a controlled manner by ANSALDO. Westinghouse completed this assessment in August 2, 1995.

The team reviewed a summary of Westinghouse's assessment activities at ANSALDO and concluded that appropriate actions had been taken to resolve this nonconformance.

2.5 Unresolved Item 99900404/95-02-02 (closed)

During the inspection, the team reviewed the VAPORE test facility calibration records which provided evidence of traceability to the appropriate ENEA controlled SIT-certified standards. This review also provided evidence of the adequacy of the facility instrumentation calibration status during each testing phase. The team found, however, that the ENEA QA program did not include adequate measures to effectively control the calibration status of reference instruments or standards used for instrument calibration, as no provisions were in place to require re-calibration by SIT at the requisite intervals. This may have resulted in the introduction of uncertainties in the adequacy of calibration of test facility instrumentation which relied on these standards to establish and maintain their accuracy.

Pending confirmation by Westinghouse that this lapse in the SIT-certified calibration interval for the ENEA standards did not undermine or adversely impact the VAPORE ADS test results, this issue remained unresolved.

In order to achieve resolution of this item, ENEA submitted the seven instruments involved in AP600 test instrument calibration to a nationally certified calibration laboratory in Italy (ERG/ING/PITER Division). Test results confirmed that VAPORE ADS test results had not been adversely impacted. During the inspection, the team verified that these actions had been completed and documented accordingly.

2.6 Nonconformance 99900404/97-01-01 (closed)

Contrary to Criterion XVI, "Corrective Action," of Appendix B to 10 CFR Part 50, WCAP-12600, "AP600 Quality Assurance Program Plan," Revision 2, dated December 15, 1993, Section 16, "Corrective Action, " and WCAP-8370, Quality Assurance Plan (QA Topical Report), Revision 12A, dated April 1992, Section 16, "Corrective Action," Westinghouse (1) did not identify, analyze, document, and correct conditions adverse to quality as required by the AP600 Quality Assurance program when such conditions were identified to Westinghouse during a July 1994 NRC structural audit of the AP600 nuclear island foundation mat, and (2) did not adequately determine and document the root cause of INITEC's basemat calculation errors nor evaluate the impact of such a condition adverse to quality on completed or related INITEC AP600 design deliverables and activities.

On July 11 through 14, 1994, the NRC performed an audit of the structural design of the AP600 at the Bechtel offices in San Francisco, California. The results of this audit were documented in a letter to Westinghouse dated August 24, 1994, "Summary of Audit of the AP600 Structural Design."

One of the key issues identified during the audit were errors found by the audit team in design calculations performed by INITEC (Calculation No. 1010-CCC-001, Rev. A). The NRC audit team identified the following deficiencies: (1) errors in shear and flexural reinforcement assessment, in the use of punching shear formula, and in the use of finite element dimension; (2) no consideration of accident pressure loads, and loads from construction sequence, and (3) out of phase overturning moment from shield and containment buildings.

In its August 2, 1994, letter to the NRC, Westinghouse acknowledged its commitment to: (1) perform an independent review of the basemat design calculations, (2) verify the adequacy of INITEC's in-house post-process computer programs used for the foundation mat design, (3) perform simplified analyses as appropriate to confirm the existing design results, and (4) provide the results of this independent review to the NRC. (This issue was identified by the NRC in the AP600 DSER as DSER Open Item 3.8.5-21).

On April 17, 1997, the inspectors reviewed an INITEC letter to Westinghouse (INI/FOK0175), dated February 15, 1995, in which INITEC provided its response to address the root causes of the quality issue relative to the Nuclear Island Basemat Calculation, identified by NRC in its July 1994 structural design audit, as well as the measures taken by INITEC in order to avoid the occurrence of similar situations during the performance of present and future structural analysis. The inspectors noted that INITEC's response had been formulated almost contemporaneously with the triennial audit being conducted by Westinghouse on February 20 through 22, 1995, at INITEC's facilities. Yet Westinghouse's triennial audit report (QLA/INI0007) did not provide any evidence that INITEC had identified this issue as a condition adverse to quality requiring root cause determination or corrective actions in accordance with INITEC's Westinghouse-approved quality assurance program. During the inspection, the inspectors inquired as to whether Westinghouse had initiated any root cause determinations or corrective actions, as required by WCAP-12600, Section 16, "Corrective Action," since this issue was first identified by the NRC in July/August 1994 or whether Westinghouse had formally accepted or rejected INITEC's proposed corrective actions identified in INITEC's February 1995 letter. Westinghouse responded that, as of April 17, 1997, this issue had not been identified as a condition adverse to quality requiring a root cause determination or corrective actions in accordance with WCAP-12600 nor had Westinghouse formally responded to INITEC's February 1995 letter.

In its June 9, 1997, response to NRC inspection Report 99900404/97-01, Westinghouse provided information related to corrective actions initiated at Westinghouse and INITEC associated with the basemat calculation. Areas noted included an identification of the error, cause, determination of the extent of the error, corrective action, action to prevent recurrences and Westinghouse oversight. Two of these areas, cause and determination of the extent of the error, were identified for further evaluation in a Westinghouse audit of INITEC in May 1997. Additional calculations were checked in the Westinghouse overview. This check confirmed that the units error was an isolated occurrence. The May 1997 audit confirmed that INITEC followed through on their corrective action which consisted of revising the calculation and strengthening their verification efforts by adding experienced personnel to their staff as well as providing additional training. The Westinghouse audit team concluded that INITEC had implemented an improved independent verification program.

In order to provide further assurance that conditions adverse to quality are being properly controlled and corrected on the AP600 program, Westinghouse initiated the following corrective actions.

1. Design deficiencies (errors) will be subject to the criteria for initiation of a corrective action document requiring quality assurance participation in the corrective action process.

2. The criteria for initiating a corrective action document requiring quality assurance participation in the corrective action process will be clarified and expanded to address the following conditions:

a. External audit issue - a finding or observation identified by an external organization during an audit or inspection, that requires a response.

b. External technical issue - a finding or observation identified by an external organization during a technical audit or review, that requires a response, and which, upon further Westinghouse review, indicated an error or deficiency in the design.

c. Internal assessment finding - a finding identified by an internal assessment that requires a response.

d. Licensing basis document inconsistency - an error or deficiency that impacts the SSAR, PRA or CDM which, upon further Westinghouse review, indicated an error or deficiency in the design.

e. Deviations related to quality procedures - recurrence of deviations in relation to approved quality procedures and/or lingering absence of appropriate procedures or work practices which should be performed in accordance with applicable quality standards.

f. Condition adverse to quality for which it is desired to determine the cause and identify action to preclude recurrence.

3. A specific AP600 Program Operating Procedure on corrective action will be generated.

4. The new corrective action procedure will be extended to all program participants.

5. Training in the corrective action procedure will be provided to program participants.

The inspection team verified that a new Westinghouse procedure, AP-16.2, "Corrective Action for Design Deficiencies or Errors, " Revision 0, (effective on August 18, 1997) had been implemented by Westinghouse to address items 1 through 3, above. For items 4 and 5, the team verified that these actions had been completed and documented accordingly.

2.7 Nonconformance 99900404/97-01-02 (closed)

Contrary to Criterion VII, "Control of Purchased Material, Equipment, and Services," of Appendix B to 10 CFR Part 50, WCAP-12600, "AP600 Quality Assurance Program Plan," Revision 2, dated December 15, 1993, Section 7, "Control of Purchased Items and Services," and WCAP-8370, Quality Assurance Plan (QA Topical Report), Revision 12A, dated April 1992, Section 7, "Control of Purchased Items and Services," Westinghouse (1) failed to adequately evaluate or assess INITEC's annual performance for a supplier of AP600 design deliverables that had been the subject of an adverse NRC audit finding, and (2) failed to conduct an evaluation of INITEC's response to Westinghouse's August 3, 1994, letter, and any associated corrective actions taken, in its February 1995 triennial audit of INITEC.

On April 17, 1997, the inspectors learned that on August 3, 1994, Westinghouse had sent a letter to INITEC (FOR/INI0181) requesting that INITEC provide its response to the NRC inspection findings. However, it appeared that Westinghouse did not consider the results of the NRC findings and concerns in its January 1995 annual review of INITEC's performance.

Additionally, the inspectors reviewed Westinghouse's Audit Report QLA/INI0007, dated March 20, 1995, that documented a triennial audit conducted on February 20 through 22, 1995, at INITEC's facilities in Madrid, Spain. Upon reviewing the report, the inspectors found that: (1) an evaluation of INITEC's response to Westinghouse's August 3, 1994, letter had not been included in the audit scope, and (2) the audit did not identify any evidence to suggest that INITEC had initiated any internal root cause analysis, and evaluation. Further, the inspectors determined that no corrective actions had been formally identified by INITEC's QA organization to determine the cause, and document the impact of the design deficiencies identified in Westinghouse's August 3, 1994, letter, on INITEC's AP600 design deliverables.

On these bases, the inspectors concluded that Westinghouse (1) failed to adequately evaluate or assess INITEC's performance, as required by WCAP-12600, for a supplier of AP600 design deliverables that had been the subject of an adverse NRC audit finding, and (2) failed to conduct an evaluation of INITEC's response to Westinghouse's August 3, 1994, letter, and any associated corrective actions taken thereof.

In its June 9, 1997, response to inspection report 99900404/97-01, Westinghouse stated that the annual performance of INITEC was reviewed by Westinghouse Quality Assurance and the responsible Westinghouse engineering manager as part of the reviews of the performance of all active AP600 design organizations during the week of January 2, 1995. During these reviews, Quality Assurance identifies suppliers who are due for re-audit based on the triennial audit requirement, and solicits input from Engineering about the status of all active AP600 suppliers, their quality and scope of work, and any other factors that would justify more frequent audits. At that time, it was determined that an audit of INITEC should be performed in 1995. INITEC was due in 1995 for a supplier audit, having been previously audited in March 1992. The audit (WES-95-211) was performed in February, 1995.

In preparation for the February 1995 audit, the AP600 manager of Plant Engineering noted that NRC had technical questions about the nuclear island basemat analysis. Since concerns regarding this analysis had been identified, the Plant Engineering group identified an action plan to address these concerns, including performing an independent and expert technical evaluation of the calculation. Based on review of the issues involved and the understanding that this issue was already being addressed in the technical arena, it was determined that the review of the analysis would assess whether there were any deficiencies inherent in INITEC's design control measures that contributed to the concerns identified. During audit WES-95-211, no deficiencies within INITEC's design control program or in meeting the requirements of NQA-1 were identified. At the time of the audit, Westinghouse Quality Assurance had not received INITEC letter INI/FOK0175 documenting INITEC's response to Westinghouse's August 3, 1994 letter, nor did it surface during the audit.

A follow-up Westinghouse audit of INITEC was conducted in May, 1997. This audit evaluated the corrective action outlined in INITEC letter INI/FOK0175. The audit team consisted of a technical specialist, two members of the Energy Systems Business Unit Quality Systems organization, and an engineer from the AP600 project group. The audit included a comprehensive review of basemat calculations. The audit team noted that INITEC had implemented an improved verification program with the addition to their staff of a qualified technical expert devoted solely to independent verification. Based on the audit team's review of the conditions surrounding the error, it was found to be an isolated occurrence. INITEC was requested to document their own evaluation of the extent to which the error may exist in other calculations. Westinghouse concurred, based on INITEC's documentation, that this error was an isolated occurrence. In order to further assure the adequacy of supplier performance evaluations and improve the integration of the performance evaluation results with the audit planning process, Westinghouse took the following actions for the AP600 program:

1. A comprehensive supplier performance evaluation checklist was developed to enhance implementation of the procedure requirements for performance evaluations. The checklist includes the following:

   - Results of open items from prior Westinghouse audits
   - Results of technical oversight of the supplier including identification of design errors
   - Results of audits of the supplier from other sources (if available)
   - Status of other committed corrective actions

2. The performance of all active suppliers was reassessed for 1997 using the new performance evaluation checklist.

3. Training on the subject of performance evaluation and audit planning was provided to AP600 project personnel.

The inspection team verified that these actions had been completed and documented accordingly.

2.8 Unresolved Item 99900404/97-01-03 (closed)

Based on the nonconformances identified in NRC Inspection Report 99900404/97-01, the NRC was concerned that those quality assurance deficiencies may have introduced a level of uncertainty on the acceptability of design deliverables provided by AP600 technical cooperation agreement participants. Of particular concern to the NRC, was Westinghouse's failure to recognize and appropriately address a condition adverse to quality, requiring a root cause evaluation and determination and appropriate corrective actions, even when such condition was identified by an NRC audit and resulted in re-design of the AP600 foundation basemat.

Westinghouse's failure to address this design and quality assurance program deficiency in a timely manner raised the issue of whether this was an isolated case and whether other design deliverables provided by AP600 technical cooperation agreement participants did in fact possess the level of integrity in design verification and quality assurance necessary to satisfy the design certification provisions of 10 CFR Part 52.

Accordingly, the NRC requested that Westinghouse: (1) determine and evaluate the impact of these nonconformances on completed or related design deliverables and/or activities performed by all AP600 technical cooperation agreement participants; (2) identify the steps that it has taken, or intends to take, to demonstrate that other design deliverables provided by AP600 technical cooperation agreement participants do in fact achieve the level of integrity in design verification and quality assurance necessary to satisfy the design certification provisions of 10 CFR Part 52; and (3) provide a list of all AP600 technical cooperation agreement participants, including a description of their AP600 program work scope and involvement. To address the NRC's concerns, Westinghouse undertook the following activities:

1. Based on recent INITEC performance findings and the findings of the quality assurance audit, Westinghouse performed a detailed management review of the INITEC activities on June 3, 1997. The purpose of the review was to evaluate whether the actions identified for INITEC were appropriate for the conditions. The reviewing group included the team that audited INITEC in May 1997, Westinghouse QA management, Westinghouse engineering management, and the Westinghouse lead reviewer for INITEC structural deliverables. There were two areas reviewed where improvements were needed: improving the accuracy of their verification process and initiating and following up on corrective actions. The verification process has been significantly augmented with additions to their staff and additional training. Improvements in the corrective action portion of the INITEC program will be defined in response to the May 1997 Westinghouse audit findings and will be monitored by a Westinghouse QA engineer. The review concluded that the actions completed, plus those identified for INITEC from the audit, were sufficient actions at INITEC, and some additional activities would be initiated at Westinghouse.

2. Westinghouse commissioned an independent audit of the Westinghouse Quality Program and quality assurance oversight of technical cooperation participants as applied to AP600 design activities. This audit, completed on May 30, 1997, was performed by individuals completely independent of the AP600 quality assurance function with the lead auditor being an outside contractor . The audit identified three findings and four recommendations for program improvement which are currently being addressed. The overall assessment of the audit was that the oversight of design activities on the AP600 was effective both from a quality assurance standpoint and from a technical standpoint and in accordance with WCAP-8370, 10CFR50 Appendix B and NQA-I.

3. A comprehensive evaluation was conducted of the methods and degree of oversight provided for all technical cooperation agreement participants performing engineering tasks at their facilities.

4. A design assurance review was initiated to assess a sample of safety related design work which is part of the AP600 licensing basis from each of the international design participants. The purpose of this review is to demonstrate, with reasonable assurance, that the design deliverables provided, achieve the required level of integrity in design verification and quality assurance. As part of the assessment, technical specialists performed an independent check for accuracy of deliverables supplied. In addition, an assessment was made of the adequacy and integrity of the original design and the design verification provided by the document originator. Conclusions reached by Westinghouse as a result of this design assurance review were documented in a summary report, "Response to NRC Inspection Report No. 99900404/97-01 of AP600," dated October 16, 1997.

The inspection team verified that these actions had been completed and documented accordingly.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 AP600 Quality Assurance Program

Chapter 17 of the AP600 standard safety analysis report (SSAR), Revision 18, describes the Westinghouse Electric Corporation quality assurance (QA) program for the design phase of the AP600 Advanced Light Water Reactor (ALWR) Plant Program. In Revision 5 to SSAR Chapter 17, Westinghouse stated that effective March 31, 1996, activities affecting the quality of items and services for the AP600 Project during design, procurement, fabrication, inspection, and/or testing would be performed in accordance with the quality plan described in Westinghouse's "Energy Systems Business unit - Quality Management System," (QMS) Revision 1.

The staff's review and approval of Revision 1 to the Westinghouse QMS was documented in a letter from Suzanne Black (NRC) to N. J. Liparulo (Westinghouse), dated February 23, 1996. Activities performed prior to March 31, 1996, were performed in accordance with the quality assurance plan described in Westinghouse topical report WCAP-8370, "Energy System Business Unit - Power Generation Business Unit, Quality Assurance Plan," Revision 12a, dated April 1992. Also, activities performed prior to November 30, 1992, were performed in accordance with the quality assurance plan described in topical report WCAP-8370/7800, Energy Systems Business Unit - Nuclear Fuel Business Unit, Quality Assurance Plan, Revision 11A/7A. Both versions of WCAP-8370 applied to all Westinghouse activities affecting quality of items and services supplied to nuclear power plants and established Westinghouse's compliance with the provisions of Appendix B to 10 CFR Part 50.

WCAP-12600, "AP600 Quality Assurance Program Plan," dated January 1997, a project-specific QA plan, was developed by Westinghouse to enhance the QMS in specific areas and to establish additional commitments needed to support the AP600 Design Certification and First-Of-A-Kind (FOAKE) program. WCAP-12600 establishes the responsibility of the Nuclear Projects Division of the Energy Systems Business Unit for AP600 Design Certification and FOAKE programs and for control of the technical interface between Westinghouse and engineering groups and suppliers providing engineering services under such programs. WCAP-12600 also addresses Westinghouse's commitments to the provisions of ANSI/ASME NQA-1-1989 Edition through NQA-1b-1991 Addenda for the AP600 project.

3.2 Review of Calculation Documents for SSAR Chapter 15 Analyses

a. Inspection Scope

During this portion of the inspection, the team reviewed and assessed the adequacy of the design input documents that provide the bases for the analyses presented in Chapter 15 of the AP600 SSAR.

b. Observations and Findings

1. Review of Design Calculation Files

During the inspection, the team reviewed the following Westinghouse design calculation files:

- SSAR-GSC-390 (Rev. 1), "AP600 Feedwater Malfunction Analysis with Loss of Offsite Power at Reactor Trip."
- SSAR-GSC-391 (Rev. 0), "Excessive Load Increase with LOOP."
- SSAR-GSC-335 (Rev. 0), "AP600 Steam Line Break - Core Response."

- SSAR-GSC-335 (Rev. 1), "AP600 Steam Line Break - Core Response with LOOP at Start of Events."
- SSAR-GSC-389 (Rev. 0), "AP600 Loss of Load Analysis with Loss of AC Power."
- SSAR-GSC-336 (Rev. 0), "AP600 Loss of AC Power for SSAR."
- SSAR-GSC-183 (Rev. 0), "AP600 Loss of Flow/Locked Rotor Analysis."
- SSAR-GSC-188 (Rev. 0), "AP600 Boron Dilution Analysis."
- SSAR-GSC-189 (Rev. 2), "AP600 SSAR Inadvertent ECCS Analysis".
- SSAR-GSC-337 (Rev. 0), "Revised SGTR Analysis for AP600 SSAR."
- SSAR-GSC-337 (Rev. 1), "SGTR Analysis for AP600 Assuming a LOOP at Start of Events."
- CN-TA-96-39, "Revision 7 of AP600 Reference Input Data for Non-LOCA Analyses."
- CN-TA-96-155, "Revision 8 of AP600 Reference Input Data for Non-LOCA Analyses."

The team's review consisted of verifying that input data and assumptions were properly documented and that an independent review was performed. The methods used by the team involved review of the calculational methods, review of values for randomly selected design parameters used for the computer input, examination of analytical results for accuracy, consistency check of calculational results and SSAR writeups, and discussion with Westinghouse technical staff for document retrieval and control.

As a result of the review, the team found that the majority of material included in the calculational files reviewed by the team was typed while information presented in a hand-written form was clearly written. For each calculational file, WP-4.5, "Design Analysis," Revision 0 (8/31/96), requires the author and the reviewer to meet the following documentation requirements via a check list:

- The purpose of the design analysis is clearly stated.
- The required inputs and their sources are provided.
- The assumptions are clearly identified and justified.
- The methods and units are clearly identified.
- The limits of applicability have been identified.
- The results of literature search, if conducted, or other background data are provided.
- All the pages are sequentially numbered and identified by the calculation file number.
- The required computer calculation information have been provided.
- The computer codes used were under configuration control.
- The computer code(s) used were applicable for modeling the physical and/or computational problems contained in this calculation file.
- The results and conclusions are clearly stated.
- Open items are properly identified and resolved.
- Approved design controlled practices were followed without exception.
- Computer input files have been transferred to the data Deck Library.
- If any of the above requirements are deviated, provide adequate justification.

The team found that each calculation clearly identified both the calculation author and the reviewer who completed the check list. The team also found that the calculational files contained sufficient information to support the Chapter 15 analyses presented in the

AP600 SSAR. The information included input parameters to computer codes, technical bases for the values selected for the computer input, discussion of the assumptions and methods used for analyses, summary of analytical results including sequences of events, and SSAR writeups to reflect the results of the analyses.

The team noted that the assumptions for analyses were consistent with appropriate Standard Review Plan guidance, the values for input parameters used in the analyses and the associated technical bases could be traced to and found to be consistent with the Westinghouse reference input data sources and the proposed technical specifications for AP600, and the computer codes used in the analyses were listed. The team found that the computer codes were either previously approved by the NRC or used in the licensing applications previously approved by the NRC. The analyses documented in calculational files were accurately reflected in the writeups of the AP600 SSAR.

2. Increase in the RCS Inventory Events (SSAR 15.5)

Upon reviewing SSAR-GSC-189, the inspector found that operator actions had been credited in the analyses of increased RCS inventory events (Westinghouse submittal, NTD-NRC-94-4175). However, Westinghouse had not addressed the following:

- Availability of (1) unambiguous alarms or indications for increased RCS inventory events, and (2) clear procedural instructions to operators to take appropriate actions within the time-frame assumed in the analyses.
- 10 CFR 50.36, "Technical Specifications," §50.36(c)(2)(ii)(C).
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) to verify the capacity of the system including the RV head vent valves that would be used by operators to prevent pressurizer overfill from occurring as assumed in the analyses.
- SSAR text providing detail of the analyses that credited the requisite operator actions.

This issue was identified as part of Nonconformance 99900404/97-02-01.

3. Boron Dilution Analysis (SSAR 15.4.6)

The team found that Westinghouse had relied on the boron mixing testing data documented in EGG-LOFT-5867 (Project No. P 394) to establish the required RCS circulation flow rate of 1000 gpm in TS 3.4.9 and to support its complete boron mixing model assumed in the boron dilution analyses (SSAR 15.4.6 and SSAR-GSC-188). However, the team noted that the AP600 design is different from the test facility configuration and conditions discussed in EGG-LOFT-5867. For example, the injection location of the unborated CVS flow is at the DVI line for AP600 while the test facility simulated a cold-leg injection. For AP600, the maximum boron dilution flow rate assumed in the analyses is 200 gpm and the required TS RCS circulation rate is 1000 gpm to assure complete mixing of the unborated water, while the test facility simulated conditions with unborated flow rate of 300 gpm and RCS circulation rates greater than 3000 gpm. The team concluded that Westinghouse had not reconciled the applicability of the boron mixing testing data to the AP600 design nor had Westinghouse validated the complete boron mixing model assumed in the boron dilution analysis. ***This issue was identified as part of Nonconformance 99900404/97-02-01.***

c. Conclusions

With the exception of the items identified above, the inspector concluded that calculation files prepared by Westinghouse and reviewed by the inspection team were technically adequate and supported the analyses presented in Chapter 15 of the AP600 SSAR. However, Westinghouse needs to evaluate the impact of the nonconformances identified above on other Chapter 15 analyses, not reviewed by the inspection team, to establish the adequacy of the design review process and integrity of the AP600 design.

3.3 QA Review of Large Break Loca; Long Term Cooling Methodology and Analysis

a. Inspection Scope

During this part of the inspection, the team reviewed various technical issues regarding documentation, technical accuracy, and validation and verification regarding large break LOCA methodology, long term cooling methodology and SSAR Section 15.6.5.4C on the AP600 long term cooling analyses results. While the inspector reviewed several documents during the inspection, these represent a small fraction of the total population of documents that provide the bases for the AP600 long term cooling analyses results.

b. Observations and Findings

During this part of the inspection, the inspector reviewed documents dealing with analyses related to the following areas: (1) the large break LOCA methodology based on WCOBRA/TRAC, (2) WCOBRA/TRAC validation for long term cooling (LTC) and (3) SSAR section 15.6.5.4C on LTC. The document selection for the large break LOCA methodology was based on code modifications, the analyses of the CCTF and UPTF experiments and the preparation of the LBLOCA input. The LTC document selection was based on the window mode calculation documentation, WCOBRA/TRAC input generation, the calculation of the collapsed liquid level and an example of test to test input changes. Finally, the SSAR document selection was based on: AP600 window mode calculations, the AP600 WCOBRA/TRAC input model generation, input changes from case-to-case and a sample computation of the AP600 vessel level and DVI flow rates with one ADS-4 valve failed. The following documents were reviewed during the inspection.

- LTCT-T2C-413, "WCOBRA/TRAC Code Modifications Regarding Check Valves," Revision 3.
- SSAR-GSC-205, "WCOBRA/TRAC Modification for Vessel Collapsed Liquid Level Sharpening," Revision 2.
- SSAR-GSC-325, "Analysis of the CCTF Experiment," Revision 0.
- MT01-T2C-260, "Geometric Data for the AP600 CMT Test."

- SEC-APS-4746-C0, "WCOBRA/TRAC Long-Term Cooling."

- SSAR-GSC-028, "CMT Implementation in WCOBRA/TRAC," Revision 0.

- SSAR-GSC-227 (SEC-APS-4964-C0), "UPTF Analysis."

- SSAR-GSC-355, "Input Parameters-AP600 SSAR Revision 3, LBLOCA," Revision 0.

- LTCT-T2C-417, "WCOBRA/TRAC Geometrical Input Data for the OSU Testing," Revision 0.

- LTCT-T2C-418, "OSU LTC Comparisons with WCOBRA/TRAC," Revision 1.

- SSAR-GSC-356, "Two-Inch Break LOCA, LTC," Revision 0.

- SSAR-GSC-377, "SBLOCA Long-Term Cooling," Revision 0.

The inspector identified discrepancies associated with the following calculation notes. ***As a result, these issues were identified as part of Nonconformance 99900404/97-02-01.***

- SEC-APS-4746-C0, "WCOBRA/TRAC Long-Term Cooling."

  The calculation note stated in the concluding section that "...variations in the initial conditions are expected to have relatively unimportant effects on the analysis results," and "Results of changing ICHP is noticeable but not large..." The inspection team noted that no basis for these conclusions was provided in the calculation note.

- LTCT-T2C-417, "WCOBRA/TRAC Geometrical Input Data for the OSU Testing," Revision 0, and LTCT-T2C-418, "OSU LTC Comparisons with WCOBRA/TRAC," Revision 1.

  In LTCT-T2C-417, pages 180-1, Figures 6 and 7, the calculation acknowledged the failure to fit DP vs $(\text{flow})^2$. However, the basis provided was that "...despite the failure to match overall agreement is reasonable." Additionally, this unquantified anomaly was used as input in LTC-T2C-418. Specifically, on page 16 of LTCT-T2C-418, a bias was applied of 0.2 psia to the atmospheric pressure (14.5 > 14.7 psia) to compensate for the disparity in DP vs $(\text{flow})^2$ in LTCT-T2C-417. LTCT-T2C-418 did not provide an explanation or justification for the use of this bias.

- SSAR-GSC-356, "Two-Inch Break LOCA, LTC," Revision 0.

  In an effort to determine single phase flow resistance, a solution of DP vs. flow was presented. The objective of the calculation was to determine the asymptotic part of the solution. The author of the calculation note observed that a harmonic oscillation was built-in to the solution, and therefore, he proposed to take the average value. The inspector, however, could not determine if the average value is equal to the asymptotic solution had the oscillation not been present. Therefore, the calculation did not address the following: (1) impact of oscillation in the asymptotic solution, (2) impact of oscillation on the flow resistance, and (3) presence of oscillation in the vessel flow, DP, and vessel collapsed liquid level solutions.

- SSAR-GSC-377, "SBLOCA Long-Term Cooling"

The inspector identified the following discrepancies in this calculation:

- The author made inappropriate use of the term "containment leakage" in place of vessel leakage and in several solutions, negative (reverse) DVI flow was calculated with no corresponding physical explanation provided.

- A break of 0.264 ft^2 was assumed on a two-sided open break. However, it did not agree with a two-inch pipe break assumed in the calculation. It appears that the break should be considered a cold leg tear.

- On pages 114-115, a significant amount (about 3000 lbs) of liquid was calculated to have leaked from the ADS 1-3 but the problem initial conditions assumed an empty pressurizer and IRWST. The inspector noted that this outcome contradicts the initial conditions specified but no explanation or clarification was provided in the calculation note.

c. Conclusions

The issues identified above represent technical deficiencies found by the inspector while performing a limited review of a larger population of analyses/calculations. The safety significance of these deficiencies needs to be addressed by Westinghouse and an evaluation of their impact, with respect to the overall long term cooling analyses results provided to the NRC in the AP600 SSAR, needs to be performed.

3.4 Review of Westinghouse Software Quality Assurance

a. Inspection Scope

The inspectors reviewed the technical and quality oversight of Westinghouse reactor system and containment safety analysis computer programs. In particular, the inspectors reviewed documentation for the NOTRUMP, WCOBRA/TRAC, and WGOTHIC computer programs. The documents were reviewed against the requirements of Westinghouse QA procedures, 10 CFR Part 50, Appendix B, NQA-1-1989, and 10 CFR 50.46 where applicable.

b. Observations and Findings

NOTRUMP

NOTRUMP is used by Westinghouse for small break LOCA analysis. The inspector reviewed documents related to model changes, error tracking and error correction required for the AP600 analysis program and interviewed the NOTRUMP cognizant engineer. The inspector reviewed SEC-APS-4838-C0, "Software Design Specifications of AP600 NOTRUMP User Externals Cycle 2," NTD-SOD-STD-95-087, an error report on Cycle 2, SSAR-GSC-301, "Software Change Specification of Version 3," and SSAR-GSC-322, "Validation Package for Version 3." The inspector noted that the documentation was clear, complete and errors were tracked and corrected. The validation testing was clearly specified and expected test results were given.

WCOBRA/TRAC

WCOBRA/TRAC is used for LOCA analysis by Westinghouse. The inspectors reviewed documents related to model changes, error tracking and error correction required for the AP600 analysis program. The inspectors also interviewed the cognizant engineer in charge of WCOBRA/TRAC.

The version of WCOBRA/TRAC used for AP600 analysis is version MOD 7A Revision 1A, which is based on operating reactor version MOD 7A, Revision 1. Several AP600 specific changes are added to this version to create MOD 7A, -Revision 1A. The software changes to create Revision 1 are specified in SEC-APS-5037-C0 and the verification of these changes are documented in SEC-SAII-5063-C0. The verification report for the AP600 specific changes to WCOBRA/TRAC did not give reasons for the specific tests chosen or for the parameters chosen for comparison in the tests.

The inspector also reviewed the code error report for MOD 7A, Revision 1, contained in NTD-NSA-SAI-96-332 and the release letter for the current operating reactor version MOD 7A, Revision 2, which contains 10 CFR 50.46 reporting information. One error was listed that affects code timestep control if a WCOBRA/TRAC run has to be restarted. The error impact on PCT was assessed as 0 degrees F for both operating reactors and AP600 because the large break LOCA methodology does not use restarts. Although the operating reactor analyses do not contain restarts, the AP600 large break LOCA analyses do in fact contain restarts. The code cognizant engineer thought that the impact of this error would probably be small on AP600 analyses. There was no evidence that the error list was evaluated for the specific case of the AP600.

The inspector also discussed the code failures with the AP600 analyst and the WCOBRA/TRAC cognizant engineer. Although the analyst discussed the failures with one of the WCOBRA/TRAC developers, the cognizant engineer stated that he was not aware of the failure. The inspector found that the code failure was not tracked in the WCOBRA/TRAC error tracking system and therefore, no official code error report was filed. ***This issue was identified as part of Nonconformance 99900404/97-02-02.***

WGOTHIC

WGOTHIC is used by Westinghouse to perform design basis peak pressure calculations for the AP600. The WGOTHIC code is somewhat different than the Westinghouse safety analysis codes used for reactor systems analysis in that part of WGOTHIC is developed and maintained by Numerical Applications Incorporated (NAI). Westinghouse then makes modifications to the code to implement special models that are used in AP600 containment calculations as WGOTHIC. Since the GOTHIC foundation of WGOTHIC is commercial software, it must go through a dedication process when it is brought into Westinghouse before it can be used in safety-related applications. The dedication, contained in CN-SMA-91-192-R0, contains descriptions of the testing process and shows acceptance by comparison to the NAI results. Although the inspector did not perform a full independent review of the dedication process, there was evidence that an independent QA review had been performed.

Westinghouse updated their GOTHIC base code to GOTHIC version 4.0 as a result of code deficiencies identified during the AP600 review, but no explanation was given in the documentation for code changes. There was also no reference to NAI documentation that would have given documentation of the code changes and Westinghouse did not run all NAI test cases to verify that the update was correct. The Westinghouse document reviewer questioned why all test cases were not run. The author responded that some of the test cases take a long time to run and that the cases run were more than enough to verify the code. There was no documentation of what functionality was tested by the test cases or whether they covered all uses of GOTHIC by Westinghouse. Stating that some test cases take a long time to run is not adequate justification for not running them. The inspector concluded, however, that the test cases used by Westinghouse were adequate to test the code functionality.

The main change made to GOTHIC by Westinghouse to make WGOTHIC is the film heat transfer package used in modeling the AP600 containment shell which Westinghouse refers to as the clime model which is a large and complicated model change. The inspector reviewed CN-CRA-93-219-R0 which is the design specification of the clime model. The beginning of the document describes what is called a complete and correct mathematical and physical model of the film energy transport but the equations are not mathematically and physically complete and correct. A complete and correct description would start out with mass, momentum and energy balances on the film and then show what terms can be neglected to obtain the final mathematical model. Several terms are obviously missing from the equations including condensation and evaporation terms which appear about 60 pages after the original "complete" model equations are discussed. There are also terms missing that depend on the time rate of change of the film thickness that result from the application of Leibniz's rule to the integral balance equations for a moving boundary problem. These missing terms may be negligible if the film thickness is changing slowly, but the assumptions incorporated into the complete equations should be clearly stated in the documentation.

In addition, related to Equation 8, an artificial thermal capacitance equal to half the thermal capacitance of the film is added to the thermal capacitance of the wall node adjacent to the film for numerical stability reasons. Adding this artificial term introduces an error in energy conservation. The term should either be removed from the equations or justification provided that the error introduced is negligible. ***This issue was identified as part of Unresolved Item 99900404/97-02-03.***

The inspector reviewed NAI GOTHIC errors identified after GOTHIC 4.0. The inspector noted that the letter for the latest release of WGOTHIC had a list of more than 100 uncorrected NAI GOTHIC errors. NAI had determined that some of these errors could affect safety determinations and may be reportable under 10 CFR Part 21. Westinghouse Procedure WP-4.19.3, "Software Error Reporting and Resolution," requires that the impact of errors must be reviewed on all work activities where the program was used during the time period the error existed and documented on an Error

Impact Review sheet within 60 days of receiving the error report. However, Westinghouse could not provide documentation to support performance of this review. ***This issue was identified as part of Nonconformance 99900404/97-02-02.***

c. Conclusions

The NOTRUMP code appeared to be adequately documented from the standpoint of software quality assurance, however several weaknesses were identified with WCOBRA/TRAC and WGOTHIC. The documentation of code changes in WGOTHIC did not allow an independent reviewer to understand why the code changes were made in all cases and the documentation of test cases for WGOTHIC and WCOBRA/TRAC did not specify why the test cases were being run or the expected results of the testing. Westinghouse also implemented major model changes into WGOTHIC with a poorly documented mathematical and physical model. The tracking and documentation of computer code errors and their impacts for WGOTHIC and WCOBRA/TRAC as applied to AP600, was not performed in accordance with the requirements of Westinghouse Procedure WP-4.19.3.

3.5 Review of NOTRUMP Small Break LOCA Code

a. Inspection Scope

The inspector reviewed selected calculation notes to determine the acceptability of the quality assurance provisions being implemented by Westinghouse for developing the code and calculation notes pertaining to the AP600 version of the NOTRUMP small break LOCA computer program.

b. Observations and Findings

- SEC-APS-4838-CO, "Software Design Specifications of AP600 NOTRUMP User Externals Cycle 2."Revision 0, dated September 9, 1995.

  This calculation disclosed that the author's response to the reviewer's comment indicating numerous errors in code parameters was that the errors were not viewed as significant and would be corrected in a later code revision. However, no basis was given to justify the conclusion that the errors were insignificant.

- SEC-APS-4837-CO, Revision 0, dated September 9, 1995, "Software Change Specification of NOTRUMP Cycle 32."

  The calculation identified that the author in response to a reviewer's comment wrote that he, "doesn't know enough about void propagation methodology of flow link property determination to determine whether unphysical interphase velocities have significant impact." The author does not give any indication of attempting to find a resource who is capable of responding to the reviewer's concerns, thus there is no evidence that the reviewer's concern was addressed.

The deficiencies discussed in the above examples were identified as part of Nonconformance 99900404/97-02-01.

c. Conclusions

With the exception of the items identified above, the inspector concluded that calculation files prepared by Westinghouse and reviewed by the inspection team were technically adequate and supported the use of the NOTRUMP computer code for application to the AP600 design. However, Westinghouse needs to evaluate the impact of the nonconformances identified above on other calculation notes pertaining to the AP600 version of the NOTRUMP small break LOCA computer program, not reviewed by the inspection team, to establish the adequacy of the AP600 design review process and integrity of the AP600 design.

3.6 QA Review of WGOTHIC Computer Code

a. Inspection Scope

The inspection team reviewed selected documents to determine the acceptability of the quality assurance provisions being implemented by Westinghouse for developing the computer program and calculation notes pertaining to the validation, verification and use of the Westinghouse WGOTHIC computer program. WGOTHIC is used to perform the design basis accidents analyses for peak containment pressure to support the AP600 design certification.

b. Observations and Findings

- CN-CRA-95-089, "Validation and Verification of INPUT Small Internal-Use Computer Program," Revisions 0 and 1.

  The document disclosed that the author's response to the reviewer comment concerning an incorrect value for an area as used in the calculation was not considered to be of sufficient consequence to warrant a code revision. No specific evaluation to support the conclusion was provided and the inspector could not assess the impact without recourse to the originator. It was also noted by the author that if other changes were found to be necessary then this error should be corrected.

  In reviewing the independent verification of the calculation performed by the reviewer, it was noted that in one case an equation was written with a "+" sign to add two terms. The reviewer's equation contained a "-" sign but the computed value was based on a summation. The FORTRAN coding for this specific equation was inspected and found to be correct. Westinghouse informed the team that the area error had been corrected and that the conclusions derived from the study remained unchanged.

- CN-CRA-94-147, "Phase 2/3 Large Scale Test Lumped Parameter WGOTHIC Base Case Deck," Revisions 0 and 1.

  The review of this document disclosed that the reviewer's comments and the author's responses were imbedded in the document. In general, the author's responses were found to include sufficient details to assess the responses. The inspection team noted that errors in the model were found after the computer analyses had been performed.

In one case, the author's response indicated that a k-loss factor was likely acceptable if other specific conditions were met. There was no statement as to the expectation of the condition being met or verified.

- CN-TA-96-153 AP600, "Steamline Break Mass and Energy," Revision 0.

  The review disclosed that errors existed in the analyses that were identified after the computer runs had been completed. The inspection team determined that some errors would be conservative and that some would be non-conservative. The author's assessment was that there was no impact associated with the errors. It was also noted that the specific errors did not occur in the limited analyses that support design certification. The inspection also disclosed that the computer program used to calculate the SSAR mass and energy releases for the steamline breaks is LOFT4AP Version 1.8 and that the values presented in the SSAR are consistent with this supporting calculation.

- 1100-SOC-001, "Containment Volumes and Heat Sinks," Revisions 0 through 4.

  The review of Revisions 2 through 4 disclosed that the development of the containment volumes and heat sinks were developed and updated based on the nuclear island general arrangement drawings starting with Revision 6 and ending with Revision 8. Rev. 0 was a preliminary scoping document that was completely revised in Rev. 2. No specific review comments were identified however each page contained a sign off block with the author and verifier (reviewer's) signatures. Minor deviations were identified such as use of "estimated" and "assumed" dimensions for non-critical components and diagrams without units or dimensions clearly identified.

  The inspector also found a summary table with an incorrect value for a surface area in Revision 0. The error was also contained in Revision 1, however the correct value was found in the current version, Revision 4.

  The inspector also determined that the thermal insulation surrounding piping and components was not considered in the development of containment volumes and flow paths (either between compartments in the below operating deck regions or between the below operating deck regions and the above operating deck region). No justification for the treatment of insulation was found in any document.

  In addition to concerns regarding the lack of margin to design pressure in the current licensing analyses presented in SSAR 6.2.1.3 and 6.2.1.4 (the MSLB limiting case with a peak pressure of 44.8 psig and the LOCA with a peak pressure of 44.0 psig as compared to the 45 psig design pressure), there are other design analyses areas that may be impacted by the insulation issue. For example, the subcompartment loads analyses provided in SSAR 6.2.1.2, and the evaluation of flooding levels within compartments and which compartments may be subjected to flooding.

With respect to SSAR 6.2.1.3 and 6.2.1.4, Westinghouse needs to:

a. Evaluate the significance of the insulation on the free volume used to determine the peak containment pressure. Provide adequate justification, as appropriate, that the free volume is conservative.

b. Evaluate the significance of the insulation on the flow path characterizations used to determine the peak containment pressure, including flow areas and form losses, for both paths connecting below operating deck compartments as well as flow paths connecting below operating deck regions to above operating deck regions. Assess the effects for each of the 4 LOCA phases as well as the MSLB.

With respect to SSAR 6.2.1.2, Westinghouse needs to:

Evaluate the significance of the insulation on the flow path characterizations used to determine the differential pressures across subcompartment walls, including flow areas and form losses, for both paths connecting below operating deck compartments as well as flow paths connecting below operating deck regions to above operating deck regions.

With respect to the flooding issues, Westinghouse needs to:

Evaluate the significance of the insulation on compartment flooding, address both timing and levels, as well as which compartments would be affected.

- CN-CDBT-92-233, "AP600 WGOTHIC Input Deck Development," Revisions 2 and 3.

The review disclosed that errors existed in the analyses that were identified after the computer runs had been completed. The specific errors identified by the reviewer in Revision 3, dated May 22, 1997, concerning errors in areas and k-loss factors, were determined to have negligible impact on the analyses and therefore reanalyses were not performed. No discussion on how this conclusion was reached was provided. Similar statements were found in CN-CDBT-92-233, Rev. 2, concerning an error in the calculation of a hydraulic diameter (although in this specific case the AP600 nodalization was modified after the error was found, however the inspection review was not detailed enough to determine if the error was promulgated into the revised model). It is also noted that to follow this nodalization change from Rev. 2 to Rev. 3, a third document, CN-CRA-97-034, Revision 1, had to be reviewed by the team since the specific justification for the change was not included in the CN-CDBT-92-233 series of reports.

CN-CRA-97-034 was not reviewed to determine if there were any errors identified in the model. The specific nodal change addressed a concern with the computed steam concentrations near the ADS. Considering the lack of margin, the potential that there are other "negligible" errors is a concern. As a result, Westinghouse needs to:

a. Provide justification that the impact of these errors and incorrect loss coefficients are conservative, or that the cumulative impact of known errors would not result in a change in the pressure calculation greater than 0.2 psig. The loss factors selected were considered to be applicable to "natural circulation" but, for the flow paths in question, the loss factors should not have been applied. Assess the effects for each of the 4 LOCA phases as well as the MSLB.

b. Provide justification for allowing known errors to remain in the licensing analyses that support design certification. Include the supporting knowledge base employed by Westinghouse that is used to assess errors to determine that, in consideration of the 0.2 psig (½ of 1%) margin in the calculated allowance to the design pressure, known errors have a negligible impact (for example only conservative errors remain, or that the cumulative impact of known errors would not result in a change in the pressure calculation greater than 0.2 psig). Consideration should be given to both accumulation of errors as well as the impact of errors in consideration of the different phenomena and characterizations for each of the four LOCA phases and the MSLB.

In each of the examples discussed, additional information is required to assess the impact of the errors identified. ***This issue was identified as part of Unresolved Item 99900404/97-02-03.***

c. Conclusions

With the exception of the items identified above, the inspector concluded that calculation files prepared by Westinghouse and reviewed by the inspection team were technically adequate and supported the use of the Westinghouse WGOTHIC computer program for AP600 design certification. However, the impact of the issues identified in the unresolved item needs to be evaluated by Westinghouse with respect to the adequacy of the AP600 design verification process and integrity of the AP600 containment design.

3.7 Review of Items Related to AP600 Program Audits and Procurement Issues

a. Inspection Scope

The inspector selected several AP600 purchase orders and audit reports to ascertain Westinghouse's implementation of the appropriate provisions in WCAP-2600 with respect to procurement document control, control of purchased services, audits, and associated corrective actions.

b. Observations and Findings

- Design Assurance Review of AP600 Vendors

  Westinghouse letter NSD-NRC-97-5370, dated October 16, 1997, documented the results of a design assurance review of vendors used by Westinghouse for the AP600 program. Items 16, 17, and 19 of Table 1 (enclosed with the above Westinghouse letter) noted additional actions that were required by Westinghouse's vendors. As a result, the inspector reviewed the following actions:

  Item 16: ENUSA (Empresa Nacional del Uranio, S.A.) FS02-VDAQ-001, Revision 1, dated October 28, 1997, related to fuel storage criticality analyses. The document was corrected as committed and associated changes made to the AP600 SSAR in revision 17. The document did not contain a revision page documenting the specific nature or purpose of the revision however the cover letter did provide some information on the reason for the change.

Item 17: A review of the root cause analysis and corrective actions associated with a draft response from INITEC on errors contained in document 1200-CCC-102, dated October 31, 1997, appeared reasonable. Westinghouse requested that INITEC provide a formal letter documenting its conclusions on November 13, 1997. The Westinghouse action appears to be appropriate.

Item 19: The team reviewed Westinghouse document SSAR-GSC-379, Revision 1, dated November 3, 1997, documenting the origin of the initial and boundary conditions used in the calculations. The document was originally prepared by Westinghouse vendor NNC, however Westinghouse prepared the revision as committed and appears to have provided the appropriate information.

- Service Contract Purchase Order Review

  The inspector reviewed the following Westinghouse initiated purchase orders during the inspection.

  - Penn State Subcontract No. MB24124H

    This contract was initiated to add non-condensibles to WCOBRA/TRAC calculations and to assess boron dilution transients. The subcontract required the establishment of an NQA-1 equivalent quality assurance program for the work performed and a quality assurance project plan was prepared. Westinghouse audited the subcontractor on October 10-11, 1995, with no problems noted.

  - ANSALDO Subcontract No. MB23889 for AP600 Long-Term Cooling Analyses

    The contract required that work performed by ANSALDO at Westinghouse be performed under Westinghouse QA WCAP-12601, and that work performed in ANSALDO's office be performed in accordance with ANSALDO QA program description ABP-001 PQEX QP001000. Work under the contract work was formally audited by Westinghouse in 1991, 1993, and 1995.

  - NNC Subcontract No. MA72217-H

    The contract was initiated to help support WCOBRA/TRAC code verification of noding of experiments UPTF Test 21 phases A and B; and CCTF Test 58. The contract required conformance to NNC QA manual QAM-1, Issue A, as audited and accepted by Westinghouse during their 1992 and 1997 audits.

  - OSU Subcontract No. MA02824-H

    This contract was initiated for the design and scaling of the OSU facility. The contract required OSU to implement a quality assurance program in accordance with NQA-1 and to submit long-term cooling test project quality plan LTCT-GAH-001 to Westinghouse. OSU was audited by Westinghouse in 1993 and 1994.

- Review of Westinghouse AP600 Internal Quality Audits

  The inspector reviewed Westinghouse Internal Audit ESBU-94-38, "AP600 FOAKE Design Control," dated October 24-28, 1994 (including assessment report dated December 8, 1994), which identified 13 findings, and Internal Audit ESBU-95-49, "AP600 FOAKE Design Control," dated December 11-15, 1995 (including assessment report dated January 5, 1996), which identified 7 relatively minor findings related to design and document control. Westinghouse concluded that none of the findings identified indicated a significant breakdown of the design control process.

c. Conclusions

Based on the above, the inspector concluded that Westinghouse was effectively implementing its AP600 QA program provisions with respect to procurement document control, control of purchased services, audits, and associated corrective actions.

3.8 Entrance and Exit Meetings

An entrance meeting was held on November 17, 1997, in which the scope of the inspection was discussed with Westinghouse management and staff. On November 21, 1997, an exit meeting was held with Westinghouse management and staff to discuss the inspection findings.

4 PERSONNEL CONTACTED

The following represents a partial list of persons contacted during the inspection:

Westinghouse Electric Corporation

Ed Cummins, General Manager, New Plant Projects Division
David Alsing, AP600 Quality Systems Manager
Rao Mandava, Manager, AP600 Plant Engineering
Bob Vujuk, Manager, AP600 Projects
Brian McIntyre, Manager, Advanced Plant Safety and Licensing
Jim Gresham, Manager, Containment and Radiation Analysis
Earl Novenstern, Consultant to Westinghouse
Eugene Piplica, Lead Engineer, AP600 Test Programs
Robert Tupper, Project Engineer
Ken Kloes, Projects Quality Assurance Engineer

5 ITEMS OPENED AND CLOSED

Opened

99900404/97-02-01	NON AP600 calculations-inadequate design review
99900404/97-02-02	NON Failure to review GOTHIC code errors
99900404/97-02-03	URI WGOTHIC calculation deficiencies

Closed

99900404/95-01-01	NON Reactor Systems Design Certification Test Program
99900404/95-01-02	NON Reactor Systems Design Certification Test Program
99900404/95-01-03	NON Reactor Systems Design Certification Test Program
99900404/95-02-01	NON As-Built Drawings for VAPORE Test Facility
99900404/95-02-02	URI VAPORE Test Facility Calibration Records
99900404/97-01-01	NON Inadequate corrective action
99900404/97-01-02	NON Inadequate quality and technical oversight of INITEC
99900404/97-01-03	URI Acceptability of AP600 design deliverables

Selected Generic Correspondence on the Adequacy of
Vendor Audits and the Quality of Vendor Products

Identifier	Title
Information Notice 98-03	Inadequate Verification of Overcurrent Trip Setpoints in Metal-Clad, Low-Voltage Circuit Breakers

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Division of Reactor Controls and Human Factors
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 Quality Assurance, Vendor Inspection and Maintenance Branch, that have been distributed to the inspected organizations during the period from January through March 1998

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)* Vendor Inspection	13. AVAILABILITY STATEMENT Unlimited 14. SECURITY CLASSIFICATION (This Page) unclassified (This Report) unclassified 15. NUMBER OF PAGES 16. PRICE

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