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LICENSEE CONTRACTOR AND VENDOR INSPECTION STATUS REPORT

QUARTERLY REPORT
January 1989 - March 1989

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Division of Reactor Inspection and Safeguards
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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PREFACE

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

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

In the case of the principal licensee contractors, such as nuclear steam supply system designers and architect engineering firms, the NRC encourages submittal of a description of corporate-wide QA programs for review and acceptance by the NRC. Once accepted by NRC, a corporate QA program of a licensee's contractor will be acceptable for all license applications that incorporate the program by reference in a Safety Analysis Report (SAR). In such cases, a contractors's QA program will not be reviewed by the NRC as part of the licensing review process, provided that the incorporation in the SAR is without change or modification. However, new or revised regulations, Regulatory Guides, or Standard Review Plans affecting QA program controls may be applied by the NRC to previously accepted QA programs.

When design and construction activities were high, firms designing nuclear steam supply systems, architect engineering firms designing nuclear power plants, and certain selected major equipment vendors were inspected on a regular basis by NRC to ascertain through direct observation of selected activities whether these design firms and vendors were satisfactorily implementing the accepted QA program. However, with the substantial decline of new plant design activities, the inspection of QA program implementation has been deemphasized. Instead, the NRC vendor inspection focus has been shifted to vendor activities associated with nuclear plant operation, maintenance, and modifications. Inspection emphasis in now placed on the quality of the vendor products including hardware fabrication, licensee-

vendor interfaces, environmental qualification of equipment, and equipment problems found during operation and corrective action. If nonconformances with NRC requirements and regulations are found, the inspected organization is requested to take appropriate corrective action and to institute preventive measures to preclude recurrence. If generic implications are identified, NRC assures that affected licensees are expeditiously informed.

In addition to the above, the Vendor Program Branch has begun inspections at licensee facilities covering the areas of procurement of replacement parts for use in safety-related systems and licensee/vendor interface programs as requested in Generic Letter 83-28. This edition of the White Book contains copies of the inspection reports of inspections completed to date. Subsequent issues will contain those reports that are issued in the quarterly report period covered by that White Book.

In the past, NRC issued confirming letters to the principal contractors to indicate that NRC inspections have confirmed satisfactory implementation of the accepted QA programs. Licensees and applicants could, at their option, use the letters to fulfill their obligation under 10 CFR 50 Appendix B, Criterion VII, that requires them to perform initial source evaluation audits and subsequent periodic audits to verify QA program implementation. However, based on the above described change in nuclear plant design and construction activities, NRC will no longer issue confirming letters to principal contractors since future NRC vendor program inspections will focus on selected areas rather than addressing the implementation of their respective QA programs. Therefore, confirming letters that have already exceeded their three year effective period will not be renewed. Confirming letters issued less than three years ago will remain in effect until the stated effective period expires. Therefore, as the confirming letters expire, licensees and applicants will no longer be allowed to take credit for the NRC acceptance of the implementation of a principal contractor's QA program. Licensees continue to be responsible for the conduct of initial source evaluation audits and subsequent periodic audits to verify QA program implementation.

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

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

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

ORGANIZATION: COMPANY, DIVISION
CITY, STATE

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

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

ORGANIZATIONAL CONTACT: Name/Title
TELEPHONE NUMBER: Telephone Number

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

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

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

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

INSPECTION BASES AND SCOPE:

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

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

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

ORCANIZATION: ORGANIZATION
CITY, STATE

REPORT NO.:	INSPECTION RESULTS:	PAGE 2 of 2

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

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

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

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

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

F. PERSONS CONTACTED: Typed, Name, Title

*present during exit meeting

SAMPLE PAGE
(EXPLANATION OF FORMAT AND TERMINOLOGY)

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INSPECTION REPORTS

ORGANIZATION: AUMA REISTER KG
MULHEIM, WEST GERMANY

REPORT NO.: 99901134/88-01	INSPECTION DATE: November 14-15, 1988	INSPECTION ON-SITE HOURS: 16

CORRESPONDENCE ADDRESS: Mr. R. Dinse, Director
Auma Reister KG
D-7840 Mulheim (Baden)
West Germany
P.O. Box 1362

ORGANIZATIONAL CONTACT: Mr. R. Dinse
TELEPHONE NUMBER: (07631) 8090

NUCLEAR INDUSTRY ACTIVITY: Manufacturer of electric motor-operated actuators for valves.

ASSIGNED INSPECTOR: _____ 12/7/88
K. R. Naidu, Program Development and Reactive Inspection Section (PDRIS) — Date

OTHER INSPECTOR(S):

APPROVED BY: _____ 12/21/88
E. T. Baker, Chief, PDRIS, Vendor Inspection Branch — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR 50 Appendix B.

B. SCOPE: Review of implementation of the Quality Assurance Program in selected areas. Review of quality assurance records for the motor operators supplied to the Trojan Nuclear Power Plant.

PLANT SITE APPLICABILITY: Trojan 50-344.

ORGANIZATION: AUMA REISTER KG
MULHEIM, WEST GERMANY

REPORT NO.: 99901134/88-01	INSPECTION RESULTS:	PAGE 2 of 8

A. VIOLATIONS:

No violations were identified during this inspection.

B. NONCONFORMANCES:

No nonconformances were identified during this inspection.

C. UNRESOLVED ITEMS:

No unresolved items were identified during this inspection.

D. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Background Information

Auma Reister KG (Auma) has been designing and manufacturing electric motor driven actuators to operate valves for approximately 26 years. Auma has a branch factory in Osrfildern, a suburb of Stuttgart, where worm gear boxes, quarter turn actuators and small actuators are manufactured. Representatives of Auma stated that they have supplied actuators for several industrial applications including fossil and nuclear power plants in several countries, including the United States. Documents indicate that SAI and SAN type actuators successfully withstood the test requirements of Institute of Electrical and Electronic Engineers (IEEE) standards 323-1974, IEEE 344-1975, and IEEE 382-1972.

SAI type actuators, which have no aluminium on their housings, are suitable for installation inside the containments of nuclear power plants. SAN type actuators are suitable for installation outside the containment. SAN and SAI type actuators were also qualified to meet Kern Tecnische Anlagen (KTA) requirements based on IEEE-382, IEEE-323, IEEE-467, EPRI NP-2129-1981, EPRI-1558, and KTA-3504.

Nine SAI type actuators were manufactured, tested, and supplied to operate stainless steel valves manufactured by Kerotest for installation in the Steam Generator Blowdown Modification System at the Trojan Nuclear Power Plant located in Portland, Oregon.

2. Plant Tour

The inspector toured the plant in the areas of machining, assembly, storage of purchased material, incoming receipt inspection, test

area, and the research and development area where end-of-life tests are being performed. The inspector interviewed component assemblers, in-process roving quality control (QC) inspectors and test technicians, and determined that the individuals interviewed were knowledgeable in the work they performed.

3. Quality Assurance Program

The inspector reviewed the quality assurance (QA) program of Auma and determined that it basically meets the requirements of Appendix B to 10 CFR 50. The principal functions of the QA manager and his representatives are the following:

a. Control of purchased materials, including audits of subvendors and receipt inspections on incoming material.

b. First piece inspections and random sampling inspections thereafter on machined parts, including roving in-process inspections.

c. Measurement and testing, including control of calibration.

d. Functional testing.

e. Final testing.

f. Packaging and shipping.

g. QA documentation.

h. Retention of documents.

4. Reviews of Control of Purchased Materials

All raw materials, including rotors and stators for motors, are purchased from vendors approved by the quality assurance department. Periodic audits are required to be performed on these vendors. Incoming receipt inspections are performed on material received. Accepted material is adequately identified and transferred to an automated storeroom. Components from the storeroom are retrieved by automated systems.

5. Review of Inspections Performed

The inspector observed the implementation of the inspection program in the following areas:

a. Receipt Inspections

Incoming receipt inspections are performed on all purchased components. Depending on the quantity and complexity, the components are either subjected to 100 percent inspection or statistical sampling. Rotor and stator assemblies for electric motors are purchased from two vendors. Procedure KV-1-5-14-2 GB Revision 2, dated March 18, 1985, entitled "Incoming Goods - Inspection of Motor Components for MDN/MDI Type Motors" is the applicable document used during receipt inspections. This procedure contains instructions for visual inspections and electrical test measurements. A check list attached to the procedure is required to be completed to document the results of the inspection. Rotors and stators failing to meet the established criteria are rejected and returned to the vendor.

b. In-process Inspections

In-process inspections were being routinely performed by roving quality control (QC) inspectors throughout the plant. The inspector observed one cast iron component being inspected after being machined. The component was rejected for flaws on the machined surface, identified as a nonconforming item and segregated. The inspector observed dimensional checks being performed on a randomly selected component on which all the machining operations had been completed. The checks were being performed by a computer-assisted, automatic comparator machine in which the critical dimensions on the relevant drawing were programmed.

The QA manager informed the inspector that if a purchased component is identified to be defective and is rejected during in-process inspections, the component is returned to the vendor. Inspections for the replacement components from the vendor are required to start from the initial step of the inspection procedure.

The measuring instruments used in the inspections were observed to have calibration stickers to indicate the date of calibration and the due date. The inspector observed that the calibration stickers on the measuring devices were current.

6. Review of Corrective Action

The inspector verified the implementation of the nonconformance process relative to rejected material by selecting a motor which was rejected on March 10, 1988. Nonconformance Report (NCR) M175, dated June 8, 1988, identified that during the final tests of the motor assembly, serial number 1840120, the measured no-load current was observed to be between 2.7 and 4.5 amperes (A) instead of 2.6 A. The 1.1 kilowatt (KW), 380 volt, 4 pole, MDI 90/4-75 type motor was intended for LOOVISA Nuclear Power Plant in Finland. The motor was purchased from Stahl Schmidt Company, Bielefeld, on March 10, 1988. Auma returned the defective motor to the vendor on June 13, 1988 along with a copy of NCR 175. The vendor examined the motor and determined that a winding in the starter was short circuited. The vendor supplied a replacement starter with serial number 184720. This motor was tested and determined to meet the specification requirements.

7. Review of Quality Assurance Records

The inspector reviewed the following quality assurance records relevant to the supply of nine SAI 25 type valve actuators to the Portland General Electric Company. The actuators are intended to operate stainless steel valve actuators manufactured by Kenotest installed in the modified Steam Generator Blow Down System at the Trojan Nuclear Power Plant located in Oregon. The serial numbers of the actuators are 4987-69293 to 4987-69301. The quality assurance records were stored in two independent buildings.

a. Inspection Records

The inspection records consisted of the following documents:

1. Incoming receipt inspection checklist to indicate that the electric motors received from the vendor were inspected to Procedures KV 1-5-14-2. The motor windings have Class H insulation.

2. Check list documenting the results of the final inspection of electrical motor assemblies performed to Procedure KV-1-5-15-2.

3. Final test results of assembled motors to Procedure KV-1-5-17-2.

4. Check lists documenting the results of tests performed in accordance to Procedure KV-1-3-33-0 Revision 0, dated November 11, 1987. This procedure, exclusively developed for the actuators supplied to Trojan, entitled "Final Test and Inspection of Auma Actuators Type SA/SAN/SAI" provided instructions for the following tests:

   a. General instructions to follow the procedure.

   b. Visual inspection of the actuator.

   c. Insulation resistance test of the motor.

   d. High voltage test.

   e. Manual operation and automatic change over to electric drive on demand.

   f. Functional tests.

   g. Locked rotor current measurements.

   h. Output torque closing and opening, measured at 70 percent of nominal voltage (under voltage) and 110 percent nominal voltage (over voltage).

   i. Setting the torque switch and the limit switches for the open and close positions. Number of turns to operate the limit switch was nine.

   j. Air-tightness test.

   k. Check the operation of all options ordered.

   l. Final inspection.

   m. Inspection of painting.

b. Certifications

The documentation contained the following certifications:

1. Certification dated January 12, 1988 certifying that the calibration of electrical and mechanical instruments used in the testing of the actuators supplied to Trojan, were current and that the instruments were calibrated in accordance to Vereinigte Deutsche Ectrotechnische Verein (VDE) and Deutsche Industrie Normen (DIN), which are German standards.

2. Certification dated January 12, 1988, stated that all material used in the manufacture of the Trojan actuators was purchased by imposing the requirements of the Auma quality assurance manual. The subvendors were subjected to audits by Auma QA personnel.

3. Certification to indicate that all inspection and test personnel were qualified and trained. Records of training given to individuals on various subjects were available.

c. Review of Design Changes

The inspector reviewed the design changes made in the manufacture of Auma actuators after the actuators were qualified to the requirements of IEEE-323, IEEE-344, and IEEE-382 in 1979. The only changes were in the materials used. The sealing material was changed from EPD to 70NBR because EPD started to swell when it came into contact with grease. To fabricate the support plate for the torque switch mechanism, X5 Cr Ni 189 type of material is being used instead of AI Mg 3F26, which was determined to be weak. The material for the pinion of the torque switch was changed from X12 Cr MOS 17 1-41-04 to X20 Cr 13 1-4021 because the latter was determined to be more resistant to wear. Auma had determined that the above changes do not invalidate the original seismic and environmental qualifications.

d. Deviation Report

One deviation report (DR), dated December 22, 1987, was included in the documentation package. The DR identified that parts list SSAI-01-02 indicated 2 x 19 Belleville springs are installed on the worm shaft of the SAI actuators. According to test

reports, TB-N 1-1-198, dated October 14, 1982 and R541/85/57, star washers were installed between the Belleville springs for the SA1-25 type actuator. Corrective action recommended was to assemble the Belleville washers in accordance with the test reports.

E. EXIT INTERVIEW:

The inspector met with persons identified in Section F and discussed the scope and findings of the inspection.

F. PERSONS CONTACTED:

R. Dinse, Managing Director
K. P. Herr, Technical Director, Sales
H. Weber, Marketing Director
G. Waldenmaier, Manager, Quality Assurance
H. G. Woesner, Research Engineer

ORGANIZATION: COMBUSTION ENGINEERING, INC.
POWER SYSTEMS GROUP
WINDSOR, CONNECTICUT

REPORT NO.: 99900401/88-01	INSPECTION DATE: 11/8-10/88	INSPECTION ON-SITE HOURS: 16

CORRESPONDENCE ADDRESS: Mr. Walter D. Mawhinney, Vice President
Nuclear Quality Systems
Combustion Engineering, Inc.
Power Systems Group
1000 Prospect Hill Road
Windsor, Connecticut 06095

ORGANIZATIONAL CONTACT: Mr. Thomas R. Swift, Manager, Nuclear Quality Systems
TELEPHONE NUMBER: (203) 285-9204

NUCLEAR INDUSTRY ACTIVITY: CE has had NSSS contracts for 16 domestic reactors, and has support service contracts for approximately 40 reactors worldwide.

ASSIGNED INSPECTOR: R. C. Wilson, Engineer, Special Projects Inspection Section — Date: 1/9/89

OTHER INSPECTOR(S):

APPROVED BY: Uldis Potapovs, Chief, Special Projects Inspection Section, Vendor Inspection Branch, DRIS, NRR — Date: 1-10-89

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Parts 21 and 50

B. SCOPE: This inspection was made as a result of an allegation concerning moisture effects on the environmental qualification of Litton-Veam electrical connectors used in Core Exit Thermocouple (CET) systems and Reactor Vessel Level Monitoring (RVLM) systems supplied by Combustion Engineering.

PLANT SITE APPLICABILITY: Numerous; see Section E.4 and Table I at end of report.

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED ITEMS:

None

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

None are applicable to the scope of this inspection; none were addressed during this inspection.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Allegation scope

The allegation consists of two specific concerns, both relating to multi-pin electrical connectors supplied by CE as portions of instrumentation systems required to be environmentally qualified to 10 CFR 50.49. The concerns are as follows:

a. It was alleged that Core Exit Thermocouple systems are subject to signal errors larger than reported by CE based on qualification test report CE NPSD-230-P. Qualification can only be accomplished by a retrofitted improvement for the original connectors or use of an improved connector, either of which prevents moisture ingress into the connector.

b. It was alleged that Reactor Vessel Level Monitoring systems supplied by CE and using the Heated Junction Thermocouple (HJTC) principle require periodic testing to maintain qualification because an undersized connector gasket can permit moisture ingress. Undersized gaskets may have been manufactured undersized or may have been compressed during service.

These two concerns were conveyed to CE and were addressed during this inspection.

2. Core Exit Thermocouple System

The CE Core Exit Thermocouple (CET) system is intended to satisfy Regulatory Guide 1.97, and must be environmentally qualified for Loss of Coolant Accidents (LOCAs) and other accidents. The area of concern is a connector that electrically and mechanically joins two multi-conductor, mineral-insulated, metal-jacketed cables, one from the core exit thermocouples and the other to the containment electrical penetration assemblies. The connector is exposed to the containment atmosphere.

CE's original CET connector environmental qualification is based on qualification report CE NPSD-230-P, "Class 1E Qualification of the Core Exit Thermocouple - Mineral Insulated Cables," April 1983 (Proprietary). It covers testing of 8-pin size 20 connectors in the CIR series manufactured by Veam Division of Litton Systems, Inc. used with cable manufactured by Electronic Resources Division Inc. of Whittaker Corp. Size 16 connectors are similar. This report had been reviewed previously during EQ inspections of licensees, and it was reviewed again during this inspection.

Report CE NPSD-230-P documents moisture ingress into the connectors tested. Post-test inspection is reported showing that the face seal gasket was completely degraded, and that the elastomeric inserts used to support pins and sockets were covered with a flaky gray material produced by interaction of the elastomer with borated water which "had obviously leaked into the connector."

A variety of anomalous behaviors occurred during the LOCA test. In evaluating them the inspector reviewed the detailed test report TR-ESE-495 (Proprietary) dated January 7, 1983 and original strip chart recordings of the thermocouple signals. Of particular concern was the impact of periodic cable insulation resistance measurements at 50 vdc, which appeared to charge the cables and cause spurious signal spikes for as long as an hour after voltage application. These tests also decalibrated the recorder amplifiers; the recorder was in fact replaced with a different type midway through the 30-day LOCA test. Thermocouples showed different time responses to temperature changes because of varying thermal masses and junction types (grounded and ungrounded); when intervals between actual temperature changes were rather small, the slowest response thermocouple signal deviated considerably from others because of obvious time lags. The temperature cycling was reduced by more closely controlling the temperature in the reference oven.

Anomalies were processed for these abnormalities as required by test procedure 00000-ESE-391 Rev. 01 (Proprietary) dated September 1, 1982. In particular, Notice of Anomaly No. 2 dated October 14, 1982 states the following reason for apparent thermocouple signal errors greater than the ±22°F allowed by the test procedure (of the order of 45°F):

> "It is proposed that Meger (sic) testing prior to initiating transients created a capacitor. When the specimen was exposed to a steam environment the Litton connector leaked and humidity or wetting of the pins created leakage paths for stored charge."

Qualification report CE NPSD-230-P also discussed this anomaly, concluding that the temporary perturbation of the thermocouple signals caused by insulation resistance checks should not be considered in the evaluation of CET system performance. Strip chart excerpts demonstrating the transient behavior are included in CE NPSD-230-P. Bench tests with line resistances simulating 180-190 foot Chromel/Alumel cables and a simulated 1000 ohms connector resistance further showed acceptable performance.

Qualification report CE NPSD-230-P and its supporting documentation thus convey the following representations concerning LOCA qualification of the Litton-Veam connectors in the CET system. The connectors are environmentally qualified for an overall system accuracy of ±22°F; the connectors experienced moisture ingress and disintegration of the face seal gasket; and transient spikes as large as 45°F had been evaluated as a test anomaly. This report was issued in April 1983.

The attention of the CE engineers then apparently focused on the Reactor Vessel Level Monitoring (RVLM) system addressed in Section E.3 of this inspection report. The RVLM system feeds heater current through the connectors, so the low insulation resistance of a moisture filled connector was recognized as unacceptable. CE then proceeded with activities intended to develop a "dry" connector for RVLM use, as described in Section E.3 below.

In October 1985 CE formed a small task force to investigate the possibility that moisture ingress into the CET connectors could generate a "battery effect," such that the LOCA test 45°F spikes

may have been real rather than anomalous performance. The task force was unable to rule out the possibility of a battery effect, and in December 1985 recommended exploratory testing. Such testing was performed late in 1986 as reported in NPSD-230-P Supplement 1 (Proprietary), undated but distributed to customers by letters dated May 21, 1987. A summary of the testing was also transmitted to the NRC by letter dated May 22, 1987. In the "battery effect test" connectors in thermocouple circuits were thoroughly wetted. The signal emfs from the thermocouples in a 400°F oven were continuously monitored while the connectors were dried in a 200°F oven, so that the effects of varying amounts of moisture down to and including a very thin film could be observed. To maximize errors, a long run of cable was coiled in the oven (1000 ohms resistance from thermocouple to connector).

The supplemental test showed that moisture ingress into the Litton-Veam connector can produce spurious signal spikes of 45°F in the CET system, as had been observed in the 1982 testing but attributed to anomalies. The May 1987 letters cited in the previous paragraph provided analyses to show the acceptability of this error in CET systems. Another CE letter issued during this inspection - November 9, 1988 - provides additional data analysis to show that in fact the originally claimed ±22°F is indeed valid. The inspector reviewed all of this material and found it to be satisfactory subject to ongoing review of the NRC Plant Systems Branch to address (1) the acceptability of the quoted ±45°F error and (2) the basis for subsequently reducing the error to ±22°F.

The inspector concludes that CE's activities with respect to the CET portion of the allegation appear adequate.

3. Reactor Vessel Level Monitoring System

The CE Reactor Vessel Level Monitoring (RVLM) system uses a combination of normal and Heated Junction Thermocouples (HJTCs) to monitor vessel level, on the premise that an HJTC will be cooler if surrounded by water than by steam. As noted above, moisture ingress into HJTC connectors is unacceptable because of the heater current transmitted through the connectors.

CE developed two alternatives to the Litton-Veam connector for RVLM use. Each uses an inorganic Grafoil gasket in place of the Litton-Veam gasket. Grafoil is a graphite gasket material manufactured by Union Carbide Corporation.

One of the alternatives is a connector supplied by the CET and RVLM cable manufacturer, Electronic Resources Division Inc. of Whittaker Corporation. This is a conventional type threaded connector that applies a positive loading force to the Grafoil gasket. The inspector reviewed the qualification documentation for this connector, including report CE NPSD-275-P "Summary Report: Class 1E Qualification Test of the Electronic Resources Division (ERD) Electrical Connectors and Mineral Insulated Cable" dated November 1984 (Proprietary). The test specimens were models 16-26-00200-2 and -4. Both CET and RVLM applications were addressed. The documentation was found satisfactory. With respect to gasket compression by repeated connects/disconnects, the test specimens were subjected to ten mechanical cycles. One test specimen admitted water during the LOCA test. This deviation was addressed in the qualification report and in NTS Hartwood Formal Report No. 558-1572, "Nuclear Qualification Testing of Class 1E ERI Mineral Insulated Cable and Connector Assemblies" dated October 31, 1984 (Proprietary) which was selectively reviewed by the inspector. Change of Procedure No. 6 and Notice of Deviation No. 8, both contained in Appendix B of the test report, document that the leaking connector had been found loosely mated during incoming inspection after the gamma irradiation test preceding the LOCA test, and was retorqued. Since the other test specimens did not leak during the 30-day LOCA test or a subsequent 24 hour submergence test, it was assumed that the leakage was caused by shipping or handling damage to or from the irradiation test. It appears reasonable to assume type qualification for ten cycles aging.

The other alternative developed by CE consists of modifying the Litton-Veam connector to (1) replace the elastomeric gasket with a Grafoil gasket and (2) add provisions for loading the Grafoil gasket with a threaded arrangement that is tightened on the connector plug after the bayonet joint between plug and receptacle is engaged.

The connector is seal-welded to the metal-sheathed cable in order to keep its mineral insulation dry. The retrofit modification was designed to be added to existing cables with integral Litton-Veam connectors, whereas the ERD connector could only be used for new manufacture. Environmental qualification of the retrofit was based on the ERD connector qualification testing supplemented by analysis addressing differences in the two designs.

The CE qualification report for the retrofit is CE NPSD-296-P, "Qualification Summary Report for the Grafoil Gasket Retrofit for Litton Connectors," May 1985 (Proprietary). This report supplements the RDE connector qualification report by providing descriptions and analyses of differences in the two designs and by describing several tests performed on the retrofit design. LOCA was addressed by analysis of torques, densities, seal pressures, and the like. Radiation and thermal aging were also addressed by analysis. Tests addressed mechanical cycling (Grafoil gasket and Belleville spring), thermal cycling, vibration aging, and seismic. One of the four test specimens had received radiation and temperature aging. The inspector concluded that qualification of the retrofit design is based on a reasonable combination of test and analysis as allowed by 10 CFR 50.49 paragraph (f)(2).

The RVLM portion of the allegation deals with the retrofit modification of the Litton-Veam connector. If the Grafoil gasket is too thin, the clamping device may simply achieve a metal-to-metal bottoming instead of loading the gasket. A possible method of assuring that the gasket is thick enough is to ensure that the travel of the threaded clamp before bottoming is less with the plug mated to its receptacle than with the unmated plug. In this regard the inspector questioned CE personnel and also reviewed installation instruction 0000-CCE-GL80-14, Revision 02, "Guidelines for Litton Connector Grafoil Gasket Retrofit," dated November 11, 1985. CE stated that control of gasket dimensions and density assure proper fit. In addition, step 8.1.3. of the installation instructions, which addresses the bayonet action of joining the plug and receptacle, states "Observe a tactile click as the coupling nut rotates past cams." The tactile click occurs when the grafoil gasket partially relieves compression as the bayonet coupling action is completed. Finally, if the gasket is loaded a gap is evident between the clamp locking sleeve and coupling sleeve (the two large, external pieces of the clamp).

Although the gap is not mentioned in the installation instruction, it is shown in Figure 1 of the instruction. With respect to compression of the gasket during use, the type qualification limit of ten disconnect cycles and the information described above appear adequate. However, modification of the installation manual to address the gap to specify a minimum dimension, and to alert plant maintenance personnel to address the tactile click and minimum gap each time a connector is reconnected, would appear to be reasonable enhancements.

The inspector concludes that CE's activities with respect to the RVLM portion of the allegation appear reasonable. Adequate qualification basis appears to exist for both the retrofit and ERD connector designs. The retrofit installation instructions also appear to be adequate to ensure Grafoil gasket loading, although they could be enhanced as noted in the previous paragraph.

The inspector noted that various other designs had been considered prior to CE's decision to proceed with the selected retrofit designed. These were not reviewed since only the retrofit design considered was supplied to customers and claimed to be environmentally qualified. Comparison of alternate designs was outside the scope of the inspection except for the similarity argument described above in qualification of the retrofit design.

4. Conclusions

Both parts of the allegations clearly address legitimate technical concerns. The inspector concludes that in each case CE has acted in a reasonable manner and has adequately addressed the area of concern. No violation or nonconformances were identified, either directly related to the allegation or in other inspected areas.

Table I of this inspection report shows the domestic plants using Combustion Engineering CET or RVLM systems, together with the type of connector believed used in each case.

The following plants are also known to have used Litton-Veam connectors in applications other than CE-supplied CET or RVLM systems:

Plant	Docket No.	Litton-Veam Connector Usage
Haddam Neck	50-213	head vent solenoid valves
Main Yankee	50-309	non-CE CET system
Millstone 3	50-423	transmitter seals, other

Documentation reviewed during this inspection clearly does not establish LOCA qualification of unclamped Litton-Veam connectors for any application other than very low voltage thermocouple circuits. Further, the connector behavior during the LOCA tests indicates unsuitability for any other use requiring LOCA qualification.

F. PERSONS CONTACTED:

W. D. Mawhinney, Vice President, Nuclear Quality Systems
*J. J. Holloway, Vice President, Nuclear Services
*T. R. Swift, Manager, Nuclear Quality Systems
*W. A. Goodwin, Director, Nuclear Quality Systems
*S. A. Toelle, Manager, Licensing
*J. M. Burger, Manager, Mechanical Systems
E. A. Siegel, Manager, Plant Structures
*J. M. Betancourt, Senior Consulting Engineer, Licensing
*C. M. Molnar, Senior Consulting Engineer, Licensing
*D. M. Amidon, Engineer, Quality Programs

*Attended exit meeting.

TABLE I. PLANT USEAGE OF CE CONNECTORS
IN CET AND RVLM SYSTEMS

Plant(s)	Docket No.(s)	CET System Conn. Type	RVLM System Conn. Clamped
Braidwood 1, 2	50-456, 457	N/A	only HJTCs
Byron 1, 2	454, 455	N/A	only HJTCs
Catawba 1, 2	413, 414	Litton-clamped	N/A (DP)
Comanche Peak 1, 2	445, 446	N/A	Yes
D.C. Cook 1, 2	315, 316	ERD	N/A (DP)
Farley 1, 2	348, 364	ERD	Yes
Fort Calhoun	285	Litton	Yes
Haddam Neck	213	Litton-clamped	N/A
Indian Point (future)	247	ERD	N/A
McGuire 1, 2	369, 370	Litton-clamped	N/A (DP)
Millstone 2	336	Litton	Yes
Millstone 3	423	N/A	Yes
Palo Verde 1, 2, 3	528, 529, 530	Litton	Yes
Prairie Island 1, 2	282, 306	Litton	N/A (DP)
St. Lucie 1, 2	335, 389	Litton	Yes
Salem 1, 2	272, 311	Litton	Yes
San Onofre 2, 3	361, 362	Litton	Yes
South Texas 1, 2	498, 499	N/A	only HJTCs
Summer	395	Litton	N/A (DP)
Trojan	344	ERD	? (DP)
Turkey Point 3, 4	250, 251	Litton	Yes
WNP-3	508	Litton	Yes
Zion 1, 2	295, 304	ERD	Yes (DP)

N/A designates system and connectors not supplied by CE.
DP designates differential pressure level instrument.

ORGANIZATION: COPES-VULCAN
LAKE CITY, PENNSYLVANIA

REPORT NO.: 99900080/88-01	INSPECTION DATE: November 7-11, 1988	INSPECTION ON-SITE HOURS: 68

CORRESPONDENCE ADDRESS: Copes-Vulcan
Martin and Rice Avenues
Lake City, Pennsylvania 16423

ORGANIZATIONAL CONTACT: Mr. Dale Mays, QA Director
TELEPHONE NUMBER: (814) 774-3151

NUCLEAR INDUSTRY ACTIVITY: The majority of Copes-Vulcan's current work is in the design and manufacturing of valve assemblies used in military nuclear applications and the manufacture of piece-part valve assembly replacement parts for commercial nuclear applications (e.g., valve trim).

ASSIGNED INSPECTOR: _____ 1/17/89
J. J. Petrosino, Reactive Inspection Section No. 1 (RIS-1) Date

OTHER INSPECTOR(S): J. A. Gavula, NRC Region III, Division of Reactor Safety
T. L. Tinkel, NRC Consultant

APPROVED BY: E T Baker 1/17/89
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: This inspection was conducted as a result of piping systems at Prairie Island that were found to be outside of their safety analysis report design limits.

PLANT SITE APPLICABILITY: All plants.

A. VIOLATION:

Contrary to Section 21.21 "Notification of failure to comply or existence of a defect" of 10 CFR Part 21, Copes-Vulcan (C-V) failed to notify all of its commercial nuclear valve assembly customers of incorrect valve assembly center-of-gravity and/or weight information on all of its drawings for valves supplied to customers prior to November 28, 1979. If the information had been adequately evaluated at the time it would have resulted in a 10 CFR Part 21 report (88-01-01).

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

B. NONCONFORMANCE:

Contrary to Criterion III "Design Control," of Appendix B to 10 CFR Part 50 , Copes-Vulcan failed to ensure that its center-of-gravity and weight determinations were checked for technical adequacy by someone other than the individual who performed the work (88-01-02).

C. UNRESOLVED ITEMS:

None.

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not reviewed during this inspection.

E. OTHER COMMENTS:

1. Entrance and Exit Meetings

The NRC inspector informed the Copes-Vulcan representative of the scope of this inspection during the entrance meeting on November 7, 1988 and summarized his inspection findings, observations, and NRC staff concerns during the exit meeting. Copes-Vulcan committed to identify all of its customers who may be affected by certain incorrect valve assembly weight and/or center-of-gravity information and notify each so that they may cause an evaluation to be performed or to notify the applicable NRC licensee facilities.

2. Background of Weight and Center-of-Gravity Issue

The NRC concern was identified in part during a 1988 Northern States Power Company (Prairie Island) engineering review regarding the applicability of the seismic issues discussed in NRC Bulletin 79-14.

During the Prairie Island engineering review of its seismic piping system analyses, design basis information for one specific C-V valve could not be found. As a result, the engineering consultant, Fluor Daniel (F-D), requested the valve design information from C-V. Prior to F-D receiving the valve design information from C-V, the original design basis documents were found by F-D, who compared the information to the original seismic analyses input and found the data to be consistent. However, after receiving and reviewing the new C-V design information, F-D noted that it contained different, nonconservative valve assembly weights and center-of-gravity (CG) data. As an example, the original small bore C-V valve assembly information showed a specific valve assembly weight to be 170 pounds and its CG to be just above 5 inches from the datum point. Conversely, the new C-V valve assembly information indicated the same valve to weigh 240 pounds and have a CG of 19 inches from the datum point.

The new C-V supplied weights and CG data were then inserted into the original stress calculation at Prairie Island. As a result of the new analyses, several Prairie Island piping system stresses were found to exceed the Final Safety Analysis Report (FSAR) requirements. In one instance stresses increased from approximately 32,000 psi to approximately 161,000 psi.

The NRC Vendor Inspection Branch reviewed the circumstances surrounding the issue. The inspection identified, in part, that C-V valve assembly weights and center-of-gravity data given on customer design drawings shipped prior to November 28, 1979 could be incorrect. There is no one reason for the incorrect values. However, a few contributors are the failure to include the mass of the operator in the weight and CG calculation, using the CG for the valve and bonnet only, and not accounting for a forged valve body instead of a cast valve body.

Note: Subsequent to the NRC inspection at C-V, the Region III NRC inspector identified that a similar condition was present at D. C. Cook Units 1 and 2. The overstress condition exceeded the D. C. Cook FSAR design limits.

3. Potentially Reportable 10 CFR Part 21 Items

Section 21.21 "Notification of Failure to Comply or Existence of a Defect," of 10 CFR Part 21 requires in part, that each individual, corporation, partnership or other entity subject to Part 21 adopt appropriate procedures to provide for evaluating deviations or

informing the licensee or purchaser of the deviation so that the licensee or purchaser may cause the deviation to be evaluated.

Contrary to this, it was identified that C-V became aware of problems with its center-of-gravity data and/or weights associated with its valve assemblies that were shipped to numerous nuclear power plants prior to November 28, 1979, and failed to either evaluate the deviation or to notify all the applicable purchasers. This issue is identified in Violation 88-01-01.

Based on a review of documents and discussions with C-V personnel, it appears that the issue of incorrect CG data and/or weights was identified in November of 1979 to a C-V sales agent in Alabama by the Farley Nuclear Plant based on a CG and/or weight concern regarding the C-V valve assembly drawings for Farley. The C-V sales agent then reviewed the C-V drawings and submitted them to the Farley architect-engineer, Bechtel Power Corporation, who then contacted the original purchaser, Westinghouse Corporation, to advise them that they had received uncontrolled revised drawings and asked Westinghouse for additional information. At this point Westinghouse (W) notified C-V that any further verbal or written correspondence or information of a technical nature must go through W. Bechtel, at this time, had identified approximately 42 questionable C-V drawings to W; however, it is unclear as to what additional W or Bechtel actions were performed based on information available at C-V.

4. Design Control

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires, in part that design control measures will provide for verifying or checking the technical adequacy of the design. The verifying or checking process will be performed by personnel other than those who performed the original design.

During a review of some C-V calculations for valve assembly CG data and associated weights it was noted that the C-V practice is to have the same engineer who performs the CG and weight calculations verify his own work. This issue is identified in Nonconformance 88-01-02.

Weight and CG calculations that were reviewed are as follows:

VALVE ID	DRAWING NO.
3/4-IA58-R	L-140209
1-IA56-RE	L-137918
1-IA56-RE	L-137968

VALVE ID	DRAWING NO.
2-IA58-RE	L-138049
1-ID56-R	B-132315
1-ID58-R	B-132318
2-ID58-R	B-132338

Additionally, some C-V valve/operator assembly analyses were reviewed. These analyses evaluated various critical valve sections for assumed horizontal and vertical static accelerations. Fundamental frequencies were also determined to verify non-participation of the valve/operator assembly during a seismic event. Simplified analytical techniques were used with appropriate assumptions. The CG's and weights of the operators, as well as other valve accessories, were properly included in these evaluations. The previously discovered issue of non-representative weight and center of gravity calculations for the valve discussed in item E.2 above was not carried over into the seismic evaluations. Some of the C-V seismic calculations reviewed were:

a. C-V calculation No. 10.3.119, dated March 31, 1977, which included valves: 3IA58RGP, 2IA58RG, 2RA56RE, 8RA36RG, and 14GM48SEZ.

b. C-V calculation No. 10.3.132, dated October 20, 1978, which included valves: 3/4 IA58RE, 1WA42DD, 1IA56RE, 1IRA58RD, 1IA58RE, 1IA38RES, 2RA42RD, 2RA56DD, 2RA56RE, 2IA58RG, 3IA58RGP, 3WA42RE, and 4RA58RGA.

c. C-V calculation No. 10.3.337, dated February 14, 1986, which included valve: D-100-160, 2 inch, class 1500.

d. C-V calculation (No ID), dated May 25, 1978, which includes valves: 2RA42DD and 2IA58RE.

5. Customer Notifications and Scope

It was determined by the NRC inspector and the C-V representative that C-V had not notified any of its customers, other than Westinghouse, of the problem regarding incorrect weight and CG data. The time frame for the problem has been established as prior to November 28, 1979. The scope of the problem includes all C-V safety-related valve assembly drawings that were provided to customers prior to November 29, 1979 that included weights and/or CG information. The significance of the problem was obvious at the Prairie Island nuclear plant where a reanalysis of some

safety-related piping systems in 1988 using the correct weights and CG's revealed piping system stresses that exceeded the design limits.

It was noted that the C-V staff implemented corrective actions in 1979-1980 on all of its in-house drawings after completion of the C-V and W meeting in November of 1979. The NRC inspector determined that the W and C-V meeting occurred on November 28, 1979 at the C-V facility in Pennsylvania. It was at this W documented meeting that W appears to have been formally notified of the problem. However, W appears to be the only C-V customer that was made aware of this deviation.

As a result of discussions between C-V and NRC staff, C-V has committed to compile a list of all its customers, with the exception of W, since W is already aware of the problem, and to notify each customer within 30 days of the NRC's exit meeting.

Following is a partial list of licensees and the type of valves supplied based on information available from C-V's valve user lists:

Acrynonyms

FW - Feedwater Valve
AO - Air Operated Valve
MO - Motor Operated Valve
FWB - Feedwater bypass valve
SV - Sampling Valve
MT - Manual Throttling
SD - Steam Dump
BOP - Balance of Plant

Plant	Known Valve Application
Beaver Valley 1	FW, AO, MO, SV, BOP
Beaver Valley 2	FW, AO, SV, BOP
Bellefonte	FW
Braidwood 1 & 2	AO, MO, SV, MT, SD, BOP
Browns Ferry 1	BOP
Byron 1 & 2	AO, MO, SV, MT, BOP
Callaway	FW, AO, FWB, BOP
Calvert Cliffs 1 & 2	SD, BOP
Comanche Peak 1 & 2	FW, AO, SV, BOP
Comanche Peak 2	FW, AO, SV, BOP
Cook 1 & 2	AO, BOP
Crystal River 3	BOP
Davis-Besse	BOP
Diablo Canyon 1 & 2	FW, AO, SD, BOP
Dresden 2 & 2	FW, BOP

Plant	Known Valve Application
Farley 1 & 2	AO, BOP
Fort Calhoun	BOP
Ginna	FW, AO, BOP
Harris	FW, AO, SV, BOP
Indian Point 2 & 3	FW, AO, BOP
Kewaunee	FW, SD
Limerick 1 & 2	BOP
Maine Yankee	FW, BOP
McGuire 1 & 2	Specific Application Unknown
Millstone 1, 2, & 3	FW, BOP
Monticello	FW
Nine Mile Point 1	FW, BOP
Nine Mile Point 2	FW, BOP
North Anna 1 & 2	FW, AO, SD, BOP
Palisades	FW, BOP
Perry	FW, BOP
Pilgrim	FW
Point Beach 1 & 2	FW, AO, BOP, FWB
Prairie Island 1 & 2	FW, AO, BOP
Quad Cities 1 & 2	FW, BOP
River Bend 1	FW, BOP
Robinson 2	FW, AO, FWB, BOP
Salem 1 & 2	FW, AO, BOP
San Onofre 2 & 3	FW
Seabrook	FW, AO, SV, BOP
Sequoyah 1 & 2	FW, AO, BOP
Shoreham	BOP
South Texas 1 & 2	FW, AO, FWB
St. Lucie 1	FW, BOP
St. Lucie 2	Specific Application Unknown
Summer 1	AO, BOP
Surry 1 & 2	FW, AO, SD, BOP
Susquehanna	BOP
Trojan	AO, SD, BOP
Turkey Point 3 & 4	FW, AO, FWB, BOP
Vogtle 1 & 2	MO, AO, SV, BOP
Watts Bar	SV, AO, SD, BOP
Wolf Creek	FW, AO, FWB, BOP
Zion 1 & 2	AO, BOP

6. Sliding Stem Friction Forces

Recently, NRC Information Notice (IN) 88-94, was issued regarding valve stem packing friction forces. Fisher Controls notified the NRC of its concern with valve stem packing friction forces which increase due to use of graphite or graphite/asbestos packing in valves that were originally supplied with teflon packing, and of licensees replacing preformed valve stem packing with graphite ribbon packing. One aspect of this inspection was to verify whether C-V explicity accounted for valve stem friction forces. It was found that C-V does account for the added forces; therefore, this area was classified as satisfactory.

F. PERSONNEL CONTACTED:

NAME	TITLE
Dale Mays	QA Manager
Tim Kunkle	Product Design Manager
Chuck Dundon	Sr. Contract Engineer
Allan Shea	Sr. Design Engineer
Norman Mattson	Valve Contract Manager
T. J. Billings	NDT Specialist
J. R. Scarpelli	QA Supervisor
J. Clifford	Sales Engineer

ORGANIZATION: EBASCO SERVICES INCORPORATED
NEW YORK, NEW YORK

REPORT NO.: 99900505/89-01	INSPECTION DATE: January 23-27, 1989	INSPECTION ON-SITE HOURS: 42

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

ORGANIZATIONAL CONTACT: Mr. Sal Sparacino, Manager, QA Engineering
TELEPHONE NUMBER: (212)839-2457

NUCLEAR INDUSTRY ACTIVITY: Design, procurement, and quality assurance activities for several nuclear projects.

ASSIGNED INSPECTOR: Robert Pettis Jr. 3/22/89
R. L. Pettis, Jr., Reactive Inspection Section No. 1, Vendor Inspection Branch — Date

OTHER INSPECTOR(S): T. Tinkel, Consultant

APPROVED BY: Edward Baker 3/23/89
E. Baker, Chief, Reactive Inspection Section No. 1, VIB — Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 50, Appendix B, Ebasco Topical Report ETR-1001 "Nuclear Quality Assurance Program Manual," and 10 CFR Part 21.

B. SCOPE: Follow-up inspection to review records, procedures and interview personnel regarding Ebasco's procurement activities at the Waterford 3 Steam Electric Station (SES) during the period 1981-1983.

PLANT SITE APPLICABILITY: Waterford 3 SES (50-382).

A. VIOLATION:

Contrary to Section 21.31 of 10 CFR Part 21, Ebasco Services Incorporated placed two purchase orders (POs), WP3-13587 and WP3-137680 for safety-related electrical material for Waterford 3 SES without specifying the requirements of 10 CFR 21 on the purchase order to the supplier (89-01-01).

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

B. NONCONFORMANCE:

Contrary to Ebasco Services Incorporated Procedure No. ASP-I-5, Issue "E," dated April 20, 1978, "Quality Assurance Evaluation of Suppliers," Ebasco placed 35 safety-related POs with the Gismo Company without the supplier being on Ebasco's Approved Suppliers List (89-01-02).

C. UNRESOLVED ITEM:

The NRC inspector was unable to review supplier evaluations performed by Ebasco New York Quality Assurance personnel for suppliers previously rejected by Ebasco but later approved and placed on Ebasco's Approved Suppliers List (89-01-03).

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not Applicable

E. INSPECTION FINDINGS AND OTHER COMMENTS:

The inspection was performed as a follow-up to an earlier inspection performed at Ebasco, New York (NRC Inspection Report No. 99900505/88-01), which was prompted by a complaint of discrimination filed with the Department of Labor (DOL) by a former Ebasco employee and referred to the NRC. The complaint alleges that Ebasco had violated 10 CFR 50, Appendix B, and its internal procedures in the evaluation and approval of certain material and component suppliers for the Waterford 3 SES. Specifically, the alleger claims that certain suppliers were found unsatisfactory for supplying safety-related components and materials in the 1981-83 time frame. Further, the alleger claims, because Ebasco had already placed POs with these suppliers and any delays could impact unfavorably on the Waterford 3 SES construction schedule, Ebasco sent other auditors (unqualified) to evaluate these suppliers who subsequently found them to be satisfactory.

To determine the validity of the above allegations, an NRC inspection was performed at the Ebasco New York office in August 1988 to review the pertinent records and related procedures. Immediately prior to the inspection, the NRC inspectors met with the alleger to identify more clearly the specific records and areas of concern. The results of this inspection are documented in NRC Inspection Report No. 99900505/88-01, dated September 28, 1988.

As a result, it was further determined by the NRC that an additional inspection be performed at the Waterford 3 SES to review Ebasco's procurement documents generated during this period in order to determine the safety-related significance, if any, associated with the alleger's concerns since these documents were not available during the August 1988 inspection. This report documents that review.

1. Review of the Ebasco New York and Waterford 3 SES Approved Suppliers Lists (ASL)

   It was alleged that Ebasco maintained a "supplementary Approved Supplier List (ASL)," known as the "Waterford 3 list," that included suppliers who could not be approved due to the lack of a 10 CFR 50, Appendix B quality assurance program. However, due to the exigencies of the construction schedule, these suppliers were nevertheless utilized to provide safety-related components and materials for the Waterford 3 SES project.

   Background

   The NRC inspectors reviewed the Ebasco files on-site that contained the list of approved suppliers used for the Waterford 3 SES project. It was determined that Ebasco maintained two separate lists for the reason that suppliers contained on the Waterford 3 SES project list were unique to the project, and that the cost incurred by Ebasco to audit these suppliers would not be shared by other Ebasco projects. The NRC inspectors reviewed both ASLs covering the period 1978-1984 and determined that suppliers utilized for safety-related materials at Waterford 3 SES were, with several exceptions noted, qualified to be included on Ebasco's ASL. The Ebasco New York ASLs and supplements reviewed were as follows: December 31, 1981; March 31, 1982; May 3, 1982; June 1, 1982; July 19, 1982; August 16, 1982; September 30, 1982; December 30, 1982; February 1, 1983; March 31, 1983; June 30, 1983; August 1, 1983; September 30, 1983; and December 30, 1983. The Ebasco Waterford 3 SES ASLs reviewed were

as follows: January 6, 1982; June 3, 1982; November 22, 1982; March 9, 1983; June 17, 1983; September 8, 1983; and December 15, 1983. Safety-related suppliers are those suppliers deemed capable by Ebasco to provide permanent plant material and equipment in accordance with the guidelines established by the Ebasco Licensing Department to be ASME Code Class 1, 2, 3, Seismic, or Electrical Class 1E. Nonsafety-related applies to all other permanent plant items.

According to Paragraph 7.1.3 of Ebasco Procedure No. QAP-9, Revision 1, dated September 29, 1978, "Quality Assurance Vendor Evaluations," qualified suppliers are placed on the ASL by an Ebasco review of their Quality Assurance (QA) Program. An initial or pre-award audit is then conducted for those first time suppliers placed on the ASL, with a follow-up audit performed three years later, as determined by the Ebasco Vendor Evaluation Group Leader. Suppliers included on the ASL, according to Paragraph 10.1.1, are those whose documented quality program and facility are considered "Satisfactory" in accordance with Ebasco Procedure No. QAP-9, Paragraph 7.7.1 and therefore authorized to receive a nuclear safety-related PO. At the end of the suppliers periodic audit date, the supplier must be reaudited or dropped from the ASL.

It was noted that not all suppliers listed on the Ebasco Waterford 3 SES ASL were required to implement or maintain a full 10 CFR 50, Appendix B QA program. In some cases, the quality requirements applicable to an order were defined in a procurement specification that was referenced in the basic PO to a supplier. The range of supplier quality programs required by these procurement specifications varied from some that were very limited to some that essentially required a complete 10 CFR 50, Appendix B program. Examples of suppliers with whom Ebasco used a procurement specification to define supplier QA program requirements included: Appleton Electric, Wheatland Tube, Picoma Industries, Conduit Pipe Products, Crouse Hinds Company, Gismo Company, and the O. Z. Gedney Company. All suppliers were included on the Waterford 3 SES "Supplemental" ASL and were scheduled for reaudit in early 1983. In each case, POs reviewed were for safety-related electrical material (i.e., galvanized rigid steel conduit, fittings, couplings and related interface material) and were placed by Ebasco during the 1978-1984 period time frame. Those POs referenced Ebasco project specifications which in-turn referenced the QA requirements in accordance with Ebasco Specification 860-80, "QA Requirements for Suppliers of Safety-Related

Equipment and Services," or Specification 860-72, "Quality Control Requirements for Suppliers of Equipment and Services." Project specifications referenced for the POs reviewed were as follows: LOU-1564.065, .066, .068, .124F, .249A, .249D, .249R, .270, and .403.

Specification 860-72 outlines the QA requirements (which resemble a 10 CFR 50, Appendix B program) for nonsafety-related applications and then invokes 10 CFR 50, Appendix B, for all safety-related applications, whereas 860-80 imposes the requirements of 10 CFR 50, Appendix B, and 10 CFR Part 21. In five of the nine LOU Specifications referenced, either 860-80, or 860-72 specifications were imposed with the balance referencing only a paragraph requiring the supplier to submit a controlled copy of his Quality Control (QC) manual with his bid for Ebasco review. Prior to contract award, a review of the supplier's facility would be performed by Ebasco to evaluate the supplier's quality program.

Ebasco's acceptance of a "Limited QA" program for suppliers mentioned above raised concerns with the alleger that Ebasco was approving suppliers without their having a full 10 CFR 50, Appendix B, QA program. It should be noted that suppliers not equipped to handle a full 10 CFR 50, Appendix B, QA program, due to the nature of the item being supplied to Waterford 3 SES (i. , galvanized rigid conduit) were required to have a "limited" QA program incorporating only those items of Appendix B applicable to the product furnished. In addition, Ebasco invoked by specification various technical and quality requirements (codes, standards, receipt inspection, tests, etc.) necessary to achieve compliance with the intent of Appendix B.

This methodology is discussed in the Introduction to 10 CFR 50, Appendix B which states, "the pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those stuctures, systems, and components." In addition, Criterion II, "Quality Assurance Program," states that the program shall provide control to an extent consistent with their importance to safety. Therefore, the "Limited QA" program applied to these suppliers in the late 1970 time frame appeared to meet applicable regulatory requirements considering the relative safety significance of the material purchased. The NRC inspectors did not review each supplier's QA file to determine the extent that their QA program complied with the applicable portions of Appendix B.

It was noted during the inspection that 35 safety-related POs were placed by Ebasco with the Gismo Company for various electrical interface material without the supplier's appearing on the ASL. These POs were placed during the period May 1978 through June 1980. A review of Ebasco's ASL also indicated that Gismo appeared on the ASL prior to and after the time period referenced. As a result, Nonconformance 89-01-02 was identified during this part of the inspection.

2. Purchase Order Review - 10 CFR Part 21

During the NRC review of safety-related POs generated by Ebasco, it was observed that the majority of the POs invoked the requirements of 10 CFR 21 on the supplier. However, two POs failed to specify that Part 21 applied: PO WP3-137680, dated December 15, 1983, to Appleton Electric and WP3-13587, dated December 4, 1983, to Crouse Hinds (both for safety-related electrical interface material). As a result, Violation 89-01-01 was identified during this part of the inspection.

3. Review of Ebasco Performed Supplier Evaluations

In a letter dated March 8, 1983 from Mr. M. Brooks (Ebasco Site) to Mr. R. Williams (Ebasco, New York), it was discussed that several "key" suppliers of safety-related electrical material, soon due for reaudit have been rejected by New York Quality Assurance due to the suppliers' inability to comply with 10 CFR 50, Appendix B requirements. The suppliers referenced were the following: Conduit Pipe Products, Picoma Industries, Wheatland Tube Company, O. Z. Gedney Company, Crouse Hinds Company, Electromark, Gismo, and Appleton Electric. A review of the Ebasco Waterford 3 SES ASL, dated December 15, 1983 indicated that all eight suppliers were included on the ASL. The basis for placing these suppliers on the ASL was supplier audits performed by Ebasco New York auditors during the period March - May 1983. As a result, these suppliers were deemed satisfactory and were placed on the ASL despite earlier rejection of these suppliers.

The NRC inspectors were unable to review the Ebasco audits performed since the files are located at Ebasco's New York office. As a result, Unresolved Item 89-01-03 was identified and will be reviewed during a future inspection at Ebasco New York.

4. Review of the Rotohammer Company

Information obtained by the NRC during the August 1988 inspection of Ebasco indicated that safety-related valve stem extension material was purchased from the Rotohammer Company who, at the time, was not an approved supplier. In addition, this breakdown was viewed by an Ebasco employee as being reportable to the NRC under the provisions of 10 CFR 21; however, the nonconformance reports generated by Ebasco indicated "not reportable." In order to review this concern, the following Ebasco documentation was reviewed by the inspectors:

a. Purchase Order NY 403568. This was the initial Ebasco PO placed with Rotohammer on January 11, 1977. A review of this purchase order indicated the following:

(1) The PO was placed for various quantities of valve stem extension equipment which was classified as nonsafety-related.

(2) The PO invoked Ebasco specification LOU-1564.124, Revision 2, dated January 11, 1977. This specification applies to valve stem extensions for non-nuclear seismic and non-seismic valves. Paragraph 5 addresses seismic considerations and states that seismic I valves shall be considered non-nuclear, nonsafety, but shall be seismically supported. The specification also invokes Ebasco specification 860-72, "Quality Control Requirements for Suppliers of Equipment and Services," which states that this document covers quality control requirements applicable to suppliers of equipment and their subvendors. In addition, 860-72 states that for safety-related material and services, the vendor's quality control program shall also meet the requirements of 10 CFR 50, Appendix B.

(3) Fourteen supplements were issued against the initial PO by Ebasco New York (Supplement 1, dated May 11, 1977 and Supplement 14, dated May 3, 1984). None of these supplements identified the material as safety-related.

b. Ebasco Nonconformance Reports (NCRs) W3-3828, W3-3930, and W3-3995 were reviewed which were generated against material ordered under Ebasco PO NY 403568.

(1) NCR W3-3828, dated May 19, 1982. A review of this NCR indicated the material was shipped to Waterford in 11 separate lots and each documented by a separate Material Receiving Report (MRR). The description of the nonconformance stated that all material was manufactured and supplied by Rotohammer and received between March 20, 1981 to May 19, 1982, during which time Rotohammer was not an approved supplier since their qualification expired in March 1980. The temporary disposition of the material was for Ebasco QC to confirm that the material was in an acceptable status at the time of issue and then to process the material on a conditional release basis. The recommended disposition was to requalify the supplier, review past history from March 1980 to June 1982 to determine any negative trends, and to obtain Certificates of Compliance (CoC) to the specifications. The NCR also indicated that the item was not reportable under either 10 CFR 50.55(e) or 10 CFR 21. The NRC inspectors requested documentation to support the basis for the item being considered "not reportable;" however, documentation could not be produced by LP&L representatives during the inspection. The NRC inspector reviewed the nature of the nonconformance and concluded that, although the material had been procured as nonsafety-related, Rotohammer material is used to operate valves from a remote location via a mechanical linkage arrangement which utilizes a yoke and universal type joint. A further review of the application indicated valves were primarily located outside of primary containment and served a passive function.

(2) NCR W3-3930, dated June 21, 1982. This material was supplied in three separate lots, and each documented by a separate MRR. The description of the nonconformance stated that safety class material received on MRRs 204245, 205234, and 206028 was from an unapproved supplier. The reportability block on the NCR was marked "not reportable."

(3) NCR W3-3995, dated June 24, 1982. This material was supplied in one lot and documented by Ebasco on MRR No. 206817. The nature of the nonconforming condition

was again that Rotohammer was an unapproved supplier. The supplier provided a CoC, dated June 11, 1982 (for MRR No. 206817), verifying compliance to Ebasco specification LOU 1564.124F and the PO.

As a result, Ebasco's decision that the nonconformances discussed in NCRs W3-3828, W3-3930, and W3-3995 were not reportable under 10 CFR 50.55(e) or 10 CFR 21, although not formally documented, appeared to be consistent with the potential safety significance of the item.

c. Documentation for PO WP3-8515

(1) PO WP3-8515. This PO was placed with Rotohammer by the Ebasco Waterford 3 SES on April 8, 1982. A review indicated that the PO was for valve stem extension gear boxes and universal joints and was ordered as safety-related. The order required a CoC stating that the materials supplied were equal to or better than material supplied on the original order (NY 403568), and the provisions of 10 CFR 21 were specified. The Rotohammer CoC, dated April 20, 1982, stated conformance to Ebasco specification LOU 1564.124F and PO WP3-8515.

(2) NCR W3-3754. This NCR was issued on May 5, 1982 due to Rotohammer not being an approved supplier and recommended reviewing their past history from March 1980 to June 1982 to determine any negative trends, and to obtain a CoC to the design specification. The NCR disposition evaluation stated Rotohammer was reaudited and found to be satisfactory per Ebasco letter E823/298, dated June 28, 1982. As stated previously, the reportability block on the NCR was checked "not reportable" and the documented basis for such a statement was not available during the inspection.

(3) Ebasco memo, dated May 18, 1982. A memo from J. Gutierrez (Ebasco QA Site Supervisor) to R. Hyme (Ebasco QA New York) discussing NCRs W3-3745 and W3-3754 was reviewed during the inspection. This memo stated that Rotohammer was initially evaluated and was considered to be an acceptable supplier; however, for some unknown reason, Rotohammer was not reaudited at the end of their three year qualification period. Recommended action was to reaudit Rotohammer and

to ensure that appropriate personnel responsible for review and approval of POs are retrained on the procedures. As a result, Ebasco Material Receipt Inspection Report (MRIR) No. 82-03654 was issued on August 12, 1982 to accept the material which had been receipt inspected on MRR No. 204352. The material was accepted by Ebasco site QC.

d. Documentation for PO WP3-8653

This PO was placed with Rotohammer by Ebasco site personnel on April 28, 1982. The order was for a valve stem extension yoke adaptor and was ordered safety-related with 10 CFR 21 invoked. A CoC was required and received on April 30, 1982 certifying that the material was equal to or better than that supplied on PO NY 403568. Again, as in the past, an NCR was issued (NCR W3-3745, dated April 30, 1982) stating that Rotohammer was not listed on the Ebasco ASL, and therefore not an approved supplier. Temporary disposition included vibroetching the material to show identification and maintain traceability prior to conditional release which was requested on Log No. 82-078 and issued on May 18, 1982. The document cited NCR W3-3745 and stated that the material was required to support project start-up requirements with the approval based on the completion of the vibroetching. Ebasco MRIR No. 82-01875 was issued on May 21, 1982 and the material conditionally accepted by Ebasco site QC.

e. Approved Supplier Status

Based on a review of the ASLs and other related documentation, the NRC inspectors confirmed that Rotohammer was not an approved supplier from March 1980 to June 1982. LP&L computer data bases ZBS and PKG were queried in an attempt to independently identify which safety-related POs were issued to Rotohammer during this period. A review of this data indicated that PO WP3-9146, dated July 1, 1982, was the next PO issued to Rotohammer after the issuance of WP3-8653 on April 28, 1982. Based on a review of other information, WP3-9146 was issued to Rotohammer after they were qualified as an approved supplier. A number of additional POs were issued, but the sequential number of the orders indicates they would have been issued after WP3-9146. It was noted that PO WP3-9146 was subsequently renumbered as WP3-9146A by a PO supplement. This was done because another PO with the same

WP3-9146 number was inadvertently issued to Capital Valve and Fittings. This review confirmed that Ebasco completely identified all Rotohammer material (POs WP3-8515 and 8653) ordered or received during the affected period when they were not an approved supplier.

f. Classification of Rotohammer Material as Safety-Related

The NRC inspectors interviewed several LP&L personnel in an attempt to better understand the application and safety significance of material ordered on the affected POs. LP&L personnel stated that the apparent reason that Rotohammer valve stem extension material was ordered safety-related by Ebasco was because of an administrative procurement definition which stated that equipment attached to safety-related valves should also be procured as safety-related. They also stated that valves with Rotohammer stem extensions are utilized outside primary containment and furthermore primarily have a passive function. In most cases, the valve stem extensions are used to allow the valves to be operated from a more convenient remote location. In some cases, the location is in a lower radiation area thus providing reduced radiation exposure during operation. From a safety standpoint, it appears that this material would not normally be ordered as safety-related since the valves are passive and could still be operated locally in the event of a failure of the Rotohammer remote operator linkage. As a result, the purchase of this material from an unapproved supplier, although a procedural nonconformance, may not constitute a reportable condition to the NRC under the provisions of 10 CFR 21. Since this condition was self-identified internally by Ebasco and proper corrective action was taken by Ebasco to satisfactorily disposition the material, a nonconformance to Ebasco procedures was not identified during this part of the inspection. The Ebasco reaudit of Rotohammer in June 1982 reviewed Rotohammer's QA program and its implementation for the period of March 1980 to June 1982, the period during which material was received by LP&L, but for which Rotohammer was not on the Approved Suppliers List.

5. Review of the J. C. White Company

During the NRC inspection at Ebasco in August 1988, additional questions concerning the activities which led LP&L to issue a Significant Construction Deficiency (SCD) Report to the NRC was

reviewed. The J. C. White Company manufactures material known as "TUBE TRACKS" and related hardware used by Waterford 3 SES to house and support instrumentation of both safety and nonsafety-related tubing. The POs reviewed were designated as safety-related with 10 CFR 21 specified and Ebasco specification 860-78 which identifies the QA requirements for nuclear safety-related items and services. This specification requires the supplier to have a documented QA program which complies with the requirements of 10 CFR 50, Appendix B. In addition, certification supporting the material's mill test reports and certification of the material's chemical and physical specifications were also required from the supplier. Two POs to J. C. White were reviewed by the NRC inspectors (WP3-2646, dated June 8, 1979 and WP3-2953, dated August 24, 1979). In both cases the material was procured safety-related with 10 CFR 21 specified. The material ordered comprised various tube track components such as yoke clamps and fasteners.

The POs specified that the material for the angles, channels, flat steel sections, and fittings be ASTM A-569 and/or A-570 and hot dipped galvanized in accordance with ASTM A-123. Solid stainless steel tube clamps were specified to be 304 stainless to ASTM A-479, while the yoke and bundle clamps were specified to be 304 stainless to ASTM A-240. During the process, additional supplements to the original PO were issued by Ebasco changing the previous material specifications and downgrading the previous PO to nonsafety-related. In addition, a later supplement to PO WP3-2646, Ebasco reclassified the material again as nonsafety-related. A similar series of events occurred for PO WP3-2953 also. An explanation for this inconsistency could not be provided by LP&L during the inspection.

J. C. White provided certification for material ordered, however, could not in all cases provided certification documenting the mill test reports, as required by the PO. This lack of certification led to concerns about the chemical and physical properties of some of the tube track material and also the affect this may have on seismic performance. As a result, NCR WP3-2749 was issued on June 10, 1981 and was forwarded to LP&L on July 21, 1981 by Ebasco as potentially reportable incident No. 49. Ebasco QA report LOU-4294 evaluated NCR W3-2749 for reportability and concluded it was reportable under 10 CFR 50.55(e) and that, until reviewed further, the physical material properties of both ASTM A-569 and A-570 are questionable due to the fact that J. C. White's quality program, conditionally accepted by Ebasco on

August 15, 1978, was not being implemented by J. C. White for the manufacture of tube track and channel for the following POs: WP3-1608, 2646, 2953, 4464, and 5829.

The recommended disposition was to review J. C. White and it's subvendor's documentation for compliance with ASTM A-569 and A-570 and to evaluate chemical properties and requirements for acceptability of material. In addition, the NCR stated that an Ebasco audit of J. C. White on May 22, 1981 noted that tube track material was not manufactured in accordance with 10 CFR 50, Appendix B, as required by the PO referencing Ebasco specification 860-78 which specifies the requirements of Appendix B. This condition eventually led LP&L to issue SCD Report No. 35 to the NRC. The final report was issued on August 29, 1983.

On December 22, 1981, Ebasco performed a follow-up QA audit of J. C. White which was documented in Ebasco letter E654/589, dated January 8, 1982. This letter stated that J. C. White's corrective actions to the May and July audits were found satisfactory and was signed by the Ebasco Vendor Evaluation Group Leader. Ebasco initiated a Design Change Notice (DCN) to the Waterford 3 construction drawings on May 4, 1982 (DCN-NYC-IC-833), including Revisions 1 and 2, dated August 19, 1982. These DCNs changed drawings to establish maximum cantilever support spans for the tube track material which presumably would not violate maximum stresses in the material based upon Waterford 3 SES receiving the least conservative material specification (i.e., ASTM A-569).

In order to provide a technical resolution that would be satisfactory for material that was already installed in the plant as well as material that was available for installation, an extensive material test program was initiated by Ebasco to determine the chemical and physical properties of the tube track material that had been supplied. Results from these tests were used to justify the use of the tube track material for any of the designed applications, including seismic.

This was documented in NCR W3-6599 issued on July 26, 1983 and provides background information relative to the material certification concerns originally identified in NCR W3-2749. The NCR stated that an independent laboratory (Lucius Pitkin) performed testing on tube track fittings and found carbon levels as high as .21 percent while ASTM A-569 specifies a maximum carbon content of .15 percent. The evaluation of the high carbon condition

indicated that due to the application of the fittings and with the existing material being weldable, the higher carbon content was acceptable. Attachment 1 to the NCR (memo to J. DeBruin to W. Yaeger, dated March 17, 1983) stated that materials application engineering (Ebasco) had been working on a program to upgrade J. C. White tube track fittings so that they can be used for Seismic 1 applications. The memo further indicated that 315 J. C. White fittings would be shipped to Pitkin Laboratories for verification of both chemical and physical material properties. Attachment 2 to the NCR (memo from L. Patrick to J. DeBruin, dated May 5, 1983) discussed the results of the Pitkin Laboratory testing of the J. C. White fittings. In summary, the memo concluded that based on the data obtained from this testing, including an evaluation of the service and load requirements for the applications, the tube track material could be used to support the Waterford 3 instrumentation tubing systems satisfactorily.

A 1986 Tube Track catalog was reviewed during the inspection to better understand products supplied commercially by J. C. White. The review indicated that with the exception of solid stainless spacer clamps, the items identified in the catalog were manufactured with commercial grade materials (e.g., 18-8 stainless and carbon steel). It appears that the main technical difference between the items listed in the catalog and the items listed in the Ebasco POs is that Ebasco invoked a material upgrade by specifying

ASTM materials for manufacturing certain items instead of allowing commercial grade materials to be used during manufacturing.

As a result of the NRC review of concerns raised with the J. C. White Company's QA program, it appears that the concerns identified were satisfactorily resolved based upon Ebasco reaudits and adequate corrective action taken by Ebasco, J. C. White, and LP&L.

F. PERSONS CONTACTED:

Louisiana Power and Light

#* L. W. Laughlin, Licensing
#* N. S. Carns, Plant Manager
#* J. J. Zabritski, Operations QA Manager
#* L. F. Lubinski, Procurement Representative
#* M. A. Triggs, Nuclear Records Manager
#* L. L. Bass
#* G. M. Davis, Events Analysis Manager
#* P. V. Prasankumar, Assistant Plant Manager
#* J. E. Howard, Procurement Programs Manager
#* D. V. Gallodoro, Procurement Engineer
J. R. McGaha
D. Gilley, Maintenance Engineer
R. Bennet, QA Supervisor
T. Gerretts, QA Manager
J. Sleger, Nuclear Safety Review Manager
W. Morgan, QA Audit Supervisor

NRC

#* Robert L. Pettis, Jr., Senior Reactor Engineer, NRC
#* Terrence L. Tinkel, NRC Consultant
W. F. Smith, NRC Senior Resident Inspector, Waterford 3
E. William Brach, Chief, Vendor Inspection Branch, NRC
* T. R. Staker, NRC Resident Inspector, Waterford 3

*Attended entrance meeting
#Attended exit meeting

ORGANIZATION: KLOCKNER-MOELLER
BONN 1, FEDERAL REPUBLIC OF GERMANY

REPORT NO.: 99901133/88-01	INSPECTION DATE: November 17-18, 1988	INSPECTION ON-SITE HOURS: 16

CORRESPONDENCE ADDRESS: Klockner-Moeller
Hein-Moeller Strasse 7-11
Bonn 1, Federal Republic of Germany
D-5300

ORGANIZATIONAL CONTACT: K. Rademacher
TELEPHONE NUMBER: (0228) 002-674

NUCLEAR INDUSTRY ACTIVITY: Manufacturer of contactors, low voltage circuit breakers, and motor control centers.

ASSIGNED INSPECTOR: _____ Date
K. R. Naidu, Reactive Inspection Section No. 1 (RIS-1)

OTHER INSPECTOR(S): None

APPROVED BY: _____ Date
E. T. Baker, Chief, RIS-1, Vendor Inspection Branch

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR 50 Appendix B

B. SCOPE: Review the implementation of the Quality Assurance Program in selected areas including design changes to circuit breakers manufactured and supplied to LaSalle County Station Nuclear Power Plant.

PLANT SITE APPLICABILITY: LaSalle County Station (50-373, 374); Grand Gulf (50-416, 417); North Anna (50-269, 270); Zion (50-295, 304) and Monticello (50-263).

A. VIOLATIONS:

No violations were identified during this inspection.

B. NONCONFORMANCES:

No nonconformances were identified during this inspection.

C. UNRESOLVED ITEMS:

No unresolved items were identified during this inspection.

D. INSPECTION FINDINGS AND OTHER COMMENTS:

1. Background Information

Klockner-Moeller (K&M), headquartered in Bonn, Federal Republic of Germany (FRG), manufactures low voltage switchgear such as circuit breakers, contactors, electric motor starters, and motor control centers (MCCs). K&M has numerous manufacturing facilities in FRG and other countries. In 1977, K&M supplied MCCs with starters, contactors, and molded case circuit breakers (CBs) for installation at LaSalle County Station (LSCS) Units 1 and 2 located in Illinois. K&M also supplied switchgear to Grand Gulf, North Anna, Zion, and Monticello nuclear power plants.

This inspection was conducted to verify the implementation of the K&M quality assurance program in selected areas, including design change control and testing.

During the preparation phase of the inspection, the inspector obtained the following model numbers on the various types of CBs, contactors and thermal overloads installed at LSCS.

a. Circuit Breakers

NZM H6 - 63/Z M6 - 6.6 - 32 - ODI - CNA
NZM H6 - 63/Z M6 - 2.1 - 12 - OBI - CNA
NZM H6 - 63/Z M6 - 3.7 - 20 - OBI - CNA
NZM H6 - 63/Z M6 - 15 - 120 - OBI - CNA

b. Thermal Overloads Used With CB's

ZO - 3.7/K - NA
ZQ - 11/K - NA

c. Contactors

ZO - 14/K - NA
Z2 - 40/K - NA
DIL - 0 - 22 - NA
DIL - 2V - 22 - NA
DIL - 3 - 22 - NA
DIL - COLB - 22 - NA

At K&M, it was explained that the contactors mentioned in c above are manufactured at a different plant located about 400 kilometers from Bonn. The NZM H6 type CBs are manufactured in Bonn; the ZO type thermal overloads are manufactured in the Werke Bayenthal facility located in Cologne, approximately 20 kilometers from Bonn.

Some of the salient features of the K&M CBs are as follows:

a. The front covers of the CBs are transparent. As such, any unusual discoloration of the main contacts is readily visible. Such CBs are removed from service and discarded. K&M stated that they do not sell replacement spare parts for their CBs. Consequently, the CBs are discarded if they are not usable.

b. Each CB is subjected to a final test. K&M does not employ sampling techniques to test assembled CBs.

c. The shunt trip coil used in the CBs are rated for continuous current operation.

2. Plant Tours

The inspector, accompanied by the QA manager, toured the facilities in Bonn and Werke Bayenthal to observe the implementation of the inspection program, segregation of material and the final tests.

a. Incoming Receipt Inspection

The plant at Bayenthal had received 10,715 springs on November 11, 1988 from Schmiede Knecht, a subvendor. These springs are utilized in the assembly of NZM H6 type CBs. The computer selected the Acceptable Quality Level for the number of springs to be tested based on the lot size of

11,000. The springs were inspected to the relevant drawing, Z23-300-62, utilizing Incoming Inspection Procedure, PV 123-60, dated February 5, 1987. The procedure provided the nominal length of the spring and the force in newtons to elongate the spring to a specified length. The tools utilized to perform the inspection were observed to have current calibration stickers. The receipt inspection did not identify any unacceptable springs within the tested sample.

b. In-Process Inspections

The inspector observed in-process inspections being performed at both the plants. Specifically, the inspector observed the inspection being performed on an auxiliary switch upper part subassembly at the Cologne plant. The smoothness of the movement of the contact was verified. The operating mechanism was turned ON and OFF three times, and to the ON and TRIP position three times. Samples of the terminals of the CBs were subjected to inspections. A test apparatus was available to test the strength of the terminals.

c. Mechanical Tests

The following mechanical tests were being performed after the operating mechanism was assembled as required by the applicable procedure. The procedure provided the minimum and maximum acceptable standards, including the units of measurement of the variables.

(1) The correct operation of the moving contacts was verified. This verification ensured that the moving contacts mate with the stationary contacts.

(2) The start and end positions of the trip bar were verified.

(3) The change of state of the auxiliary switch contacts was observed.

(4) The integrated force required to trip the CB was measured to determine if it was within the acceptable value specified in the procedure.

d. Control of Rejected Material

Components rejected during the inspections were placed in red colored baskets. In addition to the in-process inspections, final inspections are performed on each CB and other switchgear components. Each factory provides summary data on the number of switches tested and the number rejected. If the number of rejects exceeds 5 percent of the production run, production is stopped. The assembly process can commence only after the problem is identified and corrected.

3. Review of the K&M QA Program

The Quality Assurance (QA) program basically conforms to Appendix B to 10 CFR 50. In addition to the 18 criteria, the QA program covers servicing and statistical techniques. QA and quality control (QC) representatives are assigned responsibilities in each manufacturing facility in the areas of development, research, service, testing and shipping. Detailed procedures have been developed for each of the various activities for the QA/QC personnel to implement.

The changes to design drawings are controlled in the manner prescribed in their QA program and require the concurrence of QA representatives before they are issued for production. Emphasis is placed on QA during the life testing of the various products manufactured to ensure that the products perform satisfactorily. Codes to identify the date of manufacture are stamped on components such as contactors and overloads to provide traceability.

4. Review of Control of Design Changes

The inspector obtained specific details on three different types of CB's and the trip mechanisms for the CBs installed at LSCS prior to the inspection. The inspector reviewed the design changes for NZM H6 type molded case CBs and ZM6 type trip mechanisms used in the NZM H6 type CB.

a. Drawing 1Z 123 - 74-XII for molded case CBs, type NZM H6, contained the following revisions:

(1) On July 27, 1977, the drawing was redrawn.

(2) Revision i - August 18, 1982. The continuous current table was added.

(3) Revision k - March 13, 1984. The cutout view of the shunt trip was added.

(4) Revision l - November 7, 1986. The handle to operate the switch from the rear of the CB was deleted.

(5) Revision m - June 12, 1987. The label to indicate that the CB complies with the Canadian Standards Association (CSA) was deleted. The label was considered superfluous because there is a label for Underwriters Laboratories.

b. Drawing 1Z 23-70-I showing the ZM6 type trip mechanism for the NZM H6 type CB contained the following revisions:

(1) Redrawn on December 22, 1978.

(2) Revision ab - November 23, 1979. Special part for chemical industry deleted.

(3) Revision ac - December 12, 1979. The trip bar was modified.

(4) Revision ad - March 24, 1982. Special type of trip mechanism suitable for 400 HZ was added.

(5) Revision ae - July 26, 1982. K&M instruction was revised to change the calibration method.

(6) Revision af - September 8, 1982. The cover of the trip mechanism was added to the drawing.

(7) Revision ag - February 2, 1983. The screw to hold the bimetalic terminal had a washer and a spring washer. The washer was deleted.

(8) Revision ah - May 5, 1984. The surface of the three pins on the trip bar which press the trip bar to trip the CB was changed. The distance between the pin and the trip bar can be adjusted to respond to the trip current.

The inspector observed that the revisions to the drawings contained the signatures of the draftsman and the chief engineer. K&M determined by engineering judgment that the

changes were minor and would not affect the seismic qualification of the CB and the trip attachment.

5. Final Tests

Final tests are performed on each of the thermal magnetic trip units, thermal trip units, and CBs. The following are the salient features tested:

a. Trip Units

The thermal magnetic trip units and the thermal trip units have different style numbers. The design of the thermal magnetic trip unit is such that it cannot be assembled without the thermal or the magnetic element in it. The following features are verified:

(1) Air gap of the coil.

(2) Inspection to determine that the contact is adequately welded.

(3) Uniformity of the three poles.

(4) Contact screws are tight.

b. Circuit Breakers

The following are some of the tests performed on each CB manufactured.

(1) The CB is subjected to two fast and two slow ON/OFF cycles and the time periods to operate are measured. Acceptable fast operate time is 200 milliseconds (ms) and slow switch time is 3000 ms.

(2) The operation of the shunt trip unit with reduced and overvoltage conditions.

(3) The operation of the undervoltage trip unit. The undervoltage coil is energized from the moment the CB handle is turned from the "OFF" position to the "ON" position. The German Vereinigte Deutsche Electrotechnischeverein (VDE) standards require that

the voltage to the undervoltage coil be removed when the CB is in the "OFF" position.

(4) Every CB manufactured is subjected to an insulation resistance test. For one second, 2.64 KV is applied between the terminals R-U/S-V/T-W with the CB in the open position and between the terminals R-S/S-T/T-ground with the CB in the closed position, where R, S, and T are the line side terminals and U, V, and T are the load side terminals.

6. Review of Audit Performed By British Standards Institution

The inspector reviewed the audit performed by the British Standards Institution (BSI). Report 86/3856/A02, dated January 6, 1987, documents the visit of two assessors to K&M during January 6-8, 1987. The objective of the visit was to satisfy BSI that the components distributed through K&M United Kingdom (UK) are procured from a quality assured source and are regularly audited and tested. BSI also verified that the necessary authority for quality was visibly delegated by the K&M Bonn Head Office to the UK operations, and that procedures exist to ensure that the responsibility for initiation, distribution, amendment, and authorization is readily apparent and visible, for the flow of all quality and associated documents between the Head Office and other manufacturing facilities.

The report contained no adverse findings.

E. EXIT INTERVIEW:

The inspector met with persons identified in Section F and discussed the scope and findings of the inspection.

F. PERSONS CONTACTED:

*K. Rademacher	Quality Assurance Manager
*W. Lange	Quality Assurance Group Leader
H. Behr	Quality Assurance Manager, Division E
V. Vogt	Director, Werke Bayenthal
H. Goerke	Quality Assurance Manager, Werke Bayenthal
K. Pawloski	Incoming Receipt Inspector, Werke Bayenthal

* Denotes the individuals who attended the exit meeting on November 18, 1988.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99900404/88-02	INSPECTION DATE: 11/16-17/88	INSPECTION ON-SITE HOURS: 30

CORRESPONDENCE ADDRESS: Mr. Carlo L. Caso, General Manager
Westinghouse Electric Corporation
Nuclear and Advanced Technology Division
Post Office Box 355
Pittsburgh, Pennsylvania 15230

ORGANIZATIONAL CONTACT: Mr. David Alsing, Manager, Quality Assurance
TELEPHONE NUMBER: (412) 892-3708

NUCLEAR INDUSTRY ACTIVITY: Westinghouse provides NSSS components and services for nuclear power plants.

ASSIGNED INSPECTOR: Walter P. Haass 1/18/89
W. P. Haass, Special Projects Inspection Section (SPIS) Date

OTHER INSPECTOR(S): R. C. Jones, Jr., RXB/NRR

APPROVED BY: Uldis Potapovs 1-18-89
U. Potapovs, Chief, SPIS, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

A. BASES: 10 CFR Part 50.

B. SCOPE: Review records regarding an allegation concerning the Westinghouse evaluation model for reflood following a LOCA; review the Potential Item File System to assess the procedures and their implementation to resolve safety concerns; and review other files in the same technical area that could assist in the above two areas.

PLANT SITE APPLICABILITY: All nuclear plants with Westinghouse PWR-type NSSSs.

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED ITEMS:

None

D. OTHER FINDINGS AND COMMENTS:

1. Allegation on the Westinghouse evaluation model for WREFLOOD

An allegation was received by NRC that alleged the Westinghouse evaluation model for WREFLOOD was in violation of 10 CFR 50.46 Appendix K. The issues centered on the calculation of the refill period of a large break LOCA and were specifically concerned that the logic in the WREFLOOD code and the calculation of the characteristic cold leg volume necessary to establish flow at the top of the downcomer (referred to as VOLRSD) were in error.

To evaluate the specific concerns raised, the inspectors reviewed Potential Item (PI) File 86-029 entitled, "Hot Wall Delay Model in WREFLOOD," which was opened to address the concerns raised by the alleger. Westinghouse Calc Notes SEC-SA-2306-CO and SEC-SD-064-CO, which were referenced in the PI file, were also reviewed. This PI file was opened on August 29, 1986 in response to a request by another Westinghouse employee; the alleger apparently did not request that a PI file be opened. Westinghouse completed its evaluation of the issues and closed the PI file on May 12, 1988.

With respect to the possibility of a logic error in the WREFLOOD code, our review concluded that the code logic is as described in WCAP-8471-P-A, "The Westinghouse ECCS Evaluation Model: Supplemental Information." Specifically, execution of the code is delayed for a period of time associated with the transport time of the accumulator water from the injection point to the lower plenum. Following this delay period, the calculation is started

and all injected water is placed directly in the lower plenum. Thus, the alleger's observation that water is placed directly in the lower plenum is correct. However, use of the transport time delay to account for the lack of detailed cold leg modeling in the WREFLOOD code has been previously accepted by the staff and does properly account for the effects of water accumulation in the cold leg. Therefore, the inspectors concluded that there is no logic error in the code.

The calculations of the VOLRSD parameter and the transport time delay were also reexamined by Westinghouse in Calc Note SEC-SA-2306-CO. While the specific value for VOLRSD used in the Westinghouse evaluation model is provided in WCAP-8471-P-A, it was not apparent to the inspectors that the specific calculations for this parameter were examined by the staff during the initial review. Thus, the inspectors performed a review of the Calc Note to determine whether the VOLRSD parameter is calculated appropriately.

Consistent with the overall Westinghouse evaluation model approach, it was assumed that the refill/reflood transient can be viewed as a quasi-steady state process. Using this assumption, and considering the fact that the reactor coolant pumps are still coasting down, injected flow is directed towards the vessel and a void fraction within the cold leg is calculated using a standard two-phase flow model. This void fraction is then integrated over the cold leg piping volume (yielding a value of 43.92 cubic feet) which is then multiplied by the number of unbroken loops to determine the value of VOLRSD. From this result, the transport time delay is determined and used for all plant calculations. In fact, it is only this time delay which impacts the WREFLOOD calculation; the VOLRSD parameter, although input, has no impact on the results.

The constant value for the cold leg transport time delay is based on a specific piping length from the accumulator injection point to the reactor vessel, a specific diameter, and a specific injection rate. As alleged, plant specific parameters, such as the actual size for the injection nozzle or the actual injection flow rate, could impact this result. While not addressed within the Calc Note, this issue was discussed with Westinghouse personnel. Review of drawings and other plant specific information by the inspectors indicated that the assumptions made for the evaluation model calculations would likely result in a conservative estimate of the delay time for all plants. Thus, use of the same transport time delay for all plant types is acceptable.

Therefore, it was concluded that the general approach used by Westinghouse in its evaluation model for WREFLOOD is consistent with that approved by the staff and no logic error exists. The specific calculational techniques used to determine VOLRSD, and more importantly, the cold leg transport time delay, were appropriate. Thus, the allegations raised could not be substantiated.

2. Review of the Potential Item File System

The inspectors reviewed the procedures governing the processing of an employee-identified safety concern by Westinghouse. The top procedure is:

- WRD-OPR-19.0, "Identification and Reporting of Substantial Safety Hazards, Significant Deficiencies, and Unreviewed Safety Questions," Rev. 3, dated November 1, 1988

Each technical group in turn has developed and issued an instruction/guidance (IG) document that provides more specific steps for an employee reporting a safety concern. The groups that have issued IG documents on this subject include Product Licensing, Risk Management and Operations Improvement, Operating Plant Licensing Support, and Safeguards Engineering and Development. All IG documents were consistent with the top procedure with the exception of the one for the latter group which required reporting of a safety concern to the immediate manager first; this IG document reflected the earlier Westinghouse procedure and had not been revised to the later procedure. The procedure had been modified to provide greater assurance for anonymity if desired.

The inspectors also reviewed the Westinghouse video tape used for orientation of new employees and the periodic training of existing employees, and the NSD Orientation Manual for new employees. Both documents had not been revised to incorporate the new procedure for reporting of safety concerns to assure anonymity. However, Westinghouse indicated that at the time of presentation, oral corrections were given to provide employees with the latest instructions.

The inspectors noted that the proper revised procedure for the reporting of safety concerns was included in a posting of the 10 CFR Part 21 regulation that appeared in conspicuous locations for all employees to see.

The procedure involves the reporting of any condition adverse or potentially adverse to safety by the employee to the immediate supervisor, or any other manager, or the Secretary of the Safety Review Committee (SRC). If anonymity is desired, then the condition need only be reported to the SRC Secretary since the person in that position is empowered to assure the employee's confidentiality. The employee is required to complete a form or write a letter to document the concern. The preparation of a Justification for Continued Operation (JCO) is also required by the procedure. At this point, a Potential Item file is opened and responsibility for evaluating the concern is assigned to a manager and a technical person with expertise in the area of the concern.

The technical expert performs the necessary analysis and evaluation and prepares a basis for close-out or reportability. If the former is recommended, the basis is reviewed by the Secretary of the SRC, the technical expert, the immediate manager, another manager, and the concerned employee. If any one member of this group disagrees, the matter is referred to the SRC for further consideration. If the latter is recommended, the matter is forwarded to the SRC directly following review by the technical expert's manager.

If the SRC determines that the matter is reportable to the NRC, then the appropriate steps are taken to report the matter under the provisions of 10 CFR Part 21. If the SRC determines that the matter is not reportable and the concerned employee disagrees, the safety concern can be elevated to higher levels of management up to and including the division Vice-President who is responsible for reporting under Part 21. At this point, a final decision is made.

The inspectors indicated that the full extent of the procedural steps that an employee might wish to follow was not provided in documented form. Westinghouse stated that it was planned to issue an internal letter to all employees from the Vice President and General Manager as a reminder of the options available to identify and resolve employee safety concerns. The issuance of a brochure is contemplated to provide the specifics of the full process. The inspectors reviewed a prior internal letter to all employees on this subject dated February 19, 1976.

Other Westinghouse divisions including the Nuclear Components Division and the Nuclear Fuel Division have their own Safety Review Committees for analysis and evaluation of employee concerns.

however, all matters involving questions of reportability are resolved by the Nuclear and Advanced Technology SRC. All divisions are subject to the requirements of the top procedure (WRD-OPR-19.0) and have developed their own IC documents.

As noted above, NRC is not informed regarding the total PI file activity unless a safety concern has been evaluated and determined to be reportable. Westinghouse indicated that generally utilities are not kept informed regarding PI matters unless safety evaluations in process are affected, or a JCO cannot be written at the time a PI file is opened. Utilities will be informed if a commercial decision is involved; however, and when a Technical Bulletin is issued.

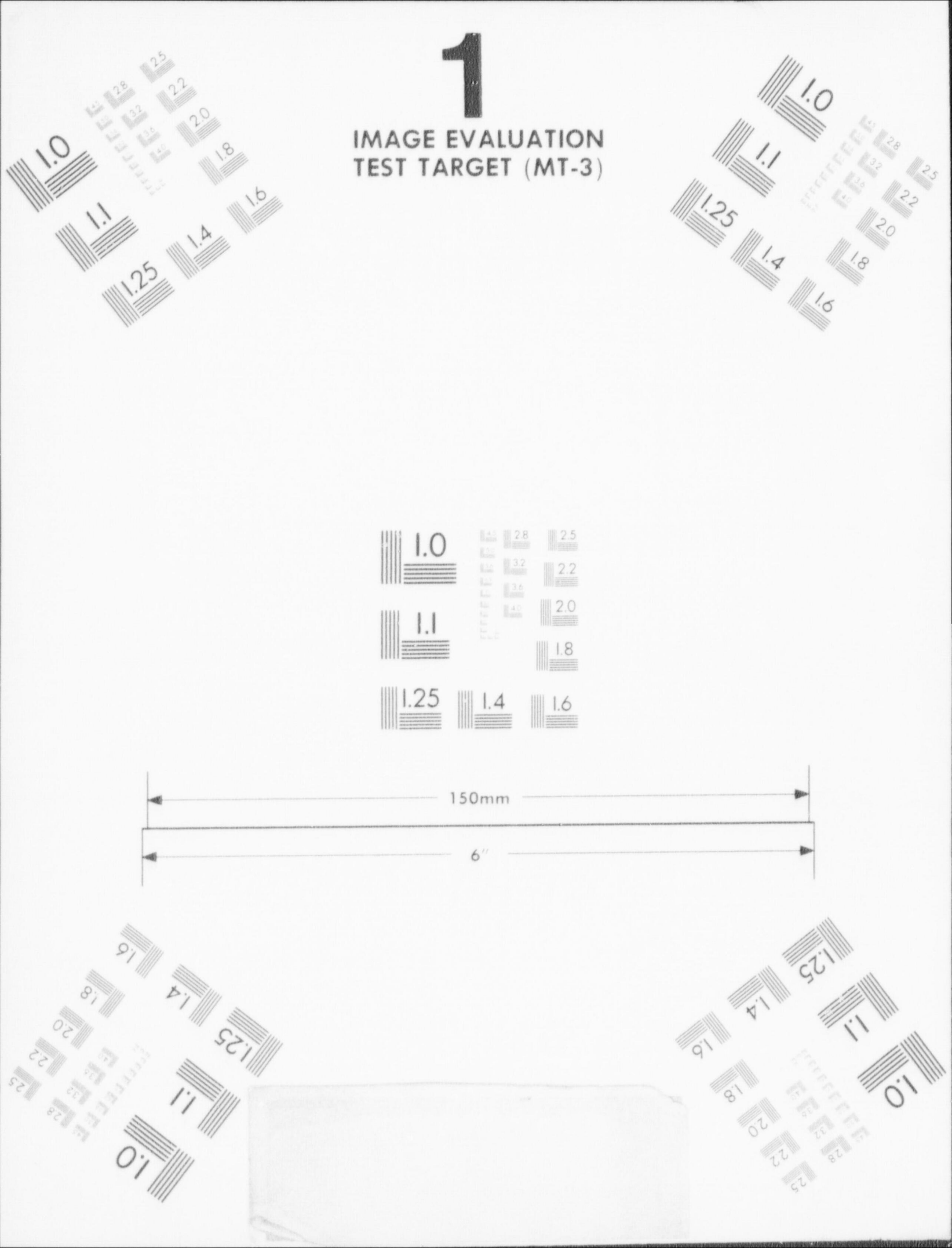
5. Review of Potential Item Files

Several PI files were reviewed to examine implementation of the PI file system. The files reviewed, and their current status were:

PI-86-012: Long Term Cooling - Generic (Open)
PI-84-264: ECCS Evaluation Model for 14 Ft Cores (Closed)
PI-85-209: RWST Temperature in LOCA Analysis (Closed)
PI-86-018: SI Time Delay Model (Closed)
PI-88-039: Hot Wall Delay in WREFLOOD (Open)
PI-87-042: Reduced Temperature Return to Power (Closed)
PI-86-024: Long Term Cooling Boron (Closed)
PI-87-039: Fuel Qualification Non-Conservation, Grid Stiffness and Seismic Spectra (Open)
PI-88-047: LOCA Limits, SG Tube Plugging Limits (Open)
PI-86-031: 3-Loop DARI Max SI (Closed)

The inspectors reviewed the contents of these PI files to ensure that JCOs were written and, for those issues which were closed or essentially resolved, that the analyses and evaluations performed were proper.

In general, the inspectors found the process for resolving PI files to be properly implemented and that sound engineering evaluations were performed. PI files 84-264, 86-018 and 87-042 appeared incomplete (letters requesting opening of the PI were missing); however, sufficient information was available, or referenced, to track resolution or current status of the issues. All JCOs reviewed were acceptable. For those PIs which were closed, the inspectors concluded that the evaluations performed were technically sound and met NRC requirements.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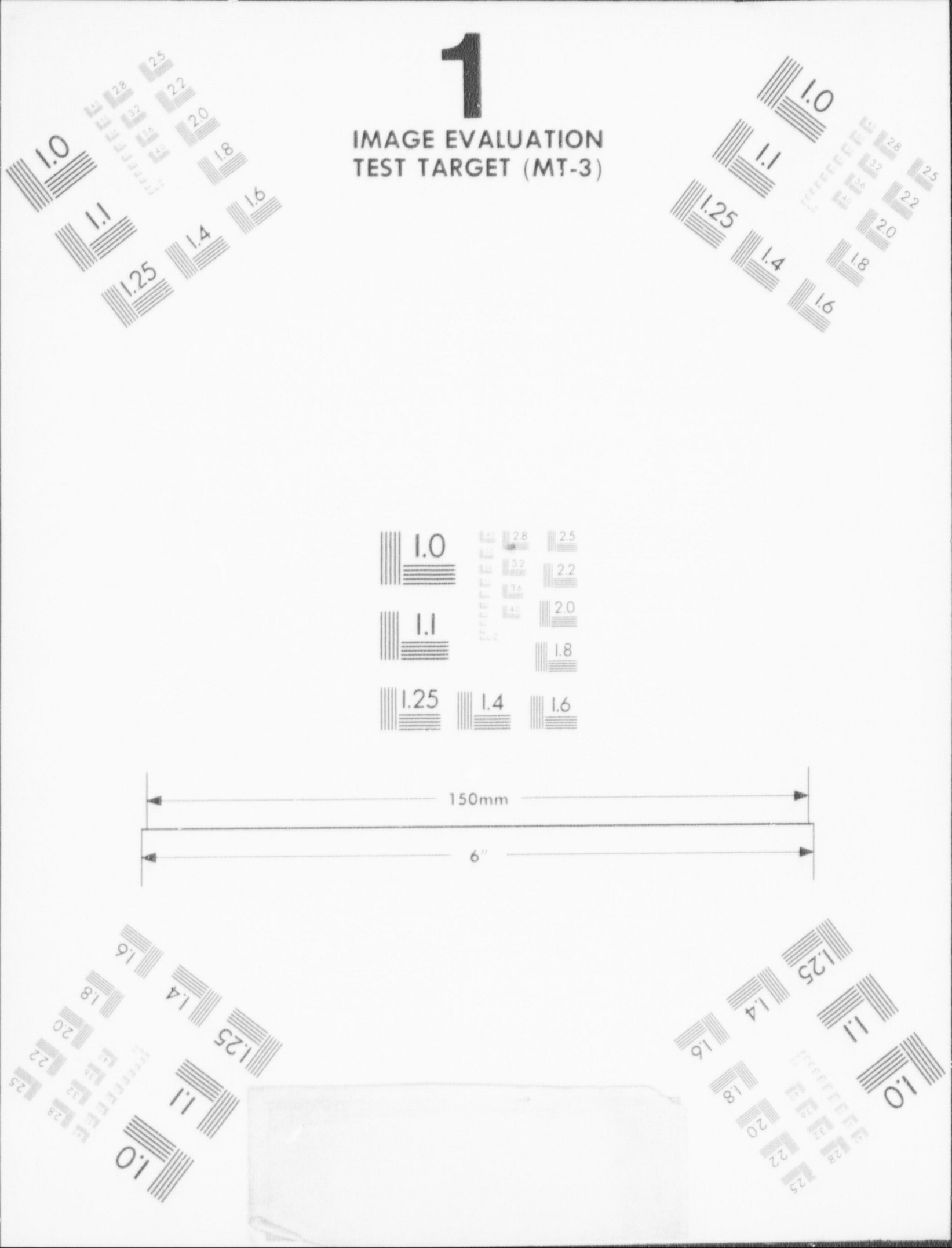
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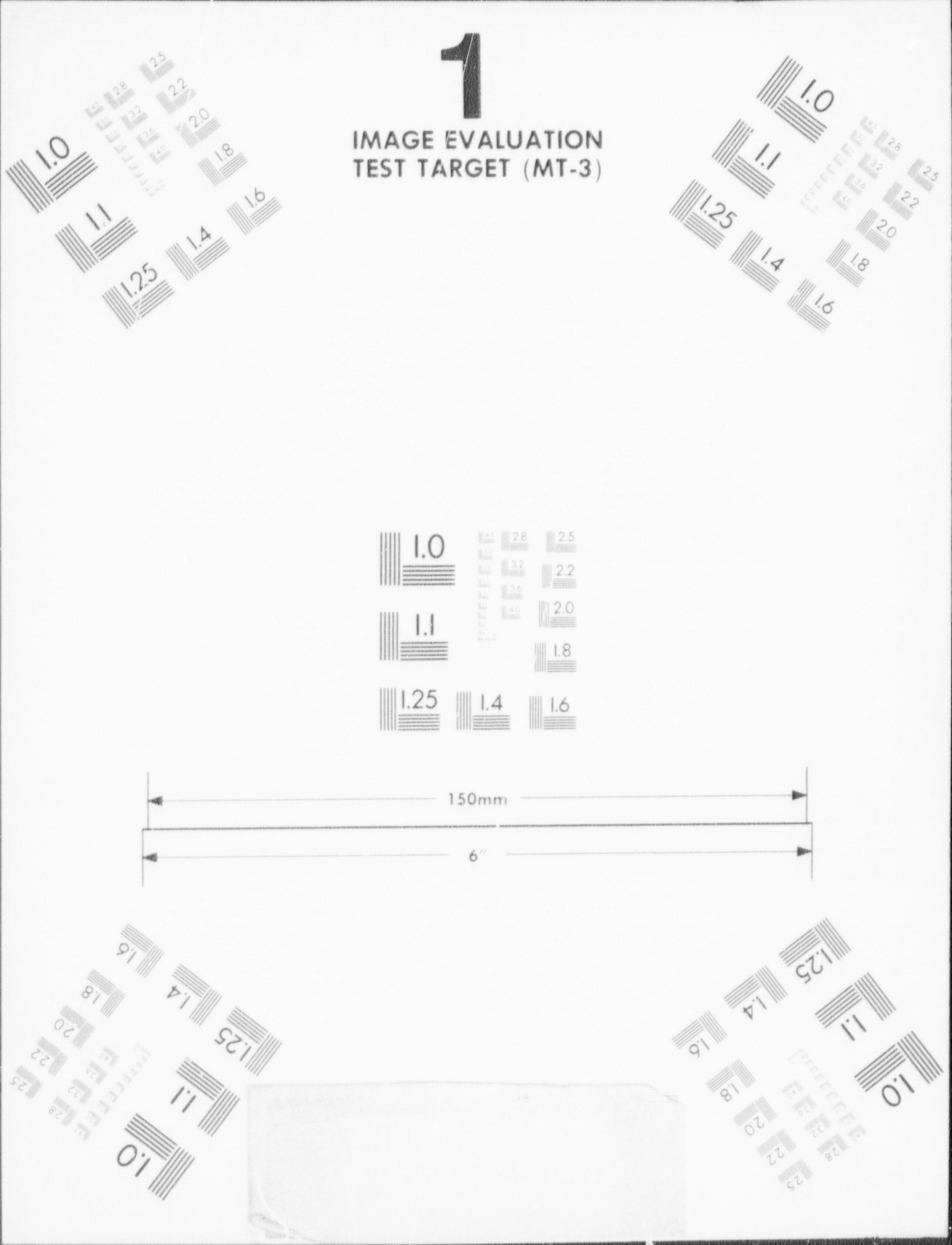
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Several generic items were raised by the inspectors, and discuss with Westinghouse personnel, based on the review of these files. One issue was related to "margin tracking." The inspectors noted that some of the JCOs and evaluations identified DNB or LOCA peak cladding temperature penalties. Plant specific safety analyses were then reviewed to assure that sufficient margin to applicable regulatory criteria existed to accommodate these penalties. Since it appeared that several penalties could have been identified, the inspectors questioned how the accumulated effect of these penalties are tracked. Westinghouse personnel described the various mechanisms used, including the use of safety analysis checklists, to assure that adequate margins are available in plant specific safety evaluations. Based upon these discussions, the inspectors concluded that the processes used for "margin tracking" are acceptable.

During the review of PI file 87-042, the inspectors noted that an employee raised concerns regarding the use of PRA methodology to resolve this issue. The specific issue raised by this PI was that LOCA loads did not consider the reduced reactor coolant system temperatures that would result during a Reduced Temperature Return to Power (RTRP) maneuver. Reduced temperatures could increase LOCA loads beyond those analyzed. This issue was resolved by first noting that the licensing basis for LOCA load calculations is based on 100 percent nominal power operation and does not need to account for normal plant maneuvers, such as startup, cooldown or RTRP, and secondly, using PRA methodology and the guidance in NUREG-0933, demonstrating that an RTRP maneuver would not result in a significant impact on public health and safety due to the limited time it is used. The inspectors reviewed the resolution provided within the PI file and concluded that the approach taken was consistent with regulatory requirements.

The inspectors discussed with Westinghouse personnel whether PRA is a general method used to resolve PI issues. It was stated that analyses and evaluations are performed in a deterministic manner for those issues within the licensing basis. For beyond design basis issues, a PRA approach may be used to determine the safety significance of the issue. Since design basis issues are being pursued in a deterministic manner consistent with the regulations, the inspectors concluded that Westinghouse is not placing undue reliance on the PRA approach to resolve employee-identified concerns.

E. PERSONS CONTACTED:

Westinghouse N&ATD:

+ Brian A. McIntyre, Manager, Product Licensing
Carl Hirst, Manager, RCS Licensing
+ Michael H. Shannon, Manager, Plant & Systems Evaluation Licensing
+ Fred Cadek, Manager, Safety Analysis Technology
+ Walter D. Tauche, Manager, Safeguards Analysis I
+ William J. Johnson, Manager, Nuclear Safety
+ David N. Alsing, Manager, Quality Engineering
+ Alex Paterson, Safety Review Committee Coordinator
+ Robert A. Wiesemann, Manager, Regulatory & Legislative Affairs
+ Raymond M. Tajc, Senior Engineer
Ronald P. DiPiazza, Manager, Operating Plant Licensing Support
Joseph V. Iannucci, Engineer
James L. Grover, Engineer
Craig Thompson, Engineer
Michael Emery, Engineer
Melita Osborne, Manager, Transient Analysis II

Nuclear Regulatory Commission:

+ Walter P. Haass, VIB/NRR
+ Robert C. Jones, Section Chief, RXB/NRR

- Attended Entrance Meeting
+ - Attended Exit Meeting

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
PENSACOLA, FLORIDA

REPORT NO.: 99900104/88-01	INSPECTION DATE: 10/4-6/88	INSPECTION ON-SITE HOURS: 60

CORRESPONDENCE ADDRESS: Mr. Jose M. Martinez
Product Assurance Manager
Westinghouse Electric Corporation
Nuclear Components Division
Post Office Box 1313
Pensacola, Florida 32514

ORGANIZATIONAL CONTACT: Mr. Jose M. Martinez, Product Assurance Manager
TELEPHONE NUMBER: (904) 474-4340

NUCLEAR INDUSTRY ACTIVITY: Westinghouse manufactures NSSS components for nuclear power plants.

ASSIGNED INSPECTOR: Walter P. Haass — 1/10/89
W. P. Haass, Special Projects Inspection Section (SPIS) — Date

OTHER INSPECTOR(S): R. W. Woodruff, NRR:OEAB
K. R. Wichman, NRR:EMTB
R. Cid, ~~CSM~~

APPROVED BY: Uldis Potapovs — 1-10-89
U. Potapovs, Chief, SPIS, Vendor Inspection Branch — Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review manufacturing and quality assurance procedures and records to identify the controls and their implementation for assuring that steam generators ready for shipment do not contain loose parts and other manufacturing debris; review examples of steam generator manufacturing work in progress; identify the interface between NCD and utility licensees and Westinghouse NSD regarding installation and servicing; and review procedures and their implementation for determining the reportability of defects to NRC under 10 CFR Part 21.

PLANT SITE APPLICABILITY: All nuclear plants with Westinghouse PWR-type NSSSs.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
PENSACOLA, FLORIDA

REPORT NO.: 99900104/88-01	INSPECTION RESULTS:	PAGE 2 of 7

A. VIOLATION:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED ITEMS:

None

D. OTHER FINDINGS AND COMMENTS:

1. General Control Techniques for Loose Parts and Debris

Assurance that loose parts and manufacturing debris are not present in the complete steam generator is provided by multiple inspections performed just prior to each step in the assembly process that results in an inaccessible area. As appropriate, barriers are inserted to preclude subsequent intrusion of such items as the assembly process continues. Assembly of the steam generator is accomplished in two major parts: the lower assembly and the upper assembly. When the lower shell and tube sheet assembly is completed, it is thoroughly cleaned and inspected for loose parts and debris prior to insertion of the wrapper and tube support plate "A" assembly. Following insertion of the wrapper assembly, inspection of the resulting annulus is performed and a protective barrier is set in place. Prior to insertion of each succeeding tube support plate, cleaning and inspection are again performed. Finally, the tubes are inserted, expanded, and sealwelded, and both ends of the lower assembly are cleaned and inspected and covered with protective barriers. In a similar fashion, the upper assembly is completed. Joining of the upper and lower assemblies is accomplished with a girth weld.

Following final assembly, access to the steam generator internals is provided by means of a manhole located in the upper assembly. Personnel entry is monitored 24 hours a day at this point in the manufacturing process and a log is maintained of all items, personal as well as manufacturing aids and installed hardware items, that ingress and egress the steam generator.

As a final check prior to shipment, the completed steam generator, while positioned horizontally, is slowly rotated while listening for falling loose parts.

Westinghouse NSD does offer a separate program for detection of loose parts in a steam generator subsequent to installation and prior to operation. It is called the Foreign Object Search and Retrieval (FOSAR) program. The extent of FORSAR use by utility licensees is not known at this time.

2. Steam Generator Responsibilities

The Westinghouse Nuclear Components Division in Pensacola performs the manufacturing of the steam generator units and also provides a manual for operation. Installation and subsequent servicing of the units as requested by the utility customer is provided by the Westinghouse Nuclear Services Division in Monroeville, Pennsylvania.

3. Cleanliness Procedures

From the time steam generators were manufactured at the Tampa facility until the present time where the units are manufactured in Pensacola, the procedures in use to assure cleanliness and removal of all loose parts and debris are as follows:

a. Process Specification 83318 PA, "Instructions for Final Cleaning Steam Generator," dated January 20, 1969 (Issue 1) through December 20, 1987 (Change 8), provides instructions for cleaning the primary and secondary sides of the steam generator prior to shipment.

b. Process Specification 83318 PE, "Cleaning of the Steam Generator Lower and Upper Shell Assemblies," provides instructions for cleaning of the lower and upper shell assemblies prior to assembly, during assembly, and after final assembly.

The above instructions were used at the Tampa facility and are directed primarily toward the removal of contamination from dye penetrant chemicals, oil, grease, metal chips, grinding dust, loose rust, paint, slag, scale, sandblasting abrasive, or dirt. "Loose parts" is not identified as an item of concern; rather,

references are made to removal of equipment, removal of all materials detrimental to the subsequent operation of the unit, general debris, and foreign debris.

c. Detailed Manufacturing Procedure DMP-5562, "Protection and Cleaning of Steam Generator Lower and Upper Shell Assemblies," dated December 21, 1987 Revision 5, provides instructions for cleaning of the steam generator lower and upper shell assemblies prior to assembly, during assembly and after final assembly.

d. Quality Inspection Procedure QIP-3364, "Cleanliness Requirements for Steam Generator Lower and Upper Shell Assemblies (Secondary Side)," dated January 18, 1988 Revision 3, provides instructions for QC-type checks of cleanliness at various steps in the assembly of the steam generator.

The above procedures are currently in use at the Pensacola facility and address similar cleanliness requirements as the Tampa instructions; however, in addition, the procedures do make specific reference to the removal of loose parts.

In summary, the evolution of the procedural controls to assure cleanliness of the steam generator units during the stages of assembly up to and including final assembly demonstrates increased concern for the presence of loose parts. It should be noted that the initial incidence of loose parts in Westinghouse steam generators attributed to the manufacturing process was observed largely in the 1982 to present time frame. Specific examples include Point Beach Unit 1 (November 1982), Watts Bar Unit 1 (March 1983), Harris Unit 1 (August 1986), and Catawba Unit 2 (February 1988). All of these steam generator units were manufactured prior to 1980 either at Westinghouse's Lester or Tampa facilities. Operating experience with more recently manufactured steam generators from the Pensacola facility is very limited in that these units generally have not reached their first refueling outage or have not operated at all.

4. Improvements in Loose Parts Control

Since the relocation of the Westinghouse steam generator manufacturing facilities from Tampa to Pensacola and the incidence of several loose parts events at operating nuclear

power plants, a greater sensitivity to the possible presence of loose parts in steam generator units ready for shipment was established. As noted above, procedures for controlling loose parts were strengthened to provide increased sensitivity to shop personnel to this concern. Also, during the assembly process, a hold point was established for the ASME Authorized Nuclear Inspector to perform a check for cleanliness and the presence of loose parts. Finally, with the joining of the lower and upper steam generator shells, a round-the-clock monitor was stationed at the manway ingress point to maintain a log of all temporary hardware items and personnel belongings to assure items were not inadvertently left in the unit.

The inspectors concluded that Westinghouse had taken some positive steps to increase the assurance that loose parts in completed steam generators would be better controlled for replacement steam generators.

5. Review of Records

The inspectors selected three steam generator units manufactured over the past 10 years for review of the routing sheets to determine the degree of conformance of cleanliness and loose parts controls relative to procedural requirements. The units selected were:

a. Catawba Unit 2 which was shipped in January 1980 from the Tampa facility.

b. Vogtle Unit 2 for which manufacturing was initiated at the Tampa facility and completed at the Pensacola facility and shipped in October 1982.

c. Indian Point Unit 3 which was completely manufactured at the Pensacola facility and will be shipped shortly.

The records indicated that at appropriate points in the assembly process the cleanliness and quality procedures were invoked and signed off. Customer hold points for a cleanliness check were noted as were those for the ASME/ANI. No significant differences in the number of cleanliness checks could be identified although as stated above there were differences in the degree of emphasis given in the applicable procedure regarding the concern for loose

parts. Despite the adequacy of controls in place, several instances have been identified in which the presence of loose parts was attributable to the manufacturing process. These instances included those steam generator units listed in Section 3 above. Westinghouse indicated that these units were manufactured prior to 1980 when the rate of steam generator manufacture was of the order of 30 units per year. The volume of production coupled with the lack of high sensitivity to the loose parts problem which arose in the 1982 to present time frame was cited as the most probable cause of the loose parts instances identified.

6. Reportability of Defects

The inspectors reviewed the procedures applicable to the Pensacola facility for the reporting of defects in safety related components under 10 CFR Part 21. The governing procedures are:

- WRD-OPR-19.0, Rev. 2, "Identification and Reporting of Substantial Safety Hazards, Significant Deficiencies, and Unreviewed Safety Questions," dated December 1, 1985.

- PQ-02-007, Rev. 2, same title as above, dated September 1, 1988.

The latter procedure is referenced in the Nuclear Components Division QA Program Manual in paragraph 16.4, "Significant Deficiencies."

The safety-related components in the steam generator are the tubes, tube sheet, shell, and lower head. The wrapper, the tube support plates, anti-vibration bars, and moisture separators are considered to be nonsafety-related.

The following examples of defects and their analyses were reviewed with regard to reportability:

- Feedwater Ring Backing Ring (1983): A concern was raised due to the potential loosening of the weld backing ring as a result of incomplete fusion during the welding process. Analysis determined that no portion of the backing ring could exit the feedwater ring (sparger) and enter the steam generator tube area, and that excess wear of the feedwater ring could not occur.

The problem was determined to be unique to the Model F design.

- Bottom Mounted Instrumentation Columns (1984): An electron beam weld failed that held an instrumentation column to the pressure vessel penetration. It was determined that this occurrence was an anomaly and an appropriate repair was performed.

- Tube Ovality (1977): Improper ovality of the steam generator tubes precluded insertion of the anti-vibration bars at the tube bend area. Corrective action was taken for the tube bending process. This was not a safety concern.

The inspectors concluded that the procedures for implementation of 10 CFR Part 21 and their applicability to the components of the steam generator were consistent with NRC requirements. Based on the review of several examples, the Part 21 system appeared to be properly controlled and the instances of defects were properly evaluated for reportability.

E. PERSONS CONTACTED

Westinghouse NCD

* + Jose M. Martinez, Product Assurance Manager
  + John P. Mortara, Technical Services Manager
  + B. R. Smelstoys, General Manager
* + Thomas A. Billman, Quality Assurance Engineer
* + E. Thompson, Manufacturing Engineer
* + John Bell, Manufacturing Engineer
* + D. Harmon, QA Engineering Manager
* D. Ford, Design Engineering Manager
* A. Owens, Area Manager
  J. Gillespie, Manufacturing Manager

NRC

* + Walter P. Haass, Senior Reactor Engineer/VIB
* + Roger W. Woodruff, Senior Reactor Systems Engineer/EAB
* + Keith Wichman, Section Leader/EMTB
* + Rafael Cid, NRC Assignee/CdeSN

* Attended entrance meeting
+ Attended exit meeting

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
COMMERCIAL NUCLEAR FUEL DIVISION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99900005/88-01	INSPECTION DATE: 12/14-15, 1988	INSPECTION ON-SITE HOURS: 16

CORRESPONDENCE ADDRESS: Mr. Mead D'Amore, General Manager
Westinghouse Electric Corporation
Commercial Nuclear Fuel Division
Post Office Box 355
Pittsburgh, Pennsylvania 15230

ORGANIZATIONAL CONTACT: Mr. R. Cost, Manager of Quality Assurance
TELEPHONE NUMBER: (412) 374-2359

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

ASSIGNED INSPECTOR: Robert L Pettis Jr. for 3/10/89
R. L. Cilimberg, Reactive Inspection Section No. 1 Date

OTHER INSPECTOR(S): None

APPROVED BY: Robert L Pettis Jr. for 3/10/89
E. T. Baker, Section Chief, RIS-1, Vendor Inspection Branch Date

INSPECTION BASES AND SCOPE:

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

B. SCOPE: Review records pertaining to the manufacture of hafnium control rods for rod cluster control assemblies (RCCAs) for Wolf Creek, Callaway, and Maanshan (Taiwan).

PLANT SITE APPLICABILITY: Wolf Creek (50-482), Callaway (50-483), Maanshan (Taiwan), and other reactor facilities with fuel supplied by Westinghouse.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
COMMERCIAL NUCLEAR FUEL DIVISION
PITTSBURGH, PENNSYLVANIA

REPORT NO.: 99900005/88-01	INSPECTION RESULTS:	PAGE 2 of 4

A. VIOLATIONS:

None

B. NONCONFORMANCES:

None

C. UNRESOLVED ITEMS:

None

D. STATUS OF PREVIOUS INSPECTION FINDINGS:

Not applicable.

E. INSPECTION FINDINGS AND OTHER COMMENTS:

1. The Nuclear Regulatory Commission (NRC) staff informed Westinghouse (W) management representatives of the scope of the inspection during the entrance meeting on December 14, 1988, and summarized the inspection findings during the exit meeting on December 15, 1988.

2. Background

Karl Hurst, Manager of RCS Components Licensing and other W staff advised the NRC by telephone on November 23, 1988, of a hafnium swelling incident that was discovered during eddy current (EC) testing of RCCAs at Wolf Creek. The EC testing was performed by Combustion Engineering to measure wear between the control rod cladding and the guide tube. During the testing, bump anomalies were discovered in the 304 stainless steel cladding on the control rods. W believes that the swelling of the cladding is caused by hydriding of the hafnium by hydrogen which evolved from the reactor coolant and diffused through the stainless steel cladding. W estimates that the swelling that could result from complete hydriding of the hafnium would cause an increase in scram time that is still below the Wolf Creek technical specification. The NRC inspection discussed in Sections 3 and 4 of this report did not identify any nonconformances with requirements for hafnium rod traceability and certification, and stated that the hafnium swelling was not related to manufacturing.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
COMMERCIAL NUCLEAR FUEL DIVISION
PITTSBURCH, PENNSYLVANIA

REPORT NO.: 99900005/88-01	INSPECTION RESULTS:	PAGE 3 of 4

3. Document Review

The inspector reviewed the computer lists in the W hafnium rod traceability system to determine what hafnium rods were used to manufacture the RCCAs which exhibited swelling in Maanshan 1, Wolf Creek, and Callaway. Hafnium ingot analysis and final product chemistry for 960 pieces of hafnium supplied to W by Western Zirconium Incorporated was selected at random for detailed review. The inspector determined that the certified data met W requirements for hafnium chemistry including a hydrogen specification of 60 parts per million (maximum). The same quality material was used to manufacture the RCCAs for Maanshan 1, Wolf Creek, and Callaway. The document review did not identify any deviations from W QA program requirements.

4. Hafnium Sweeling

The inspector met with Howard Menke, W Manager of Product Design to discuss what W had described as bump anamolies in RCCAs inspected during the third refueling outage at Wolf Creek. Mr. Menke summarized the information contained in a December 8, 1988, letter to C. E. Rossi of the NRC from Mr. W. J. Johnson, Manager of the W Nuclear Safety Department. W reviewed all manufacturing and assembly procedures as well as the fabrication inspection requirements for hafnium RCCAs. W concluded that inspection would have detected any as-built defects, therefore, manufacturing was considered to be an unlikely contributor to the bump anamolies.

Hafnium hydriding was identified as the most likely mechanism which led to the conditions reported at Wolf Creek. Hydriding occurs when hydrogen comes in contact with hafnium. Hydrogen from the reactor coolant diffuses through the stainless steel cladding and reacts with the hafnium to form hafnium hydride.

The results of the W investigation of the hafnium localized swelling phemomena covers all W supplied nuclear plants which use this type material and new information is currently being reported to the NRC. W has recommended continued operation of affected plants through three eighteen month or four annual cycles since W believes that current information does not indicate the existence of a substantial safety hazard.

ORGANIZATION: WESTINGHOUSE ELECTRIC CORPORATION
COMMERCIAL NUCLEAR FUEL DIVISION
PITTSBURGH, PENNSYLVANIA

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F. PERSONS CONTACTED:

*W. Hartnett
*J. Higginbotham
*H. Menke
*R. Pollard
*L. Reiland
*R. Ripley, Union Electric

*Attended exit meeting.

Selected Bulletins or Information Notices
Concerning Adequacy of Vendor Audits
and Quality of Vendor Products

1. Information Notice No. 89-02: Criminal Prosecution of Licensee's Former President for Intentional Safety Violations

2. Information Notice No. 89-11: Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cable Sizing

3. Information Notice No. 89-18: Criminal Prosecution of Wrongdoing Committed by Suppliers of Nuclear Products or Services

4. Information Notice No. 89-20: Weld Failure in a Pump of Byron-Jackson Design

5. Information Notice No. 89-21: Changes in Performance Characteristics of Molded-Case Circuit Breakers

6. Information Notice No. 89-22: Questionable Certification of Fasteners

7. Information Notice No. 89-23: Environmental Qualification of Litton-Veam CIR Series Electrical Connectors

8. Information Notice No. 89-29: Potential Failure of ASEA Brown Boveri Circuit Breakers During Seismic Event

VENDOR INSPECTIONS RELATED TO REACTOR PLANTS

VENDORS \ PLANTS	CALVERT CLIFFS	GRAND GULF	KEWAUNEE	LACROSSE	MIDLAND	NORTH ANNA	TROJAN	WATERFORD	WOLF CREEK	ZION
Auma Reister							X			
Combustion Engineering Power Systems Group	Numerous Facilities - Refer to Table I of Inspection Report									
Copes Vulcan	All Nuclear Power Plants									
Ebasco Services								X		
Klockner-Moeller		X	X	X	X	X				X
Westinghouse Elec. Corp. Commercial Nuclear Fuels	X								X	
Westinghouse Elec. Corp. Nuclear & Advanced Tech.	All Nuclear Plants with Westinghouse FWR-Type Nuclear Steam Supply Systems									
Westinghouse Elec. Corp. Nuclear Components Div.	All Nuclear Plants with Westinghouse FWR-Type Nuclear Steam Supply Systems									

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