U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
REGION IV

Report No. 99900400/80-01 Program No. 51100

Company: The Babcock & Wilcox Company
Nuclear Power Generation Division
P. O. Box 1260
Lynchburg, Virginia 24505

Inspection Conducted: March 11-14, 1980

Inspectors: C J Hale /for 4-15-80
D. F. Fox, Contractor Inspector — Date
Program Evaluation Section
Vendor Inspection Branch

J. M. Johnson 4/14/80
J. M. Johnson, Contractor Inspector — Date
Program Evaluation Section
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Observer: C J Hale 4-15-80
C. J. Hale, Chief — Date
Program Evaluation Section
Vendor Inspection Branch

Approved by: C J Hale 4-15-80
C. J. Hale, Chief — Date
Program Evaluation Section
Vendor Inspection Branch

Summary

Inspection on March 11-14, 1980 (99900400/80-01)

Areas Inspected: Implementation of Title 10 CFR 50, Appendix B, and Topical Report BAW-10096A including evaluation of supplier performance, audits, followup on 10 CFR 21 and 10 CFR 50.55(e) reports, and action on previous inspection findings. The inspection involved fifty-four (54) inspector-hours on-site by two (2) USNRC inspectors.

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Results: In the six (6) areas inspected, three (3) deviations from commitment were identified in two (2) of the areas and one (1) unresolved item was identified in another area.

Deviations: Follow-up on Previous Inspection Findings - QA records were not legible (Notice of Deviation, item A); Quality Assurance Audits, (1) audit checklists were not retained, (2) audit procedures do not contain committed ANSI requirements. (Notice of Deviation, items B & C)

Unresolved Items: Follow-up on 10 CFR 50.55(e) Reports - It could not be determined that CVARs receive the equivalent review and approval as the original PO technical requirements documents. (Details Section II, paragraph D.3.b.)

DETAILS SECTION I

(Prepared By D. F. Fox)

A. Persons Contacted

*C. A. Armontrout, Leader, QA Audit Section
W. T. Brunson, Principal Engineer
A. D. Chippley, Structural Engineer
D. C. Compton, Structural Engineer
*J. L. Davis, Manager, Data Management
*E. V. DeCarli, Manager, Quality Assurance
G. W. Delaney, Licensing Engineer
J. E. Galford, Manager, Systems Mechanical Analysis
A. Gharakhani, Engineer
P. Hannell, Manager, Release Administration
G. E. Hanson, Manager, Nuclear Operations Analysis
*S. H. Klein, Manager, QA Engineering
*A. L. MacKinney, Manager, General Services
D. Mars, Licensing Engineer
C. D. Morgan, Manager, Technical Staff
*P. J. Motiska, QA Engineer
K. M. Shepherd, QA Auditor
B. J. Short, Product Manager
P. Staples, Manager, Records Center
C. Taylor, QA Engineer
*J. H. Taylor, Manager, Licensing
R. O. Vosburgh, Supervisory Engineer
M. J. Yan, Principal Engineer

*Denotes those present at the exit interview.

B. Action on Previous Inspection Findings

1. (Open) Deviation (Report No. 79-04): A calculation did not include sources of design input, computer program names, versions and dates of runs. The inspector examined and found acceptable the corrective action described in the letter of response dated February 26, 1980, i.e, calculation No. 32-4057-00 was revised to include all requisite information. However, additional information is required on actions to be taken to correct and preclude recurrence of the deviation.

2. (Open) Deviation (Report No. 79-04): The HDL (Historical Document List) did not provide the correct identification

and status of a QA documentation package. The inspector examined and found acceptable the corrective action described in the letter of response dated February 26, 1980, i.e, the HDL data base was revised to reflect that QA data packages 23-1539-00 and 23-1540-00 were superceded by QA data package 23-1540-01. However, additional information is required on actions to be taken to correct and preclude recurrence of the deviation.

3. (Closed) Unresolved Item (Report No. 79-04): The identification, approval, or portions of contents did not appear to be complete or discernable on certain microfilmed QA Records.

   This unresolved item was elevated to a deviation from commitment. See Notice of Deviation, Item A, and section II.B.3.b of Inspection Report No. 99900400/79-04 for additional details.

4. (Open) Follow-up Items (Report No. 79-04): The inspector could not determine that all issued NPGD Comprehensive Administrative Manuals were complete and current during this inspection. This item will be further evaluated during a future inspection.

C. Use of Non-Conservative Seismic Response Spectra

1. Background Information

   VEPCo (Virginia Electric and Power Company) submitted a report on December 12, 1979, to Region II Inspection and Enforcement (RII) under the reporting requirements of 10 CFR Part 21. The report states that the computer code STALUM, used by B&W (Babcock & Wilcox Company) to calculate seismic response spectra, used as input to their customer's architect engineer for performing subsequent piping analysis, was incorrect. The subject code, through a misapplication of the SRSS (square root of the sum of squares) method of combining forces, incorrectly calculated the signs of all inertial forces as positive. Subsequent piping system response calculations utilizing the time history analysis method would therefore be incorrect since both the sign (direction) and magnitude (amplitude) of the inertial forces are required as input data for the response calculations.

   The licensee (VEPCo) first notified RII via a 10 CFR 50.55(e) report submitted December 7, 1979 (and subsequently by the 10 CFR Part 21 report submitted December 12, 1979). The licensee informed B&W of the error. B&W was requested to generate revised seismic response spectra and transmit it to the architect engineer (Stone and Webster Engineering Corporation) who had been instructed by VEPCo not to use the incorrect spectra previously submitted by B&W.

2. Objectives

The objectives of this area of the inspection were to verify that:

a. The available information and documentation are complete and accurate by document review or direct observation.

b. The cause and effect of the deficiency were identified, evaluated, and documented.

c. The corrective action taken was timely and that preventive measures are being planned or implemented.

d. The generic aspects have been reviewed and that affected organizations have been notified, or applicable.

3. Methods of Accomplishment

The preceding objectives were accomplished by review of the following documents.

a. The following NPGD (Nuclear Power Generation Division) Administrative Manual procedures to determine that the corporate commitments to control the design (including changes thereto) and to timely report, evaluate, and correct significant safety related deficiencies were correctly translated into procedures and instructions and are being implemented.

NPG-0402-01, Revision 5 dated October 20, 1978; Processing of NPGD Prepared Calculations

NPG-0405-22, Revision 5 dated September 20, 1979; Design Review

NPG-0503-04, Revision 9, dated March 3, 1980; Site Problem Report (SPR)

NPG-0503-07, Revision 5, dated August 20, 1979; Field Change Authorization

NPG-0902-06, Revision 3, dated January 15, 1980; Computer Program Development and Certification

NPG-1703-01, Revision 5, dated November 5, 1979; Preparation and Processing of Internal Deficiency Report/ Restraint Order/ Correction Action Request

NPG-1707-01, Revision 7, dated November 20, 1979; Processing of Safety Concerns

b. 10 CFR Part 21 Report from VEPCo to RII dated December 12, 1979.

c. B&W Preliminary Report of Safety Concern No. PSC-19-79 dated June 8, 1979 and final report dated November 1, 1979, entitled "The Generation and Use of Seismic Response Spectra Using STALUM/RESPECT Computer Codes."

d. The following B&W calculation files to determine B&W's action relating to the identification, investigation, evaluation, and corrective and preventive actions, of the identified deficiency (error) in seismic response spectra calculated using the combination of STALUM and RESPECT computer codes.

   32-7190-00 dated January 18, 1977

   32-1000687-00 dated February 3, 1977

   32-9284-00 dated August 31, 1977

   32-1102124-00 dated June 25, 1979

   32-1102447-00 dated January 20, 1979

   32-10427-01 dated March 9, 1979

   32-1105958-00 dated November 6, 1979

   32-2983-02 dated January 30, 1980

e. STALUM versions 5, 6, and 7 and RESPECT computer code abstracts and the QA Certifications thereof.

f. The following B&W Specifications and Data Transmittal Packages to verify that corrected seismic response spectra was generated and transmitted to all organizations that received the original deficient spectra.

Specifications	Data Transmittal Packages
18-1391001-02	86-2853-00
18-1391000015-03	86-2940-00
	86-5057-00

86-1100397-00

86-110397-02

86-1102488-00

g. The following letters to verify that the deficiency was accurately identified and that all affected organizations were notified by B&W that the deficiency is considered to be reportable to to US NRC.

VEPCo (TWE) to B&W (JEG) dated June 4, 1979

VEPCo (TWE) to B&W (JEG) dated June 6, 1979

TVA (RPP) to B&W (REL) dated June 5, 1979

CPCo (JMB) to B&W (Dist) dated June 19, 1979

B&W (JM) to VEPCo (SCB) dated November 19, 1979

B&W (JM) to TVA (DRP) dated November 15, 1979

4. Findings

a. Deviations, Unresolved Items and Follow-up Items

None identified.

b. Additional Comments

(1) The information examined by the inspector appeared to be complete and accurate.

(2) The cause and effect of the incorrect seismic response spectra appeared to be properly identified, sufficiently evaluated, and adequately documented.

The STALUM computer code was developed by B&W to combine and upgrade the previously used combinations of ST3D and LUM codes. Version 5 of STALUM was verified as yeilding consistent results with the previously used ST3D/LUM codes.

(a) Cause

The STALUM code was subsequently revised (to version 6) to provide additional output information to users of the code (other than as input to the RESPECT code). The revision included a misapplications of the square-root-sum-of squares (SRSS) method of combining forces. The programming error was not detected since only the "new" outputs were verified as correct according to procedures.

(b) Effect

The effect of the misapplications of the SRSS method of combining forces was that the signs of all inertial forces were calculated as positive, when in fact, the calculation would normally yeild some distribution of positive and negative inertial forces. All subsequent piping system response calculations, using the time history analysis method, would therefore be incorrect since the input data to the calculations would not contain the true distributions of the inertial forces.

(c) Corrective Action

The System Mechanical Analysis Unit of B&W (user of the code) notified the Tech Staff Unit (developer of the code) of the error in STALUM Revision 6 in accordance with procedure NPG-0902-06. Although immediate removal of the QA Certifications of the code was required by procedures, the Tech Staff Unit did not remove STALUM-6 from use until the cause of the "error" was found. B&W subsequently deemed that there were no adverse safety consequences resulting from the delay in removing the code from use and that the matter was not reportable either under the requirements of 10 CFR 50.55(e) or under 10 CFR 21.

(3) The corrective and preventive actions taken by B&W were timely (the STALUM computer code was revised (to version STALUM-7) and corrected data transmitted to all affected organizations). To provide assurance against repeated occurrences, quality assurance program procedures NPG-0303-07 and NPG-0902-06 were revised to require that all revisions (versions) of computer codes be fully verified as yeilding correct output (data) for all intended subsequent applications of the data.

(4) The generic aspects were reviewed and all potentially affected organizations were properly notified and, where

appropriate, were transmitted corrected seismic response spectra. (Note: Errors in the seismic response spectra for Consumers Power Company were detected and were corrected prior to the transmission of the data to CPCo.)

D. Containment Overpressurization Due To A Main Steam Line Break

1. Background Information

VEPCo submitted a report to RII under the reporting requirements of 10 CFR 50.55(e). The report states that the containment will be overpressurized (based on a preliminary analysis) in the event of a main steam line break and no cut off of feedwater flow to the affected generator by the operator.

The two SLB (Steam Line Break) accidents analysed by B&W appear to be the basis for VEPCo's safety concern. Except for the use of the currently approved system model for SLB evaluation, TRAP, the overall analytical approach and system actions used by B&W are equivalent to those used in previous SAR licensing submittals. The two postulated accidents can be summarized as follows:

The break is assumed to be a double-ended rupture of a 30" main steam line, inside the containment, with the plant operating at 102% of full power. Following the break, a reactor trip occurs and the ESFAS is actuated. The ESFAS isolates the main feedwater and steam lines and initiates auxiliary feedwater, Thus the affected steam generator will continue to blow down to containment. The auxiliary feedwater from a single turbine-driven pump (1520 gpm runout flow), which can feed both steam generators, is assumed to preferentially flow to the depressurized (affected) steam generator and is boiled off through the break. One of two motor driven pumps (1100 gpm runout flow) also provides flow to the affected steam generator and contributes additional steam flow out the break. This sustained, unmitigated blowdown (assuming no operator action nor automated termination of auxiliary feedwater flow) with a continuous maximum flow of 2620 gpm results in a sustained core return to critical (case one), and, eventually would result in containment overpressurization.

In the second case, the reactor core was assumed to be maintained subcritical by assuming additional shutdown margin, and the auxiliary feedwater flow to the affected steam generator was again assumed to be unmitigated. Unacceptable containment pressures occur in about 10 minutes for this case.

Based on these analyses, VEPCo filed their safety concern recognizing that some form of corrective action was necessary for their plant design in the form of either auxiliary feedwater termination or reduction.

2. Objectives

The objectives of this area of the inspection were to verify that:

a. The available information and documentation is complete and accurate by document review or direct observation.

b. The cause and effect of the deficiency was properly identified, sufficiently evaluated, and documented.

c. The corrective action taken was timely and that preventive measures are being implemented to assure against repeated occurrences.

d. The generic aspects have been reviewed in depth and that affected organizations have been properly notified.

3. Methods of Accomplishment

The preceding objectives were accomplished by review of the following documents.

a. The following Safety Analysis Reports and Plant Parameters Lists to determine the current commitments with respect to automated safeguards systems for mitigating the consequences of a postulated main steam line break with continued feedwater addition.

   BSAR for B&W Standardized Nuclear Steam Supply System - System - Section 7.3.1.1.2

   FSAR for TVA Bellefonte Nuclear Power Plants - Section 7.3.1.1

   BSAR for CPCo Midland Nuclear Power Plants - Section 7.3.3.2.6

   PSAR for VEPCo North Anna Nuclear Power Plants

   "Plant Parameter List" No. 37-6002000003-01 for WPPSS Nuclear Power Plants - Table 17.0.

b. The following NPGD Administrative Manual procedures to determine that the corporate commitments to control the design (including changes thereto), and to timely report, evaluate, and correct significant safety related deficiencies were correctly translated into procedures and instructions to assure implementation thereof:

NPG-0405-22, Revision 5 dated September 20, 1979; Design Review

NPG-0503-04, Revision 9 dated March 3, 1980; Site Problem Report (SPR)

NPG-1703-01, Revision 5 dated November 5, 1979; Preparation and Processing of Internal Deficiency Report/Restraint Order/Corrective Action Request

NPG-1707-01, Revision 7 dated November 20, 1979; Processing of Safety Concerns.

c. IE Bulletin No. 80-04, "Analysis of a PWR Main Steam Line Break with continued Feedwater Addition," dated February 14, 1980.

d. B&W "Preliminary Review of VEPCo SLB (steam line break) Safety Concern," dated September 10, 1979.

e. The following letters to verify that the safety concern was accurately identified, sufficiently evaluated, corrective and preventive measures were identified, and that all affected organizations were notified by B&W of the safety concern and the corresponding B&W recommended actions to be taken by the Licensee.

(1) B&W internal memo from the Accident Analysis Group to the Plant Design Group dated September 10, 1979.

(2) B&W internal memo to all product line Service Managers dated October 11, 1979.

(3) VEPCo letter to B&W (JM) dated October 23, 1979.

(4) B&W letter to owners of B&W 177 series NSSS systems dated November 17, 1979. (Note: Letters were transmitted to Duke Power, Metropolitan Edison, Florida Power, Arkansas P&L, SMUD and Toledo Edison.)

(5) B&W internal memo No. 582-7121-T1.2 dated March 2, 1980.

4. Findings

a. Deviations, Unresolved Items and Follow-up Item

None identified.

b. <u>Additional Comments</u>

(1) The information examined by the inspector appeared to the complete and accurate.

(2) The cause and effect of the potential overpressurization of the Containment appeared to be properly identified, sufficiently evaluated and adequately documented.

(3) The corrective and preventive actions taken by B&W were timely (an option to the B&W ESFAS system design was developed by B&W to "feed only the good generator" (FOGG) which terminates the auxiliary feedwater flow to the steam generator with the main steam line break) and recommendations were transmitted to the licensees to evaluate the potential for containment overpressurization in the unlikely event of a main steam line break and measures that could mitigate the consequences thereof.

(4) B&W management stated that:

(a) The operating plants (177 series) were designed and licensed based upon NRC established licensing requirements existing at the time.

(b) No new information (relative to design errors) has been uncovered in the VEPCo analysis of containment overpressurization due to a postulated main steam line break with continued feedwater addition in a 145 series plant.

(c) All other B&W operating plants were licensed under a myriad of assumptions which constituted an acceptable design basis (in their FSAR) for a steam line break accident. Long term feedwater boiloff to containment following a SLB (Steam line break) was considered in defining these designs. A variety of design and operating concepts for AFW (Auxiliary Feedwater Systems) following a SLB accident were evaluated for these plants. In particular, these systems were not as highly automated as in current designs and they were sized more closely for (only) decay heat removal by a single steam generator. Thus, these plants were able to effectively limit overpressurization of their containments through containment sizing, reactor building cooling capacity, and allowable operator actions.

(d) Changes may have occurred in the operating and design requirements of the AFW systems for the operating plants, either subsequent to the initial operating license or as a result of the TMI-2 accident "lessons learned." Also, licensing requirements for the AFW systems have changed subsequent to each of the operating plants being licensed. In addition, operator actions that were once considered acceptable during the licensing of the plant, may not be deemed acceptable in today's licensing climate. Thus, B&W cannot unequivocally say that there are no additional concerns or that none will be raised in the future by the NRC for the operating plants as a result of the VEPCo's findings. To allow the B&W Owner's to more fully understand the impact of their plant specific AFW system design on system overcooling and containment pressure design, B&W recommends that for each operating plant, the following additional actions be taken by each Licensee:

- Establish the assumptions used in the licensing Safety Analysis Report SLB accident analysis, including operator actions.

- Establish the design bases under which the AFW system was licensed.

- Identify the changes in AFW system design on its operation since the system was initially licensed (such as changes AFW pump capacities).

- Review the impact of any changes in the AFW system design and operation on the licensing Safety Analysis Report.

- Develop a plant specific position regarding backfit of current licensing requirements to the existing AFW system design.

(e) The Licensees were formally advised that B&W would provide engineering and/or analysis services to them on a contract basis.

(4) The generic aspects were reviewed and all affected organizations have been notified of the safety concern and were transmitted B&W's recommendations for evaluating and upgrading the ESFAS and Auxiliary Feedwater Systems of the affected nuclear power plants.

E. Audits

1. Objectives

The objectives of this area of the inspection were to verify that:

a. Audit system is established which has organizational independence, authority, and is documented in procedures and/or instructions in accordance with commitments.

b. Audit records include a written audit plan, team selection, audit schedule, and audit notification to the person or organization to be audited.

c. Members of the audit team are independent of any direct responsibility for the activities being audited.

d. Provisions exist for the reporting of the effectiveness of the Quality Assurance program to responsible management.

e. The audit includes the use of checklists or procedures, detailed audit reports, and timely identification, acknowledgement, documentation of nonconformances, and subsequent corrective action and verification.

f. Audit reports contain the audit scope, identification of auditors, persons or organizations contacted, summary of the results of the audit, the details of any noncomformances noted, the recommendations for correction, and distribution of the report to responsible management.

2. Methods of Accomplishment

The preceding objectives were accomplished by:

a. Review of the following documents to determine if objective above was accomplished:

(1) The following sections of the NRC accepted B&W Topical Report BAW-10096A (Quality Program for Nuclear Equipment) Revision 3 dated April 18, 1977 to determine the B&W corporate QA programmatic commitments relative to quality assurance audits:

2.0 Quality Assurance Program

17.0 Quality Assurance Records

18.0 Audits

Appendix A - Compliance with Applicable Regulatory Guide and ANSI Standards.

(2) The following sections of the B&W NPGD (Nuclear Power Generation Division) Quality Assurance Manual No. 19A-N.1 dated December 7, 1979, through release number 18, to determine if the corporate commitments relative to quality assurance audits were correctly translated into quality assurance requirements and procedures.

2.0, Revision 7 dated November 14, 1979; Quality Assurance Program

18.0, Revision 5 dated November 14, 1979; Audits

(3) The following procedures contained in part "A" of the NPGD Administration Manual, dated February 15, 1980, to determine that the NPGD Quality Assurance Program requirements and procedures were correctly translated into a viable quality assurance audit program.

NPG-0151-01G; Revision 1 dated July 14, 1976, Objectives and Responsiblilities.

NPG-1708-08; Revision 6, dated April 3, 1979, Quality Control Surveillance Inspection by NPGD Supplied Equipment.

(4) The following B&W Quality Assurance Instructions to determine the detailed procedural requirements for planning, scheduling, personnel qualification, preparation, execution, reporting and follow-up of quality assurance audits:

OI-1035, Revision 1; QA Section Monthly Activities Report

OI-1037, Revision 5; QA Audit Record System

OI-1040, Revision 4; QA Supplier Quality Program Audit

OI-1041, Revision 5; QA-NPGD Internal Audits

OI-1077, Revision 1; QA Special Audits/Investigations

OI-1104, Revision 1; QA Deficiency Reporting and Analysis System

OI-1142, Revision 5; QA Audit Program

(5) The NPGD Records Management Program Manual 1E, and the NPGD Records Retention Schedules and File References Manual 1E1, to determine that the NPGD commitments relative to quality assurance records were correctly translated into a viable Division records management program.

b. Review of the following documents to determine if objectives b through f. above were accomplished:

(1) The following quality assurance audits and audit files to determine that the approved procedures, instructions and management programs relative to quality assurance audits are being implemented:

Three (3) Quality Assurance Management Audits

Fourteen (14) NPGD Internal Audits

Three (3) Vendor (External) Audits

(2) Qualification and documentation records for eleven (11) QA Department and four (4) Engineering Department personnel who either are actively performing audits or who performed audits from November 12, 1979, through March 14, 1980.

3. Findings

a. Deviations from Commitment

Two (2) deviations from commitments were identified in this area of the inspection. See Notice of Deviations, Items B and C and the additional comments below.

With respect to Item C, the following additional observations were noted by the inspector:

(1) The scope of the audit was not specifically identified in six (6) of the seventeen (17) internal and vendor audit reports examined by the inspector.

(2) Certification forms and other records for six (6) auditors and lead auditors were not maintained for the time period that the seventeen (17) internal and vendor audits were performed.

(3) The acknowledgement of an understanding of the audit findings by the management of the audited organization at the post audit conference was not documented in the seventeen (17) internal and vendor audit reports examined by the inspector. In addition, the attendees of the post audit conference were not specifically identified in six (6) of them.

b. Unresolved or Follow-up Items

None identified.

c. Additional Comments

(1) Section 1.1 of the B&W Topical Report (April 18, 1977) states that the adequacy of the scope, implementation and effectiveness of the NPGD QA Program is assessed by B&W CQA (Control Quality Assurance) at least every two years.

The only documented assessment of NPGD QA by B&W CQA that was available to the inspector during this inspection was CQA Audit Report number CQA-1979, dated December 14, 1979.

(2) There did not appear to be records or other documentation that the audit team numbers were oriented by the team leader prior to the execution of eight (8) of the seventeen (17) internal and vendor audit reports examined by the inspector.

(3) The Quality Assurance Management acknowledged the non-adherences to procedural and committed ANSI requirements. The management further stated that the current procedural requirements would be reviewed and upgraded as necessary.

F. Exit Interview

An exit interview was held with management representatives on March 14, 1980. In addition to those individuals indicated by an asterisk in paragraph A of each Details Section, those in attendance were:

R. L. Bruce, Manager, Personnel
A. Hall, Special Assignment
R. H. Hide, Manager, Nuclear Parts Center
T. M. Schuler, Engineering
J. S. Tulenko, Manager, Fuel Engineering

The inspector summarized the scope and findings of the inspection. Management comments were generally for clarification only, or acknowledgement of the statements by the inspector.

DETAILS SECTION II

(Prepared by J. M. Johnson)

A. Person Contacted

T. L. Baldwin, Licensing Engineer
B. Hildenberger, Auxiliary Equipment Engineer
*S. Klein, Manager, QA Engineering
G. Lowe, Equipment Engineer
D. Mars, Licensing Engineer
*P. Motiska, QA Engineer
P. Perry, QC Section Leader
H. C. Rush, Level III, Radiographic Testing
C. Taylor, QA Engineer

*Denotes those present at exit interview.

B. Overstressed Letdown Cooler Support Brackets

Follow-up on Construction Deficiency Report for North Anna 3 and 4 reported by Telecon to Region II on January 2, 1980, concerning the letdown cooler support brackets which are overstressed. A similar problem was identified and reported earlier on WPPSS Projects 1 and 4, and a Part 21 Report was issued by Babcock and Wilcox on March 3, 1980.

1. Objectives

The objectives of this area of the inspection were to verify:

a. Deviations have been evaluated and records are maintained.

b. Methods of analysis for defect, deviation or failure to comply are clearly described and responsibilities described, and these methods were followed.

c. The information given to licensees and the NRC is complete and accurate.

d. All items have been identi...ed and generic aspects covered.

e. The cause of deficiency has been identified.

f. Status and adequacy of corrective and preventive actions.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of the following:

a. Babcock and Wilcox Topical Report No. BAW-10096A, Section 15 (Nonconforming Materials, Parts or Components) and Section 16 (Corrective Action) to determine commitments.

b. Babcock and Wilcox Administrative Policies and Procedures Manual, procedure No. NPG-1707-01, Revision 6, Processing of Safety Concerns to determine procedural requirements.

c. 10 CFR Part 21 Report dated March 3, 1980, to NRC Office of Inspection and Enforcement from B&W Licensing Manager, which indicates that NRC was notified by telephone by B&W on February 26, 1980.

d. Preliminary Report of Safety Concern No. 5-79 dated February 8, 1979, and transmittal letter dated February 9, 1979, which states that United Engineers and Constructors (UE&C) identified this deficiency on WNP-1 during a B&W/UE&C meeting on November 10, 1978.

e. Interim Report of Preliminary Report of Safety Concern No. PSC-5-79 dated March 29, 1979, indicating that the Atlas letdown cooler supports may not be adequate and must be evaluated for the loads in the B&W specification for all 205 plants, VEPCo, and Consumer's Power Company, and that this does represent a significant deficiency reportable under 10 CFR 50.55(e) for WNP-1 and WNP-4.

f. B&W memo dated April 11, 1979, from Fluid Systems Engineer to Project Managers for Midland 1 and 2, North Anna 3 and 4, Bellefonte 1 and 2, Pebble Springs, Davis Besse 2 and 3, PASNY, and Erie 1 and 2 requesting installation drawings and details from the respective customers for the support bracket NSSS/BOP (Balance of Plant) interface.

g. Telephone Call Record dated September 13, 1979, from NRC Vendor Branch inspector at Atlas Industrial Manufacturing Company to B&W cognizant engineer which discussed the fact that the 50.55(e) report filed by WPPSS appeared to indicate a generic problem, since B&W had Atlas contracts for the same letdown cooler with an additional seven (7) utilities. B&W explained that they were evaluating for these additional plants.

h. B&W memo dated January 8, 1980, from Licensing Manager to Distribution indicating that the evaluation of the remaining plants utilizing Atlas coolers is complete, and that a reportable Part 21 deficiency exists for Bellefonte 1&2, North Anna 3 & 4, and Midland 1 & 2 (in addition to WPPSS 1&4 already reported by the licensee as a 50.55(e)).

i. Babcock & Wilcox 10 CFR Part 21 Report issued March 3, 1980.

j. Atlas General Arrangement Drawing of Letdown Heat Exchanger (D3378-7) and its approval by B&W and licensee.

k. Atlas Support Point Loading and Nozzle Loads for Letdown Cooler (A-6888 (revised); A-3642 (prior)) showing forces and moments, and their approval by B&W.

3. Findings

a. Deviations and Unresolved Items

In this area of the inspection, no deviations and no unresolved items were identified.

b. Follow-up Items

During a subsequent inspection, examination will be made of the area of design interfaces to assure that adequate design information is transmitted and design interface activities are performed in accordance with established procedures.

c. Additional Comments

(1) The deviations on letdown coolers have been identified and records are maintained at B&W.

(2) Methods of analysis and responsibilities were delineated in procedure HPG-17-7-01, Revision 6, Processing of Safety Concerns, and this procedure was followed. Note that a new revision of the procedure has now been issued which is being followed for safety concerns identified after November 20, 1979.

(3) The information given to licensees and NRC is accurate.

(4) All letdown coolers supplied by Atlas to B&W have been identified. It is unknown whether Atlas is supplying this cooler to any other NSSS. Prior B&W supplied letdown coolers were skirted Graham heliocore coolers, which did not have this problem. The only similarly designed equipment

currently is the Seal Return Cooler (Class II) and B&W stated the problem did not exist on these.

(5) The cause of the WPPSS problem is stated in the Interim Report as follows: "The inadequacy is due to a combination of factors including:

(1) Limited support provided by the A/E's current design; (2) marginal thickness in the cooler base plate, and (3) inadequate and unspecified requirements for installation of the cooler."

This is primarily a design interface problem. One half of the cooler support bracket is provided by the NSSS (designed by Atlas); the other half by the licensee (designed by architect-engineer (AE) or licensee). These parts mate to provide the support. Originally it was presumed that the problem existed only on plants in which the cooler was mounted vertically. However, the same problem was found to exist on Bellefonte, in which the cooler is mounted horizontally. In each case, the licensee/ AE scope support platform did not underlap the NSSS scope support bracket, making the support a cantilevered beam or fulcrum with heavy stresses. The weight of the cooler is 4567 lbs. empty and 6295 lbs. full. The support material is 1/2" SA 36 plate.

Confirmatory calculations for WPPSS showed that specified loadings will cause bending stresses at the bolt line of the lower support of approximately three (3) times the AISC allowable for the base material. Because four different AE/licensees were involved in the mating design (UE&C for WPPSS; Bechtel for Midland; Stone and Webster for North Anna; and TVA for Bellefonte), and the interfacing designs resulted in a significant reportable deficiency in each case, it appears that the method for mounting the cooler may not have been explicitly detailed on the drawings supplied by B&W.

(6) Corrective action has been completed for WPPSS, which included the addition of three more bolts to redistribute the load on the base plate and the redesign of the licensee supplied support stand. As indicated in the Part 21 report to the NRC, corrective action has not been taken yet for North Anna, TVA, and Midland. Evaluation is underway for Midland and North Anna, and B&W has recommended the same corrective action to Bellefonte as

was taken at WPPSS. For Pebble Springs, B&W will provide the AE with an updated support design later. Erie, PASNY and Davis Besse 2 & 3 have been cancelled.

The preventive action indicated is that henceforth cooler drawings will show the required support arrangement to be provided by the AE/licensee.

(7) Although B&W evaluation of the problem and issuance of a Part 21 report was slow considering the fact that the problem was identified by UE&C on November 10, 1978, and determination of reportability for the WPPSS project was made on March 28, 1979, no finding was issued because Part 21 is silent on evaluation time. The time requirements which begin with notification to the B&W responsible officer were met.

C. Rejectable Defects on W-K-M Valves

Follow-up on a 50.55(e) Report to Region V related to rejectable defects identified during WNP 1 and 4 inspection of W-K-M valves, and on one valve returned to W-K-M for repair. Six decay heat removal valves and one core flood valve are involved. Babcock and Wilcox issued a Part 21 report on December 17, 1979. Also, another 50.55(e) report was issued by WPPSS concerning these valves due to the fact that the retaining ring for the Marotta poppet valve attached to the plug/seat was of carbon steel rather than stainless as required.

1. Objectives

The objectives of this area of the inspection were to verify:

a. Deviations have been evaluated and records are maintained.

b. Methods of analysis for defect, deviation or failure to comply are clearly described and responsibilities described, and these methods were followed.

c. The information given to the licensee and the NRC was complete and accurate.

d. All items have been identified and generic aspects covered.

e. The cause of the deficiency has been identified.

f. Status and adequacy of corrective and preventive actions.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of the following:

a. Babcock and Wilcox Topical Report No. BAW-10096A, Section 15 (Nonconforming Materials, Parts or Components) and Section 16 (Corrective Action) to determine commitments.

b. Babcock and Wilcox Administrative Policies and Procedures Manual, procedure No. NPG-1707-01, Revisions 6 and 7, Processing of Safety Concerns, and NPG-0503-04, to determine procedural requirements.

c. Babcock and Wilcox QA Manual N 19A-N.1 Section 16, Corrective Action, to determine procedur requirements.

d. Equipment Specification invoked by Purchase Order No. 029572 for Remotely Operated Valves for Auxiliary System Service, which states in Section 2.4 "All valves shall be of a backseating type" and requires Class I and Class II valves ranging in size from 2" to 14".

e. B&W letter dated December 17, 1979, reporting 10 CFR Part 21 reportable defects in a core flood system isolation valve and possible defects in six (6) decay heat removal isolation valves supplied to WPPSS. This letter indicates that all customers supplied by B&W with W-K-M valves have been notified and also that a verbal report was made to NRC on December 14, 1979.

f. B&W Preliminary Safety Concern No. (PSC) 58-79 concerning reportable defects in W-K-M valve bodies, and transmittal memo dated November 29, 1979.

g. B&W Preliminary Safety Concern Report dated December 13, 1979, for PSC 58-79.

h. ACF Industries and W-K-M Division, letter dated December 14, 1979, to B&W concerning valve body reportable defects and repair.

i. Notifications to the affected licensees (i.e. WPPSS, PASNY, TVA, and PGE) by B&W.

j. B&W Vendor Surveillance Inspection Reports dated February 13-16, 1979, February 20-24, 1979, March 8-9, 1979, October 2-4, 1979, and December 14, 1979, to assure performance of vendor surveillance at W-K-M.

k. B&W Audit Reports of W-K-M nos. 79-6 and 80-7 (which identifies five (5) deficiencies in Non-Destructive Examination (NDE) areas).

l. January 17, 1980 hold on shipment of PASNY valves pending disposition of WPPSS valves.

m. B&W W-K-M Repair Plan for reportable defects in bodies, dated February 5, 1980, issued for WPPSS approval.

n. B&W memo dated January 23, 1980, concerning RT (radiographic testing) coverage of valves by PRL (Pennsylvania Radiological Laboratory) and Precision Inspection Limited, who performed upgrade on valve bodies, including NDE examination, for W-K-M.

o. PSC 46-79 concerning W-K-M valves with Marotta valve retaining clips of carbon steel rather than required stainless steel.

p. Marotta letter to W-K-M dated July 17, 1979, notifying W-K-M of shipment of carbon steel rather than stainless steel retaining clips.

q. W-K-M/ACF letter to B&W dated August 21, 1979, notifying B&W of the retaining ring problem.

r. B&W letter No. BWUE 80-5023 dated January 29, 1980, to WPPSS concerning reportability under 10 CFR 50.55(e) of Nonconforming Retaining Clips in Gate Valves and concluding that a loose poppet valve (after corrosion of the carbon steel retaining ring) could degrade the performance of the Decay Heat Removal Pump (for valves DH-V11A&B and DH-V12A&B) or inhibit PORV closure (valve RC-V10).

3. Findings

a. Deviations and Unresolved Items

In this area of the inspection, no deviations and no unresolved items were identified.

b. Follow-up Items

One follow-up item was identified, as follows:

Examination will be made at a later date to assure that the cause of the valve body rejectable indications is identified, documented, and reported to management, and that corrective and preventive actions are completed.

c. Additional Comments

(1) The identified deviations are being evaluated by B&W and records are maintained.

(2) The methods of analysis for defects, deviations or failure to comply are described in B&W procedures and these procedures are being followed.

(3) The information given to the licensees and the NRC is accurate and complete to date.

(4) All B&W supplied items have been identified and B&W generic aspects covered. It is unknown if W-K-M is supplying similar valves to any other AE or NSSS. See Inspection Report 9990038/80-01 of W-K-M for additional information.

(5) The cause of the use of carbon steel retaining rings instead of stainless was clearly a vendor problem, identified initially by Marotta, and was the result of an incorrect part number on a Marotta drawing and use of the incorrect part (carbon steel) in fabrication and assembly. Since Marotta was a subsupplier to W-K-M, surveillance and audits were not performed by B&W. The valve body cracks were identified visually by W-K-M when the Core Flood Isolation Valve was in the W-K-M shop for replacement of the retaining ring. The rejectable indications were identified by WPPSS during pre-service inspection of the Decay Heat valves. B&W indicated that their preliminary evaluation of the cause of the cracks in the Core Flood valve was inadequate coverage by radiography of the valve body during upgrading. Note that these, by report, were off-the-shelf valve bodies which were upgraded to nuclear requirements by companies under contract to W-K-M (Pennsylvania Radiographic Laboratories and Precision Inspection Limited). Upgrading consisted primarily of NDE and weld repair of defects, if any. No liquid penetrant NDE examination was required for the Class II Core Flood valve. Both radiography and liquid penetrant examinations were required for the Class I Decay Heat valves, but both were performed during upgrading and prior to hydrotesting. Preliminary evaluation by B&W is that hydrotesting may have caused the surface indications subsequent to liquid penetrant testing. However, the causes and scope of both conditions in the valve bodies are still under evaluation at B&W, and no final determination of cause has yet been made, nor final evaluation of why the defects were not identified prior to shipment by W-K-M or by B&W vendor surveillance.

(6) Corrective and preventive actions related to the carbon steel retaining rings have been completed by B&W. The carbon steel rings have been replaced by stainless rings for WPPSS and PGE, the only affected B&W projects. For the valve body cracks and rejectable indications, neither cause nor total scope are fully known at this point. Corrective and preventive actions have not yet been taken. A plan for corrective and preventive action has been submitted by B&W to WPPSS for approval, which includes repair, radiography and evaluation of areas missed in the original radiographic examination for certain valves, and liquid penetrant examination for Class I valves. Preventive action proposed for WPPSS valves includes liquid penetrant after hydrotest and prior to assembly for Class I valves, and demonstration of W-K-M liquid penetrant testing per comparator block. Also, the specification will be changed to more clearly delineate acceptance criteria for visual inspection of valve bodies.

Further examination will be made of the W-K-M valves supplied to PASNY, TVA and PGE to determine whether rejectable indications are present, and the PASNY valves are on hold at W-K-M pending this evaluation. Also, deficiencies identified in B&W audit report No. 80-7 of W-K-M in the NDE area require correction because they could affect performance of satisfactory NDE, as indicated by the findings that (1) a Level I radiographer had signed RT (Radiographic Testing) reports as a Level II, and (2) certification for the Level III inspector for RT, PT (Liquid Penetrant Testing) and MP (Magnetic Particle Testing) expired June 30, 1979, and (3) there is no record of demonstration to the ANI (Authorized Nuclear Inspector) of any NDE procedures as required by the W-K-M QA manual and ASME Code.

D. Evaluation of Supplier Performance

1. Objectives

The objectives of this area of the inspection were to verify that procedures have been established and implemented that provide for:

a. Establishing that the purchaser and supplier understand the provisions and specifications of the procurement documents, as applicable to specific projects.

b. Requiring the supplier to identify planning techniques and processes to be utilized in fulfilling procurement requirements.

c. Reviewing documents which are generated or processed during activities fulfilling procurement requirements.

d. Identifying and processing necessary change information.

e. Establishing exchange method of document information between purchaser and supplier.

f. Initiation of pre- and post-award activities, as necessary, and as applicable to specific projects.

g. Control, handling, and approval of supplier generated documents.

h. Control of changes in items or services.

2. Method of Accomplishment

The preceding objectives were accomplished by an examination of the following:

a. Babcock and Wilcox Topical Report No. BAW-10096A, Sections 4 (Procurement Document Control), 7 (Control of Purchased Material, Equipment and Services), and 15 (Nonconforming Materials, Parts and Components) to determine commitments for WPPSS project.

b. Bellefonte Nuclear Plant Final Safety Analysis Report (FSAR), Chapter 17, Table 17.1.A-3, and Section 17.1.A.2.1.1 which commits to Revision 0 of the NRC Gray Book, to determine commitments.

c. B&W QA Manual for TVA 19A-N.1, sections 3, 4, 7, and 15 to determine procedural requirements for TVA.

d. B&W QA Manual 19A-N.1, section 3, 4, 7, and 15 to determine procedural requirements for WPPSS projects.

e. B&W Administrative Policies and Procedures Manual, procedure no. NPGD-0412-67, Processing Non - NPGD Prepared Technical Documents, to determine procedural requirements for TVA and WPPSS.

f. Documents related to Purchase Order (PO) 031990LF and change orders for Reactor Trip Switch Station from Vitro including:

(1) QA Data Sheet to determine required submittals.

(2) Vitro Assembly Drawing No. 2691-1001 to determine B&W and TVA Approval.

g. Documents related to the procurement of Reactor Trip Switch Station from Vitro for TVA on Purchase Order (PO) No. 031990LF including:

(1) Change Orders

(2) QA Data Sheet to determine required submittals.

(3) B&W Historical Document List (HDL) to determine submittals and status.

(4) Vitro Assembly Drawing No. 2691-1001 to determine B&W and TVA approvals.

(5) Vitro Outline and Installation Drawing No. 2691-1015 (B&W No. 0210306NB) and B&W and TVA approvals.

(6) Reactor Trip Switch Acceptance Test and approvals.

h. Documents related to the procurement of software package to Design and Test Calculating Module (CM) Software for Nuclear Instrumentation - Reactor Protection System (RPS) from Vitro for TVA on PO No. 032852:

(1) Qualification Report No. 58-0326-01.

(2) Specification No. SP-2915.0100, Revision B (no approvals yet).

i. Documents related to the procurement of Remotely Operated Valves for Auxiliary System Service from W-K-M for WPPSS on PO No. 029572LN:

(1) Source visitation plan for source inspection, including witness points, and Inspection Report documentation of witness activities to determine identification of processing and hold points.

(2) Historical Document List (HDL) to determine document submittals and status.

(3) W-K-M outline drawing No. 0210306NB.

(4) Radiographic Examination Procedure for Castings and Weld Repairs (B&W No. 54-8089-00/W-K-M No. 73-0012-M915) to determine B&W approval (not stamped but transmittal shows approval).

(5) Liquid Penetrant Examination Procedure (B&W No. 54-8086-00/ W-K-M No. 73-0014-M906) to determine B&W review and approval (not stamped but transmittal document shows approval).

(6) Contract Variation Approval Requests (CVARs) and B&W approval:

CVARs Nos. 87-2072-00; 87-2077-00; 87-1873-00; 87-1874-00, 87-3063-00 and 87-1791-00 (this one was unavailable during this inspection).

j. Documents related to the procurement of Letdown Heat Exchanger from Altas for WPPSS:

(1) Atlas General Arrangement Drawing - Letdown Heat Exchanger, No. D3378-7 to determine review and approval by B&W.

(2) Atlas Support Point and Nozzle Loads, No. A6888 and B&W approvals, and No. A3652 (revised) and B&W approval.

3. Findings

a. Deviations

There were no deviations identified in this area of the inspection.

b. Unresolved Items

One unresolved item was identified as follows:

It is not clear that B&W review and approval of changes is equivalent to original design approval for CVARs (Contract Variation Approval Requests) which appear to make design changes. For example, CVARS 87-2077-00 and 87-2072-00 submitted by W-K-M and approved by B&W permit the vendor to supply non-backseating valves (with modification) instead of backseating valves as required by the B&W PO and technical specification.

c. Follow-up Items

Follow-up items were identified as follows:

(1) Further examination will be made during a subsequent inspection to determine why the HDL lists CVAR 87-1791-00 dated May 19, 1978, with status BA (in

house for review), but no record or copy of the CVAR could be located during the time available in this inspection. The CVAR is to PO 029572LN (W-K-M valves).

(2) A follow-up item was identified to determine how many projectized B&W QA Manuals there are and whether all are numbered 19A-N.1, and what differences of substance exist.