UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-390/88-01 and 50-391/88-01

Licensee: Tennessee Valley Authority
6N11 B Missionary Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-390 and 50-391 License Nos.: CPPR-91 and CPPR-92

Facility Name: Watts Bar 1 and 2

Inspection Conducted: January 4, 1988 - April 4, 1988

Inspectors: G. A. Walton — 29 Apr. 88
G. A. Walton, Senior Resident Inspector — Date Signed
Construction

G. A. Walton for — 29 Apr. 88
T. B. Powell, Resident Inspector — Date Signed

Approved by: S. A. Elrod — 26 APR 88
S. A. Elrod, Section Chief — Date Signed
Office of Special Projects

SUMMARY

Scope: This routine inspection was conducted in the areas of licensee action on previous enforcement items, licensee action on previously identified inspection findings, fire prevention and fire protection, preoperational test program implementation verification, testing of pipe supports and restraints, instrument air system, cable tray installations, detailed control room design review, welding, circuit breakers, and allegations.

Results: Two violations, one involving failure to follow procedures and another involving failure to implement proper design control. Three unresolved items concerning control air quality, justifications for invalidating CAQRs, and reportability determinations for CAQRs.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

G. Toto, Site Director
*E. Ennis, Plant Manager
*R. Pedde, Deputy Site Director
G. Atwood, Division of Nuclear Engineering
*K. Ashley, Division of Nuclear Engineering
H. Bounds, Division of Nuclear Engineering
M. Brickey, Division of Nuclear Engineering
*J. Coan, Assistant Project Engineer
*J. Cromer, Project Engineer
G. Curtis, Assistant Project Engineer
*T. Dean, Compliance/Licensing
*J. Gibbs, Assistant Project Engineer
*T. Horst, Nuclear Site Representative
H. Johnson, Acting Site Quality Manager
*D. Kulisek, Acting Site Licensing Manager
*W. Leslie, Division of Nuclear Engineering
*T. McGrath, Manager of Projects
P. Metcalf, Division of Nuclear Engineering
*L. Peterson, Quality Control Supervisor
*H. Simpson, Manager of Special Projects
S. Stagnolia, Modifications Manager
*D. Stewart, Assistant Site Director
*J. Thompson, Construction Manager
*R. Tolley, Project Manager's Office

Other licensee employees contacted included engineers, technicians, nuclear power supervisors, and construction supervisors.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on April 4, 1988, with those persons indicated in paragraph one above. The inspectors described the areas inspected and discussed in detail the inspection findings listed below. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Note: A list of abbreviations used in this report is contained in paragraph 16.

Item Number	Status	Description/Reference Paragraph
390/88-01-01 391/88-01-01	Open	Violation - Failure to Follow Procedures (paragraph 15)
390/88-01-02 391/88-01-02	Open	Violation - Cable Tray Installation (paragraph 10)
390/88-01-03 391/88-01-03	Open	URI - Control Air Quality (paragraph 9)
390/88-01-04 391/88-01-04	Open	URI - Justification for Invalidating CAQRs (paragraph 15)
390/88-01-05 391/88-01-05	Open	URI - CAQR Reportability Determination (paragraph 15)
390/85-52-01 391/85-42-01	Closed	Violation - Inadequate Storage of Valves (paragraph 3b)
390/85-52-02 391/85-42-02	Closed	Violation - Failure to Follow HVAC Fabrication, Installation, and Inspection Procedures (paragraph 3c)
390/85-18-02	Closed	Violation - Corrective Action for Conduit Support CS-R1-1497 (paragraph 3d)
390/87-07-01 391/87-07-01	Open	Violation - HVAC Duct Supports (paragraph 3e)
390/85-63-02	Closed	URI - Welder Performance Qualification (paragraph 3f)
391/81-13-01	Closed	Violation - Failure to Calibrate Diesel Generator Temperature and Pressure Instruments During Preoperational Testing (paragraph 3g)
390/87-12-02	Open	URI - Performance of Preventive Maintenance Assignments (paragraph 3a)
391/84-50-01	Closed	IFI - Monitor Accumulated Dose in Operation Support Center (paragraph 5a)
391/84-50-02	Closed	IFI - Iodine Source Term Errors (paragraph 5b)
391/85-48-11	Closed	IFI - TMI II.E.1.2, Auxiliary Feedwater Initiation and Flow (paragraph 5c)

TMI II.E.1.2 (Units 1 & 2)	Open	Auxiliary Feedwater Initiation and Flow (paragraph 5c)
390/85-20-06	Closed	IFI - Site Procedures for Packaging and Shipping of Radwaste (paragraph 5e)
390/87-FRP-01	Closed	IFI - TMI II. F. 1. (3) (paragraph 5e)
390/CDR-82-80 391/CDR-82-76	Open	CDR - Shielded Power Cable Bend Radius Deficiency (paragraph 5d)
390/CDR-85-44 391/CDR-85-43	Closed	CDR - Minimum Bend Radius Cable Deficiencies (paragraph 5d)
390/CDR-85-63 391/CDR-85-59	Closed	CDR - Failure to Inspect Installed Cables for Bend Radius (paragraph 5d)
390/84-09-04	Closed	IFI - Calibration of Containment High Range Radiation Monitors (TMI II. F. 1. (3)) (paragraph 5e)
390/84-09-05	Closed	IFI - Review Procedures (TMI II. F. 1. (1)) (paragraph 5e)
Bulletin 81-03	Closed	Flow Blockage of Cooling Water to Safety System Components By Corbicula (paragraph 3h)

3. Licensee Action on Previous Enforcement Items (92702)

a. (Open) URI 390/87-12-02, "Performance of Preventive Maintenance Assignments ". During inspection 390/87-12, the following issues were identified:

- PM was performed on Safety Injection Pump 2-063-PMP-15-B on November 15, 1985, using Rev. 3 of the PM assignment sheet when Rev. 4 had been issued on October 30, 1985.

- Insulation resistance test results were not being compared with previous results as required by QCP 1.52, Rev. 5, "Preventive Maintenance", Paragraph 7.1.2.1.1.1.

A URI closure package was provided to the inspector on January 29, 1988. The closure package indicated the inspector had misread the revision number. During the inspector's review of the closure package, a copy of the PM Assignment Sheet, Rev. 4, dated October 30, 1985, was obtained. Further inspection confirmed that on November 15, 1985, the wrong revision of the PM Assignment Sheet had been used. It appears this was caused by PM assignment sheets not being controlled documents. Today, PMs are performed in accordance with CEP 1.52, Rev. 0, (formerly QCP 1.52) "Preventive Maintenance", which

requires PM assignment sheets to be controlled. No violation is being issued for this failure to use the proper revision of the PM assignment sheet for the following reasons:

a. It occurred over two years ago and a review PM records has shown that it affected only the Unit 2 Safety Injection Pumps.

b. There was no significant difference between the PM actions required by the two revisions.

c. The current revision of CEP 1.52 requires PM assignment sheets to be controlled documents.

d. The current revision of the Nuclear Quality Assurance Manual (NQAM) Part I, Section 2.6, Rev. 2, "Document Control", requires work to be performed using the latest information and requires the user to verify the correct procedure revision is being used.

Concerning the insulation resistance results, the closure package indicated that QCP 1.52, Rev. 5 stated "previous results should be compared" and the word "should" did not mean this practice had to be performed. It also indicated that CEP 1.52 did not require the comparison. The inspector reviewed CEP 1.52, which indicated that insulation resistance tests are required to be performed in accordance with IEEE 43-1974, "Insulation Resistance of Rotating Machinery", or the vendor's manual (whichever is most stringent). IEEE 43 states that the insulation resistance history of a given machine, made and kept under uniform conditions so far as controllable variables are concerned, is recognized as a useful way of monitoring the insulation conditions. It was determined that insulation resistance history was not being used to monitor insulation condition. In subsequent discussions the licensee advised the inspector that a procedure change would be made that included trending of the insulation resistance results. This URI will remain open pending review of the procedures changed.

Standard Practice WB 11.5, Rev. 7, "Plant Handling of Inspection and Audit Findings", indicates plant supervisors are to ensure factual information is provided to the NRC. The above URI is considered an example where the information supplied to the inspector was not accurate. Another example occurred with a closure package provided to the inspector for IFI 390/85-03-01; 391/85-04-01, "Deficient Training Documentation". See inspection report 390, 391/87-15. This closure package indicated the problems with DNE training records had been corrected by Rev. 1 to NEP 1.2, "Training", which was issued on January 19, 1987. During Inspection 390, 391/87-15, it was determined that Rev. 1 to NEP 1.2 appeared adequate but had not been implemented at the plant. Subsequent inspection led to the issuance of TVA-generic Violation 438, 439/87-09-01; 259, 260, 296/87-38-01; 327, 328/87-68-01; 390, 391/87-18-01, "DNE Training". These two examples of failure to provide complete and accurate information to

the NRC staff indicate a need for the licensee to improve the programs controlling the information given to NRC.

b. (Closed) Violation 390/85-52-01, 391/85-42-01, "Inadequate Storage of Valves". During Inspection 390/85-52, 391/85-42, ASME N-stamped valves were found stored outdoors - which violated the requirements of ANSI N45.2.2 - 1972, "Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the Construction Phase". The licensee's response committed to storing the valves indoors, retraining personnel, and periodic housekeeping inspections. The inspector toured the construction warehouse during this inspection period to determine the adequacy of items stored in the warehouse. The stored items were found orderly and no violations of ANSI N45.2.2 were identified. The corrective action taken in response to this violation appears adequate, therefore, this violation is closed.

c. (Closed) Violation 390/85-52-02, 391/85-42-02, "Failure to Follow HVAC Fabrication, Installation and Inspection Procedures". During Inspection 390/85-52, 391/85-42, the following deficiencies were identified:

- Loose bolts on HVAC duct flanges
- Inadequate welds on supports
- Support dimensions did not conform to the applicable drawing

The licensee's response to the violation indicated the cause was inattention to detail and failure of the craftsmen and QC inspectors to follow procedures. The licensee verified the HVAC duct bolts in the room where the violation was found and changed inspection procedures to include acceptance criteria for tightening duct bolts. QC Inspectors were retrained in the acceptance criteria for supports. Subsequent Violation 390,391/87-07-01 (item e below) identified discrepancies in HVAC supports. In response to this later violation, the licensee is performing additional inspections to determine the extent of HVAC support problems. The inadequate welds and support dimension errors will also be addressed. Since all HVAC Duct issues will be tracked by Violation 390, 391/87-07-01, Violation 390/85-52-02, 391/85-42-02 is closed.

d. (Closed) Violation 390/85-18-02, "Corrective Action for Conduit Support CS-R1-1497". During inspection 390/85-18 conduit support CS-R1-1497 was found deficient after it had passed QC inspection. It was determined that this condition was caused by rework packages that did not require retesting or inspection. The support was reworked by NCR 5974. Currently, work is controlled by CEP 1.60, Rev. 0, "Work Control", and tests and inspections are tracked by QCI-1.40, Rev. 7, "Records Accountability Programs". These procedures task the responsible engineer with assigning appropriate retests or

reinspections. The inspector has reviewed construction work packages and is satisfied that appropriate tests and inspections should be assigned as part of the work planning process if QCI - 1.40 and CEP 1.60 are followed. In addition, the licensee has established, by directive of the Watts Bar Program Team, a special program for evaluating the adequacy of conduit supports. This item is closed.

e. (Open) Violation 390, 391/87-07-01, "HVAC Duct Supports". The mechanical discrepancies on HVAC supports identified in this Violation resulted in the licensee performing sample walkdowns of supports to establish the degree of discrepancies which could exist on other HVAC supports. The sample walkdown was conducted utilizing the guidelines in NCIG-02. A multiple statistical plan was established which included 64 hangers in the initial sample. The acceptance criteria was established by utilizing existing design drawings and existing Quality Control Procedures. Of the first 42 hangers inspected, 16 failed to meet the acceptance Criteria. Since this rejection rate far exceeds the allowed limits, the inspections were discontinued. The CAQR WBN 870316, Rev. 3 was updated to include the specific findings of the 16 supports which failed to meet the appropriate acceptance standards and submitted to engineering for evaluation. This item will remain open.

f. (Closed) URI 390/85-63-02, "Welder Performance Qualification". This URI questioned the licensee's disposition as nonsignificant of an NCR, regarding welder qualification. The NCR identified one welder, qualified to a thickness range of 0.0262 inches - 0.318 inches, that welded pipe welds with a maximum thickness of 0.343 inches.

The licensee determined the welder (symbol 6GQQ) was qualified by reviewing the radiographs on three Production welds as allowed by the ASME B&PV Code, 1986 Edition, Section IX, Paragraph QW-304.1. These welds were found acceptable and were used to certify the welder to a thickness range of 0.0262 inches - 0.674 inches. Additionally, as of January 13, 1986, the licensee had reviewed 105,365 welding operation sheets for adequacy of welding, including thickness range. Sixteen NCRs involving 115 ASME welds identified problems with welder qualification, however, none involved weld thickness. This item only involved one welder who was subsequently qualified without additional testing being required. Therefore, this item is considered nonsignificant and is closed.

g. (Closed) Violation 391/81-13-01, "Failure to Calibrate Diesel Generator Temperature and Pressure Instruments During Preoperational Testing". During Inspection 391/81-13, it was identified that diesel generator temperature and pressure instruments were not properly calibrated and were not entered into a program which would ensure recalibration at the proper periodicity. This appears to have been caused by an administrative oversight when system responsibility was

transferred from construction to operations. The inspector reviewed program changes since the time this violation occurred. AI -6.2, Rev. 12, "Preoperational Testing", now requires that all installed process instrumentation utilized to verify acceptance criteria during preoperational testing be verified to be within its calibration frequency before the start of testing. AI-9.1, Rev. 3, "Maintenance Program", requires transferred instrumentation to be rescheduled for calibration within the interval specified by the Prime Instrument Calibration Status Program (PICSP). Since procedural controls on transferred instruments have been initiated, this violation is closed.

h. (Closed) Bulletin 81-03, "Flow Blockage Of Cooling Water To Safety System Components By Corbicula".

The inspector reviewed the licensee's action relative to response and commitment made to IE Bulletin 81-03. The bulletin required the licensee to determine whether Corbicula or Mytilus were present in the vicinity of the station.

The licensee's response indicated the Asiatic Clam (Corbicula) had become prominent in the vicinity of the Watts Bar site and would be controlled with low-level applications of chlorine during the clam spawning season (May through October). The licensee has advised the inspector that chlorine, injected at the intake pumping station, will treat the essential raw cooling water system and the fire protection system (the only two safety-related systems affected).

The inspector verified that the licensee initiated the program in 1981 and is currently continuing it. This item is closed.

4. Unresolved Items

Unresolved Items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. Three Unresolved Items were identified during this inspection and are discussed in paragraphs 9 and 15.

5. Licensee Actions on Previously Identified Inspection Findings (92701)

a. (Closed) Inspector Followup Item (IFI) 391/84-50-01, "Monitor Accumulated Dose in Operations Support Center (OSC)". During inspection 390/84-69; 391/84-50, it was determined there was no centralized method of monitoring accumulated dose of OSC personnel. During inspection 390/87-06, IFI 390/84-69-01 was closed. It has been determined 391/84-50-01 was not closed due to an administrative oversite. This item is closed.

b. (Closed) IFI 391/84-50-02, "Iodine Source Term Errors". During inspection 390/84-69; 391/84-50, it was identified the source term for released radioiodine was in error. During inspection 390/87-06, IFI 390/84-69-02 was closed. It has been determined 391/84-50-02 was not closed due to an administrative oversight. This item is closed.

c. (Closed) IFI 391/85-48-11, "TMI Item II.E.1.2, Auxiliary Feedwater Initiation and Flow". During inspection 391/85-48, TMI Item II.E.1.2 was identified as this IFI. During inspection 390/84-20, the licensee's response for this TMI item was reviewed along with SER paragraph 7.8.2. At that time, this TMI item was closed for Unit 1.

During this inspection, item TMI II.E.1.2 was reviewed again. It was identified that the licensee's response to this TMI item had committed the auxiliary feedwater control circuitry, including the automatic initiating circuitry, to be safety-grade, class 1E. The FSAR, in Table 7.1-1, Note 4, and in Section 8.1.5.3, Note 4, indicates class 1E equipment is qualified in accordance with IEEE 323-1971, "General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations". A review of Auxiliary Feedwater records did not identify any qualification documents that met the requirements of IEEE 323.

Failure to document Class 1E qualification is an additional example of Deviation 390, 391/87-20-01 and will be reviewed in conjunction with closure of that deviation.

TMI item II.E.1.2 is reopened for both units. IFI 391/85-48-11 is administratively closed.

d. (Open) CDR 390/82-80; CDR 391/82-76, "Shielded Power Cable Bend Radius Deficiency".

(Closed) CDR 390/85-44; CDR 391/85-43, "Minimum Bend Radius Cable Deficiencies".

(Closed) CDR 390/85-63; CDR 391/85-59, "Failure to Inspect Installed Cables for Bend Radius".

The cable bend radius issue was first identified as a URI by an NRC inspector during an inspection on June 7-10, 1982, at the Bellefonte Nuclear Plant. Subsequently, the URI was upgraded to Violation 438,439/82-22-01.

Between June 1982 and August 1982, Watts Bar Nuclear Plant reported the subject deficiency in accordance with 10 CFR 50.55(e) as CDR 390/82-80, 391/82-76. TVA submitted interim reports on August 30 and December 23, 1982. The final report for Watts Bar was submitted on September 13, 1983. NRC inspectors performed an inspection at Watts Bar on September 13-16, 1983, and closed CDR 390/82-80 and 391/82-76

based on information contained in the licensee's final report and supplemental discussions with licensee personnel.

A letter from NRC Region II to TVA, dated August 15, 1986, indicated, as a result of reviewing TVA employee concerns, that the final report for the subject CDR was incomplete. Technical content evaluation for reportability in accordance with 10 CFR 50.55(e) appeared, for this CDR, to be inadequate. Therefore, CDR 390/82-80; CDR 391/82-76 is reopened.

A TVA letter dated November 4, 1987, indicated that a final report on the entire cable bend radius issue would be issued on or about November 15, 1988. This letter also requested the issues identified in CDRs 390/85-63, 391/85-59, and 390/85-44, 391/85-43 be included in CDR 390/82-80, 391/82-76. Therefore CDRs 390/85-63, 391/85-59 and 390/85-44, 391/85-43 are closed and their respective issues are included in CDR 390/82-80, 391/82-76 which is being reopened in this report.

e. (Closed) IFI 390/84-09-04, "Calibration of Containment High Range Radiation Monitors - TMI II.F.1.(3)";

(Closed) IFI 390/84-09-05, "Review Procedures - TMI II.F.1.(1)";

(Closed) IFI 390/85-20-06, "Site Procedures for Packaging and Shipping of Radwaste";

(Closed) IFI 390/87-FRP-01, "TMI II.F.1.(3)".

These IFIs resulted from pre-operational Health Physics (HP) inspections from 1982 to 1985. Because of the subsequent delay in licensing Unit 1, HP inspections are deferred, providing no timely opportunity to review these items for closure. Due to the length of time that will have transpired between the previous HP pre-operational inspection program and projected fuel load, it is intended to perform the complete HP pre-operational program again as the new license date nears. For these reasons, these four IFIs are administratively closed.

6. Fire Prevention and Fire Protection - Unit 2 (42051)

During plant tours, the inspectors conducted observations of fire prevention and protection activities in areas containing combustible materials where ignition of these materials could damage safety - related structures, systems or components. The observations included verification that applicable requirements of Administrative Instruction (AI) 9.9, Rev. 14,"Torch Cutting, Welding, and Open Flame Work Permit", Security Procedure 2, Rev. 27, "Fire Protection Plan", AI 1.8, Rev. 12, "Plant Housekeeping"; and WBNP Construction Engineering Procedure (CEP) 1.36, Rev. 1, "Housekeeping" were being implemented with regards to fire prevention and protection.

Within this area, no violations or deviations were identified.

7. Preoperational Test Program Implementation Verification - Unit 1 (71302)

The inspectors conducted routine tours of the facility to make an independent assessment of equipment conditions, plant conditions, security, and adherence to regulatory requirements. The tours included a general observation of plant areas to determine if fire hazards existed and observation of other activities in progress, e.g., maintenance and preoperational testing, to determine if they were being conducted in accordance with approved procedures. Also observed were other activities which could damage installed equipment or instrumentation. The tours included evaluation of system cleanliness controls and a review of logs maintained by test groups to identify problems that may be appropriate for additional followup.

Within this area, no violations or deviations were identified.

8. Testing of Pipe Supports and Restraint Systems - Unit 1 (70370C)

The inspector toured areas of the Unit 1 auxiliary building and reactor building. Numerous snubbers and restraints were observed. Visual examinations were conducted to check for deterioration and physical damage of mechanical snubbers. Visual examinations were also conducted to check for damage of base support plates, fasteners, locknuts, brackets, and clamps associated with these installed pipe supports.

Within this area no violations or deviations were identified.

9. NRC Information Notice No. 87-28, "Air System Problems" (92701)

Information Notice No. 87-28 addresses problems that have occurred at numerous nuclear plants because air systems are designated non-safety-related. It points out that degradation in air quality can result in safety related equipment failures. This information notice indicates the root cause is traceable to design and/or maintenance deficiencies.

The inspector reviewed ANSI/ISA - S7.3-1975, "Quality Standard for Instrument Air", which is referenced by Watts Bar Nuclear Plant System Description N3-32-4002, Rev. 1, "Compressed Air Systems". ANSI/ISA - S7.3 establishes instrument air quality acceptance criteria and indicates that periodic checks should be made to assure high quality instrument air. It was determined no such checks were being performed at the plant.

The inspector reviewed the FSAR, Section 9.3.1, " Compressed Air Systems". The compressed air systems at Watts Bar are divided into three subsystems:

a. Service Air - used for maintenance, outage, etc.
b. Control Air - used in plant instrument and control devices.
c. Auxiliary Control Air - used as a backup to control air to supply safety related valves if control air is lost.

Auxiliary Control Air is the only safety-related portion of the compressed air system. The FSAR, in Section 9.2.1.4, "Test and Inspections", indicates periodic tests will be performed after plant startup to ensure proper operation of the Auxiliary Air System and Isolation Valves. Due to delays in starting the plant, this testing was not being performed. This issue was discussed with plant management who agreed that some type of periodic testing or maintenance should be performed. The applicability of ANSI/ISA S7.3 is not discussed in the FSAR. The presence of auxiliary control air alone could not ensure proper operation of safety-related valves. If control air quality was allowed to deteriorate, damage, such as plugging instrument orifices, could occur. The licensee agreed to consider changing procedures to include periodic tests or inspections verifying control air quality.

This is URI 390-391/88-01-03, "Control Air Quality", pending review of the licensee's disposition of this issue.

10. Cable Tray Installation (51063)

During a routine tour on January 4, 1988, of the Unit 1 reactor building annulus, the inspector selected a safety-related cable raceway (4A1916) containing orange-trained safety-related cable as the subject of further reviews to assess the adequacy of the as-installed condition. TVA engineering was requested to resolve the following comments.

- Cable tray support locations, where adjustable horizontal fittings (ZNK) are used to connect the trays, are controlled by TVA's Design Criteria WB-DC-20-21.1 and NEMA Standard VE1-1971, which require either a 12-inch maximum span from the connection to the support or analysis to show adequate support exists, or testing of the tray to show adequate support. Information for the analysis or testing was requested since the support locations were greater than 12 inches from the connection. Also, there is a question concerning how many bolts should be used in each hinge since this number varies between three and four.

- Adjustable riser (ZNB) connectors are used to align the cable trays with the curvature of the wall. ZNB connectors, like the ZNK connectors, require supports within 12 inches of each hinge unless analysis or testing is completed. Information was requested concerning this qualification and the following ZNB connector observations:

  - Some nuts have less than full thread engagement on bolts.

  - ZNB connectors are used for trays mounted on their side. This connector is normally used on horizontal, flat mounted, trays and may not be qualified for use on side mounted trays.

  - Connectors are mounted using different sizes of bolts.

- The number of bolt holes and bolts per connector varies between three and four.

- The drawing fails to show the location of all ZNB connectors. Drawing 45W869-2 currently shows only two of the several used on this cable run.

Engineering reviewed the concerns and, on January 19, 1988, was unable to provide documentation showing adequate qualification for the use of ZNB and ZNK connectors without a support located within 12 inches of each end of the connector. Cable tray support qualification B46 860325 003 is only for trays with supports within 12 inches of the connector center line. The condition observed in the field was that supports were not always located within 12 inches of the connector. This condition and the fact that the drawings do not detail the location of all ZNB fittings and bolting details for ZNB and ZNK fittings were subsequently identified by TVA on CAQR WBP 880040 (Unit 1) and WBP 880041 (Unit 2).

Documentation was provided showing qualification of the ZNB connectors in the vertical plane utilizing vendor supplied bolting. Vendor drawings were provided showing bolts that have less than full projection through the nut. Completed qualifications also show the hinge connector bolting to be acceptable when used in the vertical plane.

Additional inspections were performed on safety-related raceways in the Auxiliary Building. The following items were identified to TVA's engineering office:

- Cable trays 3B 2045, 4B 2027, and 5B 2027, located adjacent to the Unit 1 Component Cooling Pumps, utilize hinge connections with supports greater than 12 inches from the connector. The inspector questioned the qualification of these connections and asked whether there will be an inspection to locate any additional unqualified installations.

- Cable trays in the annulus that are mounted on their side utilize tie wraps to retain the cables. The inspector questioned whether the failure of these tie wraps would impact the cable tray or the cables.

- Documentation was provided showing qualification of the ZNB connectors in the vertical plane. This qualification used 45-degree mounted connectors. The applicablity to 90-degree mounted connectors or vertical (flat) raceway is uncertain.

- Qualification of non-safety-related cable tray connectors mounted above safety-related cable trays.

- 1L (position retention) seismic qualification of cable tray covers and retainers.

- One missing support for cable trays 3B 2045, 4B 2027, and 5B 2027, located adjacent to the Unit 1 Component Cooling Pumps.

On January 29, 1988, TVA advised that CAQRs WBP 880040 for Unit 1 and WBP 880041 for Unit 2 have been revised by Engineering. The revised CAQRs are intended to include the new conditions identified and to require a complete cable tray support review. Ebasco has been assigned the task package to develop corrective action for the CAQRs. Additional qualification information may be obtained from vendors.

Qualification for using tie wraps to secure cables in vertically mounted cable trays is uncertain. It appears that undue strain could be put on the cable tray or cable if the tie wrap failed. Engineering advised they will discuss this with Stone and Webster Engineering, since they had a similar condition at another nuclear facility.

Based on the above review, the inspector determined the raceways were not properly qualified. These items were subsequently identified on CAQRs dated January 19, 1988, for Units 1 and 2.

10 CFR 50, Appendix B, Criterion III, "Design Control", specifies that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. This is implemented at TVA by the QA Topical Report TVA-TR75-1A, Rev. 9, Section 17.1.3 and Table 17D-1 which commits to ANSI N45.2-1971, "Quality Assurance Program Requirements for Nuclear Power Plants."

ANSI N45.2, paragraph 4.3 states, "In those cases where the adequacy of a design is to be verified by tests, the testing shall be identified. Testing shall demonstrate adequacy of performance under the most adverse design conditions. Operating modes and environmental conditions in which the item must perform satisfactorily shall be considered in determining the most adverse conditions. If testing indicates that modifications to the item are necessary to obtain acceptable performance, the item shall be modified and retested as necessary to assure satisfactory performance".

TVA Design Criteria WB-DC-20-21.1 requires cable tray fittings and supports to be qualified either by (1) Conformance to NEMA Standard VE1-1971, Sect. 5.05, (2) Support requirements specified by the cable tray manufacturer, or (3) by analysis or testing of the fitting supported in another manner.

The following deficiencies have been identified on vertical cable trays attached to the steel containment vessel (installation details are shown on TVA drawings 48W970-1 thru -5 and 45W869- series drawings):

a. ZNK fittings for cable trays have not been qualified as required in the design criteria. DNE has not been able to locate any qualification documentation of this fitting.

b. ZNB fittings for cable trays have not been adequately qualified in the vertical position. This fitting has been qualified in the horizontal position for a single support within 12 inches.

c. In addition, DWG 45W869-Series does not clearly detail location of all ZNB fittings and bolting details for ZNB and ZNK fittings.

d. ZNB fittings, which have been qualified for a single support within 12 inches, are supported in excess of 12 inches.

e. Off set fittings for cable trays have not been qualified as required in the design criteria. DNE has not been able to locate any qualification documentation on this fitting.

f. The installed configuration of cable trays do not match design drawings.

Other similar deficient conditions may exist in the plant and the corrective action will need to address any programmatic problem.

Failure to comply with 10 CFR 50, Appendix B, Criterion III, "Design Control" as implemented by TVA's Topical Report Section 17.1.3 is identified as Violation 390, 391/88-01-02, "Cable Tray Installation".

11. Watts Bar Detailed Control Room Design Review (DCRDR) Status (37055B)

The DCRDR status was reviewed. Documents reviewed included a DCRDR Summary Report, dated September 1987, which was submitted to the NRC. The report discussed the program history, controlling documents, management and staffing, data management, methodology, and results. It defined the following:

- Human Engineering Concern (HEC) - An item designated by a DCRDR team member as a potential HED.

- Human Engineering Discrepancy (HED) - A characteristic of the existing control room that does not comply with the human engineering criteria.

HEC's were evaluated by the DCRDR staff. There were 1913 HECs identified from all sources. Of these, 1351 were assigned to HED's, either individually or in conjunction with other HECs. Of the remaining 562 (29%) HECs: 188 (10%) were already corrected; 154 (8%) were not considered valid; 22 (1%) were maintenance actions; and 198 (10%) were duplicates. Usually a group of HECs, rather than a single HEC, was assigned to an HED. The groups were established on the basis of relationships among the concerns. The grouping of concerns facilitated attention to cumulative and interactive effects in both the assessment of HED significance and the development of corrective actions. A total of 219 HEDs have been identified. Corrective actions have been established for 152 HEDs. The corrective actions include numerous hardware changes.

Also, included in the corrective actions are studies and surveys as appropriate. The remaining 60 HEDs have been evaluated as not requiring corrective action. Ongoing activities included:

- Stone and Webster had been awarded a contract to study control room labeling, lighting, communications, and computers.

- Ebasco has been selected to perform the engineering for necessary modifications. TVA's modifications group will install the modifications.

- The DCRDR activities will interface with the Containment Isolation Status Panels (CISP), Reactor Vessel Level Indication System (RVLIS), Regulatory Guide 1.97 - Post Accident Monitoring, and Q - List issues.

The inspector will continue to monitor the progress of the DCRDR program.

12. Welding On Vendor Supplied Components (55155B)

The inspector reviewed the licensee's disposition of a confirmed employee concern, identified as IN-85-372-001. The employee's concern indicated that welds on vendor supplied components are of poorer quality than TVA welds. The example cited was the manhole covers located on the 692 ft. elevation in the Unit 2 Reactor Building. The licensee determined the manhole covers (manways) in question were the Residual Heat Removal (RHR) Sump Valve Room manways. The licensee inspected the manways in question and issued Nonconforming Condition Reports (NCR) 6341 (Unit 2) and 6345 (Unit 1). The NCR's indicated the contractor welds for stiffner plates on hatch covers appear to not meet requirements of AWS D1.1. The NCR indicated the welds appear to be undersized in places and to have undercut and overlap. Engineering received the NCR for evaluation and disposition.

On October 18, 1985, engineering responded to the above referenced NCRs with the following written conclusion:

> "We have reviewed the subject NCR on substandard welds on the RHR Sump Valve Room hatch covers and concur with your recommended disposition to use-as-is. The reason for our concurrence is that the RHR sump is no longer an extension of the reactor building primary containment, therefore, these hatch covers are not safety related."

On March 1, 1988, a second letter from engineering, which discussed a third NCR (2357) that involved the RHR Sump Valve Room welds, was issued with the following conclusion:

> "The manway sleeves were repaired per NCR 2357. The doors have substandard welds, however, these have already been identified on NCRs 6341 and 6345. The disposition of these two NCRs was use-as-is... because the doors no longer serve a safety related function."

A third document, titled "Corrective Action Tracking Document" (CATD) NUMBER 80203-WBN-02, issued by the Program Manager for Employee Concern Task Group (ECTG), rejected the use-as-is disposition of NCR 6345 because the actual degree of weld nonconformance was not considered when the disposition was given. The CATD listed the following proposed corrective action:

> "No corrective action required. The original disposition was based on the description of the NCR which indicated the weld was basically per design with some possible undersize in places and some undercut and overlap. Based on this information, the following evaluation was made. The possible safety concerns would be seismic category 1(L) Position Retention, i.e. (1) the mounted cover could fall and damage safety-related equipment or (2) the possible impact of detached stiffeners on safety-related equipment. The cover is a one piece skin plate (70" x 7/16) which is bolted to a flange and it's position retention is not effected by this weld NCR. Based on the NCR description, substantial weld would remain to prevent detachment of the stiffeners from the cover. The validity of this basis was confirmed by a field inspection of the weld on 02/11/87 which indicated substantial weld is attaching the stiffeners plates to the skin plates."

An inspector review of drawing 44N355 and FSAR figure 3.8.4-16 found the manways specified to be Category 1 seismic equipment and therefore required to be safety related. On March 7, 1988, the inspector advised the licensee that dispositioning the above-referenced NCRs as non-safety-related was contrary to the drawing 44N355, Rev. 5 and FSAR Figure 3.8.4-16, which specifies the manways to be Seismic Category 1.

On March 17, 1988, another letter from engineering was issued to replace the March 1, 1988 memorandum on the subject. This letter specifies the following disposition of the referenced NCR:

> " In accordance with NCR WBNNEB 8207, the RHR Sump Valve Room no longer serves as secondary containment. However, it is required to remain intact during a seismic event to prevent possible damage to other equipment. The manway sleeves were repaired per NCR 2357. The doors do have substandard welds, however, these have been previously identified, evaluated and dispositioned to be acceptable-as-is [because the doors are "not safety related"] on NCRs 6341, 6345 and employee concern CATD 80203-WBN-02".

On April 1, 1988, another letter from engineering was issued that clarified the October 18, 1985, March 1, 1988 and March 17, 1988 letters by stating, "...these hatch covers are not primary safety related, but are secondary safety related."

The inspector reviewed the licensee's disposition of the above-referenced NCRs based on the March 17, 1988 and April 1, 1988 memoranda, issued after the inspector identified the deficiencies, and determined that proper

disposition of the undersize welds has now been made; the welds comply with FSAR requirements; and the welds meet the Category 1L Seismic requirements. Based on the revised disposition of the referenced NCRs, this item is considered acceptable.

13. NRC Bulletin 88-01, "Defects in Westinghouse Circuit Breakers" (92703)

This bulletin addresses problems in the Westinghouse series DS circuit breakers. It has been determined that the plant uses these circuit breakers. The bulletin requires plants that have not yet received an operating license to perform inspections of the circuit breakers prior to fuel load. The licensee has not scheduled the required inspections but plans to perform them before fuel load. This bulletin remains open.

14. Safety-Related Piping (Welding) Observation Of Work (55083B)

The inspector audited the following activities relative to nondestructive examinations and welding associated with the repair of ASME Section III pipe welds.

a. The inspector selected one weld (1-001A-D006-01 Main steam) where the root pass was in process and verified the following items were in compliance with the procedure and the ASME Boiler and Pressure Vessel Code (ASME Code),Section III, Class II requirements:

- Weld identification/location
- Specified weld procedure used in the root pass
- Physical appearance of weld
- Welder identification and qualification
- Evidence of QC verification of the root pass

b. The inspector selected one weld (1-003B-D003-06 Feedwater) where the welding was complete and verified that the following items were in compliance with the procedure and ASME Code Section III, Class II requirements:

- Weld identification/location
- Use of specified weld procedure
- Welder performing the welding was currently qualified for positions being welded
- Specified pre-heat, interpass temperature and post-weld heat treatment requirements were met (where applicable)
- Use of specified weld material

- Use of specified purge (if applicable)
- Starts, stops and undercuts were ground
- Procedures used to remove and repair defects and fit-up clips (if applicable)
- Physical appearance of the weld, e.g., under cutting, surface imperfections
- Periodic checks were made to assure that welding variables are within specified limits

c. The NRC inspector witnessed the calibration of the ultrasonic examination equipment and the volumetric examinations on ASME Code Class I shear lugs which are integrally attached with a full-penetration weld. The calibration and examination were performed in accordance with procedure WBEP - 0505L, Rev. 1, "Evaluation of Class I Shear Lugs". The calibration was also witnessed by the Authorized Nuclear Inspector (ANI).

In addition, the inspector reviewed the test results of the twenty Class I shear lugs installed on Class I piping. The examination results showed all welds were welded full penetration, as required.

All areas reviewed by the inspector were found acceptable.

15. Allegations (92701)

Allegation CSP 87-A-0050 (RII-87-A-0044), "Untimely Reporting of Construction Deficiency Reports (CDR) (10 CFR 50.55e Reports)".

This allegation has concerns applicable to all TVA sites. This report addresses only the concerns affecting Watts Bar.

CONCERN:

Engineering Assurance (EA) Audit 86-27, conducted in September 1986, identified a deficiency in disposition and control of "use-as-is" and "repair" CAQs. This deficiency's cumulative effect appeared to make the plant's margin of safety indeterminate. It also appeared that the Watts Bar Regulatory Compliance Organization was untimely in conducting reportability determinations.

DISCUSSION:

EA Audit 86-27 was issued September 26, 1986, and identified that the use of dispositions "use-as-is" or "repair", were not controlled by project or division procedures. This caused the quality of dispositions of "use-as-is" and "repair" to be indeterminate since justifications for the dispositions were vague or inadequate. On October 23, 1986, SCR WBN

WBP8601 was issued to correct the program deficiencies identified by the EA Audit. On January 12, 1987, the NRC was notified by phone that this issue was potentially reportable. The NRC received a written 10 CFR 50.55(e) report (CDR 390, 391/87-05) on February 11, 1987.

This issue was discussed with site licensing management who agreed the reportability determination on this issue was untimely. Licensing felt this was due to a lack of procedural guidance on the handling of CAQs at the time this issue occurred. Since then, procedures have been changed to include time requirements to be met in handling CAQs. The Nuclear Quality Assurance Manual (NQAM), Part 1, Section 2.16, Rev. 3, dated June 1, 1987, requires that a condition adverse to quality have a significance determination within 10 working days from the initiation of the CAQR. Program Management Procedure (PMP) 0600.03, Rev. 0, "Evaluation and Reporting of Construction and Design Deficiencies, 10 CFR 50.55e", requires a reportability determination to be made within 29 calendar days after classification of the CAQR as significant.

The inspector reviewed licensing's 10 CFR 50.55e files. Within the sample of 1987 issues reviewed, all issues determined reportable since PMP 0600.03 was issued, were determined within 29 calendar days after the CAQR was classified as significant. However, some issues determined to be non-reportable were not documented as non-reportable until two to five days after the 29 days expired. No violation is being issued for this procedural non-compliance only because there were no cases identified where reportable issues were determined to be reportable after the 29 days. This was discussed with site licensing who agreed better controls were needed to ensure the 29 day requirement was met in all cases.

The inspector reviewed several CAQRs and found several problems:

a. Significant CAQR WBP 871126 which identified that no procedure exists which addresses how TVA assumes design control of vendor drawings. This CAQR had been invalidated since the issue of the CAQR was considered to be only administrative in nature. This justification is of questionable validity because drawings are required to be controlled. Engineering personnel agreed that the wording used to invalidate this CAQR was inadequate and a revised justification would be considered. This is Unresolved Item 390, 391/88-01-04, "Justification to Invalidate a CAQR", pending review of the results of this reconsideration.

b. During August 1987, the NRC held an Independent Design Inspection (IDI) at TVA's Sequoyah Site. One of the findings of the IDI was that skid mounted valves are not shown on TVA flow diagrams. This resulted in safety-related/Technical Specification requirements being missed. On August 18, 1987, CAQR SQP 871347 was initiated and determined to be significant. On August 23, 1987, Sequoyah requested Watts Bar to perform a generic review. This generic review, which confirmed the issue was applicable to Watts Bar, was not completed until March 7, 1988. Watts Bar issued CAQRs WBP 880100 and WBP

880160. The NQAM, Part I, Section 2.16, Paragraph 10.5, requires potentially affected organizations to complete the generic review within 70 calendar days from the origination of the CAQR. Further inspection revealed that a Watts Bar generic review was also late on CAQR SQP 871003. Additionally, the Watts Bar generic reviews for CAQRs SQP 871066, SQP 871743, SQP 871304, and BFP 871056 were overdue.

10 CFR Part 50, Appendix B, Criterion V, as implemented by TVA's QA Topical Report, Rev. 9, Sections 17.1.5 and 17.2.5, and NQAM Part 1, Section 2.5, both titled, "Instructions, Procedures, and Drawings", requires activities affecting quality be accomplished in accordance with instructions, procedures, or drawings.

The failures to meet procedural requirements of the NQAM, Part 1, Section 2.16, is Violation 390, 391/88-01-01, "Failure to Follow Procedures".

c. CAQRs WBP 880100 and WBP 880160 were classified as "not significant". These two CAQRs were issued as a result of a generic review of Sequoyah's CAQR SQP 871347 (discussed above). CAQR SQP 871347 was classified as significant and was determined to be reportable. Since CAQRs WBP 880100 and WBP 880160 were not classified as significant, they did not receive the reviews for reportability. It appears that WBP 880100 and WBP 880160 should have been reportable.

d. CAQR WBP 870701 had been classified significant but not reportable. Licensing files contained memoranda from the various site organizations supporting the non-reportable determination. This inspection disclosed EA's CAQR evaluation (B26'870925012) which had been sent to the Site Quality Manager. This evaluation provides substantial evidence that this CAQR should have been reported. The Site Quality Manager's memorandum (T45 '871026947) to licensing did not recommend making the CAQR reportable and did not reference the EA memorandum. The justification used by QA to discount EA's evaluation had been requested.

Examples c. and d. together form URI 390, 391/88-01-05, "Reportability Determination of CAQRs", pending the licensee's explanation of the issues.

CONCLUSION:

The allegation that Watts Bar reported CDR 390, 391/87-05 in an untimely manner is substantiated. Under the new procedural controls, Licensing appears to be meeting required times for reportable items once they receive information from responsible organizations. This inspection has identified problems with the accuracy and timeliness of information submitted to Licensing by responsible organizations. These problems will be addressed by the violation and unresolved items discussed above.

16. List of Abbreviations Unit 1 and 2

ALARA	As Low As Reasonably Achievable
ANI	Authorized Nuclear Inspector
ASME	American Society of Mechanical Engineers
AWS	American Welding Society
CAQ	Condition Adverse to Quality Report
CATD	Corrective Action Tracking Document
CDR	Construction Deficiency Report
CEP	Construction Engineering Procedure
CI	Concerned Individual
CISP	Containment Isolation Status Panels
DCRDR	Detailed Control Room Design Review
DNC	Department of Nuclear Construction
DNE	Division of Nuclear Engineering
DNQA	Department of Nuclear Quality Assurance
DWG	Drawing
EA	Engineering Assurance
ECTG	Employee Concern Task Group
EQ	Environmental Qualification
HEC	Human Engineering Concern
HED	Human Engineering Discrepancy
IDI	Independent Design Inspection
IEEE	Institute of Electrical & Electronic Engineering
ISA	Instrument Society of America
LER	Licensing Event Report
MSU	Material Services Unit
NCIG	Nuclear Construction Issues Group
NCR	NonConformance Report
NEMA	National Electrical Manufacturers Association
NEP	Nuclear Engineering Procedure
NFPA	National Fire Protection Association
NQAM	Nuclear Quality Assurance Manual
OSC	Operations Support Center
OSP	Office of Special Projects
PGCE	Potential Generic Condition Evaluation
PICSP	Prime Instrument Calibration Status Program
PIR	Problem Identification Report
PM	Preventive Maintenance
PMP	Program Management Procedure
QCI	Quality Control Instruction
RVLIS	Reactor Vessel Level Indication System
SER	Safety Evaluation Report
SIS	Safety Injection System
SQA-SIL	Sequoyah Quality Assurance - Staff Instruction Letter
SQN	Sequoyah Nuclear Plant
TI	Temporary Instruction
TROI	Tracking and Reporting of Open Items Report
URI	Unresolved Item
WBN	Watts Bar Nuclear Plant
ZNB	Adjustable Riser Fitting
ZNK	Adjustable Horizontal Fittings