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

Report Nos.: 50-390/90-20 and 50-391/90-20

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: July 21, 1990 - August 17, 1990

Inspectors: K. P. Barr 9/25/90
for G. A. Walton, Senior Resident Inspector Construction — Date Signed

K. P. Barr 9/25/90
for S. P. Burris, Senior Resident Inspector Operations — Date Signed

A. R. Long 9/25/90
A. R. Long, Project Engineer — Date Signed

Approved by: K. P. Barr 9/25/90
K. P. Barr, Chief
Projects Section 3
TVA Projects — Date Signed

SUMMARY

Scope:

The inspection consisted of reviews of civil and electrical issues, employee concerns, fire prevention and protection, preoperational testing, and reviews of previously identified inspection items.

Results:

Five unresolved items (URI) were identified concerning the following issues: A QC inspector was signing and dating work documents on dates which were different than the dates the inspection occurred (paragraph 3.a); the licensee isolated a portion of the High Pressure Fire Protection System required to be operable under the Special Nuclear Material license (paragraph 3.b); the licensee found Microbiological Induced Corrosion products in a system previously determined not to be subjected to MIC (paragraph 3.c); torquing methods and values for anchor bolts installation were not being controlled (paragraph 5); and adequacy of using UT in only three locations verses 100 percent examination to investigate reported adverse conditions (paragraph 6).

Based on the above identified findings, concerns exist with the licensee's implementation and controls of work activities at the Watts Bar site.

Within the areas inspected, one violation was identified as discussed in paragraph 5.

*Unresolved Items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

P. Pace, Compliance Licensing Support Supervisor
G. Brantley, Employee Concerns Site Representative
*S. Crowe, Site Quality Manager
D. Douthit, Program Manager
E. Fuller, Chairman, Program Team
*A. Gentry, Assistant Site Representative, Employee Concerns
L. Jackson, Operations Manager
*M. Jones, Startup and Test Manager
*F. Koontz, Manager Operations Engineering
*L. Nolan, Construction Manager
C. Nelson, Acting Maintenance Support Superintendent
*J. Scalice, Plant Manager
*R. Stevens, Site Licensing Manager
*S. Tanner, Quality Control Manager
*P. Wilson, Manager, Special Projects
R. Wilson, Vice President New Projects

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

*Attended exit interview

Acronyms used throughout this report are listed in the last paragraph.

2. Review Of Corrective Actions Program For Civil Issues (51063)

On August 2, 1990, a meeting was held in Rockville, Maryland, between TVA and NRC to discuss the status of all corrective action programs (CAPs) and other associated issues related to civil issues at Watts Bar. The licensee presented information relative to the following civil issues. During this inspection further reviews were done by the inspector regarding the status of rework activities and NRC review status.

- Seismic Analysis (SEISMIC CAP)
- Hanger Analysis and Update Program (HAAUP CAP)
- Equipment Seismic Qualification (EQ CAP)
- Conduit and Supports (C&S CAP)
- Heating Ventilating Air Conditioning (HVAC CAP)
- Cable Tray And Supports (CTS CAP)
- Design Base Verification Program (DBVP)
- Special Programs (SP)
    - Soil Liquefation
    - Concrete Quality

The licensee recently completed a re-evaluation of the civil issues through independent contractor reviews. The contractors were R. L. Cloud Associates and Sygma Corporation. The re-evaluation results are discussed below:

a. Seismic Analysis

The CAP describes the following issues regarding seismic analysis of soils and structures; integration time step, soil-structure interaction, torsional effects on building analysis, and auxiliary diesel generator building.

The licensee has completed the reanalysis of nine structures identified in the CAP. This includes the completion of the floor and wall flexibility study, comparison of design basis and evaluation basis seismic responses, and documentation review of soil shear module variation basis. The items identified by the licensee to be completed are evaluation of seismic response differences and program documentation.

The NRC has completed the review and approval of the CAP methodology. Further inspections are planned to review the licensee program and implementation of this CAP.

b. HAAUP

The licensee's program in this CAP involves both large and small bore piping analysis. For large bore piping, the program includes a 100 percent walkdown and rigorous reanalysis of 300 stress problems and 8,700 pipe supports. For small bore piping the program includes verifying the acceptance of 520 stress calculations and 6,200 pipe supports. The licensee has indicated that the evaluation on small bore piping and supports will be qualified by bounding calculations.

The large bore piping walkdown and reanalysis is approximately 90 percent complete. The small bore piping program of bounding 4,000 variances of nonstandard support design is scheduled to be done. The design input documents are complete and the licensees stress qualification is in progress.

To date the licensee has issued engineering change notices for approximately 4,000 pipe support modifications of large bore pipe supports. The licensee anticipates an additional 10 percent rework to be added upon completing the program. Included in the modifications is a snubber reduction program that will eliminate approximately 350 snubbers that are presently installed. The field rework of large bore pipe support modifications is approximately 35 percent complete. The licensee is completing the modifications on the systems needed for turnover. The licensee has not completed enough of the analysis on small bore piping to determine if any rework is needed.

The HAAUP also includes the verification of the qualification of approximately 109 stress calculation and 4,400 supports for the radiation monitoring and sample line problems and the qualification of approximately 396 stress calculations and 500 supports for sense and control air lines. Additionally, the licensee plans to verify the adequacy of 4,400 typical support designs for field routed tubing to assure compliance with the updated criteria.

The NRC plans additional inspections in this area to evaluate the program adequacy and ongoing modifications. To date the NRC has performed various inspections of the field modifications on large bore pipe supports.

The NRC approved the CAP methodology based on a rigorous analysis of all safety related, and important to safety related pipe supports. The use of bounding calculations for small bore pipe support evaluations is under review.

c. Equipment Seismic Qualification

This CAP is to address approximately 2,000 models of category 1 and 1L equipment to resolve the following: problems identified with interface controls; discrepancies between design documents and installed condition; discrepancies between as-installed condition and inspection documentation; equipment mounting conditions identified from the Sargent and Lundy vertical slice review; and NRC open issues identified by site inspections.

The licensee's corrective actions are to perform walkthroughs to identify installation discrepancies and resolve the discrepancies by either engineering evaluations or field modifications. The licensee plans to review and evaluate flexible mounting conditions, generate new anchorage loads, and evaluate supporting structures and effects on equipment nozzle loading. The licensee plans to make field modifications as needed. To date the licensee has completed the support modifications for the instrument B-19 support brackets found on the Foxboro transmitters. The licensee anticipates completing the modifications in September 1990, for the Asco solenoid valves. The major rework effort for other deficiencies is scheduled to start in September 1990.

The licensee plans to perform bounding calculations for Category 1 equipment in locations where updated spectra exceeds original design spectra.

The NRC has reviewed and approved the CAP in regards to the methodology planned to resolve the issues. The bounding calculations will be further reviewed by the NRC in further inspections planned.

d. Conduits And Supports

This CAP identified that conduit and support discrepancies existed in the design bases in that the original design did not envelop all design requirements. The installed configurations do not comply with the design drawings and discrepancies exist between the installed hardware configuration and inspection documentation.

The scope of this activity involves approximately 8,000 conduit runs and 30,000 supports. To date, the licensee has completed the walkthrough inspection and documentation of 20 percent (approximately 5,100 supports) of the installed supports. For those supports which did not comply with the procedure (cookbook requirements) the licensee established a critical case (cases outside the set of established boundary) evaluation. Based on the 20 percent evaluation (5,134 supports) the licensee has established that modifications are required on 5 conduits and 6 supports. Additionally, the licensee has determined that 8 attributes will envelope the critical attributes (evaluation required to determine acceptance.) The 8 attributes include; L-cantilevers, multiple condulets (Christmas trees), conduit overspans on conduits below 2 inches, T-spans on conduits less than 2 inches, nonstandard supports, supports with missing members, typical supports fabricated to detail 55 drawing, and typical supports fabricated to detail 66 drawing. The licensee plans to walkthrough the remaining 80 percent of the conduit runs and perform inspections on the 8 critical attributes discussed above. To expedite the work in conjunction with the CRDR work in the control room, the licensee reworked all conduits and supports in the control room that were outside the procedure (cookbook) limits. Additionally, on some items such as the one hole strap, the licensee required construction to walkdown all conduit runs with the one hole strap and rework the supports to meet the qualified requirements. This work is ongoing.

The NRC has reviewed the CAP that describes the critical case analysis method and approved the methodology. However, the approval was based on the methods used to perform the first 20 percent of the inspections. The NRC has not evaluated and approved the inspection methods proposed for the remaining 80 percent of the work.

Further reviews are required to determine the acceptance of the licensee proposal for the additional 80 percent of the inspections.

e. Cable Trays And Supports

The licensee's program in this CAP is to address all Category 1 and 1L cable tray runs and 4,700 associated supports.

The issues involve the lack of full documentation of supports, undocumented design qualification for cable tray hardware, and installed hardware configurations do not comply with design output documents. The licensee is proposing to sample inspect 58 of 3,000 supports based on NCIG -01 to determine the adequacy of the inspection documentation. Additionally, the licensee is proposing a critical case evaluation of 1,700 supports (20 percent) to develop critical attributes for use in final walkthroughs (80 percent.)

For determining the adequacy of supports that have overloaded trays, the licensee is proposing to identify overloaded tray sections and group the overload conditions by configuration, amount of overload, location of overload, and existing interaction coefficients. The licensee then plans to perform worst case evaluations of the conditions. The licensee also plans to resolve the concern of support adequacy in vertical run cable tray runs. The planned resolution is similar to overloaded trays in that the licensee plans to identify vertical tray runs and then group them by height, location configuration, and overload. The licensee then plans to evaluate the worst case conditions for acceptance.

For resolving the issue of additional attachments, such as conduit supports, the licensee plans to identify a representative population of attachments found by walkthroughs and reviewing DCNs. Then they will group conditions by types of support configurations and interaction coefficients. Upon completion of this grouping, the licensee plans to evaluate the worst case conditions for acceptance then perform final walkthroughs of the cable tray supports and make modifications as necessary.

For cable trays the licensee is establishing acceptance criteria based on testing and bounding calculations, generating walkthrough procedures including critical case selection criteria, 100 percent walkthroughs to select potential critical cases based on load and configuration, then group and evaluate critical cases. Upon completing the critical case evaluation the necessary modifications will be made. Additionally, the licensee plans a rework of all cable tray covers to assure their stability during a seismic event.

The NRC reviewed and approved the methodology proposed for resolving the issues. However, the NRC has not performed enough inspections to determine the adequacy of the licensee's proposed corrective action programs in this area. Further reviews and inspections are planned.

f. HVAC Ducts And Supports

This CAP is to address acceptability of approximately 3,100 duct spans and 2,500 duct supports, and to resolve problems of discrepancies in design bases (i.e., design output documents not based on enveloping the correct design parameters, installed hardware configuration is not per design documents, and installed hardware configuration is not consistent with the inspection documentation.)

The licensee plans to perform walkthroughs and drawing reviews to determine potential critical attributes, then perform walkdowns (percent not specified) of potential critical cases. After completing this activity the licensee plans to perform the critical case evaluations and identify critical case attributes. The remaining evaluations (100 percent walkthroughs) will be based on the attributes identified in the initial assessment. Modifications will be performed where necessary based on the critical case evaluations of the critical attributes selected.

Based on the licensee's reviews the decision was made to issue DCNs to fix identified problems such as missing tie rods, cut or missing stiffeners, and excessive rivet spacing. Additionally, the licensee plans to perform special engineering and field assessments to evaluate flexible support interface and add flexible connectors where necessary, evaluate ducts subjected to high negative pressure, locate all supports constructed to typical drawing 47A055-15 (dead weight), and upgrade the supports to resist lateral loads. Lastly, the licensee has inspected and repaired all nonconforming duct and supports in the control room to expedite the CRDR modifications.

The NRC has performed evaluation of the CAP and approved the methodology of the program. Further reviews are necessary to evaluate the critical case evaluations planned by the licensee.

g. DBVP CAP

Although not specifically addressed in the DBVP CAP, the licensee is performing analysis due to the following issues with steel platforms; undocumented attachments to the platforms, verification of the adequacy of connecting members, and torsional effects on members. This review is required to determine the adequacy of approximately 100 platforms. The licensee plans to update the criteria for torsional and connection checks, evaluate the impact of interface with large bore piping loads, review the platform population, and select worst case set for evaluation. They will obtain walkdown data and actual loading for 20 selected sets of platforms, and will walkdown the worst case platforms and obtain attachment loads. The licensee then plans to analyze data on the worst case platforms, evaluate critical members and connections and implement modifications on the hardware where needed. Based on data collected from the initial evaluation of the 20 platforms, the licensee will establish critical attributes (derived from needed modifications) and walkthrough the remaining population looking at the critical attributes. The licensee will evaluate the data and make modifications as needed.

This issue was not submitted to the NRC as a CAP. To date, the NRC has not evaluated the adequacy of the licensee's program to address these issues.

h. Miscellaneous Issues With Steel Structures

The licensee identified other issues with steel structures that are assigned to the DBVP activities.

Thermal loading was not considered for all miscellaneous steel structures.

Pipe whip restraints are not always adequate in that embedded plate evaluations did not consider all design attributes. Calculations are not documented to support engineering judgements used for design and field changes and computer output for some design calculations is not retrievable. The work scope includes 115 pipe whip restraints and 136 embedded plates.

Steel containment vessel penetrations and attachment pad plates need to be verified due to revised loads on piping, conduit, and HVAC. Also, the penetration and pad plates interactions were not always evaluated and the steel containment vessel displacements require verification for thermal loads due to main steam line break considerations. The work scope includes 131 piping, electrical, and HVAC penetrations. Three-hundred and thirty pad plates require assessment.

Structural steel connections were flame cut to enlarge holes and possibly enlarged the holes beyond what is allowed. Beam copes were saw cut and resulted in square corners and notches (stress risers.) The NRC CAT inspection questioned the torque adequacy of high strength bolting.

The NRC has not performed any evaluations of the licensees planned evaluations in this area. Inspections are planned to evaluate the licensee implementation of these programs.

i. Concrete

The licensee plans to perform an overview of concrete installations to evaluate and integrate updated seismic requirements for Category 1 buildings, to evaluate large bore piping revised inputs and equipment overturning loads will be included. The NRC will review these issues as part of the DBVP inspections planned in the future.

j. Masonry Walls

The issues that will be addressed for masonry walls include the updated seismic loads, undocumented attachments connected to the walls, and analysis of nonqualified anchoring devices. NRC Bulletin 80-11 will be reopened to follow the resolution of these issues. Inspection and closure of the bulletin will be performed in future inspections.

k. Standard Design Embedded Plates

The licensee's program will address the reconciliation of large bore pipe support interface loads and verification of the structural adequacy of embedded plates. This issue will be addressed as part of the DBVP reviews planned in future inspections.

l. Equipment Anchorage and Supporting Structures

The licensee plans to evaluate the equipment anchorage and supporting structure due to revised equipment loads. The effects of TVA designed equipment supports on equipment seismic qualification and interface loads were not consistently considered. Also, concrete slabs were not initially evaluated for the effects of equipment overturning moments. This issue will be reviewed as part of the DBVP reviews planned in future inspections.

m. Geotechnical Calculations

The licensee plans to regenerate or update nonretrievable calculations on slope stability, settlement analysis, buried conduit banks, and correlation of soil properties and test data. This issue will be evaluated as part of the DBVP reviews planned in future inspections.

n. Integrated Interactions Issues

The licensee plans to evaluate the effects of seismic and thermal interactions, component protection from failure of nonsafety items, and flexibility of safety systems crossing building boundaries for relative motions. This issue will be evaluated as part of the DBVP reviews planned in future inspections.

The licensee has provided to the NRC/NRR for review marked up copies of the proposed changes to the FSAR that affect the civil issues discussed above.

3. Plant Tours

The inspectors toured various areas of the plant to ascertain whether the licensee was administratively controlling activities for the protection of equipment and personnel safety. The inspectors looked at various fire protection equipment, temporary erected structures, in-progress work activities, and general plant housekeeping and cleanliness. In general the plant is being maintained in a safe condition. For those observed areas, the inspectors did not find any areas which contained excessive or uncontrolled combustible material, or excessive accumulation of dirt and debris or unidentified out of service fire protection.

The following items will be discussed under this paragraph:

a. On August 26, 1990, during a tour of the electrical switchgear rooms, the inspectors reviewed Workplan KPO 3226A-1 which replaced various Crydom relays and filter modules per DCA P03226. During the document review the NRC inspectors noted that the QC inspector had apparently signed and dated the required QC signature and date blocks within the workplan with a date other than the actual date the record was signed. The inspectors obtained and reviewed currently approved QC procedures governing the work practices and requirements for QC inspectors. However, after review the inspectors noted that these procedures were not specific enough in nature to identify signatory and dating requirements.

   The inspectors discussed this issue with the Site QA Manager and requested that the licensee provide any justification for a QC inspector signing a required hold point and using a different date. Based on the status at the end of this inspection period, the inspectors notified the licensee that this would be identified as an Unresolved Item, 50-390/90-20-04, "QC Signature and Dating Requirements."

b. While attempting to reinstall (weld) a portion of the High Pressure Fire Protection (HPFP) piping removed earlier due to damage, the licensee inadvertently isolated a Fire Protection Hose Station (FPHS) on the refueling floor elevation (757'). This FPHS was required to be operable per the licensee Special Nuclear Material Licensee (SNM-1861) for stored new fuel in the Fuel Handling Building storage area. Initial discussions with the licensee identified the following conditions existed at the time of the event:

   - The original isolation lineup was not effective in isolating water to the weld area (one of the valve boundaries 2-26-653 was apparently leaking past its seat.)

   - To stop this leakage operations personnel isolated the next upstream valve 0-26-649 which in effect isolated the weld area in question.

   The weld area was effectively isolated, however, later review of this particular lineup found that when valve 0-26-649 was closed, it also isolated FPHS valves 2-26-671 and 2-26-672. The licensee took immediate corrective action to restore the FPHS and started a review of the circumstances which allowed this condition to be established.

   The inspectors informed the licensee that this would be identified as an unresolved item pending review of the conditions stipulated within the Special Nuclear Material Licensee (SNM-1861) and review of all pertinent documentation (hold orders, workplans, MRs, etc, . .) URI

50-390/90-20-05, "Isolation of License Required Fire Protection Equipment."

c. During repair of the HPFP system piping damaged during previous work activities, it was found that the HPFP piping was subjected to the effects of Microbiologically Induced Corrosion. The licensee is currently evaluating the degree and extent of the damage caused by MIC. The inspectors informed the licensee that this item would be identified as an URI 50-390/90-20-06, "High Pressure Fire Protection - Microbiologically Induced Corrosion."

Within this area no violations or deviations were identified.

4. Electrical Cables (51064B)

The inspector witnessed wet high pot testing of five V-4 electrical cables in a 15 foot segment of Train B conduit between the auxiliary building and the intake station. The testing was conducted on August 6, 1990, in accordance with workplan M5817-2 and TI-43.

The tested segment of conduit 2PLC1286B had been classified by the licensee as low risk with respect to potential cable damage. There had been three cable pull-bys in the conduit segment. An additional portion of the same 3,000 foot circuit had been previously classified as high risk and had been replaced. The cables which were tested were three-conductor, No. 8 cable.

After flooding the conduit from the low end and verifying it was completely filled by observing water from the high end, each conductor was tested by applying a voltage of approximately 7.2 kv. The test procedure specified that this voltage was to be reached within one minute. Leakage current was then measured at one minute intervals for five minutes. The test acceptance criterion was that the polarization index, which is the ratio of the leakage current after one minute to the leakage current after five minutes, must be greater than 1.0 (i.e., the leakage current decreases during a successful test.)

All conductors in cables 2PL2895B/WLC, 2PL3161B/WFA4, and 2V3996B/WMB passed the test acceptance criterion. Although all three conductors in cable WPL3905B/WDK passed the polarization index acceptance criterion, none of the conductors reached the test voltage within the one minute period specified in the test procedure. This information was noted in the data package for evaluation.

The black and the red conductors in cable 2V3988B/WMB failed to meet the polarization index acceptance criterion. Leakage current for the black conductor increased from 2.4 microamps at one minute to 3.05 microamps at five minutes, for an index of .786. Leakage for the red conductor was 7.9 microamps at one minute, decreased slightly, then increased to 8.1 at five minutes to yield a polarization index of .975. When these two conductors did not pass the test acceptance criterion, the licensee completed testing of the remaining cables in the conduit. The two conductors which failed the testing were then retested, as allowed by the approved test procedure. The black conductor passed the retest, with a polarization index of 1.05. However, the red conductor failed the retest due to a measured polarization index of .934.

Although the red conductor of cable 2V3988B/WMB failed the test acceptance criteria, the licensee concluded that the low magnitude of the leakage current 2V3988B/WMB provided sufficient justification that the insulation was sound. The successful retest of the black conductor was accepted as a valid indication of its acceptability. These conclusions were discussed with NRC Headquarters staff. The cable was determined to be acceptable based on these discussions.

5. Concrete Anchor Bolt Inspection (46053)

Potential deficiencies regarding the installation practices of conduit supports were brought to the attention of the resident inspector during this inspection period. The inspector selected some recently installed conduit supports for inspection to determine the validity of the concerns. The supports selected are conduit supports fastened to the wall with wedge type concrete expansion anchors.

The first concern expressed to the inspector was that during initial installation and setting of the anchor bolts, it sometimes fails to protrude through the baseplate, washer, and nut by the required amount (one thread projection through the bolt.) In these situations, it was alleged that the foreman instructed the craft to ignore the requirement of using a calibrated torque wrench and to tighten the bolt without a torque wrench until the thread was exposed the required amount. This was done without QC's knowledge. When the bolts achieved the correct protrusion, the craft were instructed to back the nut off and then notify QC. With QC witnessing the test, the nut was retightened using a calibrated torque wrench until the minimum torque (on one inch bolts the minimum torque in 250 ft. lbs.) was obtained. The concern was that a wrench was used with a three foot extension and excessive torque was applied to the bolts and damage could occur to the bolt or the concrete. It was stated to the inspector that a QC inspector had written a CAQR on the subject, but the management had invalidated the CAQR.

The second concern expressed by the individual was that on at least two other one inch diameter concrete expansion anchor bolts, the craft had attempted to torque the bolts to the 250 ft. lbs. minimum torque requirements twice and each time the bolts failed to set. The minimum torque was never obtained by that crew. He visited the area later and found the bolts were apparently set and also accepted by QC inspections. He didn't understand how the torque was achieved and was concerned that the bolts may not be torqued to the minimum torque requirements.

The third concern expressed by the individual was that some craft persons had taken the above concerns to the licensee's Employee Concern Program (ECP) over a month before and nothing had been conveyed back to them about what the ECP had done, if anything.

At the inspectors request and with the inspector present, craft personnel rechecked the minimum anchor bolt torque (250 ft. lbs.) on all 4 bolts on conduit supports number 2-CSP-292-N1631 and N1644. The bolts that supposedly never would torque to the minimum torque requirements. The bolts checked were found to be torqued to the minimum requirements of 250 ft. lbs. Additionally, the inspector had the nuts and washers removed and examined the bolt to assure the bolt was properly installed and the nut was not bottomed out on the threaded portion of the bolt. The installation of these bolts was found acceptable.

The inspector met with field engineering personnel and discussed the apparent field practice of not requiring the use of a torque wrench during the setting of the anchor bolts. The engineer advised that no upper limits exist when setting the bolts. The inspector reviewed procedure WBN-CPI-8.1.8-G-100 "Expansion, Grouted, And Undercut Anchors", paragraph H which states "Set the bolt by applying a smooth and uniform torque to the nut with a calibrated wrench until the minimum torque specified in table 5-7 has been reached. NOTE 1: Torque shall be read while nut is in a tightening motion. Anchors may be torqued to values greater than minimum, however, significantly higher installation torque (more than 20 percent) may result in anchor breakage during installation or in anchor projections exceeding requirements of the table (250 ft. lb. for one inch diameter bolts.) For example, if end of wedge bolt does not extend completely through nut when minimum torque is applied, additional tightening may be performed to increase projection of wedge bolt."

Based on the inspectors review of the above procedure, it appears that a torque wrench is required while setting the bolt and precludes the use of a wrench without torque values. Also, the procedure appears to establish upper torque limits of 120 percent when setting the bolts. The inspector also found an invalidated CAQR (WBP900213) that indicated the following, "Wedge bolts are being over torqued above the minimum at an undetermined and unmeasured value to achieve more projection of the wedge bolt to meet the one thread through the nut requirement of paragraph 5.3.3H of CPI-8.1.8-G100." This statement on the CAQR was lined out as not being applicable. The inspector interviewed the QC inspector who had written the CAQR and found the statement was lined out because he had not actually seen the over torquing occur; therefore, his management required that part of the CAQR be deleted. He advised that after that part of the CAQR was deleted, it was no longer a CAQR condition and was invalidated.

The inspector found that on August 13 another QC inspector observed craft personnel installing wedge bolt anchors, utilizing a breaker bar and socket with a cheater pipe for leverage to pull out the anchor bolts enough for the nut to achieve the required thread engagement. CAQR

WBP900377 was issued on August 10, 1990, by the QC inspector and identified the condition.

This item is identified as URI 50-390/90-20-02, "Anchor Bolt Installation Practices". The licensee should address the following questions to resolve these issues.

- What are the allowable upper torque limits, if any on wedge type anchor bolts?

- Can wedge type anchor bolts be set without using a calibrated torque wrench ?

- Is the anchor bolt installation procedure adequate for the field to properly install the anchor bolts?

Regarding the issue that the ECP apparently did not provide timely feedback information to the craft persons that entered the concern with the ECP, the inspector found that the employee concerns people knew that QC had written a CAQR on the subject. Since QC was addressing the issue they were of the opinion the CAQR would solve the problem and were waiting on the CAQR disposition before going back to the concerned individual. They apparently did not know the CAQR was invalidated. Between the time the concern was entered and the inspector inquired about the concern a month had elapsed and no feedback had been provided to the concerned individual. Based on discussions about this issue, the licensee has now changed the ECP program to include feedback to all concerned individuals on a two week basis. Also the ECP will provide each concerned individual with a card showing the ECP persons name and also the concerned individuals case number. Based on this change, the inspector is satisfied with the resolution of this issue.

The inspector performed visual inspections in the same area (auxiliary building, elevation 737) of other supports on this line and found that on conduit supports 2-CSP-292-N1615 and N1616, the anchor bolts failed to meet the requirements for complete thread engagement as required by CPI 8.1.8H-400 and CPI 8.1.8-C-501-A. These procedures require that all bolted connections, when properly tightened, shall exhibit visible evidence of complete threading through the nut by a minimum of one thread exposed. The records indicate the bolts were inspected by a QC inspector and accepted on 02-16-90 without noting or rejecting the apparent deficiency. Failure to identify and cause corrective actions on these deficient conditions is identified as a violation of 10 CFR 50, Appendix B, Criterion V, VIO 50-390/90-20-01, "Failure to Follow QC Procedures."

6. Welding (55150)

The NRC Independent Inspection Laboratory radiographed eight stainless steel welds to examine the inside for the presence of MIC. The welds selected were on the ERCW system, fabricated to ASME Class 3, and were eight-inch diameter stainless steel welds located in the reactor building annulus. The welds were not radiographed (code does not require radiography) when they were fabricated. The radiographs did not disclose any MIC in the stainless steel lines radiographed. Four of the welds did disclose defects that are being addressed by the Independent Inspection report. A follow-up review by the licensee disclosed that some of these welds were the subject of a concern identified by the QTC reviews. The concern number WI-85-050-001 identified to QTC on 08-24-85 the following: "inside the annulus, unit 1 ERCW system, eight inch stainless steel. This occurred in the latter part of 1983, or early in 1984. Located right of entry door four, eight-inch lines. Deterioration of metal, lack of metal, lack of penetration, and sugar welds--bad welds could be identified by x-ray."

To address this concern, the licensee required EG&G to evaluate the issue as part of the Weld Evaluation Project (WEP). EG&G performed the evaluation in WEP Group number 013, (report dated November 16, 1987) and used ultrasonic examination techniques at three locations around the subject piping to attempt to find the reported conditions. No defects were found using the sample test. The EG&G report does not address the statement that x-ray would find the condition. Two of the welds involved, which were radiographed by the NDE Team, are 1-067J-T608-03 and T606-04 and display the condition reported by the concerned individual to QTC and TVA. The inspector questioned the adequacy of using UT in only three locations to investigate the reported condition in lieu of a 100 percent UT or radiography. This item is identified as URI 50-390/90-20-03, "Selective UT Examination Methodology," pending the licensees further review of the employee concern issue. The technical concerns regarding the UT calibrations and the audit of record adequacy of the UT tests is discussed in report 50-390,391/90-18.

7. Prestart Test Activities (70301)

During this inspection the inspectors continued to observe WBNP Prestart Test activities. These activities included test witnessing, program status determination, and scheduling review. Each area of consideration is discussed below:

- The inspectors obtained and reviewed the licensee's test procedure for the activities associated with component cooling system valve relay testing. The Prestart Test Group performed testing in accordance with Test Instruction TI-25, Revision 7, "Slave Relay Test Panel Operating Instruction," and SI-K618-1, Revision 0, "Response Time Test - Containment Isolation Phase B Slave Relay K618 - Train A." PTP personnel tested these relays by administratively changing the plant approved operating procedures and test instructions to accomplish the test programs stated acceptance criteria. The inspectors verified that prerequisite conditions were properly established, temporary modifications to the system were identified and controlled by approved procedures, and the test director was knowledgeable of expected test conditions and results.

The TD held a pretest briefing prior to conducting the test. The test was satisfactorily performed and the required data collected for evaluation. No significant test deviations were identified during the test.

During this inspection the inspectors discussed the current PTP status with licensee supervision and management, who identified the following information:

Control Air System - All major testing activities have been satisfactorily completed, with the exception of clearing test identified deficiencies.

Component Cooling System - Currently the prestart testing activities are on hold pending the start of TVA - 20B, (RT), "Flow Balance" testing. This activity is currently scheduled to begin this month.

Functional Analysis Report Status - The PTP has a total of 58 FARs identified as necessary for program completion, with 56 FARs prepared to date. The Joint Test Group has approved 47 FARs for use at the end of this inspection period.

Emergency Raw Cooling Water System - The licensee is still evaluating system readiness for testing.

The inspectors will continue to follow the licensee's PTP progress during future inspection periods.

No violations or deviations were identified in the areas inspected.

8. Action on Previous Inspection Findings (92701)

a. (Closed) 50-390/85-BU-01, 50-391/85-BU-01, Steam Binding of Auxiliary Feedwater Pumps

NRC IE Bulletin 85-01 informed licensees and construction permit holders of a potential serious safety issue involving the inoperability of auxiliary feedwater pumps (AFW) at certain facilities as a result of steam binding. Certain PWR licensees and all PWR construction permit holders were required to take action to prevent similar events from occurring at their facilities. The bulletin described numerous events at various facilities where hot water had leaked into the AFW system and flashed to steam disabling the AFW pumps. The problem could affect both motor driven and turbine driven pumps due to common discharge and suction lines used at many PWRs. INPO issued Significant Event Report (SER) 5-84 and Significant Operating Experience Report (SOER) 84-3 as a result of the problem. In December 1984, the NRC's Office of Nuclear Reactor Regulation (NRR) determined that steam binding of AFW pumps was a generic issue assigned as Generic Issue 93, "Steam Binding of Auxiliary Feed Water Pumps." The actions required by bulletin 85-01 for licensee included:

- Develop procedures for monitoring fluid conditions within the AFW system on a regular basis during times when the system is required to be operable. This monitoring should ensure that fluid temperature at the AFW pump discharge is maintained at about ambient temperature. Monitoring of fluid conditions, if used as the primary basis for precluding steam binding is recommended for each shift.

- Develop procedures for recognizing steam binding and for restoring the AFW system to operable status, should steam binding occur.

- Procedural controls should remain in effect until completion of hardware modifications to substantially reduce the likelihood of steam binding or until superseded by action implemented as a result of resolution of Generic Issue 93.

The licensee responded to the bulletin in R. L. Gridley's letter to Dr. J. Nelson Grace (NRC) dated January 27, 1986. The licensee reported the following action:

(1) Operations Section Letter OSLA-27, "Assistant Unit Operator Work Stations," was in place and required the Assistant Unit Operator (AUO) to record the motor driven and turbine driven AFW pump casing and discharge line temperatures once each shift. Any abnormal temperatures would be brought to the attention of the Shift Supervisor. OSLA-27 requires the readings be reviewed by the Unit Operator and the Assistant Shift Operations Supervisor.

(2) System Operating Instruction (SOI)-3.2, Auxiliary Feedwater System, contained a precaution and provisions that when the pump casing(s) are found to be hot, that the pumps are vented once every four hours until the cause is found and corrected.

The NRC issued Generic Letter 88-03, "Resolution of Generic Safety Issue 93, Steam Binding of Auxiliary Feedwater Pumps." The generic letter resolved the generic safety issue by continuing the recommended procedural controls required by NRC IE Bulletin 85-01.

TVA's response to the Generic Letter 88-03 dated June 3, 1988, reiterated their previous response to bulletin 85-01 and committed to continued compliance with the generic letter.

The NRC's, Office of Special Projects issued a Safety Evaluation Report (SER) dated July 20, 1988, which accepted the TVA response.

The inspector reviewed SOI-3.2, "Auxiliary Feedwater System," Revision 12, and OSLA-27, "Assistant Unit Operator Work Stations," and found that the procedures contain the provisions discussed above. The inspector found licensee's action on NRC IE Bulletin 85-01 and Generic Letter 88-03 adequate. Item 85-BU-01, "Steam Binding of Auxiliary Feedwater Pumps," is closed.

b. (Open) CDR 50-390/87-05, 50-391/87-05, Repair Dispositioning For Construction Nonconformance Reports

In an interim report, dated February 11, 1987, the licensee identified the following deficiency:

- The Division of Nuclear Engineering (DNE) Engineering Assurance (EA) conducted an audit of Watts Bar Engineering Project (WBEP) activities related to the handling of construction nonconformance reports (NCRs). The audit placed emphasis on NCRs with "use-as-is" or "repair" dispositions to ensure that these dispositions were adequately justified and design safety margins were not compromised. The audit identified one deficiency with four concerns:

  (1) "Use-as-is" and "repair" disposition NCRs are not tracked against the affected document, therefore, in most cases for NCRs designated as not requiring a drawing change, there is no retrievable, consolidated record of the accepted variations from the drawing or original design. The cumulative effect of the design on the margin of safety is indeterminate. Very little evidence could be found to indicate that these NCRs have received the same level of independent design verification and interdisciplinary reviews as the original design.

  (2) "Use-as-is" dispositioned NCRs that come under the ASME Code that are designated as not requiring a drawing change also do not meet ASME code requirements, since the NCR cannot be readily linked to the drawing to indicate as-constructed configuration. NCRs dispositioned as requiring a drawing change did not exhibit these problems since the drawing, NCR, and ECN are all cross-referenced.

  (3) Many "use-as-is" dispositioned NCRs either do not have any justification or lack adequate justification detail (i.e., references to support documents or analysis) which make it difficult or impossible to trace the justification without recourse to someone familiar with the condition described.

  (4) There did not appear to be any project procedural guidance for the handling of NCRs. It was recognized that division guidance was also lacking, and that was referred to the

Engineering Assurance Procedures Group for resolution. The project, however, must have some interim and detailed implementing guidance to ensure NCRs are adequately and consistently handled.

The condition applied to WBN CAQs initiated by the Division of Nuclear Construction (DNC), Site Director's Office (SDO), and vendors that were sent to DNE and dispositioned by DNE as "use-as-is" or "repair." The cause was attributed to inadequate project procedures, and inadequate training of personnel to the standards of ANSI N45.2-1971.

The safety implication of the deficiency is that inadequate documentation of "use-as-is" and "repair" dispositions had potential for an adverse effect on safety. The cumulative effect of past dispositions were not documented and available for consideration in reviewing later design changes.

The licensee reported that a corrective action plan was being developed and the following actions were being considered:

- Identify the WBN CAQs that had a final disposition of either "use-as-is" or "repair."

- For the CAQs above, identify those that had no design drawings or documents issued as a result of the final disposition being "use-as-is" or "repair."

- For the CAQs identified in B, identify the design documents that contain the design requirements that were not met as described in the CAQ.

- For each design document identified in step C, perform a technical review of the latest revision of the document and consider what effect the condition described by the CAQ has on the document. Either prepare or revise a calculation to technically justify the current revision of the document and indicate what cumulative effect, if any, that the CAQs have on the document as to technical adequacy, design margin, conformance to criteria, and FSAR commitments. Revise the document to either reflect the "as-constructed" configuration represented by the CAQ or to post the CAQ number on the drawing as a reference.

- Issue a matrix drawing that cross references the CAQs identified in step B and the affected documents that were revised to incorporate the CAQs.

- Issue a memorandum from the WBEP Project Engineer to the DNC-WBN Project Manager and WBN Site Director with the matrix drawing attached with instructions to file the memorandum and matrix drawing with each CAQ listed on the matrix drawing.

Recurrence control was to be addressed by training and revision of procedures. The licensee committed to completion of all corrective action by unit 1 fuel load. The interim report stated that a schedule for the corrective action submittal would be determined as resources to conduct were acquired.

In a second interim report, dated February 29, 1988, the licensee reported the following:

- 9,655 CAQs had been screened for disposition determination. Of that number, 3,766 CAQs were dispositioned either "use-as-is" or "repair." The 3,766 was split into two groups, 3,066 for unit one and common and 700 for unit 2. Of the 3,066, 654 were CAQs which could have an impact on the Hanger and Analysis Update Program (HAAUP) and the remaining 2,412 had no HAAUP effect. Of the 654 hanger-related CAQs, 206 required some form of output document revision. None of the 206 were considered to be of a significant nature. The licensee reported similar detailed evaluations were currently in progress for the non HAAUP CAQs. The licensee also reported that a new procedure had been issued to control the process and that WBEP managers had been trained on the procedure.

In a final report dated September 14, 1988, the licensee reported the following:

- A total of 9,132 CAQs had been screened for disposition determination. Of that number, 2,753 CAQs were dispositioned either "use-as-is" or "repair." The number included 2,062 for Unit 1 and common, and 691 for Unit 2. The numbers were different from the second interim report as a result of the elimination of NCRs invalidated before issuance, redundant revisions of the same NCR, and NCRs which were dispositioned by engineering after recurrence controls were in effect. The licensee reported that of the unit 1 and common CAQs, design output documents will need to be revised to reflect the "use-as-is" or "repair" disposition of 479 CAQs. Further evaluation would be required to determine if design output document revisions will be required for an additional 478 Unit 1 and common CAQs.

The inspector reviewed the corrective action discussed above, selected records and documentation associated with the action, and held interviews and discussions with personnel responsible for the corrective action. To provide procedural guidance for the resolution of the deficiency, the licensee used an approved corrective action implementation plan which included objectives, scope, description, interfaces, assumptions, and provisions for verification and QA oversight. The inspectors reviewed the WBN engineering project implementation plan entitled, "Implementation Plan For Corrective Action to Resolve Engineering Assurance Audit Deficiency No. 86-27-01 and SCR WBN WBP 8601 - 'Use-As-Is' And 'Repair' Dispositions For Conditions Adverse To Quality," Revision 3, and concluded the plan provided appropriate controls to conduct an effective review and evaluation. The inspector discussed the mechanics of the review with the lead engineer and the traceability from the original NCRs through the design output documents that the NCRs affected.

The inspector found that Stone and Webster Engineering Corporation (SWEC) was contracted to conduct the review of CAQs prior to April 1, 1987, when recurrence controls were put in place. The review was conducted under SWEC Task Procedure, WBTP No. 005-2, "Conduct Reviews Use-As-Is/Repair CAQs," Revision 2. The review resulted in a matrix listing of CAQs which linked the disposition of "use-as-is" and "repair" CAQs to output documents. The matrix provided a QA document which would relate the disposition to an output document change, calculation, or other documented justification for the disposition. The SWEC review resulted in the identification of 1,962 unique CAQs that were classified as "use-as-is" or "repair," prior to April 1, 1987. The SWEC review resulted in the following classification:

- 914 CAQs had dispositions with an Engineering Change Notice (ECN) or Field Change Notice (FCN) issued. These CAQs were removed from the scope of the review as the ECN/FCR process provided documentation, review, and approval of the disposition and resulting change.

- 108 CAQs were evaluated as requiring no design output document change and having no affect on design or documentation. These were classified as being satisfactory "as-is."

- 447 CAQs resulted in dispositions which SWEC was unable to evaluate for reasons stated in the technical review sheet. These required resolution by TVA. SWEC prepared preliminary DCNs for TVA resolution. Ebasco evaluated about 300 of the technical review sheets and TVA performed the rest. Of the 447, about two-thirds were resolved with technical justifications and one-third by justifications and issuance of DCNs to change output documents.

- 455 CAQs resulted in dispositions which were evaluated as requiring a design output change. About 200 DCNs were issued to effect changes to output documents. The 200 DCNs enveloped changes for the majority of the 455 CAQs. CAQs that did not result in DCNs were technically justified.

- 38 CAQs were evaluated as unit 2 CAQs which would effect Unit 1 startup. The majority of the these CAQs did not affect design output documents.

The inspector found that the TVA review included screening 742 deficiencies from the preoperational test program. The screening was conducted by Ebasco and resulted in reopening 57 preoperational test deficiencies for reevaluation. TVA also reviewed 345 Westinghouse deviation notices that were "use-as-is" or "repair" and found the notices cause no changes that would affect WBN design.

The inspector found that recurrent control had evolved over the period that the reviews were being done. As reported in the licensee's interim report, recurrence control was provided by WBEP procedure EP 43.23, "Conditions Adverse to Quality - Reporting and Correcting," Revision 0. In the second interim report, the licensee had issued WBEP 3.05, "Condition Adverse To Quality Reports and Problem Identification Reports," which superseded EP 43.23. Subsequent to the licensee's final report, the project guidance in WBEP 3.05 was incorporated into plant administrative procedures and WBEP 3.05 was cancelled. Administrative procedures AI-2.8.5, "Conditions Adverse to Quality - Corrective Actions," Revision 5, AI-2.8.14, "Corrective Action," Revision 3, and AI-2.8.15, "Corrective Action - WBN," Revision 0 contained programmatic recurrence control. Nuclear engineering procedure (NEP) 9.1, "Corrective Action," provided recurrence control guidance for nuclear engineering. The inspector found that the guidance was adequate to prevent recurrence with proper implementation.

The inspector reviewed a selection of "use-as-is" resolution sheets and NCR packages from the SWEC and TVA review and discussed the packages with the responsible engineer. The information reviewed included:

- "Use-As-Is/Repair" Westinghouse Deviation Notices
- "Use-As-Is/Repair" technical resolution sheets
- NCR packages for NCRs 851, 1086, 1923, 2016, 2746, 3702, 4138, 4265, 4812, 5062, and 5484

The inspector concluded based on the sample reviewed that the resolutions were appropriate for the stated deficiencies. The inspector found that the matrix provided traceability between the NCRs, the resolution/justification for disposition and the affected design output document. Where no design output document was affected, the matrix provided traceability to the specifications, other documents affected and the technical justifications. The inspector also reviewed quality assurance audit report WBA89003, "Correction of Deficiencies (16) and Corrective Action (23)." The audit was part of the required oversight of the "use-as-is" and "repair" program. The audit report indicated that no major deficiencies in the corrective action program were identified.

The inspector found that the corrective action for the "use-as-is" and "repair" CAQ dispositioning prior to April 1, 1987, was adequately addressed by the licensee. This inspection scope for this issue was limited to the deficiency discussed above in the 10 CFR 50.55 (e) construction deficiency interim and final reports.

CDR 50-390/87-05, 50-391/87-05, "Repair Dispositioning For Construction Nonconformance Reports," is left open pending the licensees final certification of the completion of this program that is addressed by the licensee as a special program.

c. (Closed) CDR 50-390/85-20, Potential Interaction of Flux Mapping System and Seal Table

In a final report, dated July 15, 1985, the licensee identified the following deficiency:

- There was the possibility of interaction between the flux mapping system which is non-nuclear safety, and the seal table which is part of the reactor coolant system pressure boundary, due to seismic loading. The flux mapping system and the seal table were supplied as part of Westinghouse's nuclear steam supply system (NSSS). At Watts Bar Nuclear Plant (WBN) the flux mapping transfer cart is suspended from a rail car and track over the seal table. The flux mapping system components (including rail car) were not seismically qualified and potentially could fall onto the seal table during a seismic event. The interaction was not considered during the initial design, layout, and installation by the NSSS supplier (Westinghouse) or the architect-engineer (TVA.) The safety implication is potential for the flux mapping system components to fail during a seismic event and damage the flux thimble guide tubes causing a LOCA.

The following corrective actions were established by the licensee:

- TVA performed a seismic analysis to qualify the flux mapping system. Results of the analysis indicated that structural modifications would be required on the existing flux mapping system. Modifications would be accomplished through ECN 5765 for unit 1 and ECN 5766 for unit 2. TVA committed to complete the corrective action on unit 1 prior to initial criticality.

  The licensee reported that in order to prevent recurrence of this condition, Westinghouse plans to review and revise all necessary flux mapping documentation to clarify all interface criteria, including seismic and structural criteria.

The inspector reviewed ECN 5765 and the associated documentation which implemented the modification to upgrade the flux mapping system to a class I system. Supporting calculations were in place and were conducted using the Georgia Technical Research Corporation "GTSTRUDL" program. The work plan, signed off on January 28, 1987, completed the required modifications to the flux mapping system. The inspector concluded that the licensee's action was adequate. CDR 50-390/85-20, Potential Interaction of Flux Mapping System and Seal Table, is closed. CDR 50-391/85-19 for Unit 2 remains open pending completion of modification work on Unit 2.

d. (Closed) CDR 50-390/86-38, 50-390/86-25, Construction Procedure Does Not Implement G-32 Requirements

In a final report, dated April 26, 1986, the licensee identified the following deficiency:

- Nonconforming Condition Report (NCR) 6556 initiated on January 7, 1986 identified that some of the requirements of TVA General Construction Specification G-32 were not implemented in WBN Quality Control Procedure (QCP) 1.14, "Inspection and Testing of Bolt Anchors Set in Hardened Concrete and Control of Attachments to Embedded Plates." Specifically, G-32 provided requirements for spacing between concrete anchors and adjacent embedded strip inserts. If a wedge bolt anchor or grouted anchor was installed less than one inch from a strip insert, section 3.10.2 of G-32 required special installation procedures, usually grouting. The requirement was not implemented in WBN QCP-1.14. Additionally, WBN QCP-1.14 did not address G-32 requirements for attachments to strip inserts. TVA determined that the deficiency resulted from an inadequate review of upper tier documents. The safety implication is that inadequate installations could result in reduced shear capacity for anchor loading and affect plant safety.

The licensee established the following corrective actions:

(1) Perform field walkdowns of all embedded strip inserts at WBN Unit 1 and Unit 2. Support anchors which were installed less than one inch from a strip insert and which did not meet the requirements of G-32 would be reworked, or a support variance sheet submitted for engineer evaluation.

(2) To prevent recurrence, TVA will revise WBN QCP-1.14 to implement all G-32 requirements of sections 3.7.3.2, 3.7.3.3, 3.10.2.1, and 3.10.2.2.

(3) TVA stated in the final report that WBN Quality Control Instruction (QCI) 1.10, "Preparation and Control of Quality Control Instructions, Procedures, and Tests," was revised on September 20, 1985, to require responsible TVA Office of Construction (OC) engineering units to conduct a review of applicable procedures to ensure that all upper-tier requirements are incorporated.

The inspector reviewed WBN QCP-1.14, "Inspection and Testing of Bolt Anchors Set in Hardened Concrete and Control of Attachments to Embedded Features," Revision 18, and found that it had been revised to incorporate G-32 requirements on May 30, 1986, however, the procedure was subsequently cancelled. Construction Process Instructions (CPI) 8.1.8-G 100, "Expansion, Grouted, and Undercut Anchors," Revision 1, and CPI 8.1.8-H 100, "Fabrication, Installation and Documentation of Pipe Supports, Revision 3, were issued on February 15, 1989, and incorporated the guidance of the cancelled procedure. The inspector reviewed QCI-1.10, "Preparation and Control of Quality Control Instruction, Procedures, and Tests," Revision 11, addendum 2 and found that addendum 2 added a new section 6.1.3 which required engineering units to review General Construction Specifications and Project Constructions Specifications to ensure incorporation upper-tier requirements. The inspector considered the programmatic changes adequate to prevent recurrence.

The inspector reviewed work plan N6694-1, completed on October 11, 1989, which implemented the walkdowns of Unit 1. The walkdown did not identify any anchors that violated G-32 requirements regarding embedded spot or strip inserts, therefore, no rework or variances were required. Walkdowns of Unit 2 were conducted in 1986 for Unit 2 by hanger quality control personnel and no G-32 specification violations were identified. The inspector concluded that licensee action on this issue was adequate. CDRs 50-391/86-38, and 50-391/86-25, Construction Procedure Does Not Implement G-32 Requirements, are closed.

e. (Closed) URI 50-390, 50-391/88-01-03, Control Air Quality

Information Notice No. 87-28, "Air System Problems At U.S. Light Water Reactors," supplement 1 to notice 87-28, and NUREG-1275, Volume II addressed problems that have occurred at various plants because of design and maintenance practices on air systems. Design, operation, and maintenance of air systems in some cases was inadequate due to air systems not being classified as safety related. NRC inspection report 88-01 identified URI 88-01-03 due to questions regarding the requirements, maintenance, and testing of the Station Control and Service Air (SCSA) system, Auxiliary Control Air (ACA) system, and the air quality of those systems. During the inspection that identified the unresolved item, the licensee agreed to consider changing procedures to include periodic tests or inspection verifying control air quality.

Subsequent to the inspection, NRC issued Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," requiring each licensee to perform a design and operational verification of the instrument air system to include:

(1) Verification by test that actual instrument air quality is consistent with the manufacturers recommendations for individual components served.

(2) Verification that maintenance practices, emergency procedures, and training are adequate to ensure that safety-related equipment will function as intended on loss of instrument air.

(3) Verification that the design of the entire instrument air system including air or other pneumatic accumulators is in accordance with its intended function, including verification by test that air-operated safety-related components will perform as expected in accordance with all design-basis events, including a loss of the normal instrument air system. This design verification should include an analysis of current air operated component failure positions to verify that they are correct for assuring required safety functions.

- In addition each licensee/applicant should provide a discussion of their program for maintaining proper air quality.

In a detailed response to generic letter 88-14, dated February 23, 1989, the licensee reported the following (major points of licensee action have been summarized):

Action 1

The Design Basis Document and components specifications were reviewed to establish and substantiate the original design basis. The SCSA and ACA air quality was verified under preoperational test TVA-27 and 27B. The test results were consistent with design requirements. Since preoperational testing, a rotary screw compressor was added to the service air system that could be cross-connected to the SCSA system. The rotary screw compressor has a sump separator and a two stage air purifier. In addition, as service air is used for breathing, TVA's Occupational Hygiene Group (OHG) performs samples every six months. The air systems have high humidity alarms set at six percent. Plant procedures will be written to require air quality sampling on all three SCSA and ACA headers on a six month basis. A review of drawings indicated that all but 11 components had filters or filter-regulators. The 11 components were evaluated as not requiring filter installation. One of the 11, upon field inspection was found to have a filter. For the remaining 10 components (5 in unit 1, and 5 in unit 2), the 1/4-inch Asco solenoid valves were being replace with one-inch valves which have course screen filters installed. The one-inch valves were also not prone to clogging.

Action 2

Maintenance procedures are in place and provide for inspecting/replacing filters, changing desiccant, replacing soft seats in valves and replacing diaphragm in pneumatic valve actuators. Maintenance procedures incorporate manufacturers recommendations. Maintenance procedures will be further reviewed to verify compliance with vendor maintenance recommendations. Operating procedures were in place. A list of WBN air-operated, safety-related components has been prepared and will be reviewed against existing procedures. Deficiencies will be evaluated and resolved. Operators are trained yearly on a loss of control air. Air system maintenance is included in mechanical maintenance training lesson plans.

Action 3

Upon completion of the review of INPO Significant Operating Event Report (SOER) 88-1, TVA will implement improvements to its pneumatic accumulators and check valves and their maintenance testing procedures as required. Air-operated valves supplied by the ACA system were tested by preoperational test TVA-27B. Safety-related SCSA valves were not similarly tested. The testing history of SCSA valves will be reviewed and valves will be similarly tested as required. Engineering will evaluate the use of rotary screw air compressor and procedures will be revised as required based on the evaluation. A review of the fail position of valves will be conducted.

In a revised response, dated July 12, 1990, the licensee reported that all commitments and actions with Generic Letter 88-14 were complete.

> The licensee indicated that a more conservative approach to testing of SCSA valves would be taken. All safety-related SCSA valves would be tested for both a rapid and gradual loss of control air.

The licensee's corrective action provided in the initial and revised response were acknowledged by NRC letter dated July 26, 1990, as being appropriate. The acknowledgement indicated that corrective actions were subject to future audits and verification.

The inspector reviewed Information Notice 87-28, Generic Letter 88-14 and the licensee's corrective action. The inspector reviewed the following documentation which provided programmatic controls and implemented corrective action on the SCSA and ACA air system:

- MI-32.1, "Auxiliary Control Air Compressor Inspection and Rebuild," Revision 0
- SOI-32.1, "Auxiliary Air System," Revision 10

- SOI-32.1, "Auxiliary Air System," Revision 10
- SOI-33.1, "Service Air System," Revision 7
- AOI-10, "Loss of Control Air," Revision 8
- TI-27, "Cleaning and Cleanliness of Fluid Systems and Components," Revision 28
- TI-104, "Instrument Air Quality," Revision 0
- DCN P-04444-A, Air Filter Modification
- FSAR section 9.3.1.2 change submittal
- Test Scoping Document TVA-27, "Control Air System," Revision 2, units 1 and 2

The inspector concluded that the control procedures provide for adequate control and maintenance of air systems important to safety. The inspector further concluded that the licensees action in response to generic letter 88-14 was appropriate. URI 88-01-03, Control Air Quality, is closed.

f. (Closed) Violation 50-391/86-11-02, Inspection of The Flexible Conduit to Damper Operator

NRC inspection report 86-11 identified a violation for failure to adequately inspect safety related installations in that a flexible conduit was found to be interfering with the operating rod of air operated damper 1-FCO-30-55. In the response to the violation, dated July 17, 1986, the licensee admitted the violation occurred. The licensee's evaluation was that the cause was from work in the area disturbing the configuration rather than inadequate inspection during installation. The licensee completed maintenance work request A569231 to correct the deficiency on July 15, 1986. The NRC acknowledged the licensee's response to the violation in a letter dated August 4, 1986.

The inspector reviewed the violation and the licensee's response. A field inspection of the conduit installation was also performed. The conduit had been adequately rerouted to prevent the possibility of recurrence. The inspector concluded that the licensee's corrective action was appropriate. Violation 50-391/86-11-02, Inspection of The Flexible Conduit to Damper Operator, is closed.

g. (Closed) CDR 50-390/85-57, Failure to Use a Support Design Per Analysis

In a final report, dated November 25, 1985, the licensee reported the following:

- During a design review of piping analysis problem No. N3-67-9A, TVA personnel identified a support which was shown at node OD6 on analysis isometric 47W450-213 that had not been incorporated into the plant design. The deficiency was documented as Significant Condition Report (SCR) WBN CEB-8526. The support was required for the 1-1/2 inch return line of the reactor building instrument room water chillers which connects to the essential raw cooling water (ERCW) discharge lines. The cause of the omission was reported to be an isolated instance of human error which occurred during the scoping of the isometric drawing. The safety implication of the missing support would be the potential for line failure causing flooding of the penetration room at elevation 713 feet and surrounding areas.

The licensee established the following corrective actions:

(1) The missing support would be incorporated into plant design through Engineering Change Notice (ECN) 5961 and will be installed prior to unit 1 fuel load.

(2) The Pipe Support Design Manual will be revised to standardize the procedures concerning scoping analysis isometrics for need support design work.

The inspector reviewed ECN 5961 which implemented the corrective action. The ECN effected changes to the ERCW drawing 47W450-213 and added the required support to the drawing. Revision 2 to the Pipe Support Design Manual was approved on February 14, 1986, and revised section 5.5.5, Scoping Review, of the manual. The revision added additional requirements for verification that all supports required by analysis were included in the system design. The inspector reviewed work plan E5961-1 which installed the missing support and performed a field inspection on the installed support. The inspector found the support installed in conformance with the installation design drawing. CDR 50-390/85-57, Failure to use support Design Per Analysis, is closed.

h. (Closed) CDR 50-390/86-36, Failure to Perform Weld Calculations For Platforms

In an interim report, dated April 18, 1986, the licensee reported the following deficiency:

- During a review of a proposed field change request (FCR) for WBN Unit 2, it was determined that no weld calculations were computed for approximately 100 connections on WBN unit 1 structures shown on drawings 48W902 and 48W903. Nonconformance report WBN CEB-8627 documented the deficiency. The affected structures consisted of three access platforms for the 5 and 10 path rotary transfer system for the incore flux mapping probes. The rotary transfer system was also supported from the affected platforms. TVA determined that the deficiency was a design oversight. A calculation package for each platform existed, but did not address the weld calculations. The same personnel prepared the calculation packages for the affected platforms. The safety implication of the deficiency was that the transfer system directly above the seal table had potential to fail during a seismic event and cause a small break LOCA by damaging the bottom-mounted instrumentation (BMI) guide tube stubs.

As interim corrective action, TVA initiated engineering change notice (ECN) 6265 to perform an evaluation on each affected weld connection for weld adequacy.

In a final report, dated August 22, 1986, the licensee reported the following additional information and final corrective action on the deficiency:

- A calculation package for each of the affected platforms was developed, but the packages did not address weld calculations. These calculations were required for ECN 3255 which added attachments to the platforms. In performing the calculations for the ECN, personnel assumed that the additional loading would not significantly affect the weld connections. Therefore, no weld calculations were performed.

The licensee established the following corrective actions:

(1) TVA performed an analysis that indicated structural bracing was necessary to minimize the amount of rework to be done on the affected weld connections.

(2) Work plan E6265-1 which implemented ECN 6265 was completed on October 1, 1987. The work plan added structural bracing to the platform and accomplished rework on some welds.

(3) Calculations were completed to support qualification of the platform welds.

(4) The individuals responsible for the occurrence of the deficiency were notified of the deficiency and the need to eliminate that type of error.

(5) TVA's Nuclear Engineering Procedures have been comprehensively revised to provide more effective standardized requirements for design control and review. This will prevent recurrence of the subject deficiency.

The inspector reviewed the licensee's corrective action and documentation associated with the deficiency. Nuclear Engineering Procedure (NEP) 3.1, "Calculations," Revision 1, PCN 4 was revised on September 27, 1987, to include more definitive guidance on the performance of calculations. Calculations were completed for the welds and documented under RIMS B41860701952 and B41860701953. The design drawings for the platforms 48W902 and 48W903 were updated by the ECN to reflect the additional bracing and weld rework on the platforms. The inspector concluded that the licensee's actions were appropriate. CDR 50-390/86-36, Failure to Perform Weld Calculations For Platforms, is closed.

i. (Closed) CDR 50-390/88-03, Internal Bolts Corroded In Raw Water Check Valves

(Closed) 50-390/89-BU-02, Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design

In a final report, dated June 1, 1988, the licensee reported the following deficiency:

- TVA identified a deficiency at WBN involving the bolting material used inside certain Atwood and Morrill check valves installed on the Essential Raw Cooling Water (ERCW) system. The 3/8-inch bolts which secure the internal flapper assembly to the valve body were experiencing accelerated corrosion. Two valves in Unit 2 (2-CKV-67-565B and 2-CKV-67-568B) and one valve in Unit 1 (1-CKV-67-565B) contained bolts which had corroded sufficiently to cause bolt failure. Analysis by TVA's Singleton Materials Engineering (SME) Laboratory showed the bolts to be made of 410 stainless steel, used for American Society of Testing and Materials (ASTM) A-193 Grade B6 bolts. This was the material specified on the vendor drawing. According to SME, the failure mechanism involved stress corrosion cracking, and this type of martensitic stainless steel is known to be susceptible to stress corrosion cracking when stressed (torqued) and exposed to chlorides. The ERCW system constitutes an aqueous chloride environment because of the sodium hypoclorite added to the water to control Asiatic clams.

  The root cause of the deficiency was that incomplete information was provided to the vendor in the contract regarding the fluid service for the check valves being purchased. The design specification indicated raw water service but made no mention of sodium hypoclorite additions, therefore, the vendor was unaware of the chloride environment. Three valves were installed in each of four loops which supply cooling water to a lower containment vent cooler, control rod drive vent cooler, and reactor coolant pump cooler. The deficiency affects 12 valves for each unit and spares. Although the containment vent coolers were not safety-related, they are being upgraded to safety-related as a result of CAQR WBN870061 (WBRD 50-390/87-22). The safety implication of the deficiency is the potential for exceeding equipment environmental qualification temperature limits during certain accidents due to a loss of vent cooler function.

The licensee established the following corrective actions:

(1) TVA will replace the flapper assemble hold-down bolts on 26 (24 installed and 2 spares) Atwood and Morrill checks valves. TVA has coordinated with Atwood and Morrill to replace the existing bolts with bolts fabricated from ASTM F583, Allow 630, a material better suited for the service.

(2) TVA will review documentation for all other valves used in safety-related raw water systems at WBN to determine if 410 series stainless steel has been used for internal fasteners and take appropriate corrective action.

(3) Although valves less than 2-1/2 inches are compact in design and generally do not utilize stressed fasteners to secure the internals, TVA will include them in the review.

(4) To prevent recurrence, TVA will revise the following standard specifications:

- MEB-SS-10.14, "Non-ASME Section III Valves - 2.5 Inch and Larger for TVA Projects"
- MEB-SS-10.15, "ASME Code Valves - 2.5 Inch and Larger for TVA Projects"
- MEB-SS-10.18, "Non-ASME Section III Valves - 2 Inch and Smaller For TVA Projects"
- MEB-SS-10-19, "ASME Code Valves - 2 Inch and Smaller for TVA Projects"

Subsequent to the licensee's final construction deficiency report, the NRC issued NRC Bulletin No. 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design." The NRC requested the following action:

(1) For all applicants for operating licenses.

(a) All licensees of operating reactors are requested to disassemble and inspect all safety-related Anchor Darling Model S350W swing check valves supplied with internal retaining block studs of ASTM specification A193 Grade B6 Type 410 SS. Licensees should review the design of other safety-related check valves to determine if similar designs and material selection to the Anchor Darling Model S350W are used. If so, such valves should be similarly inspected. The inspection by disassembly should be performed as follows:

- If any of the internal bolting is to be reused, it should be inspected for cracks using surface inspection techniques (penetrant or magnetic particle). Cracked bolting should be replaced and a failure analysis performed including chemical analysis to confirm material type.

- If all suspect bolting is to be replaced with bolting of material and hardness specified below, surface inspection and failure analysis of the old bolting may not be needed unless an unexpected failure mechanism is evident.

- Reused and new bolting should be hardness tested for a maximum Rockwell hardness value of Rc26. Any internal preloaded bolting that does not meet the hardness requirements should be replaced by bolts of the same material with a maximum Rockwell hardness of Rc26 or by an alternate material approved by the valve manufacturer.

(b) The implementation of the actions requested is requested to be complete before fuel loading, or, if fuel loading occurs within 90 days of receipt of this bulletin, at the first refueling outage after receipt of this bulletin.

In response to NRC Bulletin 89-02, dated April 25, 1990, the licensee reported the following:

- Watts Bar Nuclear plant did not have safety-related Anchor Darling Model S350W swing check valves as described in the bulletin, but they did have valves of similar material. The problem was previously documented to NRC as stated above in the construction deficiency report.

The licensee established the following additional corrective action:

(1) As a result of the issuance of the bulletin, WBN expanded the scope of the review to include other safety related systems. Suspect material similar to 410 stainless steel was evaluated. Lists of acceptable material and suspect were developed from that evaluation. The review consisted of evaluating pins, shafts, and bolts in valves designed similar to the Anchor Darling valves. The review identified four additional 2-inch Atwood and Morrill swing check valves that required bolt replacement in the ERCW System and one 6-inch Atwood and Morrill Swing check valve that required pin replacement in the Component Cooling Water System.

(2) In lieu of testing, WBN chose to replace the suspect bolting or pins with bolting or pins of an acceptable material. The safety-related swing check valves required for unit 1 operation have had bolts or pins replaced as necessary.

The inspector reviewed the licensee's corrective action associated with the construction deficiency report and NRC Bulletin 89-02. The standard specifications associated with check valves had been revised to specify the use of ASTM A564 type 630 and ASTM A193, Grade B8M material. The ASTM F593 alloy 600 bolts are fabricated from ASTM A564 type 630 material. The revision prohibited the use of AISI type 410 stainless steel. The inspector reviewed workplan C-WBN880236-1 which implemented the unit 1 corrective action and had no questions. The licensee had documented the survey of safety-related check valves that the corrective action was based upon. CDR 50-390/88-03, Internal Bolts Corroded In Raw Water Check Valves is closed. NRC bulletin 89-02, Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design, action for Unit 1 is closed. Unit 2 action for NRC Bulletin 89-02 remains open.

j. (Open) URI 390,391/89-08-02: Cable Damage Issues

This is a continuation inspection of cable damage issues discussed in Inspection Reports (IR) 89-08 with follow-up in 89-11, 89-13, 89-18, 89-20, 90-03, 90-06, 90-09, 90-12, and 90-17.

During this inspection period the licensee notified the inspectors of a condition involving cable damage found in a section of conduit classified as low risk. The cable damage was found by the licensee when a section of cable classified as high risk (cable damage high risk) was removed in preparation for replacement.

The high risk cable was installed in a 3-inch conduit that ran from junction box 1-JB-292-1329-G to conduit 1-2PM-290-6219-G, a distance of approximately 265 feet. The cable continued in 3-inch conduit (1-2PM-290-6219-G) to tray 0-2-TRAY-290-1/2-G (a distance of approximately 16 feet) that was classified by engineering analysis as low risk.

This low risk section also was not selected for hi-pot testing because the distance is less than 20 feet between pull points, a minimum length criteria used in selecting the hi-pot sample. The low risk section was removed to facilitate removal of the high risk. The conduit contained 19 cables, Unit 1, V-2 cables, located in the Control Building. Four of the 19 cables contained apparent pull-by damage. Two of the cables had very minor type surface depressions and the outer jacket was not penetrated. On the other two cables (1-2PM-068-1026-G and 1-2PM-003-1381-G) no apparent damage was noted to the insulation of the individual conductors, but the outer jacket and shield were damaged (cut through) in one location on cable 1-2PM-068-1026-G. The other cable had one location where the outer jacket was penetrated.

The damaged location corresponds to a 90 degree elbow in the conduit run. The cable was installed with 5 pull-bys. Twelve cables were pulled on the initial installation in the conduit run from junction box 1329G to cable tray 2G1. Of those 12 cables, three were found to have cable damage. The cable with the most significant damage was in the first 12 cables installed. The next installation contained one cable. Then another single cable was installed on the second pull-by and this cable had cable damage. Three cables were installed on the third pull-by and this installation resulted in the highest cable pull tension and sidewall bearing pressure. However, the calculated pull tension and sidewall bearing pressure were both well within the acceptable range.

On the fourth and fifth (last) pull-by one cable was installed. The 19 cables were installed in the period of 05-18-79 thru 09-14-81. The licensee reported that in the high risk section of the cable that was removed, no cable damage was found. The licensee is investigating the effects of the noted damage to assess whether program changes are needed. This unresolved item remains open pending further reviews and action to resolve the cable damage issue.

k. (Closed) IFI 50-390/85-51-18: NRC to Evaluate Watts Bar Analysis Method to Determine Equivalency to Technical Specifications

The licensee is in the process of replacing their Technical Specifications with a new version. Inspection of equivalency between the new Technical Specification requirements and analysis methods will be performed during a preoperational inspection. For tracking purposes, this item is administratively closed.

9. Exit Interview

The inspection scope and findings were summarized on August 17, 1990, with those persons indicated in paragraph one. The inspectors described the areas inspected and discussed in detail the inspection results listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection. Dissenting comments were not received from the licensee.

Item Number	Status	Description and Reference
390/85-20	Closed	CDR - Potential Interaction of Flu Mapping System and Seal Table (Paragraph 8.c)
390/85-57	Closed	CDR - Failure to Use a Support Design Per Analysis (Paragraph 8.g)
391/86-11-02	Closed	VIO - Inspection of the Flexible Conduit to Damper Operator (Paragraph 8.f)

390/86-38 390/86-25	Closed	CDR - Construction Procedure Does Not Implement G-32 Requirements (Paragraph 8.d)
390/86-36	Closed	CDR - Failure to Perform Weld Calculations for Platforms (Paragraph 8.h)
390/87-05 391/87-05	Open	CDR - Repair Dispositioning for Construction Nonconformance Reports (Paragraph 8.b)
390/88-01-03 391/88-01-03	Closed	URI - Control Air Quality (Paragraph 8.e)
390/88-03	Closed	CDR - Internal Bolts Corroded in Raw Water Check Valves (Paragraph 8.i)
390/89-08-02 391/89-08-02	Open	URI - Cable Damage Issue (Paragraph 8.j)
390/90-20-01	Open	VIO - Failure to Follow QC Procedures (Paragraph 5)
390/90-20-02	Open	URI - Anchor Bolt Installation Practices (Paragraph 5)
390/90-20-03	Open	URI - Selective UT Examination Methodology (Paragraph 6)
390/90-20-04	Open	URI - QC Signature and Dating Requirements (Paragraph 3.a)
390/90-20-05	Open	URI - Isolation of License Required Fire Protection Equipment (Paragraph 3.b)
390/90-20-06	Open	URI - High Pressure Fire Protection - MIC (Paragraph 3.c)
390/85-BU-01 391/85-BU-01	Closed	BU - Steam Binding of Auxiliary Feedwater Pumps (Paragraph 8.a)
390/89-BU-02	Closed	BU - Stress Corrosion Cracking of High-Harding Type (Paragraph 8.i)
390/85-51-18	Closed	IFI - Equivalency To Technical Specifications (Paragraph 8.k)

10. List of Acronyms and Initialisms

ACA	Auxiliary Control Air
AI	Administrative Instructions
AFW	Auxiliary Feedwater
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Materials
AUO	Assistant Unit Operator
BMI	Bottom Mounted Instrument
BU	NRC Bulletin
CAP	Corrective Action Program
CAQR	Condition Adverse to Quality Report
CAT	Corrective Action Tracking
CDR	Construction Deviation Report
CEP	Construction Engineering Procedure
CFR	Code of Federal Regulation
CI	Concerned Individual
CPI	Construction Process Instructions
CRDR	Control Room Design Review
CTS	Cable Tray and Supports
C&S	Conduit and Supports
DBVP	Design Base Verification Program
DCA	Design Change Authorization
DCN	Design Change Notice
DNC	Department of Nuclear Construction
DNE	Division of Nuclear Engineering
DNQA	Department of Nuclear Quality Assurance
EA	Engineering Assurance
ECN	Engineering Change Notice
ECP	Employee Concern Program
EP	Emergency Plan
EQ	Equipment Seismic Qualification CAP
ERCW	Essential Raw Cooling Water
FCN	Field Change Notice
FCR	Field Change Request
FPHS	Fire Protection Hose Station
FSAR	Final Safety Analysis Report
HAAUP	Hanger Analysis and Update Program
HPFP	High Pressure Fire Protection
HVAC	Heat, Ventilation and Air Conditioning
IE	Inspection and Enforcement
IN	NRC Information Notice
INPO	Institute of Nuclear Power Operation
LOCA	Loss of Coolant Accident
MIC	Microbiologically Induced Corrosion
NCR	NonConformance Condition Report
NEP	Nuclear Engineering Procedure
NQAM	Nuclear Quality Assurance Manual
NSSS	Nuclear Steam Supply System
OC	Office of Construction

OHG	TVA's Occupational Hygiene Group
OSLA	Operations Section Letter
OSP	Office of Special Projects
QA	Quality Assurance
QC	Quality Control
QCI	Quality Control Instruction
QCP	Quality Control Procedure
SCR	Significant Condition Report
SCSA	Station Control and Service Air System
SDO	Site Director's Office
SER	Significant Event Report
SME	Singleton Materials Engineering Laboratory
SNM	Special Nuclear Material
SOER	Significant Operating Experience Report
SOI	System Operating Instruction
SP	Special Program
SWEC	Stone & Webster Engineering Corporation
TI	Temporary Instruction
URI	Unresolved Item
VIO	Violation
WBEP	Watts Bar Engineering Project
WBN	Watts Bar Nuclear Plant
WBP	Watts Bar Procedure
WBPT	Watts Bar Program Team