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

Report Nos.: 50-390/93-01 and 50-391/93-01

Licensee: Tennessee Valley Authority
6N 38A Lookout 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 2 through January 29, 1993

Inspectors: _____ 2/19/93
G. A. Walton, Senior Resident Inspector Date Signed
Construction

Consultants: R. Compton, Nuclear Power Consultants, Inc.
(Paragraphs 2.a - 2.i)
W. Marini, Pegasus, Inc. (Paragraphs 2.j - 2.gg)
D. Myers, Beckman and Associates (Paragraphs 2.hh - 2.oo)
J. Taylor, HCA, Inc. (Paragraphs 2.pp - 2.vv)

Approved by: _____ 2/20/93
P. E. Fredrickson, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine inspection was conducted by NRC consultants in the areas of previous inspection findings, Bulletins, Information Notices, and 50.55(e) reports (Construction Deficiency Reports).

Results:

This inspection for the closure of open items found the licensee had adequately resolved the issues for most of the packages reviewed. The quality and detail of the packages were acceptable with very minor errors found by the inspectors. Some packages reviewed still required further work or review by the licensee before closure occurs. The items left open are discussed regarding actions needed to resolve and close the issue. The closed items are discussed regarding actions taken by the licensee to resolve the issue and details of inspections done by the inspector to assure the item was resolved.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

T. Arney, Senior Quality Project Manager
*K. Boyd, Site Licensing Program Administrator
M. Bellamy, Startup Manager
J. Chardos, Manager of Projects
J. Christensen, Site Quality Manager
S. Crowe, Site Quality Assurance Manager
J. Cruise, Licensing Engineer
W. Elliott, Engineering Manager, Nuclear Engineering
N. Kazanas, Vice President Completion Assurance
A. McLemore, Modifications Engineering Manager
L. Maillet, Site Support Manager
D. Moody, Plant Manager
W. Museler, Site Vice President
C. Nelson, Maintenance Support Superintendent
*P. Pace, Compliance Licensing Supervisor
*G. Pannell, Site Licensing Manager
R. Purcell, Plant Program Manager
K. Stinson, TVA Project Manager
S. Tanner, Special Projects Manager
J. Vorees, Regulatory Licensing Manager
C. Whitehead, Project Engineer

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

NRC Consultants

R. Compton, Nuclear Power Consultants, Inc. (Paragraphs 2.a - 2.i)
*W. Marini, Pegasus, Inc. (Paragraphs 2.j - 2.gg)
*D. Myers, Beckman and Associates (Paragraphs 2.hh - 2.oo)
*J. Taylor, HCA, Inc. (Paragraphs 2.pp - 2.vv)

NRC Employees

*G. Walton, Senior Resident Inspector
*K. Ivey, Resident Inspector

*Attended exit interview

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

2. Actions on Previous Inspection Findings (92701)

a. (Open) IEB 80-11, Masonry Wall Design

As a result of the licensee's re-evaluation of various civil issues, including masonry walls, the NRC reopened IEB 80-11 in IR 390, 391/90-20 pending resolution of three issues by TVA: updated seismic loads; undocumented attachments; and non-qualified anchoring devices. Deficiencies and concerns related to the design of masonry walls and the control of attachments have been identified numerous times at WBN with corrective actions less than fully effective. TVA documented improper attachments to masonry walls on an NCR in 1983. The walls involved were subsequently evaluated and modifications made as required, and typical drawings were revised to prohibit attachments to masonry walls unless design engineering approved it. In 1987, TVA again found items attached to masonry walls with anchorages that had not been authorized (CAQR WBP870493). In 1988, the S&L vertical slice review identified that conduit supports were attached to concrete partition walls in violation of drawing requirements. CAQR WBP880766 was issued to address this deficiency. During a broad-based construction assessment conducted by the NRC in October 1989 (IR 390/89-200), additional unauthorized attachments to masonry walls were again identified (Open Item 390/89-200-29, Ineffective Corrective Action for Controlling Attachments to Masonry Walls). CAQR WBP880766 was revised to also address this issue. In response to this issue, a construction stop work order was issued on November 3, 1989, to prevent any additional field routed conduit and instrument lines being made to masonry walls without engineering approval. The licensee's general course of action to address masonry walls was detailed in Calculation WCG-1-1419, WBN Plant Seismic/Civil Validation Program Methodology Summary Report, Revision 0. Specific corrective actions taken are detailed on various DCNs, CAQRs, and engineering calculations. The inspector reviewed calculation WCG-1-1419 and considered the licensee's approach to be appropriate. The following is a discussion and assessment of the licensee's final corrective actions for each of the three elements related to IEB 80-11:

- Updated Seismic Loads

  In January 1991, all masonry walls were walked down to identify existing conditions (WCG-1-623) and the worst case walls were then analyzed using the latest seismic spectra (calculation WCG-1-767). In April and September 1991, NRR performed special calculation audits relating to the civil calculation program, including masonry walls and calculation WCG-1-767. The NRR team found the TVA calculations to be adequate and closed all open items related to the masonry wall calculations (NRR audit report dated January 31, 1992).

- Undocumented Attachments

  Corrective actions to address this issue include the walkdown, documentation, and evaluation/analysis of attachments to masonry walls; placement of signs on all masonry walls stating clearly that design engineering approval is required for all attachments; revision of design specifications and implementing construction and engineering work instructions to more clearly reflect the requirement for engineering approval of attachments to masonry walls; and conducting training to revised instructions. Many of these corrective actions have been detailed in SCARs WBP880766 (Revision 4) and WBP870493 (Revision 6) and DCNs M-15585 and M-16188.

The inspector reviewed portions of calculation WCG-1-623 detailing the walkdown scope and results. Of the total 85 masonry walls, 19 had been selected and analyzed by TVA as worst case conditions. The inspector also field verified the as-built documentation in this calculation for the following 11 masonry walls including 2 "worst case" selections:

- Control Building: 46W405-1, walls C4Q (worst case), and B5

- AUX Building: 46W405-4, walls 3A, 3B, 3C, and 3D

- AUX Building: 41N366-1, walls A, B (worst case), C, and D

- AUX Building: 41N310-1, wall J

No attachments were identified that had not been documented in the calculation as-built information. Caution signs were in place on both sides of 7 of the 11 walls inspected. The signs for the other 4 walls (46W405-4, 3A, B, C and D) had not been installed or had fallen off; adhesive backed signs installed in 1990 per MR 484638 are being replaced with permanently painted signs per DCN M-15585-A. The workplan for painting the new signs for the auxiliary building per this DCN (WP-D15585-03) had not been completed at the time of this inspection.

The inspector verified that General Design Criteria WB-DC-20-21.1, Category I Cable Tray Supports (Revision 7, February 21, 1992), WB-DC-40-31.8, Seismically Qualifying Round and Rectangular Duct (Revision 7, February 26, 1990), and WB-DC-40-31.9, Criteria for Design of Piping Supports and Supplemental Steel in Category I Structures (Revision 15, August 6, 1992) all stated that attachments are not allowed on masonry walls without formal interfacing with the NE-CE responsible engineering staff. The inspector also reviewed revised Engineering Specifications N3E-934 for Instrument and Instrument Line Installation and Inspection (Revision 2, August 21, 1992), N3E-944 for Conduit and Conduit Support Installation (Revision 0, March 29, 1991), and new

Engineering Specification N3C-946 for Attachments to Civil Features (Revision 2, November 16, 1992) which all clearly address the requirement that NE-CE formal approval is required for attachments to masonry walls. The following implementing field construction documents, MAIs, were also reviewed by the inspector to verify that the requirements for pre-installation design approval for attachments to masonry walls were addressed either directly or by reference to other controlling installation instructions:

- MAI 3.1, Installation of Electrical Conduit Systems and Conduit Boxes (Revision 6, November 16, 1992)
- MAI 3.9, Installation of Cable Tray, Cable Tray Supports and Cable Tray Covers (Revision 4, November 16, 1992)
- MAI 4.2A, Piping/Tubing Supports (Revision 5, November 16, 1992)
- MAI 4.3, HVAC Duct Systems (Revision 4, November 16, 1992)
- MAI 4.4A, Instrument Line Installation (Revision 3, November 16, 1992)
- MAI 4.4B, Instrument and Instrument Panel Installation (Revision 2, December 11, 1991)
- MAI 5.1A (Revision 2), B (Revision 6), C (Revision 5), and D (Revision 2), Various Anchor Bolt Installation Instructions
- MAI 5.9, Fabrication and Installation of Structural and Miscellaneous Steel (Revision 4, November 16, 1992)

The inspector also reviewed lesson plans and instructor notes for construction crafts installing concrete anchors (course CCM302, lesson plan CCM302.001) and for electrical raceway and supports (course CCE342, lesson plan 342.001) which indicate specific instructions are being provided to construction crafts related to the requirement for specific design approval for installations on masonry walls.

The inspector also reviewed EAI-8.07, Documentation and Evaluation for Attachments to Civil Features (Revision 1, November 1, 1992), which details the review and approval process to be used by Engineering for making attachments to civil features, including masonry walls. EAI-8.07 describes masonry walls as "critical features" to which attachments should be avoided where possible. The inspector also reviewed attendance rosters for Engineering staff training to EAI-8.07.

Pending completion of the painting of caution signs on all masonry walls, the inspector considers that the actions taken by the

licensee to prevent recurrence of the issue of undocumented attachments to masonry walls are appropriate and adequate.

Concerning non-qualifying anchoring devices, corrective actions to address this issue included the field walkdown of Calculation WCG-1-623 which documented all unauthorized installations, including expansion anchors and toggle bolts in un-reinforced masonry walls, and DCN M-16188-A which provides for replacement of all unauthorized installations. Twenty of the 85 masonry walls were identified with unauthorized anchorage installations. Twenty workplans are being written for rework, but corrective actions have not yet been implemented. The inspector considers the action taken and proposed actions to be appropriate to resolve the issue of unauthorized attachments to masonry walls at WBN.

Pending TVA completion of corrective actions detailed on DCNs M-15585-A (painting of caution signs) and M-16188-A (replacement of toggle bolt installations) and further NRC inspection of these activities. This item will remain open.

b. (Closed) URI 390/84-76-01, Missing Pipe Support Calculations

This item was initiated to track the regeneration of calculations for many EDS Nuclear Corporation pipe support designs (that had previously been destroyed) to comply with ANSI N45.2.9, Appendix A. TVA issued SCR WBNCEB8531 and 10 CFR 50.55(e) report WBRD-50-390/86-02 to address this issue by determining the existence and acceptability of the design calculations for these supports and regenerating calculations if missing or inadequate. WBNCEB8531, was superseded by CAQRs WBP890209, WBP890216, WBP890217, WBP890218, and WBP890219 which identified specific piping calculations for the various systems. These CAQRs implemented part of the corrective actions of the HAAUP CAP and included the calculation regeneration actions of the original SCR. TVA also committed to full regeneration of the missing EDS calculations in an October 1991 status update for IEB 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts.

Closure reports have been issued by TVA contractor engineering firms for the HAAUP Large Bore Program (Bechtel, December 14, 1990, RIMS B26901217547) and Small Bore Program (Ebasco, June 30, 1992, RIMS B18920630773) indicating that all calculations, including the original EDS designs, have been completed. Therefore, although the CAQRs which address the missing EDS calculations have not yet been formally closed by TVA (they include other corrective action items, including analysis of Category I(L) piping and supports which is not yet complete), the EDS calculations have been regenerated and closure report packages issued. The inspector reviewed the above referenced documentation and concluded that TVA has taken adequate corrective action to resolve this item.

The regeneration of missing EDS pipe support calculations is a specific action item for closure of currently open items CDR 86-02 and IEB 79-02 and is within the scope of the HAAUP CAP which is also an open item requiring NRC review prior to closure. Therefore, the adequacy and completeness of the calculation regeneration effort will also be evaluated by NRC during future inspections to close these issues. This item is closed.

c. (Closed) Part 21 390/86-02, During F.W. Transient 3, the SOR Differential Pressure Switches Failed to Function to Cause a Scram

This Part 21 report describing an event at LaSalle was translated into IN 86-47, Erratic Behavior of Static "O"-Ring Differential Pressure Switches; and IEB 86-02, Static "O" Ring Differential Pressure Switches. IEB 86-02 was evaluated by TVA for WBN as detailed in a letter to NRC from R. L. Gridley dated November 20, 1986. TVA's response to IEB 86-02 was evaluated by the NRC and found acceptable. IEB 86-02 was closed in NRC IR 390, 391/90-24. This item is closed.

d. (Closed) IFI 390, 391/86-02-07, Further Review of QMS Activities

This issue involves NRC inspector concerns related to the audits conducted by the Office of Engineering Quality Management Staff, specifically the lack of correlation of audit checklist items to 10 CFR 50 Appendix B criteria, lack of specificity regarding actual audit activities in audit checklists, and the failure of auditors to utilize issued CAQR reports and trending information in development of audit plans. Since 1986, the licensee's QA program has undergone significant change in both organization and process. The auditing of the engineering function at WBN is now the responsibility of the site quality manager, through the site quality audit & verification manager. Technical assistance or actual performance of site scheduled audits is provided by the corporate nuclear assurance manager through the quality programs manager. Audit program performance is currently delineated in the TVA Nuclear Quality Assurance Plan TVA-NQA-PLN89-A, as implemented through Nuclear Power Standard 3.1, Quality Assurance Program, Revision 1; WBN SSP-3.01, Quality Assurance Program, Revision 6; and five QMPs.

The inspector reviewed the above listed program, standard, and procedures and considers they adequately address the identified concerns from a programmatic perspective. The inspector reviewed the current Watts Bar Triennial Audit Schedule (1992-1994) and selected the following two completed audits of the engineering function for review:

- WBN NQA DBVP Audit WBA92208, issued in May 1992, conducted by TVA corporate QA.

- WBN Nuclear Quality Assurance and Verification Audit

WBA92214, Civil Support CAP, issued in November 1992, conducted by the WBN site QA organization.

These audits reflect that, although 10 CFR 50 Appendix B criteria are not uniquely specified, the performance objectives in the audit plan and scope sections of the audit report clearly correlate with the details in the audit reports and the audit checklists. In addition, the audit checklists and action item forms used to formally document auditor questions and concerns during the audit provide specific details concerning the scope of the items reviewed or inspected and the results. Previously issued CAQRs and identified pertinent problems were specifically noted in the audit plans to be addressed by the audit. This item is closed.

e. (Closed) VIO 390/86-22-02, Missing Calculations for Pipe Supports

This issue involved the licensee's failure to ensure that conditions adverse to quality were promptly identified and corrected. The technical issue involved was the lack of pipe support calculations for EDS Nuclear, Inc., designed Class 1 and 2 supports. The missing EDS calculations issue started in 1982 and has been addressed by numerous NCRs, CAQRs, and CDRs; URI 390/84-76-01 and VIO 390/86-22-02; TVA's response to IEB 79-02; and the HAAUP CAP. NRC evaluation of the acceptability of TVA's corrective action for the missing calculations issue will be addressed by future inspections required to close these open items. TVA's QA program has been the subject of many NRC inspections and has undergone significant change since this violation was issued. VIO 390, 391/90-27-01 identified improper identification and correction of adverse conditions. NRC IR 390, 391/90-31 detailed a special inspection of the Watts Bar corrective action program which resulted in VIO 390, 391/90-31-01 for failing to establish and implement an adequate corrective action program. This violation cited examples of failures to take timely action to determine the scope and significance of identified adverse conditions, inadequate criteria for CAQR initiation, inadequate corrective actions, and inadequate closures. The QA program issue identified in VIO 390/86-22-02 is therefore addressed fully in VIO 390, 391/90-31-01 which remains an open item requiring further NRC inspection and evaluation of the licensee's corrective actions. TVA issued CDR 390, 391/91-23 to report the issues raised in these NRC inspection reports. NRC IR 390, 391/91-29 documented a team inspection which addressed many QA program issues and determined that TVA's CAQ program provided adequate measures to identify, document, evaluate, and correct nonconforming conditions. CDR 390, 391/91-23 and VIO 390/90-27-01 were closed.

The NRC has determined the licensee's current QA program is adequately addressing the issue identified in this violation. Both the technical issue (missing calculations) and the QA

programmatic issue will receive further NRC inspection prior to closure of other open items (HAAUP, IEB 79-02, and VIO 390, 391/90-31-01). This item is closed.

f. (Closed) URI 390, 391/87-07-02, Missing Calculations

This item was originally identified during an NRC inspection of safety-related HVAC systems, documented in IR 390, 391/87-07. The licensee had not been able to produce the design calculations for two duct supports (0031-DW930-044-1092 and 0031-DW930-044-1093) that were attached to masonry block walls. The inspector verified that calculation WCG-1-623, which detailed a December 1990 walkdown of all WBN masonry walls and documentation of attachments thereto, had identified these supports (Drawing 46W405-1, Wall C4Q). The inspector verified that this wall, including the loads from these HVAC supports, had been analyzed in Calculation WCG-1-767 as one of 19 "worst case" evaluations of masonry walls and was determined to be acceptable for the existing loading conditions. This calculation, and the methodology for determining the acceptability of masonry walls at WBN, was reviewed by a team from NRR in April and September 1991, and determined to be adequate (NRR Civil Calculation Audit Report issued January 31, 1992). Calculations for the design adequacy of HVAC supports, as opposed to the walls, are ongoing under the HVAC CAP which remains an open item that will be evaluated by the NRC in a future inspection. This item is closed.

g. (Closed) IFI 390, 391/90-19-04, Verification of Consistency for CAQ Requirements in ACPs

This item was written because the procedures implementing various Administrative Control Programs identified in a new WBN corrective action program procedure, AI-2.8.15, had not been revised to reference the new procedure. The TVA and WBN corrective action program has been revised since this item was initiated. The current program is implemented at Watts Bar through SSP-3.04, Corrective Action Program, Revision 7, effective January 7, 1993. The current version of SSP-3.04 references 11 ACPs. The inspector reviewed the following governing implementation procedures for these 11 ACPs and verified that SSP-3.04 was correctly referenced:

- Maintenance Management System, SSP-6.02, Revision 8, effective January 19, 1993

- Drawing Deviation Program, SSP-2.11, Revision 2, effective January 23, 1993

- QA Program/Receipt Inspection, SSP-3.01, Revision 4, effective May 9, 1992

- Incident Investigation and Root Cause Analysis, SSP-12.09, Revision 6, effective June 29, 1992

- Radiological Awareness Report, SSP-5.05, Revision 3, effective June 5, 1992

- Reporting Safeguards Events, SSP-11.02, Revision 2, effective December 6, 1991

- Corrected on the Spot, QMP-102.14, Revision 3, effective August 14, 1992

- Finding Identification Report, SSP-3.07, Revision 8, effective January 7, 1993

- Preventive Maintenance for Non-Transferred Features, CAI-1.02, Revision 8, effective October 29, 1992

- Problem Evaluation Report, SSP-3.06, Revision 9, effective November 30, 1992

- Preoperational Test Deficiencies, SMP-9.0, Revision 7, effective January 4, 1993

This item is closed.

h. (Closed) IFI 390, 391/90-27-16, U4.3-7, Cable Tray Support Base Plate Analysis

This inspector follow-up item was written to track one of the generic issues (U4.3-7) identified during an NRC design inspection at Sequoyah Nuclear Plant in 1986 (IR 327, 328/86-27). This issue, related to potential improper design analysis of cable tray support baseplates using rigid analysis without considering baseplate flexibility, is being addressed at WBN in SCR WBNCEB8623SCA. This issue had been a part of CDR WBRD-50-390/86-39 and WBRD-50-391/86-37. The corrective action specified in both the SCR and the CDR, to review the design of cable tray supports to determine if baseplate flexibility was adequately addressed and correct any deficiencies identified, is still in progress. Because the issues are identical, and the CDR is being tracked separately by the NRC and remains an open item. This item is closed.

i. (Closed) IFI 390, 391/91-18-01, Inconsistent Information in Response to Violations

This issue involved the licensee's response to VIO 390, 391/90-33-01 in which TVA referenced a procedure change as part of the corrective action that was superseded approximately one month prior to transmittal of the response to the NRC. Although the superseding procedure change did not materially affect the

specified corrective actions, the fact that outdated references were made in formal correspondence to the NRC was of concern. TVA issued LAI-1.02, NRC Correspondence, Revision 0, dated October 19, 1991, to more formally detail processing of NRC-related correspondence and stress the importance of, and responsibilities for, completeness and accuracy in responses to the NRC. A memorandum was issued by the compliance licensing supervisor to the compliance licensing staff specifically addressing the details of this IFI and almost all of the WBN Site Licensing staff have received formal training regarding LAI-1.02. The inspector reviewed LAI-1.02, the memorandum to the compliance licensing staff, and the training attendance records for the site licensing staff regarding LAI-1.02. All of the approximately current technical personnel in compliance licensing (responsible for responses to NRC violations) and most of the technical personnel (including managers) in regulatory licensing, and special projects and Unit 2 groups, as well as the manager of the operating experience group, have completed training. The inspector considers the licensee's actions to be appropriate to prevent recurrence. This item is closed.

j. (Open) CDR 390/85-45, 391/85-44, Unanalyzed Diesel Generator Transients for a Blackout Followed by a Safety Injection Signal

This 10 CFR 50.55e report pertains to the identification of a scenario which could have led to an unanalyzed condition pertaining to DG sequential loading. The identified scenario is that of a LOOP with a subsequent safety injection signal. Two conditions led to the possibility of the unanalyzed condition. These are:

- The control circuits for the 480V shutdown board room chillers were modified to prevent their connection until 45 seconds into the loading sequence, via ECN 4480. Per design criteria WB-DC-30-1, Emergency Auxiliary AC Power System, and FSAR Section 8.3.1.1, if the LOOP sequence is being loaded on to the DG and a safety injection signal occurs, those loads not yet connected are to have their sequential timers reset and then be sequentially loaded. This requirement was not incorporated into ECN 4480.

- The above chillers, along with the control room and electric board room air conditioner compressors, are process controlled and completion of their DG sequence timing interval only enables their operation after 45 and 60 second delays, respectively. They may not actually start at those times unless their operation is needed, but at some later time. Similarly, this condition was found to exist for all process-only controlled loads powered by the DGs. They may also start at any time later than enabled by DG sequencing.

To address the worst case loading scenario of LOOP concurrent with safety injection, the licensee performed analyses to demonstrate the DG's ability to handle the worst case load combination as well as meet the frequency requirements of RG 1.9. The results of the analyses are contained in calculations EEB-MS-TI03-0012, Diesel Generator Loading Analysis, and EEB-MS-TI06-0013, Diesel Generator Voltage Analysis Profile. Revision 0 of these calculations was issued on February 2, 1990, and January 31, 1992, respectively. In Amendment 67, FSAR Table 8.3.3 was revised to reflect sequential loading times shown in these calculations.

The inspector reviewed the above calculations to determine whether the identified condition had been adequately addressed and whether the conclusions reached match current FSAR commitments. The current revisions of the above calculations are 13 and 4, respectively, and the current amendment level of FSAR Table 8.3.3 is 72. The inspector's review determined that the assumptions and methodology employed are adequate to support the conclusion that the stated DG loading sequences will not overload the capability of the DGs. However, this review also revealed that the sequencing times currently shown in the calculations do not match those stated in the current FSAR table. The calculations state that the sequencing times shown are to be implemented per DCNs M-11943-A and M-16440-A. Additional information provided by the licensee indicates that these revised times will be submitted as an FSAR amendment in the near future. The inspector reviewed FSAR change request package No. 0832 and determined that the proposed timing changes match the values shown in the latest calculations. This item will remain open pending formal submittal of the FSAR amendment package.

k. (Closed) CDR 390/86-16, Extreme Wear Shown on Westinghouse Switchgear Breakers

This deficiency was originally identified by the licensee in December 1985. The motor cutoff switch levers were found to be exhibiting extreme wear which could lead to failure of the levers. Such a failure would render the automatic function of the breakers inoperable. The licensee determined the causes of this excessive wear to be: (1) insufficient lubrication due to underestimation of the frequency of breaker operation, and (2) use of incorrect lubricant.

To correct the identified deficiencies and prevent future recurrence, the following actions have been taken:

- Following inspection of Westinghouse types DS-632, DS-416, and DS-206 breakers, the breaker cutoff switch levers for 29 breakers were replaced in April 1986 in accordance with MR A-581006. The inspector determined that, as these replacements were installed over six years ago and are physically identical to the original hardware, a physical

inspection to determine whether replacements were installed where designated would be of little value. However, the inspector reviewed the completed MR and determined that it adequately documents the lever replacement.

- Procedure MI-57.2, Annual 480 Volt Circuit Breaker Inspection, was revised to change breaker lubricant from molybdenum disulfide to Moly Kote BR2 Plus, in accordance with Westinghouse recommendations. This procedure revision also instituted a requirement to lubricate DS-632 breakers every 250 operations, DS-416 breakers every 500 operations, and DS-206 breakers every 1750 operations. The inspector reviewed the current revision of this procedure (revision 18) and determined that the above added requirements are clearly and appropriately included.

- Many of the vendor supplied breakers were originally fitted with trip counters; however, many were not. Therefore, the licensee installed trip counters on the remaining safety related breakers in accordance with DCN M-02390-B and Workplan M02390A-1 so the required maintenance and lubrication can be accomplished prior to exceeding the above stated number of breaker operations. The inspector selected a sample of breakers on which the trip counters had been added and physically inspected the installations in accordance with the installation requirements contained in the above DCN and workplan. All installations were determined to be acceptable. In addition, the inspector noted the current trip counts shown on the counters and compared them to maintenance records that document the last maintenance performed on the breakers. The results of this comparison are that no breakers within the selected sample had been operated more than 71 times since the last maintenance and lubrication had been performed. The selected sample was comprised of breakers installed in the following locations: 480V Shutdown Board 1A2-A, Cubicles 4B, 9B, 9C, & 12D; 480V Shutdown Board 1B2-B, Cubicles 2B, 3B, 3C, 3D, 7B, 8B, 9B, & 9D; 480V Shutdown Board 2B2-B, Cubicles 2B, 2D, 3B, 4B, 8B, 9C, & 10B; Reactor Trip Switchgear 1-PNL-99-1-L-115, Cubicles 1C & 2C, and Reactor Trip Switchgear 1-PNL-99-1-L-116, Cubicles 1B, 1C, 2B, & 2C.

As discussed above, the inspector reviewed the licensee's actions pertaining to this issue and determined them to be acceptable. This item is closed.

1. (Closed) VIO 390/86-21-01, Failure To Meet Post-Modification Testing Requirements

This violation pertains to the licensee's failure to adequately perform and/or document the performance of post-modification testing prior to October 1986. This item has been previously

reviewed and determined to have adequate recurrence controls to allow construction restart in IR 390/91-23.

By letter of January 21, 1992, the licensee submitted Amendment 69 to the WBN FSAR. This amendment revised Chapter 14, Initial Test Program, to reflect TVA's commitment to re-perform a RG 1.68 preoperational test program in support of licensing WBN Unit 1. This commitment was restated in a letter to NRC dated February 13, 1992, which also withdrew the Prestart Test CAP.

The total re-performance of Unit 1 prestart testing will be monitored by NRC and documented in future inspection reports. Therefore, this item is closed.

m. (Closed) IEB 88-01, Defects In Westinghouse Circuit Breakers.

This issue is identical to CDR 390, 391/91-07, which is discussed in detail and closed in paragraph 2.bb of this report. Therefore, this item is also closed.

n. (Open) CDR 390, 391/88-01, Auxiliary Control Air Compressor Control Circuits Must Be Manually Reset After Loss of Off-Site Power

As discussed in IR 390, 391/90-30 and IR 390, 391/91-15, the originally identified concern, as well as recurrence controls, were reviewed and considered acceptable. However, two questions regarding ACA system loading and valve stroke times remained to be resolved.

The licensee has since supplied the results of a comparison between the loads listed in system description N3-32-4002, "System Design Description for Compressed Air System," and those listed in FSAR Section 9.3, Table 9.3-8. Although the completed comparison showed the system description document and the FSAR to be essentially in agreement, the comparison requested in IR 390, 391/90-30 was between the FSAR Table and the Design Calculation, not the system description. Therefore, the closure package for this item has been returned to the licensee, and this item will remain open pending resolution of the two questions regarding ACA system loading and valve stroke times.

o. (Closed) IN 88-31, Steam Generator Tube Rupture Analysis Deficiency

This notice identified a possible increase in radioactivity release to the environment following a steam generator tube rupture. The licensee issued CAQR WBP880114 in response to Westinghouse letter dated December 30, 1987 (RIMS B26880104955). This CAQR was later rolled over into SCAR WBP880114SCA.

The Westinghouse Owners Group has analyzed the potential

deficiency and has issued generic report WCAP-13132, which concluded that the identified concern is not a problem for Westinghouse-designed plants and does not affect the design basis of the plant. The above report was submitted to NRR-Reactor Systems Branch, Division of Systems Technology, via letter OG-92-25, dated March 31, 1992.

Any future required actions resulting from NRR review of this issue will be addressed generically in accordance with the appropriate guidance provided by NRR. Therefore, this item is closed.

p. (Closed) URI 390/89-02-03, Calculation of All Loads on Concrete Slabs

During an NRC review of structural engineering discrepancy reports, two concerns were identified pertaining to DR-104.

- The DR reported that the seismic loads on the component cooling water heat exchanger had not been considered in the design of the floor slab below. The licensee's proposed corrective action was to review all slab calculations where heat exchangers are located. NRC staff indicated that this review should encompass slabs supporting all equipment with a weight of 2 kips or more.

- The DR reported that a later revision of the ACI-318 code than specified in the FSAR had been used in the design of the floor slabs in the area under review and that no calculations had been performed for loading cases which included seismic loads.

In response to the above concerns, the licensee developed and implemented a worst case evaluation methodology for all Category I reinforced concrete slabs. This methodology was originally presented to the NRC staff in "Seismic/Civil Validation Program Methodology Summary Report," Revision 2, on September 24, 1991. NRC staff review and acceptance of the licensee's actions was accomplished as follows:

- Between September 9-13, 1991, the staff performed "Special Calculation Audit Relating to the Civil Calculation Program and Implementation." The results of this audit are contained in Audit Report dated January 31, 1992, Section 2.1.3, "Review of Worst Case Slab Evaluations." Three unresolved concerns were identified during this audit and are identified in the report as AU-1, AU-5, and AU-10.

- Subsequent to the review of additional information provided by the licensee, the staff issued Supplemental Audit Report

dated May 26, 1992. This report, in Section 2.1, closed items AU-1 and AU-5, but left item AU-10 open pending additional review.

- In July and August of 1992 the staff conducted an IDI at WBN. The results of this IDI are contained in IR 390, 391/92-201. Section 7.2.3 of this report states that item AU-10 was reviewed and closed.

Therefore, as all of the staff's concerns pertaining to concrete slab design have been resolved in the above referenced audit/inspection reports, this item is closed.

q. (Closed) IFI 390, 391/90-03-01, Procedure Adherence "Shall/Should"

This item requested clarification by the licensee on how conflicting instructions in different procedures were being resolved and what governing instructions covered the use of the words "shall, should, may."

Procedure SSP-2.03, Administration of Site Procedures, Revision 9, addresses conflicting steps in Section 2.1 in the following steps:

"H. The activity shall be stopped if work is proceeding in violation of approved and controlling documents," and

"I. If a step cannot be performed, is not necessary, or is not required, determination and implementation of the correct disposition of the step is required."

This procedure also controls the use of "shall, should, may" in Section 2.1. The definition of these words is consistent with the industry-accepted definitions contained in ANSI-N18.7-1976.

The inspector reviewed the above licensee-supplied information and determined that adequate instructions are present to clarify resolution of conflicting instructions and control the use of "shall, should, may." This item is closed.

r. (Closed) Part 21 90-05, Swing Arm In Borg-Warner Check Valve Found Broken.

This item involves the discovery of a broken swing arm in a 4-inch 150 pound Borg-Warner bolted bonnet swing check valve installed in the service water system at CPSES in May 1989. The swing arm's collar had fractured radially in two places. Six other swing arms were then removed from service, inspected, and tested. It was concluded that the failure of the swing arm was caused by surface defects formed during the fabrication process. Other contributing factors cited were inadequate heat treatment, weld repairs, high residual stresses, and the high chloride environment of the service water system. The swing arms in question were stainless

steel castings produced from a 17-4PH alloy in accordance with Aerospace Materials Specification 5398A.

The licensee has performed a review of all safety related swing check valves at WBN. This review revealed that only two systems contain the subject B-W valves. These systems are the FW and HPFP systems. The FW valves are 6-inch valves supplied on contract 822446 with B-W drawing 455KAB1-001 (valve numbers 1-CKV-03A-638, -644, -645, -652, -655, -656, -669, -670, -678, and -679). The HPFP valves are 4-inch valves supplied on contract 822598-5 with B-W drawing 421JBB1-002 (valve numbers 1-CKV-26-1260, -1296 and 2-CKV-26-1260, -1296). Review of the above drawings revealed that the valves supplied to WBN contain machined swing arms instead of the cast swing arms which failed at CPSES. Therefore, the licensee does not anticipate failures similar to the type identified. In addition, the licensee has revised Material Specification PF-2035, Precipitation Hardening Stainless Steel Castings, to require NDE in accordance with ASME Section III, NC-2570 and Table NC2571-1, in the event that similar material is purchased for WBN in the future.

The inspector reviewed the licensee's actions discussed above and determined that they are adequate to preclude failures of Borg-Warner swing check valves similar to those experienced at CPSES. In addition, the inspector attempted to locate one of the subject valves in order to perform a visual inspection to verify that the swing arms are machined instead of cast material. However, all valves of this type currently on site are installed in piping systems presently full of water (i.e., there are currently no spares of this type in the warehouse), making such visual inspection impractical. Therefore, this item is closed.

s. (Closed) VIO 390/90-19-01, Failure To Implement Administrative Procedures

This violation is comprised of two examples. The second example pertained to processing of conditions adverse to quality and has been previously reviewed and closed in IR 390, 391/91-29. The first example pertains to the failure to cover the suction piping to AFW during work activities associated with the sandblasting and repainting of the CST. This work was performed under WP K-04335A, which failed to incorporate the necessary administrative controls to assure that sand from the blasting process would not enter AFW system piping.

To correct the identified deficiency, the following actions have been completed:

- The piping openings were sealed to prevent further contamination. This action is documented in the above referenced workplan, step 19.

- The workplan was revised to include actions necessary to determine the extent of contamination and provide for cleaning the lines upon completion of work. These requirements are contained in steps 32 through 60 of the workplan.

As stated in IR 390, 391/91-29, recurrence control pertaining to this issue has been reviewed and found acceptable. The only item remaining to be resolved is the review and verification of the completed workplan. The inspector reviewed the completed workplan and verified that QC acceptance signatures are present to document the cleanliness of the affected lines upon completion of work. During the inspection period the CST was filled with water to support the upcoming secondary system hydrostatic test. Therefore physical verification of internal tank cleanliness was not possible. However, recent flushing of the AFW lines in anticipation of the secondary system hydro provided reasonable assurance of current system cleanliness. Therefore, this item is closed.

t. (Open) URI 390, 391/90-20-06, High Pressure Fire Protection - Microbiologically Induced Corrosion

This item involves the discovery that HPFP piping was exhibiting the effects of MIC. This discovery was made during the repair of HPFP piping damaged during previous work activities. During yard modification work, heavy equipment accidently hit a buried 12" HPFP header running from the intake pumping station to the plant. When the damaged section was removed and visually inspected, many large MIC nodules were noted throughout the section of piping. The licensee was requested to provide information on their evaluation of the extent of MIC damage to the safety related portion of HPFP as well as plans for inclusion of this section of piping into the overall program for monitoring and control of MIC.

Licensee actions to identify and quantify the extent of MIC damage in HPFP are as follows:

- A section of the damaged header containing MIC nodules was split lengthwise and UT tested to determine the extent of wall thickness deterioration. The thinnest remaining wall was found to be 0.245" in a localized area beneath one nodule. The original nominal wall thickness of this piping was 0.375".

- CE performed a structural integrity evaluation to provide acceptance criteria for future UT testing and to determine the acceptability of the as-found conditions. The results of the evaluation are documented in QIRCEBWBN92057, Revision 1. The minimum wall thickness criteria provided are 0.250" for 100 percent around the pipe circumference or 0.160" for not more than 10 percent around the circumference.

- In 1991, a through-wall leak was discovered on a 6" diameter drain line, upstream of valve 0-ISV-26-683. The leak was approximately 1/8" in diameter and was located about 2" below the weld joining the 6" drain line to the 8" HPFP header. Visual inspection, along with a water chemistry test, verified the presence of MIC activity.

- CE again performed an evaluation to determine minimum wall thickness acceptance criteria for UT testing. The resulting criteria are 0.140" (original nominal wall thickness was 0.280") for 100 percent around the pipe circumference, or a total cross sectional area reduction of not more than 38 percent. These criteria are documented in QIRCEBWBN91052, Revision 0.

- The section of pipe containing the leak was then removed and UT testing was performed. The UT data obtained ranged from 0.086" adjacent to the leak site to 0.271" elsewhere on the pipe section. For sections reading less than 0.140" the <38 percent criteria was then applied. The resulting evaluation found that the worst case cross sectional area reduction was 28 percent. It was therefore determined that the piping section in question had maintained its structural integrity.

- A butterfly valve on the 10" HPFP header piping on elevation 713 of the auxiliary building was removed in order to visually inspect the adjacent piping for evidence of MIC. An extensive buildup of sediment and nodules of various sizes were observed.

- Valves were also removed at the interface point between HPFP and AFW to evaluate the condition of the piping supplying AFW. This piping exhibited minimal MIC activity evidenced by very small and scattered nodules.

The licensee has committed to continuously monitor HPFP, as well as other susceptible plant systems, in order to identify, control, and reduce MIC damage. Elements of this program include:

- Internal visual inspections are to be performed by craft during maintenance activities, in accordance with TI-27 Part III, Cleaning and Cleanness of Fluid Systems and Components, Revision 32.

- Semi-annual monitoring of piping systems to track MIC degradation is to be performed in accordance with TI-31.13, Wall Thinning Monitoring Program for Cavitation, Microbiologically Induced Corrosion and Dual Phase Erosion/Corrosion, Revision 8. HPFP system piping grids currently designated as MIC locations for monitoring under TI-31.13 are 1-FP-3, -4, -7, -8, 2-FP-5, -6, 0-FP-9, -10, and -11.

- Chemistry Manual Chapter 6.02, MIC Sampling, Revision 0, provides for as-needed and quarterly sampling of solids and liquids in susceptible systems for evidence of MIC activity.

- QIRCEBWBN92069, Calculation of Inspection Criteria Fire Protection Piping, Revision 0, was issued by CE to provide information on the highest stressed areas of HPFP piping for each applicable pipe size, a minimum wall thickness for each highest stressed node, and the maximum allowable decrease in cross sectional area for each highest stressed node. UT testing of these identified highest stressed nodes is scheduled to be performed in accordance with WR C-123367 and WR C-123368, but has not yet been completed. Areas identified by this testing that encroach on structural integrity criteria will be added to TI-31.13 for future monitoring.

- A chemical treatment program has been instituted to mitigate corrosion, prevent micro and macrofouling, control particulate deposition, and gradually remove existing corrosion tubercles and byproducts. CM Chapters 4.00, Corrosion Control, Revision 0; and 4.02, Startup and Normal Operation of the Pyrophosphate, Zinc, and Copolymer Equipment, Revision 0, describe the chemicals injected into raw water systems, techniques used to monitor the effectiveness of the chemicals, and instructions for operating the chemical injection equipment.

- NE calculation MT-WBN-90-001, Determination of Acceptability Of Using 1-Bromo-3-Chloro-5,5-Dimethylhydantoin As A Biocide In The Watts Bar Raw Water Systems, Revision 1, contains an assessment of a biocide treatment consisting of Betz Slimicide C-78P as a preventative measure for MIC activity in raw water systems; and NE calculation MT-WBN-92-001, Materials Compatibility Study and Chemical Treatment Program Evaluation WBN Raw Water Systems, Revision 0, provides a materials compatibility assessment for the chemical treatment program.

- A program of periodic flushing of HPFP has been instituted. The procedures that cover this flushing are: SI-7.15, High Pressure Fire Protection Hydraulic Performance Verification, Revision 11; SI-7.12, Fire Suppression Water System Flush, Revision 13; and TI-89, Flushing High Pressure Fire Protection System, Revision 0. In addition, the licensee has committed to perform an initial high velocity flush of HPFP piping. This flush is currently scheduled to be completed by May 6, 1993.

In SSER 8, the NRC staff published a safety evaluation of the licensee's MIC Special Program. Subsequently, in a letter dated

May 1, 1992, the licensee submitted additional information, which included the above discussed inclusion of HPFP into the MIC program. The NRC staff evaluated the additional information and concluded that, if the licensee's submitted program is properly implemented, safety related raw water systems at WBN should not lose their capability to perform safety functions due to MIC damage. This conclusion, along with a detailed discussion of the staff's review, is contained in SSER 10, dated October 1992. However, this item will remain open pending future NRC review of the licensee's implementation of the MIC Special Program.

u. (Closed) URI 390/90-20-02, Anchor Bolt Installation Practices

This item involved unacceptable methods used to torque wedge type concrete expansion anchors. The concern was that, during initial installation and setting of the anchors, the requirement to utilize a calibrated torque wrench was ignored and an un-calibrated wrench with a "cheater bar" was used. This practice has been known to occur when the anchor does not extend a sufficient length above the concrete surface to achieve the required one thread projection past the nut once the associated plate/washer/nut configuration is installed, and a cheater bar is used to pull the anchor far enough out of the hole to allow for the required thread projection. However, this practice could result in either overstressing the anchor itself and/or damaging the adjacent concrete due to excessive torque being applied.

As reported in IR 390, 391/91-23, the NRC has previously reviewed licensee actions relative to recurrence controls and procedural clarifications, and that the only actions remaining to complete the necessary corrective actions were (1) reinspection of the anchors which were observed to be torqued using a cheater bar, as well as other potentially overtorqued anchors, and (2) training of craft personnel in anchor installation requirements.

The licensee documented, in the closure of PER WBP900377PER, that all suspect anchors in the area originally identified have been reinspected (including removal of the support base plates to check for concrete damage), re-torqued, and found acceptable; and that training of craft personnel has been accomplished in accordance with lesson plan CCM302.001, Concrete anchors (SDIs, SSDs, and Wedge Bolts). In order to verify the completion of these corrective measures the inspector visually inspected the anchors and adjacent concrete associated with the originally identified support as well as three of the other suspect supports, selected at random. The supports inspected are: 2-CSP-292-N4147, -1620, -1631, and -1644. No evidence of damaged anchors or concrete was observed. In addition, the inspector reviewed the training records for the two craftsmen and the QC inspector who are shown on work Order 08-1640-45 as the personnel who performed and verified the reinspection and re-torquing of support N4147. The training records indicate that all three of these employees were

trained to the above referenced lesson plan. Therefore, as recurrence controls and procedural revisions were previously reviewed and accepted, and as the reinspection/re-torquing of the suspect supports and personnel training have been completed, this item is closed.

v. (Open) IFI 390, 391/90-24-03, Adequacy Of Labeling

This item involves labeling inconsistencies observed on the name tags of several System 82 DG pumps. The pumps in question are one 6 gpm and two 3 gpm lubricating oil pumps. As originally observed, these pumps were labeled either "Soakback Pump" or "Auxiliary Oil Pump."

Information provided by the licensee indicated that the pumps were currently tagged as follows:

- A/C Lub Oil Circ Pump (6 gpm)
- Aux AC Lube Oil Circ Pump (3 gpm, AC)
- Aux DC Lube Oil Pump (3 gpm, DC)

The inspector, after verifying that the pumps are tagged as shown above, requested documentation to substantiate that these are the correct designations for each pump. The information provided was in the form of a computerized printout, of all pumps in system 82, of the Equipment Management System. The inspector found that the designations shown on the EMS printout do not match the designations on the installed nametags. As an example, the pumps for DG 1A-A are designated in EMS as follows:

- Aux AC Lube Oil Circ Pump (6 gpm)
- Dsl Gen AC Aux Lube Oil Circ Pmp Bay 1AA (3 gpm, AC)
- Dsl Gen Aux Lube Oil Circ Pmp Bay 1AA (3 gpm, DC)

(Note: The above parenthetical information is not part of the designated nomenclature. It is included here only to provide correlation between EMS designations and as-tagged data.)

When questioned as to why the EMS designations were not the ones used when tagging the pumps, the licensee stated that a new tagging system was is the process of being instituted, and most plant equipment tags had not yet been upgraded. This new system was outlined in SSP-2.52, Replacement And Upgrade Of Plant Component Identification Tagging/Labeling, Revision 3. This system stated that the Operations Department will be responsible for the nomenclature to be included on equipment tags, providing and hanging the tags, and providing subsequent input so that EMS can be updated to show actual equipment nametag information. However, as the upgraded tagging had not yet been completed on System 82, and the installed tags were not in agreement with EMS information. This item remains open.

w. (Closed) IFI 390, 391/90-27-17, Base Plate Design Criteria

This item was opened to track an issue originally identified during a design inspection at SQN and subsequently determined to also be applicable at WBN. The concern identified was that TVA Pipe Support Design Manual (Revision April 22, 1983) section 7.18.2 stated that support base plates were analyzed as rigid members, while Civil Design Standard DS-C1.7.1 (Revision 3, November 16, 1984) stated that flexible plate analysis would be performed.

To correct the identified deficiency, PSDM was revised (Revision 3, July 23, 1984) to require flexible base plate analysis for all future support designs. During an inspection of the Hanger and Analysis Update Program CAP, the NRC staff reviewed the licensee's approach to verifying that all supports are designed to the appropriate criteria, including base plate flexibility. As reported in IR 390, 391/89-14, Section 5.3.7, the staff found the licensee's approach acceptable and stated that implementation of any resultant support modifications will be reviewed by NRC as part of final acceptance and closure of the HAAUP CAP and IEB 79-02 at a future date. In addition, the staff's acceptance of the licensee's resolution of this issue is also contained in SSER 8, dated January 1992. Therefore, this item is closed.

x. (Closed) IFI 390, 391/90-27-28, ERCW Screen Wash Pump Control

This item was initiated as a result of a design inspection conducted at SQN in 1986 and documented as Observation 06.3-3 in IR 327, 328/86-27, which noted that redundant dP switches used to initiate operation of the ERCW screen wash pumps had been disconnected, because of improper operation, via a TACF which had not been translated into a permanent design change in a timely manner. As the ERCW systems at SQN and WBN are of similar design, this item was opened to determine whether the corresponding dP switches at WBN had exhibited the same improper operation, and if so, has the problem been corrected in accordance with approved procedures.

The inspector interviewed licensee personnel (principal mechanical engineer and ERCW system engineer) who were closely associated with the design and operation of ERCW for several years. These personnel indicated that the dP switches in question have never operated improperly as had their counterparts at SQN, and no TACF was ever initiated to cause them to be disconnected. Therefore, as the concern identified at SQN has not recurred at WBN, no further action is necessary. This item is closed.

y. (Closed) URI 390, 391/91-03-01, RCS Cooldown Due To AFW Design

During SQN Unit 2 restart in 1988, RCS temperature was cooled substantially below the established no-load value after each

reactor trip from power. This was due in part to the auto level control feature of AFW combined with steam dump control set points. The generic operability review documented in the SQN CAQR improperly indicated that the CAQ related only to operating plants and was not applicable to WBN. The system design question is being tracked by this URI, while the inadequate generic applicability review is being tracked by URI 390, 391/91-03-02.

The identified concern is that a reactor trip shortly after startup after a long period of shutdown leaves little decay heat to be removed, thus causing the possibility of excessive cooling of the RCS if too much cold water is being added. The original transient analysis assumed the AFW inlet water temperature to be 120 degrees F. However, in a situation as cited above, the AFW inlet water temperature could be as low as 40 degrees F. This, therefore, constituted an un-analyzed condition that could result in a loss in shutdown margin.

Resolution of the un-analyzed condition at WBN was documented in WBPER910117. The corrective action implemented was to revise System Description N3-3B-4002, Auxiliary Feedwater System, to add a requirement to immediately put AFW on manual control in the event of a reactor trip without safety injection initiation. This will allow the operator to control AFW flow so that excessive cooling to RCS is precluded. This revision to the system was accomplished via DCN S-16879-A, which subsequently was incorporated into the system description in paragraph 4.7. Placing AFW in manual control during such a circumstance is the action recommended in the Westinghouse Owners Group Emergency Response Guidelines. This action is also consistent with Nuclear Performance Plan, Volume 4, Chapter 4, Section 2.3, which commits WBN to follow WOG-ERG for the above described situation.

The inspector reviewed the licensee's actions described above and determined them to be consistent with WOG-ERG and NPP commitments.

This item is closed.

z. (Closed) IFI 390, 391/91-04-01, Test Program for Evaluation of Shallow Undercut Anchors

This item was initiated to track the licensee's commitment to perform testing of shallow undercut anchors. This testing was intended to demonstrate that properly installed shallow undercut anchors have ultimate shear capacities exceeding design requirements. The acceptance criteria for these tests were for failed anchors to exhibit a safety factor of at least 4 if the failure was in the concrete, and a safety factor of at least 2 if the failure was in the anchor itself.

The completed test results demonstrated that all of the configurations tested passed the above acceptance criteria with

the following four exceptions: 1/2" diameter with 3-1/4" embedment; 5/8" diameter with 3-3/4" embedment; 3/4" diameter with 3-1/2" embedment; and 3/4" diameter with 5" embedment. Using these test results, the licensee has established the following minimum embedment length requirements for future installations of shallow undercut anchors: 4-3/4" for 1/2" diameter; 5-1/4" for 5/8" diameter; and 6-1/2" for 3/4" diameter anchors. The inspector has reviewed civil design standard DS-C1.7.1, General Anchorage To Concrete, Revision 5, and found that these minimum embedment requirements were incorporated via DSCN-CEB-92-02. The inspector also reviewed General Engineering Specification G-66, Requirements For The Use Of Undercut Anchors Set In Hardened Concrete During Installation, Modification, and Maintenance, Revision 3, and found that these requirements were incorporated via SRN-G-66-13, dated August 14, 1992.

By letter dated October 16, 1991, the licensee submitted, to the NRC staff, a summary report of the completed testing along with actions taken to verify that the failed configurations either had not been installed at WBN or had been identified and evaluated for acceptability on a case-by-case basis. On December 16, 1992, the NRC staff issued a Safety Evaluation which concluded that, "In view of the reasonable and responsible actions taken by TVA, the staff has concluded that the shallow undercut anchor issue is closed." Therefore, as the staff has accepted the results of the testing program and as the inspector has verified that the appropriate installation requirements have been incorporated into design output documents, this item is closed.

aa. (Closed) IFI 390/91-04-05, Condition Adverse To Quality

This item identified the inappropriate closure of a CAQR in July 1988. CAQR WBP87022 was originally initiated due to a concern over potential discontinuities in permanent plant shielding structures. As a result of plant walkdowns, penetrations were discovered that required further evaluation. The proposed disposition of the CAQR required that NEB propose remedial action for each penetration for consideration by plant management. Although the CAQR was closed on July 30, 1988, there was no evidence that the remedial action was taken. During the original review of this issue, it was discovered that QA had initiated PRD WBP900317P to address the lack of completion of hardware corrections prior to CAQR closure. However, this PRD did not address the improper closure of the CAQR. As the licensee had been tracking the hardware deficiency aspects of the inappropriate CAQR closure, this IFI deals only with the improper closure aspects of the issue.

The licensee issued PER WBPER910232 to address the identified concern. The disposition of this PER is as follows:

- The sequence of events leading to the inappropriate closure

were reviewed. It was determined that since the last required corrective action listed on the CAQR form stated to "propose remedial corrective action for each penetration for consideration by plant management," it was assumed to be acceptable to close the CAQR because the remedial corrective actions had been "proposed." In addition, since the original corrective action was to perform the evaluation and make recommendations and because the element of cost/benefit was involved in determining what, if any, field work would be required, management believed that the prerogative existed to close the CAQR once the corrective actions had been proposed.

- It was determined unnecessary to review the improper CAQR closure for actions to prevent recurrence, in accordance with this PER, as recurrence controls are being addressed by the PRD which had been issued by QA.

The inspector reviewed the licensee's actions taken to address the improper closure of CAQR WBP870252 and determined that they adequately resolve the "improper closure" aspects of the issue, and that the above referenced PRD (which has subsequently been rolled over into PER WBP900317PER) adequately addresses "hardware" and "recurrence control". This item is closed.

bb. (Closed) CDR 390, 391/91-07, Hardware Defects On Class 1E Circuit Breakers

The licensee initiated this 10 CFR 50.55e report in response to NRC Bulletin 88-01, which identified inadequate welds joining levers to pole shafts in Westinghouse DS type circuit breakers.

To correct the identified deficiency the licensee decided to replace the suspect shafts in all DS-206 and DS/DSL-416 Class 1E breakers. This replacement was accomplished in accordance with SMI-212.D, DS-Breaker-Pole Shaft Removal/Installation, Revision 1, dated December 5, 1990.

The inspector reviewed completed SMI-212.D data sheets and determined that they adequately document the replacement of the deficient shafts. In addition, as the pole shafts of installed breakers are inaccessible for visual inspection, the inspector located a type DS-416 breaker (S/N 137) that had been removed for maintenance and compared the now-installed shaft to several shafts which had been removed. This comparison showed that the new shaft-to-lever welds are acceptable and are of a noticeably higher quality than the originally supplied welds. The inspector therefore determined that the licensee has appropriately addressed and corrected the identified deficiency. This item is closed.

cc. (Closed) VIO 390, 391/91-33-01, Failure to Maintain Housekeeping Surveillances

This item was reviewed on the basis of a problem previously identified to determine if it should be re-opened since subsequent inspections have determined that many issues involved more than had been identified in the original concern. This violation was previously closed in IR 390, 391/92-26 but was re-reviewed during this inspection period to verify continued compliance with the monthly housekeeping surveillance requirements contained in SSP-12.07, Housekeeping/Temporary Equipment Control, Revision 4.

The inspector randomly selected five plant areas, as listed in SSP-12.07, Appendix A, Housekeeping Ownership Assignments. The areas selected were:

- Area 2: Unit 1 Reactor Building, Lower Containment
- Area 12: Unit 2 Reactor Building, Accumulator Rooms
- Area 13: Unit 2 Reactor Building, Fan Rooms
- Area 19: Auxiliary Building, El. 737, A1-A15
- Area 20: Auxiliary Building, El. 713, A1-A15

The inspector walked through these areas to determine if the housekeeping and cleanliness requirements of SSP-12.07 were being followed. No unacceptable areas were observed. The inspector then requested to review, for the areas selected, documentation to substantiate the performance of monthly housekeeping inspections, as is required by paragraph 2.1.1.E of SSP-12.07. These inspections are to be documented on SSP-12.07, Appendix C, Housekeeping Inspection. The inspector's review of the completed forms for these areas, for the months of October and November 1992, indicates that the required inspections are being performed. As no unacceptable practices were observed in this area, this item remains closed.

dd. (Open) CDR 390, 391/91-38, Lack Of Documentation For Fire Barrier Material In Seismic Expansion Joints.

This 10 CFR 50.55e report involves the fire barrier material in the seismic expansion joints between the auxiliary building and the reactor building at elevations 692.0, 713.0, 729.0, 737.0, 757.0, and 782.0. It was discovered that documentation could not be located to certify the asphalt covered fiberglass material to an acceptable UL or Factory Mutual standard. Without this documentation, credit cannot be taken for the as-installed configuration as either a qualified fire barrier or a qualified flood barrier.

To correct the identified condition, the following actions were taken by the licensee:

- Walkdowns were performed of accessible expansion joint wall seals to determine as-installed configurations.

- Fire testing was performed to qualify the material as a three hour fire barrier. This testing is documented in calculation SQN-00-D052/EPM-MHS-1123891. Test results confirmed that the material and configuration tested constitute a qualified three hour fire barrier.

- Flood testing was performed and is documented in TVA report WR28-4-900-253. Test results confirmed that the expansion joints are a qualified flood barrier.

- Design Criteria WB-DC-20-8, Auxiliary-Control Building Structures, was revised to more clearly identify requirements for expansion joints. These requirements were incorporated into the design criteria via DCN S-19090-A.

- To ensure that the expansion joints can continue to perform their functions as fire and flood barriers in the future, SI-7.27, Visual Inspection Of Fire-Rated Assemblies Located In Reactor Building, Unit 1, and SI-7.28, Visual Inspection of Fire-Rated Assemblies In Reactor Building, Unit 2, were revised to incorporate periodic expansion joint inspection requirements. These requirements were incorporated into revisions 12 and 5 of these procedures, respectively.

The inspector reviewed the above walkdown reports, test reports, design criteria revision, and SI revisions and determined that the actions taken adequately resolve the originally identified deficiency. However, while reviewing the walkdown documentation the inspector noted a number of items identified as not being in strict conformance with the specified expansion joint configuration. Conversations with licensee personnel indicate that these identified items have been reviewed by engineering, who determined that they were acceptable as is. The inspector has requested additional information to determine the adequacy of the licensee's use-as-is disposition. Therefore, this item will remain open pending review of the requested information. Also, the inspector has not performed field verifications of the installed fire barrier material.

ee. (Closed) CDR 390, 391/92-01, Potential Intersystem LOCA During Recovery From SBLOCA.

The potential for an intersystem LOCA was originally discovered at the D.C. Cook Nuclear Plant while conducting a simulator exercise in August 1991. The exercise involved recovery from a SBLOCA. At the time of SIS switchover from injection phase to recirculation

phase, a safety relief valve in the CCP miniflow line lifted, diverting water from the containment sump to the VCT and then to the HUT. WBN became aware of this potential deficiency when a similar simulator exercise was performed at SQN. The licensee initiated PER WBPER920004 and notified NRC on January 17, 1992, that the condition was considered potentially reportable.

The licensee performed an evaluation of the three concerns pertaining to this intersystem LOCA and determined on February 6, 1992, that this condition is not outside the design basis of the plant and is therefore not reportable. A capsulized discussion of the three concerns is as follows:

- RHR Pump Net Positive Suction Head: Westinghouse estimates the expected volume loss of containment sump water through the safety relief valve to be $\leq$ 60 gpm per CCP, or 120 gpm if both CCPs are running. WOG Emergency Response Guideline ES-1.2, Post-LOCA Cooldown and Depressurization, states that the RCS pressure can be reduced below RHR cut in pressure in 250 minutes. This would result in a potential loss of 30,000 gallons of containment sump water, or approximately 7.5 inches of water level. At the minimum containment sump level, the available suction head is greater than 15 feet above the NPSH required for the RHR pumps. Therefore, even a cooldown time twice that estimated in ES-1.2, a 15 inch drop in water level would have no adverse effect on NPSH requirements for the RHR pumps.

- Sump pH: For a SBLOCA where the RWST, RCS, and cold leg accumulators are emptied before all of the ice has melted, the concern is that the diversion of sump water could result in pH values falling below required levels after the remaining ice has melted. Following a LOCA, approximately 860,000 gallons of water accumulate in the reactor building including 380,000 gallons from the RWST and 372,000 gallons from melted ice. The remaining 108,000 gallons come from RCS inventory and the cold leg accumulators. The minimum boron concentration of the RWST and the ice is 2000 ppm and 1800 ppm, respectively. Since these are the two major sources of water to the sump, and their boron concentrations are not significantly different, it is not expected that the water in the sump will deviate from pH requirements.

- Impact of Sump Water in the Auxiliary Building: A conservative assumption would be that, with one HUT full and the other at the 50 percent level, there would be sufficient free space in the HUT to accept an additional 66,000 gallons of fluid. This is sufficient capacity to accept the flow (30,000 gallons) expected using the above referenced ES-1.2 guidelines and also sufficient to accept the more conservative flow (60,000 gallons) assuming twice the ES-1.2 guidelines. In addition, for a SBLOCA the peak clad

temperature of the fuel is assumed to be maintained at a sufficient value to preclude additional failed fuel. Therefore, diversion of sump water to the HUT will not result in radiation levels above those assumed in the original design, and equipment qualification and personnel exposure are not adversely impacted.

The inspector reviewed the above discussed justification for determining that the identified condition is not reportable and concurs in the licensee's assessment. This item is closed.

ff. (Closed) CDR 390, 391/92-02, Potential Common Mode Failure of the Auxiliary Control Air System

This 10 CFR 50.55e report identified a condition whereby two auxiliary control air system valves (0-32-241 and 0-32-281) had been improperly designated as "normally open." The piping upstream of these valves is classified as TVA Class C (ASME III Class 3, Seismic Category 1), while the downstream piping is TVA Class G (B31.1, Seismic Class 1[L]). Therefore, if the valves are normally open, the downstream piping would also have to be Class C. This improper identification of the required valve positions was instituted by DCN S-16120-A on June 6, 1991.

To correct the above deficiency, the licensee completed the following corrective actions:

- DCN S-18628-A was issued to correctly designate the identified valves as "normally closed." As these are manually operated valves, no hardware modifications were necessary. Operability of the system is not impacted because the valves should have originally been designated closed because they are drain valves from air receivers A-A and B-B.

- SOI-32.02, Auxiliary Air System, Revision 11, was issued to correctly identify the valves as "normally closed."

- An Extent Of Condition review was performed in accordance with SCAR WBSCA920004. This review determined that flow diagram 1-47W848-1 required additional changes to identify the applicable class breaks between vendor supplied and field installed equipment. The required revisions were accomplished via DCN S-18628-A.

- Formal training was conducted for engineering personnel to inform them of the identified deficiency and to stress attention to detail. This training was documented on memorandum JAS-92-007, dated January 27, 1992.

The inspector reviewed the above DCN, SOI, SCAR, and training records and determined that the licensee has adequately addressed

and corrected the identified deficiency. As no hardware corrections or modifications were required, no field verification was deemed necessary. This item is closed.

gg. (Closed) URI 390, 391/92-38-01, Overlay of Original Weld Operation Sheet Records Prior to Microfilming.

This item involves the discovery that certain weld operation sheets had been modified prior to microfilming. Due to legibility concerns, the top 1/3 of certain records had been re-written, cut, overlaid and stapled over the original record. This modified record was then microfilmed as the official record. As previously discussed in IR 390, 391/92-38, interim measures taken to assure that these modified records are not utilized as official records were as follows:

- The computer program for locating weld records was changed to revise the media field code from M (microfilm) to H (hardcopy) for all weld operation sheets held in hardcopy, and a note stating "Hardcopy in Vault" was added to the alternate field.

- A sample of weld operation sheets in the vault were reviewed to assure that the original records had been maintained.

- A sample of weld operation sheets filmed after microfilm roll W41391 was reviewed to assure that no overlay forms are contained therein.

- Filmed weld operation sheets prior to roll W41391 were pulled to ensure that only the hardcopy record in the vault was used for retrieving record information.

- A memo was sent to the N-5 review group informing them to use only the vaulted hardcopy for their review.

Subsequent actions taken to permanently correct the identified condition were as follows:

- A review of weld operation sheets on rolls prior to W41391 was completed and records containing overlays were identified.

- For all records containing overlays, the overlay sheet was removed from the original record, and both the original record and the overlay were filmed separately. These newly filmed pages are located on consecutive screens to facilitate comparison/verification. All of these refilmed records are now located on rolls E0941 and E1033.

- The computer program has been updated to accurately show the new film roll locations of the refilmed records. In

addition, the media field remains "H" and the remarks field still states "Retained In HC" to ensure that the hardcopy records are used to retrieve quality information.

To verify that the above described corrective measures have been appropriately completed, the inspector randomly selected three welds from film roll E0941 and compared the filmed version of the original sheet and the overlay to the hardcopy of these records. The welds selected for this comparison are: 0-077D-T043-23, 1-063B-D086-01, and 1-077C-D218-03B. This comparison revealed that the microfilmed version now accurately reflects the hardcopy documentation. The inspector then returned to the computer program and, for the three welds selected, verified that the media and remarks fields are noted as stated above, and that the location field accurately shows film roll E0941. Based on this selected sample, the inspector determined that the weld operation sheets which were improperly microfilmed are now correctly filmed and accurately reflect the hardcopy documentation. This item is closed.

hh. (Open) IFI 390/85-08-01, Containment Penetrations Discrepancies

Fifteen containment discrepancies were identified by resident inspectors during a comparison of the as-built plant to the FSAR. The findings were reported in IFI 390/84-52-01. In a subsequent report inspectors documented that 14 of the 15 examples had been corrected and re-inspected. One of the fifteen discrepancies had not been corrected; in addition, two new discrepancies were documented. These deficiencies constituted IFI 390/85-08-01. Specifically, the open item from IR 390/84-52 stated that the lower and upper compartment air monitor intake penetrations X-94 A,B,C, and X-95 A,B, and C, in the field both have two lines connected. The FSAR listed the penetration with three lines passing through it. The additional discrepancy listed in IFI 390/85-08-01 stated that penetrations marked as MK 059M and MK 060M were listed in the FSAR, Table 6.2.4-1, as having had three lines per penetration. However, they had only two lines installed.

The licensee initiated DCN P-01489-C which was stated to have corrected the identified discrepancies with various drawings, FSAR Figure 3.8.2-7 and FSAR Table 6.2.4-1. Amendment 65 revised FSAR Table 6.2.4-1 to show that one of the three lines in penetrations X-94 and X-95 is a spare line.

The inspector reviewed the FSAR change and found that one of the inspector's concerns had not been incorporated. The FSAR Table 6.2.4-1 lists the containment penetrations; in addition, Table 6.2.4-1 also identifies the reactor building penetration for piping that passes through both structures. In the case of penetrations X-94 and X-95, each of the assemblies had two sample lines passing through the containment and one spare line that was

truncated and capped in the annulus area. Hence, the associated reactor building penetration had only two lines passing through it. There was no installed spare as there was in the containment penetration. The FSAR table, however, still reflects that for each of the three lines in containment penetration X-95 there is a corresponding line in the MK 60M reactor building penetration (including the spare line). The X-94 and MK 59M penetrations are listed correctly. Further review of the table indicated that there may be several other such inconsistencies regarding reactor building penetrations associated with spare piping in containment penetrations.

The inspector field verified that reactor building penetration MK 60M contained only two lines.

The second part of the IFI was addressed in FSAR Amendment 72. That amendment changed Table 6.2.4-1 to reflect the fact that containment penetrations X-94 and X-95 each had two instrument lines and a spare line in each assembly. This was item 15 in the 1984 inspection report; however, the FSAR change request did not correctly change the same deficiencies on the FSAR table following 6.2.4-1. Table 6.2.4-4, Instrument Lines Penetrating Primary Containment, still refers to the spare lines in penetrations X-94 and X-95 as process lines for air monitors.

This item will remain open until inconsistencies are corrected or justified in FSAR Tables 6.2.4-1 and 6.2.4-4 for containment penetration spares and the associated reactor building penetrations.

ii. (Open) IFI 390/86-08-01, Discrepancy Between Most Heavily Loaded, Worst Condition, Shutdown Board 1B-B or 2B-B

This item was identified during a review of test procedures TVA-73B, Additional Diesel Generator Sequenced Loading, and TVA-73C, Additional Diesel Generator Qualification Test, during the previous preoperational testing in 1986. The inspector noted that test procedure TVA-73B designated the 2B-B shutdown board as the most heavily loaded, worst condition shutdown board when the FSAR documented that the 1B-B was the worst case loading. In the inspection report the inspector stated that the licensee would evaluate the issue and change either the FSAR or the test procedure.

The licensee provided a calculation, WBN EEB-MS-T103-0012, Diesel Generator Loading Analysis, Revision 13, that indicated that DG 2B was the most heavily loaded DG; however, the licensee has not updated Section 8.3 of the FSAR. Section 8.3 continues to refer to diesel generator 1B-B as the worst case loading on a DG.

IFI 390/86-08-01, Discrepancy Between Most Heavily Loaded, Worst Condition, Shutdown Board 1BB or 2BB, will remain open until the

FSAR discrepancy on DG loading is properly addressed in Chapter 8 of the FSAR.

jj. (Closed) IFI 390/86-12-06, 391/86-13-06, Follow-up on Licensee Review of Nonconformance Reports for Generic Applicability

This IFI item was opened as a result of the inspector's concern that the licensee efforts to correct identified installation deficiencies of ASCO solenoids lacked the rigor that would allow discovery of related but different problems on the same component. For example, in a 1986 tour the NRC identified a deficiency on the modification of ASCO solenoid hangers, the licensee follow-up apparently overlooked installation orientation restrictions, the effect that large electrical connector boxes attached to the valves might have on the seismic qualification of the assembly, and the effect on environmental qualification of disassembling vendor supplied valves to modify hanger assemblies. These issues were subsequently identified by the NRC.

In the inspection report, the inspector described his concern as "generic." After a detailed review of this specific issue, as well as other issues discussed in the same paragraph of the report, it is clear that the intent of the inspector's concern was to address a lack of cross-disciplinary reviews in licensee corrective action programs and, furthermore, that licensee corrective actions appeared to be driven by NRC findings rather than licensee initiative.

In reviewing the licensee's actions to resolve this issue, the inspector noted that the term "generic," as used in the NRC inspection report, was not implied by the licensee to mean multi-site reviews or similar conditions at Watts Bar.

NCR 6298 was initiated for this issue. The corrective actions showed how NCR scope was expanded as a result of investigations and NRC findings to include instruments other than ASCO solenoids (NCR W-415-P and W-416-P). These expanded investigations moved into environmental qualifications and installation orientations.

The inspector reviewed the licensee's response and determined that the scope of corrective actions was appropriate. The licensee had instituted significant corrective action program changes to address the use of cross-disciplinary reviews. The site program procedure, SSP-3.04, Corrective Action Programs, Revision 7, requires that significant corrective action reports get a review by a management reviewer. At WBN this review is performed by the Management Review Committee. This is a group of individuals having the education, experience, and training commensurate with assigned responsibilities within their respective organizations to review SCARs for validity, clarity and completeness. The NRC performed a detailed review of the SCAR process in IR 390, 391/91-29 and found the program acceptable.

The inspector determined that while each of the technical concerns raised from resident office concerns remains open, each of the items was subsequently escalated and is currently tied to another existing open NRC issue. The seismic issue of electrical connections to valves, followed by CDR 390, 391/86-59, remains open pending completion of the CAP on seismic issues. The environmental qualification issue followed by VIO 390, 391/86-02-01, reviewed in IR 390, 391/91-31, remains open pending completion of field work and the issue on the orientation of solenoids is followed by VIO 390, 391/86-18-01, Failure to Translate Vendor Requirements into Installation Instructions, remains open pending completion of field work.

The licensee has provided programmatic changes to address cross-disciplinary reviews of field identified problems. The field correction of specific and generic items related to this IFI is documented in other outstanding NRC open items. Based on these actions this item is closed.

kk. (Closed) IFI 390/86-27-01 and 391/86-26-01, Evaluation of the Licensee's New Corrective Action Program

This item was developed as a result of the review of corrective actions proposed in response to NRC Order EA 85-49 and applied to all licensee facilities. The corrective actions were to improve the methods of the identification and evaluation of conditions adverse to quality and the evaluation of the same conditions for generic applicability. The actions consisted of QA program procedure changes.

Closeout of this issue was documented in SQN IR 327, 328/87-55, under IFI 327,328/86-73-01 and applied to all facilities. Therefore, this item is closed.

ll. (Closed) IFI 390, 391/90-27-11, Valve Accelerations

This item was opened as a result of an inspection conducted on various areas of design control at SQN in which a number of design control deficiencies were identified. NRC requested that the licensee investigate the applicability of these findings to WBN.

Deviation D3.3-2 in the SQN report noted that valves were being qualified on a case-by-case basis to acceleration levels which exceeded the FSAR commitment of below 2g vertical and 3g horizontal, found in Section 3.9.2. The commitment to meet established design specifications was supported further by a note in FSAR table 3.9.2-3.

The licensee responded to the IFI and stated that the deficiency also applied to the WBN site and that there were numerous cases where increased accelerations had been approved on a case-by-case basis. The process was addressed in the licensee's piping

analysis design criteria documents but had not been included in the FSAR discussion.

The licensee submitted a FSAR change request for the applicable section. The change requested that the FSAR include the process of using engineering calculations to justify exceeding certain design limits on a case-by-case basis.

The inspector reviewed FSAR Amendment 64 and verified that the change documented the inspector's concern raised in the IFI. The change clearly permitted the process of allowing engineering calculations to justify exceeding certain design limits or commitments on a case-by-case basis.

This item is closed.

mm. (Closed) IFI 390/90-19-05, Operability Call for Missing Ring-Contact for 6900kv Breaker

This item was initiated to document a concern with the CAQR process. The CAQR process involves reviews that the licensee used to evaluate and determine the operability and generic applicability of identified deficiencies in plant equipment. A specific example was described that identified a CAQR that had been initiated during a maintenance activity that documented a missing part in a 6900 kv breaker assembly. The inspector questioned the fact that the CAQR operability statements indicated that operability was not affected. The inspector's concern was that the possible generic effect (i.e., on the operability of identical equipment at other TVA operating plants) would not be identified in a timely manner.

The inspector determined that at Watts Bar operability and generic applicability determinations for SCARs are made by the site licensing department. The process is documented in SSP-3.04, Corrective Action Program.

The most recent revision to SSP-3.04, Revision 7, effective January 7, 1993, revised Section 2.1 of the procedure. Section 2.1 described the process of SCAR initiation and evaluation of operability and generic applicability determinations. The revision established plant operations as the responsible department for operability and generic applicability determinations. The allotted times for the reviews were established as one day for WBN operability review and three days for a review of potential applicability to other TVA facilities. This revision places the evaluation of deficient conditions in the hands of the operating staff on the "front-end" of the evaluation process.

The inspector found that the issue of "determination of operability and generic issue applicability" had been identified

in related NRC inspections at Watts Bar. URI 390, 391/90-31-02 addressed the same issue. IR 390, 391/91-29 performed a detailed review of the SCAR operability and generic applicability determinations and found the licensee performance in this area acceptable. URI 390, 391/90-31-02 was closed in that report.

The inspector concurred with the earlier report and verified through the review of the program procedure SSP-3.04, which controls the SCAR processes, that recent changes, Revisions 4 through 7, do not reduce administrative controls below the levels practiced during the earlier NRC reviews. The inspector found that the recent changes enhanced the capability of the plant to identify operability issues in a timely manner by ensuring that the operations department, whose staff has the training and qualifications to make such determinations, becomes involved very early in the evaluation process.

This item is closed.

nn. (Closed) IFI 390, 391/90-27-06 Exhaust Installation D3.1-1

This issue was identified as a result of an NRC inspection of the design control processes at TVA's SQN. A number of deficiencies were identified in that inspection. The licensee conducted a review of the identified SQN design issues for applicability to WBN and determined that 25 of the 33 issues were applicable to WBN and required corrective action or further review. Item D3.1-1 in that report described a deficiency in the installation of quick exhaust valves.

Licensee reviews revealed two ECNs for WBN had been issued for modifications of control air lines on valves without documentation updating seismic qualification of the modified assembly. The ECNs, 4609 for Unit 1 and 4610 for Unit 2, installed quick exhaust valves in the tubing of the control air lines of two large control valves in the auxiliary feedwater system to improve the closing times.

The licensee responded to the IFI with corrective actions that addressed two deficiencies. The first was the overall adequacy of the identification of the need to perform seismic qualification analysis of components at WBN which permitted the issue to occur. The second was the specific corrective actions for the WBN case involving the auxiliary feedwater valves.

The TVA program that addressed the generic issues associated with the identification and documentation of equipment seismic qualification has been addressed in the design change process. The inspector reviewed EAI-3.05, Design Change Notices, Revision 10, and found that it requires the design change coordinator to review a proposed change for impact on all engineering disciplines. Appendix C of the procedure provides a checklist of

potential effects on design documents. Seismic analysis was included in that appendix.

The specific corrective actions for this IFI included the development of a calculation to analyze the effects of the modification on the seismic qualification of the level control valve assembly. TVA stated that a calculation had been performed that demonstrated that the addition of the quick exhaust valve weighing 0.6 pounds did not significantly affect the seismic operability of the control valve. The control valves in question were LCV-3-148,-156,-164, and -171. Calculation WBPEVAR8901026 was completed under corrective action of PIR WBNEEB8665, Revision 1.

The inspector reviewed in detail the licensee calculation WBPEVAR88901026 and verified that it met FSAR commitments in Chapter 3.9. The calculation, while quite explicit in justifying tubing hanger locations, was not a stand-alone document. For example, the qualification records for the quick exhaust valve itself were not contained or referenced in the calculation package. The licensee was able to provide the necessary documentation requested for the review. The inspector performed field verification of the valve configurations used in the calculations. No deficiencies were identified.

The inspector concluded, based on field observations and the documentation provided, that the seismic qualification of the modification that added quick exhaust valves to the control air piping of the auxiliary feedwater to SG level control valves had been performed adequately.

This item is closed.

oo. (Open) URI 390, 391/90-27-29, Load Calculations on De-rated Motors

This open item was initiated in October 1990 during the NRC closure of CDR 390/86-23 and 391/86-19, Diesel Generator Electrical Board Room Exhaust Fan Flow Rates. This item involves the DG, 480v Electrical Board Room Exhaust Fans 1A-A, 1B-B, 2A-A, and 2B-B, which were procured as 1.5 hp motors. The fan motors shipped by the manufacturer (Reliance) were actually 3 hp motors with nameplates that indicated the electrical current requirements and horsepower rating of a 1.5 hp motor. This resulted in higher than expected current demands, subsequently leading to circuit thermal overload device actuation and ultimately to underestimating electrical loading calculations developed for the DGs. A second example of the motor substitutions was identified where an expected 0.33 hp motor (based on nameplate data) was actually a 1.5 hp motor. The inspector's concern addressed DG loading calculations and the potential generic implications of other motors shipped by a large vendor may actually be larger than ordered.

The inspector also raised the concern of the additional DG electrical board exhaust fan needing to be reviewed as a second part of the URI.

The inspector determined that, regarding the additional DG, 480v auxiliary board room exhaust fan, the exhaust fan was purchased under a separate contract and has not experienced any tripping of thermal overload devices. Based on this, the licensee decided that there was not a reason to believe a problem existed regarding the derating of that motor or the possibility of overloading the fifth DG when it is called upon to replace any of the other four DGs. The licensee stated that preoperational testing, still to be conducted, will demonstrate the DG's ability to carry design loads.

Actions that had been completed at the time of the 1990 CDR closeout were noted in the latest response and were as follows:

The exhaust fan motors for the other DGs were replaced under workplan E6219-1. The replacement motors were 2.0 hp, 182T frame, 1800 rpm, Reliance Duty Master motors qualified in accordance with IEEE Standards 112, 323, 334, and 344. The motors were satisfactorily tested in accordance with IEEE Standard 112; As part of the CDR response, the licensee also had changed Engineering Design Specification SS-E9.2.01, "Alternating Current Induction Motors," to require the IEEE 112 testing, a program that would implement standards that would identify the actual current demands of electric motors regardless of the nameplate information.

The licensee determined that the preoperational testing (Generic Test Procedure GTEXXX-04) would identify conditions where motor running currents deviated from nameplate data. The licensee also performed a review of existing data bases to determine if a trend or generic conditions existed regarding derated motors and determined that no trend existed.

The inspector conducted field observations of the fan motor nameplate data. The inspector reviewed the generic test procedure and the IEEE standard 112 and determined that performing such tests would identify nameplate deficiencies and is adequate corrective action to prevent the recurrence of unidentified motor running currents for electric motors. The test, as written, compares data gathered under operating conditions to be compared with the nameplate data on the tested motor. This testing is capable of validating the DG shutdown board loading calculations regardless of the nameplate data on installed motors. This approach would preclude recurrence of having a motor in the field for which the running currents were based solely on nameplate information for those motors which have been tested. The inspector verified by direct observation that the motor installed on the additional DG board room exhaust was in fact provided by a

different vendor. The motors were 1.5 hp Pacemaker motors on a 145T frame.

The inspector's review found the licensee had not addressed the seismic implications of motors of a larger hp rating being labeled as smaller hp motors. This could be significant because as a "rule of thumb", as horsepower doubles the weight increases by a factor of 1.5 for small motors. For example, a 1 hp motor weighs about 65 lbs. while a 2 hp motor weighs 90 lbs, 5 hp, 145 lbs. while 10 hp is about 225 lbs. Accurate weight consideration is a factor in seismic calculations and can significantly affect the loading magnitudes for certain types of motor mountings configurations. There are many other factors beside hp ratings that can affect the weight of electric motors. The licensee could not provide information to the inspector that seismic concerns had been addressed as part of evaluation and closure process.

This item will remain open pending licensee actions to address how the seismic qualification calculation program would consider the affect of de-rated motors.

pp. (Open) URI 390/87-10-02, Use of Stick-On Electrical Wiring Fasteners

This item involved the use of ABCSMs which were found in some panels. Paragraph bb. of NRC IR 390/91-26 addressed the issue and determined that the issue had been adequately addressed by the licensee and could be closed on completion of field work associated with DCNs P-05479-A and F-09787-A. However, during the current inspection in the area, the inspector noted that in DG panel O-PNL-82-C-S some cables (including O3PP82-S-1867S,-1868S,-1870S) were resting against a rectifier heat sink due to failure of ABCSMs. The inspector advised the licensee of the possible problem with the cable bundles against the heat sink and WO C-149929 was issued. Unattached ABCSMs were also noted in 1A1-1, compartment 6, and 1B2-B, as well as other additional diesel generator panels, which appeared to be needed to perform a mechanical support function.

Review of the applicable DCNs and WPs indicated previous walkdowns had been looking for cable separation problems only and did not consider cases where the ABCSMs were used for mechanical support and training of cables. A meeting was held with licensee personnel who concurred that the corrective actions may have been too narrowly focused and agreed to re-address the issue.

Inspection for closure of this item included the following:

- Review of SAR requirements for Class 1E cable/wiring separation.

- Verified incorporation of DCN F-09787-A mounting requirements for ABCSMs into Specification G-38, Revision 11, and Drawing SD-E15.3.2.

- Review of a sample of 6 of 35 sections of WP D-05479 (sections 01, 02, 03, 15, 23, and 24) and field verification of completion of those workplan sections, including inspection of panels 1-SW-46-DC-S, 0-MCC-215-C2/DJ-S, 0-ARB-82-C-S, 0-PNL-82-C-S, 1-A1-1 (compartments 3 & 6) and 1B2-B (compartment 10). A number of workplan sections were completed without physical work required due to justified changes to specifications and exceptions to design criteria. In the sample inspected, required work had been completed.

- Review of DCN P-05479-A and Exceptions EX-WB-DC-30-4-18 and -19. It was noted that the copy of the DCN in the TSOB controlled document room was missing, but a replacement was readily obtained.

Based on this review, the inspector found the licensee's implementation adequate as far as separation was concerned but did not go far enough to completely close the issue. This item will remain open pending the licensee adequately evaluating the use of ABCSMs to provide cable restraint and training.

qq. (Closed) IN 88-49, Marking, Handling, Control, Storage, and Destruction of Safeguards Information

The NRC issued the subject information notice to advise licensees of identified weaknesses in the use and protection of safeguards information. The weaknesses were observed in the areas of marking, handling, control, storage, and destruction of safeguard material.

To assess the licensee's actions regarding the notice, the inspector reviewed SSP-11.03, Protection of Safeguards and Proprietary Information, Revision 3, TVA's response to VIO 390, 391/92-34-01, and interviewed site nuclear security, document control, training, startup test, engineering contractor, and badging personnel. Several SGI containers (Startup, Burns & Roe/Rust, Document Control, and Nuclear Security) were audited and control measures discussed with custodians. One randomly selected and two specific personnel training records were checked, and the same three personnel records checked with badging for criminal history background investigations.

From the inspectors review of SSP-11.03 it was not clear how a custodian could verify that no documents were missing from a SGI container since no detailed inventory of contents is required (procedure step 2.5.E states " Custodians...shall ensure that SGI documents removed...during the day are accounted for at the close of business"). The response to VIO 390, 391/92-34-01 stated that

accountability of SGI was inadequate unless a separate inventory listing of container contents was developed. At that time the licensee either required the custodians to develop a list for accountability purposes or was assured by the custodians that lists were already in use. From the inspectors review, a list was not evident at the startup area. In a subsequent meeting, further discussed below, the licensee clarified that the inventory list developed and implemented in the violation response was specifically for the Rust custodians involved. It was verified at that time by the licensee that all areas had an accountability list, therefore no further development of lists were required.

The inspectors audit of SGI containers at various locations found the following conditions:

1) Document Control

Proper markings were evident and access controls were in place. Custodian properly verified access clearance of the inspector by verbal discussions with the nuclear security supervisor accompanying the inspector. The nuclear security supervisor had also verified clearance via telephone to appropriate records office. The custodian properly indicated procedure would have to be referred to regarding an operational query since the container had not been accessed in approximately one year and specific details were understandably not immediately in mind. A few examples of transmittal sheets which were in the files had acknowledgement dates a month or more after transmittal (SSP-11.03 requires five days). The sheets were several years old and procedurally only require 30-day retention. All files and individual sheets randomly inspected had all cover sheets and markings as required. The inspector verified that an inventory sheet was present and was found accurate based on a random sample review of material in the container against the inventory sheet (list).

2) Startup

The inspector had a concern with the custodians attempt to open this container. It was noted that the custodian referred to a personal wallet, apparently for the container combination, in his attempt to open the container. This is contrary to SSP-11.03, step 2.5.G.6.b, which prohibits written combinations unless stored in another SGI container. The container was subsequently opened by another custodian. In discussions with the licensee and the individual involved, the inspector was informed that the number apparently referred to was only manipulation directions for the lock. Based on the fact that the custodian could not open the container and also from interviewing the individual, the inspector concluded that the custodian did

not have a correct number for the container. If the combination was in the wallet, it had been rendered obsolete by a combination change and therefore was technically not a violation. The licensee removed the individual from the custodian list. In addition, the SGI containers were transferred from control of the startup department. The inspector's review of the containers content noted the files had not been used for some time and were mostly scheduled to be transferred to Document Control. Markings and other procedural requirements appeared in order. The inspector found the licensee actions adequate to assure SGI would be adequately controlled, therefore the inspector considered this issue resolved.

3) Burns & Roe/Rust

The container was maintained in a separate combination locked room with custodial personnel manning the room. A good inventory (not procedurally required but initiated as a corrective action in above mentioned violation response) was being maintained, allowing personnel to account for each document filed, all copies, and destruction of unneeded items. Personnel were knowledgeable of the procedure and the requirement to run several blank copies after copying a SGI document to prevent subsequent possible impressions on other copied material. A program of direct control of SGI material by custodial personnel accompanying craft personnel into the field, wearing distinctive vests, was related to the inspector by the custodians. All materials observed were properly marked.

4) Nuclear Security

All materials observed were properly marked. No discrepancies were observed. Several records of audits of SGI areas were reviewed and appeared satisfactory. No examples of data processing media containing SGI were observed. The inspector was told that computer systems used for SGI were individual, personal-type computers, with required access controls in place, as required.

Review of training syllabus revealed that employees are familiarized with SGI handling in General Employee Training. Review of training records provided by the licensee for MST 003.001 showed all SGI personnel were trained within the preceding two years (oldest training September 1991) as required. SSP-11.03, step 2.2.D, requires refresher training every two years. MST 003.001 is the SGI handling course for custodians, while 303.001 is a general employee course. However, comparison of a signature list for custodian verification of inventory and combination controls issued by the licensee on January 27, 1993, contained five individuals

who were not on the training record printout. Additional records were provided indicating that three of these individuals were trained, one was the instructor of the course, and one was a corporate employee who was trained September 1991 on 003.002, designated classifier training.

The inspector performed an inspection of the badging department and verified that criminal checks had been performed on the two SGI personnel selected and not on the non-SGI person selected, per requirements. No discrepancies were noted. The badging SGI container was also audited, with container and contents sampled found properly marked, inventoried, destruction of records recorded, and logged including computer floppy disks. The inspector was assured that the two computer systems in use were not networked to any common site system, but did not verify that an access code was required for system entry, no modems or dedicated lines connecting to other systems were present, or that the internal storage device (hard drive) did not store SGI.

The inspector was informed that a revision of SSP-11.03 was in progress.

Based on review of the above items, the inspector found the licensee control of SGI is adequate. The licensee also indicated additional audits and attention would be directed to the area of SGI control.

Based on the inspectors review as documented above, this Information Notice is closed.

rr. (Closed) URI 390/89-14-04, Conduit Branch Line and Bend Design Criteria

This item resulted from an NRC/NRR Engineering Team Inspection directed at the Corrective Action Plan for equipment seismic qualification of heating, ventilation, and air conditioning. Specifically, the concern is identified from Source Issue 4 of the licensee's CAP matrix for conduit and conduit supports. The concern was over lack of clarity in how branch lines and bends were addressed by the design span load table method.

The issue has been addressed by NRR in SSER 9, Appendix S, page 13 and 14, and closed. Closure was based primarily on review of design criteria WB-DC-40-31.10, Revision 7, February 21, 1992, which eliminated the span load table method of previous revisions and allowed only equivalent static analysis and response spectrum methods of qualification of conduit systems.

Based on NRR's review as documented in SSER 9, this item is closed.

ss. (Closed) CDR 390, 391/91-19, Vendor Quality Assurance Program for Class 1E Cable (Teledyne Thermatics)

This 10 CFR 50.55(e) report involved the licensee's discovery, by audit at the manufacturer's facility, that class 1E cable purchased from Teledyne Thermatics did not comply with procurement requirements pertaining to material traceability and quality assurance records. The non-compliance affected cables bought under two contracts, 89NLB-75267A (all cable received on site) and 90NLC-75670-03, subject cable was placed on hold.

Problem Evaluation Report WBPER910187 was issued to document the deficiency. The report was rolled over into SCAR WBSCA910247. NRC IR 390, 391/91-26, paragraph R addressed the issue and found corrective actions in progress adequate for construction restart.

The licensee submitted a final report dated December 6, 1991, detailing corrective actions accomplished to resolve the issue. These included manufacturer's testing of cable samples, specification revisions for supply of raw materials by qualified suppliers, prohibiting use of stock material for nuclear products, and training for production personnel on the importance of accurate record keeping. The cables tested were from vendor's stock verified to be from the same production runs as that supplied to WBN, samples of the WBN cable, and samples of same-lot cables supplied to SQN. In addition, the licensee witnessed the vendor tests and reviewed production documentation which had not previously been made available by the vendor, and found it acceptable. Based on these test results, the onsite cable was approved for use.

The inspector reviewed the problem report, the corrective action report, vendor replies to licensee audit reports, and numerous documented cable sample test results and determined the issue had been adequately addressed.

Based on this review, this item is closed.

tt. (Closed) URI 390/92-05-06, 391/92-05-05 Soil Thermal Resistivity for Underground Duct Banks

This item resulted from an NRC audit team concern reported in IR 390, 391/92-05 about the large amount of heat from interacting underground piping being used as input in ampacity evaluations. The staff questioned the heat input value since the value used could cause soil dryout and greatly increase the soil resistivity value used in the calculations. The staff requested a re-evaluation of the calculations be made by the licensee using input from both the electrical and mechanical groups in order to arrive at a more realistic heat flux contribution from the pipes and the duct bank. The staff indicated that if the recalculation demonstrated a heat flux of no more than 100 watts/ft, then the

applied soil thermal resistivity of 90 Rho used in the ampacity calculation would be acceptable.

The inspector reviewed applicable portions of the following calculations provided by the licensee:

1) WBN-MNSA-001, Piping Heat Flux for Input to IPS Cable Bank Ampacity Calculation, Revision 3

2) WBPEVAR9003002, Intake Pumping Station Underground Duct Bank Analysis, Revision 1

3) WBN EDM-FM-082890, Heat Loss from Various Piping Systems at Specific Yard Locations, Revision 2

4) WBPE0829206002, Diesel Generator Underground Duct Bank Analysis, Revision 1

Two additional drawings not initially provided with the WBN-MWSA-001 excerpts showing ductbank/piping plan and elevations were provided on request for review. Additionally, the inspector reviewed DCN M-14874-A, which parallels existing Unit 1 reactor coolant pump cables in the duct bank with the co-located Unit 2 cables (Unit 2 cables will be rerouted via another DCN) to reduce individual cable ampacity. WPs 14874-1/18 have been assigned for preparation to accomplish the change.

The above review showed that the revised calculations demonstrated a heat flux of less than 100 watts/ft, satisfying the staff's concerns. The calculations showed that the duct bank combined heat loss was reduced from 129.9 to 58.1 watts/ft.

Based on this review, this item is closed.

uu. (Open) CDR 390/92-10, Design Control

This CDR, issued subsequent to VIO 390/92-27-06, involved failure of DCN M-12564-A to incorporate all required fuse replacements specified by calculation WBN-EEB-MS-TI 007-005, 125 VDC Vital Control System Fault Calculation. The deficiency had been identified as an example of VIO 390/92-27-06.

The licensee requested that this item be reviewed together with VIO 390/92-27-06. The inspector reviewed the violation response and determined that further information and review of the violation response would be necessary prior to closure (specifically, review of documentation of corrective actions noted). The licensee had not identified that the violation was formally ready for closure review.

Based on the above review, this item will remain open and closed with VIO 390/92-27-06.

vv. (Open) P21 89-11, Morrison-Knudsen Diesel Generator Starting Air

This item concerned diesel starting air circuit logic for DGs with tandem air starting systems. In some instances, loss of one of the air start systems would cause continuous recycling of the remaining system preventing DG start. The licensee had already noted the problem with CAQR 890376 when it was addressed by a Morrison-Knudsen Part 21 notice. The item was addressed in IR 390, 391/90-24 and left open, pending completion of circuit modifications.

The inspector reviewed the information previously assembled on the issue and IR 390, 391/90-24. Morrison-Knudsen's Part 21 report, CAQR 890376, DCN M-12839-A, DCN-08450-A (obtained on request from Licensing), and drawing 45-W760-82, Sheet 4, Revision 2, were reviewed. DCN 08450-A addresses circuit changes for the fifth DG, while DCN M-12839-A primarily addresses changes required for DGs 1-4. The DCNs appear to adequately implement the circuit corrections required, but the drawing change (sheet 4) had not yet been made. The inspector requested information on work packages issued to effect the circuit changes, but the work is apparently not scheduled until later in 1993.

This item remains open pending completion of the remaining work and subsequent review by the inspector.

11. Exit Interview

The inspection scope and findings were summarized on January 29, 1993, with those persons indicated in Paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results. Dissenting comments were not received from the licensee. Proprietary information is not contained in this report.

Item Number	Status	Description and Reference
IEB 80-11	Open	IEB - Masonry Wall Design (Paragraph 2.a)
390/84-76-01	Closed	URI - Missing Pipe Support Calculations (Paragraph 2.b)
390/85-08-01	Open	IFI - Containment Penetrations Discrepancies (Paragraph 2.hh)
390/85-45 391/85-44	Open	CDR - Unanalyzed Diesel Generator Transients for a Blackout Followed by a Safety Injection Signal (Paragraph 2.j)

390/86-02	Closed	Part 21 - During F.W. Transient 3, the SOR Differential Pressure Switches Failed to Function to Cause a Scram (Paragraph 2.c)
390/86-02-07 391/86-02-07	Closed	IFI - Further Review of QMS Activities (Paragraph 2.d)
390/86-08-01	Open	IFI - Discrepancy Between Most Heavily Loaded, Worst Condition, Shutdown Board 1BB or 2BB (Paragraph 2.ii)
390/86-12-06 391/86-13-06	Closed	IFI - Follow-up on Licensee Review of Nonconformance Reports for Generic Applicability (Paragraph 2.jj)
390/86-16	Closed	CDR - Extreme Wear Shown on Westinghouse Switchgear Breakers (Paragraph 2.k)
390/86-21-01	Closed	VIO - Failure to Meet Post-Modification Testing Requirements (Paragraph 2.l)
390/86-22-02 391/86-22-02	Closed	VIO - Missing Calculations for Pipe Supports (Paragraph 2.e)
390/86-27-01 391/86-26-01	Closed	IFI - Evaluation of the Licensee's New Corrective Action Program (Paragraph 2.kk)
390/87-07-02 391/87-07-02	Closed	URI - Missing Calculations (Paragraph 2.f)
390/87-10-02	Open	URI - Use of Stick-On Electrical Wiring Fasteners (Paragraph 2.pp)
88-01	Closed	BU - Defects in Westinghouse Circuit Breakers (Paragraph 2.m)
390/88-01 391/88-01	Open	CDR - Auxiliary Control Air Compressor Control Circuits Must be Manually Reset After Loss of Off-Site Power (Paragraph 2.n)

88-31	Closed	IN - Steam Generator Tube Rupture Analysis Deficiency (Paragraph 2.o)
88-49	Closed	IN - Marking, Handling, Control, Storage, and Destruction of Safeguards Information (Paragraph 2.qq)
390/89-02-03	Closed	URI - Calculation of All Loads on Concrete Slabs (Paragraph 2.p)
89-11	Open	Part 21 - Morrison-Knudsen Diesel Generator Starting Air (Paragraph 2.vv)
390/89-14-04	Closed	URI - Conduit Branch Line and Bend Design Criteria (Paragraph 2.rr)
390/90-03-01 391/90-03-01	Closed	IFI - Procedure Adherence "Shall/Should" (Paragraph 2.q)
90-05	Closed	Part 21 - Swing Arm in Borg-Warner Check Valve Found Broken (Paragraph 2.r)
390/90-19-01	Closed	VIO - Failure to Implement Administrative Procedures (Paragraph 2.s)
390/90-19-04 391/90-19-04	Closed	IFI - Verification of Consistency for CAQ Requirements in ACPs (Paragraph 2.g)
390/90-20-02	Closed	URI - Anchor Bolt Installation Practices (Paragraph 2.u)
390/90-19-05	Closed	IFI - Operability Call for Missing Ring-Contact for 6900kv Breaker (Paragraph 2.mm)
390/90-20-06 391/90-20-06	Open	URI - High Pressure Fire Protection - Microbiologically Induced Corrosion (Paragraph 2.t)
390/90-24-03 391/90-24-03	Open	IFI - Adequacy of Labeling (Paragraph 2.v)

390/90-27-06 391/90-27-06	Closed	IFI - Exhaust Installation D3.1-1 (Paragraph 2.nn)
390/90-27-11 391/90-27-11	Closed	IFI - Valve Accelerations (Paragraph 2.ll)
390/90-27-16 391/90-27-16	Closed	IFI - Cable Tray Support Base Plate Analysis (Paragraph 2.h)
390/90-27-17 391/90-27-17	Closed	IFI - Base Plate Design Criteria (Paragraph 2.w)
390/90-27-28 391/90-27-28	Closed	IFI - ERCW Screen Wash Pump Control (Paragraph 2.x)
390/90-27-29 391/90-27-29	Open	URI - Load Calculations on De-rated Motors (Paragraph 2.oo)
390/91-03-01 391/91-03-01	Closed	URI - RCS Cooldown Due to AFW Design (Paragraph 2.y)
390/91-04-01 391/91-04-01	Closed	IFI - Test Program for Evaluation of Shallow Undercut Anchors (Paragraph 2.z)
390/91-04-05	Closed	IFI - Condition Adverse to Quality (Paragraph 2.aa)
390/91-07 391/91-07	Closed	CDR - Hardware Defects on Class 1E Circuit Breakers (Paragraph 2.bb)
390/91-18-01 391/91-18-01	Closed	IFI - Inconsistent Information in Response to Violations (Paragraph 2.i)
390/91-19 391/91-19	Closed	CDR - Vendor Quality Assurance Program for Class 1E Cable (Teledyne Thermatics) (Paragraph 2.ss)
390/91-33-01 391/91-33-01	Closed (Reinspected)	VIO - Failure to Maintain Housekeeping Surveillance (Paragraph 2.cc)
390/91-38 391/91-38	Open	CDR - Lack of Documentation for Fire Barrier Material in Seismic Expansion Joints (Paragraph 2.dd)

390/92-01 391/92-01	Closed	CDR - Potential Intersystem LOCA During Recovery from SBLOCA (Paragraph 2.ee)
390/92-02 391/92-02	Closed	CDR - Potential Common Mode Failure of the Auxiliary Control Air System (Paragraph 2.ff)
390/92-05-06 391/92-05-05	Closed	URI - Soil Thermal Resistivity for Underground Duct Banks (Paragraph 2.tt)
390/92-10	Open	CDR - Design Control (Paragraph 2.uu)
390/92-38-01 391/92-38-01	Closed	URI - Overlay of Original Weld Operation Sheet Records Prior to Microfilming (Paragraph 2.gg)

12. List of Acronyms and Initialisms

ABSCM	Adhesive Backed Cable Support Mounts
AC	Alternating Current
ACA	Auxiliary Control Air
AFW	Auxiliary Feedwater
ANSI	American National Standards Institute
ASCO	Automatic Switch Company
ASME	American Society Of Mechanical Engineers
BU	Bulletin
B-W	Borg-Warner
CAI	Construction Administrative Instruction
CAP	Corrective Action Plan
CAQR	Condition Adverse to Quality Report
CCP	Centrifugal Charging Pump
CDR	Construction Deficiency Report
CEB	Civil Engineering Branch
CM	Chemistry Manual
CPSES	Comanche Peak Steam Electric Station
CST	Condensate Storage Tank
DBVP	Design Baseline and Verification Program
DCN	Design Change Notice
DG	Diesel Generator
dP	Differential Pressure
DR	Discrepancy Report
DSCN	Design Standard Change Notice
EAI	Engineering Administrative Instruction
ECN	Engineering Change Notice
EDS	Engineering Data System
EMS	Equipment Management System
ERCW	Essential Raw Cooling Water

FSAR	Final Safety Analysis Report
FW	Feedwater System
gpm	Gallons Per Minute
HAAUP	Hanger and Analysis Update Program
hp	Horsepower
HPFP	High Pressure Fire Protection System
HUT	Holdup Tank
IDI	Integrated Design Inspection
IEB	Inspection and Enforcement Bulletin
IFI	Inspector Follow-up Item
IN	Information Notice
IPS	Intake Pumping Station
IR	Inspection Report
LAI	Licensing Administrative Instruction
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power (Blackout)
MAI	Modification and Addition Instruction
MI	Maintenance Instruction
MIC	Microbiologically Induced Corrosion
MK	Mark
MR	Maintenance Request
NCR	Nonconformance Report
NDE	Non-Destructive Examination
NEB	Nuclear Engineering Branch
NE-CE	Nuclear Engineering-Construction Engineering
NPP	Nuclear Performance Plan
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Office Of Nuclear Reactor Regulation
PDR	Public Document Room
PER	Problem Evaluation Report
PIR	Problem Identification Report
ppm	Parts Per Million
PRD	Problem Reporting Document
PSDM	Pipe Support Design Manual
QA	Quality Assurance
QMP	Quality Methods Procedure
QMS	Quality Management Staff
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RIMS	Records Information Management System
RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SBLOCA	Small Break LOCA
SCAR	Significant Corrective Action Report
SCR	Significant Condition Report
SDI	Shell Drop-In Anchor
SG	Steam Generator
SGI	Safeguards Information
SI	Surveillance Instruction
SIS	Safety Injection System
S&L	Sargent & Lundy

SMI	Special Maintenance Instruction
SMP	Startup Manual Procedure
SOI	System Operating Instruction
SOR	Static "O-Ring"
SQN	Sequoyah Nuclear Plant
SRN	Specification Revision Notice
SSD	Shell Self-Drilling
SSER	Supplemental Safety Evaluation Report
SSP	Site Standard Practice
TI	Technical Instruction
TVA	Tennessee Valley Authority
UL	Underwriters Laboratory
URI	Unresolved Item
UT	Ultrasonic Testing
VCT	Volume Control Tank
WBN	Watts Bar Nuclear Plant
WBRD	Watts Bar Reportable Deficiency
WOG	Westinghouse Owners Group
WOG-ERG	Westinghouse Owners Group Emergency Response Guidelines
WP	Workplan