UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-390/95-70 and 50-391/95-70

Applicant: Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

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

Facility Name: Watts Bar 1 and 2

Inspection Conducted: September 10 through October 7, 1995

Inspector: Caudle Julian for 10/31/95
P. K. Van Doorn, Senior Resident Inspector Date Signed

Other Inspectors: S. Cahill, B. Crowley, F. Jape, W. Miller, J. Moorman, C. Smith, P. Taylor

NRC Contractor Inspectors: J. Cummins (paragraphs 9.0, 10.0, 11.0, 12.0, 13.6.1, 13.6.2, 13.6.3)
B. Smith (paragraph 9.0, 10.0, 11.0, 12.0)

Approved by: John P. Jaudon for 10/31/95
C. Julian, Chief Date Signed
TVA Operations Branch
Division of Reactor Projects

SUMMARY

Scope:

This routine, resident inspection was conducted in the areas of preoperational test procedure review, plant operations, operational staffing, evaluation of applicant self-assessment capability, maintenance activities, plant support, maintenance inspection open items, operating procedures review, system turnover, maintenance procedures review, actions on previous inspection report concerns, actions on open safety issues, and actions taken in response to previous inspection findings.

Enclosure 1

Results:

In the areas inspected, violations or deviations were not identified.

Preoperational Test Program: One preoperational test instruction was reviewed during the inspection period. Only minor problems were noted. (Paragraph 2)

Operations: Operators generally performed well; however, poor operator performance was noted for a loss of ice condenser cooling event. One minor housekeeping comment was noted. Conservative evaluation of fuel load readiness items was noted. (Paragraph 3)

Operational Staffing: A review of the qualifications of welding and non-destructive evaluation personnel found no problems. (Paragraph 4)

Self-Assessment: The inspector noted that plant management was still having to assure adequate root cause evaluations through the management event review process. (Paragraph 5)

Maintenance Activities: Six activities were observed. Actions noted were conservative, and personnel typically followed procedures with one minor problem noted where personnel failed to cover several ice condenser bays as required. (Paragraph 6)

Plant Support: The inspector observed poor performance in the Operations Support Center during a monthly emergency plan drill which was also noted by the applicant's management. The inspector observed an idling truck left unattended in the protected area which was contrary to the security requirements. (Paragraph 7)

Maintenance Inspection Open Items: Three weaknesses and nine deficiencies were closed. One deficiency remains open for further root cause determination, and a second deficiency remains open until further verification program omissions are corrected. (Paragraph 8)

Operating Procedures Reviews: The inspectors concluded the reviewed system operating procedures provided appropriate guidance. However, some problems were noted such as poor component locations which was a repeat problem. (Paragraph 9)

System Turnover: Review of two systems indicated that the system turnover process was effective. The inspector identified a concern with the use of variances to deviate from procedural requirements. (Paragraph 10)

Maintenance Procedures Review: A validation and review of several preventive maintenance procedures did not find any significant problems. (Paragraph 11)

Open Issues: Five previous inspection report issues were closed. Twenty-four generic safety issues were verified to be adequately addressed. Two open items were closed. No problems were identified. (Paragraph 12, 13, & 14)

REPORT DETAILS

1.0 Persons Contacted

1.1 Applicant Employees:

*R. Baron, General Manager, Nuclear Assurance and Licensing
*M. Bajestani, Assistant Plant Manager
R. Beecken, Maintenance and Modifications Manager
A. Capozzi, Program for Assurance of Completion and Assurance of Quality Project Manager
S. Casteel, Independent Review and Analysis Manager
J. Cox, Radiological Control/Chemistry Manager
S. Crowe, Quality Control Manager
*W. Elliott, Engineering Manager
*P. Hughes, Radiological Control Manager
*D. Kehoe, Site Quality Manager
*D. Koehl, Technical Support Manager
S. Krupski, Instrument Maintenance Manager
D. Kulisek, Operations Support Manager
*D. Malone, Audits and Assessments Manager
*R. McCollom, Maintenance Planning and Technical Superintendent
*R. Mende, Operations Manager
B. Obrien, Electrical Maintenance Supervisor
*P. Pace, Compliance Licensing Manager
R. Purcell, Plant Manager
J. Rupert, Engineering and Materials Manager
J. Scalice, Site Vice President
*B. Schofield, Licensing Manager
W. Skiba, Trending Manager
D. Stewart, Site Support Manager
T. Stockdale, Operations Superintendent
C. Touchstone, Licensing Engineer
D. Voeller, Chemistry Manager
J. Wallace, Human Resources Manager
*O. Zeringue, Senior Vice President, Nuclear Operations

Other applicant employees contacted during this inspection included numerous craftsmen, engineers, operators, and administrative personnel.

1.2 NRC Personnel:

*S. Cahill
B. Crowley
*P. Fredrickson
F. Jape
J. Jaudon
C. Julian
N. Merriweather
W. Miller
J. Moorman

C. Smith
P. Taylor
*P. Van Doorn
*G. Walton

1.3 NRC Contractors:

J. Cummins
B. Smith

*Attended exit interview

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

2.0 PREOPERATIONAL TEST WITNESSING AND PREOPERATIONAL TEST RESULTS REVIEW (70311, 70351, 70400)

The inspectors utilized the inspection guidance of RG 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants, Revision 2, which provides criteria for a preoperational test program. It requires that preoperational tests be designed to satisfy the test objectives, contain appropriate acceptance criteria, and require the documentation of sufficient information to permit adequate evaluation of the test results. The inspectors also utilized information contained in such documents as FSAR Chapter 14, Initial Test Program, and other FSAR sections, applicant design drawings and systems descriptions, and engineering output documents.

Several of the procedures reviewed had errors or deficiencies which resulted in the identification of several questions and comments. The questions and comments were discussed with appropriate personnel for resolution. In most cases the applicant agreed with the comment and indicated that it would be incorporated into the procedure by a change notice or a revision to the procedure. With the exception of those procedures with comments pending resolution, the PTIs reviewed were found to be adequate. The inspectors found that other than the comments documented in this report, the technical content of the procedures, required scope of testing, and acceptance criteria were consistent with design documents and FSAR commitments. In addition, the information required to be documented in the procedure was considered adequate to perform an independent evaluation of the test results. A detailed discussion of the procedure reviews and findings are summarized below.

2.1 PTI-236-03, Vital 125 VDC Power System, Revision 0

Procedure PTI-236-03 was reviewed along with the following documentation to determine if the scope of the tests, test objectives, and acceptance criteria were adequate to demonstrate that the 125 VDC power system would perform its design function.

- FSAR Chapter 14, Table 14.2-1, DC Power System Test Summary
- FSAR Chapter 8.32, DC Power System

- Design Criteria WB-DC-30-27, AC and DC Control Power Systems, Revision 19

- Electrical Design Standard DS-E3.11, Batteries and Chargers, Definitions and Capacities, Revision 1

- IEEE Standard 450-1980, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations Substations

- RG 1.68, Initial Test Programs for Water Cooled Nuclear Power Plants, Revision 2

- Surveillance Instruction 0-SI-236-55, 125 VDC Vital Battery V 60 Month Performance Test, Revision 1

Based on the above reviews, the inspector determined that the scope of the tests, as described in Procedure PTI-236-03, Section 6.0, and Instruction 0-SI-235-55, was consistent with the applicant's commitments delineated in FSAR Chapter 14, Table 14.2-1. Omission of FSAR Chapter 14, Test Acceptance Criteria 8 from Section 5 of the PTI was noted and discussed with the cognizant engineer. This acceptance criteria required demonstrating that alarms, protective devices, indicators, breakers, and interlocks would operate in accordance with vendor and design documents. The inspector was informed that these acceptance criteria were removed from the PTI in an earlier revision and performed in other tests based on prior discussions with the NRC. The inspector verified that the overvoltage protective relay and associated alarm for Charger V had been disconnected and was no longer within the scope of the preoperational test. Other alarms, indications, and interlocks required to be verified by Procedure PTI-236-03 were determined to be in accordance with design requirements upon reviewing Section 6.3, 125 VDC Vital Battery Board V Indication and Alarms.

The technical basis for determining if the test requirements and test acceptance criteria had been satisfied were contained in design output drawings listed in Procedure PTI-236-03, Section 2.2, Drawings. Additional design bases information related to the performance of Procedure PTI-236.03 were contained in Design Criteria WB-DC-30-27. The inspector reviewed these design bases documents and verified that they supported the test requirements and test acceptance criteria specified in Procedure PTI-236.03 and Surveillance Instruction 0-SI-236-55. No significant deficiencies were identified during these reviews. The inspector concluded that the procedure and surveillance instruction content, required scope of testing, and acceptance criteria were consistent with the applicant's design bases documents and FSAR Chapter 14 commitments.

The following minor deficiencies were identified during the above reviews. They were discussed with the cognizant engineer who concurred with the inspection findings.

- PTI-236.03, Revision 3

Step 4.2.4 has a typographical error where punch-list item is repeated.

Step 6.1.69 has a typographical error in the second line: delete word "to."

Step 6.1.83 similar to Step 6.1.69.

Step 6.3.93 fuse identification is incorrect; change to 0-FU-236/109 in lieu of 102.

- Surveillance Instruction 0-SI-236-55

Section 6.1, Performance Test

Requirements for independent verification of calculations using data collected during the test need to be specified. This comment is also applicable to Section 6.2.

Step 6.1(24), Low Overall Voltage Shutdown, is incorrect as shown; voltage value should be 106.75.

- Section 6.2, Battery Recharge

Establish requirements for documenting four-hour readings on suitable attachment to the surveillance instruction.

Within the area reviewed, no violations or deviations were identified.

3.0 PLANT OPERATIONS (71707)

The inspectors reviewed and observed plant operations during the reporting period to verify conformance with applicable requirements. At the start of the report period, RCS conditions were 90°F and depressurized to support RCS check valve work. Inspector observations included control room conduct, Operations support of work activities, implementation of final draft Technical Specification requirements, and clearance tagging. The inspectors routinely reviewed operator logs, shift relief records, and other documentation for Operations programs. Daily plant status meetings were attended and plant tours were routinely taken during the reporting period. The inspectors observed that control room access was generally controlled to limit the number of activities being conducted. Operators were alert, and alarms were normally acknowledged and investigated promptly and thoroughly. Communications were formal and generally thorough with adequate use of repeat backs.

The inspector identified one housekeeping concern regarding floor paint chipping being conducted on elevation 737 of the auxiliary building. The thermal barrier booster pumps were not adequately covered during this activity and received an exterior coating of dust. However, they were not running and, therefore, did not draw a significant amount of dust into the motor. The inspector expressed concern to the applicant regarding pending chipping of elevation 713. This level will be a challenge due to the large number of pumps in the area to be chipped.

At the end of the report period the RCS was 77°F, depressurized, with level at 721.5 feet and one train of RHR cooling in service. During the report period, the plant experienced a problem with a trip of the ice condenser cooling system that went undetected by the Operations staff for over seven hours on September 21.

3.1 Ice Condenser Cooling Loss

The ice condenser cooling loss on September 21 was caused by an inadvertent closure of a glycol system containment isolation valve. The worker who caused the valve closure properly notified the Operations staff who reopened the valve. However, the Operations staff members did not recognize that the valve closure automatically tripped the glycol circulation pumps and chiller units, thereby securing cooling to the ice condenser. Consequently, the operators took no further action, and the problem was not identified for over seven hours until a high temperature alarm was received on an ice condenser temperature switch. Several ice condenser temperature recorder alarms were received, and a shift turnover had taken place during this seven hours. This event demonstrated a series of operational deficiencies including a failure to respond to alarms adequately, a failure to log the valve closure and restoration, inadequate shift turnover control board walkdowns, and delays in verification of running equipment status by AUOs. The inspectors expressed concern at the similarity of this failure to verify alarms to the actions observed in an August 24 inadvertent opening of one ice condenser lower inlet door and consequent ice melt. This was documented in NRC IR 50-390,391/95-60. In that event an alarmed door remained open for over 20 minutes until a high temperature alarm was received. Incident Investigation W-95-012 evaluated the August 24 event, the results of which are documented under the self-assessment paragraph in this report. The applicant responded seriously to the September event and initiated PER WBPER950575 and performed a human performance evaluation. The applicant conducted training on the event for all operators. The inspector attended one of these sessions and concluded the training was very good and appropriately highlighted how several operational barriers were violated. The applicant also planned to perform training on recorder operation which the inspector considered beneficial because the ice condenser temperature recorder is unique and several other new recorders have recently been made operational. Although the inspectors were satisfied with the applicant's response, they concluded the operational deficiencies were examples of poor operator performance.

3.2 Fuel Load Readiness Reviews

The inspector attended a fuel load readiness review meeting conducted by the applicant. These meetings assign the tracking codes to determine if an open work or administrative item must be completed before fuel load or other applicable milestones. The inspector observed that the applicant's technical support group was recently assigned full ownership of the codes for each plant system. Code changes and initial assignments had to be approved by Operations and could only be entered into the Master Tracking System by Technical Support. The inspector observed that the coding decisions were very conservative and concluded the applicant was appropriately identifying work required prior to fuel load. The inspector identified a minor discrepancy between codes on different data bases that was corrected by the applicant. The inspector concluded that Operations and Technical Support exhibited excellent ownership of plant systems.

4.0 Operational Staffing (36301B)

The inspectors verified the adequacy of the applicant's staff qualifications in the areas of welding and NDE, as detailed below.

The applicable codes for qualification of these personnel are:

- Welding: ASME Section IX, 1986 Edition through the latest Edition/Addenda at the time of qualification

  AWS Standard Welding Code D1.1, 1986 Edition through the latest Edition/Addenda

- NDE: American Society for Nondestructive Testing (ASNT) SNT-TC-1A, 1992 Edition

4.1 Welder Qualifications

The inspectors reviewed Procedure SSP-7.52, Revision 3, Qualification, Certification, and Continuity of Personnel Performing WBS. In addition, welder qualification records, including original qualification, continuity, and current computer printout for the two most common qualifications (PQTs GT701L and SM4B3H) were reviewed for the following plant welders: 6AEJ; 6RRA; 6AEK; 1EF; 6XT; 6BX.

4.2 NDE Qualifications

Procedure IEP-200, Revision 1, Qualification and Certification Requirements For Nuclear Power (NP) NDE Personnel, was reviewed. In addition, qualification records, including records of education, experience, training, examination results, certification, and current vision tests, were reviewed for the following NDE personnel:

NDE Method	Level	Number Reviewed
PT	II	7
MT	II	5
VT (Weld)	II	7
VT (ISI)	II	2
UT	II	2

4.3 Summary

The inspectors found that welding and NDE personnel were qualified in accordance with applicable codes and the qualification records were current and in order.

No violations or deviations were identified.

5.0 EVALUATION OF APPLICANT SELF-ASSESSMENT CAPABILITY (40500)

The inspector attended the PERP meeting for II W-95-012 on September 15, 1995. This II was initiated by the applicant in response to the inadvertent opening of one ice condenser lower inlet door and consequent ice melt on August 24, 1995. The inspector noted that the investigation was conducted by only one individual versus a multi-disciplined team, although the individual received assistance from others. The inspector identified that the root cause analysis presented to management was insufficient on one issue involving marking a step in a maintenance instruction "not applicable." The II event manager focused on the worker's unawareness of the reason for the step as the root cause in his presentation. However, the plant manager had the same concern as the inspector and required further development of the root cause to evaluate how and why the workers could deviate from a procedure by marking a step not applicable if they did not fully understand the purpose of the step. The remaining deficiencies, including the inadequate alarm response by the operators, were adequately addressed by the initial II conclusions. The inspector concluded the final II conclusions were good.

No violations or deviations were identified.

6.0 MAINTENANCE (62703) (92902)

6.1 Ice Condenser Work

6.1.1 Basket Weighing And Ice Addition

In response to the August 24 inadvertent opening of one ice condenser lower inlet door and consequent ice melt documented in NRC IR 50-390,391/95-60, the applicant has undertaken a major initiative to weigh all 1944 ice baskets and add ice where required. The inspector reviewed the data from the weighing, which was approximately 90 percent complete, and verified all baskets were well above the technical specification minimum of 1214 pounds of ice. The applicant identified approximately 300 baskets which were below their administrative limit of 1425 pounds and was adding ice to these. The inspector reviewed Surveillance Instruction 1-SI-61-2, 9 Month Ice Weighing, Revision 0, for adequacy and verified the applicant was following precautions, periodically calibrating test equipment, and correctly calculating ice weights. The inspector observed various portions of the ice weighing and ice drilling of the baskets with a thermal drill which was performed in accordance with Procedure MI-61.6, Ice Addition Procedure, Revision 1. Ice weights were directly put in from an electronic load cell into a vendor software program which the inspector concluded was efficient and eliminated a possible source of transcription error. The inspector noted that housekeeping in the ice condenser upper plenum was marginal and that several opened bays were not covered with plywood as required by Surveillance Instruction 1-SI-61-2, Section 3.Q. These observations were given to maintenance management for correction. Several air handling units were frosting up quickly, causing drain pans to overflow when defrosting. This was apparently due to the excessive humidity generated by the work and constant breaching of the ice condenser. The inspector will evaluate that previous problems experienced with overflow and ice accumulation are corrected when the work is complete and the ice condenser atmosphere is allowed to reach equilibrium. The inspector observed several baskets that were frozen in place due to ice accumulation and could not be weighed. The applicant was attempting to free these baskets but had not done so by the end of the report period and was considering the option of statistically evaluating overall basket weight. The inspector is monitoring their evaluation and will verify their final actions are adequate. The inspector also observed cooldown of the ice blow line and ice transfer performed by Operations per Procedure SOI-61.02, Ice Charging System, Revision 6. Several equipment problems with the ice blower and chiller systems were encountered when preparing to blow the ice. These were corrected through the normal work process. The inspector interviewed the Technical Support system engineers overseeing the work. The inspector did not identify any notable concerns and concluded the applicant was adequately ensuring the ice condenser met technical specification weight requirements.

6.1.2 Dropped Ice Basket

A welded hook fractured on September 12 causing an ice condenser basket that was being weighed to drop approximately 10 inches. The bottom of the basket mesh was deformed from the fall. The hook was part of the rigging used to latch each ice condenser ice basket. The applicant initiated a DN against Surveillance Instruction 1-SI-61-2, Nine Month Ice Weighing, Revision 0, and initiated PER WBPER950558 for corrective action. The PER identified that the hook had not been qualified and tested as a lifting or rigging device. The inspector verified the applicant was using latching devices which were modified to a one piece design which eliminated the welded hook. The applicant planned to remove the dropped basket and replace the damaged basket section with a Unit 2 basket. This was not complete at the end of the report period. The inspector did not identify any significant concerns with the applicant's actions or response.

6.2 EDG Crab Nuts Overtorqued

The applicant identified the hydraulic torque wrench used on the head crab nuts on three of the four site emergency diesels was out of allowable calibration tolerance. The applicant generated PER WBPER950556 without any prompting to address the potential overtorque of the nuts. After consulting with the diesel vendor, the applicant performed a conservative action and retorqued all three diesels even though breakaway torque values on the first diesel were lower than expected and deemed acceptable by the vendor. The inspector reviewed the applicant's corrective actions and the vendor's recommendations. The inspector verified the calibration problem was traced to the applicant's central laboratory and was corrected so that common mode failure was addressed by using different wrenches on each train of equipment for the retorquing. The inspector witnessed portions of the retorquing and did not identify any problems with the work. However, the inspector did observe that attention to detail in the WO was poor. Several minor entries and signatures were not completed. The inspector concluded the applicant's actions in response to the head crab nuts were appropriate.

6.3 1B-B Motor-Driven AFW Pump Run

The inspector observed a run of the 1B-B MDAFW pump on September 14 to investigate vibration problems at low flows. Work had been performed to remove the flapper from pump discharge check valve 1-CKV-3-821 and to install shims in the first rigid discharge pipe support. The inspector noted that the pump run had to be delayed numerous times due to coordination with other work and concluded scheduling was poor. The pump ran well at various flow levels although it had to be prematurely secured due to a pump packing question. The inspector observed that visual vibration was acceptable and test data indicated notable improvement over previous tests. The applicant was processing a DCN to justify permanent removal of the check valve flappers. The inspector will review the DCN for adequacy when it is issued.

6.4 Reactor Head Vent Valve Stroke

The inspector observed stroking of Valve 1-68-396 open and closed after extensive work had been performed on the controller. Previous problems with this valve were documented in paragraph 3.9 of NRC IR 50-390,391/95-60. The inspector observed that the valve moved smoothly and concluded previous problems with abrupt and hesitant travel had been corrected. Some minor demand versus indicated position deviation was seen which was within allowable tolerances. The inspector reviewed as-found and as-left controller calibration data and discussed the controller problems with system engineers. The applicant obtained vendor support, utilized a vendor stroke meter test rig, and adjusted controller gain to fix the problem. The inspector concluded the applicant had effectively calibrated the valve controller. The applicant will stroke the head vent valves again during performance of Surveillance Instructions 1-SI-68-905A and B when the RCS is filled and vented.

6.5 RCS Check Valves

The inspector observed portions of the work on several more RCS check valves that had unacceptable leakage during HFT 2 testing. Work was observed at various stages of disassembly, lapping, leak testing, and reassembly. Previous observations were documented in NRC IR 50-390,391/95-60. The applicant completed work on the valves on September 26. The inspector verified cleanliness, foreign material exclusion, and quality inspection requirements were satisfied. Results were similar to previously observed valves in that slight seat warpage was found on most valves. The inspector observed that the applicant's quality assurance group frequently monitored performance of the work although their assessments documented in reports NA-WB-95-0167 and 0172 did not identify several of the discrepancies documented in IR 50-390,391/95-60. The inspector did not identify any discrepancies with the work observed in this report period.

The inspector reviewed the work documents and observed a final "information only" test the applicant performed to validate their work on the valve seats. A bonnet pressurization rig and pipe plug were fabricated to allow the applicant to pressurize the cold leg safety injection lines to 600 psi pressure for 10 minutes through each loop's primary injection line check valve. This tested for leakage by eight of the valves that had been worked as well as several others in the test boundary. The inspector observed that the applicant experienced difficulty during performance of the test. The pipe plug, which prevented flow into the reactor vessel, had to be redesigned several times to eliminate leaks. It also became stuck in the first valve the test was performed and had to be removed with a shaft knocker. The inspector was concerned that the initial actions by the plant mechanics to remove the stuck plug were not evaluated or controlled by management. However, maintenance management established control and suspended their actions without any prompting by the inspector until the knocker solution was agreed upon. The inspector relayed this concern to maintenance management and did not have any concerns with the actions that had taken place. The applicant changed the test during the review process to ensure borated water was used to fill the lines for the test. Since these are safety injection lines that would not be expected to be flushed, the inspector concluded this was an example of good reactivity awareness by the applicant. The inspector identified that the test work order did not specify any torque values for the bonnet test rig. The torque values used were discussed with the test engineer and verified allowable for the primary injection line check valves by the inspector. Results of the test were excellent. Two loops did not exhibit any leakage while the worst of the other two loops was only one gallon in 10 minutes. The inspector concluded the applicant had successfully repaired the leaking valves identified during HFT 2 and validated their results to the maximum extent practical by performing post work tests. Official post-maintenance testing will be accomplished by existing surveillance instructions which are planned as part of plant startup activities.

The applicant had initiated WBPER950488 to assess the root cause of the multitude of check valve problems. While not finalized at the end of the report period, the applicant's preliminary conclusion identified welding practices as the cause for the deformed seats. Previous work on the valves had not created further problems but had failed to correct the seat leakage. The applicant gathered industry data on the valves as well as information from the valve vendor representatives to reinforce their conclusion. The inspector reviewed the applicant's conclusion and did not identify any problems.

6.6 PMTI-35764-1, Diesel Driven Fire Pump Acceptance Test, Revision 0

PMTI-35764-1 requires functional testing of Diesel Drive Fire Pump 0-PMP-026-3150 and associated equipment. This diesel-driven fire pump was recently installed to supplement the available capacity and reliability of the electrical-driven fire pumps.

The scope and objectives of the test procedure included verification that:

- status and trouble alarms generated by the diesel fire pump controller properly indicate on Fire Detection Panel 0-M-29;

- fire pump engine can be stated manually and automatically from the designated local and remote pump start locations;

- fire pump engine can be stopped manually and automatically from the designated manual and automatic stop locations;

- fire pump can provide the required flow at the specified pressure (e.g., pump rating of 2500 gpm at 125 psi; shutoff head between 101 and 140 percent of rated head with no flow; and maximum flow of 3750 gpm between 65 and 100 percent of rated head);
- fire pump house sump pump and HVAC equipment are operable; and
- piping system meets cleanliness requirements.

The applicant's commitments concerning preoperational testing of the plant's fire protection water systems, including fire pumps, are described in FSAR Table 14.2-1, Sheets 13 and 14. The WBN Fire Protection Report, Revision 4, provides a description of the function, design criteria, and requirements for the diesel-driven fire pump. Procedure SSP-8.03, Post Modification Testing, provides requirements for the test of components installed subsequent to the completion of WBN's pre-operational test program.

The inspector reviewed Procedure PMTI-35764-1 and noted that the procedure had received the appropriate technical and management review, identified the required test scope and objectives, contained appropriate acceptance criteria, and required the documentation of sufficient information to permit adequate evaluation of the test results. The test procedure appeared to meet the applicant's requirements for post modification testing.

Within the areas examined, no violations or deviations were identified.

7.0 PLANT SUPPORT (71750)

7.1 Emergency Preparedness Drill

The inspector observed the applicant conduct a site emergency drill on September 13, 1995. The inspector focused primarily on OSC performance due to problems with radiological controls delaying team actions observed during previous drills. The inspector observed that one team took 1 hour and 19 minutes to get dispatched from the OSC to perform a simple task of verifying an AFW pump was not seized. The problems that led to delays included repeated changes to the scope of the team's job priorities, computer data base dosimetry access problems, no plant key sets available in the OSC, and several expired breathing apparatus fit tests. As the scope of the team's activities expanded, the inspector did not see that proper consideration of ALARA principles was taken. Assessments of allowable stay times in breathing apparatus were also not performed. The inspector concluded this was unacceptable performance in response to an emergency condition and discussed this with emergency preparedness management. The plant manager observed several of the same deficiencies and reached a similar conclusion. The applicant's emergency preparedness group has met with applicable members of the drill team and aggressively pursued performance improvements by conducting small scale OSC training sessions. Another site drill was conducted on October 4 and the applicant reported notable improvement in OSC performance. The inspectors will evaluate the applicant's OSC performance during a full participation drill scheduled for November.

7.2 Security

On October 3, 1995, the inspector observed an applicant's truck that was left idling without anyone in attendance in the protected area adjacent to the turbine building. This was contrary to the security requirements that all vehicles in the protected area be turned off and the keys removed when unattended. The applicant had implemented a full security lockdown so all requirements were in effect and were expected to be enforced. The inspector questioned the driver of the truck when he returned, and he indicated he was unaware of the requirement. The inspector was aware that the applicant previously had thoroughly promulgated these requirements through memorandums and bulletin board announcements in anticipation of the security lockdown. The inspector informed the security shift supervisor who initiated a security event investigation which will implement corrective action.

No violations or deviations were identified.

8.0 MAINTENANCE INSPECTION OPEN ITEMS (92902)

NRC IR 50-390,391/95-202 identified three weaknesses and 11 numbered deficiencies. Six of the deficiencies were subsequently categorized as examples of failure to follow procedures and VIO 50-390/95-202-01 was issued in a later Notice of Violation. The applicant responded to the items in a letter dated August 16, 1995, and agreed with each item. The inspector evaluated the adequacy of the applicant's corrective action and verified the implementation for each item as documented below.

The applicant has undertaken several initiatives to improve maintenance performance. Deficiencies similar to the inspection findings with work order content, planning, and procedural adherence were previously identified by the applicant's nuclear assurance group in assessments such as a maintenance performance evaluation done in May 1995. The inspector reviewed this report, NA-WB-95-0086, and verified the applicant was identifying deficiencies and implementing corrective action. The inspector attended a meeting on October 3 between the maintenance and nuclear assurance groups and concluded the nuclear assurance group was adequately assessing maintenance performance in many areas and the maintenance department was responsive to their findings. Preliminary plans were discussed for another overall maintenance assessment and a side-by-side training session where nuclear assurance inspectors would work with maintenance planners for two weeks and improve planner performance. The inspector concluded both of these efforts were proactive and would be beneficial. Additionally, the applicant generated PER WBPER950501 on August 16 to evaluate and correct overall maintenance performance problems which appeared to be a negative trend. The inspector evaluated the closure of this corrective action document and did not identify any discrepancies.

8.1 Identified Weaknesses

Each of the three identified weaknesses are addressed separately below.

8.1.1 (Closed) Maintenance Planning Work Order Preparation

This weakness involved work orders that did not specify which steps in a procedure to use and which specified test acceptance criteria without referring to approved documents that contained the acceptance criteria. The inspector verified enhancements to planner training and references were implemented. The applicant performed a job task analysis for maintenance planners to identify needs in training. The inspector also verified the applicant has implemented a temporary technical review of all safety-related work orders. Reviews by the inspectors and the applicant's Nuclear Assurance group have verified work order preparation has improved. Based on the adequacy of these corrective actions and the ongoing effort by the applicant to improve the planning process, this weakness is considered closed.

8.1.2 (Closed) Maintenance Data Base Information Insufficient for Trending

This weakness involved the maintenance history and trending program. Procedures were deemed adequate, but the maintenance planning and control data base did not contain adequate information to be useful for trending purposes, and the applicant's staff needed training in the use of the data base. Training in the maintenance planning and control system was identified and initiated as part of planner job task skill assessments. The inspector concluded that information in the data base will be accumulated as plant operation commences, which will facilitate trending analysis. Therefore, this item is considered closed.

8.1.3 (Closed) Summary of Causes for 11 Deficiencies

Four common causes were attributed to the 11 deficiencies and recognized as a weakness. Each of the causes is addressed separately below:

- Inadequate Procedures

  The applicant corrected each identified deficiency as part of their corrective action. None of the items were found by the applicant to create an extent of condition beyond the specific deficiency. The inspector agreed with this assessment and observed that maintenance procedure upgrades and other improvement initiatives were being taken. The inspector concluded the applicant's actions were adequate, and this item is considered closed.

- Not Following Procedures

  The inspector verified the applicant developed lessons learned summaries and addressed procedural adherence in appropriate training sessions. Specific non-compliances were corrected as part of the individual deficiency response. The inspector has not observed any notable procedural adherence problems since the problem was identified. The applicant adequately addressed the problem so this item is considered closed.

- Poor Completed Work Order Reviews

  The applicant has applied considerable resources to correct this problem. Ownership of package quality by the work supervisors has been reinforced and separate levels of review have been eliminated to enhance accountability. The applicant's nuclear assurance quality review organization has been performing reviews of completed work orders to assess the adequacy of post-work reviews. The inspector verified that results of these reviews have been very good. Based on the adequacy of the applicant's corrective actions and ongoing initiatives, this item is closed.

- Lack of Design Output

  The applicant developed and administered a training session on design output to all applicable engineers and planners. The inspector verified applicable personnel were trained and attended one of the sessions to evaluate its content. Although the lesson plan was simple and short, the inspector concluded it adequately promulgated the basics of when engineering output was required and nuclear engineering should be consulted. The inspector observed that the course instructors actively solicited questions from the attendees. The inspector concluded the applicant's action adequately addressed the original concern; this item is considered closed.

8.2 (Closed) VIO 50-390/95-202-01, Examples of Failure to Accomplish Activities in Accordance with Documented Procedures.

Each of the six deficiencies that constitute the violation examples are addressed separately as follows:

8.2.1 (Closed) Deficiency 95-202-01, Calibration of Standby Diesel Generator Speed Sensor Relays Without Supporting Design Documents

This deficiency involved the change of calibration setpoint acceptance criteria when the originally specified criteria could not be obtained during testing. The inspector verified that the applicant revised Procedure SSP-6.03, Preventive Maintenance Program, Revision 9, to require design output documentation when verifying safety-related parameters. The applicant revised the portion of the procedure delineating requirements for preparation of preventive maintenance packages. The inspector observed that the applicant conducted training for appropriate personnel on the use of design output. This is documented above as part of their response to identified weaknesses. The inspector verified the applicant obtained correct setpoint tolerances from the diesel vendor and incorporated them in Vendor Manual VTM-P318-0120, Revision 8. Based on the adequacy of these corrective actions, this item is considered closed.

8.2.2 (Closed) Deficiency 95-202-04, Modifications to MOVs Performed Without Supporting Design Documents

This item involved the grinding of valve housings to provide tripper finger clearances without processing design change approval. The applicant also attributed this deficiency to inadequate guidance on the use of design output and addressed the problem in training documented above in response to identified weaknesses. The inspector verified the applicant revised Procedure SSP-6.02, Maintenance Management System, Revision 15, to provide guidance to maintenance personnel on the required uses of design output. The valve vendor recommended a modification to correct the tripper finger clearance problem. The inspector verified the applicant revised Procedure MI-0.16.02, Limitorque Motor Operator Repair and Adjustment Guidelines For SMB-00, Revision 9, on September 8 to include the modification on existing valves on an as-maintained basis. The applicant also coordinated with the valve vendor on the issuance of a 10 CFR Part 21 report which was issued on July 12, 1995. Based on the adequacy of these actions, this item is considered closed.

8.2.3 (Closed) Deficiency 95-202-05, Inadequate Procedure MI-1.003, Disassembly, Inspection and Reassembly of Auxiliary Feedwater Pump Turbine, revision 2.

This item involved vendor manual guidance for the turbine-driven AFW pump oil sightglass level mark which was not incorporated into maintenance instructions and was not on the installed pump sightglass. The inspector verified the applicant incorporated the level markings into Procedure MI-1.003 and correctly marked the turbine sightglass. The applicant discussed the deficiency with mechanical planners and emphasized referring to vendor technical manuals for requirements when replacing components. The applicant did a walkdown on 18 other pumps and motors and verified vendor requirements were properly incorporated. The applicant has also initiated other maintenance program improvements as documented above in response to NRC identified weaknesses. Based upon these corrective actions, this item is considered closed.

8.2.4 (Closed) Deficiency 95-202-07, Inadequate Work Instruction in WO 94-04878-00

This item involved corrective maintenance of a 6900V circuit breaker. The specified tripping time acceptance criteria (54 ms) did not agree with the tripping time (50 ms) specified in Q-DCN 22832-A. However, the recorded test value (47 ms) was within the requirement. The applicant attributed the discrepancy to a personnel transposition error. The applicant reviewed 178 work orders that performed the Q-DCN time test and identified 17 that contained the same error due to copying the instructions containing the original error. The applicant verified these 17 breakers also were within the required 50 ms acceptance criteria. The applicant reviewed attention to detail and the details of this problem with maintenance planners. The inspector concluded the applicant's actions were thorough and appropriate. This item is considered closed.

8.2.5 (Closed) Deficiency 95-202-10, Failure of Maintenance Personnel to Issue a PER in Accordance with SSP-3.04

This item involved problems encountered during replacement of pump seals on Centrifugal Charging Pump 1B-B. The new parts were insufficient, and old parts had to be refurbished. The procedure described two possible seal configurations, but not the actual configuration which was a hybrid of the two. Maintenance personnel successfully assembled a hybrid seal package and installed it in the pump but did not record the problem on a PER for lasting corrective action. The applicant attributed the deficiency to personnel error. The inspector verified training was given to maintenance personnel on Procedure SSP-6.01, Conduct of Maintenance, Revision 3, feedback process and management reinforced expectations for use of the corrective action program and generation of PERs. The inspector observed that maintenance personnel generated PERs in response to EDG crab nut problems and an ice condenser rigging failure which are documented under maintenance in this report. These PERs were immediately initiated by the maintenance department, without prompting indicating a proper threshold for PER generation. These were examples of the heightened awareness of corrective action responsibilities the inspector has observed in maintenance personnel and verified in internal maintenance newsletters and site bulletins. The inspector verified that the applicant developed and administered a new training course, MTS343.004 - Corrective Action Program, to maintenance engineers, supervisors, and planners. The inspector reviewed the lesson plan and noted that it used each of the identified maintenance inspection deficiencies as examples. The inspector concluded the lesson plan was thorough and effective. The inspector verified that the applicant reviewed and clarified the required parts needed in a kit for a CCP mechanical seal replacement. The inspector verified Procedure MI-62.001, Centrifugal Charging Pump, Revision 16, was revised to reflect the actual seal configuration. Based upon the extent and adequacy of the applicant's corrective actions and the heightened maintenance department corrective action sensitivity the inspector has observed, this item is considered closed.

8.2.6 (Closed) Deficiency 95-202-11, Failure to Perform an Equivalency Evaluation in Accordance with SSP-10.05

This item involved vital inverter fans which were replaced without evaluating and documenting that the new fans met the original design basis. The applicant purchased the new fans as nonsafety-related, although the original fans were qualified as part of the original safety-related inverter installation. Section 2.3.3.A of Procedure SSP-10.05, Technical Evaluation for Procurement of Materials and Services, Revision 12, states that for spare and replacement parts, an equivalency evaluation must be performed and documented to ensure that such items were purchased to requirements equivalent to those specified for the original equipment. The applicant attributed the deficiency to personnel error in that an evaluation was performed by a procurement engineer to classify the fans, but the engineer did not indicate if a technical evaluation was performed or justify the conclusion. The inspector verified the applicant placed existing warehouse fan stock on hold until an equivalency evaluation was performed to dedicate the installed and in stock fans. The inspector verified the applicant conducted training for procurement engineers on technical evaluations and determination of safety classification for sub-components on a safety-related component. The inspector reviewed the training memorandum and concluded it was not very detailed but that it adequately addressed the deficiency. The applicant's procurement group reviewed a large sample of similar procurement packages involving non-safety sub-components on safety-related components to verify an adequate basis for justification existed. They identified one additional example, although it was determined not to be a discrepant item, and the applicant's response letter stated no new defective items were found. The inspector reviewed the actions for the identified example and determined it was not related to the original deficiency, and it had been corrected several years ago. The inspector verified the scope of the review and concluded it was a thorough extent of condition assessment. Based on the adequacy of these actions, this item is considered closed.

Based upon the adequate corrective action for each of the six deficiencies that constituted the violation examples, VIO 50-390/95-202-01 is considered closed.

8.3 (Closed) Deficiency 50-390/95-202-02, Inadequate Review and Closeout of WO 95-10065-10

This item involved the closure of a work order which calibrated a neutral overvoltage relay to values specified by the relay setting sheet. The completed calibration data did not meet the acceptance criteria of the setting sheet. The applicant agreed the deficiency occurred and attributed it to providing inadequate guidance to the electrical switchyard customer group for resolving setting sheet problems. The applicant claimed the disparity was resolved during testing by an engineering evaluation but the setting sheet discrepancy did not get addressed. The inspector verified the applicant revised the Customer Group Field Test Manual - Volume I (revision date July 31, 1995) to clarify acceptance criteria and problem reporting requirements. The applicant sampled 58 similar relay calibration work orders and verified they met acceptance criteria. Procedural requirements were discussed and reinforced with the customer group personnel. Based on the adequacy of these actions and the scope of the applicant's extent of condition review, this item is considered closed.

8.4 (Open) Deficiency 50-390/95-202-03, Inadequate Procedure SSP-12.06, Verification Program

Procedure SSP 12.06, Verification Program, Revision 3, was deficient because Appendix A, which delineated systems requiring second party or independent verification, did not specifically list all safety-related electrical systems. The inspector verified the applicant processed a change to the procedure which eliminated a vague reference to the Class 1E Electrical Distribution System and added the specific missing safety-related electrical systems. However, the inspector identified that Appendix A did not include several other mechanical safety-related systems. The inspector discussed this deficiency with Operations management who are responsible for defining the content of Appendix A. They initiated an evaluation which identified that further changes to Procedure SSP-12.06 were required. These changes will require further revisions to applicable system procedures such as Maintenance Instructions and Surveillance Instructions to ensure independent verification guidance is correct. This item will remain open pending the inspector's evaluations of the applicant's final corrective action.

8.5 (Closed) Deficiency 50-390/95-202-06, Failure to Revise, Review, and Approve Procedure MI-3.012, Removal, Inspection, and Replacement of Turbine Driven Auxiliary Feedwater Pump Rotor, Revision 0

This item involved an example where the craft failed to process a procedure change when deviating from the requirements of Procedure MI-3.012. A step required heating a bearing with a rosebud torch but a bearing heater was used instead. The discrepancy was not recognized during the work package closure review process. The applicant counselled the involved individuals on their responsibility to process procedure changes when appropriate. The inspector verified the applicant revised Procedure MI-3.012 to allow other heating devices to be used. The inspector concluded this was acceptable and would provide the craft with greater flexibility and prevent recurrence. The applicant also reviewed 10 percent of the previous year's work items performed by the same crew. No procedural adherence discrepancies were found. Based on the applicant's actions, this item is considered closed.

8.6 (Closed) Deficiency 50-390/95-202-08, Inadequate Review and Closeout of WO 95-04731-00

This item involved calibration of a temperature loop which required HERS data sheets to be completed. The HERS data sheets for the loop were not in the work order package. The applicant concluded the HERS data sheets for two similar loops were inadvertently interchanged during assembly of the completed packages. The applicant discussed the error with the involved individual and the inspector verified the HERS sheets were replaced in the packages. The applicant reviewed 23 similar work packages and found no other discrepancies. The inspector verified the applicant was performing further, ongoing reviews of completed work packages. The inspector concluded the applicant fully addressed the scope of the deficiency. This item is considered closed.

8.7 (Open) Deficiency 50-390/95-202-09, Inadequate Review and Closeout of WO 95-04740-07

This item involved replacement of an ERCW supply header flow scale to implement a design change. The values generated by one of the three input signals used in calibration of the flow indicator before and after replacement were outside tolerance limits. The foreman's and general foreman's review of the completed work packages failed to detect the error. The applicant attributed the discrepancy to personnel error and discussed the importance of package quality with the culpable individuals. The flow instrument was correctly recalibrated, and other scale calibration work for the same system was verified to be correct. The applicant's quality engineering group sampled similar work packages as part of ongoing corrective actions for identified weaknesses noted above. They found only minor administrative errors. However, the applicant's response did not specifically address how and why the input value was altered. The inspector considered this information critical for determining a root cause and preventing recurrence. The inspector obtained a copy of PER WBPER950314 which the applicant used to perform corrective action for the original problem. It also did not identify the altering of the input value other than attributing it to an isolated personnel error. Therefore, this item will remain open pending the inspector's verification that the root cause was sufficiently addressed.

No violations or deviations were identified.

9.0 OPERATING PROCEDURES REVIEWS (42450)

9.1 Documentation Review

In conjunction with the walkdown and review of the safety-related systems which are discussed in paragraph 11.0 of this report, the inspectors reviewed and performed a field validation of the SOIs that the applicant had implemented to operate the turned over systems. The inspectors reviewed the SOIs to evaluate the applicant's progress in developing and implementing operating procedures which would be used to control safety-related operations.

Requirements are delineated in 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings; ANSI N18.7-1976/ANS-3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, Section 5; RG 1.33, Revision 2, February 1978; TVA Nuclear Quality Assurance Plan (TVA-NQA-PLN89A), Revision 5; and selected applicant administrative procedures.

The inspectors interviewed cognizant applicant personnel and also reviewed appropriate applicant administrative procedures, vendor technical manuals, drawings, design basis document system description, and the FSAR.

During the review, the inspectors considered the following procedure attributes:

- the procedure was technically adequate to accomplish the stated purpose;
- applicable operating limits were clearly specified;
- precautions and limitations were included which prescribed activities important to the protection of the health and safety of the public and plant equipment; and
- the procedure was in the appropriate format as specified in the applicant's administrative controls.

The procedures reviewed were:

- SOI-30.06, Auxiliary Building Gas Treatment System, Revision 12
- SOI-32-02, Auxiliary Air System, Revision 12

9.2 Procedure Review Findings

The inspector discussed the following comments related to the review of Procedure SOI-30.06 with the applicant:

- Sections 8.2 [7] [a], [b], [c], and [d] should have referred to AB GEN Supply Fan, rather than AB GEN Exhaust Fan.

- The locations provided in Appendix A for some of the dampers were not accurate. Examples of this are:

  The location for damper 0-FCO-30-129 was given as A6W/729. The actual location was closer to A8X/729.

  The location for damper 0-FCO-30-130 was given as A6W/729. The actual location was closer to A8X/729.

  The location for damper 0-FCO-30-122 was given as A6W/729. The actual location was closer to A5X/744.

  The location for damper 0-FCO-30-123 was given as A6W/729. The actual location was closer to A5X/744.

  The location for dampers 0-FCO-30-137 and 0-FCO-138 was given as A6T/782. The actual location was closer to A6T/800.

  The location for dampers 0-FCO-30-140 and 0-FCO-141 was given as A10T/782. The actual location was closer to A10T/800.

- The location provided in checklist 1 for breaker 0-BKR-30-147 was cubicle 10D in 480V C&A Building Vent Board 1A1-A. The actual location of the breaker was cubicle 10D of 480V C&A Building Vent Board 2A1-A.

- The nomenclature provided in Checklist 1 for 1-BKR-235-2/19 was GAS TREATMENT FOR B-B PANEL 0-L-429. The unique identification label on the panel read GAS TREATMENT FAN B-B PANEL 0-L-428.

- In Checklist 2 the System 32 air valves, which provided operating air to the dampers identified in the auxiliary building elevation 737 table were not listed. The applicant normally included the System 32 air valve to a component in the checklists to ensure it was in the correct position when a system was lined up for operation.

The applicant stated that these comments would be reviewed and appropriate changes to the SOI would be made. In addition, the applicant committed to review and correct as necessary the other SOIs for System 30, Heating and Ventilation.

9.3 Conclusion

The inspectors concluded that the SOIs provided appropriate instructions for operating the safety related systems.

Within the areas examined, no violations or deviations were identified.

10.0 SYSTEM TURNOVER (37301)

10.1 System Walkdown and Review

The inspectors performed reviews of the ACAS and ABGTS which had been turned over from the startup test group to plant operations. The inspectors performed the system reviews to verify that the applicant's system turnover process was effective in ensuring safety-related systems being turned over to Operations were in a reasonable condition of completion to support plant Operations.

The applicant had developed and implemented Procedure PAI-5.01, System Pre-Operability Checklist, Revision 7, which delineated a systematic method for ensuring all open work items and outstanding programmatic items which could affect system operability or the operational readiness of a system to support fuel load/startup were completed or dispositioned before the system was turned over to Operations. The turnover process was referred to as SPOC (system pre-operability checklist) at WBN.

The inspectors performed the following activities:

- performed a field walkdown of selected portions of the ACAS and ABGT;

- reviewed the SPOC turnover package for the ACAS and ABGT; and

- reviewed the applicant's MTS, which documented and tracked open items for the ACAS and ABGTS.

The inspectors considered the following items during the system walkdown and review:

- deficiencies such as damaged or missing components, trash or foreign material in cabinets, and the quality of workmanship;

- Unit 1/Unit 2 system interface points had been identified and control had been established;

- open items listed on the MTS that were being turned over that could impact system operability or readiness were being adequately addressed by the applicant; and

- component labeling numbers and nomenclature reasonably matched component identifications provided on the drawings and in the SOIs.

10.2 Walkdown and Review Findings

During the ACAS walkdown the inspector noted that the solenoid operated cooling water inlet valve, pressure control valve, and temperature control valve on each air compressor had new CIDs that identified these valves as part of System 67 (ERCW). These valves had previously been identified as part of

System 32 (ACAS). The inspector verified that the applicable operating, surveillance, and maintenance instructions had been updated to reflect the new CIDs. The inspector conducted a review of the compressed air system active valves listed in DBD system description for compressed air system, N3-32-4002, and determined that these valves were still identified as System 32 (ACAS) valves. The inspector then reviewed the DCN that had implemented the CID changes (DCN S36569, Revision A). As a result of these reviews, the following concerns were identified:

- DCN S36569 had not required updating of the system description DBDs to reflect the valves removed from System 32 and added to System 67.

- The following valves were still identified as active in the active valves listed in the DBD system description for System 32 (ACAS) and were not listed under their new CID in the DBD system description for System 67 (ERCW):

Old CID	New CID	
0-FSV-32-61	0-FSV-67-1221	ERCW supply to aux air comp A
0-FSV-32-87	0-FSV-67-1223	ERCW supply to aux air comp B
0-PCV-32-68	0-PCV-67-1222	ERCW supply press cont reg vlv to aux air comp A
0-PCV-32-98	0-PCV-67-1224	ERCW supply press control reg vlv to aux air comp B
0-TCV-32-68A	0-TCV-67-1222A	reg throttle ERCW to aux air comp A
0-TCV-32-68B	0-TCV-67-1222B	reg throttle ERCW to aux air comp A
0-TCV-32-98A	0-TCV-67-1224A	reg throttle ERCW to aux air comp B
0-TCV-32-98B	0-TCV-67-1224B	reg throttle ERCW to aux air comp B

- A separate concern was identified with the PCV and TCV valves listed above. These valves were listed as motor actuated on the active valve list. These valves are not motor actuated. The PCVs listed are manually adjusted, self-actuating, pressure control valves. The TCVs listed are thermally controlled, self-actuating, temperature control devices. The applicant agreed with the inspector's finding. DCN S38221 will be issued to correct the description of these valves in the DBD system description and other documents, as applicable. This concern is considered resolved.

The inspector queried the applicant as to why the DBD system descriptions had not been updated to reflect the CID changes as required by Procedure EAI-3.05, Design Change Notices, Revision 28.

The applicant indicated that a "limited variance" had been approved for Procedure EAI-3.05 (variance T25 93 1230830) that authorized an exception to the requirement that all affected engineering controlled documents be updated as part of the S-DCN process. As a result of this limited variance, S-DCNs would only require update of Category 1 (primary and critical) drawings and the EMS. Other documents such as design criteria, system descriptions, essential calculations, FSAR, and secondary drawings would not be updated. However, the tie between the old and new CID could be found in EMS. EMS had been modified such that when an old CID was entered, the new CID would appear with a message indicating the CID had been changed.

Further, the issue of not updating engineering documents for new CIDs had been identified by the applicant during the PAC/AQ program by PACR-0425. As a resolution to PACR-0425 the applicant had committed to the following actions to update/revise other documents affected by the CID changes during the time variance T25 93 1230830 was active:

- Secondary Drawing: Corporate project "Design Change Process Automation" will, at some point (approximately 18 to 24 months), provide an electronic link between CIDs on CADAM drawings and EMS. Any CID inconsistencies between the secondary drawings and EMS will be reconciled at that time.

- EQ Binders: There have been no changes to CIDs for 50.49 components. Any potential changes will be reviewed and approved by the EQ Program Manager prior to DCN issuance to insure impacts to EQ binders have been addressed.

- Calculations: Lists of CIDs for which the unit designator has been changed have been provided for update of the Unit 2 for Unit 1 calculation. Data Systems will continue to provide updates of this list until the tagging program is complete. The Unit 2 for Unit 1 calculations will be revised to reflect these changes prior to fuel load. There is no plan to address other calculations.

- Vendor Manuals: Vendor manuals are being reconciled with EMS prior to revision, therefore, all future revisions will reflect CID changes. The long-range plan is to identify the applicable VM with each CID in EMS and do away with the applicability lists in the VMs.

- System Descriptions/Design Criteria: Lists of CID changes by system have been provided to the discipline lead engineers for incorporation, as required, in future revision of the SD/DCs. A final list will be provided when the tagging program is complete.

- FSAR/Technical Specifications: The final list of CID changes by system will be utilized to revise any CIDs that may be embedded in the text of the FSAR/Technical Specifications, as required, during the first update cycle.

The applicant stated that after system 90 (Radiation Monitoring) completes the SPOC process, limited variance T25 93 1230830 would be cancelled. At that time, subsequent S-DCNs, including those involving CID changes, would require all affected documents to be updated at the time of the S-DCN. The System Description/Design Criteria would be updated with any changes required as a result of the CID changes during the next revision of the System Description or during the applicant's "System of the Month" review, whichever occurred first.

The inspector reviewed a sampling of the CID changes that had been identified as requiring a change in EMS as of May 20, 1995, for Systems 03 (auxiliary feedwater), 18 (fuel oil), 68 (RCS), 72 (containment spray), and 74 (RHR) in order to evaluate the categories of CID changes being implemented. In general, CID changes had been identified for all types of components and sub-components that are found within these systems. Examples are pressure switches, valves, cables, snubbers, flow elements, hand switches, heaters, motors, pumps, indicators, etc.). Each specific CID change involved a change to one or more of the following items within a CID: unit, function code, system number, component address, component sub-address, train/channel or flow path, and noun description of the component. Although the identified EMS CID changes were numerous for those systems reviewed (approximately 1000), the majority involved very minor changes that would be transparent to the System Description/Design Criteria and FSAR. For example, the majority of the changes reviewed involved minor CID changes for cables and various types of instrument valves which would have significance for the category 1 drawings and various operational procedures, but not for the System Description/Design Criteria and FSAR.

The inspector determined that while some of the CID changes would require changes to the System Description/Design Criteria and FSAR, such as those identified in System 32, the majority of the changes would not. The applicant's plan for updating these documents appears to be reasonable and should not result in any adverse operational problems. This judgment was based on the fact that the category 1 drawings, operating procedures, surveillance procedures, maintenance procedures, and the EMS would be in agreement after completion of the labeling effort.

The inspector further discussed the process of using variances to change or deviate from procedural requirements. The use of variances was authorized by Procedure EAI-1.02, Preparation and Control of the EAI Manual, Revision 4. Variances are required to be reviewed by engineering personnel and approved by the engineering manager. Variances were classified as limited variance or process variance. Process variances were essentially permanent changes to the EAI procedure and were required to be incorporated in the next revision of the EAI procedure. Variances written against a specific EAI were source note referenced in the applicable EAI.

The inspector determined that active variances against a specific EAI were not incorporated into the text of the EAI or distributed with the controlled copies of the EAI. In order to determine what change or deviation each variance had authorized to an EAI, it was necessary to obtain the RIMS retrieval number from the source notes of the EAI, and then retrieve each variance individually in RIMS. The inspector was concerned that this process would make it difficult for a user of the EAI to readily determine the scope of the variances issued against the EAI.

The applicant agreed with the inspector's concern and stated that the variance process would be changed such that all variances against an EAI would be issued with the EAI such that the scope of the variances would be readily available to the EAI user. The applicant further stated that process variances would be deleted from the variance procedure as they represent permanent changes to an EAI which are done by EAI revision as a matter of practice and not by a process variance. Limited variances would still be utilized for specific issues requiring deviation from the EAI manual requirements.

The inspector concluded that the applicant's proposed actions would resolve the inspector's concern and considers this issue resolved.

10.3 Walkdown and Review Conclusions

Based on the review of the ACAS and ABGT SPOC turnover package and a walkdown of the systems, the inspectors concluded that the applicant's system turnover process was effective in ensuring systems being turned over to Operations had been adequately constructed to support plant operations and that open items were appropriately documented for correction.

Within the areas examined, no violations or deviations were identified.

11.0 MAINTENANCE PROCEDURES REVIEW (42451)

11.1 Documentation Review

The inspectors reviewed selected preventive maintenance instructions to evaluate the applicant's progress in developing and implementing procedures which would be used to perform preventive maintenance on safety-related equipment and components. Requirements are delineated in 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings; ANSI N18.7-1976/ANS-3.2, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, Section 5; RG 1.33, Revision 2, February 1978, Section 9 of Appendix A; TVA NQA Plan, Revision 3; Site Writers Manual, Revision 0; Writers Guide for Technical and Normal Operating Instructions, Revision 0; and selected applicant administrative procedures.

The inspectors interviewed cognizant applicant personnel and also reviewed appropriate applicant administrative procedures, vendor technical manuals, drawings, design basis document system descriptions, the FSAR, and the draft Technical Specifications.

The applicant had developed and implemented Site Standard Practice SSP-6.03, Preventive Maintenance Program, Revision 9. This procedure established the applicant's processes and related requirements for controlling preventive maintenance activities, including the predictive maintenance monitoring program.

During the review, the inspectors considered the following procedure attributes:

- the procedures were technically adequate to accomplish their stated purpose;

- where appropriate, the procedures included vendor technical manual recommendations for maintenance;

- the procedures were consistent with draft Technical Specifications and regulatory requirements;

- precautions and limitations were included which prescribed activities important to the protection of the health and safety of the public and plant equipment;

- the procedures were in the appropriate format as specified in the applicant's administrative controls.

- the procedures included a method for obtaining permission to work on equipment and for isolating the equipment.

The procedures reviewed were:

- 1-BD-211-B-B, File 06, Perform Thermography Test on 1-BD-211-B-B, Revision 0 (*)

- 1-BD-211-B-B, File 03, 6900V Shutdown Board 1B-B Bus Differential Relay Functional Test, Revision 0 (*)

- 1-BKR-211-1934/1-B, File 02, Calibration of AX934 and CX934 Relays for 1-BKR-211-1934/1-B, Revision 1 (*)

- 1-BKR-211-1934/1-B, File 01, 6900V Shutdown Board 1B-B Panel 1 Relay Functional Test, Revision 0 (*)

- 1-BKR-211-1914/6-B, File 02, 6900V Shutdown Board 1B-B Panel 6 Relay Functional Test, Revision 0 (*)

- 1-BKR-211-1726/11-B, File 01, 6900V Shutdown Board 1B-B Panel 11 Relay Functional Test, Revision 0 (*)

- 1364V, Direct Current HFA Relay Calibration, Revision 0

- 1-BKR-211-1914/6-B, File 01, Calibration of AX914 and CX914 Relays for 1-BKR-211-1914/6-B, Revision 0 (*)

- TI-31.07, Predictive Maintenance Thermography Program, Revision 3

- MI-57.020, 6900 Volt Switchgear Inspection, Revision 16

- MI-57.213, 6900V Shutdown Board 1B-B Indicating Instruments and Transducer Calibration, Revision 0

- 1-PMP-003-0001A-S, File 01, Lube Oil sample and Inspection of TD AUX Feedwater Pump 1A-S and Turbine 1A-S, Revision 2 (*)

- 1-TURB-001-0001A-S, File 01, Inspection, Lubrication, and, Testing of AUX FW Pump Turbine 1A-S, Revision 4 (*)

- 1-TURB-001-0001A-S, File 03, Remove and Inspect Governor Valve, Servo, and EGR, [Document reviewed was a preventive maintenance request form and it did not have a revision associated with it] (*)

- 1-SM-046-0057B-S, File 01, Replacement of Governor Valves' servo Valve and EGR Actuator Assembly, Revision 0 (*)

- 1-FCV-001-0052, File 01, Freedom of Movement Test for Aux Feedwater Pump GOV Valve, Revision 0 (*)

- MI-1.003, Disassembly, Inspection, and Reassembly of Auxiliary Feedwater Pump Turbine, Revision 2

- MI-3.012, Removal, Inspection, and Replacement of Turbine Driven Auxiliary Feedwater Pump Rotor, Revision 0

- MI-3.013, Inspection, Disassembly, and Repair or the Turbine Driven Auxiliary Feedwater Pump Rotating Element, Revision 0

- 0-COMP-032-0060, File 02, Annual Inspection of Auxiliary Control Air Compressor A-A, Revision 7 (*)

- 0-DRYR-032-0074, File 05, Replacement of Train A Auxiliary Air Dryer Afterfilter, Revision 0 (*)

  File 01, Monthly Inspection of Air Dryer Operating Cycle, Revision 1
  File 03, Inspection of Train A Auxiliary Control Air Dryers, Revision 4

- 0-MVOP-32-0082-A, File 01, Operator Diaphragm Replacement

The inspectors selected those procedures designated with an asterisk (*) and walked them down in the plant to validate the procedure against the as-constructed plant components referred to in the procedure.

11.2 Results

The inspector did not identify any concerns during the review of the preventive/predictive maintenance procedures. The inspector identified some editorial and comments during the review and discussed these with the applicant. The inspectors concluded that the applicant had implemented a preventive maintenance program and was developing and implementing procedures which could be used to perform preventive maintenance on safety-related equipment and components.

No violations or deviations were identified.

12.0 ACTIONS ON PREVIOUS INSPECTION REPORT CONCERNS

12.1 (Closed) Vital Battery Surveillance Comments (IR 50-390/94-52)

In IR 50-390,391/94-52, Attachment, Surveillance Procedure Review Comments, Section C.0, the inspector documented comments which were identified during the review of Surveillance Instruction 0-SI-236-1, 125 VDC Vital Battery Weekly Inspection, Revision 1. During this inspection period, the inspector reviewed Surveillance Instruction 0-SI-236-1 and verified that the applicant had changed documents as necessary to incorporate appropriate inspector comments.

In IR 50-390,391/94-52, Attachment, Surveillance Procedure Review Comments, Section D.0, the inspector documented comments which were identified during the review of Surveillance Instruction 0-SI-236-31, 125 VDC Vital Battery I Annual Inspection, Revision 0. During this inspection period, the inspector reviewed 0-SI-236-31 and Technical Specification SR 3.8.4.9 and verified that SR 3.8.4.9 had been corrected to read less than or equal to 80 microhms in lieu of 80 E-7 ohm. The applicant has replaced the 125 VDC vital batteries with new batteries from a different vendor and Surveillance Instruction 0-SI-236-31 was changed to include information pertinent to the new batteries.

12.2 (Closed) Hydrogen Recombiner System Operation Comments (IR 50-390/95-11)

In IR 50-390,391/95-11, Attachment, System Operating Procedure Review Comments, the inspector documented comments which were identified during the review of Procedure SOI-83.01, Hydrogen Recombiner System, Revision 8. During this inspection period, the inspector reviewed Procedure SOI-83.01 and verified that the applicant had changed the procedure to incorporate appropriate inspector comments.

12.3 (Closed) Safety Injection System Operation Comments (IR 50-390/94-38)

NRC IR 50-390,391/94-38 identified several concerns during the review of DB System Description N3-63-4001 for the safety injection system and Procedure TOP-63-02 for the safety injection system. These concerns were evaluated for applicability to the final system operating instructions, SOI-63.01. As a result of this evaluation, the following actions were accomplished:

- SOI-63.01 was revised to include instructions for draining containment penetration X-30 subsequent to each use as required by the DB system description. Safety injection system SIs were also verified to include X-30 draining instructions during restoration from use of the X-30 penetration.

- SOI-63.01 was revised to change the flow path for draining the cold leg accumulators to preclude reverse flow through kerotest valves ISV-63-610, -611, -612, and -613. Also, the DB system precaution associated with reverse flow through kerotest valves was added to the SOI.

- GO-6, Unit Shutdown from Hot Standby to Cold Shutdown, Revision 1, included instructions to close and remove power from FCV-63-8 and FCV-63-11 after cooldown to less than 350° but before placing the RHR system in service.

12.4 (Closed) Kerotest Valve Reverse Flow Concern (IR 50-390,391/94-38)

As documented in NRC IR 50-390,391/94-38, the applicant agreed to investigate the impact of reverse flow through Kerotest packless metal diaphragm Y-globe valves that were not designed for reverse flow. Reverse flow could cause severe piping, valve, and support damage due to excessive vibration. The applicant completed the inspection of the piping and the supports in the area in and around the valves on September 20, 1995. No external damage was observed. Two loose unistrut clamps were identified in accumulator rooms 2 and 3 which resulted in the initiation of WR C181045 to correct. The field inspection conducted by the applicant, as well as the procedure correction made to SOI-63.01 identified in the previous closed item, completes all the actions associated with reverse flow through kerotest valves.

12.5 (Closed) Various Operating Instruction Comment (IR 50-390,391/94-18)

NRC IR 50-390/94-18 documented comments and concerns associated with several operating instructions. The inspector evaluated the applicant's actions on the comments for Procedures SOI-74-01, SOI-62.01, SOI-68-01, and GO-1 (formally GOI-1). The applicant had either addressed the comments by changing the operating instructions or provided information as to why changes were not necessary. The inspector concluded that the applicant's actions had satisfactorily resolved the inspector's comments and concerns.

13.0 ACTIONS ON OPEN SAFETY ISSUES (92701)

Safety issues are generic matters that affect the design, construction, and operation of nuclear power plants. The NRC has developed and issued generic communications (e.g., NUREGs, Bulletins, Generic Letters, etc.) requiring action by the industry to address and resolve safety issues that may concern all applicable nuclear power plants, including the Watts Bar plant. The safety issues categories are: Three Mile Island Action Plan; Unresolved Safety Issues; Generic Safety Issues; and Multiplant Actions.

Instruction TI-2515/065, Revision 2, TMI Action Plan Requirement Follow-up, is the NRC's closeout instruction and consolidates all TMI NRC inspection requirements. Other applicable inspection criterion used during the reviews are identified with each individual safety issue.

NRR performed an independent audit to determine the completion status of all safety issues applicable to WBN Unit 1. The NRR report was transmitted to TVA/WBN in a letter dated April 6, 1995.

The NRR report considered that 65 safety issues were open and require verification that the applicant had implemented the safety issue for WBN Unit 1. The NRR report identified the following basic reasons for a safety issue to be listed as open.

- Several NUREG 0737 TMI Action Plan items were not addressed to the subtier level as specified in NUREG 0737, Enclosure 2, TMI Action Plan Requirements for Applicants for Operating License.

- Documentation of safety issues in some cases was not clearly stated in inspection reports with regard to implementation and completion status of the safety issue.

- Several safety issues were identified as open in the NRR report because the inspection guideline requirements were not fully incorporated into inspection reports (i.e., various combination of either not listing the applicable temporary instruction, inspection module, generic letter, or a bulletin, etc.).

- Several safety issues were identified in inspection reports as being reopened, thus requiring reverification of applicant's implementation.

Because of the above concerns, safety issues listed below have been evaluated for these types of deficiencies. When correction or clarification of a deficiency was determined to be necessary, this action was documented and included with the individual safety issues being inspected.

13.1 (Closed) TMI Item I.A.1.3.1, Shift Manning - Limited Overtime

(Closed) TMI Item I.A.1.3.2, Shift Manning - Minimum Shift Crew

The above TMI items were reviewed during an inspection documented in IR 50-390,391/95-37. The TMI Action Plan items in the report were not identified to the subtier level (i.e., .1 and .2 respectively) as specified in NUREG 0737 Enclosure 2, TMI Action Plan Requirements for Applicants for Operating License. A review of IR 50-390,391/95-37 does indicate that the requirements of the subtier was inspected. This safety issue remains closed.

13.2 (Closed) USI A-09, Anticipated Transient Without Trip

The 10 CFR was amended July 26, 1984, to include 10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light Water Reactors. Paragraph (c)(1) of the rule requires that pressurized water reactors have equipment, diverse from the reactor trip system, to initiate the AFW and a turbine trip under conditions indicative of an ATWS.

Inspections were performed in accordance with Instruction TI-2500/020, Inspection to Determine Compliance with ATWS Rule, 10 CFR 50.62, and documented in IRs 50-390,391/94-84 and 95-05. IR 50-390,391/95-05 completed the ATWS inspection activities, and a conclusion was reached that the installed systems met design requirements. In addition, IR 50-390,391/95-05 documented that portions of system testing were witnessed in accordance with Procedure PTI-003B-06, ATWS Mitigation System Actuation Circuitry (AMSAC) Test. No test deficiencies were identified during test witnessing. This safety issue is closed.

13.3 (Closed) USI A-24, Qualification of Class 1E Safety-Related Equipment (Unit 1 Only)

This safety issue resulted in the amendment of 10 CFR 50.49, Environmental Qualification of Safety-Related Equipment. The 10 CFR 50.49 established detailed requirements relating to the methods and procedures for assuring that mechanical and electrical equipment used to perform a safety function are capable of maintaining functional operability under all service conditions postulated to occur during its installed life and for the time it is required to operate in response to postulated accident conditions.

NRR completed their review of the applicant's EQ program and found the applicant's program acceptable. This review is documented in Section 3.11 of SSER 15.

This safety issue was reviewed during inspections documented in the following inspection reports. The inspections were performed in accordance with the inspection requirements and guidelines specified in the TIs listed within the individual inspection reports:

- IR 50-390,391/94-74, TI-2512/036, Evaluation of Applicant's Programs for Qualification of Electrical/Mechanical Equipment Located in Harsh Environments;

- IR 50-390,391/95-15, TI-2512/036, Evaluation of Applicant's Programs for Qualification of Electrical/Mechanical Equipment Located in Harsh Environments;

- IR 50-390,391/95-54, TI-2512/036, Evaluation of Applicant's Programs for Qualification of Electrical/Mechanical Equipment Located in Harsh Environments;

- IR 50-390,391/95-54, TI-2515/038, Mechanical EQ Special Program

- IR 50-390,391/95-54, TI-2515/076, Electrical EQ Special Program

IRs 50-390,391/94-74, 95-15 and 95-54 completed EQ inspection activities, and a conclusion was reached that the EQ programs were established and being implemented by the applicant in accordance with 10 CFR 50.49 requirements. IR 50-390,391/95-54 opened IFI 390/95-54-01, which involves finalizing some of the EQ binders by engineering and completing punch list items. These matters are planned for inspection during the routine and ongoing NRC inspection program. This safety issue is closed.

13.4 (Closed) USI A-44(A-22), Station Blackout

This safety issue addresses Station Blackout Rule 10 CFR 50.63, Loss of All Alternate Current Power. The station blackout rule requires that each light-water cooled nuclear power plant be able to withstand and recover from a station blackout (i.e., loss of off-site power systems concurrent with a reactor trip and the unavailability of onsite emergency ac power systems) of a specified duration.

NRR completed their review of the applicant's station blackout program and found that the applicant's methods for coping with a station blackout acceptable. This review is documented in an NRC letter dated September 9, 1995, Supplemental Safety Evaluation on Compliance with 10 CFR 50.63.

The implementation of the station blackout rule was reviewed during an inspection documented in IR 50-390,391/95-43. The inspection was performed in accordance with the inspection requirements and guidelines specified in TI-2515/120, Inspection of Implementation of Station Blackout Rule Multiplant Action Item A-22. IR 50-390,391/95-43 completed station blackout inspection activities and a conclusion was reached that the applicant had implemented a station blackout program that would meet 10 CFR 50.63 requirements once the following areas are completed:

- post-modification testing four vital batteries;
- preoperational testing the fifth vital battery;
- Operations group to issue station blackout procedures (EOPs).

IR 50-390,391/95-58 performed a reviewed of those EOPs that are to be used during a station blackout event. The EOPs were determined to be of satisfactory quality. Procedure PTI-236-03 was issued to test the fifth vital battery. The PTI was reviewed and determined to meet design testing requirements. The inspection of this test procedure is discussed in paragraph 2.1 of this report. The review of post-maintenance testing of the four vital batteries is planned for inspection during the routine and ongoing NRC inspection program. This safety issue is closed.

13.5 (Closed) GSI 67.3.3 (A-17), Improved Accident Monitoring

This safety issue involved the applicant's conformance with RG 1.97, Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.

NRR completed their review of the applicant's responses to RG 1.97, Post-accident Monitoring System, and found the applicant's design acceptable. This review is documented in Section 7.5.2 of SSER 9.

The implementation of the applicant's commitments to RG 1.97, Revision 2, was documented in IR 50-390,391/95-31. The inspection was performed in accordance with the inspection requirements and guidelines specified in Instruction TI-2515/087.

IR 50-390,391/95-31 completed the inspection activities for the post-accident monitoring system, and a conclusion was reached that the post-accident monitoring system had been installed and met design and qualification commitments of RG 1.97. IR 50-390,391/95-31 opened IFI 50-390/95-31-02, which involved the issuance of periodic calibration program procedures and testing. These matters are planned for inspection during the routine and ongoing NRC inspection program. This safety issue is closed.

13.6 (Closed) GSI 75(B078), Items 3.1.1 and 3.1.2 - Post Maintenance Testing - Rx Trip Sys Components

(Closed) GSI 75(B087), Items 3.2.1 and 3.2.2 - Post Maintenance Testing - All Other Safety Related Components

(Closed) GSI 75(B093), Items 4.5.2 and 4.5.3 - Reactor Trip System On-Line Functional Test

These safety issues were identified as specific items in GL 83-28. They involve post-maintenance testing of safety-related components which should be performed to demonstrate that the equipment is capable of performing its safety function prior to returning it to service.

NRR completed their review of the applicant's responses to GL 83-28 and found them acceptable. This review is documented in Section 15.3.6 of SSER 5.

The implementation of the applicant's commitments in response to GL 83-28 were reviewed during inspections documented in IRs 50-390,391/86-04 and 94-73. These inspections were performed in accordance with the inspection requirements and guidelines specified in Instructions TI-2515/064, Near-Term Inspection Followup to GL 83-28, Revision 1, and TI-2515/091, Inspection Followup to GL 83-28.

IR 50-390,391/94-73 completed GL 83-28 inspection activities, and a conclusion was reached that the applicant had implemented a comprehensive program of maintenance and surveillance for Rx trip breakers and other safety related components. However, the inspector identified some follow-up items in IR 50-390,391/94-73 which were reviewed during the current inspection period. Discussions related to closure of the items are provided in paragraphs 13.6.1, 13.6.2, and 13.6.3. This safety issue is closed.

13.6.1 Generic Letter 83-28 Items 3.1.1 and 3.1.2 - Post Maintenance Testing (PMT) (Reactor Trip System Components)

In IR 50-390, 391/94-73, Section 11.2.1, the inspector documented a review of the actions taken by the applicant to resolve Generic Letter 83-28 Items 3.1.1 and 3.1.2 - Post Maintenance Testing (PMT) (Reactor Trip System Components). In that documentation the inspector concurred with the determination made in paragraph 10.0 of IR 50-390, 391/86-04 that the applicant's responses for PMT of reactor trip system components were acceptable and that the applicant's documentation and suggested performance of the following procedures should be acceptable:

- MI-99.001, 480 Volt Reactor Trip Breaker Inspection and Testing,

- MI-99.002, 480 Volt Reactor Trip Switchgear Inspection and Testing.

However, the inspector stated that final control copies of these PMT procedures would require further validation and testing prior to plant startup. Also, thorough training and certification of such training in the use and performance of the listed procedures by both facility craft and Operations personnel would be necessary prior to plant startup and operation.

The applicant implemented Instruction MI-99.001, 480 Volt Reactor Trip Breaker Inspection and Testing, Revision 1, to provide detailed steps for performing the manufacturer's recommended inspection and preventive maintenance of Unit 1 Westinghouse reactor trip breakers. During September 1995, the applicant performed Instruction MI-99.001 on reactor trip breakers 0-BKR-548-RT01 (WO 951779800), 0-BKR-548-RT02 (WO 951780200), 0-BKR-548-RT03 (WO 951779900), 0-BKR-548-RT04 (WO 951780100). The inspector reviewed the work documents for breakers RT01, RT02, and RT03 and also discussed this maintenance with applicant craft personnel. Based on the review and discussion, the inspector determined that Instruction MI-99.001 was adequate to perform maintenance on the reactor trip breakers and the craft personnel appeared to have the necessary knowledge and experience to perform the maintenance.

The applicant implemented Instruction MI-99.002, 480 Volt Reactor Trip Switchgear Inspection and Testing, Revision 0, to provide steps for performing the manufacturer's recommended inspection and preventive maintenance of Unit 1 Westinghouse reactor trip switchgear. The applicant performed Instruction MI-99.002 in August 1994 (WO 941851700) to support Integrated Test Sequence PTI-99-03. The inspector reviewed the completed work instruction and determined that it was adequate to perform maintenance on the reactor trip switchgear.

The applicant implemented Instruction MI-57.002, 480-Volt Circuit Breaker Routine Maintenance, Inspection and Testing, Revision 21, to provide detailed steps to perform the manufacturer's recommended inspections and preventive maintenance on Westinghouse 480 Volt DS type circuit breakers except for the reactor trip and bypass breakers. The trip and bypass breakers at WBN were Westinghouse type DS-416. There were approximately 70 type DS-416 breakers installed at WBN in addition to the reactor trip and bypass breakers. The reactor trip breakers are different from the other DS-416 breakers at WBN in that they have undervoltage trip units installed on them. Instruction MI-57.002 was performed every two years on each of the DS-416 breakers. Many of the activities performed by Instruction MI-57.002 were the same or similar to the activities performed by Instruction MI-99.001. Instruction MI-99.001 was more prescriptive in providing detailed step-by-step instructions for the maintenance of the reactor trip and bypass breakers. Therefore, the applicant has gained experience by performing many of the same maintenance activities delineated in Instruction MI-99.001 while performing Instruction MI-57.002 on the other 480 Volt type DS circuit breakers in the plant.

Section 7.2 of Instruction MI-99.001 required the following surveillance instructions to be performed to satisfy PMT requirements:

- 1-SI-99-1, 18 Month Trip Actuating Device Operational Test of Manual Reactor Trip, Revision 0; and

- 1-SI-99-10-A, 31 Day Functional Test of SSPS Train A and Reactor Trip Breaker A, Revision 0; or (as applicable)

- 1-SI-99-10-B, 31 Day Functional Test of SSPS Train B and Reactor Trip Breaker B, Revision 0.

Section 7.2 of Instruction MI-99.002 required Surveillance Instruction 1-SI-99-1 to be performed to satisfy PMT requirements.

The applicant implemented Surveillance Instruction 1-SI-99-1 to provide detailed instructions to determine the operability of the manual reactor trip function of the Reactor Trip System.

The applicant implemented Instruction 1-SI-99-10-A to provide the detailed steps to determine the operability of the Solid State Protection System (SSPS) Train A and the Reactor Trip Breaker A.

The applicant implemented Instruction 1-SI-99-10-B to provide the detailed steps to determine the operability of the Solid State Protection System (SSPS) Train B and the Reactor Trip Breaker B.

The applicant has validated these SIs and gained practical experience by performing them in the field.

The applicant performed a SPOC validation of Instruction 1-SI-99-1 on September 2, 1994 (WO 941851700) and performed the SI again on June 7, 1995 (WO 950907100).

The applicant performed a SPOC validation of Instruction 1-SI-99-10-A on May 29, 1995 (WO 950789700), and performed the SI again on July 29,1995 (WO 951174700) and on October 13, 1995 (WO 952014500).

The applicant performed a SPOC validation of Instruction 1-SI-99-10-B on May 18, 1995 (WO 950789800), and performed the SI again on August 31, 1995 (WO 951175400) and on October 12, 1995 (WO 952046300).

The inspector observed the applicant perform SI 1-SI-99-10-A on October 12 and 13, 1995. The SI performers were deliberate in performing the SI step by step and adhering to the instructions in the SI. When problems or questions were encountered the performers stopped and resolved the problem/question before continuing. Problems/questions were identified to supervision and engineering for resolution. During the performance on October 12, 1995 a problem was encountered in Section 6.2, Logic Test of SSPS Train A of the SI in that four of the switch positions in Table 6-9 indicated bad. The performers wrote a work order to correct the problem and after correcting the problem performed the SI on October 13, 1995. The inspector discussed this problem with the applicant's cognizant engineering personnel. Engineering was monitoring performance of this SI to ensure the method in which the performers operated the logic switches (i.e., operating the switches rapidly) was not causing the logic circuits to give a bad circuit indication when there wasn't a problem.

In IR 50-390,391/94-73, paragraph 11.4, the inspector noted that the shortest time the applicant was so far able to implement the 31-day surveillance (1-SI-99-10-A, -B) was four hours, against a two-hour TS allowed outage time. The applicant stated that on October 12, 1995, they had performed Instruction

1-SI-99-10-B in approximately one hour and twenty minutes. The applicant's goal was to reduce this time to one hour which was half of the TS allowed time of two hours.

Generic Letter 83-28 Items 3.1.1 and 3.1.2 - Post-Maintenance Testing (PMT) (Reactor Trip System Components) are closed.

13.6.2 Generic Letter 83-28 Items 3.2.1 and 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components)

In IR 50-390,391/94-73, Section 11.2.3, the inspector documented a review of the actions taken by the applicant to resolve Generic Letter 83-28 Items 3.2.1 and 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components). In that report the inspector stated that final control copies of these PMT procedures will require further validation/testing prior to plant startup.

The applicant implemented Procedure SSP-6.02, Maintenance Management System, Revision 16, which established the process for initiating, planning, performing, cancelling, completing, and tracking work using the WR/WO process. Section 2.2.4.P of Procedure SSP-6.02 required the individual planning a WO to determine the post maintenance test requirements.

The applicant implemented Procedure SSP-6.03, Preventive Maintenance Program, Revision 9, which described processes and related requirements for controlling preventive maintenance activities, including the predictive maintenance monitoring program. Section 2.2.1.I.17 of Procedure SSP-6.03 required the preparer to specify post-maintenance test requirements when preparing preventive maintenance work instructions.

These SSPs functioned to ensure that appropriate post-maintenance testing was considered in the development of maintenance instructions for performing maintenance on safety-related components.

Generic Letter 83-28 Items 3.2.1 and 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components) are closed.

13.6.3 Generic Letter 83-28 Items 4.5.2 and 4.5.3 - Reactor Trip System On-Line Functional Testing

In IR 50-390,391/94-73, Section 11.3.4, the inspector documented a review of the actions taken by the applicant to resolve Generic Letter 83-28 Items 4.5.2 and 4.5.3 - Reactor Trip System On-Line Functional Testing. In that discussion the inspector stated that in a letter from Tam (NRC) to Kingsley (TVA) the applicant's responses to Items 4.5.2 and 4.5.3 were documented as being acceptable, and the inspector concurred with the statements in the letter. However, the inspector also stated that final control copies of these on-line testing procedures will require further validation and testing prior to startup.

The applicant developed and implemented Surveillance Instructions 1-SI-99-1, 1-SI-99-10-A, 1-SI-99-10-B which the applicant will use to perform on-line functional testing of the reactor trip system. As discussed in paragraph

13.6.1, the applicant has validated these SIs and gained practical experience by performing them in the field.

Generic Letter 83-28 Items 4.5.2 and 4.5.3, Reactor Trip System On-Line Functional Testing are closed.

13.7 (Closed) MPA A025, IST Review and Schedules (GL 89-04)

License inservice test programs were noted to contain a number of generic deficiencies that could affect plant safety. In an effort to address these concerns, the NRC issued GL 89-04, Guidance on Developing Acceptable Inservice Test Programs.

NRR completed their review of the applicant's responses to GL 89-04 and program for inservice testing of certain ASME Code Class 1, 2, and 3 pumps and valves as required by 10 CFR 50.55a. The NRR reviews found the applicant's responses and IST program acceptable. This review is documented in Section 3.9.6 of SSER 14.

The implementation of the applicant's commitments in response to GL 89-04 and their IST program was reviewed during an inspection documented in IR 50-390,391/95-05. This inspection was performed utilizing the guidance and requirements contained in the following documents:

- IP 73756,IST Pumps and Valves
- SSER 14, Section 3.9.6
- GL 89-04
- OM-1987, Part 1 Requirements, IST Pressure Relief Devices
- OMa - 1998 addenda, Part 6, IST of Pumps
- OMa - 1998 addenda, Part 10, IST of Valves

Instruction TI- 2515/114, Inspection Requirements for GL 89-04 Acceptable Inservice Testing Programs, was not specifically identified as being used during the inspection. A review of inspected and documented areas in IR 50-390,391/95-05 indicate adequate implementation of the TI inspection attributes. IR 50-390,391/95-52 documented the witnessing of several surveillance tests involving safety-related pumps and valves.

IR 50-390,391/95-05 opened IFI 50-390/95-05-02, which identified several concerns with the IST program implementation and procedural requirements. The applicant made changes to the program implementing procedures and IFI 50-390/95-05-01 was closed in IR 50-390,391/95-60.

IRs 50-390,391/95-05 and 95-52 completed the inspection activities for the IST pump and valve program, and a conclusion was reached that the applicant had implemented an adequate IST program. This safety issue is closed.

13.8 (Closed) MPA B031 Deep Draft Pump Deficiencies (IEB 79-15)

A number of design and manufacturing deficiencies were discovered with deep draft pumps. These deficiencies were noted to cause excessive operational vibration and bearing wear which resulted in reduced flow rates. The NRC issued IEB 79-15 requesting addresses to provide specific information regarding the number and type of deep draft pumps used in safety-related applications and a summary of pump maintenance and operating history.

NRR completed their review of the applicant's responses to IEB 79-15 and found the responses acceptable. This review is documented in Section 3.10 of SSER 4.

IEB 79-15 was reviewed during an inspection documented in IR 50-390,391/94-45 and, based on that review, the IEB was closed. IFI 50-390/91-03-04 was opened to review the applicant's corrective actions concerning low flowrates identified during surveillance testing of ERCW pumps. IR 50-390,391/93-20 documented that the applicant had enhanced operability and problem detection for all deep draft pumps by the revision of preventive maintenance instructions. IFI 50-390/91-03-04 was closed in IR 50-390,391/94-66. This safety issue is closed.

13.9 (Closed) MPA L907, Safeguards Contingency Planning for Surface Vehicle Bombs

The NRC issued GL 89-07 which requested licensees to modify their safeguards contingency procedures to address and protect against a surface vehicle bomb if such a threat should materialize.

NRR completed their review of the applicant's responses to GL 89-07 and found the responses acceptable. This review is documented in an NRC letter dated November 6, 1989.

The implementation of the applicant's commitments in response to GL 89-07 was reviewed during an inspection documented in IR 50-390,391/94-71. This inspection was performed in accordance with the inspection requirements and guidelines specified in Instruction TI-2515/102, Land Vehicle Bomb Contingency Procedures Verification.

IR 50-390,391/94-71 completed GL 89-07 inspection activities and a conclusion was reached that the applicant had appropriately implemented GL 89-07 commitments concerning safeguards contingency planning for surface vehicle bombs. This safety issue is closed.

13.10 (Closed) MPA X804, Potential Safety-Related Pump Loss (IEB 88-04)

The NRC issued IEB 88-04 requesting addresses to provide specific information regarding the design of minimum flow (mini flow) lines associated with safety related pumps. The IEB concerns involves the potential for dead-heading and subsequent loss of function for one of two pumps having a common miniflow line and whether the miniflow line capacity is adequate for serving even a single pump.

NRR completed their review of the applicant's responses to IEB 88-04 and found the responses acceptable. This review is documented in an NRC letter dated May 24, 1990.

IR 50-390,391/94-10 completed GL 89-07 inspection activities, and a conclusion was reached that the applicant had appropriately implemented IEB 88-04 commitments concerning miniflow line design and capacity, except for the AFW system miniflow line. IR 50-390.391/93-74 opened URI 50-390/93-74-05, which was concerned with the capacity of the miniflow piping design for the AFW system pumps.

The applicant evaluated the current miniflow line design capacity and determined that the existing design was acceptable for operation without abnormal pump degradation. This determination was based on several factors such as follows:

- Special tests were performed while the AFW pump was on miniflow.
- Sequoyah AFW system pump operating history (several years) was evaluated as AFW pumps operated at or near miniflow during periods of startup, shutdown, and Rx trips.
- WBN changed the AFW pump diffusers from cast iron to stainless steel to improve resistance to wear.
- Surveillance Instructions were enhanced for the monitoring/trending pump parameters during IST program testing.

The review/closure of URI 50-390/94-73-05 is discussed in paragraph 14.1 of this report. This safety issue is closed.

13.11 (Closed) MPA B110, Motor Operated Valve Testing and Surveillance (GL 89-10)

The NRC issued GL 89-10 requesting addresses to establish a program for the testing, inspection, and maintenance of safety-related MOVs and certain other MOVs in safety-related application to ensure operability of these MOVs under design basis conditions.

NRR completed their review of the applicant's responses to GL 89-10 and found the responses acceptable. This review is documented in SSER 16.

IRs 50-390,391/93-13, 94-36, 95-21, and 95-48 completed GL 89-10 inspection activities, and a conclusion was reached that the applicant had appropriately implemented GL 89-10 commitments concerning safety-related MOVs testing and surveillance.

The inspections were performed in accordance with the inspection requirements and guidelines specified in Instruction TI-2515/109, Inspection Requirements for GL 89-10, Safety-Related MOVs Testing and Surveillance. This safety issue is closed.

13.12 (Closed) TMI Item II.B.4.2 Training for Mitigation of Core Damage - Completion of Training

This safety issue was inspected using the guidance of Instruction TI 2515/065, TMI Action Plan Requirement Followup. The inspector reviewed the applicant's nuclear training manual procedures that define training programs for shift technical advisors, licensed operators, health physics technicians, instrumentation and control technicians, and chemistry technicians. The procedures require that personnel in all of the above positions who will respond to a radiological emergency receive training in mitigation of core damage. The shift technical advisors and licensed operators receive comprehensive training in the topic. Experience requirements for personnel who occupy Operations management positions encompass the requirement to have had training that included mitigation of core damage. The instrumentation and control, health physics, and chemistry technicians receive training in the topic as it is related to their specific duties. This topic is covered in initial training for each position and is also included in the continuing training program. The inspector reviewed a representative sample of class attendance rosters for instrumentation and control technicians, chemistry technicians, health physics technicians, and licensed operators. Class rosters for continuing training programs were also reviewed. Based on this review, the inspector concluded that training had been completed. IR 50-390,391/95-01, paragraph 2.1, documents the results of previous recent inspection effort in this area as it relates to licensed operators and shift technical advisers. As a result of this review, the inspector determined that the applicant had met the requirement for completion of training on mitigating core damage. This safety issue is closed.

13.13 (Closed) TMI Item II.E.1.2.1.B, Auxiliary Feedwater System Automatic Initiation and Flow Indication - Long Term

IR 50-390,391/84-20 stated that the AFW automatic initiating signals and the circuits met the safety grade requirements of TMI Item II.E.1.2. However, IR 50-390,391/88-01 revealed concerns regarding the qualification of the AFW initiating circuitry. Accordingly, TMI Item II.E.1.2 was reopened. Follow-up on this concern was completed and documented in IR 50-390,391/94-60, and the item was closed. However, there was no documentation relating to Item II.E.1.2.1.B, so this sub-item remained open.

Sub-item II.E.1.2.1.B requires the AFW automatic initiation system be installed as designed for the long term and its operability be verified.

During this inspection, the installed AFW system was verified as follows:

- The long term safety grade requirements were verified by review of the current System Description N3-3B-4002, Auxiliary Feedwater System, Revision 3, dated February 24, 1994.

- The applicant's response to the concern identified in IR 50-390,391/88-01 stated that a deviation did not exist because the AFW system control circuitry is located within a mild environment and environmental qualification was not required. This response was accepted by the NRC, but the documentation did not specifically relate this position to TMI Item II.E.1.2.1.B. Additional review of the inspection finding and the applicant's response, dated August 17, 1988, reveals that AFW control circuits are located in a mild environment and are not required to meet the documentation requirements of IEEE 323-1971. These components were purchased before 1982 and qualification is demonstrated by preventative maintenance, testing, and surveillance.

The applicant has prepared and completed Procedure PTI-003B-05, Revision 0. This test was completed on August 23, 1995. Several test deficiencies identified during the test have been addressed and closed. All required acceptance criteria were met.

During this inspection, the inspector toured the control room with the responsible test engineer to review the as-installed system. Discussions were held with the control room operators on duty and the test engineer. These people were very knowledgeable of the installed system and the safety importance of the issue. Each steam generator is provided two flow indicators within the control room, and the AFW system design and automatic initiating signals and circuits meet the safety grade requirements.

13.14 (Closed) TMI Item II.E.4.2.7, Containment Purge Valve Isolation on High Radiation

This item imposes a requirement related to the dependability of containment isolation systems. The specific requirement is to close automatically all containment purge and vent valves on a containment high radiation signal.

The applicant responded to this issue by letters dated September 14 and October 29, 1981. NRC approval was presented in a SER issued in June 1982. Subsequently, the applicant changed the plant configuration such that a containment high radiation does not cause a containment ventilation isolation. This change was completed by DCN M-13516.

FSAR Chapters 9 and 11 have been amended. The chapters describe the reactor building purge ventilation monitors and containment atmosphere monitors. These changes have been reviewed by the NRC as part of Amendments 77 and 87, which are covered by SSERs 12, 13, and 14.

Removal of the automatic containment ventilation isolation was not specifically addressed in SSERs 12, 13, or 14. The staff's review included the substance of Item II.E.4.2.7 even though it was not specifically mentioned.

The system description for the reactor building ventilation system, N3-30RB-4002, Revision 5, has been updated and containment purge system isolation is initiated by either of two signals:

- Manual - Phase A or B manual
  - SIS manual initiate

- Automatic - SIS auto initiate
  - High purge exhaust radiation (one of two sensors)

The process radiation monitoring system preoperational tests have been completed for each radiation monitor. Each monitor was source checked for proper response and output signal. A check was made to verify that the proper actions occur and the containment isolation valves close. These tests were as follows:

- PTI-090-01, Radiation Monitor
- PTI-099-04, Safeguard System Operational Test
- PTI-099-01, RPS and ESFAS Response Times

Results of these tests were satisfactory and are on file in the WBN records program.

13.15 (Closed) TMI Item II.E.4.2 (1-4), Containment Isolation Dependability Sub-Items 1-4

This item imposes specific requirements, all relating to dependability of containment isolation systems. The specific sub-items under review are:

- automatically initiate isolation by a diversity of parameters;

- re-evaluate and justify to the NRC the assignment of lines penetrating the containment into essential and non-essential categories;

- automatically isolate all non-essential systems on a containment isolation signal; and

- design the containment isolation systems such that isolation valves do not automatically open upon resetting of the containment isolation signal.

The applicant's responses to this item have been incorporated in the FSAR and thus have been reviewed by the NRC staff.

Qualified diverse containment isolation signals are provided. The system is designed to prevent release of radioactive material to the environment after an accident while ensuring that systems important for post-accident mitigation are operational. The containment isolation system is designed to prevent the inadvertent opening of an isolation valve when closed by an initiating signal. Prior to reset of the initiating signal, a valve closed by the signal can be opened only by constant operator demand with a valve's individual hand switch. The valve will return to the containment isolation position when the operator releases the hand switch. Resetting the isolation signal will not cause a containment isolation valve to change position. Each valve must be individually operated to cause a change from the containment isolation position.

The WBN design complies with NRC requirements on the automatic isolation of non-essential systems. The inspector reviewed the results of confirmatory tests which were conducted by the applicant. These tests were:

- PTI-262-01, Integrated Safeguards Test

- PTI-099-04, Safeguards System Operational Test

The results of these tests indicate that the design and installation of the containment valves do automatically isolate the containment upon receipt of the appropriate signals. Testing revealed that resetting of the isolation signal does not cause the isolation valves to re-position. Implementation and verification of this item is complete. TMI Item II.E.4.2 (1-4) is closed.

13.16 (Closed) TMI Item II.E.1.2.2.B, Auxiliary Feedwater System Flow Indication - Long Term

TMI Item II.E.1.2.2.B imposes the specific requirement that indication of AFW flow shall be provided for each steam generator using safety grade circuitry in the long term. TVA responded to this item by letter dated September 14, 1981. AFW flowrate indication and initiation were described. NRC evaluated and accepted the response as documented in Section 10.4.9 of NUREG 0847 dated June 1982.

NRC IR 50-390,391/95-15, dated April 5, 1995, documents a satisfactory review of the equipment qualification program for the AFW system.

Results of confirmatory tests which were conducted by the applicant were reviewed by the inspector. These tests were:

- PTI-003B-04, AFW Pumps and Valves Logic Test

- PTI-003B-05, AFW Pumps and Valves Dynamic Test

- PTI-003-06, ATWS Mitigation System Actuation Circuitry (AMSAC) Test

The results of these tests indicate that the design and installation of the AFW system automatically provides flow to each steam generator. Results of these tests have been reviewed and accepted by the JTG and are available through the records management program.

Implementation and verification of this item is complete. TMI Item II.E.1.2.2.B is closed.

13.17 (Closed) TMI Item II.E.3.1, Emergency Power Supply for Pressurizer Heaters

This issue requires that power to portions of pressurizer heaters be available, from either normal power supply or from station emergency power supply, to establish and maintain natural circulation. Procedures and training for the reactor operators are required to enable them to connect the power supply consistent with timely initiation and maintenance of natural circulation condition.

Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety grade requirements.

Follow-up on this TMI item was completed by reviewing test procedures and interviewing plant personnel.

The pressurizer heaters are installed as described in Chapter 8 of the FSAR. All four heater banks will trip on a SI signal. After SI reset and level recovery within the pressurizer, one backup heater bank operates automatically. The other two backup heater banks and the control bank do not come on automatically but are manually activated. Upon loss of offsite power and SI signal, two backup heater banks can be manually activated by hand switches in the control room. These manual actions are described in emergency instructions, which were inspected and reported in NRC IR 50-390,391/95-58.

IR 50-390,391/84-20 reported that the pressurizer heater power supply was from the emergency buses as described in the FSAR and that the interfaces had been qualified in accordance with safety grade requirements. During this inspection the control room was toured with the system engineer and discussion related to this issue was held with control room personnel on duty. The applicant's personnel were very knowledgeable with the pressurizer heater controls and emergency actions. Additional verification was done through review of test results of Procedure PTI-068-03, Pressurizer Heater and Spray Control. The test results met acceptance criteria. Implementation and verification of TMI Item II.E.3.1 is complete.

TMI Item II.E.3.1 is closed.

13.18 (Closed) TMI Item II.E.4.2.6, Containment Purge Valve Operability and MPA B-024, Containment Purging and Venting During Normal Operation

With the issuance of NUREG-0737, Clarification of TMI Action Plan Requirements, in November 1980, MPA B-024 essentially was subsumed by TMI Item II.E.4.2.6. Follow-up on both safety issue items was conducted concurrently since they are identical.

This TMI item and MPA B-024 item require containment purge valves be sealed closed except for those that are operable under the most severe accident flow loading and are capable of closing within the technical specification limit. The applicant's responses to this item were evaluated and on July 12, 1990, NRC provided its safety evaluations of the containment vent and purge valves, finding that all valves in the upper containment were satisfactory and that the 24-inch valves in the lower containment were satisfactory provided they were mechanically blocked to an opening angle of 50° or less. Supplement 5 to the SER, issued in November 1990, incorporated these conditions. Orientation of the 8, 12, and 24-inch valves in their preferred flow direction and restricting the opening angle to 50° has been implemented by ECN 5226. The mechanical stops are shown on vendor drawings 12407 and 12408, and the installation was verified by the inspector. A note regarding the proper valve orientation has been added to the vendor drawing for the 8, 12, and 24-inch valves. The limitation on the opening for the 24-inch lower containment valves is reflected in technical specification SR 3.6.3.7. The applicant has provided Instruction 1-SI-30-902-A and -B to periodically verify:

- the proper opening restriction of the 24-inch containment lower compartment purge supply and exhaust isolation valves is in place;

- the remote position indication of Train A and B valves in the ventilation system are accurately indicated at the primary control switch.

Instruction 1-SI-30-902-A was performed on September 5, 1995, and Instruction 1-SI-30-902-B was performed on August 29, 1995. Both tests were reviewed by the inspector and both met the acceptance criteria.

Additionally, SIs are provided to verify valve stroke times in compliance with Technical Specification 3.6.3, Containment Isolation Valves. These are 1-SI-30-901-A and -B, Valve Full Stroke Exercising During Plant Operation - Ventilation (Trains A and B).

Train A valves were tested on August 23, 1995, and Train B on July 14, 1995. Both tests met the acceptance criteria. These tests are also included in the ASME Section XI Pump and Valve Inservice Testing Program. The results of these tests were reviewed by the inspector. Implementation and verification of TMI Item II.E.4.2.6 and MPA B-024 on complete. This item is closed.

13.19 (Closed) TMI Item II.K.3.5.B, Automatic Trip of Reactor Coolant Pumps

This item resulted from the generic reviews of small-break loss of coolant accidents and loss-of-feedwater events. Based on these reviews, specific requirements were generated to provide for the automatic trip of RCPs at PWR plants.

NRC issued several documents during the early 1980s which provided information on the resolution of this issue. In 1983 and 1984, the Westinghouse Owner's Group developed guidance on alternate means of addressing requirements for tripping RCPs. The Westinghouse Owner's Guide provided technical justification to substantiate the position that RCPs not be automatically tripped but should remain operational for non-LOCA transients and other accidents where their operation was beneficial to accident mitigation and recovery.

On July 28, 1985, the NRC issued GL 85-12, Implementation of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps. This GL approved the Westinghouse Owners Group's position for the manual tripping of RCPs. The applicant's letters to the NRC dated August 29, 1985, and January 13, 1986, stated that the RCPs at WBN would be tripped manually at 1400 psig reactor coolant system pressure in the event the reactor coolant system pressure was decreasing uncontrollably. This arrangement met the Westinghouse Owners Group's position.

During telephone conference calls between NRC and the applicant on November 12, 1986, and June 8, 1990, the applicant's position for the manual trip of RCPs was also discussed. Following NRC review of the applicant's submittals and information obtained from these conference calls, the applicant's position for manually tripping of RCPs was found acceptable. The requirement to manually trip the RCPs was to be incorporated into the WBN's Emergency Operating Procedures. NRC's approval was documented in a letter to the applicant dated June 8, 1990.

The inspectors reviewed TVA's calculation WBN-OSG4-188, EOP Setpoints Verification Document, which indicated that Instruments 1-PI-68-63, 1-PI-68-64 and 1-PI-68-70, which are to be used by the operators to monitor reactor coolant system pressure, were to be set at 1500 psig. Based on the accuracy of the instrumentation, the actual required calculated setpoint is 1474.5 psig. The setpoint of 1500 psig provided an additional safety margin in the conservative direction.

The inspectors reviewed the WBN's EOPs which are included in the document entitled Emergency Instructions. These EOPs implemented the Westinghouse Owners Group's emergency response guidelines and contain instructions for manually tripping the RCPs. The first instruction on the foldout page to Emergency Instruction E-0, Reactor Trip or Safety Injection, Revision 10, contains the following criteria:

<u>RCP Trip Criteria</u>

- Phase B Isolation, OR

- One Changing pump OR one SI pump injecting AND RCS pressure decreasing uncontrolled to less than 1500 psig.

This same RCP trip criteria is contained in other EOPs where appropriate. This issue is closed.

13.20 (Closed) TMI Item II.K.3.10, Proposed Anticipatory Trip at High Power

This item required plants to perform an analysis to demonstrate that when the unit was operating below 50 percent power, the probability of a small-break loss-of-coolant accident, resulting from a stuck open PORV, would not be affected by the deletion of the reactor trip on a turbine trip function.

TVA's letter to the NRC dated April 6, 1984, stated that the anticipatory reactor trip on turbine trip feature at WBN's had been modified so that this feature operated at power levels of 50 percent or greater instead of power levels of 10 percent or above. This change was made after Westinghouse's WBN specific analysis of this modification concluded that the probability of a small-break, loss-of-coolant accident resulting from a stuck open PORV would be essentially unaffected by this change. TVA's letter to the NRC dated October 23, 1984, provided additional information for dose rate analysis, specifically steam release rate calculations and off-site dose calculations. This information indicated that a full steam release to the atmosphere through secondary safety valves would result in off-site dose consequences which were estimated to be a very small fraction of the dose specified by the 10 CFR 100 guidelines. FSAR Sections 7.1 and 7.2 were revised to reflect the provision of the 50 percent permissive on a reactor trip following a turbine trip. NRC's approval of TVA's change in the anticipatory reactor trip function is documented by SER Supplement 4, Section 7.8.4.

TVA's letter to the NRC dated February 16, 1995, provided additional information on reanalyses for a small-break, loss-of-coolant accident and other postulated events. These were reviewed by the NRC and, as documented by SER Supplement 15, Section 15.3.1, were found acceptable.

The inspectors reviewed Section 3.3.1 and Table 3.3.1-1 of WBN draft Technical Specifications and noted that reactor trip was not required at a power level of 50 percent or less. The surveillance requirements of Section 3.3.1.11 and 3.3.1.12 of the draft Technical Specifications were reviewed and found to require a channel calibration and channel operability test every 18 months on the anticipatory trip functions, including the various interlock functions, which do not automatically trip the reactor at power levels of 50 percent or less. The inspectors verified that these requirements were also incorporated into the following surveillance procedures:

- 1-SI-92-41, 18 Month Channel Calibration of Power Range Nuclear Instrumentation System Channel N41

- 1-SI-92-42, 18 Month Channel Calibration of Power Range Nuclear Instrumentation System Channel N42

- 1-SI-92-43, 18 Month Channel Calibration of Power Range Nuclear Instrumentation System Channel N43

- 1-SI-92-44, 18 Month Channel Calibration of Power Range Nuclear Instrumentation System Channel N44

Completion of these procedures are required prior to entry into Mode 1. This item is closed.

14.0 ACTIONS ON PREVIOUS INSPECTION FINDINGS (92701)

14.1 (Closed) URI 50-390,391/93-74-05, Adequacy of Auxiliary Feedwater Minimum Flow Design

The motor-driven and turbine-driven AFW pumps at WBN are designed with a miniflow piping system of 30 gpm and 50 gpm, respectively. The inspector had questioned the adequacy of this design based on industry experience and vendor precautions. This was documented in IR 50-390,391/93-74, paragraph 6. The applicant indicated that the pump diffusers had been upgraded to stainless steel and that the pumps would not be used for normal startup and shutdown because a motor-driven main feed pump was available. However, no low flow testing had been accomplished, vendor recommended enhanced monitoring during low flow testing had not been implemented, procedure cautions to limit low flow running of the pumps as much as possible were not in place, operators had not been trained to limit low flow conditions, and a program to measure the run time under low flow conditions had not been developed. The applicant has now completed each of these activities. Documentation of additional followup inspections is contained in IRs 390,391/93-85, paragraph 6.b; 94-11, paragraph 9.c; and 94-58, paragraph 12.4. During this inspection period, the inspector confirmed that Surveillance Instructions had been issued which incorporated vendor recommendations, AOI procedure precautions had been implemented, PM requirements to record low flow run time were in place, and operators had been trained. In addition, the inspector reviewed the low flow test results. The applicant decided not to perform full flow testing previously planned and described in IR 50-390,391/94-11. This was considered acceptable. The applicant has implemented adequate pump monitoring activities and controls. In addition, long term plans are in place to upgrade the miniflow line. These actions are considered adequate to close this issue.

14.2 (Closed) IFI 50-390, 391/95-40-03, Equipment and Supplies for Appendix R Cold Shutdown Repairs Were Not Available

Maintenance Instruction MI-0.047, Appendix R Safe Shutdown Repairs, provides details instructions to perform repairs needed to reach safe shutdown following an Appendix R fire. This instruction was reviewed during a July 1995 NRC inspection and found to be satisfactory, except the materials and supplies required by the procedure had not yet been obtained and stored in the plant.

The inspectors reviewed this item during this inspection and noted that the required cold shutdown repair materials had been obtained and were neatly stored in two metal storage boxes located on elevation 755 of the Turbine Building, adjacent to the entrance to the Unit 1 Control Room.

The procedure requires six electrical jumpers and five spare cables. Eight jumpers and five spare repair cables were provided: two cables (6/C, 12 AWG each) approximately 300 feet in length; one cable (4/C, 8 AWG each) approximately 300 feet in length; one cable (3/C, 12 AWG each) approximately 300 feet in length; and one cable (3/C, 10 AWG each), approximately 600 feet in length. The tools required to perform this instruction are tools typically used and available to the site electrical craft personnel. Therefore, no special tools were designated, provided or needed to be stored with the repair materials in order to perform this repair instruction. This item is closed.

15.0 EXIT INTERVIEW

The inspection scope and findings were summarized on October 6, 1995, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results. Dissenting comments were not received from the applicant. Proprietary information is not contained in this report.

Item Number	Status	Description and Reference
390,391/93-74-05	Closed	URI - Adequacy of Auxiliary Feedwater Minimum Flow Design (paragraph 14.1)

390,391/95-40-03	Closed	IFI - Cold Shutdown Repair Equipment Storage (paragraph 14.2)
390/95-202-01	Closed	VIO - Examples of Failure to Accomplish Activities in Accordance with Documented Procedures (paragraph 8.2)
I.A.1.3.1	Closed	TMI - Shift Manning - Limited Overtime (paragraph 13.1)
I.A.1.3.2	Closed	TMI - Shift Manning - Minimum Shift Crew (paragraph 13.1)
II.B.4.2	Closed	TMI - Training for Mitigation of Core Damage - Completion of Training (paragraph 13.12)
II.E.1.2.1.B	Closed	TMI - Auxiliary Feedwater System Automatic Initiation and Flow Indication - Long Term (paragraph 13.13)
II.E.4.2.7	Closed	TMI - Containment Purge Valve Isolation on High Radiation (paragraph 13.14)
II.E.4.2 (1-4)	Closed	TMI - Containment Isolation Dependability Sub-Items 1-4 (paragraph 13.15)
II.E.1.2.2.B	Closed	TMI - Auxiliary Feedwater System Flow Indication - Long Term (paragraph 13.16)
II.E.3.1	Closed	TMI - Emergency Power Supply for Pressurizer Heaters (paragraph 13.17)
II.E.4.2.6	Closed	TMI - Containment Purge Valve Operability and MPA B-024, Containment Purging and Venting During Normal Operation (paragraph 13.18)
II.K.3.5.B	Closed	TMI - Automatic Trip of Reactor Coolant Pumps (paragraph 13.19)

II.K.3.10	Closed	TMI - Proposed Anticipatory Trip at High Power (paragraph 13.20)

16.0 LIST OF ACRONYMS AND INITIALISMS

ABGTS	Auxiliary Building Gas Treatment System
ACAS	Auxiliary Control Air System
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOI	Abnormal Operating Instruction
ASME	American Society of Mechanical Engineers
AUO	Assistant Unit Operator
AWG	American Wire Gauge
AWS	American Welding Society
CADAM	Computer Augmented Drafting and Manufacturing
CCP	Centrifugal Charging Pump
CFR	Code of Federal Regulations
CID	Component Identification
DBD	Design Basis Description
DCN	Design Change Notice
DN	Deficiency Notice
DS	Design Standard
EAI	Engineering Administrative Instruction
EDG	Emergency Diesel Generator
EGR	Electro-Hydraulic Valve Actuator
EMS	Equipment Management System
EOP	Emergency Operating Procedure
EQ	Environmental Qualification
ERCW	Essential Raw Cooling Water
ESFAS	Engineered Safeguards Features Actuation System
GL	Generic Letter
GO	General Operating
GOI	General Operating Instruction
gpm	gallons per minute
GSI	Generic Safety Issue
HERS	Harsh Environment Record System
HFT	Hot Functional Testing
IEB	Inspection and Enforcement Bulletin
IEEE	Institute of Electrical and Electronics Engineers
IEP	Inspection and Examination Procedure
IFI	Inspection Followup Item
II	Incident Investigation
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
IST	Integrated Safeguards Testing
JTG	Joint Test Group
LOCA	Loss-of-Coolant Accident
MDAFW	Motor-Driven Auxiliary Feedwater Pump
MOV	Motor-operated Valve

MPA	Multi-Plant Action Item
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation, Office of (NRC)
NUREG	(NRC) technical report designation
OSC	Operations Support Center
PAC/AQ	Program for Assurance of Completion and Assurance of Quality
PACR	Potential Area of Concern/Recommendation
PAI	Plant Administrative Instruction
PCV	Pressure Control Valve
PER	Problem Evaluation Report
PERP	Plant Event Review Panel
PMT	Post-Maintenance Testing
PMTI	Post Modification Test Instruction
PORV	Power-Operated Relief Valve
PQT	Procedure Qualification Test
PTI	Preoperational Test Instruction
psig	pounds per square inch gauge
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
SD/DC	System Description/Design Change
SER	System Evaluation Report
SI	Surveillance Instruction
SOI	System Operating Instruction
SPOC	System Preoperation Checklist
SSER	Supplemental Safety Evaluation Report
SSPS	Solid State Protection System
TCV	Temperature Control Valve
TI	Temporary Instruction
TMI	Three Mile Island
TS	Technical Specification
URI	Unresolved Item
USI	Unresolved Safety Issue
V	Voltage
VDC	Voltage Direct Current
VM	Vendor Manual
WBN	Watts Bar Nuclear Plant
WBS	Welding, Brazing, and Soldering
WO	Work Order
WR	Work Request