



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

Report Nos.: 50-327/88-29 and 50-328/88-29

Licensee: Tennessee Valley Authority
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 1101 Market Street
 Chattanooga, TN 37402-2801

Docket Nos.: 50-327 and 50-328
 Facility Name: Sequoyah Units 1 and 2

License Nos.: DPR-77 and DPR-79

Inspection Conducted: June 20 - July 8, 1988

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SUMMARY

Scope: This special, announced inspection was conducted for the purpose of a Safety System Quality Evaluation for the Containment Spray system and included a review of the TVA Nuclear Performance Plan functional corrective action areas identified in the Sequoyah Unit 1 and Unit 2 restart program matrix. The inspection consisted of an in-plant review in the mechanical, electrical, civil, structural, and instrumentation and control disciplines in order to verify that the CS system as currently constructed and installed is in accordance with the licensed design bases, system design specifications, applicable drawings, system modifications and temporary alterations. In addition, the operational capability of the CS system was evaluated by reviewing the system operating instructions and procedures, surveillance and testing requirements, corrective and preventive maintenance activities, human factors, emergency operating instructions and operator training. The inspection team evaluated, on a sampling basis, portions of the TVA Nuclear Performance Plan functional corrective action areas.

Results: Based on a review of the Containment Spray System there appears to be adequate program implementation in the following areas to support Unit 1 startup without further detailed NRC inspection:

- Design Basis Verification Program
- TVA As-Constructed Walkdowns
- Drawing Control Program
- Inplant Configuration Control and System Alignment
- Surveillance Instructions
- ASME Section XI
- Restart Test and Functional Performance Program
- Design Change and Modifications Programs
- Cable Routing and Cable Loading
- Equipment Qualification and Seismic Programs
- Preoperational Test Program
- Employer Concerns
- CAOR Including the QA Audit Process
- PRO and LER
- NER
- Instrument Line Slope
- System Operating and Emergency Operating Instructions
- Alternate and Rigorous Support Analysis
- Maintenance (including Trending, Material Control, Preventive Maintenance and Housekeeping)
- Operability Lookback
- Platform Thermal Growth

- Cable Tray Supports
- Welding (including Pipe, Structural, and Civil)
- Operator Training

However, some of these areas will be included in a scheduled operational readiness inspection.

Nuclear Performance Plan implementation requiring additional NRC review is as follows:

- Critical Calculations Regeneration Program (as part of violation 327, 328/88-29-01 response)
- Appendix R
- Electrical System SER-Related Issues
- Functional Test Observation of Pump Flow and Component Logic
- 10 CFR 50 Appendix J Testing

Within the areas inspected, the following violations were identified:

- 327,328/88-29-01: Incomplete Design Basis Calculation (paragraph 1)
- 327,328/88-29-02: Structural Walkdown Issues (paragraph 6)
- 327,328/88-29-03: Maintenance of Safety-Related Electrical Equipment (Paragraph 2)
- 327,328/88-29-04: Inadequate procedures (paragraph 1).

The violations were determined to be Sequoyah Unit 1 related.

Two Unresolved Items (URIs) were identified:

- 327,328/88-29-03: Containment Spray Check Valve Testing, paragraphs 1.h.(2)
- 327,328/88-29-06: System Design Deficiencies

Resolution of items 327,328/88-29-01 through 06 is necessary prior to the startup of Unit 1. These two URIs are Sequoyah Unit 1 startup related.

Deficiencies: Several deficiencies were identified within the report. These issues do not constitute programmatic issues, violations or deviations and because of their low safety significance, are not required to be resolved prior to the startup of Unit 1. These deficiencies are being identified for completeness and their resolution could improve overall plant efficiency and performance.

Commitments: The licensee committed to the following actions during the exit conducted on July 8, 1988:

- to test the CS pump flow characteristics including a multiple point test prior to the startup of Unit 1;

- to test the ESF pump valve logic performance as demonstrated in surveillance instruction SI 68 prior to the startup of Unit 1;
- support NRC review of a new TS indicating the 143 psid required to insure 4750 gpm flow from CS pumps prior to the startup of Unit 1. Verify this parameter prior to the startup of Unit 1;
- determine what the actual values are for heat exchanger differential pressure in order to resolve restart test functions 72-003 and 72-018 prior to the startup of Unit 1; and
- include in the next scheduled update of the CS training lesson plans information on the manual swapover of the CS system and the interlocks associated with the system. This issue was determined not to be startup related.

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

REPORT DETAILS

1. Mechanical Inspection

The design aspects of the inspection evaluated the system and components against applicable standards, the references cited and listed in this report, and the SYSTEMS/design basis reports for the Unit 1 CS system.

Inspectors performed a walkdown of portions of the Unit 1 CS system and performed a comparison between the as-constructed drawings and the actual installed system. The walkdown was conducted on system piping, valves, and components inside containment, the annulus, and the auxiliary building. Additional information for the walkdown was drawn from isometric details, design documents, and vendor data packages.

a. Conformance of the Containment Spray System With the As-Constructed Drawings

A walkdown of portions of the CS system was performed in the auxiliary building, annulus, and containment comparing the installed system with drawings 47W437, sheets 1-6, and 47W812, sheet 1. The following system attributes were considered during the walkdown and drawing review:

- Pipe sizing and class
- Reducers
- Flanges/fittings/spool pieces
- Location of vents, drains, thermowells
- Isometric routing
- Unidentified/undocumented valves, pipes, instrumentation
- Interferences
- Support/restraint location
- Valve flow directions

The inspectors identified no major discrepancies during the system walkdown. The discrepancies observed by the inspectors had previously been identified by the licensee prior to the inspection as part of the OSLA 107 walkdown program, implemented as part of the licensee's SSQE inspection preparation, and did not affect system operability.

b. Associated System Interfaces

A verification was performed of the following associated system interfaces with the CS system, both on the drawings and on the installed system:

RHR HX 1A & 1B to CS (drawing 47W811, sheet 1)

ERCW to CS HX A & B (drawing 47WC45)

Boric Acid Blender to CS (drawing 47W809, sheet 2)

CS Pumps 1A & 1B suction relief valve discharge (drawing 47W811, sheet 1)

CS Trains 1A and 1B suction to containment sump (drawing 47W811, sheet 1)

The inspectors noted that several skid mounted valves supplying component cooling water for cooling of the CS pump mechanical seal and the oil bearing cooler were unlabeled and not on the flow drawings. TVA had previously committed in their response to NRC Inspection Report 327,328/87-52 to add skid mounted valves to the SI and SOI checklists prior to Unit 1 startup. These valves were identified on SOI checklists 72.1A-1 and 72.1A-2 for CS Pumps 1A and 1B, respectively, with the valve numbers listed as N/A. During NRC system alignment inspection 327,328/87-66, TVA had labeled all Unit 2 skid mounted valves with tags having descriptions matching those in the SOI checklists. This was necessary to ensure that the operators using the checklists would position the proper unnumbered skid valve. Since this had not yet been accomplished for Unit 1, the inspectors obtained a commitment from TVA to label all Unit 1 skid valves with descriptive tags prior to establishing configuration control for Unit 1 restart. This commitment is being tracked under Violation 327,328/87-52-01 corrective action.

c. Shield Building Penetrations

The inspector reviewed the following Unit 1 shield building mechanical penetration seals:

Penetration 1X-48 B at Elevation 729'
Penetration 1X-49 B at Elevation 729'

Penetration 1X-48 B is a 16 inch pipe sleeve which accommodates the 12 inch line to CS header 1-B. Penetration 1X-49 B is a 12 inch pipe sleeve which accommodates the 8 inch line to RHR spray header 1-B.

The mechanical seal penetrations are shown schematically on TVA drawing No. 47W812-1, Flow Diagram/Containment Spray System, Revision Y, dated April 11, 1988. Penetration 1X-48 B is shown on TVA drawing No. 47W437-5, Mechanical Containment Spray System Piping, Revision F, dated October 21, 1985. Penetration 1X-49 B is shown on TVA drawing No. 47W437-4, Mechanical Containment Spray System Piping, Revision D, dated April 1, 1980.

At the time of the inspection, these penetrations were being modified in accordance with the boot seal detail shown on sheet 99 of ECN L7382B. (For penetration seals above elevation 724' which is the flooding level). The seal type was designated as Category F. Category F penetration seals are defined as seals with thermal

movements which exceed 1/4 inch, and with installed configurations which allow for the degradation of the fire and pressure barriers. ECN L7382 requires re-booting these penetrations before Unit 1 restart with fire and pressure rated boots which can accommodate maximum thermal and Safe Shutdown Earthquake (SSE) pipe movement

For this modified mechanical seal penetration detail, the inspector evaluated the design basis loads, and the qualification of the penetration materials and boot assembly to the design basis loads.

The penetration assemblies are subject to the following design basis loads:

- Fire
- Radiation
- Environmental temperature and pressure
- Pipe fluid operating temperature
- Piping movements due to thermal and SSE

TVA was able to provide the inspector with copies of environmental drawings and test reports to confirm that the seal assembly is qualified to the above design basis loads, with the following exception.

TVA did not have readily retrievable documentation to confirm that the penetration seal assembly materials were qualified to the total 40 year integrated dose of 10^6 rads specified on TVA drawing No. 47E235-51, Revision B, dated October 18, 1984, or to the 400°F pipe fluid design temperature specified in the table on sheet 95 of ECN L7382B. TVA asked Insulation Consultants and Management Services, Inc., to provide the appropriate qualification documentation, and was able to provide the inspector with a comparable document which ICMS prepared for the same penetration seal materials used at another plant.

To show that penetration seal assemblies 1X-48 B and 1X-49 B are qualified to the negative (0.5 inch water) annular pressure differential and the transient tornadic differential pressure drop of 3 psi specified on TVA drawing 47E235-51, TVA provided the team with ICMS report No. HT-M05-34, Hydrostatic Test for Mechanical Boot Seals, dated May 22, 1986. The ICMS report summarizes a 2-hour hydrostatic test conducted for a 2-inch pipe/10-inch sleeve to a maximum hydrostatic pressure of 28 psi, to confirm the ability of the mechanical penetration seal assemblies that are installed below elevation 724 feet to withstand the design basis flood. This test condition would envelope the design pressure environment for penetrations 1X-48 B and 1X-49 B. The TVA technical staff have indicated that penetration seal assemblies installed below flood level are subject to a maximum differential hydrostatic pressure of about 18 psi.

It was noted that TVA had not considered the axial thrust induced in the pipe due to the differential hydrostatic pressure on the seal which must be restrained by the pipe supports adjacent to the penetration. For penetrations 1X-48 B and 1X-49 B the loads would be small, but for penetrations subjected to 18 PSI the loads are higher. As an example, a penetration assembly which consists of an 8-inch pipe and a 12-inch sleeve appears capable of generating a 4-5 KIP thrust due to a hydrostatic pressure of 18 psi. During the course of the review it was found for penetrations with significant dP across them that TVA did not account for the additional axial load imparted to the pipe by the loaded area of the seal. This issue is designated URI 327,328/88-29-06, (Example a). Adequate resolution for the above URI will include Engineering Assurance review and approval of the design documentation and requires resolution prior to the startup of Sequoyah Unit 1.

TVA has indicated that Construction Technology Laboratories Report, Fire and Hose-Stream Tests for Penetration Seal Systems (NMP2-PSS6), dated March 1986 qualifies the penetration assembly to the 3-hour fire barrier requirement imposed by 10 CFR 50, Appendix R.

In addition, the penetration assembly boot material has been proportioned to accommodate the radial and axial pipe movements due to thermal and SSE movements which are listed on sheet 95 of ECh L7382B.

The inspector's conclusion is that shield building penetrations 1X-48 B and 1X-49 B meet the design bases. Documentation was initially not available within TVA to justify that the penetrations were qualified to meet the radiation and temperature design bases. Documentation was generated for the inspectors and appeared to be adequate.

d. System Pressure Boundaries

Section 2.1.6.a, Systems Integrity for High Radioactivity, of NUREG 0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, dated July 1979, requires that licensees implement a program to reduce leakage from systems outside containment that includes: 1) Immediate leak reduction by implementing all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment and measuring actual leakage rates with system in operation and reporting them to the NRC; and 2) Continuing leak reduction by establishing and implementing a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

To assess TVA's implementation of these NUREG requirements, the inspector reviewed the design changes and surveillance criteria which TVA prepared and implemented for the CS system.

TVA letter L51 791031 913 dated November 1, 1979, established the initial guidelines which the TVA technical staff used to implement the NUREG requirements.

ECN No. 2586 installed welded nipples and threaded caps on a number of drains, vents and test valves in the CS system and other systems. TVA made these design changes by reviewing the as-designed flow diagrams and identifying drains and vent lines which did not have a secondary boundary. For the CS system, the design changes prepared under ECN 2586 were incorporated into the following as-designed TVA drawings:

- TVA drawing No. 47W812-1, Flow Diagram/Containment Spray System, Rev. 9, dated September 18, 1979.
- The following TVA mechanical CS system piping drawings:
 - 47W437-1, Rev. 17, dated September 12, 1979
 - 47W437-2, Rev. 14, dated September 12, 1979
 - 47W437-5, Rev. 10, dated September 12, 1979
 - 47W437-6, Rev. 11, dated September 12, 1979

Field Change Request SQ-FCR-001 was prepared on February 23, 1980 to revise Rev. 9 of the flow diagram when a subsequent review of the drawing indicated that not all of the design changes had originally been incorporated into the drawing.

The inspector confirmed that the comparable as-constructed flow diagram and piping physicals indicated the addition of these nipples and caps.

On April 2, 1980, TVA provided the NRC with the surveillance procedures to be used to monitor system leakage (A27 800402 008). Surveillance instruction procedure SI-632.0, Auxiliary Building Combined Systems External Leakage, Rev. 0, dated January 17, 1980, documents the combined external leakage to the auxiliary building which is monitored by the separate implementation of system-specific surveillance procedures such as SI-632.1, Auxiliary Building Containment Spray System External Leakage, Rev. 0, dated January 17, 1980.

TVA provided the NRC with the results of the leakage tests for the Unit 1 systems monitored outside of containment. Supplement No. 5 to the SER dated May 1981 indicates that the results of the tests which TVA submitted to the NRC for Unit 1 were satisfactory. The inspectors concluded that TVA's actions to implement NUREG 0578 Section 2.1.6.a for the CS system were satisfactory.

e. System Alignment

The inspectors reviewed SOI checklists 72.1A-1 and 72.1B-1 for adequacy and conformance with the system drawing. The inspectors verified that the system was either aligned per the SOI checklists or the valve position was documented in the configuration log. During the walkdown the inspectors noted that valves 72-515, 522 and 524, which are reach rod operated valves, had hold order tags attached indicating that the valves were open when the remote position indicators showed that the valves were shut. The inspectors verified locally that the valves were actually open. Through discussions with the licensee it was determined that the problem with the reach rod indication had previously been identified by the licensee as part of an ongoing effort to identify and correct problems with reach rod operated valves throughout the plant. The licensee currently requires operators to verify valve position locally on reach rod operated valves as well as through remote indication.

f. Component Marking and Accessibility

A verification was performed to ensure that the equipment identification, tagging, and nomenclature used in the CS system was consistent with drawings and procedures. TVA has a tagging/labeling program in progress. Components necessary for the operation of the system were determined to be accessible and adequately identified. Minor discrepancies noted by the inspectors had been previously identified in the TVA program.

g. Material Traceability

The inspectors verified that the name plate data for both containment spray heat exchangers, both CS pumps, both CS pump motors, and pump suction relief valve 1-72-513 were in accordance with vendor data packages and design documents. The inspectors verified that the size imprint on the recently modified orifice plate in the header piping to the spray nozzles agreed with the size specified on ECN L7381A and work plan 7381-01. The ECN and work plan resized the orifice and replaced the orifice to a full flow pipe size.

h. Surveillance Requirements, Emergency Operating Procedures, and Functional Testing

(1) The following equipment surveillances and surveillance records were verified to support the requirements of the TS as noted:

*SI-34, Containment Spray System Valve Position Verification

SI-37.1, Containment Spray Pump 1A-A Test, Unit 1

*SI-37.2, Containment Spray Pump 1B-B Test, Unit 1

SI-158.1, Containment Isolation Valve Leak Rate Test, Unit 1

SI-166.39, Disassembly and Inspection of SIS/RHR/CS/UHI Check Valves During Refueling Outages, Unit 1

SI-186, Locked Valve Position Verification Per NRC Commitment, Containment Inspection, Unit 0, Unit 1 (Note: Unit 0 is a designation for a common system)

*SI-267.72.1, Functional Pressure Test of Containment Spray System, Unit 1

IMI-99 RT-16.6, Response Time Test Procedure of Containment Pressure Channels I and II

IMI-99RT-643B, Response Time Testing Engineered Safety Feature Actuation Slave Relay K643

SI-166.1, Full Stroking of Category "A" and "B" Valves During Operation

SI-166.3, Full Stroking of Category "A" and "B" Valves During Cold Shutdown

SI-166.15, Containment Spray Check Valve Test Performed During Operation

SI-251.1, Channel Calibration of Class 1E Motor Operated Valve Overload Relay Heaters

*SI-68, Functional Test of Containment Spray Pumps and Associated Valves

SI-138, Containment Spray - Spray Nozzle Test

SI-2, Shift Log

*The inspector field verified the appropriateness of these procedures.

Through a mixture of field inspection and review of the last test performance, the inspectors determined that the tests listed above met the following surveillance requirements:

- 4.6.2.1.6
- 4.6.2.a.c.1
- 4.6.2.1.c.2
- 4.6.2.1.d
- 4.3.2.1.1.d.2a (In Part)
- 4.3.2.1.3 Table 3.3-5 (In Part)
- 4.3.2.1.1.A.2.c
- 4.6.1.2.d (In Part)

The inspector reviewed SI-138, Containment Spray - Spray Nozzle Testing. Step 6.2.3 in the procedure requires the operator to:

Close valve 72-545 upon completion of CSH "A" nozzle verification. If testing of Train "B" is not to be performed immediately following Train "A", shut down the hot air compressor to avoid heating/pressurizing the piping.

Step 6.2.4 of SI-138 then states:

Open valve 72-546 and start hot-air compressor (if necessary). This will allow air flow through CSH "B".

Appendix A of SI-138 gives the following recommendations for Air Compressor Rental:

Supplier: Atlas Copco Comtec, Inc.
2346 Mellon Ct.
Decatur, GA 30035

Description: Compressor, Air, 100% Oil-free Air, @ 300 degree F. 1500 CFM @ 125 psig max., with relief valve set @ 100 psi.

Lead Time: 5-7 weeks
Previous Requisition: 453185

CAUTION: If compressor furnished does not have relief capability, appropriate measures should be taken to ensure relief capability is provided or steps shall be taken to prevent potential overpressurization of piping by rearrangement of procedure steps using temporary change forms per AI-4.

The closure of Valve 72-545 in step 6.2.3 isolated the CS system piping and applies full compressor air pressure to a section of 100 psi rated pipe. Therefore, until the opening of 72-546 in step 6.2.4, the CS piping is relying on the compressor relief valve for protection. If the compressor does not have relief capability, the piping will overpressurize. It is imprudent at best to close 72-545 until 72-546 is open in either case.

Revision 7, the current revision to SI-138, requires the nozzles to be inspected by use of an infrared camera to verify that each of the 312 nozzles are open and pass air freely. Results for all unobstructed nozzles are documented by checking one blank. The procedure also states, "If desired, photos may be made for future reference."

If one or more nozzles are nonfunctional, the number of nonfunctional nozzles are recorded and a sketch is made of the location of each. Verification is made by the Test Director and a second party.

The technique used to conduct the surveillance requires the test director to observe the nozzles for the lack of an infrared signal. Previous revisions of the procedure required each nozzle to be identified and checked off that a positive infrared signal exists.

Thus the previous revisions of SI-138 required the director to look for a positive signal as opposed to the lack thereof. The previous revisions also created more auditable records of the inspections by requiring the test director to document that each nozzle flowed freely. TS Surveillance Requirement 4.6.2.1.1, requires that each CS spray train shall be demonstrated OPERABLE by verifying each spray nozzle is unobstructed. The verification for each spray nozzle should be documented.

Revision 7 of SI-138 has never been performed. Therefore, all previous tests have included appropriate procedures and documentation. This SI should be revised to include more appropriate documentation prior to its next required performance.

During the review of SI-274.900, Engineered Safety Feature Response-Time Verification, the inspector became aware of a potential problem with the response timing of the Containment Spray actuation system caused by inaccuracies in the Agastat relay. The Agastat relay is a 0-300 second timing relay used in the sequencing of EDG loads. The vendor stated accuracy of the relay is plus or minus 5% for a specific repeatability. The licensee stated that the actual accuracy over a test range was about plus or minus 10%. Even though the relay is capable of operating over an entire range, it is operated in the CS system only at a specific point. Therefore, the inspector requested the licensee to field verify the accuracy of the relay for its specific application in the Containment Spray System.

In response to this request, on June 29, 1988, the licensee performed a bench test of one of the Agastat relays for installation in Unit 1. The relay was calibrated for 180 seconds. The relay was independently measured and its repeatability determined to be accurate within 1% which was considered acceptable.

- (2) The inspector reviewed the following Emergency Procedures for the Containment Spray System:

E-0, Reactor Trip or Safety Injection
 E-1, Loss of Reactor or Secondary Coolant
 ES-1.2, Transfer to RHR Containment Spray

ES-1.2 required operators to verify the CS pump suction to the containment sump at an RWST level of less than or equal to 8% indicated level. The procedure recommended that the operator verify ECCS lineup prior to this swap, if time allowed. The safety analysis for containment pressure control assumes that swapper occurs before ice melt, therefore the time dependence of this swapper was questioned. The inspector determined that in all cases the swapper should occur prior to complete ice melt.

No violations or deviations were identified.

- (3) The inspector questioned the adequacy of the testing of valves 72-547 and 72-548 in that they are not type "C" leak rate tested per 10 CFR 50 Appendix J. The adequacy of the containment isolation design with respect to GDC-56 was reviewed by the staff during the review of the nuclear performance plan and is documented in the May 1988 SER. The inspector will review the leak rate testing of these valves during future resident inspection activities. This item is identified as URI 327,328/88-29-05.
- (4) A sample of the records for the following valves were examined to assure that inservice testing and MOV thermal overload protection requirements were met. These requirements are contained in the FSAR (6.2 and 9.2), TS (4.8.3.2, 4.0.5, 3.6.2.1), 10 CFR 50 Appendix A (General Design Criteria - Section V) and ASME Section XI (IWV).

Valves: 1-FCV-72-2, 13, 20, 21, 22, 23, 34A, 39, 40, 41
 Check Valves: 1-72-506, 507, 547, 548, 555, and 556

In an SER issued in May 1988 (NUREG 1232, Volume 2) the NRC stated, "Since certain penetrations, including the containment spray and RHR spray, are part of the systems required to operate following an accident, it is imprudent to follow the explicit requirements of GDC 56 and automatically isolate or lock closed the isolation valves. In those instances where post-accident operation is required, remote manual valves are acceptable for meeting the GDC as described by SRP section 6.2.4 and the ANSI standard. For the containment spray and RHR spray line penetrations, TVA has identified additional outboard valves that have remote manual closure capability as containment isolation valves. The designation of those valves as containment isolation valves brings the isolation design for these penetrations into compliance with the staff guidelines for meeting GDC 56 contained in the SRP."

The system is provided with a check valve inside containment (1-FCV-72-547 and 1-FCV-72-548) and a "remote manual" isolation valve outside containment (1-FCV-72-2 and 1-FCV-72-39) for each spray header.

The licensee requested and was granted relief in April 1985 (SER) from exercising valves 72-547 and 72-548 (containment spray header check valves) in accordance with the requirements of ASME Section XI, contingent upon providing a method for verifying full flow capability of the valves. Testing these valves with water would deluge containment, causing potentially significant damage and cleanup requirements to equipment and structures. The licensee proposed testing these valves with air during the spray header nozzle test required by TS 4.6.2.1 at least once every five years. The NRC position stated that this method could not ensure full stroking of the CVs. As an alternate to full flow testing, one of these four CVs will be disassembled each refueling outage on a rotating basis. If any valve is found to be inoperable and the cause determined to be potentially generic, the other valves must also be disassembled and inspected before being declared operable.

The disassembly of these valves is performed under SI-166.39, Disassembly and Inspection of SIC RHR/CS/UHI Check Valves During Refueling Outages, Unit 1. The inspector reviewed documentation on the last performance of this SI dated May 1, 1986 and found it to be acceptable.

SI-158.1, Containment Isolation Valve Leak Rate Test, verifies that valves 1-FCV-72-2 and 1-FCV-72-39 have acceptable leakage rates for containment isolation. The latest performance of this SI dated September 9, 1985, was reviewed and found to be acceptable.

The inspector reviewed the status of thermal overload protection devices installed in the containment spray system MOV motor starters. All thermal overload protection devices were removed or bypassed with the exception of those in starters associated with valves 1-FCV-72-20, 21, 22, 23, 40 and 41. These devices are tested in accordance with SI-251.1, Channel Calibration of Class 1E Motor Operated Valve Overload Relay Heaters. This SI implements the requirements of SR 4.8.3.2.

The inspector reviewed documentation of the most recent performances of SI-251.1 on valves 1-FCV-72-20 and 1-FCV-72-21 and found these tests to be acceptable.

The inspector also randomly selected valves and verified that they were included in the Section XI program and currently tested per that program.

- (5) The following design basis required functions were reviewed to determine if surveillance or other functional testing adequately documents the ability of the CS system to meet the design function.

The operability of the CS pump protective circuit was evaluated. This circuit protects the pump by allowing pump discharge to be circulated back to the pump intake if flow in the discharge line drops below that required for pump protection (1650 gal/min) as measured by flow elements FE-72-34 or 13, or if upon starting flow is not achieved in the spray header within a preset time interval (10 seconds). It was determined that construction and calibration criteria were established. This system capability was tested in TVA-21B and again in WP 12358 following modifications.

A review for the existence of an interlock between FCV-72-23 and 22 for CS train "A" and FCV-72-20 and 21 for CS train "B" was performed. The function of this interlock is to prevent the CS pump from taking suction from the RWST and the containment sump at the same time. This item is discussed further in Section 1.(j) of this report.

Automatic activation of the CS system is based on activation of two out of four of the containment hi/hi pressure switches. The inspectors requested to observe a surveillance which would demonstrate this system function. Due to the CS system being drained for maintenance, these SIs were not performed during the inspection period. It will be necessary for two testing functions to be observed prior to the startup of Unit 1:

- Pump flow characteristics including a multiple point test as well as the performance of the current revision of SI-37.1.
- ESF pump/valve logic performance as demonstrated in SI-68.

1. ASME Code Section XI Testing

The inspector evaluated the implementation of the Section XI testing program for the 1A-A CS pump for consistency with TS requirements and design requirements.

The current TS SR 4.6.2.1.1.b requires that the licensee verify that, "each containment spray train shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 140 psig when tested pursuant to TS 4.0.5."

Baseline flowrate data for the Unit 1 containment spray pumps will be established during future performances of SI-37.1 and SI-37.2. These have not been established in the past because it is not a regulatory requirement and measurement of pump flowrate is not required by the 1974 edition through the summer of 1975 addenda to the ASME Boiler & Pressure Vessel Code Section XI (Code of record for Sequoyah).

Previous tests of the pumps met the allowable ranges of inservice test quantities identified in Section XI and during baseline tests. These tests also verified the letter of the TS. However, these tests never verified that 4750 gpm would be supplied to the spray nozzles during an accident as assumed in the design basis calculations. This flowrate is required for the system to meet its design basis.

The licensee has submitted a TS change request to require that the pumps be tested to deliver 4750 gpm at a dP of 143 psid.

Following the performance of the current revision of SI-37.1 the plant will be ready for restart with respect to Section XI and surveillance testing for the 1A-A CS pump.

10 CFR 50.55a(g) requires inservice testing of pumps and valves in accordance with ASME Section XI to verify Operational Readiness. ASME Section XI, IWV-2100 defines relief valves as category C valves and ASME Section XI, IWV-3511 requires category C valves to be tested on at least a five year interval in accordance with Table IWV-3510.1.

Contrary to the above, the licensee failed to include ASME Section XI requirements for testing of the containment spray system suction relief valves (72-512 and 72-513) in the instructions for inservice testing which are provided in Section 6.8 of the Sequoyah Final Safety Analysis Report. These valves were however, tested in other surveillances and were maintained operable. This is violation 327,328/88-29-04 example 2.

j. Functional Design Parameters

The following sample of functional design parameters was reviewed during the inspection.

- (1) The design parameters of the CS piping are shown on drawing 47W812-1, Rev. 16 (Reference 1, report section 1.k). The inspector reviewed the calculations which determine these design parameters for the piping. These calculations are References 2, 3, 4 and 5.

Reference 2 was prepared on May 27, 1988. References 3, 4 and 5 were prepared in June 1988.

The inspector reviewed these references. The review revealed that several pressure and temperature (design conditions) boundaries as shown in Reference 1 were incorrect. The results of the review are as follows:

- ° The calculation of Reference 2 was performed as part of ECN L6673, dated June 17, 1986. It changes the design conditions of the lines from the flow restrictor downstream of RWST to valves FCV-72-21 and FCV 72-22 to 40 psi and 150 Degrees F. Reference 1 depicts the old conditions which were 100 psi and 100 Degrees F. The parameters 42 psi and 150° degree F were added to the design conditions in design condition No. 5 of Reference 2.
- ° Reference 3, addresses the pump suction, discharge, miniflow and test lines. This calculation, completed during the inspection, identified that the design condition boundaries shown at valves 72-503 and 72-504 are incorrect because the pressure just downstream of these valves could be 170 psig, which is higher than the current 100 psig rating. The design condition boundary will be moved to valve 72-502.
- ° Reference 4, addresses the containment spray ring headers and lines downstream of the isolation valves FCV-72-2 and FCV-72-39. This calculation identified that the design condition boundary should be moved from the outlet of these valves to the inlet side of the containment penetration since the pressure at the outlets of these valves could be 127 psig, higher than the current 100 psig rating.
- ° Reference 5, addresses the RHR spray ring headers and lines downstream of isolation valves FCV-72-40 and FCV-72-41. This header and piping are considered part of the Containment Spray system. The calculation identified that the design condition boundary should be moved from the inlet side of these valves to the inlet side of the penetration since the pressure at the penetration will not exceed 100 psi. By implementing the change, the penetration will be at design conditions which are in agreement with its nameplate rating (100 psig).

CAQR SQP 880387, Revision 0, was written on June 24, 1988, to address the discrepancy between Ref. 1 (Fig. 6.2.2-2 of the Sequoyah FSAR which currently shows R12) and the nameplate rating of the pressure of the fluted heads for containment penetrations X49A and X49B. The 100 psi nameplate rating was found as a result of a system walkdown performed by TVA on December 13, 1987. The results of Reference 5 indicate that the 220 psi rating is not required as the design pressure of the steel containment

penetration and that the maximum sustained operating pressure per ANSI B31.1 is below 100 psi.

Therefore, this is considered a documentation problem and not a component deficiency. This CAQR is applicable to both Units 1 and 2. Due to the above mentioned discrepancies, the following corrective actions were recommended in the CAQR:

- Perform calculations to support design parameters on drawings 47W812-1, R16 (Reference 1).
- Resolve any discrepancies identified between values listed on drawings and results of the calculation.
- Reevaluate adequacy of components, revise drawings, determine impact on pipe analysis, and verify hydrotest records as necessary if design parameters on the flow diagram cannot be supported by calculations.
- Determine if other design calculations for pressure and temperature are missing on other systems in order to establish and resolve the full extent of problem.

An ECN/DCN will be prepared by TVA to update the design documentation to reflect the changes addressed above and will be completed prior to Unit 1 restart.

The failure to have pressure and temperature calculations to define pressures and temperatures at various points in the Containment Spray System is considered to be a violation of 10 CFR 50, Appendix B, Criterion III, Design Control, and is identified as violation 327,328/88-29-01, Design Basis Calculations.

The missing pressure and temperature calculations were re-generated during the inspection. As a result of the new calculations, several components and associated piping are in a higher pressure rating. TVA is currently assessing the effect of these changes. As of June 27, 1988, no hardware had been identified as affected by the resulting shift in the location of pressure boundaries. This issue is designated as URI 327,328/88-29-06 Example b., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

As part of assessing whether fluid flow acceleration or deceleration (water hammer) in the CS system has been considered by TVA in terms of its resulting dynamic loading of the system, the inspector reviewed a study entitled, Evaluation of Fluid

Dynamic loads on the Containment Spray System, dated May 26, 1987. This study is incorporated as Appendix A to problem Number 0600104-01-02 Containment Spray System, Units 1 and 2, Sequoyah Nuclear Plant. The lack of existence in 1987 of such a calculation and its subsequent generation by TVA was addressed in Inspection Report 50-327, 328/87-28.

The following are the results of this calculation review:

- (a) The methodology used in the calculation is simplistic with potentially inaccurate results.
 - (b) The derivation of maximum force and rise time formula on page 7 of Appendix A is not given; however, based on similar studies, it appears that the magnitude of the force is reasonable.
 - (c) The derivation of the maximum load on the ring header is not given. It is stated that the maximum load occurs at the time that one-half the header is filled. The inspector considers that both the magnitude and time of occurrence are incorrect. Similar studies have shown that the maximum load on the header could be one order of magnitude higher than the one calculated in Appendix A. Moreover, the time of each occurrence is the time at which the two water slugs which fill the the header in a symmetric fashion from the two opposite ends meet each other. A comparison of the support loads due to water hammer versus other loads is given in Appendix A. Although accurately calculated water hammer loads may still be substantially smaller than other loads on the system, there may be support locations where such loads are not negligible. Appendix A, indicates that the CS System is scheduled for reanalysis following Unit 2 restart. The inspectors consider that a more accurate methodology for calculating water hammer loads should be used. This issue is designated URI 327,328/88-29-06 Example c and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.
- (2) A review of sample piping runs was performed. Two pipe stress analysis problems, 0600104-01-02 and N2-72-1A & 2A, were selected for review using the piping detailed on physical drawings 47W812, Sheet 1 and 47W437 Sheets 1-6. Also included in the review were all outstanding deficiencies previously identified by TVA. These analysis problems were reviewed considering the walkdown attributes listed in Section 1.a in addition to the design and stress analysis items listed below which were verified.

- The stress isometric of record agreed with the current piping physical drawings.
- All pipe supports were identified on the stress isometric including type and direction.
- Equipment nozzle loadings were properly considered.
- Results of the latest system walkdown were considered and properly accounted for in the analysis.
- All anchor and restraint point displacements due to thermal and seismic effects were properly considered.
- Design input parameters such as temperature, pressure, pipe material and size, seismic anchor movements and response spectra were properly considered.
- Proper modeling considerations such as valve motor operator, system interconnection and overlap, elbow and tee type, flanges, concentrated masses, e'c., were made.
- All pertinent loading conditions were considered, including thermal deadweight, seismic, fluid dynamic, and steel containment vessel thermal displacement.
- Pipe stresses were within the specified allowables for all conditions analyzed.

Containment spray pipe stress problems N2-72-1A & -2A, Rev. 5, dated May 18, 1988, contained the analysis for piping routed from the containment spray pumps 1A-A and 1B-B discharge nozzles to the containment spray heat exchangers 1B and 1A intake nozzles. The system was divided into two problems N2-72-1A and N2-72-2A as shown on isometric 47K437-50. Problems N2-72-1A and N2-72-2A are not connected and do not overlap with any other piping system.

During the review of the above problems the following items were discovered. Page "a" of the summary of piping analysis N2-72-1A, 2A indicates that Rev. 5 voided page 12B; however, the page was not indicated as voided. TVA confirmed that page 12B belongs in the analysis of record package and will modify page "a" of the analysis accordingly. Another area which requires attention is that the piping analysis isometric of record for Unit 1 identifies pipe supports using Unit 2 support identifiers resulting in confusion when trying to review piping analyses. TVA currently has a program which should, in the near future, update the piping isometrics to reflect both the Unit 1 support identifiers and also include the current as-built system geometry and support locations.

These issues were identified as deficiencies and provided for licensee information.

On page A.18 of the calculation a value of 14 1/2 inches was measured in the field as the distance from the pipe center line to the weld location for a 12 inch long radius elbow. Since this elbow standard dimension should be 18 inches, either the field measurement was incorrect or the elbow was modified. TVA should review this discrepancy and take appropriate action. This issue is designated URI 327,328/88-29-06 Example d., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

During the review of the pipe stress analysis walkdown evaluation shown on page A.17 of the calculation an apparent anomaly in the reported as-built piping lengths was discovered. The walkdown results reported a total deviation of 5 feet, 5 and 1/4 inches between the as-analyzed and as-built dimensions of a length of pipe run from node point 211 to the control point of the elbow at point 215 as shown on drawing 47K437-50, R4. A field verification of this length performed by the NRC determined that no deviation exists. TVA has used these erroneous lengths in calculations to justify the adequacy of the as-built piping system and supports. TVA should review the walkdown data for this system and modify the calculations as required. This issue was identified as a deficiency and provided for licensee information.

Containment spray pipe stress problem 0600104-01-02, Rev. 14, dated May 18, 1988, contains analysis for piping routed from the outlet side of containment spray heat exchanger 1B thru steel containment vessel penetration 1X48B to the containment spray header 1-B. The system was overlapped with problems N2-72-3A and EM 0600104-01-01 as shown on pipe stress isometric drawing 0600102-01-02, Rev. 12.

During the review of the above problem the following items were identified which require further TVA action. On page B.26 of the summary of analysis for system 0600104-01-02, Revision 14, dated May 18, 1988, the evaluation of a pipe support location discrepancy did not consider the effects on the X-direction seismic restraint located at Node 120. The loading on the restraint at Node 120 would increase due to the new location of the adjacent X-direction restraint (CSH-31) located at Node 66. TVA should determine the load increase and evaluate its effect on the seismic restraint located at Node 120. Also, the effects of moving the support at Node 66 on the loading of the Containment Spray Heat Exchanger 1B has not been considered by TVA.

The as-analyzed length between Node 60 and Node 63 in the X-direction was 14 feet 6 inches and the piping physical drawings detail a length of 13 feet 6 inches. This issue is designated URI 327,328/88-29-06 Example e., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

In summary, from the design standpoint all items and attributes listed above, with the exceptions discussed, were determined to have been adequately addressed by TVA. The analysis of record for these piping problems based upon the attributes reviewed are considered adequate and meet FSAR and design commitments.

- (3) The team reviewed Containment Spray Pumps 1A-A and 1B-B to assess the design and procurement of these pumps with respect to FSAR commitments and design criteria.

The CS pumps are shown on the following as-designed TVA drawings:

- ° 47W812-1, Flow Diagram/Containment Spray System, Rev. 17, dated June 14, 1988.
- ° 47W437-1, Mechanical/Containment Spray System Piping, Rev. 24, dated May 3, 1988.

FSAR Section 6.2.2.2 specifies that each pump is rated for 4750 gpm flow at a design head of 370 feet. FSAR Table 6.2.2-1 specifies additional pump design parameters. FSAR Section 6.2.2.2 also details the functional requirements for the 700 HP pump motors.

The functional requirements for the pump and pump motor are reiterated in Design Criteria No. SQN-DC-V-27.5, Containment Spray System, Rev. 2, dated July 22, 1987.

The design parameters of the CS System pumps are provided in Reference 6. The calculation considers only flow from the RWST. Only one pump is assumed operational. The calculation shows that each of the pumps, 1A-A and 1B-B, must develop a head of 328.29' or 142.1 psi at its rated flow. This is lower than the manufacturer value of about 160 psi. A similar calculation for Unit 2 resulted in a required head of 328.88' (Reference 7). TVA could not provide a similar calculation for the required head when the pump takes suction from the sump during the recirculation mode for Unit 1. Since the RWST is at a higher elevation from the sump and the piping geometry on the suction side is different for the two cases (RWST vs. Pump) it could be expected that during recalculation the required head might be

higher. On the other hand, since the containment pressure exerted on the sump assists the pump during recirculation it is likely that the required pump head during recirculation will be smaller. Without a calculation it is not apparent which case might control. Consequently, the team considers that a calculation should be performed by TVA to document the head required under recirculation mode for Unit 1.

A Technical Specification change has been submitted to the NRC that will replace the requirement that "on recirculation flow, each pump develops a discharge pressure of greater than or equal to 140 psig" to the requirement that "on recirculation flow, each pump develops a differential pressure of greater than or equal to 143 psid at greater than or equal to 4750 gpm".

Moreover, Surveillance Instructions SI-37.3 and SI-37.4 for Unit 2 have been revised to reflect the 143 psid (Reference 12). Surveillance Instructions SI-37.1 and SI-37.2 for Unit 1 have not as yet been revised to reflect 143 psid, although it is stated in the revision log that the revised differential pressure was incorporated (Reference 13).

It appears that little or no margins have been incorporated in the calculations nor have the requirements of ASME Code Section XI been fully considered by the calculations used to set the required pressure differential across the pump. These issues are designated URI 327,328/88-29-06 Example t., and requires resolution prior to the startup of Sequoyan Unit 1. Adequate corrective action for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

In both References 6 and 7 the recommendation is made that a comprehensive pre-operational test be performed to establish a set of performance points for the pumps. The origin of this request stems from the test deficiencies experienced in the original pre-op program conducted in 1980.

This subject was addressed previously by the NRC through URI 327,328/87-50-03. Specifically, CAQR SQP 870860 was issued by the licensee to document the fact that the preoperational test for the CS pumps was not satisfied, in that, the head may not be adequate to provide the required system flow. The CAQR stated that preliminary analysis indicated that although the preoperational test for pump performance was not satisfied, the impact on containment integrity was minimal. Initially, the reportability of this CAQR and supporting potential reportable occurrence (PRO) report was determined to be "indeterminate." The CAQR was later determined, after approximately two months of engineering evaluation, to be reportable. The inspector determined that the licensee has scheduled the technical issue

for resolution prior to plant restart. However, the use of the term "indeterminate" for situations where the licensee knows that a value used in TS and FSAR accident analysis can not be satisfied by installed equipment is questioned. This issue was discussed with the licensee in a management meeting conducted on September 24, 1987.

As part of the resolution of the above CAQR, the licensee performed special test instruction (STI) STI-65, Containment Spray Pump Performance for Unit 2. The intent of the test was to reestablish a pump performance curve and verify that pump performance is adequate to provide the needed system flow. Additionally, this test was to measure actual heat exchanger differential pressure (dP) and compare it to the value of 10 psid used to size the pump. Due to problems with installed flow instruments (ANNUBAR), the licensee has had to resort to the use of ultrasonic flow instruments during testing. The test results of STI-65 indicated that the 2B pump satisfied the manufacturers pump performance curve. However, the 2A pump failed to provide the required flow during testing. It was later determined that the ultrasonic flow instrument used during testing of the 2A pump failed it's post use calibration. A second test was performed using another ultrasonic flow instrument and again the pump flow curve failed to satisfy the pump head curve; however, on the second test the pump did deliver the required 4750 gpm minimum flow. A CAQR was issued to document the pump failure.

At the time of the SSQE inspection, it was TVA's intention to perform the recommended testing on the pumps using a single point test at the 4750 gpm flow rate. The team considers that because pump performance appears to be marginally rdequate and that the resolution of this identical issue for Unit 2 included a three point pump curve flow test, that a three point pump curve flow test should be performed for the Unit 1 pumps also. A licensee commitment was obtained to accomplish this. This issue is designated as URI 327,328/88-29-06 Example f., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

The following additional CS Pump functional design parameters were reviewed:

- ° Accuracy of the calculation EPM-DAB-040498 which determines the required CSS pump head for 4750 gpm.

The inspector requested that TVA perform a comparison between the piping lengths appearing in the subject calculation and the as-built drawings. The inspector

considered that due to the importance of the subject calculation and the lack of margin in the use of the generated pump head, such a comparison was crucial. The inspectors performed a limited review of TVA's comparison. The results indicated that the subject calculation was acceptable.

- Consistency of the geometrical configuration and flow loss coefficient between calculations EPM-DAB-040488 Pumphead and SQN-SQS4-0107, Pump NPSH.

In order to review the consistency of the input information used in essential calculations by various preparers, the inspector requested that TVA perform a comparison between the geometrical configuration and flow loss coefficients. Pipe lengths, fittings, components and the corresponding losses were compared. A limited review of TVA's comparison indicated that overall, more conservative numbers were used for the calculation of the required NPSH in calculation SQN-SQS4-0107. Since the available NPSH is higher than the required NPSH, the choice of conservative numbers is acceptable.

- Clogging of Spray Nozzles.

The head loss through the spray nozzles is a significant contributor to the total system loss. Moreover the spray nozzles should be kept clean so that they can pass the required flow. Due to the above considerations, the inspector reviewed TVA's practices in ensuring that these nozzles remain clean.

Technical Specification Surveillance Requirement 4.6.2.1.1.d (and 4.6.2.1.2) require that the CS System (and RHR System) spray train shall be demonstrated operable at least once per five years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed. This is accomplished through the Surveillance Instruction SI-138, Rev. 6, which was found to be acceptable in section 1.h of this report.

- Opening times of Valves FCV-72-2 and FCV-72-39.

The inspector reviewed the data sheets on the stroke time of these valves. The data indicated that both valves open fully in about 15 seconds. The maximum allowable stroke time is about 20 seconds. Considering that the CSS pumps reach full speed in about five seconds and that both valves and pumps receive the signal simultaneously, the valve opening times are acceptable.

- o Effect of the Opening of the Miniflow Line on CS System Pump Ability to Deliver the Required Flow of 4750 gpm.

The Annubars used to measure the flow in the CS System are susceptible to clogging, particularly during the recirculation mode. Such clogging could result in a low flow indication which in turn could result in opening minimum flow valve FCV-72-13. The orifice in this miniflow line is sized to pass 250 gpm under deadhead conditions (201.82 psid). The inspector questioned whether the CS System pump can deliver the required flow of 4750 gpm to the spray header while simultaneously feeding the miniflow line 250 gpm. The total flow through the pump will be 5000 gpm. TVA performed a calculation which indicates that, under these conditions, the pump can deliver up to 5250 gpm. The inspector performed a limited review of this calculation and has found it to be correct. Therefore, the opening of the miniflow valve will still allow the required flow to the spray header.

- (4) A review was performed for the CS pumps relative to net positive suction head (NPSH). The available NPSH for the CS pumps is calculated based on the assumptions that the sump fluid is subcooled (190 degrees F) and that NPSH available is equal to the containment pressure prior to LOCA plus the pump static head minus the vapor pressure head and the line loss. Therefore, the applied methodology meets the intentions of Regulatory Guide 1.1.

The NPSH calculations for the CS pumps are provided in References 8 and 9 (section 1k). These calculations are common to both units. Reference 8 compares the net positive suction head available (NPSHA) to the net positive suction head required (NPSHR) during the RWST injection mode. An adequate margin is computed. This calculation used the rated flow rates for the pumps. The maximum flow rates should have been used instead.

A similar comparison of NPSHA and NPSH is made in Reference 9 for a large LOCA. Maximum flow rates are used. An adequate margin is computed. The maximum flow rates for the CS System pumps are calculated in Reference 10 for Unit 1 and Reference 11 for Unit 2. Reference 11 is a detailed calculation. Reference 10, dated June 15, 1988, simply states that, due to minor differences in geometry between Units 1 and 2, the maximum CSS flow rates are the same for both plants. This issue is designated URI 327,328/88-29-06 Example g., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

- (5) The inspector reviewed Containment Spray Heat Exchangers 1A and 1B to assess the design and procurement of these heat exchangers with respect to FSAR commitments and design criteria.

The inspector verified that the CS heat exchangers were vertical shell, U-tube type heat exchangers with tubes welded to the tube sheet. These heat exchangers are shown on the following as-designed TVA drawings:

- ° 47W812-1, Flow Diagram/Containment Spray System, Revision 17, dated June 14, 1988.
- ° 47W437-1, Mechanical/Containment Spray System Piping, Revision 24, dated May 3, 1988.

SN FSAR Table 3.2.1-2 specified the containment spray heat exchangers as TVA Class B (tube)/C(shell) seismic category I components, the tube side designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, and the shell side in accordance with Section VIII of the ASME Code.

FSAR Table 6.2.2-2 specified the following design parameters for the CS heat exchangers:

Heat Transfer/Unit:	64X10 ⁶ BTU/Hour
Flow Shell Side:	5,000 gpm
Flow Tube Side:	4,750 gpm
Tube Side Inlet Temperature:	135.8°F
Shell Side Inlet Temperature:	83°F
Tube Side Outlet Temperature:	108.5°F
Shell Side Outlet Temperature:	109°F
Design Pressure Shell/Tube:	150/300 psig
Design Temperature Shell/Tube:	200/300 psig

Table 3.7-3 of Design Criteria No. SN-DC-V-27.5, Containment Spray System, Rev. 2, dated July 22, 1987, reiterates these design criteria.

A detailed review of the system functional capability of the CS heat exchangers is presented elsewhere in this report.

TVA procured the CS heat exchangers in accordance with the design criteria contained in TVA purchase specification No. 71C33-92645, Containment Spray Heat Exchangers, which TVA prepared on November 19, 1970.

TVA Specification 1152 for Containment Spray Heat Exchangers for Sequoyah Nuclear Plant Units 1 and 2 forms a part of the referenced purchase specification for the heat exchangers. Specification 1152 reiterates the requirements that the tube side of the heat exchangers be designed in accordance with

Section III of the ASME Boiler and Pressure Vessel Code for Class C Nuclear Vessels, and that the shell side be designed in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

As noted in Section 19 of Specification 1152, Conditions of Service, TVA procured the CS heat exchangers to the following design criteria:

Design pressure, shell, psig	150
Design temperature, shell, F	200
Design pressure, tubes and bonnets, psig	300
Design temperature, tubes and bonnets, F	300

Section 19 specified two limiting conditions (Condition A and Condition B) related to heat transfer and heat sink flow parameters as follows:

	Condition A	Condition B
Quantity of containment spray water, gpm	4750	4750
Quantity of cooling water, gpm	6028	6028
Temperature of containment spray water in, F	156	146
Temperature of containment spray water out, F	115	106
Temperature of cooling water in, F	91	83
Temperature of cooling water out, F	123	115
Maximum allowable pressure drop shell side, psi	15 (max)	15 (max)
Maximum allowable pressure drop tube side, psi	10 (max)	10 (max)
Fouling factor for tube inside, Hr, F, ft ² BTU	0.0003	0.0003
Fouling factor for tube outside, Hr, F, ft ² Btu	0.001	0.001
Duty, Btu/hr	97,385,000	95,000,000

These design conditions meet or exceed the design conditions specified for the CS heat exchangers in the FSAR and Design Criteria.

Section 13 of Specification 1152, Seismic Requirements, details the seismic criteria which the heat exchanger vendor is required to address in order to seismically qualify the heat exchangers.

The CS heat exchanger is shown on Industrial Process Engineers Drawing No. F-6663-2, Rev. B, dated January 6, 1972.

TVA provided the following Industrial Process Engineers calculations:

- ° TVA - Sequoyah Nuclear Plant Units 1 and 2/Containment Spray Heat Exchangers/Code Calculations, dated March 5, 1971 (RIMS No. A26 870728 602).
- ° TVA - Sequoyah Nuclear Plant Units 1 and 2/Containment Spray Heat Exchangers/Seismic Analysis, dated October 4, 1971 (RIMS No. A26 871020 705).
- ° TVA - Sequoyah Nuclear Plant Units 1 and 2/Containment Spray Heat Exchangers/Weights - C.G. - Lifting Lugs, dated October 19, 1971 (RIMS No. illegible).
- ° TVA - Sequoyah Nuclear Plant Units 1 and 2/Containment Spray Heat Exchangers, dated August 24, 1971 (RIMS No. illegible).

These vendor calculations provide some evidence that the CS heat exchangers were qualified to the governing mechanical and seismic criteria, but are not sufficiently legible to permit detailed review.

However, based on three generic deficiencies which the NRC identified during inspection 327,328/87-28, Deficiency D3.4-3, CCW Heat Exchanger Calculation, Deficiency D3.4-4, CCW and CS Heat Exchanger Nozzle Loadings, and Deficiency D4.6-1, Discrepancies Between Design Calculations and Construction Drawings, TVA has prepared CAQR No. SQP870199, Rev. 0, dated October 8, 1987. The CAQR indicated that component analysis and "as-built" anchorages were not consistent and in agreement with component qualification. The CAQR addressed equipment installed in Units 1 and 2.

To address the CAQR, TVA, in part, prepared the following calculations:

- ° Calculation No. CEB-CQS-312, Inclusion of Nozzle Shear Loads in the Qualification of the Containment Spray Heat Exchangers on contract 71C33-92645, Rev. 0, dated August 27, 1987 (RIMS No. B41 870827 002).
- ° Calculation No. MCL C12 et al, Structural Evaluation of As-Modified Containment Spray Heat Exchangers 2A and 2B, Rev. 3, dated February 22, 1988 (RIMS No. 88 0223 310).

TVA closed out CAQR No. SQP870199 on January 12, 1988.

On May 27, 1988, TVA prepared CAQR No. SQP 880363, Rev. 0, to indicate that CAQR No. SQP 870199 had been closed for Unit 1 without completely documenting the qualification of the CS heat exchangers and the associated supports and anchorages, as well as additional Unit 1 heat exchangers.

TVA asked Impell to compare the applicability of the Unit 2 heat exchanger calculations to the Unit 1 heat exchangers.

Impell's letter to TVA dated June 16, 1988, indicates, in part, that CS Heat Exchanger 1A requires separate qualification due to significant differences in the supporting structures and nozzle loads, and that CS Heat Exchanger 1B requires additional evaluation due to differences in the nozzle loads and as-built conditions.

TVA is currently considering Impell's proposal to implement the scope of work outlined in the letter.

The team therefore notes that TVA's actions to re-qualify the components installed in Unit 1 with respect to the generic deficiencies which the NRC identified during the IDI inspection conducted on Unit 2 during the latter part of 1987 are incomplete at this time. This issue is designated URI 327,328/88-29-06 Example h., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

The maximum operating pressure of the CS system heat exchangers is calculated in Reference 31 (section 1.1) as 155 psig. This is below the 220 psi rating of the system.

A study of required ERCW flow rate (shell side) to remove the heat from the CS system under various ERCW inlet temperatures and various heat exchanger tube plugging is given in Reference 32. The results from the reference are used to adjust the ERCW flow rates when Surveillance Instruction SI-566 is implemented (Reference 33). A maximum of 10% tube plugging is used in SI-566.

CAQR SQP 870105, Rev. 1 (reference 34), revises FSAR Table 6.2.1-1 sheets 9 through 12. According to TVA, the revised data agree with HX calculations and the Westinghouse/HX vendor data. Some inconsistencies between the HX parameters in the current FSAR and the HX parameters in the Design Criteria of the CS exist.

The pressure drop across the HX tubes is measured via SI-37.1 and SI-37.2, Containment Spray Pump Tests. During these tests, conducted as part of the Unit 2 restart test program, a pressure differential of about 5 psid was developed for the required flow rate of 4750 gpm.

A recent TVA calculation on the ERCW system performance following the Loss of Down Stream Dam, Reference 35 concludes that an ERCW supply temperature as high as 83.2°F will

adequately remove the required heat load. The team performed a limited review of this calculation as it relates to CSS. An unverified assumption is used which relates to data received from Westinghouse. Some discrepancies were identified between the unverified assumption and heat removal rates used in previous calculations. Inspectors did not perform a review of the justification of the clarification on the differences by TVA.

Due to the importance of the CS System HXs, TVA should review in more detail the HX calculations and their conformance to component specification 1152, the FSAR, and the CSS design criteria. This issue is designated URI 327,328/88-29-06 Example i., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

- (6) An evaluation was conducted to determine if a hazard analysis had been performed for the CS system.

Four issues were addressed by the inspectors.

- ° Effect of a High Energy Line Break on CS System Operability.

The inspector evaluated the effect of high energy line breaks (HELB) inside and outside containment on the CS system. Two concerns were addressed: Pipe whip and flooding from such HELB which could potentially incapacitate the CS system.

Regarding pipe whip for a HELB inside containment, drawings 47W200-12, R5, and 47W200-13, R5, "Equipment, Reactor Building" and drawing 47W2500-1 through 12, R3, "Composite Piping" show that the top of the steam generator cavity, the refueling floor and the control rod drive missile shield provide a physical separation between any piping in the lower containment and the CS system piping and components. Therefore, such an interaction is not credible.

A HELB outside containment will not require the actuation of the CSS. Moreover, a simultaneous HELB inside containment is beyond the DBA and need not be analyzed.

Regarding flooding, there is no CSS equipment inside containment that could be affected by a HELB inside containment. Additionally, for a HELB outside containment, the CSS is not required to operate.

- o Verification of the Secondary Design Basis of the CS System.

The FSAR on page 6.2-87, states, "The secondary design basis for the Containment Heat Removal Spray Systems is the suppression of steam partial pressure in the upper volume due to operating deck leakage from a small break before a full loss-of-coolant accident. The requirement is that the Containment Spray Systems be able to absorb the steam leakage through the operating deck at the maximum possible long-term deck differential pressure of one pound per square foot equivalent to the ice condenser door opening". The team requested the analysis which verifies that the containment spray will quench the leaking steam. TVA's response was provided in writing to the inspector on June 30, 1988. According to this response, the secondary design basis has been deleted from the CS design criteria SQN-DC-27.5, with Westinghouse's concurrence.

The secondary design basis addresses the protection of the containment from a double accident that is a small break, which initiates the CS system, followed by a large break. In the TVA response, the argument is made that this scenario goes beyond current NRC requirements. Moreover, the Technical Specifications' action statements would be entered and safety injection would be actuated before the containment sprays would be activated. The inspector found this logic acceptable.

According to TVA, the secondary design basis will be omitted from the FSAR in the next yearly update, scheduled for April 15, 1989, to make it consistent with the design criteria.

- o Effect of CS System Pump Startup Delay, Due to Diesel Loading Sequence, on the Accident Analysis.

The delay time in the FSAR analysis assumed for loading the spray pumps on the diesels is 30 seconds. This time has been changed to 180 seconds to account for random loads that might occur during the safeguards loadings sequence. To assess the impact of this change, the containment pressure (due to LOCA) and temperature (due to mainsteam line break) analyses currently presented in the FSAR have been reviewed by TVA. As a result of this review, TVA concluded through a qualitative evaluation that the delay has no impact on either analyses.

Westinghouse has concurred with TVA's conclusion. The team has reviewed both TVA's evaluation and the concurrence letter from Westinghouse and found them both acceptable.

- Modeling of the Delay Time for Change Over from Injection to Recirculation.

In changing over from injection to recirculation, the CSS pumps are shut off. The delay time for changeover from injection to recirculation is given in Table 6.3.2-5 of the FSAR as the summation of actions 13 through 18. This time is equal to 110 seconds. Westinghouse LOCA analysis assumed 310 seconds. The inspector determined that this modeling assumption was conservative.

- (7) An evaluation was performed to ascertain that ventilation having airborne radioactivity originating in one pump compartment will not be transmitted to the other pump compartment or to other vital areas within the auxiliary building.

The design features of the ventilation system which would prevent the transport of radioactivity from one pump room to the other were reviewed. Review of drawings 47W866-1, -2 and -10, "Flow Diagram, Heating and Ventilating/Air Flow", indicated the following:

- Backdraft dampers prevent backflow in the ducts. Each room is equipped with one such damper.
- These rooms are normally exhausted by general ventilation. In case of radiation release, the auxiliary building gas treatment system is used.

These design features appear to be acceptable.

j. Preoperational Functional Testing

- (1) An inspection was performed to verify that the following preoperational functional testing was performed and to determine whether or not the testing was adequate.

Test TVA-21B, Containment Spray System.

Pumps were operated at reduced flow through the minimum flow recirculation lines and essentially heated flow through the test line to the RWST. Pump performance values were derived.

Valve interlocks in pump suction lines between the containment sump and RWST were verified to be operable in accordance with SQN-47W610-72-1, Mechanical Logic Diagram.

The capability of manual operation from the control room and the auxiliary control room was verified.

Test W-6.1A1 - Integrated Flow Testing of the Safety Injection (SI) System

This test demonstrated adequate net positive suction head (NPSH) during integrated operation of the CS and SI systems during the recirculation mode.

The inspector reviewed Preoperational Test TVA-21B, Containment Spray System, and verified that the preoperational test operated the pumps at reduced flow through the minimum flow recirculation line and at essentially rated flow through the test line to the Refueling Water Storage Tank, and that pump performance values were generated. The inspector verified that the preoperational test tested the Containment Spray valve interlocks on the pump suction lines and the RHR Containment Spray injection valves. The inspector also verified that the preoperational test verified that the motor operated valves could be operated from the local, remote and auxiliary control stations. The inspector also reviewed Preoperational Test W-6.1A1 - Integrated Flow Testing of the Safety Injection System, as it pertained to the Containment Spray System. The inspector noted the following:

- TVA-21B did not adequately verify the valve interlocks on the RHR containment spray discharge valves 1-FCV-72-40 and 1-FCV-72-41. The inspector determined that this condition had been identified by the licensee as part of the restart test program. The inspector, however, determined that the valve interlock had been tested and verified as part of the ASME Section XI Program.
- Step 5.6.14 of TVA-21B required that inboard and outboard bearing temperatures and the motor temperature be recorded at 10 minute intervals until the bearing temperature stabilizes. The step contains a note that a stable temperature exists when three successive readings do not vary more than 3%. Review of the bearing temperature data recorded in the pre-operational test determined that the bearing temperatures did not meet the 3% criteria as required. Section XI testing requires that pump bearing temperature be monitored and that three successive readings be within 3%. The inspector determined that the pump bearing temperatures recorded in the pre-operational test were not excessive and that TVA has received exemption from the requirement for monitoring pump bearing temperatures on the containment spray pumps based on inaccurate temperature measurements of the bearings and the fact that the other test parameters provide sufficient information about pump condition. The inspector believes this deficiency does not present a pump operability issue and that the licensee's Section XI testing provides sufficient information about pump condition.

- k. The following are additional references that were used during the mechanical portion of this inspection:
- (1) Flow Diagram, Containment Spray System Drawing 47W612-1, Rev. 16, February 16, 1988.
 - (2) Design Pressure and Temperature Calculation for RWST Suction Header and RHR Return Line EMP-SMJ-022886, Rev. 2, May 27, 1988, Units 1 and 2.
 - (3) Containment Spray Pressure and Temperature Requirements, EPM-STM-062088, Rev. 0, June 24, 1988, Units 1 and 2.
 - (4) RHR Spray Header Pressure Requirements at Containment Penetrations, EPM-STM-061388, Rev. 0, June 20, 1988, Units 1 and 2.
 - (5) RHR Spray Header Pressure Requirements at Containment Penetrations, EPM-LFG-061088, Rev. 0, June 16, 1988, Units 1 and 2.
 - (6) Containment Spray Pump Test Requirements EPM-DAB-040488, Rev. 0, April 13, 1988, Unit 1.
 - (7) Containment Spray Pump Test Requirements EPM-DLB-050487, Rev. 3, November 6, 1987, Unit 2.
 - (8) NPSH Calculations for the CCP, SIP, CSP, and RHR Pumps Operating in the RWST Injection Mode following a LOCA, Rev. 0, April 29, 1988, Units 1 and 2.
 - (9) NPSH Calculations for the RHR and CSS Pumps Operating in the Recirculation Mode for a Large LOCA, Rev. 3, May 6, 1988, Units 1 and 2.
 - (10) Containment Spray Pump Maximum Flow EPM-STM-060388, Rev. 0, June 15, 1988, Unit 1.
 - (11) Containment Spray Pump Maximum Flow EPM-DLB-060587, Rev. 0, July 7, 1987, Unit 2.
 - (12) Surveillance Instructions 37.3 and 37.4 "Containment Spray Pump 2A-A Test" and "Containment Spray Pump 2B-B Test", Unit 2, Rev. 1, February 25, 1988.
 - (13) Surveillance Instructions 37.1 and 37.2 "Containment Spray Pump 1A-A Test" and "Containment Spray Pump 1B-B Test", Unit 1, Rev. 1, June 7, 1988.

- (14) Tubular Exchanger Manufacturers Association, Class R Heat Exchanger, Tube Side, ASME Boiler and Pressure Vessel Code Section VIII.
- (15) ANSI 16.5, Steel Pipe Flanges and Flanged Fitting.
- (16) ANSI B 31.1, Code for Pressure Piping with inspection and test requirements to ANSI B 31.7 Code for Nuclear Piping in lieu of applicable Nuclear Code Cases.
- (17) SSOC 1.3, "System Standard Design Criteria (SSDC)," Revision 2, Westinghouse Electric Corporation, dated April 15, 1974.
- (18) E-Specification 678765 - Motor Operated Valves for TVA Sequoyah Nuclear Plants Units 1 and 2, and G-676258 Motor Operated Valves, Westinghouse Electric Corporation.
- (19) E-Specifications 67863 - Control Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and E-Specifications 676270 - Control Valves, Westinghouse Electric Corporation.
- (20) E-Specifications 67869 - 2 Inches and Below Manual "T" and "Y" Globe and Self-Actuated Check Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and 678724 - 2 Inches and Below Manual "T" and "Y" Globe and Self-Actuated Check Valves, Westinghouse Electric Corporation.
- (21) E-Specifications 678760 - Manual "T" and "Y" Globe, Manual Gate, and Self-Actuated Check Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and G-676241 - Manual "T" and "Y" Globe, Manual Gate, and Self-Actuated Check Valves, Westinghouse Electric Corporation.
- (22) E-Specifications 67868 - Auxiliary Relief Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and G-676257 - Auxiliary Relief Valves, Westinghouse Electric Corporation.
- (23) SQNP-47W812-1, Flow Diagram, Containment Spray System Powerhouse, Units 1 and 2.
- (24) SQNP-47W610-72-1, Mechanical Control Diagram, Containment Spray System.
- (25) SQNP-47W611-72-1, Mechanical Logic Diagram, Containment Spray System.
- (26) SQNP-47A366-72-Series, Tabulation of Valve Marker Tags.
- (27) SQNP-47W437-Series, Containment Spray System Piping.
- (28) SQN-DC-V-27.1, Design Criteria for Ice Condenser System.

- (29) SQN-DC-V-3.1, Classification of Piping, Pumps, Valves and Vessels
- (30) Regulatory Guide 1.1, NPSH for ECCS and Containment Heat Removal Pumps.
- (31) "Operating Pressure of CSS Heat Exchangers (Tube side)", Rev. P, SQN-72-0053, EPM-DLB-121987, February 10, 1988.
- (32) "Containment Spray System Heat Exchanger - Tube Plugging", Rev. D, SQN-72-0053, EMP-KBO-017067, February 29, 1987.
- (33) Surveillance Instruction, SI-566, ERCW Flow Verification Test, R18, June 15, 1988
- (34) CAQR SQF 870105, R1, 5-13-88, "Revise FSAR Table 6.2.1-1 sheets 9 through 12, Heat Exchanger Data".
- (35) PIR SQNME986:32, RO. 4-4-87.
- (36) ERCW System Loss of Downstream Dam Flow and Temperature Calculations, SQN-67-0053, HCG-GEB-051088 RO, 6-7-88.

2. Electrical Inspection

Design document and in-plant field observation were integrated in order to evaluate the CS system and components for proper design and design implementation. As discussed in each of the following sections, the inspectors evaluated the system and components against the applicable standards listed at the end of this report and the SYSTEMS/Design basis reports for the Unit 1 CS system.

The inspection of electrical components of the CS system included pumps, motors, breakers, motor operated valves, and associated cabling and control devices. The inspection was conducted on a sampling basis by a comparison of physical installations to Sequoyah Unit 1 as-constructed drawings. TVA staff personnel accompanied inspectors for most of the electrical walkdowns and all findings and comments were discussed with appropriate personnel. Components selected for sampling were necessary in supporting the analyzed design, function, and operation of the CS system and therefore provided an adequate basis for determination of design, compliance, and performance.

a. Electrical Design

(1) General

The design of the station auxiliary electrical systems was evaluated to ascertain whether they would provide reliable power to control and operate the Unit 1 containment spray system and the associated support systems in accordance with the

requirements of General Design Criteria (GDC) 17 of 10 CFR 50 Appendix A. Additionally, the design was reviewed for conformance to the commitments of the Final Safety Analysis Report (FSAR), the requirements of the TS and license conditions.

This design review was made for normal station operating modes including the following postulated conditions:

- Loss of Coolant Accident (LOCA) - Normal offsite power.
- Loss of Coolant Accident - degraded offsite grid.
- Loss of Coolant Accident - loss of offsite power (LOOP)
- Loss of Coolant Accident - electrical fault

The review was made to determine if the steady state and transient current and voltage were within the systems component design ratings.

Correlation between the electrical parameters of selective electric components was reviewed. These electrical parameters were specified in the purchase specifications of electrical equipment, vendors test results, field verified equipment name plate data, and input data to electrical calculations.

The protection coordination of the electrical systems, during postulated fault conditions, was reviewed, relative to the containment spray system, to assure fault removal with the minimum disturbance to the unaffected portions of the electrical systems. Field verification of selected protective relay types and settings was made. These data were compared to the relay calibration test data and to the protection coordination study.

An operability evaluation of the electrical systems was made relative to and including the containment spray system. This evaluation was made by reviewing selective surveillance test records. These tests and records are required as specified in the Technical Specifications. Also reviewed were the operability and design of the containment spray pump motor space heaters. A followup review of the emergency generator alternator space heater problems previously identified was completed. The effects of low voltage during a postulated grid condition, relative to the containment spray system was reviewed. This review included both the electric motors and the motor operated valve motor control center contactor operability.

(2) Scope

The design review included portions of the station auxiliary electric power system as follows:

- Power supplied to the unit transformer 1A from both the main generator and main transformer, to the 6.9KV unit

board 1B, to the 6.9KV shutdown board 1A-A, to the 480V shutdown board 1A1-A & 1A2-A, to the 480V reactor motor operated valve (MOV) board 1A1-A & 1A2-A.

- Power supplied to the 6.9KV shutdown board 1A-A from the 6.9KV emergency diesel generator 1A-A.
- Power supplied from 125VDC batteries 1 and 2 to their associated distribution systems.
- Power supplied from 120VAC vital inverters 1 and 2 to their respective distribution systems.

Power is supplied to the common station service transformer A from the 161KV switchyard to the 6.9KV unit board 1C, to the 6.9KV shutdown board 1B-B, to the 480V shutdown board 1B1-B & 1B2-B, to the 480V reactor MOV board 1B1-B and 1B2-B. Power supplied to the 6.9KV shutdown board 1B-B from the 6.9KV emergency generator 1B-B. Power supplied from 125VDC batteries 3 and 4 to their respective distribution systems. Power supplied from 120VAC inverters 3 and 4 to their respective distribution systems.

The design review included the containment spray system and specific components as follows:

- containment spray pumps (CSP) 1A-A & 1B-B
- motor operated valves:
 - 1-FCV-72-22 RWST to CSP 1A-A
 - 1-FCV-72-23 SUMP to CSP 1A-A
 - 1-FCV-72-34 CSP 1A-A Recir
 - 1-FCV-72-39 CSP 1A-A Disch. Header Isolation
 - 1-FCV-72-21 RWST to CSP 1B-B
 - 1-FCV-72-20 SUMP to CSP 1B-B
 - 1-FCV-72-13 CSP 1B-B Recir
 - 1-FCV-72-02 CSP 1B-B Disch. Header Isolation

(3) Auxiliary Electrical System Analysis

The TVA Electrical Loading Matrix (ELMs) study of the electrical system load flow, fault current and voltage considered the following plant conditions:

- Unit 1 normal - Unit 2 normal (condition 1)
- Unit 1 full load rejection (FLR) - Unit 2 FLR (condition 2)
- Unit 1 FLR - Unit 2 Safety Injection (SI) phase A (condition 3)
- Unit 1 FLR - Unit 2 SI phase B (condition 4)

The above conditions were analyzed at time zero and five seconds including electric power supplied from the main generator (source-1) and from the 161KV system (source-2). During the thirty seconds after a FLR electric power will be supplied to the station service auxiliary power system from the main generator then transferred to the 161KV source.

The ELM study with the reverse of the above conditions for conditions 3 and 4 and with Unit 1 full load rejection was not available for review. The summary and conclusions of the completed ELM study is presently scheduled to be submitted by TVA to the NRC by July 15, 1988.

The emergency diesel generator loading study was not available for review. This study for two unit operation will be submitted by TVA to the NRC prior to Unit 1 restart.

The review of these studies are necessary for the completion of a site electrical systems Safety Evaluation Report.

(4) Fault Currents

A review of the ELM electrical system study revealed the following:

The 6.9 KV unit board load breakers interrupt design rating are exceeded for a postulated fault condition on a load cable next to the breaker.

This condition is valid during normal plant operation when the unit boards are supplied power from the main generator. The interrupt values that the load breakers would be required to interrupt are 584 MVA for the smallest motor and 550 MVA for the largest motor on the unit boards. The installed breakers are ITE 7.5HK500 which have a design interrupt rating of 500 MVA and a performance guarantee rating of 525 MVA. ITE has tested this breaker type for 550 MVA interruption.

The unit board load breaker interrupt requirements are higher should a fault occur when the emergency diesel generators (EDG) are being tested. Only one EDG is tested at a time. The test frequency is once per month for a one hour duration per EDG.

The postulated fault currents in the ELM study are based upon a three phase bolted fault with no fault impedance. This type of fault has a low probability of occurrence. The value of the fault currents would decrease due to both fault impedance and distance of the fault from the bus toward the load due to the increased cable impedance.

The 6.9KV shutdown board load breaker interrupt requirements are 525 MVA which is equal to the breaker performance guarantee rating.

The ELM study did not list the momentary current for the fault condition. Both the interrupt and momentary fault conditions were analyzed for Unit 2 restart and are discussed in the Safety Evaluation Report (SER) for Sequoyah, NUREG-1232, Volume 2. The interrupt value stated in the SER for Unit 2 unit board load breakers was more than 560 MVA. TVA has committed to resolve this problem of the 6.9KV unit board load breaker fault interruption. This commitment was given to the NRC in a letter of August 10, 1987. The NRC staff has requested that TVA provide a detailed description, analysis, and installation schedule for implementation of the corrective actions. TVA has committed to provide this information before June 30, 1989.

The review of this study is necessary for the completion of a site electrical systems Safety Evaluation Report and TVA should make this information available to the NRC reviewer.

The postulated fault current values in the ELM study, at the 480 volt portion of the auxiliary power system, did not exceed the 480 volt breaker interrupt rating and are acceptable.

(5) Voltage

The voltage at the 6.9 KV shutdown boards, for time zero, conditions 1 through 4, with source 1 and 2, was adequate to maintain the 6.6 KV motors within the motors' design rating.

During condition 4 when the shutdown boards are supplied power from source 2, the voltage at time zero drops below the setpoint of the degraded voltage relay. This set point is 6560 volts, plus or minus 33 volts. After five seconds the voltage on shutdown board 1A-A recovers to 6662 volts which is 68 volts above 6560 volts plus 33 volts. Associated with the degraded voltage relay is a ten (10) second time delay before system separation. Although a ten second ELMs study was made, it was not available for review. The concern is the voltage relay dead band. TVA was asked to provide this information. The value of voltage that must be reached to stop the time delay relay must be known to assure that the voltage at the shutdown board has recovered above the dead band before ten seconds to preclude unnecessary system separation for the offsite source. The information provided by TVA from the PSO relay calibration sheet indicates the undervoltage relay would reset and stop the timer at 6600 volts and the shutdown board voltage recovers to 6662 in 5 seconds.

The review of this study is necessary for the completion of a site electrical systems Safety Evaluation Report and TVA should make this information available to the NRC reviewer.

The 480 volt motors did not fall below their 80% starting limits with one exception. A motor operated valve (MOV) had 79% voltage. The MOV had been specified to start at 75% voltage. TVA will be asked to show that all 6.6 KV and 460 volt motor voltages recover to the minimum of minus ten percent of motor rated voltage after either a condition 3 or 4 when supplied power from source 2.

The review of this data is necessary for the completion of a site electrical systems Safety Evaluation Report and TVA should make this information available to the NRC reviewer.

A review was made for adequate voltage at the containment spray system motors and motor operated valve motor contactors during a steady state degraded voltage condition.

The voltage value used was the setpoint of the degraded voltage relay which is 6560 volts plus or minus 33 volts. Using the low side of 6560 volts minus 33 volts, 1 volt was added for a value of 6528 volts. The containment spray pump motor terminal voltage was within the motor rating. The motor operated valves, associated with the containment spray system, also had terminal voltage within their design. The voltage at the MOV motor contactor coil was above the TVA test value of 80 volts. During the contactor minimum voltage test the pickup current was 987 milliamperes. The worst case control current was compared to the type FRN-1 fuse which is a 1 ampere time delay fuse. The worst case contactor current was 70% less than the fuse opening current, at the 10 seconds setpoint of the degraded voltage condition, where system separation occurs.

(6) Protection Coordination

The protective relay coordination provides selective tripping during a fault condition to minimize deenergizing electrical equipment. The load breakers should open for a load fault without opening a supply breaker unless there is a failure of its protective relays or the load breaker fails to open.

A review was made of protective relays associated with the unit transformer and incoming supply breaker to the 6.9 KV unit board 1B. These relay setpoint curves were compared with the protective relay setpoint curves of reactor coolant pump 2 for proper coordination. Reactor coolant pump 2 is the largest horsepower motor supplied power from the unit board 1B. This coordination review contained for the protective relays associated with unit board 1B tie breaker to shutdown board 1A-A

including those relays associated with the incoming supply breaker to shutdown board 1A-A. The relay setpoint curves associated with the incoming power supply to shutdown board 1A-A were compared with the protective relay setpoint curves of the largest horsepower motor on the bus, which is containment spray pump 1A-A. The coordination review continued from shutdown board 1A-A through load center transformer 1A1-A to the 480 volt shutdown board (1A1-A) then to the 480 volt reactor motor operated valve (MOV) board (1A1-A). Reactor MOV board 1A1-A supplies power to the containment spray MOVs associated with containment spray pump 1A-A.

Relay types, trip current setpoints, and time lever settings were found to be the same between the coordination curves, relay calibration sheets and field verified at the panels. The coordination curves indicated that the electrical systems protective relay setpoint was adequate.

(7) Electrical Operability Surveillance

There were fifteen electrical surveillances considered for review. These are tests required by the Technical Specification. Due to the extensive data and the review time available during this inspection a proper review could not be made at this time. However, the tests that are identified in the documents reviewed listing will be given an additional review both for adequacy of test methods and content.

It was noted in the review for SI-7, Diesel Generator, that starting the diesel required pulling fuses, removal of relay covers, and pushing a relay to make contact. Also, the acceptable measurement in the seven day tank, related to the 62,000 gallons required by the Technical Specification, was given in feet in the Surveillance Instructions. TVA has not responded as to why the diesel generator batteries Surveillance Test did not include both a service and capacity test as did the vital station batteries.

The review of this information is necessary for the completion of a site electrical systems Safety Evaluation Report. TVA should make this information available to the NRC reviewer.

(8) Electrical Data

There were no differences between electrical parameters listed in the following documents reviewed:

- o purchase specification requirements
- o vendors fill in data of purchase specifications
- o vendors test data

- o electrical drawing
- o relay calibration sheets
- o electrical studies
- o field verified name plate data:
 - main generator
 - main transformer
 - unit transformer 1A
 - common transformer A
 - emergency diesel generator 1A-A & 1B-B
 - containment spray pump 1A-A
- o field verified protective relay setpoints

(9) Electrical Components

The time delay relays used for the emergency diesel loading and degraded voltage time delay are of the electric pneumatic type. These relays require that air bleed off to complete the time delay function. TVA was asked to provide data that these relays would not be adversely affected during a tornado created atmosphere depression. TVA provided a study, SQN-CSS-019, that indicated that these time delay relays were not adversely affected during a tornado.

b. Electrical Components Inspection

(1) Motor Operated Valves

A wiring verification and inspection was performed on five Containment Spray motor operated valves. Work was observed on two additional motor operated valves that had their valve operators removed and in the mechanical maintenance shop for repair. Selected valves in the Containment Spray system were inspected for proper wiring configuration, qualified wire, correct termination and crimping, limit switch condition, cable and conductor damage, valve cleanliness and condition, and environmental qualification. The wiring was verified to be in accordance with the "As Configured" wiring diagrams. The inspection included an electrical verification of valve condition in the limit switch compartment. Prior to the NRC field inspection of the motor operated valves, the licensee had performed a pre-inspection of all CS valves predicated by the impending NRC inspection. NRC inspectors noted many of the same findings that were discovered during the pre-inspection. NRC inspectors found the additional discrepancies that were not discovered by the licensee during the pre-inspection:

- o = NRC additional discrepancies
- * = pre-inspection discrepancies

- Motor lead T1 bend radius was not in accordance with requirements.
- The limit switch cover gasket seating surface was coated with surface rust.
- Wire 53 and 55 terminal lugs on rotor terminal 2 were bent in excess of 90 degrees between the ring and the lug.
- * Crimps on CL1 (red) and 60 (black) conductors of 1V3155B, insulation not under insulation on barrel.
- * Motor leads T1 and T2 and wire 25 (white) conductor of cable 1A5335 have cable repairs using electrical tape rather than Raychem.
- * Cable identification tag was missing from cable 1A5335.
- * 2 Terminals on rotor 1, contact position 1 were incorrectly labeled as #53 and #55.
- No additional findings.

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- * Crimps on all conductors of cable 1V1842B (except blue), insulation not under insulation on barrel.
- * Hairline crack on unused rotor finger block.
- * Conduit identification tag broken on cable 1V1840B.
- * Cable 1A5388 was not tagged correctly.
- * Cable 1A1842B was not tagged.
- * Spare conductors were spared using electrical rather than Raychem.

1-FCV-72-22

- Cutoff terminal lug laying loose in limit switch compartment.
- Loose tiwrap laying in limit switch compartment.
- * Crimp on white-Black conductor on cable 1V1872A, insulation not under insulation on barrel.
- * T1 and T2 motor leads were charred.

- Motor lead T1 bend radius was not in accordance with requirements.
- The limit switch cover gasket seating surface was coated with surface rust.
- Wire 53 and 55 terminal lugs on rotor terminal 2 were bent in excess of 90 degrees between the ring and the lug
- * Crimps on CL1 (red) and 60 (black) conductors of 1V1558, insulation not under insulation on barrel.
- * Motor leads T1 and T2 and wire 25 (white) conductor of cable 1A5335 have cable repairs using electrical tape rather than Raychem.
- * Cable identification tag was missing from cable 1A5335.
- * 2 Terminals on rotor 1, contact position 1 were incorrectly labeled as #53 and #55.
- No additional findings.

1-FCV-72-20

- * Crimps on all conductors of cable 1V1842B (except blue), insulation not under insulation on barrel.
- * Hairline crack on unused rotor finger block.
- * Conduit identification tag broken on cable 1V1840B.
- * Cable 1A5388 was not tagged correctly.
- * Cable 1A1842B was not tagged.
- * Spare conductors were spared using electrical rather than Raychem.

1-FCV-72-22

- Cutoff terminal lug laying loose in limit switch compartment.
- Loose tiwrap laying in limit switch compartment.
- * Crimp on white-Black conductor on cable 1V1872A, insulation not under insulation on barrel.
- * T1 and T2 motor leads were charred.

- * Cable 1A5394 missing identification tag.
- * Spare conductors were spared using electrical tape rather than Raychem.

1-FCV-72-40

- o Flexitite conduit for cable 1A671 pulled out of fitting with sharp edge resting on conductors.
- o Conduit fitting loose on flexitite for 1V2150A.
- * Crimp on T3 motor lug, insulation pulled out from under insulation on lug barrel.
- * White wire (89) of cable 1A3224 has the same problem as above.
- * Conduits 1A3224 and 1A671 missing identification tags.
- * Cable 1V2743A missing cable identification.
- * Wire 53 (green) and wire 55 (red) were terminated incorrectly on terminal 16 of rotor #4 rather than 15 as required by configuration drawings.
- * Spare conductor spared with electrical tape rather than Raychem.

1-FCV-72-41

- o Terminal nut laying inside limit switch compartment.
- o washer laying loose inside limit switch compartment.
- o A spare rotor block jumper wire 3 inches long with bare terminal at both ends was left loose inside limit switch compartment.
- * Crimp on white conductor (#89) of cable 1A3236 wire insulation not crimped under lug barrel insulation.
- * Conduit 1A3236 missing identification tag.
- * Cable 1V1910B and 1M237 missing identification tags.
- * Spare conductor spared with electrical tape rather than Raychem.

1-FCV-72-2 and 1-FCV-72-39

Valve operators for these 2 valves were removed and were being overhauled. Inspectors surveyed the valve location in the Unit 1, 714 ft level penetration room. The valve operators had been disassembled in place and the operators were in the mechanical shop. The limit switch covers, limit switches, torque switches, terminal boards, motors, nuts, bolts and washers were located in the general work area on top of hanger 1CSH-5 near the valves about 8 feet above floor level. Several deficiencies were noted with regard to proper in-process handling, storage, and protection of safety related material and equipment. This was discussed with electrical supervisory personnel; however, the condition remained unchanged during the 2 week inspection period. The following was noted.

- The limit switches, torque switches, terminal boards, and wiring remained exposed and unprotected in an upside down limit switch compartment cover for both valves.
- Nuts, bolts, washers, and other parts of the operator which were not tagged or identified were stored loose under the torque switch, limit switch and terminal blocks in the limit switch compartment cover.
- Lubricated gears on the torque switches were not protected from damage.
- Exposed lubricated stems on the valves were not protected.
- Both operator motors were sitting on the hanger cross beams untagged with exposed unprotected lubricated gears turned upward. One motor was tied off. The other motor was cradled between hanger beams.
- The hold tag for 1-FCV-72-39 was attached to the disconnected cable over the valve rather than to the valve as required.
- All disconnected cables were hanging loose with no protection for the safety related terminations.
- Insulation damage was noted on the control power cable conductors for 1-FCV-72-2.

During the period of the inspection, groups of 2 to 4 maintenance personnel conducted replacement of mechanical penetration seals directly above the valves and exposed parts. The work platform for part of the maintenance were the hanger beams that all of the parts were sitting on.

(2) Electrical Control Boards

Containment Spray portions of the 6.9 KV shutdown boards and reactor boards were inspected to determine that system circuits, relays, breakers, fuses, and switches were properly installed and that corrective and preventative maintenance had maintained the electrical boards in accordance with procedures and drawings. Prior to the NRC inspection of the electrical boards TVA had performed a pre-inspection of the same boards using teams composed of DNE, maintenance, QC, modification, and system engineering personnel. NRC inspectors performed a field inspection verifying the TVA findings and in addition, found several additional discrepancies that were not identified by TVA. NRC field inspections were conducted with electrical maintenance supervisory and craft personnel. The following items were noted:

- = NRC identified
- * = Identified during TVA pre-inspection.

6.9 KV Shutdown Board 1A-A, compartment 13.

- Front compartment extremely dirty (up to 1/4 inch dust and dirt in compartment bottom).
- Rear compartment had been cleaned after the pre-inspection but cleaning was inconsistent. The front of current transformer insulators was clean and wiped down, rear of current transformer insulators was dirty, front of bus bar insulators was clean, rear still had dust, etc.
- There was improper bolting in front panel between motor starter relays. The lock washer and flat washer were cocked preventing full contact with the panel.
- One of two hinge pins on the front panel was not fully engaged (seismic concern).
- * Some "A" phase current transformer insulator screws were missing.
- * Rear panel compartment needed cleaning.
- * One rear panel screw was missing.

6.9 KV Shutdown Board 1B-B, compartment 13.

- One of two hinge pins on front panel door was not fully engaged.
- Improper bolting between motor starter relays.

- * Front and rear compartment needed cleaning.

- * One wire was disconnected with no tag.

480V Reactor MOV Board 1B1-B, compartment 13A

- o Green vertigree on breaker staves.

- o Cutoff terminal lug laying in rear of compartment 12 (seen from rear of compartment 13)

- o One of two hinge pins on front panel door was not fully engaged.

- o Bend radius violation on jumper wire for 1FU4-72-2B.

- * Front and rear compartments needed cleaning.

480V Reactor MOV Board 1B1-B, compartment 13B.

- o One of two hinge pins was not fully engaged.

- o Cutoff tiwrap was laying on top of a motor starter relay.

- * Loose screw was found on compartment floor.

- * Rear compartment needed cleaning.

- * Time delay relay (Agastat) was labeled as setting 8.0 to 8.5 seconds, drawing stated 10 seconds.

480V Reactor MOV Board 1B1-B, compartment 13C.

- o Bend radius violations on fuse block jumper wiring.

- o Front panel wiring loop had a broken wire support, tiwraps were substituted.

- o One of two hinge pins on front panel door was not fully engaged.

- * Rear compartment required cleaning.

480V Reactor MOV Board 1B1-B, compartment 13E.

- o One of two hinge pins on front panel cover was not fully engaged.

- * Rear of compartment needed cleaning.

480V Reactor MOV Board 1A1-A, Compartment 4E.

- The T3 motor lead had a bend radius violation.
- The breaker indicating light mounting bracket had one loose screw.
- One terminal screw was laying in the bottom of the rear compartment.
- * Pre-inspection noted no discrepancies.

480V Reactor MOV Board 1B1-B, compartment 14A.

- Inner frame member support in rear compartment not engaged with inner frame.
- Bend radius violations on fuse block jumper wires.
- Several terminal lug connections on the board side of the MOV control power terminal block do not meet acceptance criteria for lug insulation crimped over wire insulation.
- The 7 conductor control power cable for the MOV had a bend radius adjacent to the terminal board.
- There was green vertigree on the breaker staves.
- * Rear compartment required cleaning.

M and AI - 7, Cable Terminations, Splicing, and Repairing of Damaged Cables implements TS 6.8.1 for establishment, implementation, and maintenance of procedures for the termination and repair of safety related electrical components was reviewed. Contrary to sections 3.4 and 5.2 of M&AI-7, motor lead T1 on flow control valve 1-FCV-72-13 was not trained in accordance with the required bend radius. In addition, motor leads T1 and T2, and white conductor wire 25 of cable 1A5335 have cable repairs using electrical tape. This is a violation 327,328/88-29-02, example 1, failure to maintain safety related electrical equipment.

Green wire 53 and red wire 55 on 1-FCV-72-40 were not routed in accordance with Drawing 45N1749-15. This is a violation, 327,328/88-29-02, example 2.

Standard Practice SQA 66, Plant Housekeeping, implements TS requirements and Nuclear Quality Assurance Manual part II, Section b 1.2 Requirements for Procedural Control of Work Activities. Section 5.3.2 of SQA 66 states that if work extends beyond one shift, and is not continuously worked the craftsman shall ensure the work area is left clean. Tools, parts, and equipment must be properly identified with area barrier tag or individual pink tags. It also states that special care shall be

taken when opening or disassembling sensitive electrical equipment which may be damaged by dust or moisture. Contrary to this requirement, components for valves 1-FCV-72-2 and 1-FCV-72-39 were not tagged correctly, nor covered. These components were stored in an area where penetration seal work was being conducted directly overhead. This is a violation, 327,328/88-29-02, example 3.

Maintenance Instruction MI-6.20, Configuration Control During Maintenance Activities, implements TS procedural requirements for controlled reassembly of safety related components. MI-6.20 states that when a configuration change is returned to normal the accuracy shall be verified and documented. Contrary to this procedure, during an internal inspection of the limit switch component of valves 1-FCV-72-41 and 1-FCV-72-22, loose extraneous material was identified in the internal of the limit switch. This is violation 327,328/88-29-02, example 4.

The loose material which was identified in violation 327,328/88-29-02, example 4 also constitutes a question with respect to the maintenance of the seismic qualification of the equipment in accordance with IEEE 344, Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations. IEEE 344 states that it must be demonstrated that the equipment is capable of performing its safety function throughout its qualified life including its functional operability during and after an SSE at the end of life. It further states that justification must be provided to show that the equipment to be qualified is similar to the data base equipment. If extraneous material has been left within the qualified equipment, then the installed equipment may no longer be similar to the data base equipment, test or calculation that was originally used to support seismic qualification.

In addition to the discrepancies listed in violation 327,328/88-29-02, examples 1-4, the discrepancies listed in the MOV and control board sections of this report relate to inadequate implementation of several sections of M&AI-7, Cable Terminations, Splicing and Repair of Damaged Cables, SQM-2, Maintenance Management System, and SQA-66, Plant House Keeping.

The additional discrepancies identified above require resolution prior to the startup of Sequoyah Unit 1. Adequate corrective action for violation 327,328/88-29-02, Failure to Maintain Safety-Related Electrical Equipment, will include correction of identified deficiencies, evaluation of root cause and appropriate action to preclude recurrence. In addition adequate corrective action for the above mentioned violation shall include a Quality Assurance review of the TVA pre-SSQE walkdown discrepancies and the applicability of those discrepancies to other components.

(3) Cabling

Inspectors reviewed cable routing records for the Containment Spray system cables and performed on a sampling basis, a physical inspection of 6.9 KV Containment Spray pump power cables, and cable tray system associated with that system. Cables and trays inspected were reviewed for divisional separation, segregation, identification, loading, associated cabling and cable condition. Mechanical aspects of the cable tray supports are addressed in the support section of this report. TVA performed a pre-inspection of the cables and trays for the 6.9 KV power cables for the 1A-A and 1B-B Containment Spray pumps. During the pre-inspection, cable 1PP637B (CS pump 1B-B 6.9 KV power cable) was found to be routed in the wrong tray for a distance of about 20 feet. The cable left the specified tray AA-B between nodes 5 and 6 and entered tray AO-B between nodes 13 and 14. The cable exited tray AO-B between nodes 15 and 16 and entered conduit 1PP637B.

Inspectors performed a physical inspection of the 1A-A, Containment Spray pump power cable tray AM-A between nodes 25 and 28. Correct cable count and identification were verified for this tray section. No deficiencies were noted.

Inspectors performed a physical inspection of the misrouted cable in tray AO-B on the 669 ft. level of the auxiliary building. Inspection of this tray section verified a portion of the TVA walkdown on cable 1PP625A. Inspectors performed a physical inspection of cable trays AO-B between nodes 13 and 16 and AA-B between A10 and A12 and verified the misrouted cable. Although the cable was misrouted and not in accordance with the approved routing schedule, inspectors concluded that the routing was satisfactory with respect to separation, voltage segregation, associated cabling, and Appendix R considerations. Based on tray fill and cable count of tray AO-B in the area of the misroute, it appeared that tray loading and ampacity would be satisfactory. It appeared that the misrouting was from original plant construction. Tray AA-B, the correct tray, turned away from the area the cable needed to be routed and tray AO-B continued in the required direction to about 5 feet from the conduit entry point for Containment Spray pump 1B-B cable.

Inspectors concluded that the misrouted cable presented no operability or safety problems and that based on the tray location and routing, the misrouting may have been an original construction walkdown drawing error. Discussions with TVA staff indicated that a Drawing Deviation request and a CAQR were being prepared to document and resolve the issue. This issue was identified as a deficiency and provided for licensee information.

c. Documents Reviewed

The inspector reviewed the documents listed as reference 1-19 during the design review of the electrical systems identified in the scope:

- (1) Final Safety Analysis Report (FSAR) - Chapter 8.
- (2) Technical Specification, Amendment 64 - 3/4.8 Electrical Power Systems - Table 3.3-4 Engineered Safety Features Actuation System (ESFAS) Instruments Setpoints.
- (3) Safety Evaluation Report on Sequoyah Nuclear Performance Plan, NUREG -1232, Vol. 2, May 1988.
- (4) Sequoyah Unit 2 Integrated Design Inspection (IDI), November 6, 1987.
- (5) 6900V Unit Board Load Coordination Study PSO Plant Section RS Calculation, Revision 3/88.
- (6) 480V AC Class 1E Load Coordination Study, Revision 11.
- (7) Sargent & Lundy ELMS AC Program - Load Flow, Short Circuit Currents & Voltage, Run 6/20/88.
- (8) DS-EB.1.1, Electrical Design Standard for Substitution of Low Voltage Power and Control Fuses, Revision 8/18/87.
- (9) AI-16, Administrative Instruction for Fuse Control, Revision 12.
- (10) SQEP-34, Engineering Procedure for Implementation of the Electric Fuse Tabulation, Revision 10.
- (11) Data Sheets, Sequoyah Fuses in System 72 Containment Spray System.
- (12) Purchase Specifications:
 - 9617 Steam Turbogenerators and Reactor Feedpump Turbines.
 - 9841 Main Power Transformers and Neutral Reactors
 - 9877 Common and Unit Station Service Transformers.
 - 1166 Diesel Engine Driven Emergency Power Packages.
 - 1101 6900 volt Auxiliary Power Switchboards.
 - 1135 480 volt Switchboard and Transformers.
 - 1200 480 volt Motor Control Centers (MCC).
 - 1153 Electric Motor Driven Containment Spray Pumps.
 - 9923 Principle Piping Systems and Appurtenance. (original MOV (purchase)
 - MEB-SS10.10 Motor Operated Valve Motor Operator. (replacement)

(13) Vendors Test Data:

main generator
 main transformer
 unit & common station service transformers
 emergency diesel generators
 6.9KV switchgear
 480V switchgear
 480V motor control center (MCC)
 250V MCC control fuses

(14) L6883, Engineering Change Notice for Moldec Case Breaker Replacement and Thermal Overload Bypass.

(15) SQN-CSS-019, Agastat Accuracy During a Tornado Depressurization.

(16) FIRL No. F-A5844, Franklin Institute Research Report for AMERACE Corporation on Depressurization Tests of Agastat Series E 7000 Time Delay Relays, September 28, 1983.

(17) TS Surveillance Tests:

- SI-7, Diesel Generator (DG), revision 41 -DG 1B-B 6/11/88.
- SI-26.1A, Loss of Offsite Power with Safety Injection, Revision 13, DG 1A-A 6/22/87, 6/30/87/7/4/87.
- SI-26.1B, same as above except Revision 9, DG 1B-B 10/13/85
- SI-238 DG Battery (BAT) System Operability, Revision 19, DG 1A-A 4/7/88, DG 1B-B 6/10/88.
- SI-238.1 DG BAT Weekly Test, Revision 14, All DG BAT 11/13/88.
- SI-238.2 DG BAT and Charger Performance Test, Revision 7, DG 2A-A 5/6/87, DG 1A-A 5/6/87.
- SI-238.3 DG BAT Annual System Inspection, Revision 0, DG 1B-B 8/20/87.
- SI-100.1 125V Vital BAT Weekly Inspection, Revision 17, all BAT 4/11/88.
- SI-100.3 125V Vital BAT Annual Inspection, Revision 0, Vital BAT 4 3/1/88.
- SI-105 125V Vital BAT 60 Month Performance Test, Revision 15, Vital BAT 1 Test 7/22/85, Vital BAT 2 Test 5/28/85, Vital BAT 3 Test 3/27/85, Vital BAT 4 Test 2/15/85.

- SI-251.1 MOV Thermal Overload Test, Revision 0, 1-FCV-72-20 Test 12/19/83, 1-FCV-72-21 Test 12/6/83.
- SI-270.1 Fuse for containment penetration conductor overcurrent protection surveillance, Revision 0, 10% rotating sample 18 month test 3/10/87.
- SI-256 Periodic calibration of overcurrent and ground fault relays on reactor coolant pumps and backup devices on 6.9 KV unit board, Revision 10, 72 months test.
- SI-258 480V circuit breaker containment penetration conductor overcurrent protection, Revision 0, 10% rotating sample 18 month test.
- SI-266 6.9 KV circuit breaker inspection and preventative maintenance reactor coolant pump A test.

(18) Cable Data for the following:

Node Number	Cable Number	Node Number	Cable Number
8	1PP625A	122	1PL5075B/1
8	1PP756A	122	1PL5076B/1
8	1PP759A	122	1PL5077B/1
8	1PP750A	122	1PL5078B/1
9	1PP637B	122	1PL4945B
9	1PP762B	247	1V1870A
9	1PP765B	247	1V1880A
9	1PP753B	247	1V3160A
13	1PP105S2/1	247	1V2820A
13	1PP106S2/1	249	1V2830B
13	1PP107S2/1	249	1V3150B
14	1PP110S1/1	249	1V1840B
14	1PP111S1/1	249	1V1850B
14	1PP112S1/1	449	1PL5047A/1
119	1PL5051A/1	449	1PL5048A/1
119	1PL5052A/1	449	1PL5049A/1
119	1PL5053A/1	449	1PL5050A/1
119	1PL5054A/1	450	1PL5055A/1
119	1PL4935A	450	1PL5056A/1
120	1PL5059A/1	450	1PL5057A/1
120	1PL5060A/1	450	1PL5058A/1
120	1PL5061A/1	453	1PL5063B/1
120	1PL5062A/1	453	1PL5064B/1
120	1PL4938A	453	1PL5065B/1
121	1PL5067B/1	453	1PL5066B/1
121	1PL5068B/1	455	1PL5071B/1
121	1PL5069B/1	455	1PL5072B/1
121	1PL5070B/1	455	1PL5073B/1
121	1PL4942B	455	1PL5074B/1

(19) Drawings

- 15E500-3, Revision A, Transformer Taps & Voltage Limit APS
- 15E500-1, Revision H, Key Diagram One Line APS
- 15E500-2, Revision I, Key Diagram One Line ASP
- 45N713, Revision K, Station Serv. Trans. & Bus
- 45N721-1, Revision W, 6.9KV Unit Boards (BD) 1A & 1B
- 45N721-3, Revision E, 6.9KV Unit BD 1C & 1D
- 45N724-1, Revision Z, 6.9KV Shutdown (SD) BD 1A-A
- 45N724-2, Revision Z, 6.9KV SD BD 1B-B
- 45N749-1, Revision RO, 480V SD BD 1A1-A
- 45N749-2, Revision QQ, 480V SD BD 1A2-A
- 45N749-3, Revision TT, 480V SD BD 1B1-B
- 45N749-4, Revision RO, 480V SD BD 1B2-B
- 45N751-1, Revision TT, 480V reactor motor operated valve (RMOV) BD 1A1-A sheet 1
- 45751-2, Revision JJ, 480V RMOV BD 1A1-A sheet 2
- 45751-5, Revision QQ, 480V RMOV BD 1B1-B
- 45N700-1, Revision R, Key Diagram 120VAC & 125VDC Vital Plant Control Power System
- 45N703-1, Revision UU, 125V Vital Battery (VB) BD 1
- 45N703-2, Revision RC, 125V VB BD 2
- 45N703-3, Revision RO, 125V VB BD 3
- 45N703-4, Revision OO, 125V VB BD 4
- 45N763-7, Revision J, 6.9KV Unit Auxiliary Power
- 45N763-1, Revision J, 6.9KV Unit Aux Pwr DC
- 45N763-2, Revision P, 6.9KV Unit Aux Pwr
- 45N765-1, Revision Q, 6.9KV Unit Aux Pwr Sheet 1

- 45N765-2, Revision Q, 6.9KV Unit Aux Pwr Sheet 2
- 45N765-3, Revision AA, 6.9KV Unit Aux Pwr sheet 3
- 45N765-4, Revision R, 6.9KV Unit Aux Pwr Sheet 4
- 45N765-5, Revision CC, 6.9KV Unit Aux Pwr Sheet 5
- 45N765-7, Revision RO, 6.9KV Unit Aux Pwr sheet 7
- 45N765-8, Revision O, 6.9KV Unit Aux Pwr Sheet 8
- 45N765-9, Revision J, 6.9KV Unit Aux Pwr Sheet 9
- 45N765-18, Revision II, 480V SD Aux Pwr
- 45N779-8, Revision II, 480V SD Aux Pwr
- 45N779-10, Revision RO, 480V SD Aux Pwr Sheet 10
- 45N779-11, Revision V, 480V SD Aux Pwr Sheet 11
- 45N779-25, Revision W, 480V SD Aux Pwr Sheet 25
- 45N779-26, Revision T, 480V SD Aux Pwr Sheet 26
- 45N1749-15, Revision W, 480V RMOV BD 1A1-A Sheet 8
- 45N1750-5, Revision F, 480V RMOV BD 1B1-B Sheet 5
- 45N1750-13, Revision K, 480V RMOV BD 1B1-B Sheet 7

3. Instrumentation and Control (Design Evaluation)

In each of the following sections the design phase of the inspection evaluated the system and components against select requirements as indicated in the report.

a. Instrument Verification

The design aspects and application of the following instruments were reviewed:

- PDT 30-42 Containment/Annulus Differential Pressure Channel IV
- PDT 30-43 Containment/Annulus Differential Pressure Channel III
- PDT 30-44 Containment/Annulus Differential Pressure Channel II
- PDT 30-45 Containment/Annulus Differential Pressure Channel I

b. Interlock and Control Functions

- (1) The operability of the CS pump miniflow protective circuit was evaluated. This circuit protects the pump by allowing pump discharge to be circulated back to the pump intake if flow in the discharge line drops below that required for pump operability. The flow is measured by flow elements FE-72-34 or 13, and if upon starting, flow is not achieved in the spray header within a preset time interval, the circulation is back to the intake. The flow elements are ANNUBARS.

The design evaluation of the miniflow control channels revealed the following: inspectors reviewed the specifications for the flow channels used for measuring containment spray flow; this measurement is used for flow indication, and at the lower end of its range, provides a signal for miniflow control. Inspectors also reviewed schematic diagram 45N779-26 Revision T, dated January 16, 1988, vendor loop diagram (GE) D-30COK13-513 sheet 9, Revision E, instrument accuracy calculation 1-FT-72-13 Rev. 4 dated April 30, 1988, and ECN 6674 Revision 1 dated April 30, 1988.

The inspector identified several miniflow control issues. These issues regarded the ability of the miniflow control to function as described in the FSAR under all design basis conditions, the accuracy of the flow indication presented to the operator, and the implications for any other safety-related systems using Annubars.

First, the use of an Annubar for measuring flow during the containment sump recirculation phase does not assure a functional system, since the ports and plenum of the element could become blocked by post-accident debris or particulates from the sump. The TVA specification did not stipulate debris and particulate as a design basis. If the upstream ports were blocked, a false low flow measurement could result, which would open the miniflow valve at normal flow values rather than at the low flow setpoint. This would divert flow from the spray headers, and could occur concurrently for both CS system trains. Overriding the flow signal from the control room (i.e., closing the valve) must be done by holding down the momentary action control switches, making corrective action difficult.

The consequences of diverted miniflow are discussed in Section 1.j.3 of this report. In addition, inaccurate flow indications in both CSS trains would be presented to the operator for verifying containment spray flow; this verification is required by Emergency Instruction ES-1.2, "Transfer to RHR Containment Sump", Rev. 5, dated January 12, 1988. This issue is designated URI 327,328/88-29-06, Example j., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

Second, ECN 674 reported structural inadequacy of the original Annubars which had one-sided support; that ECN replaces the original elements with Annubars having two-sided support. Discussions with TVA staff indicated that one original Annubar had been damaged in service. Inspectors requested that TVA provide a root cause and extent determination for the original failure of the Annubar element to provide assurance that problems do not exist for other safety-related systems. This issue is designated URI 327,328/88-29-f/s, Example k., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

Third, it was noted that it was not apparent that element errors due to process upstream/downstream flow conditions were accounted for. These errors would take into account installed upstream/downstream straight run, tees, elbows, and other significant disturbances that could result in repeatability errors related to the velocity profile. Subsequent documentation of the installed conditions was retrieved by TVA, indicating that the downstream conditions were not within the reference conditions stipulated by the vendor for the repeatability specifications quoted.

This issue was identified as a deficiency and provided for licensee information.

It was also noted that the setpoint calculation concludes that a setpoint of at least 1200 gpm is required, due to an assumed 5:1 turndown ratio for the element (due to the square root relationship of differential pressure to flow). However, the actual setpoint indicated on the current instrument tabulation is 500 gpm, which is below the region of accurate measurement. TVA explained that the instrument tabulation will be updated to reflect the higher setpoint after the modification is implemented. Inspectors concluded this action was acceptable and in accordance with TVA procedures.

- (2) An inspection was performed to verify the existence of an interlock between FCV-72-23 and 22 for CS A train and FCV-72-20 and 21 for CS B train. The function of this interlock is to prevent the CS pump from taking suction from the RWST and the containment sump at the same time.

Inspectors reviewed schematic diagrams 45N779-8 Rev. II, dated February 15, 1988, 45N779-25 Rev. W, dated February 15, 1988, and 45N779-11 Rev. V, dated March 11, 1988, that describe the interlock provisions for RWST outlet valves 72-21 and 72-22, as well as containment sump isolation valves 72-20 and 72-23. The

review included three inquiries to TVA regarding the manual suction transfer scheme.

- ° The inspector noted a safety evaluation assumption cited in SQN-DC-V-27.5, Rev. 2, paragraph 3.9.1, wherein 110 seconds is assumed from CS pump shutoff to pump restart. TVA was asked to provide the basis for a manual switchover (rather than an automatic switchover) in light of this apparent requirement for operator action within less than 2 minutes. TVA clarified the statement in SQN-DC-V-27.5, stating that the 110 seconds is not a limiting value for the manual switchover, but rather represents their evaluation of the operator action sequence time as described in FSAR Table 6.3.2-5. It was further stated that the limiting value as determined by the containment pressure analysis is 310 seconds, per Table 1 of the "Westinghouse Report for the SNP Units 1 and 2 Containment Pressure Calculations with an Extended Containment Spray Pump Loading Delay" dated June 1987. Considering that the switchover is a manual preplanned operation prescribed by procedures, the inspector concluded that the requirement for manual actuation within 310 seconds was acceptable.

- ° The inspector noted that a postulated single failure to the MOV circuits of interest might result in connection of the containment sump and RWST through a single train of valves. The inspector asked that TVA address the consequences of this postulated event and demonstrate that the consequences were within the design basis. TVA demonstrated that the consequences of the postulated failure would be an early switchover to sump recirculation (since the RWST level would drop more rapidly prior to switchover). The FSAR in section 7.6.6 takes credit for measures that would preclude spurious actuation of SIS sump valves 63-72 and 63-73; therefore, we do not include spurious actuation of these valves in our postulated scenario. Accordingly, these valves may be assumed to remain closed until recirculation. Following switchover, TVA demonstrated that no loss of inventory would result, spray flow would not be interrupted, and spray temperatures would be within design values (since ice melt would not be complete). TVA was asked to demonstrate that any release paths to atmosphere (via the RWST) during this postulated single failure have been adequately addressed, taking into account check valve leakage. This issue is designated URI 327,328/88-29-06, Example 1., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

- ° The interlocks that prevent operation of the sump isolation valves are bypassed by a "seal-in" while the sump valve is opening. Consequently, there is a very brief interval where the RWST outlet valve could be reopened from the control room, defeating the interlock. However, this vulnerability is limited to a short interval during the first (nominal) 5% of sump valve travel, during which the postulated overriding operator action would need to be taken to defeat the interlock. Also, if this were to occur, the corresponding interlock in the RWST valve circuit would be quickly reinstated, limiting the RWST valve travel for this event. Emergency Instruction (EI) ES-1.2, Transfer to RHR Containment Sump, Rev. 5, dated January 12, 1988, explicitly prescribes the sequence of operator actions required to achieve the manual transfer. Operating the valves in the postulated manner would violate the Emergency Instruction.

TVA's response noted that check valve 72-506 is provided as a second isolation point for the postulated release path (Reference FSAR Table 6.4.2-8). They stated that this check valve was manufactured and tested to an acceptance criteria of 24 cc/hour of seat leakage, and the valve is maintained under the ASME Section XI valve testing program to assure operability. In addition, a water leg is maintained from the RWST that provides an additional barrier to the atmosphere. The inspector found TVA's responses to be acceptable.

The inspector concluded that this limited vulnerability for defeat of the interlocks is acceptable since defeat would require violation of the EI by the operator and would only be possible for only a very brief time interval.

Based on the foregoing review, the design of the interlocks appears to be acceptable pending a satisfactory resolution of URI 327 328/88-29-06, Example 1.

- (3) An inspection of the design of the provisions for the automatic activation of the CS system was performed. The system is based on activation of two out of four of the containment hi/hi pressure channels.

The inspector reviewed several design attributes for these channels against governing design requirements. FSAR 7.3 includes the channel functional, performance, and testing requirements. The team determined from TVA that the rack mounted analog instrumentation, actuation logic, and ESF test cabinets supporting the CSS actuation signals were of a generic design provided by the NSSS vendor and previously reviewed by

the NRC staff. On that basis, we focused our review on the field instrumentation.

TVA verified that no high energy line breaks requiring mitigation by containment spray or high containment pressure initiate safety injections would cause common mode failure of the containment/annulus differential pressure instruments. On this basis, the physical separation provided by the 4 field instrument racks appears acceptable. With regard to seismic supports for the impulse line tubing, TVA is verifying this and other field routed tubing as a part of TVA's longer term commitment to verification of field routed instrument lines against engineered installation criteria. With regard to potential effects on response time, TVA reported that the sensing lines are 1/2" schedule 80 piping having maximum length of about 20 feet, the penetrations are 3/4" schedule 160, and no restrictions exist in the lines. On that basis, inspectors concluded the lines would not adversely affect response time.

The inspector noted that TVA is currently replacing the differential pressure transmitter 1-PDT-30-44 with Foxboro Model NE13DM. The demonstrated accuracy calculation dated April 10, 1988, for these new transmitters indicated a range of -1 to 13 psig which was less than the range stipulated in FSAR Table 7.5.1-2. TVA determined that this was an isolated typographical error appearing once in the calculation and did not affect the results. TVA will correct the calculation document.

It was also noted that the accuracy calculation establishes the tolerances for post-accident indication and not for the actuation signal setpoint. TVA indicated that the tolerance for the actuation signal value is determined by WCAP 11239 Rev. 2, dated September 1986. TVA was asked to confirm that the new transmitters are bounded by WCAP 11239 and to more explicitly demonstrate or clarify the margin between the actual setpoint value and the measured value. TVA retrieved NSSS vendor documentation demonstrating to the team that these aspects of the tolerance calculations were properly covered by the vendor's support of WCAP 11239.

TVA was asked to provide assurance that obstruction or isolation of the lines would not go undetected, noting that the transmitter process input simulation for test purposes is reported to be done at the instrument rack. If the untested portions of the lines were blocked and not detected, automatic initiation of containment spray and automatic initiation of safety injection on high containment pressure would be defeated. This issue is designated URI 327,378/88-29-06, Example m., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering

Assurance review of the design basis information related to this issue.

- (4) An inspection was performed to verify the discharge isolation valve/pump interlock for one CS train.

Inspectors reviewed schematic diagrams 45N779-10 Rev. 0 dated 6/1/88 and 45N765-7 Rev. 0 dated 4/8/88. The interlock were in conformance to the requirements of FSAR 6.2.2.5 and SQN-DC-27.5 Rev. 2, para. 3.9.1, and acceptable on that basis.

c. The following CS instrumentation design criteria were evaluated:

- (1) The response time design characteristics of the CS system with a containment pressure initiation signal was evaluated.

Inspectors examined the overall design of the CSS actuation signal channels, noting that the hardware and configuration of the analog signal conditioning and actuation logic was of a generic design provided by the NSSS vendor and previously reviewed by the NRC staff. On that basis, the response times of the analog signal conditioning, function modules, logic circuits, and master/slave output relays would be expected to have an acceptable design basis for use in the containment spray system. Similar hardware in the system is used to provide reactor trip and safety injection signals, for example.

The design of the containment/annulus differential pressure transmitters and impulse lines was also examined (as reported in item 3.b.(3)), and it was concluded that the design basis for the response time of these instruments appears to be bounded by the design basis described in FSAR 7.3, and is therefore acceptable.

- (2) The design relationship between the SI system and the CS system while in the recirculation mode is described in design document SQN-OSG7-008. This design document addresses the containment sump minimum level at the time of switchover to the recirculation mode and the allowable margin for RWST level instrument inaccuracy for a large LOCA.

- (3) Design Evaluation of RWST Level Channel Accuracy

Inspectors checked instrument accuracy calculation B43871001915, RWST Level, Rev. 5, dated October 1, 1987. This calculation established the demonstrated accuracy for RWST level channels 1-LT-63-50, -51, -52, 53 which are used for automatic switchboard and low-low alarm. These instrument channels also include level indicators.

The TVA instrument calculation appears to establish different (but more conservative) values than WCAP 11239 Rev. 2, and TVA

was ask to demonstrate that there were no inconsistencies in the approach and that the TVA methods are intended to be generally consistent with WCAP 11239 methodology.

In addition, a review was conducted of interfacing instrument accuracy assumptions embodied in TVA calculation SQN-0SG7-0008, Containment Sump Minimum Level at Time of Switchover to Reconciliation Mode for a Large LOCA, Rev. 4, dated May 6, 1988.

The review of these assumptions identified that slightly less conservative accuracies were assumed in SQN-0SG7-0008 for the automatic switchover (low level) and alarm (low-low level) signals than were demonstrated by the latest revision (Revision f) to the instrument calculation. This appears to be a case where the latest revision of the instrument calculation was not used. The consequences of this error are nonconservative, but are insignificant with respect to safety (less than 2000 gallons volume and approximately 2 inches water of NPSH). However, the programmatic issue of maintaining current calculation cross references should be addressed by TVA. We understand that TVA will issue a PIR addressing and correcting this inconsistency.

It was noted by the team that if the indicators shown in the instrument calculation were used for post-accident monitoring (PAM), the indicator channel accuracy does not appear to be within the plus or minus 3% of span specified in FSAR Table 7.5.1-2. TVA was asked to demonstrate that the PAM RWST level indicators meet the FSAR requirements. In addition, TVA should demonstrate that the RWST level indication used for TS operability determination (i.e., assurance of adequate RWST inventory) has been properly assessed for demonstrated accuracy.

This item remains open pending TVA's clarification of the WCAP 11239 results, issuance of a PIR regarding the calculation discrepancy, and demonstration that the RWST level indication channels and allowable values for PAMs and TS values have been properly assessed for the effects of instrument errors. This issue is designated URI 327,328/88-29-06, Example n., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

4. Component Environmental Qualification Inspection

- a. Field walkdowns were conducted on selected Containment Spray motor operated valves, instruments, motors, penetrations and electrical equipment. All Raychem splices encountered in motor operated valves were inspected for evidence of deterioration, damage, overlap, bend radius, and nuclear sealant at the wire/Raychem interface. Limit

switch covers, gaskets and gasket sealing surfaces were inspected for condition and damage. Instruments were inspected for broken conduits, loose connection boxes, and loose transmitter/indicator covers. Junction boxes and conduit covers were not removed to inspect Raychem splices. Containment spray pump motors, penetrations and other electrical equipment were physically inspected during the walkdown.

Issues identified in the environmental qualification walkdowns were limited to improper wire repairs within Containment Spray motor operated valves. The specific details are discussed in section 2 of this report. The issues involved use of electrical tape to repair damaged MOV internal wiring.

- b. EQ binders for the following components were sampled for anomalies with EQ design bases and other engineering criteria (such as environmental data and component functional requirements):
- (1) Containment Spray Pump Motor (SQNEQ-MOT-001 Rev. 10, March 11, 1988).
 - (2) MOV 1-FCV-72-20: Sump isolation to CSP suction MOV 1-FCV-72-22: RWST isolation to CSP suction (SQNEQ-MOV-004 Rev. 15, May 6, 1988).
 - (3) 1-FT-72-13, -34: CSS Header Flow Transmitters (SQNEQ-XMT12-003, Rev. 13, May 19, 1988).

No anomalies were found in the binders, and they appeared comprehensive and readily auditable. However, a potential programmatic problem was identified regarding possible use by engineering of "as constructed" environmental data drawings (47E235 series). In the review, the inspector used "as-constructed" drawings provided by TVA, and noted technical errors on those drawings regarding post-accident integrated dose for certain plant areas; these areas were incorrectly identified as mild radiation ($<10^4$ Rad). More current "as-designed" drawings subsequently retrieved by TVA had correct information. The inspector had a concern regarding the possibility that a design engineer might mistakenly use the obsolete "as-constructed" drawings and incorrectly specify the service environment for new equipment. The licensee currently has a comprehensive drawing control and upgrade program in progress.

This issue was identified as a deficiency and provided for licensee information.

5. Maintenance and Housekeeping Inspection

Inspectors performed a verification on a sampling basis of the adequacy of the maintenance program as applied to the CS system. The following was considered:

- Management/supervisory involvement.
- Maintenance instruction enhancement.
- Preventative maintenance.
- Maintenance training.
- Adequacy of recent work requests performed on the system.
- Review of work requests encountered during walkdowns.

Inspection for maintenance condition of system components (leakage, integrity, bent stems, missing or improper fasteners, mounting, preservation, hazards) during walkdowns.

The inspector verified the adequacy of all hold orders associated with the Containment Spray System. The inspector reviewed selected completed work requests for adequacy and reviewed the licensee's work control printout of all open work requests on the Containment Spray System. No discrepancies were noted in any of the maintenance activities reviewed by the inspector.

The inspector reviewed the licensee's preventative maintenance program as it applies to the containment spray system. The inspector determined through discussions with the licensee that a PM upgrade program is presently in progress to improve and standardize the PM program and procedures. The first phase of this improvement program is scheduled for completion prior to the end of 1988. The program appeared to be adequately implemented.

The inspectors conducted a housecleaning inspection of the Unit 1 containment spray pump rooms, heat exchanger (HX) rooms, and pipe chases and had the following observation. Large amounts of insulation were scattered about the pump and HX rooms and the 690 pipe chase. The insulation removal was a result of ongoing work. The overall condition of the plant was acceptable with respect to housekeeping with the exception of maintenance discrepancies on motor operated valves discussed in Section 2 of this report.

6. Structural Supports

Design and in-plant field observation phases of the inspection were used to evaluate Unit 1 for proper design and design implementation relative to structural supports. In each of the following sections, the design phase of the inspection evaluated the system and components against the applicable standards listed at the end of this report and the SYSTERS/design basis reports for the Unit 1 CS system.

The inspectors walked down the CS system and selected a number of hydraulic and mechanical snubbers, other pipe supports, cable tray supports, and equipment foundations for a detailed review. The detailed review was performed to verify whether or not the installation was in accordance with design drawings and that the installation was technically adequate.

The inspectors reviewed the construction specifications related to structural steel and support activities, including welding, to ascertain whether the specified technical requirements conform to the commitments contained in Chapter 3 and 5 of the FSAR, design document SQN-DC-V-1.0, and other design documents.

a. Pipe Snubbers and Other Pipe Supports/Restraints

TVA's program to review the rigorously analyzed piping and supports for Sequoyah Unit 1 was presented to the NRC on April 14, 1988. As noted in TVA's presentation, TVA's program philosophy for this review uses the similarity between Units 1 and 2, addresses open NRC Bulletin 79-14 items, and performs an integrated evaluation of as-built data, nonconformances (CAQRs, NCRs, PIRs, SCR), and as-built discrepancies.

The specific elements in TVA's review program include field walkdowns, a review of the calculations of record, and revision or regeneration of these calculations as warranted, and modifications to supports performed pre- or post-restart.

The scope of TVA's program includes the review of 25 safety-related piping systems, 162 piping analyses, and approximately 2900 pipe supports. Specific program procedures, originally implemented for TVA's program to regenerate the pipe support calculations for Unit 2, include:

- (1) Civil Engineering Branch Instruction CEB-CI21.80, Program Plan for Calculation Regeneration of Pipe Supports on Rigorously Analyzed Category I Piping - Sequoyah 2, Revision 1, dated August 28, 1987.
- (2) CEB-DI21.81, Generation and Control of Rigorous Analysis Problem Connectivity Diagrams for Category I Piping: Sequoyah 1 and 2, Revision 2, dated March 17, 1988.
- (3) CEB-DI21.83, Functional Verification of Supports for Rigorously Analyzed Category I Piping: Sequoyah Unit 2, Revision 4, dated March 17, 1988.
- (4) CEB-CI21.84, Control of Correspondence and Transmission of Design Documents Between TVA and Engineering Services Contractors, Revision 2, dated April 26, 1988.
- (5) CEB-DI21.85, Generation of Pipe Support Design Data - Sequoyah 1 and 2, Revision 3, dated April 25, 1988.
- (6) CEB-CI21.88, Control of Input and Output from the Sqn Hanger Tracking Subprogram of CCRIs (Sequoyah Nuclear Plant Only) Revision 1, dated October 19, 1987.

- (7) CEB-CI21.89, Modification Priorities for Pipe Supports on Rigorously Analyzed Category I Piping - Sequoyah Units 1 and 2, Revision 3, dated April 7, 1988.
- (8) CEB-CI21.90, Gang Hanger and Terminal Anchor Procedure - Sequoyah Units 1 and 2, Revision 1, dated March 17, 1988
- (9) CEB-CI21.91, Handling of Pipe Support Calculation Review/Regeneration Results - Sequoyah 2, Revision 0, dated December 18, 1987.
- (10) CEB-CI21.92, Red Lining of Pipe Support Drawings, Sequoyah 2, Revision 0, dated December 14, 1987.

Revision 4 of CEB-DI 21.83 notes that piping functional verification of Unit 1 will be performed in accordance with TVA SQN Special Maintenance Instruction SMI-0-317-69, Performance of Walkdowns for Verification of Plant As-Installed Configuration, Revision 0, dated November 14, 1987.

Revision 3 of CEB-DI 21.85 notes that the procedure for the generation of the remaining support loads for Unit 1 piping is primarily defined in the Gilbert Commonwealth Report Task R0055, Rigorously Analyzed Piping Program/Program Document, Revision 0 (preliminary issue).

Revision 3 of CEB-DI 21.87 added TVA's commitment to assess the adequacy of the pipe support calculations prepared by Gilbert Commonwealth. TVA will perform a general technical review of at least ten percent of the calculations and perform a thorough line-by-line review of at least 50 pipe support calculations. TVA's overview of Gilbert Commonwealth's work parallels TVA's overview of the pipe support calculations which Bechtel and Stone and Webster prepared for Unit 2.

Design Criteria No. SQN-DC-V-24.2, Supports for Rigorously Analyzed Category I Piping, Revision 2, dated November 30, 1987, establishes the design requirements for the evaluation or design of pipe supports for rigorously analyzed Category I piping.

Re-evaluated pipe supports which cannot meet the requirements of Design Criteria SQN-DC-V-24.2 can be evaluated to the interim criteria detailed in CEB-CI 21.89. Supports which meet these interim criteria can be reviewed to SQN-DC-V-24.2 after restart and modified. Support designs which cannot meet the interim design criteria of CEB-CI 21.89 must be reconciled, either by enhanced analysis or modification, prior to restart.

On June 30, 1988, TVA indicated that approximately 2800 of 2900 pipe supports had been reviewed, that 2430 pipe supports required no modification, that 200 supports required modifications before

restart, and that 170 pipe supports required modification post-restart. TVA has indicated that slightly more than half of the scheduled pre-restart modifications have been completed to date.

The following snubbers and rigid pipe supports were design reviewed considering the attributes listed below:

<u>Snubber Mark No.</u>	<u>Size</u>	<u>Pipe Stress Iso. No.</u>
1-CSH-100	10	0600102-01-03
1-CSH-77	10	0600102-01-03
1-CSH-78	10	0600102-01-03
1-CSH-48	10	0600102-01-02
1-CSH-99	3	0600102-01-04
1-CSH-74	10	0600102-01-03
1-CSH-95	10	0600102-01-04
1-CSH-96	10	0600102-01-04
1-CSH-75	10	0600102-01-03
1-CSH-44	10	0600102-01-02
1-CSH-14	10	0600102-01-01
1-CSH-18	35	0600102-01-01
1-CSH-15	10	0600102-01-01
1-CSH-47	10	0600102-01-02
1-CSH-17	10	0600102-01-01
1-CSH-36	10	0600102-01-01
1-CSH-35	10	0600102-01-02

<u>Restraint Mark No.</u>	<u>Drawing No.</u>
1-CSH-444	1-H21-468
1-CSH-413	1-H21-426
1-CSH-403	1-H21-406, 1H21-407A
1-CSH-400	1-H21-401, 1H21-401A

The specific attributes which were reviewed for the above pipe supports are listed below:

- ° The pipe support was designed to restrain the pipe loadings in the proper direction and location.
- ° The proper loading value, direction and orientation was transmitted to the pipe support designer from the pipe stress analyst.
- ° The structural analyst checked the pipe support stresses against the proper allowables for all structural shapes in axial, bending, shear, combined stress interaction, web crippling, etc.
- ° Local stresses induced into the piping by welded attachments were properly considered.

- Torsional stresses were properly considered.
- Concrete anchor bolts, including base plate flexibility, were addressed.
- Support rigidity/frequency or deflection limits were considered properly.
- Component standard supports properly applied.
- Structural computer program used in an appropriate manner using proper units, loadings, properties, geometry, etc.
- The as-analyzed pipe support configuration agrees with or has been properly compared with the as-built condition.
- Weld configurations and sizes used in the analysis agree with the as-built configurations and sizes.
- All structural analysis computer output has been properly reviewed.
- That proper consideration has been given to special design considerations such as thermal environmental conditions.

In general the pipe support calculations reviewed were prepared in such a manner that enabled the work to be reviewed with little outside explanation. The calculations were considered to be auditable and generally well documented. During the review of the above pipe supports several items were identified which require TVA action.

On page 6 of the pipe support calculation for 1-H21-17 the loadings are identified in a local direction. However, the piping isometric and stress analysis calculations clearly indicate that the direction of restraint is the global Z-direction. Existing Unit 2 calculations were used to qualify the Unit 1 support and the Unit 2 calculation was performed using the proper load orientation. Therefore, this item results in only a documentation error which TVA should correct.

This issue was identified as a deficiency and provided for licensee information.

For support calculation 1-H21-100, it was not apparent that the as-installed direction of the snubber is in agreement with the restraint direction indicated in the pipe stress analysis. It was necessary to perform a calculation to insure that the restraint was installed in the proper direction. Generally, a pipe support's direction of restraint is obvious and can be determined from just looking at the support drawing. However, for supports with complicated geometry such as supports connecting to the steel

containment vessel, a calculation may be required when the restraint direction is not clear from just looking at the support drawing. TVA should consider performing the necessary calculations where required to assure that the pipe support has been designed to restrain the proper loading direction.

This issue was identified as a deficiency and provided for licensee information.

Page 16 of pipe support calculation 1-CHS-96 contains a weld calculation which indicates a weld size of 0.132 inch is required and a 0.155 inch weld is provided. However, the calculation also states that a significant portion of the weld was determined to have a size of only 0.08 inch. Therefore, the weld as analyzed would result in an overstressed condition. Also the weld calculation is not very clear in its determination of weld size. TVA was notified of this issue and should resolve the inconsistency.

This issue is designated URI 32/323/88-29-06, Example c., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

The inspector's review of pipe restraints 1-CSH-400, -403, -413 and -444 indicated no deficiencies with respect to the design attributes.

In conclusion, the sample of rigorously analyzed snubber and rigid pipe supports were reviewed for the attributes listed above and with the exceptions discussed, the supports were determined to have been adequately addressed by TVA. The analyses of record for these pipe supports reviewed were considered adequate and meet the FSAR and design commitments relative to the attributes reviewed.

The inspectors selected seven pipe snubbers and five other pipe supports/restraints associated with the CS system and performed visual inspections with the aid of measuring devices and inspection mirrors to verify the installations were as depicted in as-constructed drawings. These installations were inspected for:

Deterioration and Leakage.

Correct structural member, bolts, and fasteners properly installed.

Moving/rotational parts were free to move.

Alignment.

Interferences.

Fluid level (hydraulic).

The following snubbers were inspected:

Mark No.	Size
1-CSH-36	4 (hydraulic)
1-CSH-7	10
1-CSH-37	10
1-CSH-65	10
1-CSH-66	10
1-CSH-67	10
1-CSH-470	3

Seven discrepancies were noted on five snubbers. No discrepancies were noted on two snubbers. The following table provides a detailed description of the NRC inspection findings.

Snubber/Drawing/Discrepancies

1-CSH-7 (H21-7)

- ° Undersize weld. Rear bracket to plate (piece 7) is 3/16" vs 1/4" as required.
- ° Undersize weld. Wide flange (piece 9) to horizontal wide flange is 3/16" vs 1/4" as required.
- * Piece 7 is 6 3/4' by 6 3/4" rather than 7" by 7" as required by the drawing.

1-CSH-65 (H21-65)

- ° Snubber indicator tube/cap assembly is bound up in pipe clamp.

1-CSH-66 (H21-66)

- ° No deficiencies identified.

1-CSH-470 (H21-499, 500)

- ° No deficiencies identified.

1-CSH-67 (H21-67)

- ° Piece 3 in bill of material not shown on detail.

1-CSH-37 (H21-37)

- ° As-configured drawing specifies 1-1/4 inch wedge anchor bolts. Actual installed bolts are 1 inch.

1-CSH-36 (H21-36)

- * Drawing does not show a weld or weld details for attachment of rear bracket to piece 2.

Discrepancies marked with an asterisk were identified in both NRC and TVA walkdowns.

In addition, TVA performed an inspection of 10 CSS snubbers prior to the NRC inspection. The following table categorizes the discrepancies noted during both inspections:

<u>Discrepancy/Resolution Category</u>	<u>NRC (7 samples) # Observed</u>	<u>TVA (10 samples) # Observed</u>
Inadequate/incorrect drawings	1	1
Hardware/installation discrepancies noted by TVA, evaluation acceptable, drawings. Not yet changed.	1	2
Hardware/installation discrepancies not previously identified.	5	2

The inspectors also selected five other types (struts, springs) of pipe supports/restraints. During the inspection of the five supports/restraints, additional discrepancies were noted on two adjacent Auxiliary Feedwater System supports.

Twenty one discrepancies were noted on these seven supports/restraints. At least one discrepancy was noted on each support/restraint examined. The following table provides a detailed description of the NRC inspection findings:

1-CSH-400 (H21-400,401,401A)

- *1. Drawing does not show weld details for clamp stiffeners and bracket to clamp joint for west strut.
- *2. Spacers, piece 3 on drawing, are not installed (or required).
- *3. Expansion anchor/bolt assembly for west baseplate is not identified on bill of material.
- *4. Connection welds for piece 12 exhibit poor weld contour.
- *5. Piece 17 is 6 by 6-3/4 inches. Drawing specifies 6-1/2 by 7 inches.

1-CSH-401 (H21-402, 403)

1. As-constructed drawing specifies size 11 spring cans. Actual spring cans installed are size 9.
2. Spacer plates are not centered as show on drawing. Drawing does not show/specify orientation of spacer plates. Drawing does not specify the required weld length between piece 7 and spacer plates. Drawing does not specify weld size, length or location for spacer plate to embed weld.
3. Piece 7 wide flange undersize. Drawing specifies W6X20. Installed flange is W6X15.5.

1-CSH-408 (H21-415, 417)

1. As-constructed drawing specifies size 12 spring cans. Actually installed are size 9.
2. Beam attachment load bolts are actually 3/4 inch diameter. Vendor catalogue specifies a 7/8 inch bolt.
3. Washer plates are not welded to back channels on outboard ends. Drawings specify an all around weld.
4. Fabricated U-bolt (piece 3) is bent on both sides and thus center to center spacing specified on the drawing as 1 ft 1-3/4 inches is actually 11-5/8 inches.
5. Weld attaching wide flange piece 6 to existing wide flange exhibits poor weld contour.
6. Drawing provides no weld details for attaching piece 6 to existing wide flange.
7. As-constructed drawing specifies a 3/4 inch rod and beam attachment. The vendor catalogue specifies a 1 inch diameter rod and beam attachment for the size 12 spring can detailed on the drawing.

1-CSH-413 (H-426, 427)

1. Undersize wide flange. Piece 1 is W5X15.5. Drawing specifies W6X20.
- *2. Drawing does not specify weld details for the rear bracket to wide flange weld.

1-CSH-449 (H21-473)

- *1. Undersized welds for pieces 2 and 3 to baseplates. Drawing specifies 1/4 inch, actual is 3/16 inch.

1-AFDH-300 (H3-329, 330)

1. The 5/8 inch beam attachment that is installed requires a 3/4 inch diameter load bolt. Actually installed is a 1/2 inch diameter bolt.
2. The drawing is in error in that a 1/2 inch diameter rod and beam attachment assembly is specified for a size 7 spring can. This size spring can requires a 5/8 inch diameter bolt per the vendor's catalog.

1-AFDH-301 (H3-332)

1. Drawing specifies a 5/8 inch beam attachment with bolt. The catalog requires a 3/4 inch diameter load bolt for this attachment. Actually installed is a 5/8 inch bolt that is threaded full shank. Grinnel supplied load bolts are not threaded full shank.

In addition, TVA performed an inspection of 14 non-snubber CSS supports/restraints prior to the NRC inspection. The following table categorizes the discrepancies noted during both the NRC and TVA inspection.

<u>Observation/Resolution</u>	<u>NRC (7 samples)</u>	<u>TVA (14 samples)</u>
<u>Resolution</u>	<u># Observed</u>	<u># Observed</u>
Inadequate/incorrect drawing.	6	5
Hardware/installation discrepancies previously noted by TVA, evaluated as acceptable but not yet included in drawings.	2	9
Hardware/installation discrepancies not previously identified.	13	19

The discrepancies identified by TVA in their pre-SSQE walkdown included undersized welds, missing locknuts, undersized material and dimensional discrepancies. The new discrepancies identified by the NRC inspector included undersized welds, load bolts, structural shapes, concrete expansion anchors and spring can assemblies. Nineteen of the 29 discrepancies identified by the NRC had not been identified during the previous TVA walkdown programs or by the walkdown TVA conducted prior to the SSQE inspection.

Numerous instances were identified where design features, such as weld details, were not specified on the drawings used for construction and inspection. Undersized welds, expansion anchors and load bolts indicate that either inadequate modifications have been performed or the supports had been inadequately inspected. In addition, numerous discrepancies had not been identified earlier during previous TVA walkdown/inspections either due to a different scope of inspection or an inadequate inspection. Inadequate design change controls are evident by issuance of as-configured drawings indicating that ECN 5277/WP9911 had been completed on 1-CSH-401 and 408 when the required larger spring hanger assemblies had not been installed.

The failure to install pipe supports and restraints 1-CSH-401, and 408 in accordance with design drawings is violation 327, 328/88-29-03, example 2, Structural walkdown discrepancy.

This issue requires resolution prior to the startup of Sequoyah Unit 1. Adequate corrective action for violation 327,328/88-29-03, Failure to Install Components in Accordance With Design Documents, will include retrieval, generation or regeneration of sufficient system operability determination information necessary to resolve this issue. In addition, adequate corrective action for the above mentioned violation shall include a Quality Assurance review of the TVA pre-SSQE walkdown discrepancies and the licensee's previous field inspection/walkdown programs for this type of component.

b. Equipment Foundation Inspection

The inspector selected two equipment foundations on the CS system and performed visual inspections and measurements to verify compliance with design drawings and support documents. A design evaluation was performed on the CS Heat Exchanger 1A and CS Pump 1A-A and 1B-B supports.

- (1) CS Heat Exchanger Support 1A calculation SCG 1S 180 (B25 880113-801), Rev. 0, dated January 13, 1988, indicated that the Unit 1 Containment Spray Heat Exchanger support adopts calculation SCG 1S 179, Rev. 0, dated December 19, 1987, which analyzed the identical heat exchanger supports in Unit 2. These calculations were performed as a result of the discovery of inadequacies in the original calculations as identified by CAQR SQP 870188, dated March 11, 1987. This CAQR identified the inadequacy of the calculations in that improper vendor loads had been used. Review of Drawing 48N1231 demonstrated that the heat exchanger foundation supports are identical except they are mirror images. The Unit 2 corrective measure was reviewed by the previous NRC IDI team and found to be acceptable (Inspection Report 327,328/88-13, May 26, 1988). The IDI report also outlined the past sequence of activities which led the inspectors to the current issue which is identified as improperly considered nozzle loads.

TVA completed the same field modification for Unit 1 as was completed for Unit 2. Subsequently, it was determined that Unit 1 nozzle loads differed considerably from those of Unit 2. The team was informed that this condition is documented in CAQR SQP 880363, dated May 27, 1988. This has been determined to be a restart item by TVA.

This issue is designated URI 327,328/88-29-06, Example p., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

- (2) SQN FSAR Table 3.2.1-2 specifies the containment spray pumps as TVA Class B seismic Category I components, designed in

accordance with the Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, dated 1968, and March 1970 Addenda.

TVA procured the CS pumps in accordance with the design criteria contained in TVA purchase specification No. 71C30-92646, Pumps, Centrifugal, Electric-Motor-Driven, which TVA prepared on November 9, 1970.

TVA design specification No. N2M-46 R0, Sequoyah Nuclear Plant Units 1 and 2/Centrifugal Pumps for Containment Spray, dated May 18, 1972, forms a part of the CS pump purchase specification, and contains detailed design provisions in specification 1153 for Electric-Motor-Driven Centrifugal Pumping Units for Containment Spray Service for Sequoyah Nuclear Plants 1 and 2. TVA prepared this specification in compliance with paragraph N-141 of the draft ASME Code for Pumps and Valves for Nuclear Power, dated November 1968.

The functional requirements for the pump and motor detailed in Specification 1153 are in accordance with the pump and motor design parameters specified in the FSAR. Section 22 of Specification 1153, Seismic Requirements, notes in part that:

The pump-motor assembly and all individual parts of the pump shall be designed to operate satisfactorily during earthquake forces resulting from acceleration in the horizontal and vertical directions. The forces are 1.0 g horizontal and 0.67 g vertical, applied simultaneously at the center of gravity. The entire assembly must be designed to receive and transmit these forces through the supports to the foundation.

TVA procured the CS pumps prior to TVA's formal implementation of Appendix F, Design Criteria for Qualification of Seismic Class I and Seismic Class II Mechanical and Electrical Equipment, which TVA issued on February 11, 1971, and which TVA subsequently used to specify the seismic requirements for safety-related mechanical and electrical equipment.

The CS pump is shown on Weise & Monski Outline Drawing No. UE 032-12.50-2, Rev. 1, dated June 23, 1972.

Weise & Monski calculation No. TP-001-2, Seismic Calculation of Strengths, Containment Spray Pump, Rev. 1, dated July 27, 1972, (RIMS No. 870824T0771) was originally prepared to qualify the CS pumps to the design criteria detailed in TVA Specification 1153.

However, based on two generic deficiencies which the NRC identified during an inspection of SQN Unit 2 during the latter part of 1987, Deficiency D3.4-6, Vendor Seismic Qualification

Reports, and Deficiency D3.6-1, Equipment Anchorages, TVA prepared the following calculations:

- TVA calculation No. SCG-4M-168, Containment Spray Pump, Rev. 1, dated June 20, 1988 (no RIMS No.).
- TVA calculation SQN-CEB-SCG-2E-00S-375, Seismic Qualification, Equipment, Rev. 0, dated February 12, 1988 (RIMS No. B25 88 0215 319).

United Engineers (UEC) prepared Rev. 0 of the first calculation for TVA on November 24, 1987, to compute the anchor bolt loads for the Unit 2 CS pump. TVA technical staff revised the calculation to incorporate the Unit 1 CS pumps.

Impell prepared the second calculation for TVA to re-qualify the Unit 2 pump to the governing mechanical and seismic loads in accordance with the ASME design code of record. As noted in the Impell calculation, the original qualification report which Weise & Monski prepared for the CS pump was prepared in accordance with German standards applicable to commercial pumps, and did not provide documented evidence that these criteria were consistent with the ASME code requirements for the CS pump materials and pump assembly.

Review of the first TVA calculation indicated that UEC computed the seismic component of the anchor bolt tension and shear by using the zero period acceleration (ZPA) loads instead of the 1.0g horizontal, 2/3g vertical accelerations specified for the CS pump in the purchase specification. The ZPA loads are about 15 percent of the specified seismic loads. UEC chose to use the ZPA loads based on the assumption that the CS pumps are rigid. However, UEC's use of the ZPA loads to compute the seismic component of the anchor bolt loads is unconservative with respect to Specification 1153. The inspector noted that Impell's calculation to re-qualify the pump assembly uses the correct seismic loads.

TVA's design of the pump foundation pad, which uses the anchor bolt loads as input design loads, also needs to be reviewed.

The second calculation, which Impell prepared for TVA, requalifies the CS pump assembly, in part, to the manufacturer's allowable suction nozzle loads and to the Unit 2 discharge nozzle loads calculated in the piping analysis of record by TVA.

However, the calculation does not address the seismic qualification of the Westinghouse motor, or the qualification of the bolts which restrain the motor to the pump baseplate.

The CS pump motor is shown on Westinghouse drawing No. 269CG80, TVA contract No. 71-92646.

To address this concern, TVA prepared calculation MCL A09/SCG-4M-00457, Seismic Qualification of Containment Spray Pump Motor, Rev. 0, dated June 24, 1988. However, the calculation does not confirm the seismic qualification of the motor and the motor hold-down bolts to the 1.0g, 2/3g design seismic loads.

TVA provided the inspectors with an additional calculation entitled Sequoyah Nuclear Plant, Containment Spray Pump Nozzle, Rev. 1, dated February 23, 1982 (RIMS no. CEB 82 0225 002). The team recommends that this calculation be voided, since the calculation appears to be superseded by the calculation which Impell recently prepared for TVA.

In summary, the team notes that TVA needs to perform the following:

- Check the CS pump base plate anchor bolts for the design seismic loads.
- Review the design for the CS pump foundation pad for these revised loads.
- Requalify the CS pump for the Unit 1 nozzle loads.
- Provide evidence that the CS pump motor and motor hold-down bolts are qualified for the design seismic loads.

This issue is designated URI 327,328/88-29-06, Example q., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

The inspectors reviewed a calculation for pump foundation supports SCG1S173x106 (B25 88-1029-482), dated January 29, 1988. Drawing 41N307-3 locates pump and mark number 41D307-1 provides the bolt details. Drawing 41N353-1 Revision 4, contains dimensions of the pump concrete foundation.

The pump is held down to a 2'-3" reinforced concrete slab by six 1" diameter A307 bolts. There is 5/4" of grout and a 6" thick concrete pad between the concrete slab and the bottom of the pump support frame.

An anchor bolt is designed for maximum of 16570 lbs. of tension and 9844 lbs. of shear. These loads are results of a separate pump analysis referenced as SCG-4M-00168. The bolt stress

calculation was done in accordance with the methodology provided in AISC Sec. 5-1.6.3 and concluded that they are within allowable stresses. The inspectors performed a simplified and conservative independent calculation and came to the same conclusion. The anchor bolt concrete capacity was investigated for tension pullout load. The grout pad and 6" concrete pad were not considered, which is conservative.

The result indicated that there exists a safety factor of more than four which is acceptable.

(3) The following foundations were inspected in the field:

CS Heat Exchanger 1B per drawing 48N1231, FCR 6873 R1.

CS Pump 1B-B per drawings UE 032-12.50-2, 41N353-1, 41N307, 41N307-3, 41N309, and 41N309-1.

The lower support for the Heat Exchanger 1B was found to be installed in accordance with design drawings with regard to bolting, member size, configuration, and weld size, location and quality. The following discrepancies related to bolting of the heat exchanger to the upper support structure were identified:

- Six of eight fasteners were loose, two with only 1/2 nut engagement.
- One assembly had no washer.
- The remaining seven fasteners had flat washers installed on the sloped inside surface of the structural channel flange. The American Institute of Steel Construction (AISI) Manual of Steel Construction and TVA Modifications and Addition Instruction M & AI 9, "Tightening, Inspection, and Documentation of Bolted Connections", require the use of beveled washers for surfaces that slope greater than 1:20.

A followup inspection by TVA also identified that 3/4 inch diameter bolts were installed. The heat exchanger mounting feet have holes for one inch bolts.

The failure to install CS Heat Exchanger 1B in accordance with design drawings and site procedures is Violation 327,328/88-29-02, example 1.

CS Pump 1B-B was generally installed in accordance with design drawings and site procedures. However, the inspector identified that the holes in the mounting bracket of the vendor supplied pump assembly had been enlarged (slotted), apparently to aid in the alignment of the holes with the anchor bolts embedded in the concrete foundation pad. Vendor drawing UE 032-12.50-2

specifies 1-1/8 inch diameter holes. At least four holes had been enlarged a minimum of an additional 7/16 inch. This minimum dimension is based on measured gaps visible outside the washer and assuming the bolt is in contact with the bracket on the opposite side (nuts were not removed for the inspection). TVA staff indicated that they were unable to provide any documentation to show that this condition had been previously identified, documented, or evaluated for effect on the seismic design basis. As a result of this inspection, TVA is performing new seismic calculations to determine the technical acceptability of the as-installed condition. CAQR SQN 880392 was issued by TVA to address this matter.

The failure to properly control the installation and design changes to CSS Pump 1B-B installed foundation brackets is Violation 327,328/88-29-03, example 3.

The walkdown discrepancies identified in the structural section of this report involve the as-built configuration of the plant. Adequate corrective action for violation 327,328/88-29-03 will include retrieval, generation or regeneration of sufficient system operability determination information necessary to resolve this issue. In addition, adequate corrective action for the above mentioned violation shall include a Quality Assurance review of the TVA pre-SSQE walkdown discrepancies and the licensee's previous field inspection/walkdown programs. These additional corrective actions for violation 327,328/88-29-03 are required to be completed prior to Unit 1 startup.

c. Platform Thermal Growth

TVA found, in May 1985, that structural and miscellaneous steel were designed and installed without proper consideration of thermal loading from a postulated DBA (Staff Safety Evaluation Report, NUREG 1232, Volume 2, on TVA Sequoyah Nuclear Performance Plan - May 1988). Subsequently, TVA completed corrective measures which resulted in several structural modifications introducing connections with slotted holes to allow thermal expansion. The staff reviewed and approved the TVA corrective actions for Unit 2 restart (see above noted staff SER). The modifications were common to both Unit 1 and Unit 2.

The inspector found no unreviewed potential thermal growth interference of structural steel and concrete within the containment spray system. However, the inspectors reviewed the thermal growth interaction between the containment spray piping and the steel containment shell.

It was noted during the inspector's field walkdown that several platforms were attached to the containment spray pipes. Piping isometric drawing 0600102-01-02 details the horizontal restraint of

the pipe at the location of the platforms. These platforms are acting as pipe supports, (Drawing 482412-1, R1) and they are in turn supported by the steel shell containment. Review of shell stress summary report (SCG-CSG-88-091, Rev. 0, B25-88-0227-308) revealed that no stresses from thermal growth were accounted for at the location of pipe supports.

Design basis related thermal growth of containment shell will be reflected in the piping reactions and these reactions are to be incorporated into shell stress analysis. This is required by FSAR Section 3.8.2.3.2., items 4A and 4D. TVA stated that it is their belief that the calculation did not include piping reactions because they were considered to be small.

TVA presented several aspects of conservatism inherent in their shell stress calculation including conservative shell wall thickness close to where the piping is supported. TVA also stated that when they determined that the contribution of piping reactions to shell stresses was considered to be large, such effects were included in the total stress calculation in the final stress report, thus meeting the FSAR requirements. The inspector was unable to complete a review of TVA's information and related calculational packages CSG-87-058, Rev. 0, B41-87-0605-006 SQN Unit 2 - Steel Containment Vessel - Pad Plate Analysis - Containment Spray System Supports and CSG-87-037, Rev. 1, B41-87-1019-008 SQN Unit 1 - Steel Containment Vessel - Pen. X-48A Shell and Nozzle Evaluation.

This issue is designated URI 327,328/88-29-06, Example r., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

d. Cable Tray Support

Cable tray supports for CS pump motor 6900 V power and control cables were inspected on a sampling basis. Selected supports were inspected to the specifications of drawings 48N1370 and 48N1360. The cable tray supports supported power and control trays for CS cables in 6900 V shut down board rooms 1A-A and 1B-B and the surrounding areas.

- (1) Eight cable tray supports on tray AM-A auxiliary building elevation 734 were inspected for proper spacing and location, membersize, configuration and orientation, weld size, attachment location on embedded plates and tray attachment to supports. The following drawings provided the acceptance criteria for this inspection:

48N1338, 45N828-3, 48N1340, 48N1360, 48N1361

One potential discrepancy was noted concerning relocated expansion anchor bolts and enlarged base plate holes. This issue appeared to have little safety significance and was referred to TVA for resolution.

This issue was identified as a deficiency and provided for licensee information.

- (2) The inspector reviewed the TVA design of cable tray supports. Design criteria SQN-DC-V-1.3.4 entitled Category I Cable Tray Support System, Revision 1, December 22, 1986 (B05-861230 501), included Appendix A "Interim Acceptance Criteria for Reevaluation of Category I Cable Tray Supports". The purpose of this appendix is to provide an interim acceptance criteria for the reevaluation to be completed prior to the restart of the Sequoyah units. A set of calculations SCG1S52x1, (Rev. 0, B25 860913 825), SCG1S47x1 (Rev. 0, B25 860913 801), SCG1S47x2 (Rev. 1, B25-861113-818) and SCG1S29x46 (Rev. 1, B25 861113 817) constitutes such a reevaluation.

As stated in NRC staff safety evaluation, NURE6-1232, Volume 2, May 1988, several audits have been completed and both interim criteria and TVA calculational methodology were approved with TVA's commitment that original FSAR criteria for the affected cable tray supports will be restored in an orderly manner after restart. Interim acceptance criteria for cable tray supports are less stringent than those in the FSAR. TVA calculations were performed on worst case bases. The SER stated that regarding the selection methodology, the staff finds that TVA has used good engineering judgement in its selection of the worst cases and found the approach used acceptable for restart.

The inspector selected two cable tray supports for review. They are MK-1B in 48N1337, R 10 and MK-2F in 48N1338R9C. TVA's reevaluation calculation of the cable tray supports also selected MK-1B as a worst case sample (SCG-1S-47x1) even though it is at a different elevation (drawing 48N1360). Therefore, the inspector's review of support MK-1B in 48N1337 constituted a review of the TVA reevaluation calculation. No inadequacies were identified.

Next, the inspectors reviewed support MK-2F to determine if this support is bounded by the worst case support, MK-1B. Consideration of the number of cable trays at the support, span distances to the next supports, support member sizes, and its unsupported length as prescribed in the TVA worst case selection methodology demonstrated that support MK-1B bounds MK-2F. This issue is closed and the inspectors concluded that design of cable tray supports is adequate for restart.

7. Weld Inspection

The inspectors performed field inspections, observed nondestructive examinations (NDE), and reviewed welding records (including RT film, inservice inspection data, etc.) for the following nozzle, piping, and structural welds related to the CS system on a sampling basis. Welds were examined in the field for size, contour, and surface conditions. Documentation was examined for welder qualification, weld procedure qualification, and NDE results of compliance with design requirements.

The inspectors performed inspections in the following areas:

a. Pipe Welds

(1) Field Welds	Drawing (Welding Map)
1199, 1200, 1201 & 1202	CS-4
F-21, 22, 23, 28, 28A & 29	NAVCO A-7204

- Detailed welding procedures were checked.
- Welder identification, qualification, and welding continuity were examined.
- NDE inspections were signed off.
- Qualifications of NDE inspectors were examined.

The NRC inspectors accompanied by licensee welding QC inspectors performed the following reinspections:

<u>Weld No.</u>	<u>Pipe Diameter</u>	<u>Type of Weld</u>	<u>Inspection Performed</u>
1-CSF-21	12"	Butt Weld	Visual and PT*
1-CSF-22	12"	Butt Weld	Visual and PT
1-CSF-23	12"	Butt Weld	Visual and PT
1-CSF-28	12"	Butt weld	Visual and PT
1-CSF-28A	12"	Butt weld	Visual and PT
1-CSF-29	12"	Butt weld	Visual and PT
1-CSF-30	12"	Butt weld	Visual only
1-CS-1199	2"	Socket weld	Visual only
1-CS-1200	2"	Socket weld	Visual and PT
1-CS-1201	2"	Socket weld	Visual only
1-CS-1202	2"	Socket weld	Visual and PT

*PT - dye penetrant inspection.

- (2) The inspectors reviewed the documentation on spool piece Nos. ICS 14 and 15 (Drawing NAVCO A-7204).

b. Equipment Nozzle Welds

The inspectors performed the following inspections on one nozzle for CS pump 1A and two nozzles for CS heat exchanger 1A:

Weld No.	Pipe Diameter	Type of Weld	Inspection Performed
1-CSF-42	12"	Nozzle weld	Visual and PT
1-CSF-31	12"	Nozzle weld	Visual and PT
1-CSF-20	12"	Nozzle weld	Visual and PT

The inspectors reviewed the radiographic film for the following field fabricated welds:

CSC-X23	12" Butt Welds
CSF-28A	12" Butt Welds
CSC-29	12" Butt Welds
CSF-20X1	12" Butt Welds
CSC-31	12" Butt Welds
CSF-42	12" Butt Welds
CSC-28	12" Butt Welds
CSF-30	12" Butt Welds
CSC-21	12" Butt Welds
CSF-22	12" Butt Welds

The radiographic film was reviewed to determine compliance with USAS B31.7 code in the adequacy of weld quality, weld coverage, film density, penetrometer size and location, sensitivity, and geometric unsharpness. The following discrepancies were identified.

The radiographic data sheets which accompanied the file packet failed to reference the radiographic procedure used. However, the licensee was able to reconstruct this information through review of weld data sheets and determining the time frame the welds were welded and radiographed. From the inspection dates the licensee determined the document applicable to the radiography was G-29 Process Specification 3.M.3.1, Rev. 3, Specification For Radiographic Examination Of Welded Joints. As specified in Table 1 of this procedure, in the weld thickness range of 1/4" through 3/8" a number 7 penetrometer with 2T sensitivity is required to show on the radiographic film. The review of the following welds with a nominal wall thickness of .365 and .375 inches revealed a number 10 film side penetrometer was placed on the material.

-	1-CSF-X23	.375 wall
-	1-CSF-28A	.375 wall
-	1-CSF-29	.375 wall
-	1-CSF-20X1	.375 wall
-	1-CSF-31P4	.375 wall
-	1-CSF-42	.375 wall
-	1-CSF-28	.375 wall
-	1-CSF-30	.375 wall

Initially the licensee was unable to provide a basis for using a

number 10 penetrometer when a number 7 was specified except they believed the weld, with reinforcement, was actually thicker than .375 inches. To determine the actual wall thickness, the inspector requested an ultrasonic wall thickness of the following welds.

- 1-CSF-28 Measured minimum wall thickness .372
- 1-CSF-31 Measured minimum wall thickness .246
- 1-CSF-23 Measured minimum wall thickness .381
- 1-CSF-28A Measured minimum wall thickness .426
- 1-CSF-29 Measured minimum wall thickness .462

As noted above welds 1-CSF-28, 1-CSF-31 were below the .375 wall thickness and a number 7 penetrometer is required to meet the requirements specified in USAS B31.7

Table 3.2.2.2 of the FSAK states the code applicable for fabrication, and nondestructive examinations of TVA Class B piping is B31.7.

The licensee informed the inspector that Code Case 115 was approved and allowed the substitution of ASME Section III requirements for piping weld. ASME Section III allows the use of a number 10 penetrometer in the thickness ranges discussed above. Based on the applicability of Code Case 115, the penetrometer selection is acceptable.

The licensee performed calculations of the pipe weld to heat exchanger nozzle weld (CS-1-00-31) with a measured wall thickness of .246 inch thick to determine seismic and pressure/temperature adequacy. Calculation number SCG-4M-00461 determined the weld is seismically adequate. Calculation number SQN-72-D053 determined a minimum wall thickness requirement of .151 inch. Therefore the weld thickness is considered adequate. The inspector determined the wall reduction from the original design nominal wall thickness of .375 inches occurred when grinding was performed during fabrication to remove surface defects found by radiography. The licensee was successful in producing a defect free weld that passed the radiography examination; however, they failed to consider the minimum wall thickness requirements. Quality Control inspection for wall thickness reduction was not included in the inspection program during fabrication of the field piping welds.

Failure to comply with the committed to B31.7 weld standard as implemented by TVA procedure G-29 is violation 327,328/88-29-04 example 1. Adequate corrective action for this violation will include TVA review to determine if minimum wall design requirements were met on other field fabricated pipe welds.

In addition to the above radiographic film review, the licensee re-radiographed the following listed welds for the inspectors review:

- 1-CSF-23
- 1-CSF-28A
- 1-CSF-29

The inspector's review included a comparison of the original radiographic film to the new film to determine if the original weld radiographed matched the weld number shown on the drawing. The new film was also reviewed for weld quality and a determination if intergranular stress corrosion cracking (IGSC) and/or microbiological intrusion corrosion (MIC) had occurred. No fabrication defects were observed and no IGSC or MIC were apparent.

c. Structural Welds

The inspectors reviewed the pipe support calculations for pipe restraints 1-CSC-15, -47, -400, -403, -413 and -444, in part to confirm that the pipe support detail sheets properly indicated weld sizes where required, and that welds were checked to confirm structural adequacy with respect to the forces imposed by the connected members. The inspectors did not identify any deficiencies from this review.

The inspector reviewed a weld calculation of the heat exchanger 1A foundation supports. The calculation required two 1/4 inch fillet welds 6 inches long on each side of the web of a M4 X 13 diagonal member (see SCG 1S 180 Rev. 0, B25-88113-801, dated January 3, 1988, which in turn references SCG 1S 179, B25-88-0223-310, dated February 22, 1988 - P 118 for 1/4 inch welds). The M4x 13 section is a newly added member to strengthen the foundation support required by a recent modification (see CAQR SQP 870 188, 3/1. '87). This weld connects the web of the M4x12 to the flange of vertical member of 8WF35. The angle between two members is shown to be 56.5 degrees.

The theoretically available maximum length for the fillet weld due to geometric constraints including 56.5 degree angle between the members as well as TVA's subsequent field as-built measurement indicated that it is not possible to have more than 4 inches of weld. TVA admitted that 6 inch length in the calculation does not reflect field measurement and a CAQR is being issued reflecting the error in the calculation. TVA has initiated a modification to the calculation with the purpose of showing that the current as-built weld of approximately 4 inches of weld continues to be adequate to satisfy design requirements committed to in the FSAR.

The issue will remain open subject to a review of the calculation. TVA has determined that this issue is to be resolved prior to restart of Unit 1. TVA indicated that revised calculations would include evaluation of calculational conservatism as well as conservatism in the value for the weld allowable stress. In addition, an entire weld calculation of the heat exchanger foundation supports is being re-evaluated for a complete accuracy check. It should be noted that a previous IDI report (NRC Inspection Report 50-327-88-13, May 26, 1988) noted several calculational errors and Rev. 3, of SC GIS 179 that the current inspector has reviewed is supposed to have addressed the items with "line by line review comments". This issue is

designated URI 327,328/88-29-06, Example s., and requires resolution prior to the startup of Sequoyah Unit 1. Adequate resolution for the above URI shall include an Engineering Assurance review of the design basis information related to this issue.

8. Operational and Experience Review Issues

The following areas were reviewed to determine if systemic operational or experience review issues existed.

a. Restart Test Program (RTP)

The inspector reviewed the Unit 2 CS system Restart Test Program (RTP) test matrix and compared the matrix to the Unit No. 1 CS system functions. The inspector also compared the general Unit 1 program against the Unit 2 completed program. The inspection objectives for the CS along with the inspector's findings are provided below.

(1) Unit No. 1 CS Restart Test Program Review

The objectives of this inspection were as follows:

- To verify that the Restart Test Group (RTG) functional review process is being adequately implemented.
- To verify that component/system functions that are identified as requiring testing are properly dispositioned.
- To provide a sample assessment of the technical adequacy of several portions of previously completed preoperational tests that are being used to satisfy the functional testing requirements.
- To provide a sample assessment of the correctness of the FSAR as it related to system functional requirements.

The inspectors reviewed the identified system package to verify compliance to the specified program. Specifically, the following items were addressed during this review:

- Verify that the functional analysis report (FAR) matrix package complied with the following documents as applicable and contained the necessary information:

- Function Review Process - Unit 1 (SIL-6)
- Function and Punchlist Tracking - Unit 1 (SIL-7)
- Test Analysis Report - Unit 1 (SIL-8)
- Function Analysis Report - Unit 1 (SIL-9)
- RTP Interface Report - Unit 1 (SIL-9A)
- Modification Review Report - Unit 1 (SIL-9B)

RTG Generated Testing Implementation Unit 1 (SIL-10)
RTG Closure Reports - Unit 1 (SIL-11)

- Review (10-20%) Division of Nuclear Engineering (DNE) documents to Restart Test Engineer (RTE) which list component/system functions and verify that the functions were listed on the functional review matrix (FRM).
- Determine if RTE has identified any additional component/system functions as a result of the reviews and ascertain the reason the functions were not listed by DNE. Verify that any additional functions identified during the review were listed on the Punchlist and determine if they were properly identified to DNE and if the item resulted in a Condition Adverse to Quality Report (CAQR).
- Discuss with RTE their background experience and verify qualifications, documented training, and required reading are in accordance with SIL-1.
- Review the FAR, including the punchlist report and FRM to verify that the above documents are in agreement as to number of identified retests/tests to be performed, the disposition of punchlist items, and the resolution of identified interface items. Additionally, the conclusions reached by the RTE should be evaluated and discussed with the RTE. The following points should be considered when performing the above review:
 - (a) If the function has never been tested: is testing planned; what type of function (i.e., control, indication, safety, etc.); will a special test be written or will the existing SI be modified? If a safety function is involved, was existing SI inadequate? Was CAQR issued?
 - (b) If function was last tested during preoperational testing, should it be included in an existing SI as a requirement or an enhancement, added to a preventive maintenance program or ISI program, etc.?
 - (c) Are TS, FSAR, and/or design criteria document changes necessary? What method has TVA used to identify/track these changes?
- Evaluate the supervisory and JTG review and approval of the system package
- Verify that the FRM reflects the functions listed in the applicable FSAR and TS section.

The inspector determined that for the CCS (system 72) the requirements of the Unit 1 restart test program were properly implemented. The inspector did question the following:

- DNE provided functions 72-052 and 72-053 required the local handswitches (1-HS-72-2B and 1-HS-72-39B) to open or close their respective valves from the local control station. These functions were not verified as part of the review process and the FAR indicated that DNE concurred with the RTE that the functions were neither normal nor safety functions for the system.

Subsequent discussion indicated that the ability of the local switches to open the valves in question was verified during preoperational testing. However, the closing function of these local switches was not verified. The licensee indicated that the JTG had advised the RTE that as long as the specified function did not require additional testing DNE agreed that the local switches did not provide any safety or normal control function and were installed for maintenance purposes only. Given that the switches are located in an area that would be inaccessible during an accident condition, the inspector agreed with the licensee position that this function would be outside the intended scope of the RTP.

- Functions 72-002 and 72-017 require that the CS pumps deliver 4750 GPM with a discharge head of 143 psid. For Unit No. 2, RTG had determined that during preoperational testing the vendor pump curves had not been properly validated and required in STI-65 that a three point flow test be performed. However, for Unit 1, the licensee indicated that the function would be validated by SI-37.1 & 37.2 for the 1A-A & 1B-B pumps respectively. These two SIs only require a single point verification of pump performance as required by ASME Section XI. The inspector questioned the acceptability of validating pump performance with a single point test when the licensee's preoperational test for pump performance was never satisfied.

The licensee was requested to justify the adequacy of a single point test to validate this function for the 1A-A & 1B-B pumps.

- Functions 72-003 and 72-018 specify CS heat exchangers 1A & 1B differential pressure (DP) as 10 psid. The inspector determined that the 10 psid specified was in error because the results of recent CS flow calculations for both Unit 1 and Unit 2, which were performed to verify the adequacy of the modified system to deliver the required 4750 GPM,

indicated that maximum heat exchanger DP could not exceed 6 psid at 4750 GPM.

- o The inspector requested that TVA determine if the FAR functions were incorrectly stated and if so determine if the FRM should be modified to accurately reflect a maximum of 6 psid. Additionally, the FAR should be changed to reflect the correct maximum heat exchanger DP.

Resolution of these issues is a restart item and was committed to by the licensee at the inspection exit held on July 8, 1988.

5. Comparison of Unit 1 RTP to the Unit 2 Completed Program

The purpose of this comparison was to determine the adequacy of the modified Unit 1 RTP as contrasted to the Unit 2 program that was accepted by the NRC and documented in NUREG-1232 Volume 2, Safety Evaluation Report of TVA Sequoyah Nuclear Performance Plan.

The RTP for Unit 1 was essentially the same as that for Unit 2 and the evaluation and conclusions discussed in the SER mentioned above are considered valid for both units. However, the Unit 1 program scope was reduced from that used for Unit 2 based on lessons learned, and as a result of modifications to other Unit 1 programs that were inputs to the RTP. These differences along with the inspector comments are provided below:

- o Once the design functions were established, the review of the impact of previous modifications was performed by the RTE utilizing SIL-9B to generate the modification review report. This was different from the Unit 2 program which utilized the DBVP output for the list of modifications which may effect the system.

The inspector identified a possible weakness with this approach specifically, the Unit 2 program had also used red line drawings to depict the as constructed system at the time the preoperational tests were performed. Combining the DBVP output (i.e., mods since time of licensing) with the red line drawings, the Unit 2 program evaluated the adequacy of post modification testing of all modifications subsequent to successful preoperational testing. In comparison, the Unit 1 program which did not include the red line drawing process created a gap involving the adequacy of post modification testing between the time the preoperational test was performed and the time of issuance of the operating licensing (OL).

The above mentioned problem only affected those functions where the licensee was taking credit for preoperational tests to validate adequate testing of the specified function.

Subsequent to the inspector's identification of this problem, the licensee determined that 274 modifications fell into the post preoperational testing and pre OL category. Of these, 190 modifications were reviewed as part of the modifications review for Unit 1 and 16 were Unit 2 only which left 68 modifications to be reviewed. Two of the 68 modifications were determined to have a potential impact on previously tested equipment and both of these modifications were determined to be adequately tested and had no impact on the function involved.

- o The Unit 2 program requirement to review the results of the post maintenance test survey was not included in the Unit 1 program. This decision was based on lessons learned from the Unit 2 program which indicated that approximately 6% of the MR reviewed indicated either a lack of adequate test documentation or a lack of adequate testing. Additionally, the post maintenance test survey was not conducted for Unit 1 as part of DBVP; therefore, the RTP could not use it as an input to their process.

The inspector was provided with a copy of memo (RIM S16 880624 890) dated June 24, 1988, from the RTP manager to the JTG which provided statistics from the Unit 2 effort and provided the basis for not including it in the Unit 1 program. Additionally, the inspector was informed that the additional testing controls, put in place at the station as a result of the Unit 2 maintenance program upgrade should also reduce the impact of possible inadequate post maintenance testing on the validity of previous functional test.

- o The Unit 2 requirement to review the impact of the piece parts review was also deleted from the Unit 1 program. The licensee provided the inspector a copy of a February 10, 1988 letter to the NRC (RIM L44 880210 800) which indicated that based on the Unit 2 program lessons learned the scope of the Unit 1 piece parts program would be reduced. The letter indicated that less than four hundredths of one percent (5 of 13,000) of the reviewed parts required change out.

The S. , based on the above statistic, did not identify a need to review the output of the piece part program for impact on functional test validation. Additionally as stated above the licensee feels that the improved maintenance program would ensure that any part replaced as a result of the piece parts review would be adequately tested.

As stated earlier based on the above minor program implementation changes, the evaluation and conclusion for the Unit 2 program as stated in the SER appear to generally bound the Unit 1 program.

With the exception of the three point pump test, the RTP appears to be adequately implemented.

- c. Quality Assurance (QA) audit or surveillance items which may be applicable to Unit 1

During this inspection the licensee was requested to provide the results of any quality assurance audits or surveillances which had been conducted on the CS System. Discussions with licensee personnel concerning this subject revealed that no audits or surveillances had been conducted specifically on the CS system. Audits are conducted to verify programmatic controls and surveillances are conducted on specific site activities. Further discussion indicated that, if deficiencies on the CS system had been identified by other programmatic audits or activity surveillances, the results would be available on the TROI computer tracking system which was reviewed by the inspector. As a result of the above, the licensee was requested to provide the results of any audits concerning operational readiness conducted prior to the restart of Unit 2. The licensee provided the audits and the inspector reviewed them for proper corrective action and closeout (as appropriate). The following audits were included in this review:

SQA-87-0020	Restart Test Program
SSA-87-0019	Control of Replacement Items
SQA-87-803	RTI-1.1 Master Test Sequence
SQA-88-804	Revised Procedural Change Review System and USQD Process
SQK-88-804	Correction of Deficiencies
SQA-86-007	Calibration of CSSC Instruments
WBA-87-0018	Generic Reviews of WBN Categories
SSA-88-807	Generic Reviews of WBN CAQRs for Impact on SQN

No violations or deviations were found in this area.

- d. Employee Concerns CATD items which are specifically applicable to the Unit 1 or Unit 2 CS systems

The inspector reviewed issues which had been presented to the New Employee Concerns program for significant items applicable to the containment spray system. The New Employee Concerns program was formed on February 1, 1986, in order to resolve problems identified after that date.

The licensee was requested to identify any ECP issues which could affect the CS system. One case, ECP-87-SQ-510-09, was identified as being relevant to containment spray. The report for this open case was to supersede ECP report ECP-86-SQ-253-01 and provide a revised response to NRC allegation RII-84-A-0187, which involved the adequacy and implementation of procedures concerning heat number validation on structural material. The original licensee investigation had

concluded that the concern in ECP-86-SQ-253-01 was unsubstantiated, but this conclusion was later invalidated and the case reopened. The reinvestigation reviewed the heat number validation program for the time period from 1977 through 1984 as well as the information from previous investigations.

The ECP-87-SQ-510-09 investigation identified 31 receipt inspection data cards for QA Level I structural steel on which material inspectors certified that the inspection was accomplished according to procedure when it had not been. In addition, ECP identified a number of violations of heat number validation procedures, including a seven year period of routinely verifying heat numbers by using an indexed listing which was unauthorized and contrary to procedures. Use of the indexing listing has been evaluated for impact on the quality of pressure boundary construction but not for structural construction.

The inspector reviewed a draft of the ECP report, and a TVA letter dated May 27, 1986, from the Site Quality Manager to the Site Director in response to the ECP report draft. This letter stated that the ECP report brought into question the adequacy of the material controls for the structural steel installed at Sequoyah during construction. The letter also outlined a proposed plan of action to assure the required traceability. The planned course of action had not been finalized at the time of the inspection.

All information obtained during the inspection regarding this Employee Concern will be forwarded to the Allegation Coordinator, NRC HQ Office of Special Projects, for inclusion into the overall resolution of heat code traceability issues.

The inspector also reviewed the listing of all safety significant employee concerns and open files, and identified approximately thirty additional cases involving general issues which could possibly affect containment spray. The licensee provided the inspector with a summary of the resolution or the status of each of these cases. With one exception, the cases identified by the inspector had either been unsubstantiated, were restricted to systems other than the Unit 1 CS, had not affected equipment operability, or were being addressed programmatically and adequately resolved.

ECP-88-SQ-658, concerning wall thickness on inaccessible tubing, appeared to have possible relevance to containment spray. This concern resulted from an allegation made at Bellefonte that Sequoyah may not have adequately handled a nonconformance identified at Bellefonte. The issue identified that Sequoyah had a problem with a failure to verify wall thickness on inaccessible tubing as identified on BLN NCR 4658. In response to the inspector's request to evaluate the concern for possible applicability to Unit 1 CS, the licensee determined that the issue was being addressed through PIR SQNCEB874 and that corrective action and closure will be a post restart item

for both units. The PIR does affect the Containment Spray System, in that tube steel was used in the supports for that system. The maximum decrease in tube steel wall thickness identified in this PIR is 6.2% less than required. CEB-C1 21.82 permits a 20% increase in allowable stress in determining restart status. Since the area of steel increases linearly with the wall thickness, and the section modulus could increase with the square of the wall thickness the worst case would be $(1-.062)^2=.88$, or a 12% reduction in section modulus. Therefore, under the maximum design stress conditions, the tube steel would qualify for restart.

e. Maintenance History and Trending

In response to violations identified in Inspection Reports 327,328/85-45 and 327,328/86-37, TVA implemented in 1986 a maintenance history and trending program intended to improve the timeliness and effectiveness of corrective actions for equipment failures and out-of-tolerance conditions. As described below, the maintenance history and trending consisted of an "Operability Lookback" at pre-1986 issues, and an ongoing program consisting of several computer data-bases. The inspector reviewed portions of the history and trending for components in the CS system to identify possible equipment operability issues and to assess the effectiveness of the licensee program.

Within this area of inspection, no violations or deviations were identified. Effective use of historical maintenance records to identify and resolve recurring problems had been made by the licensee on several occasions. In the future, as more information is added to the data bases for periods of plant operation, and the presently planned refinements have been fully incorporated, the tracking and trending program should prove more useful.

(1) Operability Lookback Review

To aid in identifying potential operability questions resulting from past undetected repetitive failures, the licensee "Operability Lookback" review was conducted to identify and evaluate equipment problems which occurred prior to the initiation of the tracking and trending program in 1986. The objectives of the lookback program included the identification of adverse conditions associated with equipment operability, the evaluation of these conditions for significance with respect to safety, documentation of the existence and effectiveness of corrective actions, and the proposal of additional or modified corrective actions. The Operability Lookback utilized data obtained from PROs from both Sequoyah units, and interviews with senior plant employees. The review process evaluated operability issues involving generic equipment groups, as well as problems with specific individual components.

The inspector reviewed each PRO evaluation from the CS system portion of the Operability Lookback. The Operability Lookback review had identified eleven significant component failures in containment spray. Of these, the following were classified by the licensee as isolated cases:

- The 1B-B containment spray pump failed to manually start because the breaker locking lever was not adequately lubricated. The PRO evaluation included a review of prior work requests (A529411, A38689, A232/41, A157697) and concluded that there was no evidence of repetitive failures for the same root cause. In response to this finding, a recommendation was made to revise maintenance instruction (MI-10.4) to require lubrication of breaker locking levers. The inspector confirmed that this procedure modification had been made.
- Containment Spray pump 2B-B failed a surveillance test because the timer was out of tolerance.
- Containment spray pump 2A-A was declared inoperable due to a failure of undervoltage relay BCTA-72-27. The PRO review noted that there had been a similar previous undervoltage relay failure on DG 1A, but the two failures were not considered to represent a trend.
- Check valve 72-525 failed SI-166.15 twice. Because the SI was run every 92 days and the valve only failed twice, the PRO concluded that the problem was not significant, and additional corrective action was not recommended.
- Check valves 72-506 and 507 failed a surveillance test because the valve internals had not been replaced after flushing the system.
- Flow transmitter 2-FT-72-13 failed upscale due to air being trapped in the sense lines. The PRO evaluation documented that another failure of this flow transmitter had occurred on 4/11/83 due to a failed power supply.
- Containment spray mini-flow valve would not open due to a Buchanan plug coming loose on the control room handswitch.

The following issue was also identified in the lookback review:

- Possible deterioration of the containment spray heat exchanger tubing could go unnoticed. The recommended corrective action was to perform eddy current testing of the heat exchanger during every outage. The PRO evaluation package, dated 1/27/87, states that this eddy current testing had been performed at least once since 1979. The

inspector verified through discussions with the licensee that the eddy current testing had been performed during subsequent outages.

The following CS issues had been incorporated into the Operability Lookback generic equipment concern packages for Arrow Hart and Limitorque:

- Flow Control valve 1-FCV-72-24 failed to open due to dirty breaker contacts (Arrow Hart).
- Failure of BCTD-072-2A (Arrow Hart).
- 1-FCV-72-20 failed to open due to loose bolts on the torque switch (Limitorque). The PRO review generic package for this issue included one additional example of a valve which failed to stroke due to loose bolts.

The generic PRO review package for Arrow Hart contactors included at least eight PROs addressing failures of these components. The generic review, dated 3/13/87, stated that problems with the contactors had occurred since early plant operation, and a corrective maintenance plan had been documented in LER 84014 R1. After implementation of this corrective maintenance plan, additional failures occurred due to dirty contacts and dirt in the lubrication. DNE and Electrical Maintenance then determined that the lubrication appeared to create problems which negated the benefits. Laboratory testing was performed which showed that unlubricated contacts performed reliably for five times the number of cycles expected during the 40-year life of the plant. As a result, the lubrication was removed, and MI-10.40 was revised to require inspection of the breaker contacts. The inspector discussed the current status of this issue with Electrical Maintenance personnel, and the problem appeared to have been resolved.

The inspector observed that once the Operability Lookback issues had been identified through a search of the PROs, the PRO reviews made effective use of available maintenance tracking and trending data records when evaluating the issues for repetitive or generic failures. However, the analyses appeared to focus primarily on failures of a particular component having the same root cause, and the review could therefore have failed to fully identify the significance of repetitive failures due to different causes. Frequent failures of a particular type of component for different reasons or due to failures of different subcomponents could indicate the need to increase the testing frequency for that component.

In addition, because the Operability Lookback was based primarily on a review of existing PROs, the potential existed

for repetitive failures apparent from WR records not to be identified. The inspector performed a cursory review of the WRs issued during the period of the lookback study, and noted possible patterns which were not picked up by the Operability Lookback. In particular, an unusually large number of WRs were observed for the flow transmitters and flow indication instrumentation, including calibration and other problems. This was not reflected in the Operability Lookback. However, previous trends, if they continue and are of significance, are expected to be identified by the new ongoing tracking and trending program.

This issue was identified as a deficiency and provided for licensee information.

Inspection Report 327,328/87-24 identified that the findings of the Operability Lookback program were being tracked to completion but were not being directly factored into the new tracking and trending program data base. The licensee had responded that the operability lookback issue summary packages would be made readily available for utilization by those groups reviewing the trending and tracking data for repetitive instrument deficiencies. The inspector noted that although the Operability Lookback equipment failure issues were not directly factored into the new program, those issues addressed in WRs were included in the maintenance history records and thus available for incorporation in future trending reviews.

In conjunction with the review of the new maintenance history and trending program, the inspector reviewed the history and trending records for subsequent failures of selected components flagged in the operability lookback review. The results are documented below.

(2) Maintenance History and Trending Program

ANS 3.2/ANSI N18.7, Section 4.1.4 requires that a program be established which detects trends in activities affecting plant safety which may not be apparent to the day-to-day observer. Procedural requirements for trending the required information obtained from WRs, special reports, and out-of-calibration reports are specified in SQM-58, "Maintenance History and Trending". The inspector reviewed Revision 6 of the procedure.

The Maintenance History and Trending program documented in SQM-58 was implemented to satisfy the ANSI requirements by providing comprehensive maintenance history records for major plant components, in a readily retrievable format useful for detecting failure trends. Maintenance history tracks three categories: 1) NPRDS reportable items, 2) Class 1E and 50.49

components, and 3) Other components (including CSSC) falling outside of the other two categories.

Maintenance history information is tracked in three data bases from which it can be trended:

- (a) NPRDS: Program Procedure 1601.02 documents that ANSI N18.7 history and trending requirements will be satisfied through use of the NPRDS program. The NPRDS program, managed by INPO, provides maintenance trending and reliability information which is both specific to Sequoyah and common to the industry. The components tracked by NPRDS are prescribed by the program description. Semi-annual trending reports are prepared based on this data base, each covering a twelve month period so that all significant data will be included. The trending analysis incorporates criteria for identifying both repetitive and generic component failures.
- (b) EQIS: Program Procedure 1601.02 documents that 10 CFR 50.49 tracking requirements will be satisfied through the use of the EQIS data base. EQIS is used to store NPRDS reportable activities, Class 1E and 50.49 failures, and certain other failures documented on WRs. The data entered into EQIS is primarily failure related. EQIS was implemented in January 1986 and contains information processed after that date.

The 1E and 50.49 components are trended annually with EQIS using the same criteria as for the NPRDS data. CSSC components which are non-NPRDS and non-1E/50.49 are also trended annually using EQIS. The capability exists to use EQIS to trend the NPRDS components but this is not routinely done.

EQIS trending is also required whenever a component failure results in a reactor trip, turbine trip, load reduction, or LER. Each time a work request is entered, the data base is reviewed for similar failures of that component and components with three or more failures are flagged for further review.

- (c) Maintenance History: The Maintenance History data base contains records of all maintenance activities requiring documentation for tracking, both failures and non-failures. This data base can be sorted and trended using the SEEK program to obtain comprehensive maintenance records for particular components, but the program is not designed for routine use for identifying repetitive failure trends.

The threshold criteria which indicate a repetitive or generic failure are specified in SQM-58 as: 1) Any component failing two or more times in 12 months, 2) Any component model number with failures on more than 3% of the components during 12 months, and 3) Any item of the same function made by the same manufacturer with 5% failures in 12 months. SQM-58 requires that when repetitive or generic component failure trends are identified, an evaluation be performed in accordance with the appropriate attachments to the procedure.

The inspector reviewed selected WRs to independently identify issues relevant to equipment operability, and to assess how effectively these issues were being evaluated and trended by the licensee. The licensee was requested to provide a list of all WRs issued on the Unit 1 CS system since 1985 (the approximate time period applicable to EQIS entries), including those WRs which were either active or completed but in the review process. From the list which was provided, the inspector reviewed those WRs listed below for generic issues which affected equipment operability and could indicate design problems, a need for increased preventive maintenance, or a need for an increase in surveillance testing frequency. For purposes of the inspector's review, an equipment failure was defined as any condition which could prevent the equipment from performing its intended function. This included out of calibration conditions. When it was not apparent from the WR whether or not the equipment was actually found out of tolerance during a calibration, the IM calibration cards were consulted. In some cases the available information was not sufficient to warrant a failure determination.

<u>WR #</u>	<u>Component</u>	<u>Description</u>
B278242		Active WR - Repair or replace throttling valve orifice (WR dated 12-4-88)
A560117	PMP-72-10	No failure. Oil leakage created a room hazard. Cause of the oil leak was a loose plug in the reservoir and a poor oil level sight glass design, which resulted in overfilling and leakage. Replacement of the sight glass with different design was recommended. (WR dated 8/2/85)
A528784	PMP-72-10	No failure. Oil leakage was being caused by a loose reservoir plug. (WR dated 3/21/85)

B117523	PMP-72-10	Failure, Active WR - During four performances of SI-37, the pump had come close to exceeding the acceptable ASME Section XI vibration range. The pump motor was aligned to the pump. The cause of the problem was attributed to normal operating conditions over a period of time causing a gradual misalignment. (WR dated 5/20/86)
B232275	PMP-72-10	Active WR - Change grease in coupling, verify no leaks per SQM 66 to clear CAQR SQP-880035. (WR dated 4/27/88)
B232274	PMP-72-27	Active WR - Change grease in coupling, verify no leaks per SQM 66 to clear CAQR SQP-880035. (WR dated 4/27/88)
A089626	MTR-72-10B	Failure, Surveillance had indicated inboard bearing was bad. Vibration analysis had indicated progressive worsening. The motor bearings were replaced and the motor was retested successfully. The problem was attributed to normal wear. (WR dated 1/15/85)
A548339	MTR-72-10B	No failure. Performed insulation check per MI-10.20. (WR dated 12/5/85)
A291498	MVOP-72-02	No failure on Unit 1. The Unit 1 operator was replaced after it was used to replace the operator in Unit 2. (WR dated 2/1/85)
B295983	MVOP-72-21B	No failure. Rebuild Limitorque operator and replace gear box grease per MI-11.2. (WR dated 2/23/88)
B234170	MVOP-72-22A	Rebuild Limitorque operator and replace gear box grease per MI-11.2. (9/22/87)
B295970	MVOP-72-39A	Rebuild Limitorque operator and replace gear box grease per MI-11.2. (1/4/88)

B234173	MVOP-72-20B	Rebuild and regrease of Limitorque. (WR dated 9/22/87)
B103544	MOVOP-72-34	Active WR - Valve stem broke during functional test (WR dated 1/6/86)
B234171	MVOP-72-23A	Rebuild and regrease of Limitorque. (WR dated 9/22/87)
B228011	MVOP-72-0039	No failure. Sampled grease and replaced plugs. (4/4/87)
B247230	MVOP-72-13	No failure. (10/27/87)
B784921	VLV-72-misc	Active WR - Specifies for listed CS system valves, visually inspect, stroke to ensure no binding, check for packing leakage or damage, repair as necessary. (WR dated 5/27/88)
B784807	VLV-72-512 VLV-52-513	Active WR - Removal of valves from system, performance of setpoint and leakage test, repair as necessary, and reinstallation. (WR dated 5/27/87)
B114204	VLV-72-503	Active WR - Valve binds when operated. (WR dated 3/14/86)
B233705	VLV-72-502S	No failure. Excessive force had been required to open and close the valve. When the valve stem was cleaned and lubricated, it functioned properly. (WR dated 4/18/87)
B233706	VLV-72-504	No failure. Excessive force had been required to open and close the valve. When the valve stem was cleaned and lubricated, it functioned properly. (WR dated 4/18/87)
B751301	FCV-72-40	Failure, Active WR - FCV-72-40 failed the maximum stroke time for SI-166.6 (PMT on MOVATS) with a stroke time of 11 seconds as compared to a limit of 10 seconds. (WR dated 5/10/88)

B119812	FCV-72-40	Failure. Valve FCV-72-40 failed the SI-166.6 stroke time test. The problem was corrected by resetting the open limit switches per MI-11.2B and the stroke time acceptance criterion was then met. (WR dated 4/14/86)
A529253	FCV-72-2	Failure. The valve had failed to open during the performance of an SI. The problem was corrected by cleaning the contacts at the starter. (WR dated 1/26/85)
B100508	FCV-72-34A	Failure. The valve stem coupling broke while attempting to handcrank the valve open during a functional test. The failure was attributed to stripped coupling bolts caused by excessive force. The bolts were replaced. (WR dated 1/14/86, Duplicate of WR 103544)
A116682	FCV-72-22	Replacement of Crydom relay 1A1-153, which had burned up. (WR dated 10/25/85)
B292544	FCV-72-13	Active WR - Rework tubing to resolve SMI-1-317-26 FCV-72-34 discrepancies. (WR dated 3/9/88)
B784949	FCV-72-misc	Active WR - For specified CS valves, WR specifies checking and cleaning and lubricating valve stem, stroking test position indication and smooth travel, inspection of packing condition, repair as necessary. (WR dated 5/27/88)
A524367	FE-72-34	Removal and reinstallation of insulation for ISI inspection. (WR dated 1/9/86)
A524366	FE-72-34	Removal and reinstallation of insulation for ISI inspection. (7/1/85)
B119627	FT-72-13B	During an outage, the flow indicator indicated flow when pump was off. The instrument (Rosemont) was found to be within the allowed bands

		during the recalibration. (WR dated 5/2/86)
B118633	FT-72-13B	Calibration Check for SI-37 B. (WR dated 4/4/86)
B218755	FT-72-13B	Failure. During an outage, the Flow transmitter was providing a 600 gpm signal to the indicator, when no flow was present in the system. The threads on the tee connection were defective, and the tee was replaced. (WR dated 1/3/87)
B132532	FT-32-13B	No Failure. Calibrated for SI, and as-found was within specifications. (WR dated 5/4/86)
A550904	FI-72-13	Instrument was calibrated for SI-32 Part B. (WR dated 10/22/85)
B237669	FI-72-34	Active WR - Flow Indicator showing approximately 1000 gpm flow with pumps off and valves closed. Calibrate or Repair as needed. (6-12-88)
B221760	FM-72-13A	WR stated that the instrument would not calibrate below 25% of normal span due to wrong input resistor. IM calibration showed instrument was found in tolerance. (WR dated 3/2/87)
B227289	RLY-72-34A RLY-72-13B	WR stated that time delay relays were not within acceptance criterion. (4/30/87)
A529411	BCTA-72-10	Failure. CS pump 1B-B did not operate because breaker locking lever did not fully close. Problem was resolved by repairing the lever and lubricating. The problem was caused by racking in the breaker too tightly. Note: This issue was included in the Operability Lookback. (WR dated 3/3/85)
B285372	BCTD-72-39	Replace breaker wires with broken strands. (2/3/88)

B299963	PdI-72-16 Fdi-72-33	No Failure, Active WR - The licensee vertical slice walkdown identified that these two containment spray pump B startup strainer differential pressure indicators were swapped. The instruments were to be removed and swapped. (WR dated 6/13/88)
B104134	FTG-72-misc	Tube fittings were leaking due to boron buildup. (WR dated 2/10/86)
B261795	Air Test Line	Active WR - Investigation, evaluation, and repair (if necessary) of an arc strike on containment spray pipe. (WR dated 5/28/88, was not planned yet at the time of the inspection.)

This WR review, in conjunction with the Operability Lookback information in the above paragraph did not indicate any current generic or repetitive instrument problems, other than possible repetitive problems with the flow instrumentation.

The EQIS records were reviewed for each component for which a failure was identified in a WR. No repetitive failure trends were identified. The inspector requested the licensee to provide the failure records on all Unit 1 and Unit 2 plant components with the same manufacturer and model numbers as selected components in the CS system. These component model numbers were determined by the licensee to only be found in containment spray. The results of the model number scan were as follows:

- CS Pumps: No failure entries
- CS Pump Recirculation Flow Control Valves (72-13, 72-34):
 - B100508: 1-FCV-72-34 stem coupling failure (11/14/85)
 - A119737: 2-FCV-72-34 improper operation due to dirt and lack of lubrication (11/29/84)
 - A242957: 1-FCV-72-34 did not close due to trip on thermal overload for unknown reasons (12/14/83)
- RWST to Spray Header Flow Control Valves (72-21, 72-22) and Containment Spray Header Isolation Valves (72-2, 72-39):
 - B208087: 2-FCV-72-21 had leak and boron buildup due to worn packing (10/31/86)

- A040290: 2-FCV-72-22 had boron acid residue due to worn packing (09/29/84)
- B219517: 2-FCV-72-22 had packing leak (01/15/87)
- Manual Isolation Valves (72-500, 502, 503, 504, 533, 534):
 - B114204: 1-VLV-072-503 worn internals (3/3/86)
 - B208153: 1-VLV-072-500 worn packing (10/13/86)
 - B233708: 1-VLV-72-500 would not operate due to lack of lubrication (4/23/87)
 - B223815: 2-VLV-72-502 packing leak (5/5/87)
 - B103103: 2-VLV-72-502 packing leak (1/25/86)
 - B115481: 2-VLV-72-504 packing leak and boron buildup (4/2/86)

The above information was not considered by the inspector to indicate any generic equipment problems.

To assess whether component failures were reliably being entered into the EQIS and NPRDS data bases, the Unit 1 EQIS data base was searched for records of those WRs considered by the inspector to constitute component failures. A number of the above WRs had either been filed prior to the inception of EQIS, or had not completed the review process at the time of the inspection. The remaining component failures identified from the WRs by the inspector were all properly identified in EQIS and reported to NPRDS when required. The licensee had also implemented an independent engineering review of the component failure designations for added assurance that all applicable data would be tracked and trended as required, and be properly classified.

The licensee identified to the inspector that CAQR CHS 88001 had been written to address the fact that a number of data entries had been accidentally deleted from the EQIS data base and actions were being taken to restore the information to the records. This loss of data did not affect the trending commitments with respect to data being trended through NPRDS, but potentially affected the trending of 50.49 components with EQIS. Licensee corrective action for this problem was ongoing at the time of the inspection, and appeared to be adequate.

- The inspector noted that PRO 1-86-076, dated 4/8/86, identified that 1-FM-72-13A and 1-FM-72-13B were found to

be out of tolerance. The EQIS data base did not contain a failure entry for the associated work request, B118633.

The inspector briefly reviewed with the licensee the evaluations of repetitive or generic failures identified in the annual EQIS reviews for 1E/50.49 components. No issues were identified as applicable to the CS system.

NPRDS failure reports identified only one potential generic or repetitive failure pertaining to the CS system. The report for the period from July 1985 through June 1986, prepared by the TVA Performance and Analysis Section, identified multiple problems involving Kerostat valves similar to FCV-72-34. The inspector reviewed the licensee evaluation of this issue, which concluded that the failures were due to unrelated causes (dirt in the system, broken stem coupling, worn bearings, scratched end seat). No further corrective action was recommended.

The inspector reviewed four additional review and evaluation packages for generic or repetitive equipment failures identified through the trending programs for systems other than containment spray. The packages which were reviewed addressed generic problems with Quincy Air Start Compressors, Foxboro transmitters, and Asco pressure switches. Each of these reviews were considered by the inspector to be thorough and comprehensive, and produced meaningful and significant results and recommendations.

The inspector noted that some of the reviews of repetitive failures (primarily those in the Operability Lookback review) appeared to focus primarily on root cause determination, and possibly deemphasized the importance of repetitive failures from different causes. Numerous repetitive failures resulting from different root causes could indicate a need for an increased testing frequency. Licensee plans included more fully implementing this type of review in the future as the data base is expanded.

The inspector noticed in the review of the EQIS records that no CS pump failures were logged, although failures of subcomponents were logged which could have affected the operability of the CS pumps. The licensee follows the NPRDS guidelines concerning which components are to be tracked and where the failures should be entered. Effective trending of failures of major components due to different root causes must be accomplished by trending the major component together with all applicable subcomponents.

- f. Potential Reportable Occurrences (PRO) which were applicable to Unit 1 or Unit 2 CS systems (from August 1985 to present)

The licensee was requested to provide the inspection team with copies of all PROs which were written on the CS system for both units from August of 1985 to the present. These reports were provided and were included in the Operation Experience Review. The PROs were reviewed to determine any trends which may have existed concerning proper equipment operation and reliability. Additionally, the review included an assessment of: the evaluation and corrective actions for all deficiencies, the root cause analysis determination and actions to prevent recurrence (where applicable), the reportability of deficiencies, the operability of equipment, and the generic applicability of reported deficiencies, where appropriate. The following is a listing of PROs and LERs included in this review:

<u>PROs</u>		<u>LERs</u>
1-86-076	1-87-117	2-88-001
1-86-125	1-87-177	1-87-050
1-86-216	1-87-256	1-87-069
1-86-301	1-87-257	66-028
1-87-027	1-87-396	87-010
1-86-361	1-88-137	
1-87-049	2-87-012	
1-87-053	2-87-016	
1-87-055	2-87-017	
1-87-110	2-87-018	
2-88-5	2-88-138	

During the review of PRO 1-86-125 several concerns were identified regarding the installation and testing of relief valves on the containment spray system. One concern resulted in a problem area requiring licensee management attention and corrective action as follows:

- During the review of PRO 1-86-125, documentation provided indicated that the suction relief valves (72-512 and 72-513) were not included in the sites inservice testing program. Further investigation of this concern with the licensee revealed the following facts regarding this concern:
 - a) 10 CFR 50.55a(g) requires inservice testing of pumps and valves in accordance with ASME Section XI to verify operational readiness.
 - b) ASME Section XI, IWV-3511 requires category C valves to be tested in accordance with Table IWV-3510-1 (at least on a five year interval).
 - c) The valves on both units were tested as required by ASME Section XI. Documentation was provided by the licensee (Reference work plans 6813-01 and 12309).

- d) The licensee's Section XI pump and valve program, Section 6.8 of the FSAR, does not require valves 72-512 and 72-513 to be tested in accordance with ASME Section XI.

This issue is addressed as violation 327,328/88-29-04.

- g. Condition Adverse to Quality Requests (CAQRs) which were applicable to Unit 1 or Unit 2 CS systems (from August 1985 to present)

The licensee was requested to provide the inspection team with copies of all CAQRs (Conditions Adverse to Quality Reports) which were written on the CS system for both units from August of 1985 to the present. These reports were provided and were included in the Operation Experience Review. The CAQRs were reviewed to determine any trends which may have existed concerning proper equipment operation and reliability. Additionally, the review included an assessment of: the evaluation and corrective actions for all deficiencies, the root cause analysis determination and actions to prevent recurrence (where applicable), the reportability of deficiencies, the operability of equipment, and the generic applicability of reported deficiencies, where appropriate. The following is a listing of CAQRs included in this review:

CAQR

SQP 87-0570	SQP 88-0344
SQP 87-0713	SQP 88-0287
SQP 87-0697	SQP 87-1543
SQP 88-0212	SQE 870R01003
SQP 87-1481	SQP 87-1697
SQP 87-0603	SQP 87-1559

No violations or deviations were found in this area.

- h. Preoperational Test Deficiency Resolution

The inspector reviewed Preoperational test W-6.1A, SIS-Integrated Flow Testing, as it related to the CS and Preoperational Test TVA-21B, Containment Spray System for the purposes of evaluating the TVA resolution of test deficiencies. The specific test along with the inspection findings are listed below:

- W-6.1A1 - Of the eight deficiencies listed in this test package only deficiency DN-5 related to the containment spray system. This deficiency involved suction pressure gage PI-72-33 being found defective during testing and required test gages to be installed to complete testing. Subsequent to the test the gages were recalibrated and reinstalled thereby, resolving the test deficiency.

- TVA-21B - This test comprised the majority of testing associated with the CS. The review of the test results for this test revealed that 10 test deficiencies were written during the course of this testing which was conducted in the January 1979 time frame. Of the deficiencies written, 8 involved equipment failure and after repairs or replacement, the equipment was successfully retested. However, deficiencies DN-9 and DN-10 involved the fact that both the 1A-A and 1B-B CSS pumps failed to meet the manufacturer's pump curve and exceeded the expected starting current. The starting current deficiency was evaluated by DNE and found to be acceptable. The pump curve deficiency, however, was never properly resolved. The licensee resolved this issue for Unit 2 startup by performing STI-65. STI-65 required that a three point flow test be performed so that the manufacture's curve could be validated. However, for Unit 1 the licensee indicated that SI 37.1 and 37.2 would be performed to verify proper pump performance.

The inspector's review of SI 37.1 and 37.2 indicated that only one flow data point was being verified at the required flow of 4750 GPM. This issue was previously discussed in section 8.a of this report.

1. Industry Nuclear Experience Review (NER) issues specifically applicable to the Unit 1 or Unit 2 CS systems (Note: this included SER, SOER, IEB, IEN, NSRS, NMRG, and NSRB items from August 1985 to present)

Other items concerning industry nuclear operating experience issues with the Containment Spray System were reviewed by the inspector (NERs, SERs and IEBs). The following items were included in this review:

- NER 88-0250
- NER 88-0196
- NER 87-0683
- NER 870464002
- NER 850314001
- SER 30-84
- IEN 84-39

The corrective actions for three of these items was determined to be weak or nonexistent: corrective action for NER 850314001 which concerned inadvertent actuations of the containment spray system at various other utilities did not indicate that a thorough review of instrumentation/controls, procedures and personnel training had been conducted. No documentation of corrective actions was provided by the licensee for SER 30-84 (Inadvertent Actuation or containment spray at another utility), nor IE Notice 84-39 (Inadvertent Isolations of Containment Spray Systems at other utilities).

This issue was identified as a deficiency and provided for licensee information.

j. Training of Licensed Operators and Auxiliary Unit Operators Which are Specifically Applicable to the Unit 1 or Unit 2 CS Systems

As part of this inspection the inspector requested and reviewed Power Operations Training Center (POTC) course outlines/lesson plans associated with the CSS. A review of these lesson plans to determine the detail and type of questions provided during this training for both AUO and licensee operators was performed and is discussed below:

- ° Course outline OPN 017.027 (PWR), Student III, Step 1B - Reactor Technology (SQN-WBN) which is taught to AUOs during the course of their training prior to being assigned to an operating plant. This course has an 8 hour duration and contained basic system function description.
- ° Course outline OPN 218.067, Student III, Step II - Containment Spray (system 72) which is taught to AUO during the last part of their training phase was reviewed. This course, listed as an 8 hour program, provided a basic system function description as well as providing system operating information.
- ° Course outline OPL271C024, SQN Operator Certification Training - Containment Systems, which is taught to licensed operators during a part of their qualification and requalification training, was also reviewed. This instruction also provided basic system description and operating instruction.

Based on a review of the above lesson plans the inspector provided the following observation:

- ° None of the above three lesson plans discussed the interlocks associated with this system. These interlocks involving are very important to proper system operation.

This observation was discussed with the POTC PWR training manager who indicated that a training letter discussing these interlocks would be issued within seven days followed by a revision to the above training plans during the next scheduled update. This is a commitment which the licensee agreed to at the exit meeting conducted on July 8, 1988.

In addition to the above, the inspector reviewed past training records to determine if students demonstrated any generic weakness on the CSS. Additionally, the inspector reviewed approximately 20 student feed back forms required by Procedure SQ-OTIL-14 in an attempt to identify any student suggestions to improve the training on the CSS. No problems associated with these reviews were identified.

9. Additional References (some referenced in text)

- Drawing CB-1, sheets 74 & 75, Containment Pipe Supports
- Drawing 47W611-72-1, Mechanical Logic Diagram, Containment Spray System
- Drawing 47B16-2, Piping Class Drawing
- Drawing 47B601-72, Mechanical Instrument Tabulation
- Drawing 47W437, Mechanical Containment Spray System Piping
- Drawing 47W812-1, Flow Diagram, Containment Spray System
- Drawings 47A30, 47A31, Pressure Indicators Tap Drawings
- Drawing 47A37, Temperature Connections
- Drawings 47B001 series, Auxiliary Piping Installation drawings
- Drawings 47A053 series, 2" or Smaller Field Run Piping
- Drawings 45N751-1, 2, 5, 6, MOV Electrical Configuration
- Drawing E-48840, Aloyco, 300 PSI, 12" MOV
- Drawing E-48836, Aloyco, 300 PSI, 8" MOV
- Drawing E-48848, Aloyco, 300 PSI, 12" MOV
- Drawing TV-D-9909MO-(2), Kerotest, Containment Spray Recirculation Flow MOV
- Drawing 15-476-2411, Limitorque Wiring Diagram
- SOI-72.1, Containment Spray Systems
- TVA Detailed Design Criteria, SQN-DC-V-27.5, Containment Spray System
- TVA Detailed Design Criteria, SQN-DC-V-3.0, Classification of Piping, Valves, and Vessels
- SI-34, Containment Spray System Valve Position Verification, Units 1 & 2
- SI-37.1, Containment Spray Pump 1A-A Test, Unit 1
- SI-37.2, Containment Spray Pump 1B-B Test, Unit 1
- SI-158.1, Containment Isolation Valve Leak Rate Test, Unit 1 & Unit 2

- SI-162.1, Snubber Visual Inspection (Hydraulic and Mechanical), Unit 1 & Unit 2
- SI-162.2, Snubber Functional Testing (Hydraulic and Mechanical), Unit 1 & Unit 2
- SI-166.39, Disassembly and Inspection of SIS/RHR/CS/UHI check valves during refueling outages, Units 1 & 2
- SI-186, Locked Valve Verification Per NRC Commitment, Containment Inspection, Units 0, 1, 2
- SI-267-72.1, Functional Pressure Test of Containment Spray System, Units 1 & 2
- SI-604, Essential Instrumentation Operability Verification
- Technical Specifications, Unit 1, Section 3/4.6.2, Depressurization and Cooling Systems
- ASME Boiler & Pressure Vessel Code, Sections III, VIII, IX, XI
- ASME Draft Code for Pumps and Valves for Nuclear Power, November 1968
- Hydraulic Institute, Section B, (Centrifugal Pumps)
- Tubular Exchanger Manufacturers Association, Class R Heat Exchanger, Tube Side, ASME Boiler & Pressure Vessel Code Section VIII
- National Electrical Manufacturers Association, NEMA - MG - 1 (Motors), 1963
- ANSI 16.5 Steel Pipe Flanges & Flanged Fittings
- ANSI B 31.1 Code for Pressure Piping with Inspection and Test Requirements to ANSI B 31.7 Code for Nuclear Piping in Lieu of Applicable Nuclear Code Cases
- SSDC 1.12, System Standard Design (SSDC), NSSS Layout Guidelines, Westinghouse Electric Corporation, dated March 1971
- SSDC 1.14, System Standard Design Criteria (SSDC), Nuclear Steam Supply System Containment Isolation, Revision 3, Westinghouse Electric Corporation, dated September 1981
- SSDC 1.15, Systems Standard Design Criteria (SSDC) NSSS and Related Systems Equipment Safety Classification, Revision 3, Westinghouse Electric Corporation, dated May 1978
- SSDC 1.3, System Standard Design Criteria (SSDC), Revision 2, Westinghouse Electric Corporation, dated April 15, 1974

SS1.3X, System Standard (SS) 1.3X Nuclear Steam Supply System Auxiliary Equipment Design Transients for all Standard Plants, Revision 0, Westinghouse Electric Corporation, dated September 1978

IEEE 279-1971, Standard Criteria for Protection Systems for Nuclear Power Generating Stations

IEEE Std. 379-1972/ANSI N 41.2, Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems

Containment Sump Minimum Level at Time of Switchover to Recirculation Mode and Allowable Margin for RWST Level Instrument Inaccuracy for a Large LOCA (SQN-OSG7-008). Unit 1 corollary

E-Specification 678765 - Motor Operated Valves for TVA Sequoyah Nuclear Plants Units 1 and 2, & G-676258 Motor Operated Valves, Westinghouse Electric Corporation

E-Specifications 67863 - Control Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and E-Specifications 676270 - Control Valves, Westinghouse Electric Corporation

E-Specifications 67869 - 2 Inches and Below Manual "T" and "Y" Globe and Self-Actuated Check Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and 678724 - 2 Inches and Below Manual "T" and "Y" Globe and Self-Actuated Check Valves, Westinghouse Electric Corporation

E-Specifications 678760 - Manual "T" and "Y" Globe, Manual Gate, and Self-Actuated Check Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and G-676241 - Manual "T" and "Y" Globe, Manual Gate, and Self-Actuated Check Valves, Westinghouse Electric Corporation

E-Specifications 67858 - Auxiliary Relief Valves for TVA Sequoyah Nuclear Plant Units 1 and 2, and G-676258 - Auxiliary Relief Valves, Westinghouse Electric Corporation

SQNP-47W812-1, Flow Diagram, Containment Spray System Powerhouse, Units 1 and 2

SQNP-47W610-72-1, Mechanical Control Diagram, Containment Spray System

SQNP-47W611-72-1, Mechanical Logic Diagram, Containment Spray System

SQNP-47A366-72-Series, Tabulation of Valve Marker Tags

SQNP-47W437-Series, Containment Spray System Piping

SQNP-47B601-72-Series, Mechanical Instrument Tabulation

SQNP-47B16-2, Piping System Classification

Westinghouse Drawing 110E338, Sequoyah Unit 1 - Safety Injection System, Flow Diagram

SQN-47W811-1, Flow Diagram, Safety Injection System Powerhouse, Units 1 and 2"

SQN-DC-V-10.1, Design Criteria Mechanical Unit Control Panels - January 11, 1971

SQN-DC-V-10.3, Design Criteria Mechanical Auxiliary Instrumentation (Room) Panels - July 14, 1971

SQN-DC-V-10.4, Design Criteria Mechanical Local Panels for Class I Equipment - January 10, 1972

SQN-DC-V-1.0, General Civil Design Criteria

SQN-DC-V-2.16, Single Failure Criteria for Fluid and Electrical Safety-Related Systems

SQN-DC-V-10.5, Separation of Instrument Sensing Lines and Instrument Air Lines

SQN-DC-V-11.2, 125-V DC Vital Battery System

SQN-DC-V-11.3, Power Control and Signal Cables For Use In Category I Structures

SQN-DC-V-11.6, 120-V AC Vital Instrument Power System

SQN-DC-V-11.4.1, Normal and Emergency Auxiliary Power Systems

SQN-DC-V-12.2, Separation of Electrical Equipment and Wiring

TVA Sequoyah Nuclear Plant Unit 2, Final Safety Analysis Report, Amendment 3, filed on 6/16/86

TVA-TR75-1A, Quality Assurance Program Description for Design Construction and Operation of TVA Nuclear Power Plants, Revision 8, October 1984

TVA Nuclear Quality Assurance Manual

SQN-DC-V-21.0, Design Criteria for Environmental Design

SQN-DC-V-27.1, Design Criteria for Ice Condenser System

SQN-DC-V-3.0, The Classification of Piping, Pumps, Valves and Vessels

SQN-DC-V-2.3, Containment Vessels

SQN-DC-V-27.6, Design Criteria for RHR System

SQN-DC-V-2.15, Containment Isolation

SQN-DC-V-7.5, Fire Protection Systems

SQN-DC-V-7.6, Proprietary Protective Signal Systems for Fire Alarm and Supervisory Service

SQN-DC-V-27.3, Design Criteria for Safety Injection System

SQEP-29, Procedure for Preparing Design Basis Document for Sequoyah Nuclear Plant

SQNP-DC-V-7.4, Essential Raw Cooling Water System

Civil Design Guide DG-C1.3.4 - Extreme Wind and Tornado Wind Forces on Structures

Response to High Containment Pressure, Functional Restoration Guideline FR-Z.1

Quality Assurance Plan Westinghouse Nuclear Energy Systems Divisions, WCAP-8370, Revision 7A, February 1975

Westinghouse Water Reactor Divisions Quality Assurance Plan, WCAP-8370, Revision 8A, September 1977

Westinghouse Water Reactor Division Quality Assurance Plan, WCAP-8370, Revision 9A, October 1979

Nuclear Fuel Division Quality Assurance Program Plan, WCAP-7800, Revision 5, December 1977

Westinghouse Water Reactor Divisions Quality Assurance Plan, WCAP-8370, Revision 9A, Amendment 1, February 1981

Westinghouse Water Reactor Divisions Quality Assurance Plan, WCAP-8370/7800, Revision 10A/6A, August 1984

CE-CPA-546 - System Functional Requirements for Systems Safety Injection System Actuation: a) SIS Actuation and Reactor Trip, b) Containment Spray Actuation, M. A. Mangan, R. M. Reymers, May 6, 1970, TVA-300/6

Nuclear Engineering Procedure NEP 9.1, Corrective Action

10 CFR Part 50.59, Changes, Tests and Experiments

Regulatory Guide 1.105, November 1975, Instrument Set Points for Safety Related Items

Regulatory Guide 1.29, Seismic Design Classification

Regulatory Guide 1.53, June 1973, Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

10 CFR 50.59, Equipment Qualification

10 CFR 50, Appendix A, 1970 Draft Version

10 CFR 50, Appendix J

10. Persons Contacted

Licensee Employees

- *S. A. White, Senior Vice President, Nuclear Power
- *J. T. Bynum, Assistant Manager, Nuclear Power - Operations
- *H. L. Abercrombie, Site Director
- *J. T. La Point, Deputy Site Director
- *S. Smith, Plant Manager
- *J. Patrick, Operations Group Manager
- R. J. Prince, Radiological Control Superintendent
- *M. J. Ray, Licensing Group Manager
- L. E. Martin, Site Quality Manager
- *P. G. Trudel, Project Engineer
- R. W. Olson, Modifications Branch Manager
- J. M. Anthony, Operations Group Supervisor
- R. V. Pierce, Mechanical Maintenance Supervisor
- M. A. Scarzinski, Electrical Maintenance Supervisor
- H. D. Elkins, Instrument Maintenance Group Manager
- R. S. Kaplan, Site Security Manager
- J. T. Crittenden, Public Safety Service Chief
- R. W. Fortenberry, Technical Support Supervisor
- J. H. Sullivan, Regulatory Engineering Supervisor
- J. L. Hamilton, Quality Engineering Manager
- *H. R. Rogers, Plant Operations Review Staff
- M. A. Cooper, Compliance Licensing Supervisor
- R. Mills, EQ Engineer
- Roger Field - Principle Engineer, CEB/CSG
- Nat Foster - Technical Supervisor, CEB/CSG
- Kreis Lester - Technical Supervisor, CEB/CSG
- Charlie Johnson - Lead Engineer, CEB/CSG
- Carl Barker - Technical Supervisor, CEB/CSG
- Mike Edward - Technical Supervisor, CEB/CSG
- Orhan Gurbuz - Consultant, Bechtel
- Chang Chen - Consultant, Gilbert Commonwealth
- *Raj Kundelkar - Assistant Lead Engineer, CEB/CSG
- Coleman Haskin - Engineer, CEB/CSG
- George East - Section Manager, SWEC
- Debbie Burch, Mechanical Engineer
- Calvin Burrell, Mechanical Engineer
- Stan Duke, Mechanical Engineer

Chris Fulwider, Principal Mechanical Engineer
 Roger Gisl, Mechanical Engineer
 Mike Hammond, Mechanical Engineer
 Roy Hoekstra, Principal Civil Engineer
 Ken House, Section Supervisor
 T. J. Means, Design Engineer Associate Mechanical
 *Ken Mogg, EMG Lead Engineer
 Bill Roberts, Principal Civil Engineer
 George B. Sanders, Project Engineer (G/C)
 *Mark Serhal, Nuclear Engineer
 Jim Southers, Design Engineer, Associate Mechanical
 *Ed Steinhauser, Lead Engineer
 J. M. Warren, Mechanical Engineer (G/C)
 Charles W. Whitehead, Project Engineer (G/C)
 *R. C. Williams, Electrical/I&C Team Leader
 M. Bowman, Electrical Engineer
 R. Hall, Principal Electrical Engineer
 A. Pal, Electrical Specialist
 J. Hutson, Assistant Chief Electrical Engineer
 J. Edwards, Electrical Group Leader
 S. Gallager, Electrical Engineer
 A. Raju, Electrical Engineer
 S. Jackson, Mechanical Engineer
 Joseph Drago, Engineering Specialist
 Robert Adkison, Civil Engineer
 Rick Daniels, Lead Mechanical Engineer
 Bob Bryan, Nuclear Engineering Staff Specialist (Knoxville through
 Mark Serhal)
 Randy Devault, Nuclear Engineer
 *Frank Denny, Engineering Assurance
 Aubrey Coleman, Mechanical Engineer

Other licensee employees contacted included technicians, operators, shift engineers, security force members, engineers and maintenance personnel.

NRC Representatives

*J. G. Partlow, Director, Office of Special Projects (OSP)
 *F. R. McCoy, Assistant Director for TVA Inspection Programs, TVA Projects
 Division, OSP
 *S. Black, Assistant Director for TVA Projects, TVA Projects Division, OSP
 *R. Pierson, Branch Chief, Plant Systems Branch, OSP
 *J. N. Donohew, Project Manager, Sequoyah Restart, OSP

*Attended exit interview

11. Exit Interview

The inspection scope and findings were summarized with the Plant Manager and members of his staff on July 8, 1988. Four violations described in this report's Summary paragraph were discussed. No deviations were

discussed. The licensee acknowledged the inspection findings. The licensee did not identify as proprietary any of the material reviewed by the inspectors during this inspection. During the reporting period, frequent discussions were held with the Site Director, Plant Manager and other managers concerning inspection findings.

12. Acronyms and Initialisms

ABGST	-	Auxiliary Building Gas Treatment System
ABSCE	-	Auxiliary Building Secondary Containment Enclosure
AFW	-	Auxiliary Feedwater
AI	-	Administrative Instruction
AOI	-	Abnormal Operating Instruction
AUO	-	Auxiliary Unit Operator
ASOS	-	Assistant Shift Operating Supervisor
BIT	-	Boron Injection Tank
C&A	-	Control and Auxiliary Building
CAQR	-	Conditions Adverse to Quality Report
CCP	-	Centrifugal Charging Pump
CCTS	-	Corporate Commitment Tracking System
COPS	-	Cold Overpressure Protection System
CSH	-	Containment Spray Header
CSSC	-	Critical Structures, Systems and Components
CVI	-	Containment Ventilation Isolation
DC	-	Direct Current
DCN	-	Design Change Notice
DNE	-	Division of Nuclear Engineering
ECCS	-	Emergency Core Cooling System
EDG	-	Emergency Diesel Generator
EI	-	Emergency Instructions
ELM	-	Electrical Loading Matrix
ENS	-	Emergency Notification System
ESF	-	Engineered Safety Feature
FCV	-	Flow Control Valve
FSAR	-	Final Safety Analysis Report
GDC	-	General Design Criteria
GL	-	Generic Letter
HIC	-	Hand-operated Indicating Controller
HO	-	Hold Order
HP	-	Health Physics
HX	-	Heat Exchanger
ICMS	-	Insulation Consultants and Management Services
IN	-	NRC Information Notice
IFI	-	Inspector Followup Item
IM	-	Instrument Maintenance
IMI	-	Instrument Maintenance Instruction
IR	-	Inspection Report
KP	-	Kilopound Thrust
KVA	-	Kilovolt-Amp
KW	-	Kilowatt

KV - Kilovolt
 LER - Licensee Event Report
 LCO - Limiting Condition for Operation
 LOCA - Loss of Coolant Accident
 MI - Maintenance Instruction
 NB - NRC Bulletin
 NOV - Notice of Violation
 NRC - Nuclear Regulatory Commission
 OSLA - Operations Section Letter - Administrative
 OSLT - Operations Section Letter - Training
 OSP - Office of Special Projects
 PMT - Post Modification Test
 PORC - Plant Operation Review Committee
 PORS - Plant Operation Review Staff
 PRO - Potentially Reportable Occurrence
 QA - Quality Assurance
 QC - Quality Control
 RCS - Reactor Coolant System
 RG - Regulatory Guide
 RM - Radiation Monitor
 RHR - Residual Heat Removal
 RWP - Radiation Work Permit
 RWST - Reactor Water Storage Tank
 SER - Safety Evaluation Report
 SG - Steam Generator
 SI - Surveillance Instruction
 SOI - System Operating Instructions
 SOS - Shift Operating Supervisor
 SQM - Sequoyah Standard Practice Maintenance
 SR - Surveillance Requirements
 SRO - Senior Reactor Operator
 SSQE - Safety System Quality Evaluation
 STI - Special Test Instruction
 SYSESR - System Evaluation Report
 TACF - Temporary Alteration Control Form
 TROI - Tracking Open Items
 TS - Technical Specifications
 TVA - Tennessee Valley Authority
 URI - Unresolved Item
 USQD - Unreviewed Safety Question Determination
 WCG - Work Control Group
 WP - Work Plan
 WR - Work Request

*** Print Diagnostics for: 36SD 3577

Total Formatting Exceptions = 2 Total Listed Below = 2

The Following Two Formats Will Be Used:

<u>Page/Line</u>	<u>Format Exception Message Found By The IBM 5520</u>
<u>Sheet Number</u>	<u>Format Exception Message Found By The Printer</u>
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74.0.0/32	0020-Line Is Too Long To Be Justified

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JUS 90 1
*PRO
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03 INPUT 10 400
04 INPUT 12 401
14 061PUT 87
05 OUTPUT 193
1. OUTPUT 224
28 JUS
29 ADJ
27 PLA
*END
#123