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SEQUOYAH NUCLEAR POWER PLANT

DESIGN CALCULATION REVIEW PROGRAM

INSPECTION REPORTS 50-327 AND 50-328

1 INTRODUCTION AND BACKGROUND

The Division of Nuclear Engineering (DNE) developed the design calculation review program because the findings of past audits and other reviews had shown that the design bases for the nuclear power plants operated by the Tennessee Valley Authority (TVA) were not adequately documented with supporting calculations, or that such calculations were no longer retrievable. This program augmented the Sequoyah Nuclear Power Plant (SQN) design baseline and verification program (DBVP) by including a technical adequacy review of supporting calculations, a feature not included in the DBVP.

The design calculation review plan was initially described in an enclosure to a TVA letter from R. L. Gridley of January 20, 1987, (Reference 1) and Revision 1 to Section III.4 of Sequoyah's Nuclear Performance Plan (Reference 2). The design calculation review plan was subsequently updated in an enclosure to the TVA letters from R. L. Gridley of July 31, 1987, and August 21, 1987 (References 3, 4). The design calculation review plan addressed the essential calculations required to support the SQN design basis in the four technical branches of DNE.

Before the restart of Sequoyah Unit 2, the U. S. Nuclear Regulatory Commission (NRC) staff conducted three inspections of the design calculation review program and documented the results of these inspections in Inspection Reports (IRs) 50-327, 50-328/87-06; 50-327, 328/87-27; and 50-327, 328/87-64 (References 5-7). In addition, the NRC staff conducted an Integrated Design Inspection (IDI) of the Essential Raw Cooling Water (ERCW) system. The results of these inspections are documented in IRs 50-327, 328/87-48; 50-327, 328/87-74; and 50-327, 328/88-13 (References 8-10). The NRC staff conducted a final pre-restart inspection to address the open restart items in the civil calculation program and documented the results in IR 50-327, 328/88-12 (Reference 11). Although this inspection resolved the open restart items for Sequoyah, several unresolved items still remained open in the area of civil calculations. TVA submitted an initial set of responses on each of the unresolved items (Reference 12-22).

2 INSPECTION SCOPE AND OBJECTIVES

The NRC staff performed this inspection to review TVA's responses and corrective actions on the unresolved items involvi mechanical components and piping that were identified in IR 50-327, 328/88-12. Seven of the eleven unresolved items identified in this IR involved mechanical components and piping and piping supports. The seven unresolved items (URI's) reviewed during this inspection were 88-12-01, 88-12-02, 88-12-03, 88-12-06, 88-12-08, 88-12-10 and 88-12-11.

3 SUMMARY

During the inspection, only one item, URI 88-12-01, out of the seven items reviewed was closed. Five of the remaining six items require additional TVA actions. The requested TVA actions on these items are described in Appendix A f this report. The remaining item, URI 88-12-06, is open pending the development of a staff position on the acceptance criteria for the feedwater water hammer analysis. In addition to the seven unresolved items reviewed during this inspection, the NRC staff held a meeting with TVA to discuss the status of the remaining four items in the civil area. These four civil items have been the subject of ongoing discussions between TVA and the NRC staff. On the basis of additional staff review one of these items, URI 88-12-07, was closed. The status of the remaining items is also presented in Appendix A of this report.

4 DETAILED INSPECTION FINDINGS

The detailed inspection findings for each unresolved item are discussed in Appendix A of this report. No new topics were reviewed during this inspection.

5 MEETING SUMMARIES AND REFERENCES

A summary of attendees of the entrance and exit meetings is provided in Appendix B. A list of references is provided in Appendix C.

APPENDIX A

LICENSEE ACTION ON PREVIOUS UNRESOLVED ITEMS

(Closed) Unresolved Item URI 88-12-01, Thermal Monitoring of Supports

In Inspection Report 50-327, 328/88-12, the NRC staff identified that, as part of the pipe support calculation regeneration effort, TVA developed a set of restart criteria, CEB-CI 21.89 (Reference 23). The staff accepted these criteria subject to restrictions (Reference 24). TVA revised these restart criteria based on additional discussions with the NRC staff (Reference 25). One of the revisions to the restart criteria allowed TVA to monitor snubber swing angles during plant heatup to verify that thermal binding would not occur. These measurements were to be used instead of using the calculated piping thermal movement for computing the angular swing for comparison with allowable tolerances. TVA identified 13 supports to be monitored during the heatup (TVA memorandum from Hosmer to Abercrombie of December 18, 1987, RIMS No. B25 871218 020). In its review, the staff identified that four supports, 2-H63-2, 2-H63-3, 2-H63-4 and 2-H63-5, were being monitored by strain gage to obtain thermal loads. These strain gage measurements were not part of the agreed criteria (Reference 25). TVA responded (Reference 26) that the four supports met the allowable stress criteria in CEB-CI-21.89. TVA also proposed using the measured loads to qualify the supports to the long term design criteria SQN-DC-V-24.2. The staff indicated that it would not accept this long-term solution unless TVA would commit to reanalyze the entire piping analysis problem to determine a new load distribution of all supports. This item remained open pending TVA's response specifying the method that would be used to qualify the four supports to long-term criteria.

TVA responded to the open item (Reference 12) by stating that it had initiated a reanalysis of the affected piping problems for both units, and that it would modify the supports as necessary to meet long-term design criteria specified in SQN-DC-V-24.2 by the end of the cycle 4 outage for each unit.

To confirm TVA's actions to address URI 88-12-01, the team performed a programmatic review of the following TVA piping analysis and pipe support calculations for SQN Unit 2:

1. TVA Calculation No. 0600154-03-01, "Summary of Analysis 0600154-03-01," Revision No. 2, dated January 23, 1990 (RIMS No. B87 900207 012).

2. TVA Calculation No. MCLC09/2H630002/2RHR0002, "System 74/Calculations for Pipe Support 2-H63-2," Revision No. 3, dated October 11, 1989 (RIMS No. B87 891011 011).

3. TVA Calculation No. 2RHRH0003/2H630003, "Residual Heat Removal Calculations for Pipe Support 2-RHRH-3," Revision No. 1, dated January 5, 1989 (RIMS No. B25 890112 800).

4. TVA Calculation No. MCLC09/2H630004/2RHRH0004, "System 74/Calculations for Pipe Support 2-H63-4," Revision No. 3, dated October 11, 1989 (RIMS No. B87 891208 018).

5. TVA Calculation No. MCLC09/2H630005/2RHRH0005, "System 74/Calculations for Pipe Support 2-H63-5," Revision No. 2, dated October 11, 1989 (RIMS No. B87 891011 009).

The portion of the Residual Heat Removal System which the TVA piping analysis (Item No. 1) addresses is shown on TVA Drawing No. 45-M-47K432-57," Reactor Building/Unit 2/Problem: 0600154-03-01/14" RHR from RCL Hot Leg #4 to Pen. X-107; 2" & 6" SI to RC HL & 4" RV Disch., Revision No. 1, dated April 28, 1987.

TVA subsequently prepared design change notice (DCN) M01057A, dated November 9, 1989 (RIMS No. B25 891109 001) to require, in part, that pipe support 2-H63-2 be braced, and that pipe support 2-H63-3 be physically removed. TVA has indicated that the DCN will be implemented during the Cycle 4 outage for SQN Unit 2, which is currently scheduled to begin on October 1, 1990.

Based on the team's review of the referenced TVA documents provided by TVA during the week of April 9, 1990, the team accepted TVA's actions to address URI 88-12-01. Therefore, URI 88-12-01 is closed.

(Open) Unresolved Item URI 88-12-02, Allowable Loads for Standard Component Supports

In IR 50-327, 328/88-12, the NRC staff discussed the results of a review of TVA's development of allowable loads for U-bolts. TVA's design criteria for piping supports, SQN-DC-VC-24.2, Figure I-7 (Reference 24) contained a table of load ratings for U-bolts. Based on a review of the pipe support criteria in September 1987, the staff questioned the basis for the allowable loads used for U-bolts. TVA provided the basis for the U-bolt allowable loads as a Brown Ferry test report, CEB-85-06. According to CEB-85-06 the load ratings were developed based on the winter addenda to the 1983 Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME). The staff review of CEB-85-06 questioned whether the allowable loads have been appropriately derived using the ASME Code criteria. TVA presented an additional basis for the allowable U-bolt loads. The additional basis included a comparison of the allowable loads with a load rating procedure using the factor of safety quoted in industry standard MSS SP-58. This standard is referenced in the piping code USAS B31.1 - 1967. The load rating calculations also included a check of deflection criteria in SQN-DC-V-24.2. The staff did not agree with the appropriateness of the deflection criteria used for the lateral load test. TVA had used an average value from tests of cinched and uncinched U-bolts. Based on further staff questions, TVA responded that they did not have cinched U-bolts in field installations. TVA typical drawing 17 W586-3, Revision 23 showed a gapped U-bolt configuration. A TVA check at the field during the inspection also identified that a gapped U-bolt configuration was typical. TVA then demonstrated that the U-bolts could meet a reduced allowable load based on test data using the uncinched U-bolt tests. The staff considered this action by TVA to be acceptable for restart.

In addition to the concern with U-bolts, the NRC staff had identified the development of allowable loads for standard component supports as an open issue requiring further review (Reference 24). Inspection Report 50-327, 328/88-12, stated that the staff had accepted these allowable limits for restart (Reference 27) but the staff still had an open issue concerning TVA's demonstration that these allowable limits meet the Sequoyah FSAR requirements.

TVA's response to the open items (Reference 13) provided the results of its additional evaluations of the standard allowable loads for component supports. TVA's response addressed the issue of standard allowable loads for component supports, in four separate categories: U-bolts, Unistrut clamps, snubbers, and other standard support components.

To address concerns about U-bolt and Unistrut clamp allowable loads, TVA has issued the following design standards.

1. TVA civil design standard DS-C1.6.13, "Design of U-Bolt Clamps for Piping and Tubing - BFN," Revision No. 0, dated March 20, 1989. A design standard change notice that TVA issued on April 5, 1989 extends the applicability of this design standard to SQN.

2. TVA civil design standard DS-C1.6.14, "Design of Unistrut and B-Line Clamps for Piping and Tubing," Revision No. 0, dated May 23, 1989.

TVA is also conducting separate studies to confirm that the support configurations installed in SQN that use U-bolts or Unistrut clamps meet the allowable loads tabulated in the referenced design standards. TVA has indicated that these reviews will be completed on or about July 1, 1990. This portion of URI 88-12-02 remains open pending TVA's submittal of the results of these completed studies.

TVA's load ratings for snubbers are tabulated in Figure I-2 of TVA design criterion No. SQN-DC-V-24.2, "Supports for Rigorously and Alternately Analyzed Category I Piping," Revision No. 3, dated November 23, 1988. Figure I-2 tabulates separate upset and faulted load capacities for the pre-NF(not manufactured to the requirements of Subsection NF of the ASME Code) hydraulic snubbers, pre-NF mechanical snubbers, and post-NF mechanical snubbers installed in SQN. The NRC team did not indicate a concern with TVA's use of the faulted load capacity tabulated in the Figure for pre-NF hydraulic snubbers. However, the team has not accepted TVA's use of the faulted load capacity for pre-NF mechanical snubbers, which is tabulated in Figure I-2. The team asked TVA to provide justification for the use of the faulted load capacity that TVA specifies in the referenced design criteria for the pre-NF mechanical snubbers installed in SQN. The staff had expressed its concern with the allowable limits for these snubbers proposed by TVA in September 1987 (Reference 24). The staff addressed the same issue during the review of the design criteria at Browns Ferry Nuclear Power Plant. For Browns Ferry, TVA proposed an allowable limit for the pre-NF snubbers that the staff considered acceptable. This issue is discussed in IR 50-260/89-15 (Reference 28). During this inspection, the team stated that TVA's use of the same allowable limits that were used for Browns Ferry would also be acceptable for Sequoyah. The team also stated that if TVA wanted to justify higher allowable limits for Sequoyah, then TVA should also demonstrate that sufficient margins exist in the pre-NF snubber allowable limits to accommodate the loads from the site-specific earthquake discussed in Section 2.5 of NUREG-0011, Supplement 1 (Reference 29). This portion of URI 88-12-02 remains open pending TVA's submittal of this justification to the staff. The team asked

TVA to confirm that the faulted load capacities specified for the post-NF snubbers nstalled in SQN have been assigned in accordance with the snubber manufacturer's load capacity data sheets (LCDs). This portion of URI 88-12-02 also remains open pending TVA's submittal of the result of this study.

TVA has also demonstrated that other standard support components installed in SQN, such as rods, struts, hangers and clamps, exhibit maximum faulted stress levels less than the maximum permissible nine-tenths yield stress of the component material at operating temperature.

To verify TVA's conclusion, the team programmatically reviewed the following TVA calculations:

1. TVA Calculation No. CD-Q0999-890866," Capacity of Pipe Support Standard Components/Linear Faulted Condition for TVA-BFN," Revision No. 0, dated March 13, 1989 (RIMS No. B22 890329 212).

2. TVA Calculation No. SCG1M0700/CAQR SQP 890351, "Qualify CSS Calculation for CAQR SQP 890351; SCG1M0700," Revision No. 0, dated August 8, 1989 (RIMS No. B87 890808 006).

3. TVA Calculation No. SCG1M0719,SCG1M01719, "Operability Review Check of Pipe Support Components," Revision No. 0, dated August 10, 1989 (RIMS No. B87 890815 004).

As a result of these calculations, TVA has derated the faulted load capacities of some standard components, such as clamps, and has required that the faulted load capacities of other standard components, such as saddles without center stiffeners, be established by analysis. The referenced calculations demonstrate that the maximum faulted stresses for the derated standard components installed in SQN are less than the allowable faulted stresses for the loads of record. On August 10, 1989, TVA issued a design input memorandum (DIM) to design criteria No. SQN-DC-V-24.2 to specify revised allowable faulted load ratings for the standard component supports installed in SQN. Based on the team's review of the referenced TVA documents, this portion of URI 88-12-02 is closed.

(Open) Unresolved Item URI 88-12-03, DBA ZPA Effects

In IR 50-327, 328/88-12 the NRC staff stated that its review of Employee Concerns Element Report 221.2(B) identified that TVA had not followed the recommendations in civil engineering report CEB 80-58 for evaluating the zero period acceleration (ZPA) effects for the containment load caused by a design basis accident (DBA). In response to this concern, TVA contracted Bechtel North America Power Corporation (Bechtel) to evaluate a sample of five piping systems attached to the steel containment vessel. The evaluation had only looked at the containment penetrations for this load case. This evaluation demonstrated that the penetrations were adequate for the increased loads caused by the DBA ZPA effects. The staff requested the results of the rest of the piping analyses including the supports. TVA stated that because of the low level of deflection caused by the ZPA loads, the supports would not be loaded because of the support construction gaps. The staff disagreed with TVA's reasoning on this issue. TVA used two contradictory sets of assumptions for the analysis. In determining that the piping had a rigid response, TVA assumed the supports were active. Then TVA assumed the supports were not active for the loads generated assuming the piping response was rigid. In response to the staff concern, TVA completed an evaluation f the supports where loads increased by more than 10 percent on the five sample piping analyses (Reference 30). The results of this evaluation demonstrated that the supports met either the interim or long term criteria. Inspection Report 50-327, 328/88-12, stated that TVA should complete the evaluation of the remaining piping systems attached to the containment to demonstrate that these systems meet the FSAR allowable limits specified in the Final Safety Analysis Report (FSAR).

Before this inspection, TVA had responded to the open item (Reference 14) by affirming the adequacy of the Bechtel study that was previously performed on five piping analysis problems. During the inspection, TVA noted that it was developing DBA loading based on the "Leak Before Break" (LBB) criteria for Sequoyah. However, the NRC staff had not yet reviewed the new DBA analysis based on LBB criteria for application to Sequoyah design. Therefore, the team did not include consideration of these criteria during the inspection.

During the inspection, TVA identified two cases of modification that may result from inclusion of the DBA ZPA load. One case involves three pre-start modifications identified for a hydrogen collection (HC) system pipe located in Unit 2. TVA is determining whether these modifications were actually required, as none are required for the comparable Unit 1 piping. Nevertheless, because these modifications had been identified before the restart and were determined to be caused by DBA ZPA loading, it is not clear why they were not considered in the Bechtel study, Bechtel North America Power Corporation report, "Evaluation Review of DBA ZPA Effects for Design Basis Accident Zero Period Acceleration for Sequoyah Nuclear Plant Unit No. 2," dated February 25, 1988 (RIMS B41 88 0302 300). The Bechtel study had concluded that no modifications would be required to accommodate this load. The second case involves a Condition Adverse to Quality Report (CAQR) that identifies four problems in the piping of the containment spray (CS) system that may require modifications because of the inclusion of the DBA ZPA load. The HC and CS system piping includes relatively large bore pipes that are located at higher containment elevations where the DBA ZPA load tends to be the greatest. The Bechtel study sample included neither large bore pipe nor, apparently, pipe located at higher containment elevations. This exclusion suggests that the Bechtel sample study may not adequately represent all piping systems attached to the containment. During this inspection, the NRC team learned that TVA has reanalyzed 18 problems related to the piping attached to the steel containment vessel after the restart. Based on these analyses, TVA has performed 15 modifications to installed hardware. From its consideration of the reanalyses and modifications, the NRC team attempted to determine whether the design modifications resulted from the DBA ZPA loads. However, the team could not determine whether the modifications resulted from the inclusion of the increased DBA ZPA load or from other physical changes (such as valve change) that had also been incorporated in the reanalyses. Also, the available piping design margins do not indicate that a significant increase in piping loads that could result from inclusion of the DBA ZPA load would exceed design criteria. Additional discussion of the evaluation of piping attached to the steel containment vessel is included in the summary of URI 88-12-10.

The NRC team identified concerns relating to samples used in the pre-restart study of DBA ZPA load effects on piping qualification; and concerns relating to the small design margins, discussed in URI 88-12-10, that are available in current piping qualification analyses. Based on these concerns the team concluded that this item is open pending TVA's submittal of the results of a more rigorous evaluation of the effects of the DBA ZPA.

(Open) Unresolved Item URI 88-12-04, Containment DBA Spectra

In IR 50-327, 328/88-12, the NRC staff stated that, based on a safety evaluation performed on the use of ASME Code Case N-411 damping (Reference 31) the staff concluded that the use of the code case damping for evaluation of the piping systems that are attached to the SCV under the load caused by the containment vibratory motions associated with a DBA was acceptable provided the DBA response spectra at various locations on the SCV have been generated by conservative analysis techniques. TVA's generation of the DBA response spectra for the SCV was documented in Report No. CEB-86-20-C, RO, entitled, "Sequoyah Nuclear Plant Design Basis Accident Non-Axisymmetric Pressure Loading Dynamic & Static Analysis of the Steel Containment Vessel and Response Spectra for Attached Equipment." In the inspection report, the staff concluded that the existing DBA response spectra were acceptable for restart and requested TVA to take the following two post-restart actions:

(1) Verify the adequacy of the double differentiation technique used in the computer code SUPRPOS by comparing the acceleration response spectrum generated directly from the acceleration time history (the results of SUPERSHELL computer code analysis) at the 0-degree azimuth nodes with the corresponding response spectrum generated from the SUPERSHELL displacement time history by the computer code SUPRPOS with the double differentiation technique, and

(2) Verify that the existing DBA response spectra did not miss the real maximum response because the analysis was cut off at the end of 0.9 second.

TVA committed to perform these post-restart actions in a letter dated March 2, 1988 (Reference 26). In a letter dated November 9, 1989 (Reference 15), TVA responded to these two action items. The staff review of this letter identified four concerns: (1) The need to consider the equivalent orthotropic elasticity properties of the SCV in the horizontal direction as they were considered in the vertical direction; (2) The lack of clear theoretical bases for the double differentiation technique used in the computer code SUPRPOS; (3) The need to check for a possible spectral peak (DBA spectra) in the high-frequency range (higher than 10 Hz); and (4) The need to provide justification to show that the maximum response of the SCV will not occur beyond the cut-off time duration 0.9 second while the forcing function was carried to 2 to 3 seconds. These concerns were discussed with TVA during a conference call on February 20, 1990.

In a telephone conversation on March 15, 1990, TVA provided the following schedule for the responses to the concerns mentioned previously:

(1) SCV orthotropic elasticity properties	March 26, 1990
(2) Theoretical basis for the double differentiation technique	March 23, 1990

(3) Existence of spectral peak beyond 10 Hz March 23, 1990

(4) Maximum SCV response beyond 0.9 second April 30, 1990

At a meeting during this inspection, TVA rescheduled its responses to the four concerns until the week of May 1, 1990. TVA has subsequently rescheduled its responses to June 8, 1990. This item remains open pending TVA's submittal of the additional responses discussed previously.

(Open) Unresolved Item URI 88-12-05, ERCW Pumphouse

In IR 50-327, 328/88-12, the NRC staff stated that during the review of the ERCW pumping station core samples, it discovered that one of the samples indicated that approximately seven feet of void existed between the two northern-most intake lines. To justify the strength of the concrete in the ERCW access cells, TVA submitted a report, "Rock and Concrete Investigation Report" dated January 1978. This report indicated that a total of eight holes were drilled. Two holes were core-drilled and six were percussion-drilled. Between the elevations of 620 and 630 feet (the bottom of the cells), six of the samples indicated cavities. TVA took sonic cross hole measurements on the two northern and the two southern holes. These measurements indicated that the cavities were not continuous at these two locations. However, these holes were not in the area of concern. TVA performed an analysis assuming areas of sound concrete, soft concrete, and voids or gravel pockets. The results of the analysis indicated that the stress levels were acceptable when subjected to loads experienced during a safe shutdown earthquake (SSE). This analysis gave reasonable assurance that the ERCW pumping station would not fail or be subject to excessive deflections when subjected to the postulated SSE event. This evaluation was considered acceptable for restart.

In IR 50-327, 328/88-12, the NRC staff stated that TVA agreed to perform the following additional post-restart evaluations of the ERCW pumping station concrete (Reference 32):

1. TVA will submit an evaluation program to the staff for review and approval. This program will include special emphasis on determining the number and size of the cavities or gravel pockets.

2. Once the as-built condition is determined, TVA will perform the following:

    a. Review the seismic qualification of ERCW equipment,

    b. Re-evaluate the effect of ERCW pumping station deflections on ERCW piping,

    c. Confirm that the design requirements are satisfied for an operating basis earthquake (OBE) that occurs when the water level is at an elevation of 704 feet.

For the first concern in the letter dated October 20, 1988 (Reference 16), TVA proposed to perform a limited exploration program for the pumphouse foundation by taking more core samples adjacent to Hole No. 39 (the original exploratory hole number) at Cell Element D from which the seven feet void was identified. The purpose of this program was to show that the actual founding materials exceed the description contained in the original exploration program report and o confirm that the assumptions made in the pre-restart analysis were reasonable nd conservative. In order to confirm that there are no cavities and loose materials (gravel, sand, or other material) in other cell foundation elements, TVA planned to take two additional core samples in Cell Element B. This program was discussed in detail during the meeting held on November 28, 1988 (Reference 33), and was revised by TVA through its letter dated July 10, 1989 (Reference 34). Based on its review of Reference 34, the staff found that the proposed exploration program is acceptable (Reference 35).

In a letter dated March 1, 1990 (Reference 36), TVA proposed to revise its schedule for the completion of the pumphouse foundation drilling program to September 1, 1990. The NRC staff finds the extension of the program to be acceptable. In addition, in a letter dated April 13, 1990 (Reference 37), TVA revised the locations of the holes to be drilled in Cell Element D and proposed for in Element B that if sound concrete is found from the first drilling, then the second hole will not be drilled. The staff finds that the revised locations of the holes in Cell Element D are acceptable. However, upon completion of the first drilling in Cell Element B, TVA should inform the staff of the soundness of the concrete prior to discontinuing drilling of the second hole.

From the examination of the core samples drilled in Cell Element D and the review of the video tapes taken after the core drills were completed during the site visit conducted on February 15, 1990, the staff found that TVA had demonstrated that there is no cavity or void in Cell Element D. However, TVA identified loose gravels in these samples. Therefore, the staff concludes that the actual founding materials are better than those described in the original exploration program report, and the analysis results performed for restart are acceptable for long-term plant operation if the results of the new exploration program shows no cavity (or void) or large volume of loose gravel in other cell elements.

To answer the second concern, TVA should complete its seismic evaluation of ERCW equipment and piping and confirm that the design requirements of the OBE with a water level at elevation 704 feet are satisfied.

This URI remains open pending TVA's submittal of the results of the proposed exploration program and the results of the evaluation of the second concern.

(Open) Unresolved Item URI 88-12-06, Feedwater Waterhammer

In IR 50-327, 328/88-12, the NRC staff stated that during the design calculation review, the team discovered that TVA had performed an analysis of a waterhammer caused by a feedwater check valve closure event, but had not formally issued the report. The original issue was identified as observation MEB-3 in IR 50-327, 328/87-06. TVA had originally performed an analysis of the piping and the check valves to demonstrate that they could withstand the pressure associated with the check valve closure event, but TVA had not evaluated the piping system for the flow transient event. Although TVA contended that its original evaluation met its licensing commitments and was adequate, TVA performed an additional analysis of the feedwater piping using forcing functions developed for the waterhammer flow transient (Reference 38). The staff considered this additional analysis of the feedwater check valve closure event to be acceptable for restart. Inspection Report 50-327, 328/88-12 stated that the staff has not resolved the issue as to the appropriate long-term criteria for this analysis.

In response to the open item (Reference 18), TVA stated that it considered the feedwater line waterhammer analysis already performed as an appropriate basis for the long-term evaluation of the check valve closure event. TVA further stated that the analyses were consistent with the design loading combinations identified in Section 3.9.2.2 and Tables 3.9.2-2, 3.9.2-5 and 3.9.2-6 of the FSAR. During this inspection, TVA discussed the analyses that had been performed for the check valve waterhammer event. According to TVA, it had performed three separate analyses of loop 1. The first analysis assumed that all supports were active, and the results of this analysis showed that several supports would be overloaded. The second analysis assumed that selected pipe whip restraints were active, based on the results of the first analysis, and the results showed that the piping would not meet strain limits that had been developed based on criteria contained in Appendix F of the ASME Boiler and Pressure Vessel Code. The third analysis assumed that all pipe whip restraints were active, and the results showed that the piping met the strain criteria limits. Although the third analysis showed that the piping met the strain criteria, an analysis that allows support failures are not a normal design practice for piping system design. Because the issue of feedwater check valve waterhammer has been raised on another facility, this item still remains open pending development of a generic NRC staff position on the appropriate acceptance criteria. No TVA action is required on this issue at this time.

(Closed) Unresolved Item URI 88-12-07, HVAC Duct Support Calculations

IR 50-327, 328/88-12, contained the results of a review of TVA's support calculations for heating, ventilation, and air conditioning (HVAC) ducts. The NRC staff had originally identified concerns with deficiencies in HVAC duct support calculations in Observation CEB-16 of IR 50-327, 328/87-27. In response to the observation, TVA wrote CAQR SQT870843. To resolve CAQR SQT870843, TVA selected five worst case duct systems that were qualified by computer analyses. TVA contracted Gilbert/Commonwealth (TVA Consultant) to perform analyses on the five duct samples. The results of this evaluation are contained in the Gilbert/Commonwealth (G/C) report for Task R0006.

In the review of this report, the NRC staff found that TVA used the 7-percent damped amplified response spectra (ARS) to determine the seismic loads for both OBE and SSE in 4 of the 5 duct samples, namely ducts 1, 3, 4, and 5. The use of this damping violated the FSAR in which the use of 2-percent and 5-percent damped ARS was required for steel structures with bolted connections during cases of OBE and SSE loading. During the inspection, TVA performed additional calculations using the loads calculated from the 5-percent damped ARS to show that the HVAC ducts and duct supports meet the restart criteria. TVA's use of 5-percent damping for SSE loading condition was acceptable to the staff for the HVAC duct evaluation. The results of these preliminary calculations showed that the sampled HVAC systems met the restart criteria requirements. TVA submitted the final calculations for review (Reference 39). The staff reviewed these calculations and found them acceptable for restart.

For Sample 2, the staff reviewed the support calculations (RIMs Nos. B25 871120 450, B25 871120 453 through B25 871120 455) and also reviewed the preliminary G/C calculations in which overstresses in connection welds and drilled-in anchors were identified. The staff review of the G/C calculations found that the overstressed welded connections were adequate to transfer the axial loads and would act as pinned connections rather than fixed connections, as modelled in the computer analyses. The staff requested that these preliminary calculations for qualifying the overstressed welds be revised to reflect the pinned end connections before the restart. TVA submitted the finalized G/C calculations (RIMS Nos. B25 880224 308 through B25 880224 313), which considered the welded connections as pinned connections (Reference 39). The staff reviewed these calculations and found the results acceptable for restart. Also, the staff accepted that the drilled-in anchors for sample 2 met a short-term safety factor of 2.0.

In IR 50-327, 328/88-12, the NRC stated that TVA should compute the evaluations of the five duct samples to the long-term criteria and that TVA should select additional samples from other duct systems to evaluate to the long-term criteria.

In a letter dated June 30, 1989 (Reference 40), TVA proposed to use 4-percent damping (for OBE) and 5-percent damping (for SSE) for non-welded companion-angle duct construction and 7-percent damping (for both OBE and SSE) for pocket-lock-type duct construction for the long-term analysis and design of HVAC systems at SQN. These damping values were accepted by the staff in its letter dated August 25, 1989 (Reference 41).

At a meeting during this inspection, TVA informed the staff that the long-term HVAC (including supports) analyses and design calculations were completed and agreed to submit the design calculations for the staff review.

TVA provided calculations which qualified the five worst case samples to the long-term criteria. TVA Calculation SCG-5S-88-005, "HVAC Ducts and Supports - Study of Seismic Qualifications with 5% vs 7% System Damping," Rev. 1, 5/3/89 (RIMS B25 890503 801), showed that ducts and duct supports in samples 1, 3, 4 and 5 met the long-term criteria for the safe shutdown earthquake (SSE) condition using damping values of 5-percent for non-welded companion-angle duct construction and 7-percent for pocket-lock-type duct construction. In addition, TVA calculation CSG2S89-051, "Task R0099-Resolution of Post-Restart Ducting Issues," Rev. 0, 11/3/89 (RIMS B87 891106 001) showed that for samples 1, 3, 4 and 5, SSE loading condition goverened the design of the HVAC systems over the operating basis earthquake (OBE) loading condition. As a result of the TVA evaluations, no modifications were necessary for ducts and duct supports in samples 1, 3, 4 and 5.

Worst case sample 2 was also evaluated against the long-term criteria. However, several duct supports in this sample required modifications to baseplates and anchor bolts. TVA has issued Design Change Notice (DCN) M-01419A (RIMS B85 890918 015) to implement these modifications. This DCN shows that 7 supports in sample 2 duct system were modified by TVA.

In order to select additional samples from other duct systems, TVA has conducted a review of sample 2 to determine attributes that caused modifications in this duct system. This review, which concluded that modifications to duct supports in sample 2 were caused by the presence of large valves/dampers, is documented in Gilbert/Commonwealth Letter, W. J. Leininger (G/C) to P. G. Trudel (TVA), dated 4/27/89 (RIMS B25 900327 023). In accordance with the findings reached in this report, TVA performed a horizontal review of drawings to identify those ductwork systems which contained large valves. This horizontal review identified only one duct system which contained large valves and was not previously evaluated by TVA. The details of this horizontal review is included in TVA Calculation

SCG-2S89-141, "Resolution of Duct and Duct Support Qualification Task R0006," Rev. 1, 3/19/90 (RIMS B87 900320 001). This duct system (called sample 6 by TVA) was then evaluated by computer analysis to long-term criteria, as shown in TVA Calculation SCG-2S89030, "HVAC System 6 - Analysis Model and Ductwork Evaluation," Rev. 0, 3/26/90 (RIMS B87 900326 030). As a result of this analysis, one support in this duct system was modified to meet long-term criteria.

The staff found the actions taken by TVA to resolve this URI 88-12-07 adequate. Therefore, URI 88-12-07 is closed.

(Open) Unresolved Item URI 88-12-08, Component Damping Values

In IR 50-327, 328/88-12, the NRC staff stated that during the Integrated Design Inspection (IDI) review of the component cooling water heat exchanger calculation, the team discovered that TVA was using damping values for component qualification from Regulatory Guide 1.61 instead of the damping values specified in FSAR Table 3.7.1-3 for welded structures. The original issue resulted from the IDI follow-up review of deficiency D3.4-3 documented in IR 50-327, 328/88-13. In IR 50-327, 328/88-12, the staff stated that it had considered TVA's use of current licensing criteria acceptable for restart. However, in the IR, the staff stated that the issue of appropriate damping values for mechanical components is an open post-restart issue.

TVA's response to the open item (Reference 20) stated that it considered the damping values appropriate for components. TVA's basis for its position was that the NRC had required TVA to demonstrate compliance with the Institute of Electrical and Electronic Engineers (IEEE) standard, IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The IEEE standard specifies the same damping values as Regulatory Guide 1.61. TVA further stated that it was revising the Sequoyah FSAR to incorporate these values.

The background on TVA's use of equipment damping values for Sequoyah was discussed during the inspection. To procure equipment, TVA had used Appendix F of the Quality Assurance Manual, "Design Criteria For Qualification of Seismic Class I and Seismic Class II Mechanical and Electrical Equipment, February 11, 1971. This document had been identified by TVA as a design criteria document during the design baseline verification program at the Sequoyah Nuclear Power Plant. The document specifies the same damping value contained in FSAR Table 3.7.1-3 for welded structures that had been identified during the NRC's IDI as the applicable damping value.

During the Sequoyah licensing review, the NRC staff and TVA had discussed the implementation of the new revision to the IEEE standard, IEEE 344-1975. These discussions focused on the new testing requirements specified in IEEE 344-1975. In September 1976, the staff performed a seismic audit on TVA's equipment qualification (Reference 42) to assess TVA's implementation of the new seismic qualification requirements of IEEE 344-1975. According to TVA, the majority of the equipment at Sequoyah was purchased according to the older revision of the IEEE standard, IEEE 344-1971, and TVA had upgraded the procurement requirements to IEEE 344-1975 in October 1974. TVA's response to the seismic audit performed by the NRC staff (Reference 43) made it clear that TVA considered its licensing basis to be IEEE 344-1971.

The Sequoyah FSAR, Section 3.9.2.1, also emphasizes that equipment was qualified in accordance with guidelines established by IEEE 344-1971. The staff's safety evaluation report, NUREG-0011, Supplement No. 1, states that the staff reviewed the equipment qualification at Sequoyah to address the concern of whether or not the original testing or analysis could be justified in light of the current criteria (IEEE Standard 344-1975). Therefore, the Sequoyah licensing documents do not support TVA's contention that the staff required TVA to demonstrate compliance with IEEE 344-1975.

The concern with the appropriate damping values for the qualification of mechanical components is based on establishing the appropriate level of safety consistent with Sequoyah's licensing basis. During the licensing of Sequoyah, the staff raised a concern with the level of conservatism in the earthquake input used for design. To address this concern, TVA developed a new site-specific earthquake input. TVA used the new site-specific earthquake to evaluate selected structures, systems, and components, and to demonstrate that the design margins were adequate to accommodate the larger site-specific earthquake input. As part of its evaluation of the effect of the site-specific earthquake on structures and components at Sequoyah, the staff had considered the conservatism in the damping values used in the analysis of structures and components at Sequoyah compared to those values contained in Regulatory Guide 1.61. The staff's evaluation is contained in Section 2.5 of NUREG-0011, Supplement No. 1. In response to concerns expressed by the Advisory Committee on Reactor Safeguards (ACRS), the staff performed additional reviews of the seismic margins of equipment. The additional staff review was documented in an internal TVA Meeting/Trip Report dated April 1, 1982. According to TVA, most of the equipment reviewed met the normal design allowables using the input from the site-specific earthquake. However, the results of the NRC staff review were not documented in formal correspondence. Since TVA has subsequently reanalyzed mechanical components using the higher damping values, TVA has altered the design basis reviewed at licensing to determine the acceptability of equipment to withstand the site-specific earthquake.

TVA also provided a copy of a recent staff safety evaluation on the replacement items program (Reference 44). Although this safety evaluation addressed replacement items for seismically-qualified equipment, the safety evaluation did reference IEEE 344-1975 as the basis for the qualification of replacement parts. However, the safety evaluation also stated that the issue of equipment qualification for the site-specific earthquake is still open.

On the basis of the review of the background information on the seismic qualification of equipment, the team considers TVA's use of the current damping values for mechanical components referenced in IEEE 344-1975 to be a relaxation of the original design basis for Sequoyah. For mechanical components, TVA should either reanalyze those items for which higher damping values were used or demonstrate that those items meet the appropriate design criteria using the site-specific earthquake input and the higher damping values. This item remains open pending TVA's submittal of an acceptable plan for the resolution of this item.

(Open) Unresolved Item URI 88-12-09, <u>ERCW Pumping Station Access Cells</u>

In IR 50-327, 328/88-12, the NRC staff stated that during the IDI review the team identified a concern with TVA's assumptions used in the evaluation of the ERCW access cells. The concern was originally identified as Deficiency D4.2-1 in IR 50-327, 328/87-48.

As discussed in IR 50-327, 328/88-12, the original seismic analysis of the access cells assumed that the six cells and the interconnecting cells will act as a single "J-shaped" unit. Contrary to this assumption, the design calculations predicted that shrinkage will occur in the interior concrete fill. This shrinkage would cause a gap between interior concrete and the exterior steel sheet piling. TVA design criterion SQN-DC-V-104.5 contains the statement, "the sheet pile sections serve only as forms for the tremie concrete; therefore, quality assurance is not required for these sheet pile sections." The calculations also predicted that there would be vertical movement between adjacent cells. Beams were designed to tie the cells together horizontally but not vertically. In fact, compressible material was placed above and below these beams to preclude load transfer in the vertical direction. A TVA internal memorandum from J. H. Coulson, Principal Civil Engineer, to the Civil Engineering and Design Branch files, dated October 13, 1977, stated, "cells A through F and the ERCW pumping station are individual rigid bodies capable of moving vertically with respect to each other."

The inability to transfer vertical shear between the cells made the original assumption of a single J-shaped unit invalid. Furthermore, even if the assumption was valid, torsional loads should have been considered in the analysis and design because the J-shaped unit is not symmetrical. The calculations also stated the following: "RJH & RDG analyzed cells as both individual cells and as a unit. The former case showed the cells were unstable and the latter case showed a stable unit acting as a rigid body." The calculations also showed that cells are unable to transfer vertical shear, making the original assumption of a single J-shaped unit invalid. The calculation also contained the statement that the cells are not stable if they act as individual cells. In addition, cores taken in November 1977 in several cells indicated that the concrete at the bottom of these cells was soft, crumbly, or contained gravel pockets and cavities. TVA reanalyzed the ERCW access cells using a non-linear seismic time history response analysis. The revised seismic analysis for the ERCW cell was based on a two-dimensional non-linear time history analysis method in which the foundation was represented by discrete springs and dampers with no tension capability in the vertical direction (Reference 45). The analyses using the lower-bound concrete modulus at the ERCW pipe elevation revealed a maximum base uplift and maximum lateral displacement of about 83 percent and 0.89 inches, respectively. The maximum toe pressure was about 800 psi. The staff was concerned that the magnitude of the seismic response toe pressure with respect to the low strength of the soft concrete at the base of the cell. TVA performed additional analysis which assumed that the soil surrounding the cell and sheet pile would interlock to confine the soft concrete and gravel. The results of this analysis indicated a factor of safety of 1.05 against failure.

The staff had considered the results of these evaluations acceptable for restart. In IR 50-327, 328/88-12, the staff requested that TVA submit an evaluation program after the restart to evaluate the stability and deflections of the access cells using as-built conditions and ensure that the ERCW piping will not be overstressed because of access cell deflections.

In the letter of October 20, 1988 (Reference 16), TVA provided its long-term evaluation report of the access cells and concluded that the ERCW pumping station access cells are structurally adequate and will function as intended under the design loads, including during an earthquake. The staff and its consultant reviewed this report and, during the meeting held on November 28, 1988 (Reference 33), raised the following three concerns that require further justifications:

(1) The use of a friction coefficient of 1.0 for tremic concrete-to-rock interfaces,

(2) The use of allowable bearing stress on rock equal to 1500 psi, and

(3) The use of allowable bearing stress on gravel equal to 600 to 800 psi.

On December 28, 1988, TVA provided its justification to the three concerns (Reference 17), and provided the final evaluation report for the access cells on March 1, 1990 (Reference 36). In the review of these two documents, the staff found that TVA's justification for the allowable bearing stress of rock and gravel appeared reasonable, and the final access cell evaluation report was acceptable, except for the use of a friction coefficient of 1.0 for the evaluation of sliding against earthquake loading. TVA applied the concept of shear friction for reinforced concrete and used the coefficient of friction recommended in the ACI 318-71 Code (Section 11.15.4) for the dynamic stability (sliding) evaluation of the cells (Reference 17). The staff did not consider this acceptable because the concept of shear friction and the friction coefficients are applicable only when enough reinforcement is provided across the interface of the two elements. For the case of access cells (tremic concrete with sheet-pile form work) rested on bedrock, it is not possible to develop shear friction at interface between the cells and bed rock. Therefore, the staff does not accept the use of a friction coefficient of 1.0, and URI 88-12-09 remains open pending TVA's submittal of an acceptable response to address the staff's concern.

(Open) Unresolved Item URI 88-12-10, Seismic Analysis of the Steel Containment Vessel

In IR 50-327, 328/88-12, the NRC staff stated that TVA's review of the steel containment vessel vertical response spectra for the time step issue evaluated the effects on a sample of piping analysis problems attached to the affected structures including the reactor coolant loop. The concern with the vertical response spectra was originally discussed in deficiency D4.2-3 that was documented in IR 50-327, 328/87-48. TVA had originally developed the response spectra using the computer code RESPONSE and a time step of 0.01 second for the integration. When TVA used a smaller time step (0.005 second) for the integration, there was a significant increase in the vertical response in the frequency range of 20 to 30 Hz. TVA had evaluated the piping analysis sample using the new vertical response spectra generated with the smaller time step (Reference 46). TVA also provided an evaluation of a sample of piping systems attached to the reactor coolant loop for the time step issue (Reference 47). In IR 50-327, 328/88-12, the staff stated that the sample analyses were adequate for restart. This IR also contained a recommendation for TVA to complete the evaluation of the remaining piping analyses as a post-restart item.

In response to the open item (Reference 21) before the inspection, TVA confirmed the adequacy of the sample evaluation and stated that no further evaluation of the remaining piping analysis problems was necessary. During the inspection, the team learned of additional TVA activities that would affect the evaluation of this item. These activities are the post-restart revision of RCL spectra and the reanalysis of a substantial number of problems including both SCV-attached and RCL-attached piping. Parts 1 and 2 of the following discussion address the effects of the SCV and RCL spectra revision on piping qualification, respectively.

Part 1 - Revised SCV Spectra Effects

Since restart, TVA has revised 18 SCV-attached piping analyses and has identified modifications for 15 of the 18 analyses. TVA attempted to determine which of the modifications were caused by the revision to SCV spectra. However, corollation between SCV spectra revision and required modifications could not be determined because the reanalyses incorporated other revised design information including many system modifications not associated with design issues. During this inspection, the team requested that TVA identify available design margins in other SCV-attached analyses. In response to this request, TVA reviewed additional SCV-attached analysis packages that had not yet been revised for the new spectra. This review identified numerous examples of high design load-to-allowable load ratios for pipe supports, with a maximum ratio of 0.99. The ratios suggest some supports could require revision to reconcile revised input loading such as revised SCV seismic spectra.

Because TVA identified small design margins available in current design, the team concludes that TVA should expand the pre-restart study of SCV spectra revision to justify current design for the balance of SCV-attached piping. This part of the issue is open until TVA submits a response describing the results of an evaluation of the remaining piping systems attached to the SCV.

Part 2 - Revised RCL Spectra Effects

Following the restart, TVA revised the Sequoyah RCL analysis to provide a supporting calculation for the design basis spectra because they could not retrieve the original calculation from its records. TVA also performed this calculation to answer a concern regarding damping associated with the spectra, and to help resolve a concern regarding unconservative analyses of RCL seismic anchor motions. From the reanalysis, TVA revised its spectra and anchor motion loads, and invalidated the conclusions of the earlier Bechtel evaluation of the time step issue for RCL-attached piping (Reference 47). The revised spectra peak accelerations occur at different frequencies and significantly exceed the previous spectra acceleration levels. During this inspection, the team did not review the revised RCL analysis but did review the effect of the revised spectra on piping qualification.

To evaluate the revised spectra effects, TVA considered two samples of RCL-connected piping analysis problems. One sample is included in the Bechtel North America Power Corporation report, "Final Report on Reactor Coolant Loop Spectra Evaluation, Sequoyah Nuclear Power Plant Units 1 and 2," dated April 1990. In this study, all RCL-attached piping analysis problems were ordered in accordance with a conservative estimate of stress. The stress estimate was based on the then current stress levels factored by spectral acceleration increases for significant participating modes. Six problems were selected based on "worst case" estimated stress. The sample included analysis problems with a relatively high number of planned modifications and excluded analysis problems that were undergoing revision. Each of the six sample problems was revised using the latest design information. The second sample of problems consisted of problems for which reanalyses were already completed or were underway.

TVA performed an evaluation for all resultant modifications of both samples to determine whether modifications resulting from the reanalyzed problems were caused by the new spectra. A test analysis was performed on those reanalyzed problems for which modifications had been identified. The test analysis included the same input that was used in the reanalyses except that the previous spectra revision was used in the test cases. This allowed TVA to identify modifications caused by the spectra revision. This effort demonstrated that the spectra revision did not cause any of the modifications required for the six sample problems from the Bechtel study. However, TVA determined that three modifications associated with one of the sample problems of the second set resulted from the spectra revision.

The team concluded that TVA efforts have not demonstrated that piping design is unaffected or that additional modification will not be required because of the revised RCL spectra. The significantly increased peak accelerations, peak frequency shifts, and identified modifications indicate additional evaluation of the spectra is required. This part of the issue remains open pending TVA's submittal of a response describing the results of an evaluation of the remaining piping systems attached to the RCL.

(Open) Unresolved Item URI 88-12-11, Diesel Generator Exhaust Piping

In IR 50-327, 328/88-12, the NRC staff stated that in their review of TVA's evaluation of piping in the diesel generator building for the effects of the time step issue on the seismic response spectra, the staff identified that TVA had used interim criteria that had not been reviewed or approved by the staff. The concern was identified in TVA calculation "Evaluation of CAQRSQF879242, N2-870242-Misc," Revision 1, dated February 25, 1988 (RIMS B25 880226 800). In Stress problem N2-82-3A, the staff identified that TVA used interim criteria from CEB-CI-21.89 to qualify hanger 17A586-01-001. The criteria used by TVA involved a modified fatigue evaluation for the secondary load case. This criteria had not been accepted by the staff for general use unless a case-by-case review and approval was obtained (Reference 24). According to TVA, the criteria had only been applied at the welded attachments to the typical diesel generator exhaust lines. The staff had considered the use of these criteria to be acceptable if TVA would visually inspect the affected support attachments on the diesel generator exhaust lines for damage. The IR stated that TVA should provide the results of this inspection to the NRC.

TVA's response to the open issue (Reference 22) stated that Gilbert/Commonwealth (G/C) had performed a field inspection of five of the eight geometrically similar exhaust lines. G/C did not inspect the remaining three lines because the insulation had not been removed. G/C's memorandum to TVA dated March 7, 1988, noted that the measured gaps between the pipe lugs and the diesel generator building roof sleeves varied between 1/2 inch and 1-1/4 inches for the gaps most susceptible to the lateral thermal growth of the exhaust lines. G/C indicated that the inspected exhaust lines showed no apparent damage caused by contact between the pipe lugs and the roof sleeves. On April 12, 1990, the team performed a similar inspection of five of the eight exhaust lines that included the three lines that G/C had not previously inspected. The team confirmed G/C's conclusion that no signs of damage to the inspected exhaust lines were apparent. The eight diesel generator exhaust lines are depicted in plan and elevation on TVA Drawing No. 45-M-4-17W586-1, "Mechanical Exposed Oil, Air, Water & Misc. Piping," Revision No. 17, dated March 16, 1984. The referenced drawing designates the diesel generator exhaust lines as Seismic Category I, TVA Class G piping.

The team also reviewed TVA's piping analysis and pipe support calculations of record for the diesel generator exhaust lines to confirm that TVA had qualified the piping, lugs, and supports in accordance with the latest revisions of TVA Design criteria No. SQN-DC-V-13.3, "Detailed Analysis of Category I and I(L) Piping Systems," and Design Criteria No. SQN-DC-V-24.2, "Supports for Rigorously and Alternately Analyzed Category I Piping."

TVA Calculation No. N2-82-03A, "Summary of Analysis for N2-82-03A," Revision No. 1, dated December 20, 1988 (RIMS No. B25 881221 805) summarizes the detailed piping analysis that generically qualifies the eight exhaust lines. The following documents specify the qualifications for the piping-to-lug interface: TVA Calculation No. N2-870242-Misc, "Evaluation of CAQRSQF870242, N2-870242-Misc," Revision No. 1, draft issue (Revision No. 0 RIMS No. B25 880226 800), and TVA Calculation No. 17A58601001/N2-82-03A/MCLC09, "System 82/Calculations for Pipe Support 17A58601001," Revision No. 2, dated May 19, 1989 (RIMS No. B87 890525 004).

Based on a programmatic review of the referenced TVA calculations, the team recommended that TVA incorporate (or confirm the incorporation of) the following design attributes into the referenced calculations:

1. Address the minimum as-built gap of 1/2 inch that G/C documented.

2. Limit the maximum permissible lateral pipe movement caused by the combined effect of thermal and seismic loads to the minimum as-built gap.

3. Consider the effect of the relative lateral seismic movement of the diesel generator roof and slab on the size of the minimum as-built gap.

4. Consider the effect of the radial thermal growth of the 22-inch diameter exhaust line on the size of the minimum as-built gap.

5. Confirm that the latest design spectra of record for the diesel generator building were used to analyze the exhaust lines.

6. Confirm that the exhaust line piping, lugs, and supports have been analyzed in accordance with the requirements of design criteria documents SQN-DC-V-13.3 and SQN-DC-V-24.2 for TVA Class G Seismic Category I piping and supports.

7. Evaluate the piping configuration for the thermal case alone, with friction.

9. Confirm that the axial growth of the exhaust line silencer has been included in the piping analysis.

On April 12, 1990, TVA agreed to confirm or implement these recommendations to close URI 88-12-11. This item remains open pending TVA's submittal of a response providing the results of the additional evaluations.

APPENDIX B

C.1 ENTRANCE MEETING - APRIL 9, 1990

NAME	ORGANIZATION	TITLE
John R. Fair	NRC	Team Leader
Robert E. Serb	NRC	Consultant
A. V. duBouchet	NRC	Consultant
John K. McCall	TVA	Chief Civil Engineer
Tracy A. Flippo	TVA	QA Manager
Marci Cooper	TVA	Acting Site Licensing Manager
Karl S. Seidle	TVA	Engineering Mechanics Manager
M. Von Schimmelmann	TVA	Site Audit Group
Dick Connelly	TVA	Public Affairs
Paul Harmon	NRC	Resident Inspector
D. L. Lundy	TVA	Lead Civil Engineer
Paul G. Trudel	TVA	Project Engineer
M. G. Maxwell	TVA	Senior Civil Engineer
J. R. Bynum	TVA	V. P. Nuclear Production
C. A. Vondra	TVA	Plant Manager
W. L. Byrd	TVA	Manager Project Controls Financial Services

C.2 EXIT MEETING - APRIL 13, 1990

John R. Fair	NRC	Team Leader
A. V. duBouchet	NRC	Consultant
John K. McCall	TVA	Chief Civil Engineer
Carlo Brillante	TVA	Lead Engineer
Karl S. Seidle	TVA	Engineering Mechanics Manager
Kenneth A. House	TVA	Principal Civil Engineer
W. L. Byrd	TVA	Manager Project Controls Financial Services
J. R. Bynum	TVA	V. P. Nuclear Production
Tracy A. Flippo	TVA	QA Manager
William R. Bibb, Jr.	TVA	Engineering Specialist
George Sanders	G/C	Engineering Manager
M. G. Maxwell	TVA	Senior Civil Engineer
S. J. Patel	TVA	Principal Civil Engineer
Marcia A. Cooper	TVA	Acting Site Licensing Manager
Paul G. Trudel	TVA	Project Engineer
D. L. Lundy	TVA	Lead Civil Engineer
Robert E. Serb	NRC	Consultant
J. N. Donohew, Jr.	NRC	Senior Project Manager
D. Terao	NRC	Chief, Engineering Branch
L. J. Watson	NRC	Chief, Project Section 1
S. W. Spencer	TVA	Licensing Engineer
M. J. Burzynski	TVA	Unit 2 Cycle 4 Outage Manager
J. W. Proffitt	TVA	Acting Licensing Manager

APPENDIX C

REFERENCES

1. Letter from R. L. Gridley (TVA) to B. J. Youngblood (NRC), transmitting the Design Calculation Review Plan, January 20, 1987.
2. Letter from S. A. White (TVA) to NRC, transmitting Revision 1 to the Sequoyah Nuclear Performance Plan, April 1, 1987.
3. Letter from R. L. Gridley (TVA) to NRC, transmitting additional information on the Design Calculation Review Plan, July 31, 1987.
4. Letter from R. L. Gridley (TVA) to NRC, transmitting the program plan for regenerating pipe support calculations, August 21, 1987.
5. Inspection Report (IR) 50-327, 328/87-06, forwarded by S. Ebneter letter, April 8, 1987.
6. IR 50-327, 328/87-27, forwarded by S. Ebneter letter, August 24, 1987.
7. IR 50-327, 328/87-64, forwarded by S. Richardson letter, February 23, 1988.
8. IR 50-327, 328/87-48, forwarded by S. Ebneter letter, November 6, 1987.
9. IR 50-327, 328/87-74, forwarded by S. Richardson letter, February 22, 1988.
10. IR 50-327, 328/88-13, forwarded by S. Ebneter letter, May 26, 1988.
11. IR 50-327, 50-328/88-12, forwarded by S. Richardson letter, June 24, 1988.
12. Letter from M. Ray (TVA) to NRC, providing TVA's response to Unresolved Item (URI) 88-12-01, June 8, 1989.
13. Letter from R. Shell (TVA) to NRC, providing TVA's response to URI 88-12-02, July 19, 1989.
14. Letter from R. Gridley (TVA) to NRC, providing TVA's response to URI 88-12-03, March 30, 1989.
15. Letter from M. Ray (TVA) to NRC, providing TVA's response to URI 88-12-04, November 9, 1989.
16. Letter from R. Gridley (TVA) to NRC, providing TVA's response to URI 88-12-05 and URI 88-12-09, October 20, 1988.
17. Letter from R. Gridley (TVA) to NRC, providing TVA's response to URI 88-12-05 and URI 88-12-09, December 28, 1988.
18. Letter from M. Ray (TVA) to NRC, providing TVA's response to URI 88-12-06, July 31, 1989.

19. Letter from R. Gridley (TVA) to NRC, providing TVA's response to URI 88-12-07, March 15, 1989.

20. Letter from R. Gridley (TVA) to NRC, providing TVA's response to URI 88-12-08, March 10, 1989.

21. Letter from M. Ray (TVA) to NRC, providing TVA's response to URI-88-12-10, June 8, 1989.

22. Letter from M. Ray (TVA) to NRC, providing TVA's response to URI 88-12-11, May 10, 1989.

23. Letter from R. Gridley (TVA) to NRC, regarding the regeneration of Sequoyah Unit 2 pipe support calculations, August 31, 1987.

24. NRC summary of September 1, 1987 meeting regarding the Sequoyah pipe support criteria, September 4, 1987.

25. Letter from R. Gridley (TVA) to NRC, providing supplemental information on the Sequoyah Unit 2 pipe support restart criteria, dated November 17, 1987.

26. Letter from R. Gridley (TVA) to NRC, providing additional information on swing angle allowables, design basis accident spectra and U-bolt allowables March 2, 1988.

27. NUREG-1232, Volume 2, "Safety Evaluation Report on Tennessee Valley Authority: Sequoyah Nuclear Performance Plan," May 1988.

28. IR 50-260/89-15, forwarded by B. D. Liaw letter, May 18, 1989.

29. NUREG-0011, Supplement 1, "Supplement No. 1 to the Safety Evaluation Report by the Office of Nuclear Reactor Regulation; U. S. Nuclear Regulatory Commission In the Matter of Tennessee Valley Authority, Sequoyah Nuclear Plant, Units 1 and 2; Docket Nos. 50-327 and 50-328," February 22, 1980.

30. Letter from R. Gridley (TVA) to NRC, Subject: "Effect of Zero Period Acceleration (ZPA) on piping during the design basis accident (DBA)" March 2, 1988.

31. NRC Safety Evaluation, Subject: "Piping Analysis Damping - ASME Code Case N-411" forwarded by letter, February 8, 1988.

32. Letter from R. Gridley (TVA) to NRC, providing additional information on the essential raw cooling water (ERCW) pumping station concrete, March 3, 1988.

33. NRC summary of November 28, 1988 meeting regarding the ERCW pumphouse foundation and roadway access cells, January 27, 1989.

34. Letter from M. Ray (TVA) to NRC, Subject: "Sequoyah Nuclear Plant (SQN) - Essential Raw Cooling Water (ERCW) Pumphouse Foundation and ERCW Pumping Station Access Cells," July 10, 1989.

35. Letter from S. C. Black (NRC) to O. D. Kingsley, Jr. (TVA), Subject: "Essential Raw Cooling Water Pumphouse Foundation And Roadway Access Cells," August 4, 1989.

36. Letter from E. G. Wallace (TVA) to NRC, Subject: "Sequoyah Nuclear Plant (SQN) - Essential Raw Cooling Water (ERCW) Pumphouse Foundation and ERCW Pumping Station Access Cells," March 1, 1990.

37. Letter from E. G. Wallace (TVA) to NRC, Subject: "Sequoyah Nuclear Plant (SQN) Essential Raw Cooling Water (ERCW) Pumphouse Foundation and ERCW Pumping Station Access Cells," April 13, 1990.

38. Letter from R. Gridley (TVA) to NRC, providing additional response (Observation MEB-3) to IR 50-327, 328/87-27 February 18, 1988.

39. Letter from R. Gridley (TVA) to NRC, providing additional information on conduit and HVAC duct support calculations March 2, 1988.

40. Letter from M. Ray (TVA) to NRC, Subject: "Sequoyah Nuclear Plant (SQN) Units 1 and 2 Heating, Ventilating, and Air Conditioning (HVAC) Duct Support Calculations - Unresolved Item 88-12-07," June 30, 1989.

41. Letter from S. Black (NRC) to O. D. Kingsley, Jr. (TVA), Subject: "HVAC Duct Calculations, Unresolved Item 88-12-07," August 25, 1989

42. Letter from S.A. Varga (NRC) to G. Williams, forwarding the trip report on the seismic audit of TVA equipment, November 16, 1976.

43. Letter from J. E. Gilleland (TVA) to S. A. Varga (NRC), providing TVA's response to the NRC seismic audit of equipment, February 7, 1977.

44. Letter from J. G. Keppler (NRC) to S. A. White (TVA), providing the staff position on the replacement items program seismic screening methodology, October 29, 1987.

45. Letter from M. Ray (TVA) to NRC, providing a supplemental response to IDI Item D4.2-1, March 2, 1988.

46. Letter from M. Ray (TVA) to NRC, providing additional information on IDI Item D4.2-3, March 2, 1988.

47. Letter from R. Gridley (TVA) to NRC, providing additional information on the effect of the time step concern on the RCL spectra, March 2, 1988.

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U.S. NUCLEAR REGULATORY COMMISSION

INSPECTOR'S REPORT

Office of Nuclear Reactor Regulation

PRINCIPAL INSPECTOR (Name, last, first and middle initial): FAIR, JOHN R.

REVIEWER: D. TERAO

[INSP]ECTORS: A.V. duBouchet (Consultant); R.E. Serb (Consultant)

LICENSEE/VENDOR	TRANSACTION TYPE	DOCKET NO (8 digits) OR LICENSE NO (BY PRODUCT) (13 digits)	REPORT NO	SEQ	NEXT INSPEC DATE MO	YR
TVA	X I - INSERT	05000327	18	A		
	M - MODIFY	05000328	18	B		
	D - DELETE			C		
	R - REPLACE			D		

PERIOD OF INVESTIGATION/INSPECTION: FROM 04 09 90 TO 04 13 90

INSPECTION PERFORMED BY: 1 - REGIONAL OFFICE STAFF; 2 - RESIDENT INSPECTOR; 3 - PERFORMANCE APPRAISAL TEAM; X OTHER: HQ Team

ORGANIZATION CODE OF REGION/HQ CONDUCTING ACTIVITY (See IEMC 0530 Manpower Reporting—Weekly Manpower Reporting for code): REGION | DIVISION | BRANCH

REGIONAL ACTION (Check one box only): 1 - NRC FORM 591; 2 - REGIONAL OFFICE LETTER

TYPE OF ACTIVITY CONDUCTED (Check one box only): 02 - SAFETY; 03 - INCIDENT; 04 - ENFORCEMENT; 05 - MGMT AUDIT; 06 - MGMT VISIT; X 07 - SPECIAL; 08 - VENDOR; 09 - MAT ACCT; 10 - PLANT SEC; 11 - INVENT VER; 12 - SHIPMENT EXPORT; 13 - IMPORT; 14 - INQUIRY; 15 - INVESTIGATION

INSPECTION/INVESTIGATION FINDINGS (Check one box only): A: X, B: X — 1 - CLEAR; 2 - VIOLATION; 3 - DEVIATION; 4 - VIOLATION & DEVIATION

TOTAL NUMBER OF VIOLATIONS AND DEVIATIONS: A B C D

ENFORCEMENT CONFERENCE HELD: A B C D — 1 - YES

REPORT CONTAIN 2.790 INFORMATION: A B C D — 1 - YES

LETTER OR REPORT TRANSMITTAL DATE: NRC FORM 591 OR REG LETTER ISSUED: 05 25 90; REPORT SENT TO HQ FOR ACTION: MO DAY YR

MODULE INFORMATION

REC ORD TYPE	MODULE NUMBER INSP	SEQ	DIRECT INSPECTION EFFORT IN STAFF HOURS EXPENDED THIS INSPECTION	PERCENTAGE COMPLETED TO DATE	Description
B	30703	A			Entrance and Exit Meetings
		B	3		
		C			
		D			
B	09902S	A	18		Review of Post Restart IDI Items
		B	40		
		C			
		D			
B	537700[6]	A	18	25	Review of Post Restart Calculation Program Items
		B	40	50	
		C			
		D			

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