UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

# SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

# FOR EMPLOYEE CONCERNS

# SEQUOYAH NUCLEAR POWER PLANT UNITS 1 AND 2

# ELEMENT REPORT EN 215.1(B) "CIVIL/STRUCTURAL DESIGN-SEISMIC CRITERIA"

## I. SUBJECT

Category: Engineering

Subcategory: Civil/Structural Design

Element: Seismic Criteria

Concern: 00-85-005-009

Sequoyah Nuclear Plant is sited on an earthquake fault that runs from around Chattanooga to north of Knoxville. If there were an earthquake, power plant structures could fail.

## II. SUMMARY OF ISSUE

1. Sequoyah plant is on an earthquake fault that runs from Chattanooga to Knoxville.

2. Plant structures could fail in an earthquake.

## III. EVALUATION

As discussed in the Subject Element Report and the SER issued in March 1979 (Reference 1), the nearest regional fault to both the Sequoyah and Watts Bar Nuclear Power Plant is the Kingston Fault, which lies about one mile to the northwest of the plant at its closest approach. This fault is about 150 miles long, strikes northeast and dips at least 30° to the southeast. Projection along the dip of the fault would place it at least 2000 feet beneath both plant sites.

The Kingston Fault is one of numerous low angle thrust faults that characterize the Southern Valley and Ridge Tectonic Province. These faults range in length from several tens of miles to more than 100 miles. They were formed during the Applachian Orogeny in the late Paleozoic Era (more that 250 million year ago). There is no evidence that these faults have been active since that time, however, outcrops that expose cross-cutting relationships between the faults and overlying younger strata are rare. The following are the bases presented

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in the Clinch River SER (Reference 2) to support the staff's conclusion that these faults are not capable in the meaning of Appendix A, 10 CFR Part 100:

1. Extensive filed research has been conducted in the region with the intent of finding evidence for recent displacement along these faults to explain current seismicity, and none has been found.

2. Triassic dikes mapped in Virginia penetrate Valley and Ridge Province structures without being offset.

3. In Alabama where Coastal Plain deposits overlie the southern part of the Valley and Ridge Province structure there is no evidence of offset.

4. Where subsidiary faults of the major thrust faults have been mapped in relation to overlying ancient terrace deposits, those terraces have not been offset (i.e., Phipps Bend and Watts Bar site fault investigation; TVA, 1975; TVA, 1974).

5. Radiometric age dating of gouge taken from the Copper Creek Fault, which is similar to the Kingston Fault and strikes parallel to it several miles to the east, indicates an age of at least 280 million years before present.

Seismological studies of instrumentally recorded earthquakes in eastern Tennessee and some of their aftershock sequences indicate that their hypocenters occur predominantly in the Precambrian basement well below the Paleozoic thrust faults. Fault plane solutions of these events suggest that the earthquake source mechanisms are inconsistent with the structural trends and the sense of predominant displacement on these low angle thrust faults that are characteristic of Valley and Ridge. On the other hand, trends and senses of motion of these earthquake are consistent with structures imaged in geophysical data taken within the Precambriam basement.

## IV. CONCLUSION

For the reasons stated above, the staff reaffirms its conclusions made in previous licensing activities regarding sites in eastern Tennessee, specifically Sequoyah and Watts Bar, that the regional low angle thrust faults, including the Kingston Fault, do not represent a ground displacement or seismic hazard to nuclear power plants in that region and concurs with the conclusion drawn in the subject element report.

## V. REFERENCES


1. NUREG-0011, "Safety Evaluation Report Related to Operation of Sequoyah Nuclear Plant, Units 1 and 2" dated March 1979.

2. NUREG-0968, "Safety Evaluation Report Related to Construction of Clinch River Breeder Reactor Plant" dated March 1983.

SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2
SAFETY EVALUATION REPORT FOR EMPLOYEE CONCERNS
ELEMENT REPORT 215.2-SQN
"CIVIL/STRUCTURAL DESIGN, CUT REBAR CONTROL"

I. Subject

Category: Engineering

Subcategory: Civil/Structural

Element: Civil/Structural Design, Cut Rebar Control

Concerns: IN-85-297-005, IN-85-868-004

The bases for Element Report 215.2-SQN, Rev. 1, dated January 27, 1987, are Sequoyah Employee Concerns IN-85-297-005 and IN-85-868-004 which questioned the structural integrity of the containment and the crane walls inside the reactor building because of over 2000 known releases for core drills due to penetrations of ducts, conduits and pipes.

These concerns were evaluated by TVA as potentially nuclear safety-related and potentially applicable to Sequoyah (generic).

II. Summary of Issues

The stated concerns as defined by TVA are: (a) cutting of rebar in the reactor containment and the crane walls inside the reactor building could have weakened the structure; (b) there are over 2000 known releases for core drills; and (c) procedural control/assessment of cut rebar to ensure structural integrity of concrete is in question.

III. Evaluation

Investigations by TVA personnel and consultants found that all three issues were valid and that there was a lack of procedural controls and adequate assessment of the cumulative effect of cut rebars. The NRC staff reviewed TVA's investigations and concurred with their findings. To resolve the three issues, TVA developed a corrective action plan (CAP) that included three pre-restart and three post-restart action items. The pre-restart CAP was to (1) revise existing plant procedures to ensure coordination between plant operations and TVA's Division of Nuclear Engineering, and develop a new procedure for documenting and controlling future rebar cuts, (2) develop a baseline map of cut rebars that ocurred in the reactor building during the construction phase, and assess the structural integrity of the shield wall and crane wall for the cumulative effect of both cut rebars and hanger loads (see Element Report 215.6-SQN for employee concern on hanger loads), and (3) assess the structural integrity of the most critically affected concrete elements in the auxiliary building - slabs at Elev. 714', 734' and 749', U-line wall, and other critical shield walls - for the cumulative effect of both cut rebars and hanger loads. The percentage of cut rebar was assumed to be the same as the worst percentage of cut rebar developed from the data for the corresponding concrete elements of the auxiliary building at the Watts Bar Nuclear Plant (WBN) because the concrete design is similar for the two plants and because such plant specific data for the SQN auxiliary building were not available. The assessments were done on the basis of the ultimate strength method as specified in design criteria SQN-DC-V-1.3.3.1, and the combination of dead, live, and FSAR OBE or SSE loads was considered.

TVA has completed the implementation of the pre-restart CAP. To assess the adequacy of the scope and implementation of the pre-restart CAP, the NRC staff performed a walkdown of the plant and audited a representative sample of the results of TVA's implementation. In addition, TVA was requested to compare the percentage of cut rebars between the SQN and WBN reactor buildings based on the available data from both plants. The comparison showed that the percentage of cut rebars in the reactor building was similar between the two plants, and the NRC staff accepted TVA's assumption for pre-restart CAP item (3) regarding the similarity in percentage of cut rebars between the SQN and WBN auxiliary buildings. TVA was also requested to verify that the structural assessments, which considered the seismic loads from the FSAR OBE and SSE, provided sufficient safety margins with respect to the seismic loads from the site-specific (84-percentile) SSE by evaluating the two most critically stressed locations of the slab in the auxiliary building at Elev. 714'. The evaluation results obtained by TVA demonstrated that the floor does possess sufficient margin to withstand the 84-percentile SSE. Based on the above evaluations, the NRC staff found the scope and implementation of the pre-restart CAP to be acceptable.

For the post-restart CAP, TVA will (1) develop a plant-specific baseline of cut rebars for all Category I concrete structures at SQN to facilitate the long term assessment of the cumulative effect of cut rebar and hanger loads, and also review the WBN cut rebar data and evaluations in detail because they were already complete, (2) revise Section 3.8 of the Sequoyah FSAR to clarify the use of the ultimate strength method as specified in design criteria SQN-DC-V-1.3.3.1 for the evaluation of the reactor building and auxiliary building because the ACI working stress method was the original FSAR criteria for the design of concrete for these two buildings, and (3) evaluate and document future cut rebar requests based on procedures developed from the pre-restart CAP. The NRC staff found the scope of the post-restart CAP to be sufficient.

## IV. Conclusions

The NRC staff reviewed TVA's investigation of the employee concerns and the CAP developed by TVA to resolve such concerns, and found they were adequate. TVA's implementation of the pre-restart CAP was also found acceptable. The NRC staff therefore believes TVA's resolution for the concerns as described in Element Report 215.2-SQN, Rev. 1, is acceptable.

SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2
SAFETY EVALUATION REPORT FOR EMPLOYEE CONCERNS
ELEMENT REPORT 215.6-SQN
"CIVIL/STRUCTURAL DESIGN, HANGER LOADS ON STRUCTURES"

I. Subject

Category: Engineering

Subcategory: Civil/Structural

Element: Civil/Structural Design, Hanger Loads on Structures.

Concerns: IN-85-220-003, IN-86-173-001

The bases for Element Report 215.6-SQN, Rev. 1, dated January 27, 1987, are Sequoyah Employee Concerns IN-85-220-003 and IN-86-173-001 which questioned the structural integrity of the supporting walls/floors in the Unit 2 reactor building annulus areas in particular, and in other concrete structures in general, due to the weight of an excessive number of hanger attachments.

These concerns were evaluated by TVA as potentially nuclear safety-related and potentially applicable to Sequoyah (generic).

II. Summary of Issues

The stated concerns as defined by TVA are: (a) structural integrity of concrete walls/slabs in the annulus area of the Unit 2 reactor building is questionable due to excessive number of hangers; and (b) design calculations have not evaluated individual and cumulative effects of hangers on concrete walls/slabs.

## III. Evaluation

Investigations by TVA personnel found both issues to be valid and identified one additional deficiency, i.e., lack of control and documentation for hanger loads. The NRC staff concurred with the findings from TVA's investigations. To resolve the employee conerns and the related deficiency, TVA developed a corrective action plan (CAP) which consisted of both pre- and post-restart corrective actions. The pre-restart CAP was to (1) perform live load evaluation of all Category I structure concrete slabs, (2) perform evaluation for two worst case shield walls, the reactor building shield wall, and the auxiliary building U-line wall, and (3) revise DNE and plant procedures to control approval for all future hanger attachments, and develop a program plan for the long term evaluation of remaining Category I concrete walls not covered by the pre-restart evaluation. For the concrete elements in the reactor and auxiliary building, the cumulative effects of both hanger loads and cut rebar were considered simultaneously in the evaluations, as was discussed also in Element Report 215.2-SQN, Rev. 1. The implementation of the pre-restart CAP is complete, and the assessment of concrete structural elements was based on the ultimate strength method specified in design criteria SQN-DC-V-1.3.3.1, considering the combination of dead, live and FSAR OBE or SSE seismic loads. The NRC staff's evaluation included a walkdown of the plant and an audit of representative samples of TVA's implementation results. The scope and implementation of the pre-restart CAP items were found acceptable. The assumption regarding the similarity in percentage of cut rebar between the SQN and WBN auxiliary buildings and the assessment of the concrete structural elements were found acceptable as discussed in the staff safety evaluation for Element Report 215.2-SQN, Revision 1.

Regarding the post-restart CAP, TVA has committed to (1) revise Section 3.8 of Sequoyah FSAR to clarify the use of the ultimate strength method from design criteria SQN-DC-V-1.3.3.1 for the structural integrity assessment of the reactor building and auxiliary building, and (2) perform the long term evaluation of Category I concrete walls not included in the pre-restart assessments, based on the program plan developed in pre-restart CAP item (3). The NRC staff found the scope of the the post-restart CAP to be sufficient.

IV. Conclusions

The NRC staff reviewed TVA's investigation of the employee concerns and the CAP developed by TVA to resolve such concerns, and found they were adequate. TVA's implementation of the pre-restart CAP was also acceptable. The NRC staff therefore believes TVA's resolution for the concerns as described in Element Report 215.6-SQN, Rev. 1, is acceptable.

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# SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2

# SAFETY EVALUATION REPORT FOR EMPLOYEE CONCERN

# ELEMENT REPORT 215.9 (B), STRUCTURAL STEEL CONNECTION DESIGN"

## I. Subject

Category: Engineering (20000)
Subcategory: Civil/Structural Design (21500)
Element: Structural Steel Connection Design (21509)

The basis for Element Report 215.9 (B) Revision 1, dated January 13, 1987 is employee concern IN-85-297-003 which states:

> "Structural steel connections (I-beam to embed plates) are both welded and bolted. One method is for vibration and the other is for dead loads. Both type connections are being used on the same I-beam and these _are not supposed to be mixed'. Construction Dept. concern. CI declined to provide futher information."

This concern was evaluated by the licensee as potentially nuclear safety-related and potentially applicable to Sequoyah (generic). A similar concern was investigated under Sequoyah Element Report 222.5(B) entitled "Pipe Support Weld Design - Bolts Replaced by Weld."

## II. Summary of Issues

One issue was defined by the licensee:

> Bolted and welds are used in the same connection to transfer loads from structural steel members to concrete walls. These are not supposed to be mixed.

## III. Evaluation

The employee's concern about mixed connections was evaluated as related to welding and bolting at the same connection. There is nothing unusual about a welded connection at one end of a beam and a bolted connection at the other end.

The FSAR and design criteria for the Sequoyah Nuclear Plant commit the licensee to design structural steel in accordance with the American Institute of Steel Construction (AISC) code. Section 1.15.10 of the AISC code contains the design criteria for the use of bolts in combination with welds:

> "A-307 bolts, or high strength bolts used in bearing-type connections shall not be considered as sharing the stress in combination with weld. Welds is used shall be provided to carry the entire stress in the connection."

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The intent of this rule is that the relatively rigid weld will carry the shear load when connections have a combination of bolts and welds. According to this rule, replacing one bolt in a four bolt connection with a weld means that the weld must accommodate the shear stress formerly taken by all four bolts.

However, the licensee also creates the potential for using a weld of insufficient size with Note 2 on Drawing 47A050-2 Revision 5 which states:

> "Mechanical Hangar Drawings - General Notes - Note 2 - Where a bolt anchored plate overlaps an existing embedded plate, bolt anchors may be replaced by a minimum of 2 in. of 5/16 in. weld for each bolt eliminated due to some portion of the bolt hole being obstructed by the embedded plate. Engineering design approval is required. This note is not applicable to 1-1/4 in. wedge bolts or to any size of grouted anchors."

Significant Condition Report SQN CEB 8601 noted that this note had existed since 1978, but design calculations to justify this statement did not exist. Similar notes are found in the licensee's drawing series 47A051, 47A052, 47A054, 47A055 and 47A056 which cover general notes for all seismic category 1 support structures for piping, electrical conduits and trays, HVAC ducts, and instrument tubing. ' licensee review found several structural steel supports with these mixed connections, e.g., supports for a large duct at elevation 710 that circles the reactor cavity wall.

The licensee submitted recent licensee calculations which concluded that all systems will be able to perform their intended functions and no failures would occur as a result of the drawing note. The licensee's evaluation team observed that the loads were shared by the welds and bolts contrary to AISC rules and there were several cases where the yield stress of the weld was exceeded based on allowable design stresses. In addition, the shear strength of the base metal was not considered in determining the load-carrying capacity of the weld. While this is a programmatic error, the NRC staff noted that it has a negligilbe effect for these welded base plates.

The licensee's evaluation team substantiated the employee concern that welds and bolts are used on the same connection. The team also found that the calculations do not demonstrate licensee conformance to FSAR commitments.

For corrective action, Sequoyah randomly selected 60 baseplates with mixed connections that represent the structures throughout the plant. These baseplates were analyzed by considering all of the shear forces applied to the baseplate as acting on the weld or welds. Policy memorandum PM-86-17 was issued to provide instructions for designing these mixed connections and to prevent a reoccurrence of this type of problem. The plan was to strengthen deficient welds, but none were found based on actual loads. Since no welds were deficient, TVA claimed a 95% confidence level inthe integrity of these types of connections at Sequoyah.

IV. Conclusions

The NRC staff believes that the licnesee's investigation of the concern was adequate, and their resolution of the concern as described in TVA Employee Concerns Special Program Report Number 215.9 (B) Revision 1 dated January 13, 1987, entitled "Civil/Structural Design - Structural Steel Connection Design" is acceptable for Sequoyah. The licensee admitted that the expansion anchors are designed to carry shear loads with welds contrary to AISC code requirements. Sequoyah performed a random sampling program of 60 baseplates and performed a stress analysis based on the as-measured dimensions of the connections. No connections required weld strengthening and the sample gave a 95% probability that less than 5% of all of these connections at Sequoyah may need strengthening. The licensee issues a policy memorandum to provide instructions for designing these mixed connections and to prevent a reoccurrence of this type of problem.

Technical Contact: Paul Cortland (301) 492-8734

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# SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

## EMPLOYEE CONCERNS PROGRAM

## TENNESSEE VALLEY AUTHORITY

## SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

## DOCKET NOS. 50-327 AND 50-328

## I. INTRODUCTION

The issues addressed in this Safety Evaluation Report (SER) for the Sequoyah Nuclear Plant (SQN) are in the civil/structural and pipe support design areas. This report provides an evaluation of 2 individual concerns categorized in the following 2 element and/or subcategory reports:

| ELEMENT/SUBCATEGORY | DESCRIPTION |
|---|---|
| 21510/25000 | Feedwater Heater Monorail Design |
| 22110/22100 | Use of Snubbers |

If determined to be valid these issues must be resolved for the Sequoyah Plant.

## II. EVALUATION

The NRC consultant, Parameter, Inc., has reviewed the 2 employee and/or subcategory reports and prepared the attached Technical Evaluation Reports (TER).

The staff has reviewed the TERs and concurs in their bases and findings. There were no allegations identified during the review pertinent to those reports.

Those elements that were initially submitted as non-restart justification issues were reviewed as part of a sub-category report. The review included the evaluation of the employee concerns as well as addressing the SQN restart issue.

Where corrective action has been warranted, the staff's acceptance is based upon satisfactory fulfillment of all commitments as described in the TVA corrective action plan.

## III. CONCLUSION

Based on the staff review of the attached TERs relating to the employee concerns program for SQN, the staff concludes that TVA has adequately addressed the employee concerns and that their conclusions and corrective actions are acceptable.

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Certain corrective actions have been implemented for SQN Unit 2 only. It is the responsibility of TVA to assure that acceptable implementation of such corrective action will be performed for Unit 1. Any additional program changes should be submitted for staff review and should not be implemented prior to review and approval by the staff.

SEQUOYAH NUCLEAR POWER PLANTS, UNITS 1 AND 2

TECHNICAL EVALUATION REPORT FOR EMPLOYEE CONCERN ELEMENT REPORT 21510(B), "FEEDWATER HEATER MONORAIL HANGER DESIGN"

I. SUBJECT

| | |
|---|---|
| Category: | Engineering (20000) |
| Subcategory: | Civil/Structural Design (25000) |
| Element: | Feedwater Heater Monorail Hanger Design (21510) |

The basis for Element Report 21510(B), Rev.0, 12/2/86 is employee concern LDA-86-001, which questions the structural integrity of the feedwater heater monorail hangers.

II. SUMMARY OF ISSUE

The structural integrity of hangers for the feedwater heater monorails located in the turbine building is questionable.

III. EVALUATION

TVA subcategory report 25000, Rev.2, 10/26/87, and TVA element report 21510(B), Rev.0, 12/2/86, identify the issue as not safety related because of the monorail function and location within the turbine building. The TVA reports also identify the issue as not valid. The chronology of events affecting this issue is given as follows:

-The concern was expressed orally on or before August 5, 1985.

-On August 6, 1985, the concerned employees met with the TVA design engineer who explained the design approach and details of the monorail hangers. In a statement documenting the meeting, it is recorded that the employees expressed satisfaction and gave their assent to closing the issue.

-A TVA structural engineer made an independent review of the feedwater heater monorail design on August 13, 1985 and affirmed its adequacy.

-A 3rd party review was made on August 19, 1985 by Impell Corporation, which confirmed the design as adequate.

-The scope of these reviews and the conclusion reached are stated within subcategory report 25000 as: "The design calculations and drawings were reviewed for assumptions, logic, analysis, code interpretations, member selections, connections, and clarity of presentations. The evaluation team found the design documents well organized, complete, and meeting the AISC requirements."

-TVA performed a load test of the system on August 25, 1985, using a load 40% heavier than the operating load to be carried. The test was successful.

## IV. CONCLUSION

TVA evaluation, action, and resolution of the expressed concern is adequate and acceptable for Sequoyah Units #1 and #2 restart.

SEQUOYAH NUCLEAR POWER PLANTS, UNIT 2

TECHNICAL EVALUATION REPORT FOR EMPLOYEE
CONCERN ELEMENT REPORT 22110(B), "USE OF SNUBBER"

## I. SUBJECT

| | |
|---|---|
| Category: | Engineering (20000) |
| Subcategory: | Pipe Support Design (22100) |
| Element: | Use of Snubber (22110) |

The basis for Element Report 22110(B), Rev.1, 12/30/86, is employee concern SQN-86-001-02 which states that the Upper Head Injection System vertical riser requires a rigid support where a snubber was used.

## II. SUMMARY OF ISSUE

A rigid type support is specified in the piping analysis for a specific location on the vertical riser of the Upper Head Injection (UHI) system, but the detail drawings and as-built condition show use of a snubber at this location. UHI has a plant safety-related function.

## III. EVALUATION

TVA element report 22110(B),Rev.1, 12/30/86 recognized the employee concern as valid. In a subsequent letter J.A. McDonald (TVA) to B.J. Youngblood (NRC), 2/17/87, responding to an NRC request for additional information, the root cause of this disparity between the pipe support analysis and the as-built condition was given as a lack of attention to detail, specifically, that an engineering judgement was made regarding support orientation and design without proper documentation and communication to interfacing groups. The letter also identified a 100% engineering review of all snubbers in the plant against the piping analyses, and confirmed this instance to be a single, isolated case. The report indicates that TVA re-analysis of the UHI pipe restraint at this location utilizing a snubber demonstrated the use of the snubber to be an adequate design, able to sustain required seismic and thermal stress levels. The TVA evaluation identified this as an acceptable resolution in the report, but also described TVA's decision and commitment to replace the snubber type support with a rigid type support prior to re-start. TVA recognizes the necessity to fulfill applicable requirements of design control and configuration control of 10 CFR 50, Appendix B, Criterion III and ANSI N 45.2.11 in TVA work performed to SQN Pipe Support Design Manual (PSDM), Volume III.

The depth and extent of the evaluation team review of this issue is adequate, including identification of root cause of the problem. Corrective actions regarding both the engineering design activities and replacement of the pipe support are adequate.

The replacement action has been tracked under Corrective Action Tracking Document (CATD) 22110 SQN01, and is reported as completed and verified for Sequoyah Unit #2 only, on 8/27/87.

IV. CONCLUSION

TVA evaluation and resolution of this employee concern is adequate, acceptable and appropriate for Sequoyah Unit No. 2 restart.

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# SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2

# SAFETY EVALUATION REPORT FOR EMPLOYEE CONCERNS

# ELEMENT REPORT 218.1(B), REVISION 1

# "PIPE STRESS CALCULATIONS

# THERMAL ANALYSIS OF PIPING

# SUBJECTED TO TEMPERATURE LESS THAN 120°

## I. Subject

| | |
|---|---|
| Category: | Engineering (20000) |
| Subcategory: | Pipe Stress Calculations (21800) |
| Element: | Thermal Analysis of Piping Subjected to Temperature Less than 120°F (21801) |
| Concerns: | SQN-86-002-03, SQN-86-001-03, IN-85-038-001, IN-85-039-001, IN-85-039-002 |

The bases for Element Report 21801, Revision 1 dated December 19, 1986 are Employee Concern Nos. SQN-86-002-003, SQN-86-001-03, IN-85-038-001, IN-85-039-001 and IN-85-039-002, which question the thermal analyses of piping performed by TVA.

## II. Summary of Issues

The Employee Concerns Task Group (ECTG) report identified the following six issues from the employee concerns:

a. Current operating mode drawings were not used for all subsequent analyses.

b. Site group stress analysts were not allowed to evaluate the significance of the current operating mode definitions in the analysis of record.

c. The environmental temperature in the annulus area may reach 150°F but site group stress analysts were not allowed to evaluate the effect of the environmental temperature on piping in that area.

d. The operational mode procedure does not require evaluation of previously performed thermal analyses when thermal conditions change.

e. Not all stress-analyzed piping included a code-required evaluation of thermal expansion.

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f. Excessive levels of pipe support loads and pipe stress due to thermal expansion have been observed for some piping where the system operating temperatures were between 40°F and 120°F and a thermal expansion evaluation was not performed.

III. Evaluation

A technical review of Employee Concerns Element Report 218.1(B), Revision 1 was performed by NCT Engineering, Inc. under NRC Contract No. 05-86-156. The results of this review are summarized in the attached NCT technical evaluation report dated November 30, 1987 on Employee Concerns Element Report 218.1(B), Revision 1.

Element Report 218.1(B), Revision 1 found that only issue d contained a valid concern for rigorously analyzed piping systems. The report further stated that based on the results of a sampling program at Watts Barr, the thermal operating modes used for Sequoyah were adequate and no corrective actions were required. The report referred to Element Report 218.4(B) for evaluation of issues e and f for alternate analysis piping. The NCT review of Element Report 218.1(B), Revision 1 found that the ECTG evaluations of issues a, b, c and d was acceptable. Based on a finding by the NRC's Integrated Design Inspection (IDI) the NCT report concluded that issues e and f should remain open until the IDI finding is resolved. The staff concurs with the conclusions presented in the NCT technical evaluation report.

The NCT technical evaluation report identified that an additional item has been raised by the ECTG based on a revised version of an employee concern. This new issue has not been transmitted to the NRC and has not been reviewed. The NCT evaluation also identified that TVA committed to issue new operating mode drawings for all Unit 2 piping systems and recommended that this effort be completed in a timely manner. TVA's implementation of this commitment to evaluate operating mode drawings should be reviewed as a post restart item for both Units 1 and 2.

IV. Conclusions

Based on the review of Employee Concerns Element Report 218.1(B), Revision 1 the staff concludes that Employee Concerns SQN-86-002-03, SQN-86-001-03, IN-85-038-001, IN-85-039-001 and IN-85-039-002 have been, in general, adequately addressed for rigorous piping analyses for Sequoyah restart. Final resolution of these concerns is contingent on the resolution of the NRC's Integrated Design Inspection finding on the ERCW thermal analysis. Alternately analyzed piping is addressed in the evaluation of Element Report 218.4(B). In addition, the new issue raised by ECTG should be reviewed by the NRC staff prior to the Sequoyah restart to determine whether the new issue has any impact on the conclusions of this evaluation. TVA's implementation of the commitment to issue new operating drawings should be reviewed by the staff as a post restart item.

V. Addendum

See next page.

V. Addendum (continued)

The safety evaluation report for this element report specified the review of the revised employee concerns report as a restart item. This element report was revised based on the identification of an additional technical concern by the employee concerns task group. The additional issue involved TVA's failure to consider secondary stress range for alternate analysis at Watts Bar. The resolution of this item for Sequoyah as described in Element Report 218.1, Revision 2, is to address the issue in the Phase II alternate analysis program. The Phase II alternate analysis program will be performed after the Sequoyah Unit 2 restart. The resolution of secondary stresses due to stress range considerations in the post restart effort is consistent with the staff safety evaluation on alternately analyzed piping and is acceptable.

SEQUOYAH NUCLEAR POWER PLANT, UNIT 2
TECHNICAL EVALUATION REPORT FOR EMPLOYEE CONCERNS
ELEMENT REPORT 218.1(B), REVISION 1
"PIPE STRESS CALCULATIONS
Thermal Analysis of Piping Subjected to Temperature Less than 120 F"

SUBJECT: This report summarizes the NRC audit of TVA investigation of SQN piping operating mode identification, control and evaluation concerns.

By: Robert E. Serb
Consultant
NCT Engineering, Inc.

Date: November 30, 1987

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NCT ENGINEERING, INC.

3850 MT. DIABLO BLVD. P.O. BOX 1937 LAFAYETTE CA 94549 (415) 283-0471

SEQUOYAH NUCLEAR POWER PLANT, UNITS 1 & 2
TECHNICAL EVALUATION REPORT FOR EMPLOYEE CONCERNS
ELEMENT REPORT 218.1(B), REVISION 1
"PIPE STRESS CALCULATIONS
Thermal Analysis of Piping Subjected to Temperature Less than 120 F"

I. Subject

Category: Engineering (20000)
Subcategory: Pipe Stress Calculations (21800)
Element: Thermal Analysis of Piping Subjected to Temperature Less than 120 F (21801)
Concerns: SQN-86-002-03, SQN-86-001-03, IN-85-038-001, IN-85-039-002

The bases for Element Report 21801, Revision 1 dated December 19, 1986 are Employee Concern Nos. SQN-86-002-03, SQN-86-001-03, IN-85-038-001, IN-85-039-001 and IN-85-039-002. The concerns regard the identification, control, and pipe stress analysis of pipe system operating and design modes (op modes) for TVA nuclear sites and are potentially applicable to the Sequoyah Nuclear Power Plant (SQN).

II. Summary of Issue

The ECTG report has translated the concerns into the following six issues:

a. Current operating mode drawings were not used for all subsequent analyses.

b. Site group stress analysts were not allowed to evaluate the significance of the current operating mode definitions in the analysis of record.

c. The environmental temperature in the annulus area may reach 150 F but site group stress analysts were not allowed to evaluate the effect of the environmental temperature on piping in that area.

d. The operational mode procedure does not require evaluation of previously performed thermal analyses when thermal conditions change.

e. Not all stress-analyzed piping included a code-required evaluation of thermal expansion.

f. Excessive levels of pipe support loads and pipe stress due to thermal expansion have been observed for some piping where the system operating temperatures were between 40 F and 120 F and no thermal expansion evaluation was performed.

## III. Evaluation

Issue "a"

In order to assure that correct op mode data have been used at SQN, the ECTG reviewed the incorporation of op mode data in five calculations and for one of these verified that the proper data was evaluated in the piping analysis. The ECTG review identified no instances in which erroneous data was used. Although the ECTG identified no op mode data errors, the concern remains that some piping analyses performed prior to initiation of op mode drawing development may not have used correct operating temperatures. This issue as well as procedural aspects of op mode data development, control, and application are addressed in the discussion of Issue "d" below.

Issue "b"

To investigate the concern that analysts were not allowed to evaluate "currect" op modes, the ECTG separately interviewed four TVA pipe stress analysts. All four analysts said they were not aware of any instance in which anyone was instructed not to use current op mode drawings.

Issue "c"

The ECTG identified TVA Drawing 47E235-47, Revision 3 as the basis of containment annulus temperatures and that drawing was reviewed during the NRC audit of this concern. The drawing defines the SQN containment annulus maximum normal and abnormal environmental temperatures as 120 F. That drawing also defines a maximum faulted condition environmental temperature of 134 F. SQN FSAR Table 3.9.2-2 specifies Safety Class B,C & D component support load combinations. That table does not specify a thermal load for the faulted load condition. FSAR Table 3.9.2-3 specifies stress limits for Safety Class B,C & D component load conditions and states that expansion stresses need not be evaluated for the faulted condition. The FSAR is consistent with the requirements of the ASME Section III code which does not require evaluation of piping thermal expansion for plant faulted conditions. Therefore, the ECTG conclusion that annulus area piping environmental condition thermal analyses should be based on the normal/abnormal maximum 120 F temperature is considered aceptable. Evaluation of pipe thermal expansion due to the normal/abnormal 120 F annulus and other temperature conditions is addressed in the discussion of Issue "e" of this report.

Issue "d"

The ECTG reviewed the Operating and Design Modes section of the SQN Rigorous Analysis Handbook, Section No. SQN-RAH-207 to verify that TVA procedures require piping design reevaluation when new op mode data is issued. That procedure section requires design and operating modes to be issued in accordance with Mechanical Design Guide DG-M5.1.1. It also requires op mode information to be issued on drawings which must be referenced on all new analysis and reanalysis problem isometrics. Therefore, consideration of current op modes has been proceduralized for analyses performed subsequent to issuance of the handbook.

In 1984, NCR SQNCEB8205 addressed three areas of nonconformance regarding issuance, revision control, and incorporation of op mode data in pipe analyses for analyses performed prior to the implementation of SQN-RAH-207. The NCR identified the requirement to include op mode handling methodology in the Rigorous Analysis Handbook. The adequacy of op mode data used in analyses prior to the use of op mode drawings was also addressed in the NCR. The data was determined to be adequate based on the the following as excerpted from the NCR:

- WBN [Watts Bar], a Westinghouse sister plant to SQN, is a section III plant. Availability of operating modes and design transients for WBN has resulted in a better understanding and better analysis of SQN systems than would normally be expected for that vintage plant.

- Operating temperature and pressures were furnished by Westinghouse and systems engineers. As indicated the controlling equipment for these data was firmed up very early, and significant changes have been evaluated.

- In an effort to further ensure operating mode adequacy for SQN, a commitment was made to review any discrepancies determined by the WBN sampling program for their effect on SQN operating modes. A sample of WBN problems was decided in part because the more detailed consideration of operating modes at WBN would have turned up nonconforming items in some cases where the SQN procedures would not.

- The WBN sampling program was performed and a WBN report, NCR WBNCEB8215 R5 (revised final), was completed March 7, 1984. Only one problem was located for WBN (CEB 831025 003) and was immediately evaluated for SQN. The problem was determined to be qualified for SQN (PWP 831202 016). No other discrepancies were noted. Therefore, the adequacy of the consideration of SQN operating modes has been assured.

Based on the results of the WBN sampling program evaluations which demonstrated acceptable piping stresses, the ECTG concluded that thermal operating modes had been adequately defined for SQN.

During the NRC audit of employee concerns TVA commited to issue op mode drawings for all Unit 2 rigorous analysis problems by December 1988. This was documented in the undated TVA Memorandum from W.J.Kagay to Rick Daniels; SEQUOYAH NUCLEAR PLANT - UNIT 2 - GENERATION OF OP MODE DRAWINGS FOR RIGOROUS PIPING ANALYSIS PROBLEMS. Per that memo, the potential effect of op mode drawing data on piping problems will be evaluated and the problems reanalyzed as required.

Issue "e"

Consideration of thermal expansion for SQN Alternate Analysis scope piping is addressed by TVA Element Report 218.4(B), Revision 2. Apparently, thermal expansion was not always considered and, therefore, this concern is valid for alternate analysis scope piping. TVA has developed an alternate analysis review program to address deficiencies in alternately analyzed piping and this program has been the subject of other NRC review. The discussion which follows is therefore limited to TVA rigorous analysis scope piping.

TVA Design Criteria No. SQN-DC-V-13.3, Detailed Analysis of Category 1 Piping Systems, dated March 10, 1975 requires piping systems analyses to consider design and operating conditions. The TVA memorandum from R.O.Barnett to CEB Files dated March 20, 1987, SEQUOYAH NUCLEAR PLANT (SQN) - DESIGN INPUT MEMORANDUM FOR DETAILED ANALYSIS OF CATEGORY 1 PIPING SYSTEMS, SQN-DC-V-13.3 specifies that thermal ranges for limiting operating modes are considered. In addition, SQN Rigorous Analysis Handbook, Section SQN-RAH-207 requires evaluation of all op modes. Therefore, TVA piping analysis procedures do require evaluation of thermal expansion for rigorously analyzed piping.

The NRC Integrated Design Inspection (IDI) identified a deficiency in the implementation of these procedures. An NRC letter to Mr. S.A.White dated October 9, 1987, "ITEMS IDENTIFIED BY THE INTEGRATED DESIGN INSPECTION REQUIRING RESOLUTION PRIOR TO RESTART OF SEQUOYAH UNIT 2" identifies a "Draft Deficiency" regarding the failure to include a 35 F cold thermal mode in the op modes defined for an ERCW piping problem. Evaluation of TVA operating mode procedure implementation should remain open pending resolution of that deficiency.

Issue "f"

Consideration of operating temperatures between 40 F and 120 F for SQN Alternate Analysis scope piping is addressed by TVA Element Report 218.4(B), Revision 2. Concern regarding consideration of all operating condition thermal expansion effects for rigorously analyzed piping is addressed at Issue " e" above.

IV. Conclusions

Based on review of the ECTG report and TVA criteria for rigorous analysis piping system qualification, the ECTG evaluation of Issues "a", "b", and "c" is considered acceptable.

TVA has commited to issue operating mode drawings for all Unit 2 piping systems. This should assure evaluation of these piping systems for current operating modes. It is recommended that evaluation of the drawing data and performance of resulting reanalyses, if any, be completed in a timely manner. The ECTG evaluation, and the op mode drawing development and evaluation are sufficient to consider Issue "d" evaluation acceptable.

The IDI dificiency regarding failure to consider the correct operating temperature affects conclusions regarding Issues "e" and "f". The evaluation of these issues will be considered acceptable upon resolution of the IDI finding.

During the NRC audit of these concerns, an additional issue was identified by the ECTG based on receipt of a revised version of an employee concern. Therefore, additional review of this concern will be necessary when the revised ECTG evaluation is completed.