TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401

5N 157B Lookout Place

JUL 27 1990

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - NRC INSPECTION REPORT NOS. 50-327/90-18 AND 50-328/90-18

As requested in the subject inspection report, this letter submits TVA's response to the unresolved items (URIs) which remain open from Inspection Report Nos. 50-327/88-12 and 50-328/88-12. Although the responses in Enclosure 1 are titled by the 88-12 URI numbers, they address the specific concerns raised by the subject inspection report.

With the exceptions of design basis accident and zero period acceleration effects, component damping values, and seismic analysis of the steel containment vessel, TVA is in agreement with NRC on the scope of work required to address the concerns. For these three issues, additional information is provided to support TVA's position.

Summary statements of commitments contained in this submittal are provided in Enclosure 2. Please direct questions concerning this issue to Kathy S. Whitaker at (615) 843-7748.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. H. Shell

for E. G. Wallace, Manager
Nuclear Licensing and
Regulatory Affairs

Enclosures
cc: See page 2

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U.S. Nuclear Regulatory Commission

JUL 27 1990

cc (Enclosures):

Ms. S. C. Black, Deputy Director
Project Directorate II-4
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

Mr. J. N. Donohew
Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

Mr. B. A. Wilson, Project Chief
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE 1

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

U-Bolt and Unistrut Allowables

TVA has completed the study to confirm that the Category I pipe support configurations installed at Sequoyah Nuclear Plant (SQN) that use U-bolts meet the allowable loads tabulated in Design Standard DS-C1-6.13. The 185 Unit 1 installations have been evaluated and qualified to the standard. The 323 Unit 2 installations have been evaluated, and 299 have been qualified to the standard. The 24 unqualified U-bolt support configurations will be modified before restart from the Unit 2 Cycle 5 refueling outage.

A study has also been completed to confirm that the Category I pipe support configurations installed at SQN that use Unistrut clamps meet the allowable loads tabulated in Design Standard DS-C1-6.14. The 90 Unit 1 installations have been evaluated, and 85 have been qualified to the standard. For Unit 2, the 262 installations have been evaluated, and 237 have been qualified to the standard. The five unqualified Unit 1 Unistrut clamp support configurations will be modified before restart from the Unit 1 Cycle 5 refueling outage. The 25 unqualified Unit 2 Unistrut clamp support configurations will be modified before restart from the Unit 2 Cycle 5 refueling outage.

Pre-NF (Not Manufactured to the Requirements of Subsection NF of the American Society of Mechanical Engineers [ASME] Code) Mechanical Snubbers

TVA has determined that the current faulted load capacity of 2.0 for pre-NF snubbers will be reduced to 1.33. For Unit 1, walkdowns must be completed during the Cycle 5 refueling outage to identify pre-NF snubbers. After a review to determine which snubbers do not meet the 1.33 capacity factor, TVA will submit a schedule for the necessary Unit 1 modifications six weeks after restart from the Unit 1 Cycle 5 refueling outage. Five pre-NF mechanical snubbers not meeting the 1.33 capacity factor have been identified in the initial review for Unit 2. The necessary modifications will be completed for Unit 2 before restart from the Unit 2 Cycle 5 refueling outage.

Post-NF Mechanical Snubbers

The appropriate vendor-certified load capacity data sheets were used for post-NF mechanical snubbers installed at SQN. Attachment 1 to this enclosure is Vendor Figure BE-416N; it was used by TVA and provides rated loads for NF mechanical snubbers. With the exception of Model Number PSA-1, all allowables are lower than the ASME Section III allowables. The BE-416N allowable for PSA-1 is 2,320 pounds versus the ASME Section III allowable of 2,300 pounds. This difference is less than 1 percent, which is considered insignificant. (Figure BE-416N does not provide a faulted allowable for PSA-100, but there are no safety-related PSA-100 snubbers in Category 1 structures at SQN.)

URI 88-12-03, Design Basis Accident/Zero Period Acceleration (DBA/ZPA) Effects

Prior to the restart of SQN Unit 2, TVA contracted with Bechtel North American Power Corporation to generically address the NRC concern related to the ZPA effect on piping for the containment DBA analysis. Bechtel responded to this concern by performing a study of five piping analysis problems of lines attached to the steel containment vessel (SCV). The attributes and bases used to select piping analyses for review were described in Reference 1. Each of the attributes was selected to not only obtain a sample representative of the population of piping attached to the SCV, but also a representation of the most severely affected piping. The results of the study demonstrated that the sampled piping, supports, and SCV penetrations are still acceptable when the DBA/ZPA effects are considered. Based on this study, TVA concluded that a high degree of confidence had been provided that other piping systems not included in the sample would likewise be acceptable and no additional evaluations were required. TVA committed to incorporate DBA/ZPA effects into future piping reanalysis.

During the 90-18 inspection, NRC challenged TVA's DBA/ZPA study in two areas; (1) the representativeness of the original sample and (2) the high design load-to-allowable load ratio for supports on SCV attached piping.

The first general area questioned by NRC related to the representativeness of the original five-problem study; specifically, NRC questioned the exclusion of the hydrogen collector (HC) and containment spray (CS) piping from the original sample since support modifications were identified or expected from inclusion of DBA/ZPA effects.

TVA has reviewed the process used in the development of the original five-problem sample study. The sample selection was focused on selecting representative piping having worst-case attributes. The HC and CS piping are the only piping systems rigidly attached to the SCV greater than 6 inches in diameter and the only piping systems with extended, long axial runs. The HC and CS piping are thus not representative of SCV attached piping. Inclusion of the HC and CS piping would have inappropriately skewed the conclusions of the sample evaluations. Also, each of these uniquely configured piping systems were identified to require full reanalysis, incorporating DBA/ZPA and revised vertical spectra effects. TVA concludes that the apparent basis for excluding the HC and CS from the Bechtel five-problem sample is appropriate. To more completely document this conclusion, TVA plans to amend the sample study report to address exclusion of the HC and CS piping.

To give additional confidence and credence to the representativeness and conclusions of the original sample, TVA contracted with Gilbert Commonwealth, Incorporated (G/C) to undertake a summary study of the 21 SCV attached piping problems formally reanalyzed since 1988. These piping problems, selected because DBA/ZPA and revised vertical spectra effects were included in their reanalysis, were performed to document design changes and to resolve conditions adverse to quality. The effects of revised vertical SCV spectra, discussed in URI 88-12-10, were considered jointly with DBA/ZPA effects since the scope of piping is the same for both issues.

The purpose of the sample study was to conclusively determine if modifications of pipe supports resulting from the reanalysis were caused by the inclusion of DBA/ZPA and vertical spectra effects. Modifications were first reviewed to determine if conservatism in the original support evaluation could be removed and the status of the support changed to no modification required. If this review determined a support modification was still required, the piping was reanalyzed without the DBA/ZPA and revised vertical spectra effects. This reanalysis was then compared to the existing analysis with DBA/ZPA and revised vertical spectra effects included. Pipe support load changes were assessed and modifications reviewed to determine if load changes beca f DBA/ZPA and vertical spectra effects caused the modification. A summary the scope, methodology, and results of this review is provided in Attach it 2 to this enclosure. G/C has concluded that the 21 reanalyses of SCV attached piping since 1988 resulted in no pipe support modifications driven by DBA/ZPA and revised vertical spectra effects. This summary study substantiates the original five-problem study and provides additional evidence that the DBA/ZPA and revised vertical spectra effects do not have a significant potential to impact existing qualification of pipe supports on SCV attached piping.

The second area questioned by NRC addressed the high design-to-allowable load ratio of pipe supports adjacent to the SCV. TVA acknowledges that the current design margin of supports near the SCV are potentially low and that support margin was not a sample attribute in the original Bechtel five-problem study. TVA, however, considers the original study results to still be valid. This conclusion is based on the following:

1. The G/C summary study of 21 reanalyses concluded that the DBA/ZPA and revised vertical spectra effects did not significantly impact the existing pipe support design.

2. Current pipe support design margins are normally based on conservative evaluation techniques. Refined support analytical methods generally provide increases in support design margin.

3. Additional pipe support design margin is available if the benefical effects of the leak-before-break (LBB) technology is included in the pipe support qualification. To obtain an indication of this potential load reduction, TVA contracted with G/C to evaluate the piping analyses in the 21-problem study with and without LBB DBA spectra. The results of this study indicate an average decrease in support load of 23 percent.

4. The SQN design bases criteria requires pipe stress and support evaluations consider the simultaneous occurrence of a DBA and a design basis earthquake. The maximum load effects of these two extremely low probability events must be individually determined and combined absolutely. A more reasonable approach to account for time phasing of the events and probability of event occurrence would be to combine the effects of these two events by the square root of the sum of the squares (SRSS) method. The G/C 21-problem study determined that using the SRSS load combination in lieu of the absolute combination resulted in an average support load reduction of 13 percent.

5. The standard linear response spectrum modal analysis used for evaluation of rigorous piping at SQN has been shown to be from 11 to 42 percent more conservative than linear elastic time-history analysis as described in Reference 2.

   In Reference 3, the effects of nonlinearities present in pipe supports are combined with an evaluation of the response spectra versus time-history methods. This study indicated even more pronounced pipe support load reductions.

6. Dynamic testing of piping systems initiated by Electric Power Research Institute (EPRI) and NRC is described in Reference 4. The authors conclude that current code rules based on static collapse load concepts for dynamic load considerations are overly conservative. Other reports of dynamic testing of piping systems under dynamic loadings, such as those in References 5 and 6, conclude that piping systems and supports are capable of withstanding several times their rated capacities under dynamic loadings.

Considering the significant conservatisms noted above in the current pipe support load development and the conclusions from the sample studies, pipe supports with high interaction ratios on SCV attached piping do not represent a safety concern at SQN.

References

1. TVA letter to NRC dated March 30, 1989, "Sequoyah Nuclear Plant (SQN) Units 1 and 2 - NRC Inspection Report Nos. 50-327/88-12 and 50-328/88-12 - Response to Unresolved Item (URI) 88-12-03"

2. "Comparison of LMFBR Piping Response Obtained Using Response Spectrum and Time-History Methods" by G. M. Hubert, presented at the ASME PV&P Conference in Dever, Colorado in June of 1981.

3. "Seismic Analysis of Piping with Nonlinear Supports" by D. A. Barta, et al, ASME PV&P Conference, August 1980.

4. "Seismic Analysis and Testing of Piping Systems and Components" by S. W. Taggart, et al, PVP-Volume 144.

5. "Experimental Study of Piping Stability During Strong Earthquakes" by N. Ogawa, et al, PVP-Volume 150.

6. "High Level Seismic Tests of a Piping System at the HDR Facility" by L. Malcher, et al, PVP-Volume 182.

URI-88-12-04, Containment DBA Spectra

In the referenced letter, TVA provided the requested additional information concerning (1) the steel containment vessel orthotropic elastic properties, (2) the theoretical basis for the double differentiation technique, (3) the existence of a spectral peak beyond 10 Hertz, and (4) the maximum steel containment vessel response beyond 0.9 seconds.

Reference

TVA letter to NRC dated June 11, 1990, "Sequoyah Nuclear Plant (SQN) Units 1 and 2 - NRC Inspection Report Nos. 50-327, 50-328/88-12 - Unresolved Item (URI) 88-12-04."

URI-12-05, Essential Raw Cooling Water (ERCW) Pumphouse

As committed in References 1 and 2, TVA will complete the limited exploration program by September 1, 1990. The satisfaction of the design requirements for the operating basis earthquake (OBE) concurrent with water level at Elevation 704 will be confirmed as committed in Reference 3.

References

1. TVA letter to NRC dated December 28, 1988, "Sequoyah Nuclear Plant (SQN) - Essential Raw Cooling Water (ERCW) Pumphouse Foundation and ERCW Pumping Station Access Cells"

2. TVA letter to NRC dated March 1, 1990, "Sequoyah Nuclear Plant (SQN) - Essential Raw Cooling Water (ERCW) Pumphouse Foundation and ERCW Pumping Station Access Cells"

3. NRC letter to NRC dated March 3, 1988, "Sequoyah Nuclear Plant - Essential Raw Cooling Water (ERCW) Pumping Station Concrete"

URI 88-12-08, Component Damping Values

During the 90-18 inspection it was agreed that the remaining NRC concern for URI 88-12-08 was limited to mechanical components satisfying the following parameters.

1. Required for safe shutdown as defined for the 1979-1982 SQN seismic-margins review.

2. Welded construction.

3. Located in buildings listed in the Final Safety Analysis Report (FSAR) Table 3.7.1-1.

4. Non-rigid (fundamental structural frequency less than 25 Hertz).

5. Floor/wall mounted (not line mounted).

6. Seismically analyzed after March 1982 using safe shutdown earthquake (SSE, SQN's design basis earthquake) damping values greater than 2 percent of critical.

These limiting parameters were established based on the recognition that NRC's concern relates specifically to the SQN seismic-margins review for mechanical components. This review was completed in March 1982 and is briefly discussed in Supplement 1, Section 2.5 of the SQN Safety Evaluation Report NUREG-0011. The specific concern is that 3 percent damping was used for some recent design basis SSE analyses whereas the original vendor seismic qualification used 2 percent in accordance with the TVA procurement requirements, thereby reducing margins implied by the seismic-margins review in 1979.

TVA used Appendix F of the Quality Assurance (QA) manual to establish vendor requirements for seismic qualification of equipment prior to September 1, 1974. After that date, the vendor's seismic-qualification requirements were established by WB-DC-40-31.2, Revision 1. This information was reviewed in the seismic-qualification-review team (SQRT) audit in 1976 and 1977 and documented by References 1 and 2. This change in vendor requirements was made to comply with TVA's design basis licensing commitment to apply the criteria of Institute of Electrical and Electronics Engineers (IEEE) 344-1975 to the fullest extent reasonably possible. WB-DC-40-31.2, Revision 1 specified equipment damping values consistent with IEEE 344-1975. In addition, as documented by References 1 and 3, the SQRT required demonstration of equivalent seismic qualification to IEEE 344-1975 for equipment procured to IEEE 344-1971 (e.g. Appendix F) requirements. TVA completed the required equivalency demonstration in June 1979.

TVA pointed out in Reference 4, and NRC concurred in Reference 5 that SQN is not an A46 plant because it was audited to IEEE 344-1975 or equivalent as its seismic-design basis for mechanical and electrical equipment. Since this was the case, use of 2 and 3 percent damping for seismic analysis of floor- or wall-mounted equipment and components is in accordance with the SQN design basis.

TVA recognizes the need to clarify the FSAR and issue updated seismic-design criteria to more clearly reflect the design-basis-licensing commitments for seismic qualification of SQN mechanical and electrical equipment. An updated design criteria to replace Appendix F of the QA manual will be prepared by October 1, 1990. TVA will update the FSAR to clarify the design basis relative to compliance with IEEE 344-1975 or equivalent for mechanical and electrical equipment and components in the next annual update.

TVA conducted a seismic-margins review as requested by NRC and the Advisory Committee on Reactor Safeguards (ACRS), in 1979 to 1982. The objective of the review was to demonstrate that adequate margins exist in the SQN designs to withstand an 84th percentile site-specific earthquake. This review did not constitute a change in the SQN design basis.

The initial review results were documented in Reference 6. Piping was evaluated by analysis of selected systems using Regulatory Guide (RG) 1.61 damping values and allowable pipe stresses in accordance with ASA B31.1. The conclusions as stated in Reference 6 for piping and equipment were: "The seismic margins were not quantified for selected mechanical and electrical equipment in safe shutdown systems but the equipment was evaluated against the revised floor response spectra and is considered qualified. Based on the results of the reanalysis of the selected systems and the reevaluation of the electrical and mechanical equipment, we concluded that the piping systems and mechanical equipment in safe shutdown systems are sufficiently conservative in design to meet current licensing criteria."

The ACRS recommended a low-power license based on these results on December 11, 1979, but also recommended an expanded study to determine the seismic-design margin of all structures and equipment necessary for safe shutdown. NRC staff accepted this recommendation and required expansion of the seismic-margins review.

TVA submitted a seismic-margins-program plan by Reference 7. This program was completed by TVA, and NRC was notified of completion by Reference 8. The results were that the original qualification analysis or test for the equipment was generally sufficiently conservative to envelope the 84th percentile site-specific floor response spectra at the equipment locations. These results were presented to NRC on March 29-30, 1982, in fulfillment of the August 11, 1980, commitment.

Although precise acceptance criteria for SQN's seismic-margins review were not defined, criteria equivalent to that used for assessing seismic margins for the systematic-evaluation program plants would have been reasonable. This criteria is defined in NUREG/CR-0098 entitled "Development of Criteria for Seismic Review of Selected Nuclear Power Plants." It is based on current earthquake definitions (RG 1.60 or 84th percentile site-specific) and recommended qualification techniques for those earthquake definitions. NUREG/CR-0098 includes a table of recommended damping values that is provided as Attachment 3 to this enclosure. The values in the left column of this table are lower-bound values and those in the right column are average values for the type and condition of structure. According to this table the lower-bound and average-damping values for welded-steel-construction components subjected to an 84th percentile site-specific earthquake would be 5 and 7 percent of critical. The corresponding values for vital piping would be 2 and 3 percent of critical.

Additionally, NUREG/CR-0098 recommends acceptable ductility factors of 1.0 to 1.3 for vital equipment, components, and structures. This implies a steel support/anchorage allowable load increase of 25 percent or more over the design-basis limit of 0.9 yield. Other features of NUREG/CR-0098 would also permit more realistic prediction of dynamic response, for example Section 7.4 on the effects of inelastic action and Section 7.9 on the response of equipment and attachments. The approaches taken by TVA to demonstrate adequate seismic margins were more conservative than those recommended in NUREG/CR-0098.

The largest potential effect of equipment/component damping is indicated by the peak of the response spectra. A comparison follows of the peak spectral accelerations in the auxiliary and control building where the majority of the Category 1 equipment is located. This comparison is for design basis SSE spectra peaks at 3 percent damping versus 84th percentile site-specific spectra at 5 and 7 percent damping. The 3 percent design-basis damping is in accordance with IEEE 344-1975, and the 5 and 7 percent damping are in accordance with NUREG/CR-0098. The building spectra were derived for the 84th percentile site-specific earthquake using RG 1.61 damping values as part of the seismic-margins review described above. In accordance with RG 1.61, 7 percent building damping was specified. (This damping value corresponds with the NUREG/CR-0098 recommended lower-bound value for reinforced concrete.)

The following table compares North-South and East-West horizontal peak values at the building base, Elevation 713.5 (where the CCW heat exchanger and other major heat exchangers are located), and Elevation 791.25, which is the top of the building. Comparisons are made for the horizontal direction only since the mechanical components are typically rigid in the vertical direction so that damping is not a factor for vertical response.

Direction	Elevation	Spectra Peaks		
		Design Basis SSE	84th Percentile (%) Site Specific	
		3% Damping	5% Damping	7% Damping
North-South	669.0 (Base)	0.50g	0.65g	0.56g
East-West	669.0 (Base)	0.50g	0.65g	0.56g
North-South	713.5	1.34g	1.42g	1.20g
East-West	713.5	1.69g	1.68g	1.40g
North-South	791.25 (Top)	2.76g	2.70g	2.31g
East-West	791.25 (Top)	3.17g	3.06g	2.50g

From this comparison it is clear that qualification of the welded construction mechanical components of concern for the design basis SSE, 3 percent damping, and design basis allowable stresses effectively ensures compliance with NUREG/CR-0098 criteria without any further evaluation. This provides verification of the seismic-margins review results in that regard. It also ensures sufficient seismic margin for the TVA-designed equipment/component supports and anchorages which were not evaluated in the 1979-1982 seismic-margins review. These supports and anchorages were evaluated and verified as satisfying design-basis criteria in 1987 and 1988 as documented in TVA's response to IDI audit deficiency D4.6-1. Equipment/component damping values of 2 and 3 percent were used for OBE and SSE, respectively, in those evaluations.

In conclusion, TVA believes that the licensing basis for SQN is compliance with IEEE 344-1975 or equivalent for the design-basis OBE and SSE. The use of damping values from IEEE 344-1975 in some recent work to resolve NRC questions is not inconsistent with the licensing basis. Furthermore, the fact that the damping values are larger than the original calculations does not invalidate any of the seismic-margins review work previously done. TVA has demonstrated that the licensing basis is met and that sufficient seismic margins still exist even in the revised evaluations. As a result, further work addressing the issue of seismic margin does not appear to be a prudent use of resources, and TVA considers this issue closed.

References

1. NRC letter to TVA dated November 16, 1976, concerning the seismic team for SQN.

2. TVA letter to NRC dated February 7, 1977, "Seismic Qualification of Safety-Related Equipment - TVA Meeting with NRC Seismic Team September 28 through 30, 1976."

3. TVA letter to NRC dated June 22, 1979, concerning the seismic-qualification data package.

4. TVA letter to NRC dated March 19, 1986, concerning seismic qualification of equipment at Sequoyah Nuclear Plant.

5. NRC letter to TVA dated October 29, 1987, "Sequoyah Nuclear Plant (SQN) - Replacement Items Program (RIP) Seismic Screening Methodology."

6. NUREG-0011, Supplement 1, Section 2.5

7. TVA letter to NRC dated May 5, 1981, concerning the seismic-margin program plan.

8. TVA letter to NRC dated March 1, 1982, concerning the seismic-margin program plan.

URI 88-12-09, ERCW Pumping Station Access Cells

In Reference 1, TVA provided a response to NRC's concern regarding the use of a friction coefficient of 1.0 for tremie concrete-rock interfaces. The response referred to American Concrete Institute (ACI) Code 318 and shear-friction coefficients. For the ACI 318 shear-friction method, the normal force can be considered as either the internal member force on the cross section or the forces caused by tensile deformation of the reinforcing bars crossing the shear plane. Although no reinforcing steel is present in the access cells or pumping-station foundations, a normal force will exist that will provide shear resistance during a seismic event. The 1.0 value is intended for use at a contact surface with an amplitude of roughness of about 1/4 inch. The use of this value is considered to be conservative, since the actual amplitudes are nearer to one foot.

Additionally, TVA has completed an evaluation that further justifies the use of a 1.0 friction coefficient for both concrete-rock and rock-rock interfaces.

The rock surface at the access cells is composed of interbedded shale (56 percent) and limestone (44 percent). The rock surface is very rough with asperities close to each other because of the inclination of the interbedded rock of different hardnesses. Based on mapping by divers, the angles of inclination (i) of these asperities are generally in the range of 20 to 25 degrees.

For the assessment of the concrete-rock contact surface, Reference 2 recommends a concrete-rock sliding friction coefficient $\tan \mu = 0.7$ (μ = 35 degrees) where μ is defined as the angle of friction. This value is applicable for clean, sound rock on a relatively smooth surface for a horizontal plane. However the actual frictional resistance of the concrete-rock contact surface is the combination of concrete-rock sliding friction and the frictional resistance provided by the asperities. This resistance is calculated by the following equation, which is found in Reference 3.

$$\text{friction coefficient} = \tan(\mu + i)$$

Using a value for μ of 35 degrees and a conservative value for i of 10 degrees (one-half of the lower-bound value of the angle of asperities), the friction coefficient for the concrete-rock contact surface is 1.0.

In addition, field-shear tests of concrete footings and blocks against various types of rock have been conducted at 44 dam sites in various parts of the world. The results as reported in Reference 4 indicate that the friction coefficient exceeded 1.0 for concrete-rock contact surfaces.

For the assessment of the rock-rock friction coefficient, the evaluation related the actual unconfined compressive strength test data (Reference 5) and the angles of internal friction from published literature (Reference 4) on rock similar to those at the ERCW access and pumping station cells. TVA conservatively ignored the cohesion measured for limestone and shale. The friction coefficient for the rock-rock contact surface calculated by this approach is 0.96. If a cohesion value for the shale and limestone had been taken into account, the friction coefficient for the rock-rock contact surface would exceed 1.0.

Based upon the TVA evaluation, a literature search, and the interpretation of ACI 318, it is concluded that a friction factor of 1.0 is an appropriate value for use in the sliding stability analysis of the ERCW cells.

References

1. TVA letter to NRC dated December 28, 1988, "Sequoyah Nuclear Plant (SQN) - Essential Raw Cooling Water (ERCW) Pumphouse Foundation and ERCW Pumping Station Access Cells"

2. NAVFAC DM-7.2 Foundations on Earth Structures, Department of the Navy, Alexandria, VA., 1982

3. R. D. Lama and V. S. Vutukuri, "Handbook on Mechanical Properties of Rocks; Volume IV Trans Tech Publications, Aedermannsdorf, Switzerland, 1978

4. R. D. Lama and V. S. Vutukuri, "Handbook on Mechanical Properties of Rocks; Volume III Trans Tech Publications, Aedermannsdorf, Switzerland, 1978

5. SQN FSAR Section 2.5

URI 88-12-10, Seismic Analysis of the SCV

Revised SCV Spectra Effects

Before the restart of SQN Unit 2, TVA performed a detailed study of the effect of the time-step issue on the SCV vertical spectra and the resulting impact on the qualification of piping systems attached to the SCV. In this study TVA evaluated ten piping analysis problems to assess the effects of the time-step change. Four of the ten piping problems selected to be evaluated contained worst-case attributes (low-allowable stress margins, modal frequency in the region of spectra value increase, high contribution of seismic load and support qualification to Interim Criterion CEB-CI-21.89) selected to maximize the impact of the time-step issue. The remaining six analyses evaluated had worst-case attributes (low-stress margins, high-elevation SCV rigid attachments, etc.) for a prior SCV vertical spectra change and were considered representative of SCV attached piping analysis for purposes of the time-step evaluation.

The ten-problem study showed either a decrease or a negligible increase in pipe stress, SCV penetration loads, support loads, and valve accelerations. TVA concluded from this study that the increase in pipe support loads because of the time-step change is insignificant and that no additional evaluation was required.

In response to NRC's current concern on the justification of the effect of the time-step issue on recent (1988-1990) reanalyses of SCV attached piping, TVA has since initiated a summary study of 21 recently reanalyzed piping problems. The study, described in the response to URI 88-12-03 and further detailed in Attachment 2 to this enclosure, evaluated the impact of the revised vertical spectra and DBA/ZPA effects. This study demonstrated conclusively that none of the 21 piping problems evaluated resulted in pipe support modification driven by revised vertical spectra and DBA/ZPA.

NRC also expressed concerns about the high design load-to-allowable load ratios for pipe supports on piping attached to the SCV. This issue is also addressed in the response to URI 88-12-03.

Revised Reactor Coolant Loop (RCL) Spectra Effects

TVA developed a three-phase program to address the effect of the revised RCL spectra on loop attached piping. The first phase of the program consisted of developing a worst-case impact ranking of loop attached piping and the selection of a sample of the most severely impact piping. The second phase of the program required the evaluation of this worst-case sample to determine if modifications were required by the revised spectra. If no modifications were identified, the impact of the revised spectra would be considered bounded and no further evaluations required. If modifications were identified, the third phase of the program would conduct a horizontal review of other RCL attached piping having similar worst-case attributes to determine if additional modifications would be required.

At the time of the NRC 90-18 inspection, TVA had completed only the first phase of the evaluation program. Each of the attached piping problems was ranked in order of descending anticipated impact. The impact was determined by factoring the current stress levels by the spectral acceleration increases for significant participating modes. Secondary worst-case rankings were also developed for existing piping stress interaction ratios and existing pipe support margin (as indicated by identified post-restart modifications). To ensure the study was also representative of the population of loop attached piping, consideration was also given to selection of a variety of pipe sizes and attachments to each of the major loop components.

Using the worst-case rankings, TVA selected six piping-analysis problems in addition to ongoing reanalyses (because of design changes) of other RCL attached piping to assess the revised RCL spectra impact. Either directly or by similarity, these piping analysis captured (1) the 14 most severely impacted stress analysis based on projected increased pipe stress; (2) six of the top seven piping problems ranked on the number of post-restart modifications; (3) five of the top seven piping analysis ranked on existing pipe stress margin; (4) pipe sizes between 3/8 inch and 10 inches in diameter and (5) attachments to the reactor coolant pump, steam generator, hot leg, cold leg, and cross-over leg.

To implement the second phase of the program, TVA contracted with Bechtel to evaluate the six additional piping analysis problems and with G/C to assess the piping reanalyses resulting from ongoing design changes. The Bechtel evaluation was completed in April of 1990 and was provided to NRC during the 90-18 inspection. Bechtel concluded from this study that the change in RCL spectra did not cause any of the modifications associated with the reanalysis of the six worst-case sample piping problems. The G/C review of ongoing design changes was incomplete at the time of the 90-18 inspection. G/C has since completed this review with only one RCL spectra-driven modification identified. Phase three of the program, horizontal expansion in the failed sample population, has also been completed. The horizontal sample expansion revealed no additional support modifications were required.

Based on the results of this review program, TVA considers the effects of the revised RCL spectra to be bounded and no further evaluations required. The final report on this open issue will be completed and submitted to NRC by September 28, 1990.

URI 88-12-11, Diesel Generator Exhaust Piping

NRC has requested that TVA incorporate the following design attributes into the applicable 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. Although punching-shear design and lug design are not specifically addressed in the design criteria, qualification of these support components will be addressed and documented in the calculations.

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

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

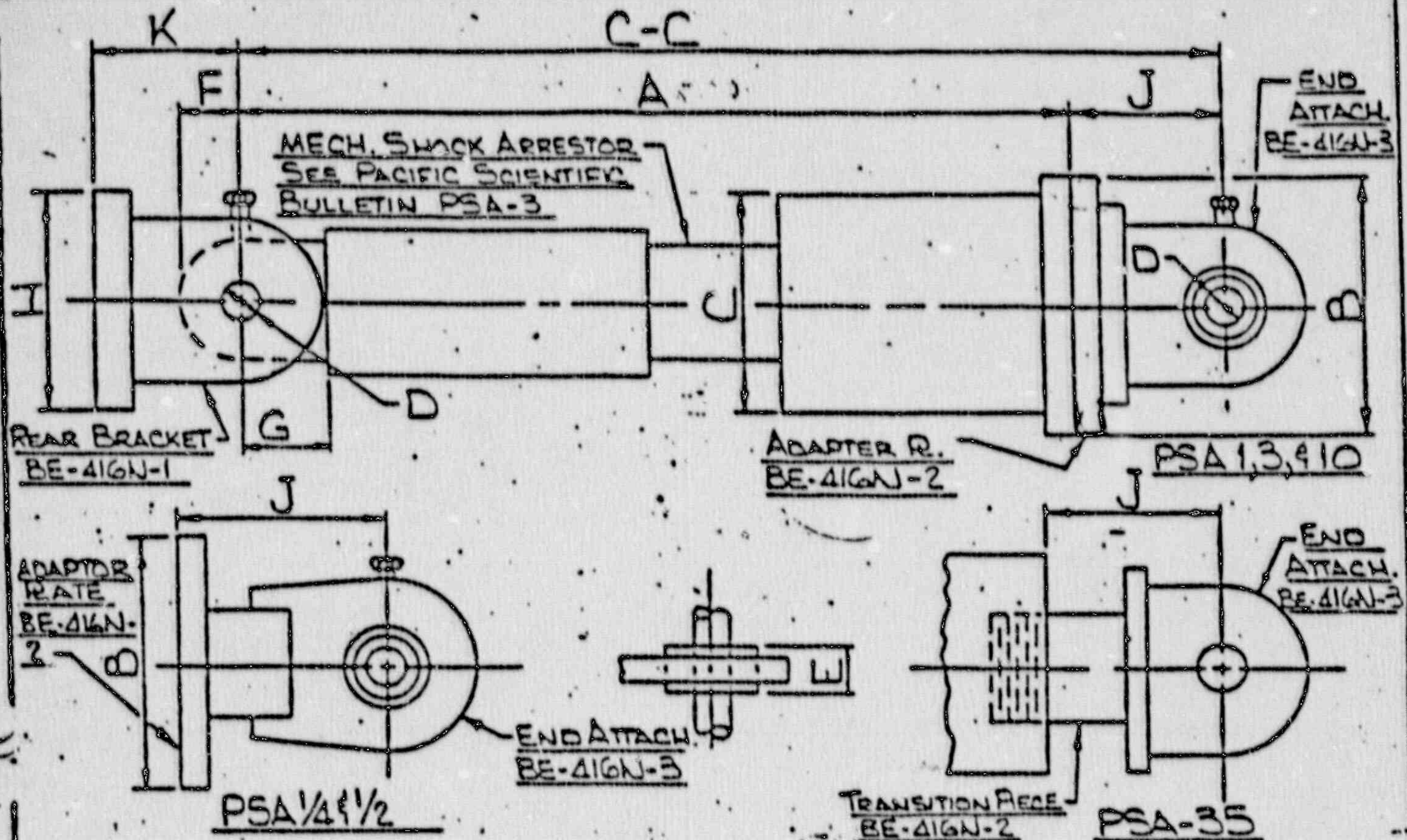
All of these eight attributes will be incorporated into piping analysis and pipe support calculations by September 28, 1990.

ATTACHMENT 1

BASIC ENGINEERS

DIVISION OF NATIONAL VALVE AND MANUFACTURING CO.

MECHANICAL SHOCK ARRESTOR · SHORT UNITS ASS'Y	FIGURE NUMBER: BE-416N



*MAX. RATED LOAD (LBS) @ 300°F
D.N.U.C = DESIGN NORMAL & UPSET CONDITION
E.C = EMERGENCY CONDITION
F.C = FAULTED CONDITION

Δ FAULTED CONDITION LOADS ARE LISTED FOR ATTACHMENTS FURNISHED BY BASIC ONLY. FAULTED LOADS FOR THE SNUBBER UNIT ARE NOT PUBLISHED BY PACIFIC SCIEN.

MODEL PSA-	RATED LOAD* D.N.U.C	E.C.	F.C.Δ	TRAV	WT.	A		B	C	C-C	D	E	F	G	H	J	K
1/4	350	420	540	4	5.6	9	13	2 1/4 DIA.	2 1/4	11 3/8 - 15 3/8	3/8	.406	.50	.85	2	2 3/8	1 1/4
1/2	650	780	1000	2.5	4.7	7 1/2	10	2 1/4 DIA.	2 1/4	9 7/8 - 12 3/8	3/8	.406	.50	.85	2	2 3/8	1 1/4
1	1500	1800	2320	4	13.0	11	15	3.5 SQ	3 1/2	13 5/8 - 17 5/8	1/2	.500	.99	.99	2 1/2	2 5/8	1 3/4
3	6000	7200	9300	5	36.4	14 1/2	19 1/2	5.0 SQ	4.63	18 1/2 - 23 1/2	3/4	.750	1.37	1.35	3 1/2	4	2 3/8
10	15000	18000	23250	6	65.4	17.1	23.2	6.0 SQ	5.75	21.35 - 27.45	7/8	.875	1.65	1.65	4	4 1/4	2 3/4
35	50000	60000	77500	6	185	24	30	—	6	29 3/4 - 35 3/4	1 1/2	1.375	2.85	3.10	5 1/2	5 3/4	4
100	1[illegible]0000	144000	—	6	475	29	35	—	7 1/2	32.8 - 38.8	2 1/2	2.062	3.50	3.80	10x20	3.80	6.28

DRWN. TJW DATE 9-22-76	CHKD. PGS DATE 9-27-76	REV.	1	2	3	4	H-58

ATTACHMENT 2

ATTACHMENT 2

Unresolved Item (URI) 88-12-03 identifies a concern with the exclusion of design-basis accident/zero period acceleration (DBA/ZPA) from Sequoyah Nuclear Plant (SQN) rigorous piping analysis performed before 1988. URI 88-12-10 Part 1 identifies a concern with the time step used to generate the steel containment vessel (SCV) vertical seismic spectra. TVA has developed new vertical seismic spectra curves for the SCV, which have been in use since 1988. In 1989 new leak-before-break (LBB) DBA spectra were developed, which reduce conservatisms inherent in the previously used DBA spectra and DBA-ZPA values.

The 21 rigorous piping analysis problems rigidly attached to the SCV were formally reanalyzed during the period from 1988 to 1990. A study of the support modifications required by these reanalyses has been performed to determine if the modifications were driven by the inertial response spectra cases associated with the SCV. The 21 problems can be grouped into four categories: two hydrogen collector lines, seven upper containment vent cooler lines, eight lower containment vent cooler lines, and four other miscellaneous reanalyses.

Hydrogen Collector Lines

The two hydrogen collector lines in Unit 2, Problem Numbers N2-14-3R and N2-14-4R, were reanalyzed and unitized in 1988. The civil calculation regeneration program for Unit 2 designed three load-driven modifications from the output of these reanalyses. The Unit 1 calculation regeneration program used the Unit 2 loads for support design, as the systems were identical and the Unit 2 reanalyses were state-of-the-art.

The Unit 1 program designed no modifications. A review of the Unit 1 calculations shows that the modifications in Unit 2 could have been eliminated by neglecting the loads generated by differential SCV movements in the static load cases. Neglecting these loads is justified since any two of the attachment points are only separated by a few feet. Therefore, the modifications designed for N2-14-3R and N2-14-4R were not driven by either of the inertial load cases related to the SCV and were not required to achieve design criteria compliance. The support design margin in these lines was adequate to accomodate the revised SCV vertical spectra, missing mass effects from seismic-inertia cases, and DBA/ZPA.

Upper-Containment Vent-Cooler Lines

Seven upper-containment vent-cooler lines were reanalyzed in response to Significant Condition Report (SCR) SQN CEB 8626 in 1988. The seven piping analysis problems are: N2-67-10A, N2-67-11A, N2-67-12A, N2-67-15A, N2-67-16A, N2-67-37A, and N2-67-38A.

The piping configurations of the seven lines are very similar, but there are differences in the support schemes. The three analysis problems with the largest number of support modifications from these reanalyses were chosen for study of SCV inertial effects. These piping problems are N2-67-11A, N2-67-12A, and N2-67-15A. The N2-67-11A analysis had a piping stress-driven modification as well as support load-driven modifications, and was therefore thought to be one of the most severe cases among the eight.

N2-67-15A

The modifications designed in the 1988 reanalysis did not arise from load increases. The loads from the 1988 reanalysis are lower overall than the loads that existed before the analysis. The modifications were driven by design criteria changes and revisions to vendor-supplied component allowables. The same modifications would have been required, if no reanalysis had been performed, by the calculation-regeneration program. Modifications of this nature have been programmatically identified and addressed by the calculation-regeneration program and are not related to any SCV inertial-case issues.

N2-67-12A

The loads on supports in this analysis problem increased under the 1988 reanalysis. A piping run was made with the 1988 geometry and new LBB spectra, and the support loads were evaluated for their effect on the previous configuration of the supports. The modification of the support at Node R1 would not have been necessary with the loads produced by the LBB runs. The modification at Node 141 was not load driven, the original support was adequate for the applied load. The modification was caused by a swing angle problem. The modification at Node 144 was driven by an interference issue. It is concluded that adequate design margin for the loads generated by the current LBB piping run existed at the time of the 1988 reanalysis.

N2-67-11A

The two support modifications on problem N2-67-11A were driven by load increases caused by configurational changes identified by SCR SQN CEB 8626. Several piping runs have been generated to determine the relative effects of the piping-configuration change, the DBA/ZPA effects, and the SCV vertical spectra change. These runs demonstrate that the bulk of the load changes was because of the configuration change, and the loads generated purely by these changes would have driven the modifications, independent of other issues. On the support at Node 17 the load more than doubled, from 2,556 to 6,465 pounds, simply because of the revision of the geometry. The addition of missing mass effects and new vertical spectra raised the 6,465-pound load to 6,614 pounds, an increase of 2 percent. The support at Node 110 was changed to a three way to take load off the support at Node 17 and reduce overall stresses in the system. These two modifications were driven by configurational changes in the pipe routing and were not significantly affected by missing mass or SCV vertical seismic spectra changes.

The conclusion of the study of the three upper-containment vent-cooler lines is that the support modifications were not driven by the revised SCV vertical spectra, seismic missing mass, or DBA/ZPA effects, but by the configuration changes identified by the SCR. The three lines studied effectively enveloped the support configuration of all seven lines, therefore it is believed that the conclusion applies to all seven lines.

Lower-Containment Vent-Cooler Lines

Eight of the lower-containment vent-cooler lines were reanalyzed in 1988 to incorporate a design change to the isolation valves inside containment. The isolation check valves had been failing leak rate tests and were relocated and replaced with motor-operated butterfly valves. The eight analysis problems were: N2-67-1R, N2-67-3R, N2-67-5R, N2-67-7R, N2-67-9R, N2-67-10R, N2-67-12R, and N2-67-26R.

The eight lines are geometrically similar, but the support configurations vary. Three of the eight lines have been chosen to represent the various support configurations, N2-67-3R, N2-67-5R, and N2-67-7R. These three analyses have been studied for the effects of the inertial cases associated with the SCV.

N2-67-3R

A piping analysis identical to the 1988 Code of Record Analysis was performed using the old SCV SSE vertical spectra and neglecting missing mass in both the seismic and DBA cases. The faulted loads for the supports modified under the 1988 reanalysis have been compared to the loads from this run, and no variation of more than 7 pounds was observed. It is concluded that the modifications required by the 1988 reanalysis were not caused by DBA/ZPA or by the revised SCV vertical spectra but by the relocation and change to the valve.

N2-67-5R

A piping analysis identical to the 1988 Code of Record Analysis was performed using the old SCV SSE vertical spectra and neglecting missing mass in both the seismic and DBA cases. The faulted loads for the supports modified under the 1988 reanalysis have been compared to the loads from this run, and no variation of more than 8 pounds was observed. It is concluded that the modifications required by the 1988 reanalysis were not caused by DBA/ZPA or by the revised SCV vertical spectra but by the relocation and change to the valve.

N2-67-7R

A piping analysis identical to the 1988 Code of Record Analysis was performed using the old SCV SSE vertical spectra and neglecting missing mass in both the seismic and DBA cases. The faulted loads for the supports modified under the 1988 reanalysis have been compared to the loads from this run, and no variation of more than 21 pounds was observed. This 21-pound increase was on a 6,000-pound load and is insignificant. It is concluded that the modifications required by the 1988 reanalysis were not caused by DBA/ZPA or by the revised SCV vertical spectra but by the relocation and change to the valve.

The conclusion of the study of the three lower-containment vent-cooler lines is that the support modifications were not driven by the revised SCV vertical spectra, seismic missing mass, or DBA/ZPA effects but by the configuration changes relating to the valve relocations. The three lines studied effectively enveloped the support configuration of all eight lines, therefore the conclusion should apply to all eight lines.

Miscellaneous Other Reanalyses

N2-67-2A, N2-67-3A4, N2-77-1R (Unit 1), and 0600104-04-01 (Unit 2) have been reanalyzed since 1988 using new SCV vertical seismic spectra and missing mass techniques for both seismic and DBA load cases, and no modifications in the vicinity of the SCV were required. The conclusion drawn is that sufficient design margin existed in pipe supports to accomodate the two SCV-related URI issues.

SEQUOYAH NUCLEAR PLANT
PIPING ANALYSES WITH RIGID CONNECTIONS TO THE SCV - REANALYZED SINCE 1988

PROBLEM #	DESCRIPTION	UNIT	PEN #	ELEV	AZIMUTH	PIPE SIZE	NO. OF SUPPORTS
N2-14-3R	HYDROGEN COLLECTOR	2	-	736'-848'	250°	12"	14
N2-14-4R	HYDROGEN COLLECTOR	2	-	736'-848'	295°	12"	14
N2-67-10A	UPPER ERCW CONTNMNT	1	X69	772'	301°	2"	7
11A	VENT COOLERS	1	X70	774'	301°	2"	10
12A	"	1	X71	776'	301°	2"	10
15A	"	1	X74	774'	301°	2"	13
16A	"	1	X75	775'	301°	2"	13
37A	"	1	X68	770'	301°	2"	8
38A	"	1	X72	769'	301°	2"	8
N2-67-1R	LOWER ERCW CONTNMNT	1	X58	697'	7°	6"	17
3R	VENT COOLERS	1	X60	697'	173°	6"	17
5R	"	1	X62	697'	187°	6"	17
7R	"	1	X56	697'	353°	6"	17
9R	"	2	X56	697'	353°	6"	17
10R	"	2	X60	697'	173°	6"	17
12R	"	2	X62	697'	187°	6"	17
26R	"	2	X58	697'	7°	6"	17
N2-67-2A	ERCW SUPPLY HEADER	1,2	X58	697'	7°	6"	157
			X56	697'	173°	6"	
			X60	697'	187°	6"	
			X62	697'	353°	6"	
N2-67-3A4	ERCW DISCHARGE HEADER	2	X59	697'	8°	6"	31
			X63	697'	189°	6"	
N2-77-1A	WASTE DISPOSAL	1	X39A	697'	280°	1"	7
0600154-04-01	DISCH FROM EXCESS LETDOWN HX TO SCV	2	X35	697'	301°	6"	18

06/29/90

ATTACHMENT 3

TABLE 1. RECOMMENDED DAMPING VALUES

Stress Level	Type and Condition of Structure	Percentage Critical Damping
Working stress, no more than about ½ yield point	a. Vital piping	1 to 2
	b. Welded steel, prestressed concrete, well reinforced concrete (only slight cracking)	2 to 3
	c. Reinforced concrete with considerable cracking	3 to 5
	d. Bolted and/or riveted steel, wood structures with nailed or bolted joints	5 to 7
At or just below yield point	a. Vital piping	2 to 3
	b. Welded steel, prestressed concrete (without complete loss in prestress)	5 to 7
	c. Prestressed concrete with no prestress left	7 to 10
	d. Reinforced concrete	7 to 10
	e. Bolted and/or riveted steel, wood structures, with bolted joints	10 to 15
	f. Wood structures with nailed joints	15 to 20

ENCLOSURE 2

LIST OF COMMITMENTS

Unresolved Item (URI) 88-12-02

1. The 24 unqualified U-bolt support configurations will be modified before restart from the Unit 2 Cycle 5 refueling outage.

2. The five unqualified Unit 1 Unistrut clamp support configurations will be modified before restart from the Unit 1 Cycle 5 refueling outage.

3. The 25 unqualified Unit 2 Unistrut clamp support configurations will be modified before restart from the Unit 2 Cycle 5 refueling outage.

4. TVA will submit a schedule for the necessary Unit 1 modifications on pre-NF (not manufactured to the requirements of subsection NF of the American Society of Mechanical Engineers (ASME) Code) mechanical snubbers 6 weeks after restart from the Unit 1 Cycle 5 refueling outage.

5. The necessary modifications on pre-NF mechanical snubbers will be completed for Unit 2 before restart from the Unit 2 Cycle 5 refueling outage.

URI 88-12-08

1. An updated-design criteria to replace Appendix F of the Quality Assurance (QA) manual will be prepared by October 1, 1990.

2. TVA will update the Final Safety Analysis Report (FSAR) to clarify the design basis relative to compliance with Institute of Electrical and Electronics Engineers (IEEE) 344-1975 or equivalent for mechanical and electrical equipment and components in the next annual update.

URI 88-12-10

1. The final report on the revised reactor coolant loop spectra effects will be complete and submitted to NRC by September 28, 1990.

URI 88-12-11

1. All of NRC's eight requested design attributes will be incorporated into piping analysis and pipe support calculations by September 28, 1990.