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EGG-EA-6016

August 1982

TECHNICAL EVALUATION REPORT

HADDAM NECK PLANT SEISMIC DESIGN

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ABSTRACT

A Systematic Evaluation Program was initiated by the Nuclear Regulatory Commission (NRC) to bring eleven older operating nuclear power plants to a level of safety consistent with current standards of acceptability. Northeast Utilities Services Company (NUSCO) personnel and their consultants analyzed the Haddam Neck Plant's safety related piping, mechanical and electrical equipment, and component supports. NRC personnel and their consultants from EG&G Idaho, Inc. formed a review team that evaluated the licensee's analyses. The analyses presented to the review team by NUSCO and their consultants were generally acceptable with the exception of minor suggestions, comments, and questions. The results were obtained through working level meetings and telephone conversations with NUSCO personnel and their consultants. The results indicate that modifications may be required to bring this plant to an acceptable level of safety.

SUMMARY

A Systematic Evaluation Program (SEP) was initiated by the Nuclear Regulatory Commission (NRC) with the goal of bringing eleven older nuclear power plants to a level of safety consistent with current standards of acceptability. The Haddam Neck Nuclear Reactor is one of these plants. The NRC and their consultants from EG&G Idaho, Inc. formed a review team and evaluated the acceptance criteria and analyses presented by the Northeast Utilities Services Company (NUSCO) and their consultants. These analyses were performed on the safety related equipment required to function during a Safe Shutdown Earthquake (SSE).

The information was obtained through working level meetings between NUSCO personnel, their consultants, and the review team. Piping, mechanical equipment, electrical equipment, and component support analyses were evaluated with the review team formulating suggestions and open items at the conclusion of each of the presentations. The review team developed an acceptance criteria for guidance in evaluating these analyses. Documents sent to the review team and telephone conversations with NUSCO personnel and their consultants also aided the review team in obtaining the required information for the evaluation of the plant.

This report was divided into individual sections covering the balance-of-plant piping, the reactor coolant loop piping, electrical equipment, the balance-of-plant mechanical equipment, the reactor coolant loop mechanical equipment, and component supports. These sections contain procedures utilized by NUSCO or their consultants for the analyses performed. Each section also contains the review team's evaluation of the analyses presented.

The analyses and procedures presented by NUSCO and their consultants to the review team were generally acceptable. However, some open items still remain and must be addressed for this review to be complete. The results indicate that modifications may be required to bring this plant to an acceptable level of safety.

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INTRODUCTION

In October of 1977, the Office of the Nuclear Reactor Regulation (NRR), an office of the Nuclear Regulatory Commission (NRC), initiated a Systematic Evaluation Program (SEP) by selecting eleven older operating nuclear power plants with the goal of bringing these plants to a level of safety consistent with current standards of acceptability. These plants were divided into two groups based on their original seismic design. The Haddam Neck Plant, operated by the Connecticut Yankee Atomic Power Company (CYAPCO), is included with the Group II plants. A reanalysis was performed to demonstrate that the structural integrity of the safety related piping systems and their supports, mechanical equipment, and electrical equipment would not be impaired when subjected to a Safe Shutdown Earthquake (SSE) combined with other normal design loadings.

The Haddam Neck Nuclear Reactor is a pressurized light water moderated and cooled system. The plant initially produced 1,473 MW of heat and 490 MW of gross electric power. The plant was designed to produce 590 MW of gross electric power. The containment structure was designed to the ACI Building Code and the ASME Boiler and Pressure Vessel Code; state and national codes were utilized as guides to arrive at a safe design criteria. Both the reactor vessel and components were analyzed to the ASME Code. The original design criteria used for analysis of this plant's primary piping system was the ASA B31.1 Code for pressure piping.

A decision was made by the NRC to review the reevaluation analyses performed by the licensee and their consultants rather than performing their own analyses on the plant. A review team consisting of NRC staff personnel and NRC consultants from EG&G Idaho, Inc. evaluated the piping, mechanical, and electrical equipment analyses. The licensee and their consultants were required to present their seismic reevaluation criteria, typical analyses, and results to the review team.

The audit review consisted of working level meetings between the review team, Northeast Utilities Services Company (NUSCO) personnel, and their consultants. These meetings proved to be an efficient method of exchanging information among the review team, licensee, and their consultants with a minimum of formal written communication. The review team obtained a general idea of methods utilized by the licensee through these meetings. Sample analyses and calculations were presented and reviewed in detail. Questions, comments, and open items were formulated and submitted to the licensee at the conclusion of each working level meeting. Before these working level meetings were initiated, the review team developed an audit plan (Appendix A) and presented it to the NUSCO personnel. This plan was developed to aid the utility and their consultants in presenting information the review team considered important.

The review team developed an acceptance criteria for guidance in evaluating the analyses. The licensee was requested to justify major deviations which appear less conservative than those in the review team acceptance criteria.

The scope of review for the seismic reevaluation program included the systems, structures, and components (including emergency power supply and distribution, instrumentation, and actuation systems) with the following functions:

1. The reactor coolant pressure boundary as well as the core and vessel internals. This also includes those portions of the steam and feedwater system extending from and including the secondary side of the steam generator up to and including the outermost containment isolation valve and connected piping of 2-1/2 inch or larger nominal pipe size, up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of normal reactor operation.

2. Systems or portions of systems that are required for safe shutdown as identified in the SEP safe shutdown review (SEP Topic VII-3). The system boundary includes those portions of

the system required to perform the safety function and connected piping up to and including the first valve that is either normally closed or capable of automatic closure when the safety function is required.

3. Systems or portions of systems that are required to mitigate design basis events, i.e., accidents and transients (SEP Topics XV-1 to XV-24). The functions to be provided include emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, as well as support systems, such as cooling water, needed for proper functioning of these systems.

4. Systems and structures required for fuel storage (SEP Topic IX-1). Integrity of the spent fuel pool structure including the racks is needed. Failure of the liner plate due to the safe shutdown earthquake must not result in significant radiological releases, or in loss of ability to keep the fuel covered. Failure of cooling water systems or other systems connected to the pool should not permit draining of the fuel pool. Means to supply make-up to the pool as needed must be provided.

5. Structures that house the above equipment.

For the Haddam Neck Plant, the review team required the following systems, and associated structures, and components to be addressed.

(a) Reactor Coolant System (RCS)
(b) Portions of Main Steam System
(c) Portions of Main Feedwater System
(d) Portions of systems directly connected to the RCS up to and including isolation valves
(e) Control Rod Drives
(f) Auxiliary Feedwater System
(g) Residual Heat Removal System (including ECCS recirculation mode)

(h) Portions of Chemical and Volume Control System
(i) Portions of Service Water System
(j) High Pressure Safety Injection System
(k) Low Pressure Safety Injection System
(l) Containment Cooler System
(m) Spent Fuel Pool and Makeup

As discussed previously, a "system" also includes the power supply, instrumentation and actuation systems.

This report was divided into individual sections covering balance-of-plant and reactor coolant piping, electrical equipment, balance-of-plant and reactor coolant mechanical equipment, and component supports. Each section explains in detail NUSCO's or their consultant's analysis procedures, acceptance criteria, and typical analyses. Each section also contains the review team's evaluation of the analyses performed by NUSCO or their consultants. The review team's conclusions were based upon NUSCO's and their consultants presentations and documents.

BALANCE-OF-PLANT PIPING SYSTEMS

Licensee Evaluations

NUSCO performed most of the analyses required for the safety related piping systems of the Haddam Neck Plant. However, the Nuclear Steam Supply System (NSSS) consultant also performed piping analyses. The NUSCO analyses were performed in accordance with their "Piping Stress Analysis Procedure For Seismic Qualification of Safety Related Piping at Connecticut Yankee" (Appendix B) and results were compared with the "Connecticut Yankee Atomic Power Company Safety Related Piping Seismic Qualification Program Criteria Document" (Appendix C).

The safety related piping systems (defined in Appendix C) analyzed for the SEP were:

1. Reactor coolant loop
2. Main steam
3. Feedwater
4. Auxiliary feedwater
5. Residual heat removal
6. High pressure safety injection
7. Chemical and volume control
8. Service water
9. Fuel oil
10. Compressed air.

The "Piping Stress Analysis Procedure" provided a guide for modeling wall and floor penetrations, valves, non-standard fittings, flanged joints, branches, and anchors. It also discussed spring hangers, single acting restraints and loading conditions.

During the working level meetings, much discussion was devoted to finding an acceptable method of modeling single acting restraints. NUSCO's approach required the analyst to examine each of these supports individually. When deadweight and thermal analyses without single acting supports incorporated into the model were reviewed, only the single acting supports at points with downward displacements were utilized. Single acting restraints were not included when the seismic analyses were performed. Even though NUSCO did not include single acting restraints in their seismic analyses, they did account for impact loading on these restraints. A factor of 1.5 times the deadweight reaction times the peak value of acceleration from the appropriate floor response spectra was substituted for the impact load.

Wall and floor penetrations were also examined on an individual basis. If gaps existed around the pipe at these locations, thermal movements were considered in determining whether or not a penetration acted as a restraint. If the thermal movement was greater than the size of the gap, the penetration was considered a restraint. In addition to modeling a restraint at these penetrations, a displacement was imposed on the pipe which was as large as the penetration gap. These gap closures were not considered for seismic load cases unless a restraint was required due to overstressed piping in the area, for which case the penetration gap was closed by shimming. If the pipe was embedded in concrete, the penetration was considered either an anchor or a four way restraint. The penetration was considered an anchor if an anchor ring or lugs were welded to the pipe.

NUSCO also defined several areas on the piping systems as anchors. These areas are:

1. Equipment nozzles

2. Piping interface where the moment of inertia of the run pipe exceeds that of the connecting line by a minimum factor of ten

3. An anchor, i.e. a six way restraint

4. Two or more restraints in proximity such that the effects of piping on either side of the support group are isolated.

This list was intended as a guide for the individual analyst. Engineering judgement was also used in considering anchor points.

NUSCO obtained valve dimensions, weights, and centers of gravity from the original valve drawings. If these drawings were not available, standard valve drawings, the manufacturer's catalog, or other plant records were investigated.

The stress intensification factors for non-standard fittings were obtained from either the manufacturer's data or engineering judgment. The reinforcement area of fabricated branch connections was reviewed in accordance with ANSI B31.1, Paragraph 104.3.1.

If the moment of inertia of a run line was ten times greater than the moment of inertia of a connecting branch line, the branch line was analyzed separately and the run line was considered as an anchor.

Spring hangers were considered as external loads applied during the weight analyses. The piping thermal and seismic displacements were used in determining if the spring hanger remained in its working range. Spring hangers that were considered unacceptable were replaced.

Other modeling techniques found in the piping stress analysis procedure were also utilized. These methods were also reviewed. The piping criteria document also indicated that realistic support stiffnesses were applied in appropriate directions.

NUSCO defined several loading conditions and analyzed the safety related piping to these conditions. These loading conditions were:

1. Design pressure plus deadweight plus temperature

2. Safe Shutdown Earthquake (SSE) plus maximum operating pressure plus deadweight.

Weight loads included the weight of the piping and components, insulation, and contents.

The piping system evaluations were based on the guidelines stated in ANSI B31.1 Power Piping Code 1973 Edition, Summer 1973 Addenda. The loading combination and stress limits utilized by NUSCO are summarized in Table 1 of Appendix C. The stress limits used for the SSE loading condition correspond to the faulted condition allowables.

Most of the seismic analyses were performed using lumped mass dynamic models with the appropriate floor response spectra at two or three percent damping. If the piping system ran between floors, the response spectra utilized was an envelope of the individual floor response spectra. Simultaneous three directional input was utilized and the results of each mode were combined by the square root sum of the squares (SRSS) method in conformance with Regulatory Guide 1.92. A static load case analysis was performed to account for the effects of missing mass. The equivalent "g" (acceleration) value for the static load case was equal to the quantity one minus the modal effective mass fraction times the "g" value corresponding to the cutoff frequency.

NUSCO also reserved the option to perform an equivalent static seismic analysis. The static analyses for each direction were combined by the SRSS method. The equivalent static "g" load was equal to the maximum value of acceleration from the appropriate floor response spectra times 1.5.

For small piping systems (2-1/2 in. and less), NUSCO felt a chart method of analysis was sufficient. No details on the chart method were presented to the review team.

In general, the piping systems were analyzed on ADLPIPE, February 1977, Revision 1B. Some consideration was given to utilizing NUPIPE-II, Version 1.5, for piping models too large for ADLPIPE. NUSCO has not presented any piping system analyses utilizing NUPIPE-II at the present time.

An approximate 10% increase in the number of supports was required on the piping systems analyzed to date. NUSCO was required to present a complete report of the balance-of-plant piping to the NRC by the end of August, 1982. A summary of NUSCO's completed results are contained in Appendix D.

Review Team Evaluations

The Acceptance Criteria for Piping provided by the NRC review team is contained in Appendix E. If Class 2 analytical procedures are used, two Equation 9 stress allowables are required. Stresses in piping considered as Class 1 must not exceed 1.8 S_h. Stresses in piping considered as Class 2 must not exceed 2.4 S_h. Other stipulations are also stated in the NRC's Acceptance Criteria for Piping.

In general, the methods (defined in Appendix A and B) applied by NUSCO in their piping reanalyses are acceptable. The modeling techniques utilized by NUSCO provide a complete and practical representation of the piping systems. Uplift and support buckling were considered on one-way supports. Pipe deflections were checked to assure that spring hangers remained in their working range. The percent of damping utilized by NUSCO for the response spectra was in accordance with Regulatory Guide 1.61.

The mass point spacing suggested by Arthur D. Little, Inc. for use with the ADLPIPE computer program was examined and considered acceptable by the review team. The ADLPIPE guide for masspoint spacing is contained in Appendix F.

Fifteen piping system analyses were reviewed in detail at one of the working level meetings. The systems reviewed were the Service Water System lines SR6S, SR6R, SR7S, SR7R, SR8S, SR8R, SR9S, SR9R; Feedwater lines 7, 8, 9, 10; Main Steam lines 24 in. SMP-601-1 through 4; and HPSI Loops 1 and 2. These piping analyses were considered acceptable. The review team assumed the piping analyses that were checked were general examples of the remaining analyses. The selected analyses that were reviewed appeared to be complete and they were performed in an acceptable manner.

NUSCO's method of analyzing wall or floor penetrations with gaps was considered acceptable provided only one penetration was contained in the model. However, some concern exists as to whether or not models with several wall or floor penetrations containing gaps will be properly analyzed. If not properly analyzed, the piping system may undergo unanticipated thermal movements. These thermal movements could possibly be overlooked in the piping analysis allowing the existence of undetermined stresses. The analytically imposed displacement may also create a loading on the piping that the free thermal movement would not create. Individually displacing the pipe at one gapped penetration at a time will help define the situation at the other penetrations. Several iterations may be required to define the true thermal expansion of the piping system. Although the review team has not evaluated an analysis under these circumstances, NUSCO has acknowledged that they recognize the potential problems when more than one gapped penetration is contained in a piping system.

NUSCO's piping criteria [illegible]d the review team's acceptance criteria. NUSCO used Class [illegible]ures and analyzed the equivalent of Class 1 piping to 1.8 S_h and Class 2 piping to 2.4 S_h. There was no deviation from the NRC staff's piping criteria. Although NUSCO's piping criteria did not mention the inclusion of seismic anchor movements, later telephone conversations have verified that seismic anchor movements were included in the analyses performed.

NSSS PIPING SYSTEMS

Licensee Evaluations

The seismic reevaluation analysis of the primary reactor coolant loop (RCL) is currently being performed by the licensee's NSSS consultant. The analysis of this system is not complete; however, preliminary results have been provided and are contained in Appendix G. From this preliminary analysis, the piping stresses were determined to be within allowable stresses for both normal design pressure plus deadweight and SSE plus operating pressure plus deadweight load combinations. This analysis was based upon existing pipe support conditions with no support modifications required.

The criteria used to evaluate the RCL piping was based on the rules of the ANSI B31.1-1973 Code, Summer 1973 Addenda. The allowable stress for the faulted load combination of SSE plus deadweight plus operating pressure was reduced from $2.4S_h$ to $1.8S_h$ to account for the differences in stress indices between ASME Class 1 and Class 2 analyses.

This analysis was performed using three-dimensional static and dynamic models, depending on the loading under consideration. The computer code WESDYN was used to perform these static and dynamic analyses. The dynamic seismic analysis was performed using the response spectra method. The seismic analyses were performed with simultaneous spectra input accelerations for the two horizontal components and one vertical component of the earthquake. Modal responses were combined using the SRSS method in accordance with the requirements of Regulatory Guide (RG) 1.92. The spectra input for each direction was the envelope of the applicable floor spectra at the different support locations at a value of 4% critical damping. A maximum masspoint spacing of five feet for piping with a 30-35 in. outside diameter and a thickness of 2-1/4 to 3-1/4 in. was utilized.

The three-dimensional lumped mass model of the RCL system was based on as-built isometric piping drawings and equipment drawings. In addition to the RCL piping, the following equipment was included in the RCL system model: RCL valves, steam generators, reactor coolant pumps, and the reactor pressure vessel. Nonlinear support conditions occurred at two locations in the RCL. These two nonlinear conditions are: (1) the potential uplift of the reactor vessel from its supports and (2) the lower steam generator supports which are capable of resisting only compression loads. The analyses results indicated that uplift of the reactor vessel from its supports will not occur. The seismic support loads at this location are less than the normal operating compression loads due to pressure, weight, and thermal expansion. The second nonlinear condition was treated by assuming only two of the lower steam generator supports were present. Two analyses were performed to bound this condition. One analysis was performed with two of the adjacent loop steam generators supported and the other two free. Another analysis was performed in the same manner with two other adjacent loop steam generators supported. These two fixed steam generators were perpendicular to the two assumed fixed in the first analysis. Simultaneous three directional earthquake response spectra were input for both analyses. From these two analyses, the highest stresses were used to evaluate the RCL system piping regardless of whether they were for a loop which had the steam generator lower support modeled or not.

The surge line from the RCL hot leg to the pressurizer was not included in the RCL system model. It was analyzed separately using the same response spectra method as used for the RCL analysis. The stresses from this piping system were also found to be within allowable limits. One spring hanger was found to be overstressed. A proposal was made for this spring hanger to be replaced with a rigid rod hanger of suitable strength. The surge line was reanalyzed for this support condition and piping stresses were still found to be within the allowable limits.

Review Team Evaluations

The criteria and methods proposed for the RCL piping system and surge line analyses appear reasonable and consistent with the requirements for the SEP. The analytical methods utilized conform with current practice for performing seismic response spectra analyses of piping systems. However, one exception is that the damping value used is 4% which is higher than the requirements of R.G. 1.61. Justification for use of the higher damping value of 4% of critical damping is required. The masspoint spacing utilized for the RCL piping analysis was considered acceptable.

As previously mentioned, the RCL results are preliminary at this time. A detailed review of these analyses was not possible at the audit meetings since the analyses were not complete. Changes to the RCL mechanical equipment supports could be such that the RCL system piping results may change from current preliminary results. A detailed review of the final RCL system surge line piping analyses is recommended. In addition, the anchor displacements of the main steam and feed water piping attached to the steam generator should be reviewed to determine if these were adequately addressed for the piping analyses.

ELECTRICAL EQUIPMENT

Licensee Evaluations

A consultant to NUSCO performed the seismic reevaluation analyses for selected safety related electrical equipment of the Haddam Neck Plant. The analyses were performed in a conventional manner utilizing marketed finite element computer programs coupled with insitu vibration testing. These analyses were compared against the Allowable Stress Criteria and Damping Values (Appendix H) list. This document is an itemized list of all mechanical and electrical equipment, the allowable stresses, and the damping values utilized for each item.

Only a sampling was made of the electrical equipment analyzed. The consultant felt these analyses sufficiently covered all of the necessary types of safety related electrical equipment. The safety related electrical equipment selected for the seismic reevaluation analysis for the SEP are:

1. Battery rack
2. Motor Control Center (MCC) No. 1
3. Switchgear (D. G. Room)
4. Control panel (D. G. Room)
5. Engine mounted control panel (on diesel generator)
6. 4160-480 V switchgear
7. Transformers (switchgear room)
8. MCC No. 5 and 6
9. Battery charger
10. MCC No. 3
11. Main control board
12. Emergency power control board
13. MCC No. 8.

The consultant proposed utilizing conventional modeling techniques coupled with low impedance, insitu vibration testing for analyzing the selected electrical equipment. Three dimensional lumped mass finite element models were developed for performing seismic reevaluation analyses using the response spectra method. The computer code SAP IV was used to perform these analyses. The frequencies and mode shapes of the computer analysis were verified from those determined by low impedance, insitu testing. An evaluation of the equipment was performed considering normal design conditions combined with the postulated SSE loading. A general written analysis procedure for electrical equipment was not provided.

The seismic analyses were performed utilizing the appropriate floor response spectra with damping specified in the list found in Appendix H. The electrical equipment evaluation was based upon the guidelines stated in ASME Section III, Division 1, Appendix XVII and Appendix A.

The MCC No. 1 is the only piece of electrical equipment that has been analyzed to date (Appendix I). It was analyzed using the computer program SAP IV. A response spectra analysis on a finite element beam and plate model was performed. A damping value of 7% was utilized. The results of this analysis indicated the MCC No. 1 maintained structural integrity during and after a seismic event.

Review Team Evaluations

The allowable stress criteria for the individual pieces of electrical equipment was evaluated by the review team. These allowables were considered acceptable for each piece of equipment. In general, the methods proposed for the electrical equipment analyses are acceptable and consistent with SEP requirements. The modeling techniques utilized provide a complete and practical representation of the electrical equipment. Proposed analysis methods adequately address evaluation of proper load combinations. Damping values proposed are also consistent with current accepted practice for seismic analysis of this type of equipment.

The MCC No. 1 analysis was the only electrical equipment analysis completed and available for detailed review. This analysis was adequately performed and documented. This seismic reevaluation analysis was performed using the response spectra method. The three dimensional lumped mass model representation of this piece of equipment was adequately verified by comparing the mode shapes and frequencies determined from the computer analysis to those determined by low impedance, insitu testing. This method of performing such an analysis is acceptable current standard practice. From the review of the details, this analysis was determined to have been adequately performed for SEP.

From discussions with the consultant performing the electrical equipment analyses, the remaining analyses will be completed using the same methods as those used for the MCC No. 1 analysis. Detailed review of the remaining analyses is recommended. In addition, it is recommended that safety related electrical equipment similar to electrical equipment requiring design modifications be reevaluated for seismic loading.

BALANCE-OF-PLANT MECHANICAL EQUIPMENT

Licensee Evaluations

A consultant to NUSCO performed the seismic reevaluation analyses for selected Balance-Of-Plant (BOP) safety related mechanical equipment of the Haddam Neck Plant. These analyses were performed in a conventional manner utilizing marketed finite element computer programs. The results of these reevaluation analyses were compared against the Allowable Stress Criteria and Damping Values (Appendix H) list.

Only a sampling was made of the mechanical equipment analyzed. The consultant felt these analyses sufficiently covered all the necessary types of mechanical equipment. The safety related mechanical equipment selected for the seismic reevaluation analysis for the SEP are:

1. ESW pump
2. Diesel exhaust duct
3. Diesel air start-up tanks
4. CVCC regenerative heat exchanger
5. Diesel generator
6. Boric acid pump
7. High pressure safety injection pump
8. RHR pump
9. RHR heat exchanger
10. Boric acid tank
11. Demineralized water storage tank
12. Refueling water storage tank
13. Steam driven auxiliary feedwater pump
14. Underground 5,000 gallon oil tank
15. Clean diesel oil day tank
16. Volume control tank
17. Containment fan coolers.

The selected BOP mechanical equipment was analyzed using conventional computer models and hand calculation techniques. Generally, these analyses were performed using three dimensional lumped mass computer models. Where justified, two dimensional stick beam models, such as for axisymmetric structures, were developed to perform these analyses. The seismic analyses were generally performed using the response spectra method, however, static equivalent seismic analyses were utilized where justified. For the response spectra analyses, floor response spectra with damping listed in Appendix H was used as the SSE seismic input. The mechanical equipment was evaluated considering normal design conditions combined with the SSE loading. The evaluation was based upon the guidelines stated in ASME Section III, Division 1, and ASME Appendix XVII and Appendix A.

Only four of the selected mechanical equipment analyses were completed and available to the review team for detailed evaluation. Analyses were performed on the diesel exhaust duct, boric acid tank, refueling water storage tank, and steam driven auxiliary feedwater pump. The analyses results are contained in Appendix I.

The diesel exhaust duct was analyzed using the finite element multi-purpose program, DYNAFLEX. A response spectra analysis applied to a three dimensional pipe element model was performed. A damping value of 4% was utilized. The transition piece from the diesel exhaust nozzle to the duct was analyzed separately using an unspecified multi-purpose finite element program. The results of this analysis indicated the diesel exhaust duct maintained structural integrity during and after a seismic event.

The boric acid tank was analyzed using the finite element multi-purpose program, SAP IV. A response spectra analysis utilizing a two dimensional stick beam dynamic model was performed. A damping value of 7% for impulsive loads and 0.5% for sloshing loads was utilized. The results of this analysis indicated the boric acid tank maintained structural integrity during and after a seismic event. The tank was also considered leak tight.

The refueling water storage tank was analyzed using the finite element multi-purpose program, SAP IV. A response spectra analysis utilizing a two dimensional finite element stick beam dynamic model was performed. This model appeared as a beam attached to soil springs. A damping value of 7% for impulsive loads and 0.5% for sloshing loads was utilized. The results of this analysis indicated the refueling water storage tank maintained structural integrity during and after a seismic event except for the anchorage system. Design modifications required to fix the anchorage system were not completed or available for review.

The final mechanical equipment analysis presented for review was the steam driven auxiliary feedwater pump. This piece of equipment was analyzed using the finite element multi-purpose program, SAP IV. A response spectra analysis applied to a three dimensional finite element multi-degree of freedom dynamic model was performed. A damping value of 7% was utilized. The results of this analysis indicated the steam driven auxiliary feedwater pump maintained structural integrity during and after a seismic event. The pump was also considered leak tight and capable of operation after an earthquake.

Review Team Evaluations

The proposed scope of the BOP mechanical equipment seismic reevaluation analysis does not include all of the plant safety related mechanical equipment. The scope includes a sampling of equipment which represents all the types of safety related mechanical equipment. This similarity approach is acceptable provided adequate justification is presented to assure worst-case bounding conditions have been included in the analyses.

The allowable stress criteria provided for the BOP mechanical equipment contained in Appendix H was evaluated by the review team and determined to be acceptable for SEP. These allowables are expressed in terms of yield strength only, but they are consistent with the Class 2 mechanical equipment allowables developed for SEP which are contained in Appendix E.

In general, the methods utilized in the BOP mechanical equipment analyses are acceptable. The modeling techniques utilized provided complete and practical representations of the mechanical equipment. Load combinations evaluated were proper in that the normal design loading combined with the SSE loading was evaluated. The seismic analyses of tanks properly addressed both impulsive and sloshing loads. Damping values used for the dynamic analyses are consistent with current accepted values for these types of equiment. Assessment of leak tightness was adequately performed for the mechanical equipment analyzed.

As previously mentioned, only four of the selected seventeen BOP mechanical equipment analyses were completed and available for review at the final Haddam Neck Plant SEP audit meeting. These completed analyses were for the diesel exhaust duct, the boric acid tank, the refueling water storage tank, and the steam driven auxiliary feedwater pump. The details of these analyses were reviewed at the final Haddam Neck Plant SEP audit meeting. It was determined that these analyses were adequately performed. From discussions with the consultant performing these analyses, it is assumed that the analyses of the remaining thirteen pieces of mechanical equipment will also be performed in a similar acceptable fashion. A detailed review of the completed analyses for the remaining thirteen pieces of mechanical equipment is recommended. Review of the proposed design modification required to fix the refueling water storage tank anchorage system is also recommended. In addition, it is recommended that safety related mechanical equipment similar to selected mechanical equipment requiring design modifications be reevaluated for seismic loading.

NSSS MECHANICAL EQUIPMENT

Licensee Evaluations

The seismic reevaluation analyses of the NSSS mechanical equipment is currently being performed by the licensee's NSSS consultant. The NSSS mechanical equipment being reevaluated for seismic loading consists of the reactor vessel, steam generators, reactor coolant pumps, and pressurizer. The analyses for this equipment were not complete, however, preliminary results have been provided and are contained in Appendix G. From these preliminary analyses results, the equipment stresses have been determined to be within allowable stresses for the SSE loading plus normal design pressure plus deadweight. The allowable stress criteria used to evaluate this equipment's pressure boundary is based upon the ASME Code Class 1 vessel requirements for faulted conditions.

The seismic reevaluation analysis of the reactor vessel was performed in two steps. The reactor vessel nozzles, shell, and supports were evaluated using loads from the RCL piping system analysis. An analysis of the reactor vessel internals will be performed using a more detailed model of the reactor vessel and internals. This analysis will not be completed for several months. From the preliminary evaluation of the reactor vessel shell and nozzles, reactor vessel stresses were determined to be within allowable stress limits.

The seismic reevaluation of the steam generators and pressurizer was also performed in two steps. The shell, nozzle, and support loads were obtained from the RCL piping system analysis. Stresses for internals were determined from more detailed lumped mass finite element models. The seismic analyses utilizing these more detailed models were performed using the response spectra method. It has been determined from the preliminary results of the analyses of this equipment that stresses for the vessels and internals are within allowable stresses.

The seismic reevaluation of the reactor coolant pumps was similarly performed. The pump nozzle and support loads were obtained from the RCL piping system analysis. The remainder of the pump internals were evaluated based on the time history seismic analysis performed for pumps of the same model for San Onofre Nuclear Generating Station Unit 1 (SONGS 1). The response spectra developed from the time history acceleration used for the SONGS 1 pump analysis envelope the response spectra developed for the Haddam Neck Plant pumps. The response spectra for the Haddam Neck Plant pumps was developed using the Singh method. Pump stresses were determined to be within allowables for the SONGS 1 pump anlaysis. The Haddam Neck pump housing stresses, including nozzle and support loads from the RCL piping system analysis, were also within allowable stresses.

Review Team Evaluations

The scope of the NSSS mechanical equipment seismic reevaluation analyses is adequate since all NSSS mechanical equipment will be reevaluated. In general, the methods utilized in this reevaluation are acceptable. The acceptance criteria used to evaluate the equipment's pressure boundary is consistent with that developed for the SEP which is contained in Appendix E. The allowable stress criteria utilized in evaluating equipment internals also appears reasonable. The load combinations evaluated for the NSSS mechanical equipment are proper in that the normal design loading combined with the SSE loading was evaluated. Sample values used for detailed steam generator and pressurizer analyses are not specified in the preliminary reports.

All NSSS mechanical equipment stress results were within allowable limits, however, these results were preliminary. Design modifications to some of the supports were proposed and may affect reevaluation results of the equipment. A detailed review of the NSSS mechanical equipment final analyses is recommended after the equipment support design modifications are completed. An additional concern with regard to the steam generator analysis is that impacting of the steam generator lower supports was not adequately addressed. This impact loading should either be evaluated or additional justification should be provided as to why evaluation of this impact loading was not required.

The reevaluation of the Haddam Neck Plant's primary coolant pump internals was based upon the acceleration response spectra being less than the acceleration response spectra for SONGS 1 primary coolant pumps. These spectra were not provided, therefore, a detailed comparision could not be made by the review team. Copies of these spectra were requested. In addition, qualification of the Haddam Neck Plant primary coolant pump internals is contingent upon approval of the analyses performed for the SONGS 1 primary coolant pumps. A detailed review of that analysis has not yet been completed. Upon approval of the SONGS 1 analysis, approval will also be granted for the Haddam Neck Plant primary coolant pumps.

COMPONENT SUPPORTS

Licensee Evaluations

Component support analyses were required for all piping systems, mechanical equipment, and electrical equipment. The analyses performed by NUSCO were done in accordance with their piping stress analysis procedure and the safety related piping seismic qualification program.

As mentioned earlier, NUSCO examined each single acting support individually. When deadweight and thermal analyses without single acting supports incorporated into the model were reviewed, only the single acting supports at points with downward displacements were utilized. Single acting restraints were not included when the seismic analysis was performed. NUSCO did examine impact loading on single acting restraints. NUSCO considered spring hangers as external loads applied during a weight analysis. The spring hangers were checked and those that exceeded their working range seismically or thermally were replaced. Rigid supports utilized in piping analyses were analyzed by applying the load combinations specified in Table 2 of Appendix C. NUSCO considered thermal expansion and thermal anchor movements as primary loads during the support analyses.

Stress limits for supports were also supplied in Table 2 of Appendix C for the NUSCO component support criteria. No criteria was specified by NUSCO regarding concrete attachments.

NUSCO has presented a limited number of component support analyses for review. These calculations were included with the piping system analyses. If the applied support loads did not exceed the original design loadings, NUSCO did not reanalyze the support.

Evaluation of the NSSS piping and mechanical equipment component supports was performed by NUSCO's NSSS consultant. These supports were evaluated for normal design and normal design plus SSE loading. The loads required for performing this evaluation were obtained from the RCL piping system analysis. Since the RCL piping system analysis was performed assuming linear elastic behavior for all component supports, one-way supports (tension only or compression only) were evaluated individually to assure uplift and impacting does not occur, thus invalidating the analysis assumptions. In all cases uplifting was not found to be a problem. It was determined that some of the RCL pump support spring hanger cans were bottoming out due to combined SSE and normal design loading. Using energy balance considerations, new spring can loads were determined accounting for this bottomed out condition. Design modifications to these supports were required because of this increased loading. Other NSSS component supports requiring design modifications include: 1) pressurizer truss support, 2) surge line pipe supports, and 3) steam generator lower hold down bolts. With the exception of the surge line supports, design modifications required for these component supports were not finalized. For the surge line, two spring hanger supports were found to be overstressed. The proposed design modification was to replace these spring hangers with rigid struts. The surge line piping analysis has been reanalyzed for this proposed support condition and found to be within acceptable limits. Preliminary results for all NSSS component support evaluations are contained in Appendix G. The acceptance criteria used to evaluate the NSSS component supports is in accordance with the requirements of the ASME Code. Subsection NF. Criteria for buckling and anchorage were not provided.

Seismic reevaluation of the BOP mechancial and electrical equipment supports will be performed by the same consultant performing the seismic reevaluation analyses of the BOP safety related electrical and mechanical equipment. Currently only five out of 31 selected pieces of equipment have been analyzed. This equipment will be reevaluated considering normal design loading combined with SSE loading. For most of this equipment, the normal design loading consists of deadweight only. The allowable stress criteria, utilized to evaluate the BOP equipment support structures is contained in Appendix H. One-way supports (tension or compression supports only) will be evaluated individually to assure uplift and impacting do not occur. Of the five pieces of BOP equipment analyzed, structural adequacy was demonstrated for all of the equipment supports except for the refueling water storage tank which lacks adequate anchorage. Design modifications required to remedy this situation have not been finalized.

Review Team Evaluations

The acceptance criteria for component supports provided by the review team is contained in Appendix E. The acceptance criteria was divided into Class 1 and Class 2 subsections. The Class 1 support criteria was written in terms of principal stress intensity. The Class 2 support criteria was written in terms of principal stresses. Other stipulations are also stated in Appendix E for the NRC acceptance criteria for component supports. The acceptance criteria for concrete attachments is also included in Appendix E.

NUSCO provided the review team with sample component support analyses. These calculations were contained in the fifteen piping analyses previously mentioned. Rack support calculations were not reviewed since they will not be initiated until all of the piping analyses are completed. The review team requested NUSCO to present a typical rack support calculation as soon as one is available. When NUSCO presented their component support analysis to the review team, an unsymmetrical bending problem arose that had not been analyzed as such. The review team requested assurances that unsymmetrical bending problems be analyzed properly. At the next meeting, NUSCO presented the same and similar problems. The analyst stated that stresses were sufficiently small (based on Mc/I) to warrant an unsymmetrical bending calculation but they would perform unsymmetrical bending analyses on other component supports if engineering judgment deemed it necessary.

NUSCO's component support criteria paralleled the review team's original Class 1 component support acceptance criteria. However, the review team revised their acceptance criteria several times after the working level meetings were held for the Haddam Neck Plant. NUSCO did not provide a concrete attachment acceptance criteria.

The seismic reevaluation methods utilized for the NSSS piping and mechancial equipment supports were generally adequate and consistent with SEP requirements. The scope of this effort is adequate in that all NSSS component supports will be evaluated. The analyses of the NSSS component supports properly considered the normal design loading and the normal design loading combined with the SSE loading. The acceptance criteria used to evaluate the component supports was generally adequate, however, discussions with the consultant performing these analyses revealed that buckling was not adequately addressed for all component supports. Buckling did not appear to have been properly evaluated for the reactor vessel support (Neutron Shield Tank), the steam generator support skirt, and the pressurizer support truss. The licensee should perform buckling evaluations for these component supports utilizing buckling criteria equivalent to that contained in the reevaluation guideline contained in Appendix E. Anchorage allowable stress criteria was not provided for the NSSS component support reevaluation. This must be provided before a complete assessment of the NSSS component supports reevaluation can be made. Another concern with the steam generator and steam generator support analyses was that impacting of the steam generator lower supports was not adequately addressed. This impact loading should either be evaluated or additional justification should be provided as to why it need not be evaluated.

The NSSS component support analyses presented were preliminary. Design modifications are being proposed for the following equipment supports: steam generator, pressurizer, RCL pumps, and surge line piping. The proposed modifications to these supports appear reasonable, however, a detailed review of the final analyses accounting for the required design modifications is recommended.

The BOP mechanical and electrical equipment support analyses generally appeared adequate. Supports and anchorage were evaluated for all of the selected safety related BOP equipment which were reevaluated for seismic loading. The methods for performing the evaluation for these equipment supports were considered adequate. The load combinations considered in the evaluation were adequate. The allowable stress criteria used to evaluate the BOP equipment supports and anchorage was generally more conservative than that specified in the acceptance criteria contained in Appendix E. The allowable stress for passive component support structures was based on material yield strength. Care must be taken when evaluating general primary membrane stress or general primary membrane stress intensity, in that the allowable stress based on ultimate strength may govern rather than the allowable stress based on yield strength. The BOP equipment support allowable stress criteria should be specified in terms of both yield strength and ultimate strength to be consistent with the acceptance criteria contained in Appendix E.

The BOP safety related equipment and equipment support seismic reevaluation analyses will be performed for 31 selected pieces of equipment. To date only five of these analyses have been completed. A detailed review of the remainder of these analyses is recommended upon their completion.

CONCLUSIONS

BOP Piping System Analyses

NUSCO presented copies of their acceptance criteria and modeling techniques to the review team at the working level meetings. Typical piping analyses were also presented at the meetings. The review team evaluated this information and concluded that NUSCO's overall analysis techniques and piping criteria are reasonable. NUSCO's support criteria was also evaluated and appeared to parallel the review team's support criteria. However, the review team's acceptance criteria was modified since the meeting with NUSCO.

Several open items still remain and must be addressed before the reevaluation of the BOP piping system is complete. These items are:

1. All remaining analyses results should be submitted to the NRC.

2. NUSCO's concrete anchorage (expansion and embedded anchor bolts) acceptance criteria should be submitted to the NRC.

3. NUSCO's chart method and typical analyses should be submitted to the NRC if this method is used for piping system analyses.

BOP Electrical and Mechanical Equipment Analyses

Seismic reevaluation analyses of the Haddam Neck Plant BOP safety related electrical and mechanical equipment are being performed for 30 selected pieces of equipment representative of all types of safety related electrical and mechanical equipment. At the time of the final Haddam Neck SEP audit meeting, June 22-23, 1982, preliminary seismic reevaluation results were available for only five of the 30 selected pieces of BOP mechanical and electrical equipment. The preliminary results for these five analyses indicated that four out of the five pieces of equipment are structurally adequate for combined normal design loading plus postulated SSE loading. The anchorage system for the refueling water storage tank was found to be inadequate.

After reviewing the five preliminary BOP equipment analyses (contained in Appendix I) it was determined that these analyses, in general, were adequately performed. The modeling techniques utilized provide a complete and practical representation of the equipment. The load combinations evaluated were adequate. The allowable stress criteria used to evaluate the equipment and equipment supports was generally consistent with the SEP criteria contained in the Reevaluation Guideline (Appendix E). However, the passive component support structure allowable stress criteria used may not be consistent for all cases.

For BOP equipment, the following open items must be addressed before the reevaluation of the BOP equipment is complete. These items are:

1. The scope of the reanalysis efforts for BOP electrical and mechanical equipment is inadequate in that:

    a. all equipment required for safe shutdown and emergency core cooling has not been included,

    b. instrument and control reevaluation to assure that adequate parameters are available to the plant operator are not included,

c. air systems and air operators required to insure that necessary safety functions are met are not included.

Should the licensee desire to use similarity arguments for demonstrating the structural adequacy of safety related equipment for seismic loading, justification based on sound engineering principles is required.

2. Provide justification that the allowable stress criteria utilized (yield strength) for the passive component support structures do not exceed the allowable stress criteria contained in Appendix E. The stress criteria contained in Appendix E is based on both yield strength and ultimate strength.

3. Submit all final BOP equipment and equipment support analyses to the NRC for review.

4. For equipment requiring design modifications, such as the refueling water storage tank, provide a detailed description of the design modifications.

5. For BOP equipment similar to the selected BOP equipment requiring design modifications, provide SEP seismic reevaluation analyses or provide justification for not performing these analyses.

NSSS Piping and Mechanical Equipment Analyses

Seismic reevaluation analyses are being performed for the Haddam Neck Plant RCL. Preliminary results of these analyses demonstrate structural adequacy of all the components for combined normal design loading plus SSE loading. However, several component supports were determined to be structurally inadequate. They are: 1) two surge line piping supports, 2) steam generator hold-down bolts, 3) primary coolant pump supports, and 4) the pressurizer support truss.

After reviewing the preliminary analyses, it is concluded that the NSSS piping, mechanical equipment, and component support analyses are generally being performed in an adequate manner. The scope of these analyses is adequate since the entire NSSS is being reevaluated. The modeling techniques being utilized to perform these analyses provide a complete and practical representation of the RCL and equipment components. The load combinations being evaluated are adequate since both normal design loading and normal design loading plus SSE loading are being evaluated. However, possible impact loading of the system generators with their lower supports was not addressed. The RCL system piping analyses are being performed using 4% of critical damping which is not consistent with RG 1.61. The allowable stress criteria being used for reevaluation of the NSSS piping and mechanical equipment components are adequate and consistent with the SEP criteria contained in Appendix E. The allowable stress criteria being used for reevaluation of the component supports does not adequately address buckling and does not specify allowable stresses for anchorage.

For NSSS piping and mechanical equipment, the following open items must be addressed:

1. Justification must be provided for using 4% of critical damping for the RCL piping system analyses.

2. Damping values used to perform the detailed model analyses of the pressurizer and steam generators were not provided. The licensee is requested to provide these values.

3. Impact loading of the steam generators with the steam generator lower supports should be addressed or justification as to why this item is not accounted for should be provided.

4. Buckling of reactor supports (neutron shield tank), steam generator support skirts, and the pressurizer support truss member should be addressed using allowable stress criteria consistent with that contained in Appendix E.

5. Component support anchorage allowable stress criteria was not provided. The licensee is requested to provide this allowable stress criteria.

6. Submittal of final reevaluation analyses is requested.

7. For NSSS components and component supports requiring design modifications, a detailed description of the proposed design modifications is requested.

APPENDIX A

HADDAM NECK AUDIT PLAN FOR SEP SEISMIC QUALIFICATION
OF PIPING, MECHANICAL, AND ELECTRICAL EQUIPMENT

HADDAM NECK AUDIT PLAN FOR SEP SEISMIC QUALIFICATION OF PIPING, MECHANICAL, AND ELECTRICAL EQUIPMENT

I. Background

In October, 1977, the office of Nuclear Reactor Regulation (NRR) initiated Phase I of the Systematic Evaluation Program (SEP) to determine the margin of safety relative to current standards for eleven selected operating nuclear power plants and to define the nature and extent of retrofitting required to bring these plants to acceptable levels of safety if they are not already at these levels. Phase I of SEP involved Group I plants, where Phase II involves Group II plants, consisting of San Onofre 1, La Crosse, Big Rock Point, Yankee Rowe, and Haddam Neck. The review for seismic requalification of SEP Group II plants will be performed by two teams. One team consisting of NRC staff personnel and NRC consultants from Lawrence Livermore National Laboratory (LLNL) will evaluate the Group II plants' structures. A second team consisting of NRC staff personnel and NRC consultants from EG&G Idaho, Inc., will evaluate the Group II plants' piping, mechanical, and electrical equipment important to safety. This audit plan provides a description of how the SEP seismic requalification of Haddam Neck piping, mechanical, and electrical equipment important to safety will be reviewed.

II. Scope

The scope of review for the SEP seismic re-evaluation program will include the systems and components (including emergency power supply and distribution, instrumentation, and actuation systems) with the following functions:

1. The reactor coolant pressure boundary as well as the core and vessel internals. This should also include those portions of the steam and feedwater system extending from and including the secondary side of the steam generator up to and including the outermost containment isolation valve and connected piping for

all safety related systems up to and including the first valve that is either normally closed or is capable of automatic closure during all modes of normal reactor operation.

2. Systems or portions of systems that are required for safe shutdown as identified in the SEP safe shutdown review (SEP Topic VII-3). The system boundary includes those portions of the system required to perform the safety function and connected piping up to and including the first valve that is either normally closed or capable of automatic closure when the safety function is required.

3. Systems or portions of systems that are required to mitigate design basis events, i.e., accidents and transients (SEP Topics XV-1 to XV-24). The functions to be provided include emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, as well as support systems, such as cooling water, needed for proper functioning of these systems.

4. Systems and structures required for fuel storage (SEP Topic IX-1). Integrity of the spent fuel pool structure including the racks is needed. Failure of the liner plate due to the safe shutdown earthquake must not result in significant radiological releases, or in loss of ability to keep the fuel covered. Failure of cooling water systems or other systems connected to the pool should not permit draining of the fuel pool. Means to supply makeup water to the pool as needed must be provided.

For the Haddam Neck plant, the following systems, and components should be addressed:

1. Reactor Coolant System (RCS)

2. Portions of Main Steam System

3. Portions of Main Feedwater System

4. Portions of systems directly connected to the RCS up to and including isolation valves

5. Control Rod Drives

6. Auxiliary Feedwater System

7. Residual Heat Removal System (including ECCS recirculation mode)

8. Portions of Chemical and Volume Control System

9. Portions of Service Water System

10. High Pressure Safety Injection System

11. Low Pressure Safety Injection System

12. Containment Cooler System

13. Spent Fuel Pool and Makeup

As discussed previously, a "system" also includes the power supply, instrumentation and actuation systems.

III. General Criteria and References

The criteria contained in the following documents will be the bases used to evaluate the SEP seismic re-evaluation of Haddam Neck Plant piping, mechanical, and electrical equipment important to the plant's ability to safely withstand the effects of a postulated safe shutdown earthquake event.

1. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," N. M. Newmark and W. J. Hall, May 1978.

2. Standard Review Plan, Sections 3.2, 3.7, 3.8, 3.9, 3.10.

3. Regulatory Guides, 1.29, 1.48, 1.60, 1.61, 1.89, 1.92, 1.100, 1.124, 1.130.

4. ANSI/IEEE Standard 344-1975.

5. ASME Boiler and Pressure Vessel Code Section III, 1980 Edition or subsequent.

6. AISC, "Manual of Steel Construction," Eighth Edition.

The intent of Phase II of SEP is to demonstrate that the structural integrity of the systems and components being re-evaluated will not be impaired when subjected to a postulated Safe Shutdown Earthquake (SSE) in combination with other normal design loadings. As a minimum, component primary stresses must be evaluated using current criteria provided in the above standards for Level D (faulted) service limits.

IV. Review Procedures

A. General

The review team (NRC and NRC consultants) will perform the review effort parallel with the licensee's seismic re-evaluation efforts. A minimum of three working level meetings among the review team, licensee, and licensee's consultants are anticipated. This method of review has been selected in order to expedite the review. The working level meetings will permit an exchange of information which will minimize formal written communication, thus expediting the program. One of the meetings will be conducted at the plant so the review team can perform a field inspection of the equipment being re-evaluated.

The review process will be accomplished in three steps. The first step will consist of the review team reviewing the details of the seismic re-evaluation program plan submitted by the licensee. A substantial portion of this review will be performed prior to the first working meeting. Any concerns the review team has with the program plan will be discussed and preferably resolved at the first working meeting.

The next step of the review will consist of review of analyses performed by the licensee or licensee's consultants. This review will be performed by one or more of the following methods: (a) The review team will perform a review of seismic re-evaluation analyses at the working meetings. (b) The review team will perform review of seismic re-evaluation analyses at their offices. These analyses will either be given to the review team at the working meetings or transmitted by mail to the review team upon completion. (c) The review team will perform independent analyses for some components and systems. Information necessary to perform these analyses will be supplied by the licensee at the working meetings or transmitted later. The depth of review of analyses will vary depending on the complexity of the item being evaluated. The analysis review guidelines are contained in Appendix A.

The third and final step of the review process will consist of the review team preparing and submitting a technical evaluation report (TER) which identifies the results of the seismic re-evaluation review.

3. <u>Audit Meeting Agenda</u>

As previously mentioned, the SEP will require working level meetings among the review team members, licensee, and licensee consultants to be held either at the plant or at licensee's engineering offices. For the meetings at the engineering offices, the following agenda is anticipated:

1. Detailed presentation of seismic re-evaluation program plan by licensee or licensee's consultants.[a]

2. Discussion and resolution of concerns which the review team has with the program plan.[a]

3. Presentation of licensee's progress towards completion of seismic re-evaluation program by licensee.

4. Presentation of anticipated schedule for completing program by licensee.

5. Summary presentation of seismic re-evaluation analyses results (include identification of systems and components which require retrofitting) by licensee.

6. Detailed review of completed seismic re-evaluation analyses for selected systems and equipment (include detailed review of required retrofits).

7. Exit briefing identifying acceptable areas of review and areas of concern requiring additional information to resolve by review team.

For the meeting at the plant, the following agenda is anticipated:

1. Presentation of licensee's progress towards completion of seismic re-evaluation program by licensee.

2. Presentation of anticipated schedule for completing program by licensee.

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a. Required at initial meeting only.

3. Summary presentation of seismic re-evaluation analyses results (include identification of systems and components which require retrofitting) by licensee.

4. Field inspection of selected equipment being re-evaluated by review team and licensee.

5. Detailed review of newly completed seismic re-evaluation analyses, by review team (include detailed review of required retrofits).

6. Exit briefing identifying acceptable areas of review and areas of concern requiring additional information to resolve, by review team.

V. Review Team Members

The SEP review team for Haddam Neck nuclear powr plant will consist of the following NRC and EG&G Idaho, Inc., personnel.

NRC

Thomas M. Cheng

EG&G Idaho, Inc.

Tom L. Bridges
Sheryl L. Busch
D. Keith Morton[a]
Tommie R. Thompson

a. First working meeting only.

VI. Review Schedule

The anticipated schedule for completing Phase II of SEP for Haddam Neck nuclear power plant is as follows:

1.	First working meeting	Week of 04-19-82
2.	Plant visit	Not Scheduled
3.	Final working meeting	Not Scheduled
4.	Complete TER	08-31-82

APPENDIX A ANALYSIS REVIEW GUIDELINES

The following is a list of guidelines to be used in reviewing analyses for the SEP Group II Plants. Although the list may not be all inclusive, it does provide the areas of interest pertaining to the SEP review.

I. Analysis Audit Format (Piping)

1. What computer codes were used in the analyses?

   a. How were the above computer codes verified?

2. Is the proper input forcing function being utilized?

   a. If response spectra method is used:

      (1) Is correct spectra and damping utilized?

      (2) Have sufficient modes been used to adequately describe system response?

      (3) Is spectra properly broadened?

      (4) Do system frequencies straddle any peaks?

   b. If time history method is used:

      (1) Is sufficient system response achieved?

      (2) Is an adequate time step utilized?

      (3) Is proper damping utilized?

c. If static equivalent method is used:

(1) Is justification provided for performing a static equivalent analysis?

(2) How was required level of input determined?

3. Has the piping system been properly modeled?

a. Have valves been properly modeled including any eccentricity?

b. Has adequate mass point spacing been utilized?

c. Are adjacent element length ratios reasonable?

d. Have all significant branch piping systems been included?

e. Have all supports been specified with correct imposed loads (if any), direction and stiffness?

f. Have supports with significant nonlinear characteristics been properly handled?

g. Have correct pipe sizes, geometry, thicknesses, and uniform weights been specified?

h. Have correct design and operating pressure and temperature data been specified?

4. Has the piping system been evaluated against proper criteria?

a. Has a proper minimum thickness check been performed?

b. Have excessive deflections been considered?

c. Have proper stress intensification factors been utilized?

d. Have proper load combinations been analyzed?

e. Have proper allowable stress limits been selected in order to assure the required operation of the piping?

f. Were standard or nonstandard components used?

g. What criteria were used in evaluating adequacy of supports?

II. Analysis Audit Format (Mechanical Equipment)

1. Is the equipment rigid or flexible?

a. How were the natural frequencies determined?

b. If flexible, is its response single-directional or multi-directional?

c. If flexible, is its response at one predominant frequency or at several frequencies?

2. What type of analysis was performed?

a. Static g level

(1) How was required level of input determined?

b. If response spectra method is used:

(1) Is correct spectra and damping utilized?

(2) Is sufficient system response achieved?

(3) Is spectra properly broadened?

(4) Do system frequencies straddle any peaks?

(5) How were directional components of input applied (combined)?

c. If time history method is used:

(1) Is sufficient system response achieved?

(2) Is an adequate time step utilized?

(3) Is proper damping utilized?

(4) How were directional components o input applied (combined)?

d. If testing was used for requalification:

(1) What type of test was performed?

(2) What justification is provided for the type of test used?

(3) How were system natural frequencies determined?

(4) How was the required response spectra (RRS) determined?

(5) How does the test response spectra (TRS) compare to the RRS?

(6) What g level was used in the test?

(7) Were support and boundary conditions, including anchor bolts, properly simulated in the test?

(8) How was functional operability verified during the test?

(9) What criteria were used in evaluating the adequacy of the test results?

3. What computer codes were used in the analyses?

   a. How were the above computer codes verified?

4. Has the system been properly modeled?

   a. Has adequate mass point spacing and distribution been used?

   b. Have all supports and boundary conditions, including anchor bolts, been properly modeled?

   c. Have significant nonlinear effects been properly handled?

5. Has the system been evaluated against proper criteria?

   a. Have the proper load combinations been analyzed?

   b. Have proper stress intensities been evaluated?

   c. Have deflections been considered?

   d. Have proper allowable stress limits been selected?

   e. How were computer output responses combined (directional and modal)?

III. Analysis Audit Format (Electrical Equipment)

1. Is the equipment rigid or flexible?

   a. How were the natural frequencies determined?

   b. If flexible, is its response single-directional or multi-directional?

   c. If flexible, is its response at one predominant frequency or at several frequencies?

2. What type of analysis was performed?

   a. Static g level

      (1) How was required level of input determined?

   b. If response spectra method is used:

      (1) Is correct spectra and damping utilized?

      (2) Is sufficient system response achieved?

      (3) Is spectra properly broadened?

      (4) Do system frequencies straddle any peaks?

      (5) How were directional components of input applied (combined)?

   c. If time history method is used:

      (1) Is sufficient system response achieved?

(2) Is an adequate time step utilized?

(3) Is proper damping utilized?

(4) How were directional components of input applied (combined)?

d. If testing was used for requalification:

(1) What type of test was performed?

(2) What justification is provided for the type of test used?

(3) How were system natural frequencies determined?

(4) How was the required response spectra (RRS) determined?

(5) How does the test response spectra (TRS) compare to the RRS?

(6) What g level was used in the test?

(7) Were support and boundary conditions, including anchor bolts, properly simulated in the test?

(8) How was functional operability verified during the test?

(9) What criteria were used in evaluating the adequacy of the test results?

3. What computer codes were used in the analyses?

a. How were the above computer codes verified?

4. Has the system been properly modeled?

   a. Has adequate mass point spacing and distribution been used?

   b. Have all supports and boundary conditions, including anchor bolts, been properly modeled?

   c. Have significant nonlinear effects been properly handled?

5. Has the system been evaluated against proper criteria?

   a. Have the proper load combinations been analyzed?

   b. Have proper stress intensities been evaluated?

   c. Have deflections been considered?

   d. Have proper allowable stress limits been selected?

   e. How were computer output responses combined (directional and modal)?

APPENDIX B HADDAM NECK SEP PROGRAM PLAN REVIEW SUMMARY

At the first SEP audit meeting for the Haddam Neck plant, the licensee and its engineering consultants provided a detailed presentation of the Haddam Neck plant SEP program plan for seismic re-evaluation of piping, mechanical, and electrical equipment important to safety. In general, the program plan presented satisfies the SEP requirements reasonably well with a few exceptions. For mechanical and electrical equipment, the following open items must be addressed by the licensee:

1. Provide sample calculations to justify the damping compatible with the stress levels used in the equipment evaluation.

2. The NRC position is that all equipment on the safe shutdown list should be qualified. NUSCO needs to clarify their intent on the remaining equipment not specified by Stevenson and Associates.

3. What are the capabilities to cool the spent fuel pool?

4. What safety related equipment is Westinghouse covering and what are their criteria and methods? What is the schedule for completion?

5. Provide soil properties for the evaluation of the field erected tanks, buried tanks, and buried piping. Justify the modeling in conjunction with these soil properties.

For the piping analyses being performed, the following items require additional attention:

1. The piping stress allowables are currently undecided. The NRC and NRC consultants will make a decision on this issue and will transmit the decision with the trip report.*

2. The licensee is requested to provide assurance that unsymmetrical bending will be addressed for piping supports where applicable.

3. The licensee is requested to provide clarification of support load combinations. It appears that the algebraic sum of weight plus thermal plus seismic is not always consistent with acceptable criteria.

4. What items is Westinghouse covering with regard to safety related piping? What is the schedule for completion?

In addition, the scope of SEP includes seismic re-evaluation of emergency power supply and distribution, instrumentation, and actuation systems. Clarification of the licensee's schedule for completion of this effort is requested.

---

* NRC and its consultants have made a decision as to what the piping allowable stresses should be. The allowables should correspond to ASME Code service level D (faulted) allowables. If the piping system is a Class 1 system and a Class 1 analysis is being performed, then the primary allowable stress should be 3.0 S_m. If the piping system is a Class 2 system and a Class 2 or ANSI 31.1 analysis is being performed, then the primary allowable stress should be 2.4 S_h. If the piping system is a Class 1 system and a Class 2 or ANSI 31.1 analysis is being performed, then the primary allowable stress should be 1.8 S_h to account for the difference in stress indices between the two types of analyses where:

S_m = ASME Code allowable stress intensity

S_h = ASME Class 2 or ANSI 31.1 allowable stress.

APPENDIX B

PIPING STRESS ANALYSIS PROCEDURE FOR SEISMIC QUALIFICATION OF SAFETY RELATED PIPING AT CONNECTICUT YANKEE

8-1

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

short sections
8 #9

PIPING STRESS ANALYSIS PROCEDURE

FOR

SEISMIC QUALIFICATION OF SAFETY RELATED PIPING

AT

CONNECTICUT YANKEE

PRELIMINARY

PREPARED BY: ______________________ DATE ____________
Thomas J. Mawson
Piping Systems Engineering
Generation Engineering Department

REVIEWED BY: ______________________ DATE ____________

APPROVED BY: ______________________ DATE ____________

NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
THE HARTFORD ELECTRIC LIGHT COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

PSE PROCEDURE FOR CY

SEISMIC QUALIFICATION OF SAFETY RELATED PIPING

PRELIMINARY

1.0 PURPOSE

This procedure provides an outline of the criteria and methodology to be employed by Generation Piping Systems Engineering in the seismic qualification of Category I safety related piping at Connecticut Yankee.

2.0 SCOPE

This procedure applies to all aspects of the system engineering evaluation including the following.

2.1 Piping analysis.

2.2 Evaluation of the adequacy of existing supports and design of subsequent modifications.

2.3 Design of new support functions as determined by the piping analysis.

2.4 Review of equipment nozzle loads.

2.5 Review of fabricated branch connections.

3.0 REFERENCES

3.1 ANSI B31.1 Power Piping Code, 1973 Edition, Summer 1973 Addenda.

3.2 AISC Specification, Manual for Steel Construction, 7th Edition, 1970.

3.3 ACI Standard 318-77, "Building Code Requirements for reinforced Concrete".

3.4 CYS-579, Revised December 10, 1965, "Specification for Shop Fabricated Piping for Secondary Plant and Primary Plant Waste Disposal and Other Miscellaneous Systems".

3.5 CYS-1550, Revised July 21, 1965, "Specification for Shop Fabricated Nuclear Piping".

3.6 CYS-579A, Revised January 7, 1966, "Supplement to Piping Specifications CYS-579 and CYS-1550 Covering Field Erection".

3.7 Welding Research Counsil (WRC) Bulletin 107, "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings".

3.8 American Petroleum Institute (API) Standard 610, "Centrifugal Pumps for General Refinery Services".

3.9 W. G. Counsil letter to D. M. Crutchfield, dated August 11, 1981.

3.10 D. M. Crutchfield letter to W. G. Counsil, dated September 28, 1981.

3.11 CYS-500, Revised May 20, 1966, "Summary of Design Conditions".

3.12 NUSCO Procedure GE&C 4.04, "Preparation, Review, Approval, and Control of Design Analyses, Technical Evaluations, and Manual and Computer Calculations".

3.13 American National Standard ANSI N45.2.11, Section 4.2.

3.14 Northeast Utiltiies Quality Assurance Program Topical Report, QAP 6.0.

3.15 Connecticut Yankee - Inservice Inspection Boundary Diagrams.

PRELIMINARY

4.0 NOMENCLATURE

4.1 Line Designations

Piping is identified on the various isometrics, plans, and sections by a unique line designation. The line designation is comprised of a line size and a line number in conjunction with a fluid designation and pipe class. Examples are given below.

Size	Fluid Designation	Pipe Class	Line Number
10"	AC	601R	56
2"	CH	151N	186
12"	WFPD	601	10

Once the pipe class is identified, the pipe schedule and material can be determined from References 3.4 or 3.5.

PRELIMINARY

4.2 Support Function

Type	Isometric Abbreviation
Anchor	Anc
Lateral Restraint	Lat
Axial Restraint	Axial
Vertical Restraint	Vert
Spring Hanger	S.H.
Rod Hanger	R.H.
Sliding Support	S.S.
Lateral Shock Suppressor	LSS
Axial Shock Suppressor	ASS
Vertical Shock Suppressor	VSS
Vertical Support	VS

Multiple support functions are represented by a combination of the above symbols; e.g., VERT-LAT.

5.0 GENERAL

Safety related piping systems will be divided into several individual stress problems based on analytical terminal points, such as structural anchors and equipment nozzles. For each stress problem the piping geometry will be based on as-built isometric piping drawings developed under the I&E Bulletin 79-14 program.

Support information shall be derived from the I&E Bulletin 79-02 and I&E Bulletin 79-14 hanger inspection packages.

6.0 MODELING/TECHNICAL CONSIDERATIONS

6.1 Single Acting Restraints

Single acting restraints, such as rod hangers and sliding supports, shall be evaluated on an individual basis applying the following method.

PRELIMINARY

A deadweight and "free" thermal analysis shall be run with all single acting restraints removed to determine pipe movement under both load cases. This unrestrained movement will then be combined at each restraint location. The restraint shall be considered active only if the calculated net displacement is negative (i.e., downward).

Single acting restraints will not be considered in the seismic analysis. However, during a seismic event an additional load could be imposed on these restraints. The evaluation of single acting restraints, therefore, will include a seismic load equal to a dynamic load factor of 1.5 times the deadweight reaction times the peak value of acceleration from the appropriate floor response spectra.

6.2 Wall/Floor Penetrations

Thimble drawings shall be reviewed to determine gaps around the pipe. Thermal movement at this location will be compared with the gap to determine if the penetration will impede the total thermal movement. If so, then the penetration shall be incorporated into the thermal analysis as a restraint in the appropriate direction once the gap is closed. DISPLACE CARD FOR MOVEMENT

~~SAME THING FOR SE~~ NOT CONSIDERED FOR SEISMI

Where the pipe is embedded in the concrete, the penetration will be considered to act as one of the following restraints for all load cases. 3/32

6.2.1 Full anchor: for embedded lines with anchor rings or lugs welded to the pipe.

6.2.2 Four-way restraint with no axial and torsional restraint; for embedded lines without welded rings or lugs.

6.3 Valves

Dimensions, weight, and center of gravity of valves shall be taken from the original valve drawing. In many instances, however, individual valve drawings are not available. In lieu of original drawings, valve properties may be obtained from other appropriate sources, such as a standard valve drawing, the manufacturer's catalog, or other plant records.

Where the valve wall thickness cannot be determined, then an assumed value of four (4) times the nominal pipe wall thickness shall be used.

For piping two inches (2") nominal and larger, the valve center of gravity will be modeled to consider the effects of the eccentric mass. If the center of gravity is not given, then one will be calculated using the methods shown in Figure 1.

6.4 Non-Standard Fittings

Stress intensification factors for non-standard fittings such as weldolets shall be obtained from the manufacturer. If manufactorer's data is not available, then the stress intensification factor may be determined by engineering judgement.

Reinforcement area of fabricated branch connections shall be reviewed in accordance with ANSI B31.1, Paragraph 104.3.1.

6.5 Flanged Joints

Flanged joints shall be evaluated to include the effect of moments and forces acting on the joint as a result of load conditions other than internal pressure.

The methods of ASME Section III, Subsection NC, 1980 Edition, Winter 1980 Addenda, Paragraph NC-3658, shall be used for this review.

6.6 Branch Lines

Where the moment of inertia of the run pipe is a minimum factor of ten (10) times greater than than of the branch line, the branch line may be analzyed separately. For these cases the run will then be considered to act as an anchor with respect to the branch line.

6.7 Anchors

PRELIMINARY

PRELIMINARY

7.0 LOAD CONDITIONS

7.1 Primary Loads

7.1.1 Pressure

The longitudinal pressure stress shall be calculated using the system design pressure.

7.1.2 Weight

The weight of piping and components, insulation, and the contents under normal operating conditions, shall be considered. Using the line designation, this information can be determined from the pipe fabrication specifications and a table of standard pipe properties.

NOTE: Hydrostatic test loads will ~~now~~ not? be considered in this evaluation.

Spring hangers shall be represented in the analysis as an external force equal to the "HOT" load. The "HOT" load will be determined from the IEB 79-14 hanger inspection information. Where the inspection package specifies a "COLD" load, then the "HOT" load shall be calcualted as follows.

HOT Load = COLD Load - (thermal displacement)
x (spring constant)

This information shall be recorded on the Spring Hanger Summary Sheet (Attachment 1) and the results evaluated to determine if the load range is within the load carrying capability of the spring. If the calculated load range falls outside the spring range, then adjustment or replacement of the spring shall be considered.

7.1.3 Seismic

The structural integrity of safety related piping under a safe shutdown earthquake will be evaluated using one of the following methods.

7.1.3.1 Dynamic

A dynamic analysis may be performed using lumped mass dynamic models with the appropriate amplified floor response spectra as input. Zero period acceleration (ZPA); i.e., missing mass effects shall be considered as a static load case in conjunction with the inertial response of the system.

Where the stress problem involves piping that is supported at different floor elevations or runs between two separate structures, the response spectra in each direction shall be an envelope of the applicable floor spectra.

The analysis shall consider a simultaneous input of the two horizontal components and the one vertical component of the earthquake. Results for each mode shall be obtained by the square root sum of the squares (SRSS) method.

Output from the dynamic analysis shall be reviewed to determine the cutoff frequency for each direction. This will coincide with the mode for which the deflection in the given direction is less than .001 inches. The corresponding modal effective mass fraction can now be established.

To account for the effects of missing mass a static load case shall be performed for each of the two horizontal and the one vertical direction. The equivalent "g" value for this load case will be equal to the quantity one minus the modal effective mass fraction times the "g" value corresponding to the cutoff frequency from the appropriate floor response spectra. The results shall then be combined by the SRSS method.

7.1.3.2 Static

In lieu of dynamic analysis an equivalent static seismic analysis may be performed. Each of the two horizontal and the vertical shall be addressed in a separate static analysis. The results shall then be combined by the SRSS method.

The equivalent static "g" loading shall be calculated by multiplying the maximum value of acceleration from the appropriate floor response spectra by a dynamic load factor of 1.5.

PRELIMINARY

7.2 Secondary Loads

7.2.1 Thermal

Forces and moments resulting from thermal expansion or contraction including the thermal displacements of nozzles and anchors shall be evaluated. The analysis shall take into account the complete range of system and plant operation.

Clearances between the pipe and a rigid restraint or building penetration shall be compared to the unrestrained thermal movement of the piping to determine if the pipe movement is restricted. Where the "free" thermal movement is less than the clearance the restraint or penetration need not be considered in the thermal analysis.

7.2.2 Seismic Anchor Movement

The effects of relative seismic anchor displacements shall be considered in the evaluation. Movements will be assumed to occur out-of-phase between anchor points.

The results of this load case shall be combined with the results of the thermal analysis.

PRELIMINARY

APPENDIX C

CONNECTICUT YANKEE ATOMIC POWER COMPANY SAFETY
RELATED PIPING SEISMIC QUALIFICATION PROGRAM CRITERIA

CONNECTICUT YANKEE

ATOMIC POWER COMPANY

SAFETY RELATED PIPING

SEISMIC QUALIFICATION PROGRAM

CRITERIA DOCUMENT

REVISION 1

AUGUST 10, 1982

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ANALYSIS OF SAFETY-RELATED PIPING SYSTEMS

A. SCOPE

The purpose of this document is to present the analytical methods and stress criteria which will be used for the Connecticut Yankee safety related piping seismic qualification program. The program will include static analysis of the piping system/support scheme for maximum operating thermal, pressure, and deadweight loads, 1
along with dynamic analysis for seismic loads. Stress criteria will be presented for the piping and supports.

B. BACKGROUND

In the years since the Connecticut Yankee generating station was designed, seismic analysis methods have become more rigorous and the ASME Boiler and Pressure Vessel Code (BP&V) Section III, Nuclear Power Plant Components, has been published reflecting changes in analysis, design, and quality control techniques. The purpose of this criteria document is to establish requirements for performing the upgrading seismic analyses of safety related piping systems applying current technology.

The original design criteria used for analysis of this plant's safety related piping systems was the 1955 Edition of the American Standard Code for Pressure Piping, ASA B31.1.

For the purposes of this document, safety related piping shall be considered to consist of portions of those systems listed herein including connecting piping two and one-half inches (2½") or larger nominal pipe size, up to and including the first valve that is normally closed or is capable of automatic closure during all modes of reactor operation.

1. Reactor coolant loop attachments -- Up to first isolation valve.

2. Main steam -- Up to and including the outermost containment isolation valve.

3. Feedwater -- Up to and including the outermost containment isolation valve.

4. Auxiliary feedwater -- From the primary feedwater lines to the demineralized water storage tank.

5. Residual heat removal.

6. Low pressure safety injection -- Residual heat removal system up to the refueling water storage tank.

7. High pressure safety injection -- Reactor coolant loops up to the refueling water storage tank.

8. Chemical and volume control -- Reactor coolant loops up to the volume control tank, volume control tank up to the boric acid tank (supply lines only).

9. Service water -- Supply lines to safety related equipment.

10. Fuel oil -- Emergency diesel generators to the emergency diesel generator storage tanks (TK-33-2A and 2B).

11. Compressed air -- Emergency diesel generators starting air motors up to air compressors C-14-1A and 1B.

C. LOADING CONDITIONS

Plant safety related piping and associated supports/restraints will be analyzed for the following loading conditions.

1. Design pressure, deadweight, and maximum operating temperature range. 1

2. Safe shutdown earthquake (SSE) combined with operating pressure and deadweight.

3. Analyses will not consider coincident LOCA and SSE. 1

D. STRESS CRITERIA

1. Above Ground Piping 1

(a) General 1

The piping analysis that will be performed for the Connecticut Yankee evaluation is based on the rules of the ANSI B31.1 Power Piping Code, 1973 Edition, Summer 1973 Addenda.

The loading combinations and associated stress limits to be used for the piping systems, which are part of the seismic qualification program, are given in Table 1. The stress limits used for the SSE condition correspond to faulted condition allowables as defined in the ASME Section III Code. The piping stresses are to be calculated using formulas given in ANSI B31.1, 1973 Edition, Summer 1973 Addenda. 1

Piping materials are to be identified on the basis of the Connecticut Yankee piping line list and pipe fabrication specifications.

(b) Supports/Restraints 1

Existing supports/restraints will be evaluated using the as-built configuration to determine if the assembly is capable of sustaining the new piping analysis loads.

Where the new loads exceed the original design load, the support/restraint will be reviewed to determine if modifications are required.

Additional piping support/restraint functions that are identified in the piping analysis shall be designed to meet the required load capacity.

Evaluation of component standard supports will be performed using manufacturer's published allowable load data. Piping support/restraint structural steel will be reviewed/designed in accordance with the AISC Specification, "Manual for Steel Construction", 7th Edition, 1970.

The effects of friction shall be considered where the thermal movement of the pipe relative to the piping support exceeds one-sixteenth inch (1/16"). The coefficient of friction to be used in this analysis is 0.3 for steel-on-steel. The frictional force acting on the structure shall be equal to the greater of the deadweight load or the deadweight plus thermal load, multiplied by the coefficient of friction.

C-6

(c) Small Bore Piping 1

Small diameter (2½" nominal and less) piping systems needed for the safe operation of the plant will be reviewed using chart methods which are demonstrated to result in piping stresses within Code allowable limits; i.e., same as D.1. Computer analyses may be performed on select small bore lines.

Vent, drain, and sampling lines which are not considered vital to safe system operation are not included in this program.

2. Underground Piping

The analysis of underground piping conforms to the criteria outlined by Newmark and Hall[A] and to the method proposed by E. C. Goodling[B], and by Shah and Chu.[C]

This method addresses primarily the axial stresses induced in pipe runs which are parallel to the direction of soil strain as recommended by Wang.[D] Since a buried pipeline reacts to seismic inputs through the medium of the surrounding soil, its response behavior is influenced by its physical parameters and by the governing geotechnical and seismological parameters. These parameters are manipulated to determine the forces, moments, and 1
stresses on the pipe element. This method neglects strains induced by ground curvature as recommended by Newmark. Also, since the dynamic effects on buried piping response have been found to be negligible, they are not considered.

This analysis method involves four (4) distinct phases.

1. Physical, geotechnical, and seismological data are assembled. The physical and seismological data are readily available, but because Connecticut Yankee was designed before the emphasis on rigorous seismic analysis developed, there may be some difficulty obtaining the geotechnical data from the plant site. However, using engineering judgements, data available from a geologically similar site, will be combined with tabulated parameter values for the soil covering the pipes to form the geotechnical data base.

2. Intermediate parameters such as the soil pipe interaction constant and maximum length, L', over which friction effects 1 are important, are calculated.

3. Pipes are then classified as long or short based on the magnitude of L' relative to the length at which there is negligible additional influence on the forces and moments. This classification allows for flexibility assumptions which will facilitate the evaluation of stresses.

4. Finally, deflections, forces, moments, and code stresses are calculated and compared to allowables.

E. ANALYSIS PROCEDURES

1. Computer Codes

In general, the following QA qualified computer codes will be utilized in this analysis.

Piping Stress	ADLPIPE	Version 1D
Support/Restraint Structure	STRUDL	Version 2 Model 2

If necessary, other QA qualified programs may be employed.

2. Response Spectrum Analysis Procedures

Analysis will be performed assuming that the seismic event is initiated with the plant at normal full power condition. The damping values that will be used for the SSE condition are shown below as a percent of critical damping.(E) Since the piping can be supported at different floor elevations within the containment building, the response spectrum in each direction shall be an envelope of the applicable floor spectra. The floor response spectra* utilized in this analysis is based on ground response spectra previously submitted for NRC review.(F)

Large diameter piping systems, pipe diameter greater than twelve inches (12") nominal. (3%)

Small diameter piping systems, diameter equal to or less than twelve inches (12") nominal. (2%)

The analysis shall be performed with a simultaneous input of the two (2) horizontal components and one (1) vertical component of the earthquake. The modal response for each item of interest (i.e., force, displacement, stress) shall be obtained by the square root of the sum of the squares method.

*The ground response spectra used to develop the floor response spectra is very close to the Staff's ground response spectra forwarded by D. M. Crutchfield's letter to all SEP owners (except San Onofre) on June 8, 1981.

$$R_T = \{ \sum_{i=1}^{3} R_i^2 \}^{1/2}$$

where: $$R_i = \{ \sum_{j=1}^{N} R_{ij}^2 \}^{1/2}$$

where: R_T = total combined response at a point

R_i = value of combined response of direction i

R_{ij} = absolute value of response for direction i, mode j

N = total number of modes considered

For systems having modes with closely spaced frequencies, the above method shall be modified to include the possible effect of these modes. Combined total response for system which have such closely spaced modal frequencies will be obtained in accordance with Regulatory Guide 1.92.

F. MODELING TECHNIQUES

The piping system and support scheme are to be represented by an ordered set of data which numerically describes the physical system.

Each system will be analyzed by one or several stress problems. In general, the analytical terminal points for each stress problem will be one of the following, based upon engineering judgement.

1. Equipment nozzle.

2. Piping interface where the moment of inertia of the run pipe exceeds that of the connecting line by a minimum factor of ten.

3. An anchor; i.e., a six-way restraint.

4. Two or more restraints in proximity such that the effects of the piping on either side of the support group are isolated. 1

The spatial geometric description of the model is to be based upon the as-built isometric piping drawings developed under the I&E Bulletin 79-14 program. Node point coordinates and incremental lengths of the members are determined from these drawings. The geometrical properties along with the modulus of elasticity, E, the coefficient of thermal expansion, α, the average temperature changes from the ambient temperature, ΔT, and the weight per unit length, ω, are specified for each element. Supports are represented by a stiffness applied in the appropriate direction to define the restraint characteristics of the supports.

The models used in the static analyses are to be modified for use in the dynamic analyses by including the mass characteristics of the piping. The lumping of the distributed mass of the piping systems is to be accomplished by locating the total mass at points in the system which will approximately represent the response of the distributed system.

The effect of eccentric masses, such as valves and extended structures, are considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis and the moments caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria applicable to the safety class of the piping.

TABLE 1

LOADING COMBINATIONS AND STRESS LIMITS FOR PIPING

LOADING COMBINATIONS	STRESS LIMITS	
1. Normal:		
(a) Design Pressure + Deadweight	$\leq S_h$	
(b) Maximum Operating Temperature + Seismic Anchor Movements	$\leq S_A$	1
or		
Design Pressure + Deadweight + Maximum Operating Temperature + Seismic Anchor Movements	$\leq (S_H + S_A)$	1
2. SSE:		
(a) Maximum Operating Pressure + Deadweight + Maximum Potential Earthquake Loads (SSE)	$\leq 2.4\ S_h$	1

Where: S_A = allowable stress range

$= 1.25\ S_c + .25\ S_h$

S_c = material allowable stress at minimum temperature from ANSI B31.1, 1973 edition, summer 1973 addenda

S_h = material allowable stress at design temperature from ANSI B31.1, 1973 edition, summer 1973 addenda 1

TABLE 2

LOADING COMBINATIONS AND STRESS LIMITS FOR SUPPORTS

LOADING COMBINATION	LINEAR TYPE SUPPORT LIMITS[3]	PLATE AND SHELL SUPPORT LIMIT	
Greater of:			
D + T or D	Working Stress[4]	$P_m \leq 1.0\ S_m$	1
		$P_m + P_b \leq 1.5\ S_m$	
Greater of:			
D + T + E or D + E	Within lesser of:	$P_m \leq 1.2\ F_y$[1]	1
	$\frac{1.2\ F_y}{F_t}$ or $\frac{0.7\ S_u}{F_t}$	$P_m + P_b \leq 1.8\ F_y$[2]	
	times working limits[4]		

Where: D = deadweight 1

T = thermal maximum operating temperature 1

E = SSE

F_y = material yield strength

F_t = allowable tensile stress per ASME Section III, Appendix XVII

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[1]Not to exceed $0.7\ S_u$

[2]Not to exceed $1.05\ S_u$

[3]Compressive axial member loads should be kept to less than 0.9 times the critical buckling load

[4]Working stress allowables per Appendix XVII of ASME III.

REFERENCES

A. Newmark, N. M., and Hall, W. J., 'Development of Criteria for Seismic Review of Selected Nuclear Power Plants", N. M. Newmark Consultant Engineering Services, Urbana, Illinois, 1977.

B. Goodling, Jr., E. C., "Flexibility Analysis of Buried Pipe", American Society of Mechanical Engineer, New York, 1978.

C. Shah, H. H., and Chu, S. L., "Seismic Analysis of Underground Structural Elements", Journal of the Power Division, Proceedings of ASCE, Vol. 100, No. P01, July 1974.

D. Ru-Liang Wang, L., "Some Aspects of Seismic Restraint Design of Buried Pipelines", Presented at Third National Congress on Pressure Vessels and Piping, San Francisco, California, 1979.

E. "Damping Values of Nuclear Plant Components", Regulatory Guide 1.61.

F. W. G. Counsil letter to D. M. Crutchfield, dated September 15, 1980.

APPENDIX D

CONNECTICUT YANKEE PIPING REEVALUATION STATUS REPORT

PA 81-057

STATUS REPORT: CONNECTICUT YANKEE PIPING REEVALUATION DATE: JUNE 18, 1982

SYSTEM	PIPING STRESS PROBLEMS: ACCESSIBLE QUANTITY	ACCESSIBLE % COMPLETE	INACCESSIBLE QUANTITY	INACCESSIBLE % COMPLETE	COMBINED QUANTITY	COMBINED % COMPLETE	SUPPORTS: ACCESSIBLE QUANTITY	ACCESSIBLE % COMPLETE	INACCESSIBLE QUANTITY	INACCESSIBLE % COMPLETE	COMBINED QUANTITY	COMBINED % COMPLETE	REVIEW & APPROVAL OF STRESS REPORT (%)	WEIGHTED PERCENT COMPLETE OF SYSTEM EVALUATION	COMMENTS
MAIN STEAM	2	58	4	100	6	86	62		16	100	78	20	66	74	
FEEDWATER	6	78	4	100	10	87	89		40	100	129	31	40	67	
AUXILIARY FEEDWATER	3	64	1	93	4	71	25		2		27		25	50	
RESIDUAL HEAT REMOVAL	2	66	2	65	4	66	61		68		129			36	
CHEMICAL & VOLUME CONTROL	6	8	5	64	11	33	200		34		234			22	
SAFETY INJECTION	3	55	3	89	6	72	42		21	19	63	6	17	49	
SERVICE WATER	10	48	8	100	18	71	212		4		216			43	
REACTOR COOLANT	-		6	18	6	18	-		71		71			11	
EMERGENCY DIESEL GENERATOR	6	55	-		6	55	33		-		33			33	
TOTAL	38	48	33	70	71	57	724		256	22	980	6	15	43	

NOTES:
1. ANALYSIS OF REACTOR COOLANT LOOPS INCLUDING SURGE LINE BY WESTINGHOUSE (CJGLADDING).
2. ANALYSIS OF PIPING TO PRESSURIZER CODE SAFETY VALVES AND ION PRESSURE RELIEF VALVES BY S&W (LFSTARNER).
3. ANALYSIS OF ANNULUS PIPE RACKS IS ESTIMATED TO BE 10% COMPLETE.
4. NEXT REPORT 7/9/82.

APPENDIX E

REEVALUATION GUIDELINE FOR SEP GROUP II PLANTS
(EXCLUDING STRUCTURES)

REEVALUATION GUIDELINE
FOR
SEP GROUP II PLANTS
(EXCLUDING STRUCTURES)

INTRODUCTION

In support of NRC's Systematic Evaluation Program (SEP) for Group II Plants, the following Reevaluation Criteria have been established. These criteria include recommended load combinations with allowable stresses and/or loads for piping systems, component supports, concrete attachments, and equipment. These criteria are based on linear elastic analyses having been performed. The acceptance criteria are generally based on the ASME Code. For situations not covered by these criteria, (i.e. items constructed of cast iron) compatible criteria shall be developed by the licensee and will be reviewed on a case-by-case basis. The licensee is requested to justify major deviations in criteria which appear less conservative than those specified used herein. 1

DEFINITIONS

Code = ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1980 Edition, Winter 1980 Addenda.

σ_m = General membrane stress. This stress is equal to the average stress across the solid section under consideration, excludes discontinuities and concentrations, and is produced only by mechanical loads. 1

σ_b = Bending stress. This stress is equal to the linear varying portion of the stress across the solid section under consideration, excludes discontinuities and concentrations, and is produced only by mechanical loads.

P_D = Design or maximum operating pressure loads and design mechanical loads.

SSE = Inertial loads due to Safe Shutdown Earthquake (SSE) and design mechanical loads where applicable.

T = Loads due to thermal expansion of attached pipe (constraint of free end displacement).

W = Loads due to weight effects.

AM = Loads due to SSE anchor movement effects.

S_{bk} = Critical buckling stress.

S_m = Allowable stress intensity at temperature listed in ASME Code.

S_y = Yield strength at temperature listed in ASME Code.

S_u = Ultimate tensile strength at temperature listed in ASME Code.

σ_ℓ = Local membrane stress. This stress is the same as σ_m except that it includes the effect of discontinuities. 1

S = ASME Code Class 2 allowable stress value. The allowable stress shall correspond to the metal temperature at the section under consideration.

P_m = General Primary Membrane Stress Intensity. This stress intensity is derived from the average value across the thickness of a section of the general primary stresses produced by design internal pressure and other specified Design Mechanical Loads, but excluding all secondary and peak stresses. Averaging is to be applied to the stress components prior to determination of the stress intensity values. 1

P_ℓ = Local Membrane Stress Intensity. This stress intensity is the same as P_m except that it includes the effects of discontinuities.

P_b = Primary Bending Stress Intensity. This stress intensity is derived from the linear varying portion of stresses across the solid section under consideration produced design pressure and other specified design mechanical loads. Secondary and peak stresses are not included. 1

SPECIAL LIMITATIONS

1. Critical buckling loads (stresses) must be determined taking into account combined loading (i.e., axial, bending, and shear), initial imperfections, residual stresses, inelastic deformation, and boundary conditions. Both gross and local buckling must be evaluated. Critical buckling loads (stresses) shall be determined using accepted methods such as those contained in NASA Plates and Shells Manual or ASME Code Case N-284. 1

2. Where stresses exceed material yield strength, it shall be demonstrated that brittle failures and detrimental cyclic effects are precluded, and that dynamic analysis assumptions are not nonconservatively affected. Where significant cyclic effects are identified, it shall be demonstrated that the structure or component is capable of withstanding ten full peak deformation cycles.

3. Where results of analysis indicate that the allowable stresses of the original construction code are exceeded in any of the load combinations specified herein, it shall be demonstrated that the in-situ item was designed and fabricated using rules compatible with those required for the appropriate ASME Code Class (Subsection NX2000,

4000, 5000, and 6000). In cases where compatibility with the appropriate ASME Code Subsections was not substantially achieved, appropriate reductions in these limits shall be established, justified, and applied.

ACCEPTANCE CRITERIA FOR PIPING

Using Code[a] Class 2 analytical procedures [Equation (9), 1
NC-3653.1], the following stresses are not to be exceeded for the specified piping:

Class 1: $P_m + P_b = |W + P_D| + |SSE| \leq 1.8\ S$ 1

Class 2: $P_m + P_b = |W + P_D| + |SSE| \leq 2.4\ S$

The effects of thermal expansion must meet the requirements of Equation (10) or (11) of NC-3653, including moment effects of anchor displacements due to SSE if anchored displacement effects are omitted from 1
Equation (9) of NC-3653. Class 1 analytical procedures (NB-3600) can also be utilized if appropriate allowable stresses specified in NB-3650 are used.

Branch lines shall be analyzed including the inertial and displacement input due to the response of the piping to which it is attached at the attachment point.

---

a. The references to ASME Code equation and paragraph numbers on this page correspond to the 1980 edition of the code, 1981 winter addenda. This was done in order to avoid confusion introduced by the initial 1980 edition of 1
the code which renumbered the equations differently from past and present editions of the code. Equation numbers presented on this page reflect common nomenclature utilized in the nuclear industry.

ACCEPTANCE CRITERIA FOR CLASS 1 COMPONENT SUPPORTS 1

Imposed Load Combinations	Acceptance Criteria[a] Linear	Acceptance Criteria[a] Plate and Shell[b]	
The higher of:			
$\lvert W \rvert$	Code Subsection NF Design, Level A, and Level B Limits	$P_m \le 1.0\ S_m$	
or			
$\lvert W + T \rvert$		$P_\ell + P_b \le 1.5\ S_m$	1
The higher of:	- - - - - - - - - -		
$\lvert W \rvert + \lvert SSE \rvert + \lvert AM \rvert$	Code Subsection NF Level D Limits	$P_m \le 1.5$ Sm or 1.2 Sy[c] not to exceed 0.7 Su	1
or			
$\lvert W + T \rvert + \lvert SSE \rvert + \lvert AM \rvert$		$P_\ell + P_b$ 2.25 Sm or 1.85 y[c] not to exceed 1.05 Su	

In addition to the above criteria, the allowable buckling stress shall be limited to 2/3 S_{bk}, where S_{bk} is determined in accordance with Special Limitation 1.

---

a. These load combinations shall be used in lieu of those specified in ASME Code Subsection NF. In addition, for brittle types of material not specified in the Code, appropriate stress intensification factors for notches and stress discontinuities shall be applied in the analysis.

b. The 1.5 Sm value from NB 3221 on which these are based (Code Appendix F 1323.1) shall be limited by Code Section NB 3221.3. 1

c. Use larger of.

ACCEPTANCE CRITERIA FOR CLASS 2 COMPONENT SUPPORTS

Imposed Load Combinations	Acceptance Criteria[a]	
	Linear	Plate and Shell
The higher of:		
$\lvert W \rvert$		$\sigma_\ell \leq 1.0\ S$
or	Code Subsection NF Design, Level A, and Level B Limits	
$\lvert W + T \rvert$		$\sigma_\ell + \sigma_b \leq 1.5\ S$
The higher of:	- - - - - - - - - - -	
$\lvert W \rvert + \lvert SSE \rvert + \lvert AM \rvert$		$\sigma_\ell \leq 1.5\ S$ or
or	Code Subsection NF Level D Limits	$0.4\ S_u$ (b)
$\lvert W + T \rvert + \lvert SSE \rvert + \lvert AM \rvert$		$\sigma_\ell + \sigma_b \leq 2.25\ S$ or
		$0.6\ S_u$ (b)

In addition to the above criteria, the allowable buckling stress shall be 1
limited to 2/3 S_{bk}, where S_{bk} is determined in accordance with Special Limitation 1.

---

a. These load combinations shall be used in lieu of those specified in ASME Code Subsection NF. In addition, for brittle types of material not specified in the Code, appropriate stress intensification factors for notches and stress discontinuities shall be applied in the analysis.

b. Use lesser of.

ACCEPTANCE CRITERIA FOR CONCRETE ATTACHMENTS

I. Concrete Expansion Anchor Bolts[a]

Load Combinations: Same as for component supports.

Acceptance Criteria:[b]

Wedge type: 1/4 ultimate as specified by manufacturer.

Shell type: 1/5 ultimate as specified by manufacturer.

II. Grouted Bolts: Replace[a],[b],[c]

III. Concrete Embedded Anchors[a]

Load Combinations: Same as for component supports.

Acceptance Criteria[b]: 0.7 S_u

---

a. Base plate flexibility effects must be considered.

b. Both pullout and shear loads must be considered in combined loading situations.

c. Unless stresses in the bolts and structure to which they are attached are shown to be sufficiently low to preclude concrete/grout/steel interface bond failures. Load combinations are the same as those for component supports.

ACCEPTANCE CRITERIA FOR CLASS 1 MECHANICAL EQUIPMENT

Component	Loading Combination[b]	Criteria[d] [g]
Pressure vessels and heat-exchangers	$\lvert W + P_D\rvert + \lvert SSE\rvert + \lvert \text{Nozzle Loads}\rvert$	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (e) $(P_m \text{ or } P_\ell) + P_b \leq 3.6\ S_m$ or $1.05\ S_u$ (e)
Active pumps and other mechanical components[a][d]	$\lvert W + P_D\rvert + \lvert SSE\rvert + \lvert \text{Nozzle Loads}\rvert$	$P_m \leq 1.2\ S_m$ or S_y (f) $(P_m \text{ or } P_\ell) + P_b \leq 1.8\ S_m$ or $1.5\ S_y$ (f)
Inactive pumps and other mechanical components	$\lvert W + P_D\rvert + \lvert SSE\rvert + \lvert \text{Nozzle Loads}\rvert$	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (e) $(P_m \text{ or } P_\ell) + P_b \leq 3.6\ S_m$ or $1.05\ S_u$ (e)
Active valves[a],[c],[d]	$\lvert W + P_D\rvert + \lvert SSE\rvert + \lvert \text{Nozzle Loads}\rvert$	$P_m \leq 1.2\ S_m$ or S_y (f) $(P_m \text{ or } P_\ell) + P_b \leq 1.8\ S_m$ or $1.5\ S_y$ (f)
Inactive valves[c]	$\lvert W + P_D\rvert + \lvert SSE\rvert + \lvert \text{Nozzle Loads}\rvert$	$P_m \leq 2.4\ S_m$ or $0.7\ S_u$ (e) $(P_m \text{ or } P_\ell) + P_b \leq 3.6\ S_m$ or $1.05\ S_u$ (e)
Bolt stress shall be limited to:		Tension = Sy or $0.7\ S_u$(e) Shear = 0.6 Sy or $0.42\ S_u$(e)

1

a. Active pumps, valves, and other mechanical components (e.g., CRDs) are defined as those that must perform a mechanical motion to accomplish a system safety function.

b. Nozzle loads shall include all piping loads (including seismic and thermal anchor movement effects) transmitted to the component during the SSE.

c. Scope and evaluation of pumps and valves are to be in accordance with NB 3411, NB 3412, and NB 3546 of the Code, including seismic and thermal anchor movement effects. 1

d. For active mechanical equipment contained in safe shut down systems, it shall be demonstrated that deformation induced by the loading on these pumps, valves and other mechanical components (e.g., CRDs) do not introduce detrimental effects which would preclude function of this equipment following a postulated SSE event. For valve operators integrally attached to valve bodies, binding can be considered precluded if stresses in the valve body and operator housing and supports are shown to be less than yield. In these evaluations, all loads (including seismic and thermal anchor movement effects) shall be included. 1

e. Use lesser of two values.

f. Use greater of two values. 1

g. The 1.5 Sm value from NB 3221 on which these are based (Code Appendix F 1323.1) shall be limited by Code Section NB 3221.3.

ACCEPTANCE CRITERIA FOR CLASS 2 MECHANICAL EQUIPMENT

Component	Loading Combination[b]	Criteria[d]	
Pressure vessels and heat-exchangers	$W + P_D + SSE$ + Nozzle Loads	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_\ell) + \sigma_b \leq 2.4\ S$	
Active pumps and other mechanical components[a],[d]	$W + P_D + SSE$ + Nozzle Loads	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_\ell) + \sigma_b \leq 1.8\ S$	
Inactive pumps and other mechanical components	$W + P_D + SSE$ + Nozzle Loads	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_\ell) + \sigma_b \leq 2.4\ S$	
Active valves[a],[c],[d]	$W + P_D + SSE$ + Nozzle Loads	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_\ell) + \sigma_b \leq 1.8\ S$	1
Inactive valves[c]	$W + P_D + SSE$ + Nozzle Loads	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_\ell) + P_b \leq 2.4\ S$	
Bolt stresses shall be limited to:		Tension = Sy or $0.7\ S_u$[e] Shear = 0.6 Sy or $0.42\ S_u$[e]	

---

a. Active pumps, valves, and other mechanical components (e.g., CRDs) are defined as those that must perform a mechanical motion to accomplish a system safety function.

b. Nozzle loads shall include all piping loads (including seismic and thermal anchor movement effects) transmitted to the component during the SSE.

c. Scope and evaluation of pumps and valves are to be in accordance with NC 3411, NC 3412, and NC 3521 of the Code, including seismic and thermal anchor movement effects.

d. For active mechanical equipment contained in safe shut down systems, it shall be demonstrated that deformation induced by the loading on these pumps, valves and other mechanical components (e.g., CRDs) do not introduce detrimental effects which would preclude function of this equipment following a postulated SSE event. For valve operators integrally attached to valve bodies, binding can be considered precluded if stresses in the valve body and operator housing and supports are shown to be less than yield. In these evaluations, all loads (including seismic and thermal anchor movement effects) shall be included. 1

e. Use lesser of two values.

ACCEPTANCE CRITERIA FOR TANKS

Load Combinations: Normal Operating Loads + SSE Inertia Loads + Dynamic Fluid Pressure Loads[a]

Acceptance Criteria: Smaller of S_y or 0.7 S_u. In addition, the allowable buckling stress shall be limited to 2/3 S_{bk}, where S_{bk} is determined in accordance with Special Limitation 1.

---

a. Dynamic fluid pressure shall be considered in accordance with accepted and appropriate procedures; e.g., USAEC TID-7024. Horizontal and vertical loads shall be determined by appropriately combining the loads due to vertical and horizontal earthquake excitation considering that the loads are due to pressure pulses within the fluid. These loads shall also be applied, in combination with other loads, in tank support evaluations.

APPENDIX F

LUMPED MASS LOCATION--A SUGGESTED GUIDE FOR ADLPIPE USERS

F-1

Reference 16
Revised 9/80

LUMPED MASS LOCATION

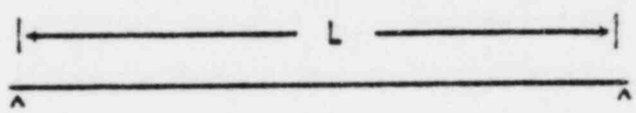
A Suggested Guide for ADLPIPE Users

Piping systems are usually composed of flexible components with distributed mass. A shock loading will cause this system to vibrate and the amplitude of vibration will be dependent on two parameters: (1) the natural frequencies, and (2) the energy distribution in the shock spectra.

The natural frequencies and mode shapes of the system are dependent on the mass and stiffness of the piping, which, in turn, is dependent on the support locations and material properties. Imposing a shock loading (earthquake) to a piping system excites all the natural frequencies and an accelerometer would record a response which was composed of a combination of modal motion. Shock loadings of earthquakes cause the low frequencies of the pipe system to be excited, as the earth motion is dominant in a band of 0.1 to 20 hz. Earth motion seems to attenuate above 50 hz.

An approximate calculation of the system natural frequencies can be made by considering the distributed mass of the piping to be lumped at key points in the system. (This is the ADLPIPE technique.*) Questions arise continually about the location of these lumped masses as ADLPIPE has the capability of selecting the lumped mass points. A spacing criteria is set forward here which is based on accurate computation of the frequencies in the significant frequency bandwidth of 0.1 to 50 hz.

Take, for instance, a long, straight piping system supported at the ends.



The natural frequencies of this distributed mass system are:

$$\omega_n = a_n \sqrt{\frac{EI}{mL^4}} \qquad (1)$$

where $m = \frac{w}{g}$

w = unit weight, #/in
g = 386 in/sec^2
L = length, in
E = modulus, lb/in^2
$I = \pi(d_o^4 - d_i^4)/64$
d_o = outside diameter, in
d_i = inside diameter, in
ω_n = frequency, rad/sec

*The ADLPIPE technique used to compute the mass magnitude is explained in the Appendix.

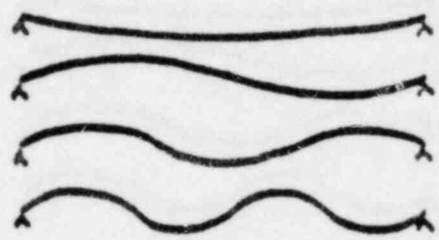
The coefficient, a_n, has an infinite number of values; the first four which represent the lowest frequencies and mode shapes are:

$n = 1$ $\qquad a_1 = \pi^2$

$n = 2$ $\qquad a_2 = 4\pi^2$

$n = 3$ $\qquad a_3 = 9\pi^2$

$n = 4$ $\qquad a_4 = 16\pi^2$



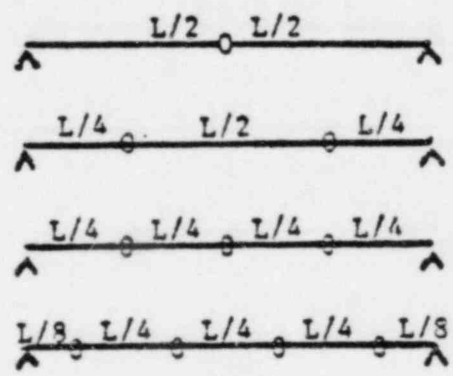
This system can be approximated by one or more lumped masses. Equation (1) remains the same if the masses are located as shown below:

one mass equal to $\frac{mL}{2}$

two masses, each equal to $\frac{mL}{2}$

three masses, each equal to $\frac{mL}{4}$

four masses, each equal to $\frac{mL}{4}$



The coefficients for the first four modes (one mode per mass degree of freedom) are:

	a_n uniform	a_n 1 mass	a_n 2 masses	a_n 3 masses	a_n 4 masses
n = 1	9.896	9.797	9.7964	9.867	9.867
= 2	39.478		27.626	39.185	39.185
= 3	88.826			83.088	83.138
= 4	157.91				110.85

Inspection of the table shows that the error is slight for the first frequency. As the number of masses are increased, the error decreases for the lower frequencies. Three and four masses give highly accurate frequencies for the first two modes with increasing error in the higher frequencies.

Therefore, the programmer can model the long beam-like piping system with three masses if the third frequency is <u>above</u> the frequency bandwidth which is considered to be significant.

For example, most earthquakes cut-off at 50 hz. Substituting a conservative 100 hz for the cut-off frequency:

$$\omega_{cutoff} = 88.826 \sqrt{\frac{EI}{mL^4}} \quad \text{(third mode)}$$

$$L^4 = \left[\frac{88.826}{\omega_{cutoff}}\right]^2 \cdot \frac{EI}{m}$$

$$\omega_{cutoff} = 628.3 \text{ rad/sec}$$

$$L^4 = 0.02 \frac{EI}{m}$$

The mass spacing will be $\frac{L}{4}$, therefore:

$$L^4_{span} = 7.81 \quad 10^{-5} \frac{EI}{m} \qquad (2)$$

ADLPIPE will calculate the lower two frequencies with an error of less than 2% with this mass spacing:

Numerical Example:

8 inch, schedule 40 steel pipe

d_o = 8.625"
d_i = 7.981"
I = 72.5 in^4
E = 30.10^6
w = 28.56 lb/ft = 2.38 lb/in

$$L^4_{span} = \frac{7.81 \quad 10^{-5} (30.10^6)(72.5)}{\frac{2.38}{386}}$$

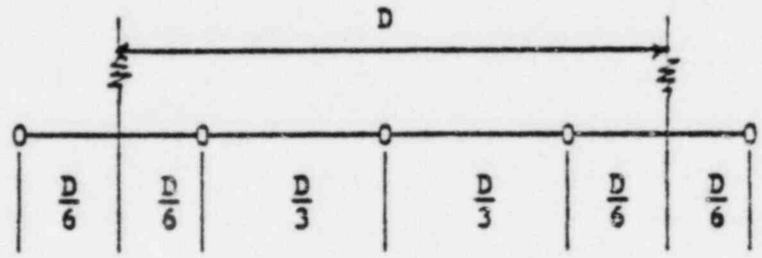
L_{span} = 72.4 in, or approximately 6 ft

An ADLPIPE problem with the above data gives the following results:

	Eq (1)	ADLPIPE 1 mass	ADLPIPE 3 masses
f_1	11.107 cps	10.991	11.073
f_2	44.429		43.551
f_3	99.966		90.627

Heavy components such as valves should be individually lumped. The above mass point spacing should be used for the connecting pipe.

Hanger spacing is important also and has to be taken into effect. If hanger spacing is less than the calculated lump mass spacing, then the mid-span spacing should be reduced so that one mass is between two hangers. If hanger spacing is greater than the calculated mass spacing, then the number of lumps should be spaced between hangers so that the actual spacing is less than the calculated mass spacing, L_{span}. For example, in the sketch below of three masses, this spacing is satisfactory if D/3 is less than L_{span}.



APPENDIX

Automatic Lumping in ADLPIPE

If a uniform beam is modeled by a single lumped mass of magnitude, M, the natural frequency may be calculated:

$$\omega_1 = \sqrt{\frac{K}{M}} \qquad (3)$$

$$K = \frac{48EI}{L^3}$$

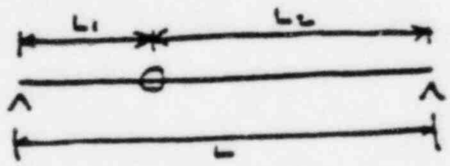
The first frequency of the uniform beam equation (1)

$$\omega_1 = 9.896\sqrt{\frac{EI}{mL^4}}$$

Set $\omega_1 = \omega_1$, and solve for M

$$M = .4901\, mL \approx \frac{mL}{2}$$

Based on this relationship, the lumped mass at a point calculated by ADLPIPE is equal to one half the span $(\frac{L}{2})$ times the distributed mass of the uniform beam. It is possible with the WEIGHT card or the VALVE card to have a concentrated mass between the two lumped mass points. ADLPIPE lumps the concentrated mass as follows:



L = span between lumped masses

L_1, L_2 location of concentrated mass (not a lumped mass point)

The frequency of a simply supported beam with the mass on center is equation 3:

$$\omega_1 = \sqrt{\frac{48EI}{M_c L^3}}$$

The frequency of a simply supported beam with this mass off center is:

$$\omega_1 = \sqrt{\frac{3EIL}{M_o L_1^2 L_2^2}}$$

Let $A = 2L_1/L$ and solve for the ratio of the masses

$$\frac{M_c}{M_o} = A^2 (2 - A)^2$$

This relationship has been used to accumulate the effect of concentrated masses not at lumped mass points.

For a mass off center at the quarter span $L_1 = \frac{L}{4}$, A = .5 the magnitude to the center mass to obtain the same frequency is:

$$M_c = .5625\ M_o$$

APPENDIX G

NSSS PRELIMINARY RESULTS

SEISMIC REEVALUATION OF THE HADDAM NECK PLANT REACTOR COOLANT SYSTEM COMPONENT SUPPORTS

M. S. THOMAS
L. A. HARDING

REVIEWED:

APPROVED: ______________________
S. A. Palm, Manager
Structural Engineering I

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SECTION 1
INTRODUCTION

As part of the Haddam Neck Plant systematic evaluation program, the primary piping/support system has undergone a static analysis for normal operating thermal, pressure, and deadweight loads along with a dynamic analysis for seismic loads. This report presents the structural evaluation of the Reactor Coolant System (RCS) component supports under all design loading conditions. Evaluation of the RCS piping system is presented in another report[1]. The applicable criteria and methods of analysis were submitted to the Nuclear Regulatory Commission via NRC Docket 50-213, da nuary 17, 1980[2].

This report presents results of the component support evaluation and the proposed modifications necessary to adequately qualify the supports for the evaluation conditions. Supports included are for the reactor vessel, steam generators, reactor coolant pumps, pressurizer and surge line.

---

(1) P. J. Kotwicki, "Structural Analysis of the Primary Reactor Coolant Loop System for the Haddam Neck Nuclear Power Station".

(2) Connecticut Yankee Atomic Power Company, NRC Docket 50-213, Haddam Neck Plant Systematic Evaluation Program Seismic Reevaluation, January 17, 1980.

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SECTION 2

EVALUATION LOADING CONDITIONS AND STRESS CRITERIA

2-1. LOADING CONDITIONS

The structural stress analyses done on the RCS supports consider the loads resulting from deadweight, thermal expansion and safe shutdown earthquake. Two loading combinations are examined: a static combination of deadweight and thermal expansion (normal condition) and a dynamic combination of deadweight, thermal expansion and SSE (faulted condition). The loads applied to the supports are obtained from an integrated model of the reactor coolant loop. This model consists of all the RCS components, main coolant piping, and stiffness values representing the component supports and piping restraints.

2-2. STRESS CRITERIA

The stress criteria used in evaluating the component supports are in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, 1977 Edition through Summer 1979 Addenda. Allowable stress limits are dependent on support type and loading condition. Linear type supports subject to normal operating conditions must meet working stress allowables per Appendix XVII of ASME III. These working stress limits are increased by the lesser of 1.2 F_y/F_t or 0.7 S_u/F_t for the faulted condition per Paragraph F-1370 of Appendix F.

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SECTION 3
ANALYTICAL CONSIDERATIONS AND METHODS

3-1. ANALYTICAL CONSIDERATIONS

The static and dynamic structural analyses assume linear elastic behavior for all component supports. The analysis of each support considers only the critical support components described in Section 3-2.

3-2. SUPPORT DESCRIPTION

Reactor Vessel

The reactor vessel is supported by four block assemblies per Stone and Webster drawings 10899-FV-4A, 4B, 4C and 4D. The blocks rest on a hollow, cylindrical shell surrounding the vessel. The shell is reinforced by 16 vertical, radial plates.

Steam Generator

Drawings 10899-FV-33A and 33B and 10899-FV-34A and 34B show the steam generator support to consist of a cylindrical skirt transferring load from the generator to the concrete floor. Four studs attach the generator feet to sliding blocks at the skirt's top. The top sliding blocks allow for radial thermal expansion of the generator while eight lower sliding blocks allow for expansion away from the reactor vessel. All blocks transfer load via steel balls to rigid blocks which are held down by bolts.

Reactor Coolant Pump

Three spring hangers support each reactor coolant pump. The spring supports provide a means for transferring load from their attachment points on the pump shell to the civil structural steel. Figures 3-1 and 3-2 show details of the two types of spring hangers and corresponding

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support mark numbers. Drawings 10899-FP-14A and 10899-FC-38L show the support attachment points to the pump shell and civil structural steel, respectively.

Pressurizer

Four upper guide supports, three rod hangers and an earthquake truss comprise the pressurizer support (Drawings Nos. 10899-FS-35A and 35B). The guide supports provide lateral support at the top of the pressurizer; the earthquake truss provides lateral restraint to the pressurizer base. Vertical restraint is supplied by the rod hangers.

Surge Line

Spring hangers RC-H-17 and RC-H-18 provide deadweight support for the surge line. Hanger details are shown in Figures 3-3 and 3-4.

3-3. METHOD OF ANALYSIS

The loads used to evaluate the RCS supports are obtained from the analysis of the reactor coolant loops as described in Section 2-1. Each support was analyzed for an algebraic combination of deadweight and normal operating thermal loads and a maximum combination of deadweight, normal operating thermal and SSE loads. These member loads were then used to find member stresses for each loading condition. Comparison with the appropriate ASME allowable led to a verification of the support's adequacy.

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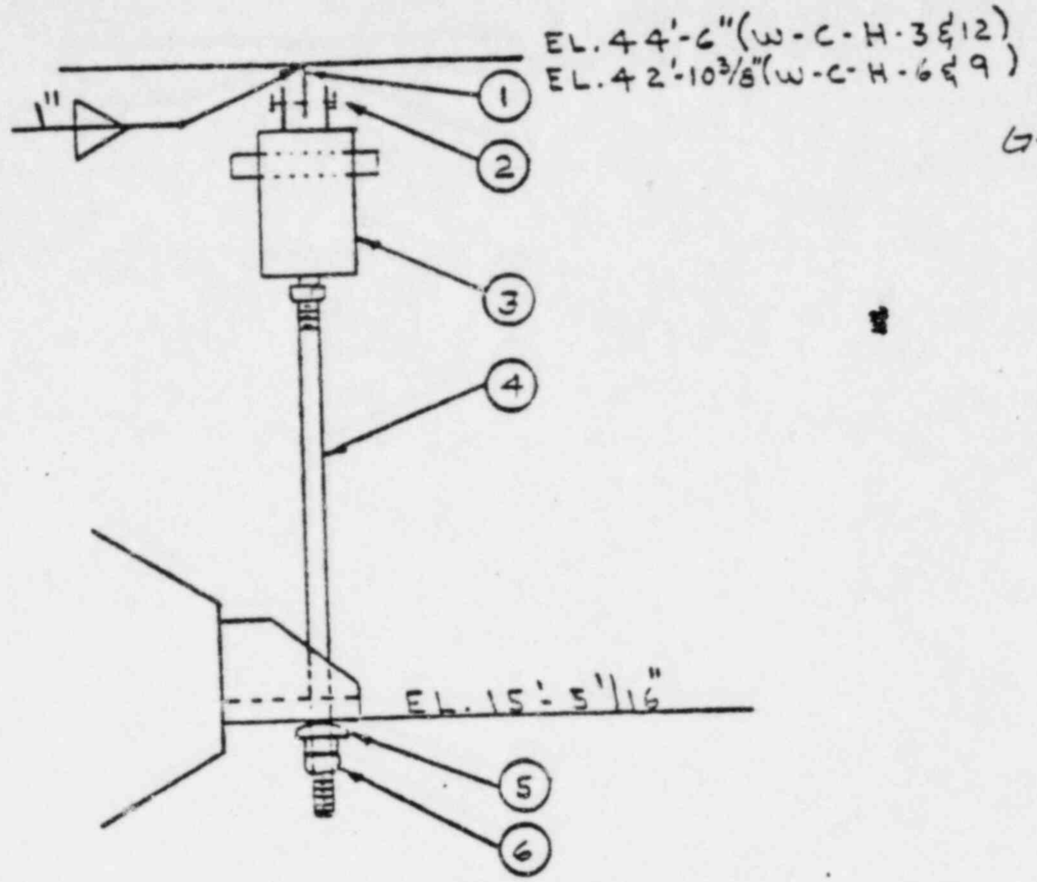

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MATERIALS AND OPERATIONS	ITEM NO.
HANGER ASSEMBLIES CONSISTING OF:	
Lugs/Sk. #104	1
Pins/Sk. #103	2
Size #824, Type "C", Pre-engineered Disc Spring Hanger w/Cold Set,	3
HL-70,500#, CL-78,800#, S-2"	
4" x 28'-1" Fig. 140 w/10 1/2" T.O.E. & 16" T.O.E.	4
Mark: (1) W-C-H-3	
(1) W-C-H-12	
4" x 26'-6" Fig. 140 w/10 1/2" T.O.E. & 16" T.O.E.	4
Mark: (1) W-C-H-6	
(1) W-C-H-9	
Rocker Washers/Sk.# 102	5
4" Hex Nuts	6
Mark: (1) W-C-H-3 (1) W-C-H-9	
(1) W-C-H-6 (1) W-C-H-12	

FIGURE 3-1: RC PUMP SUPPORTS

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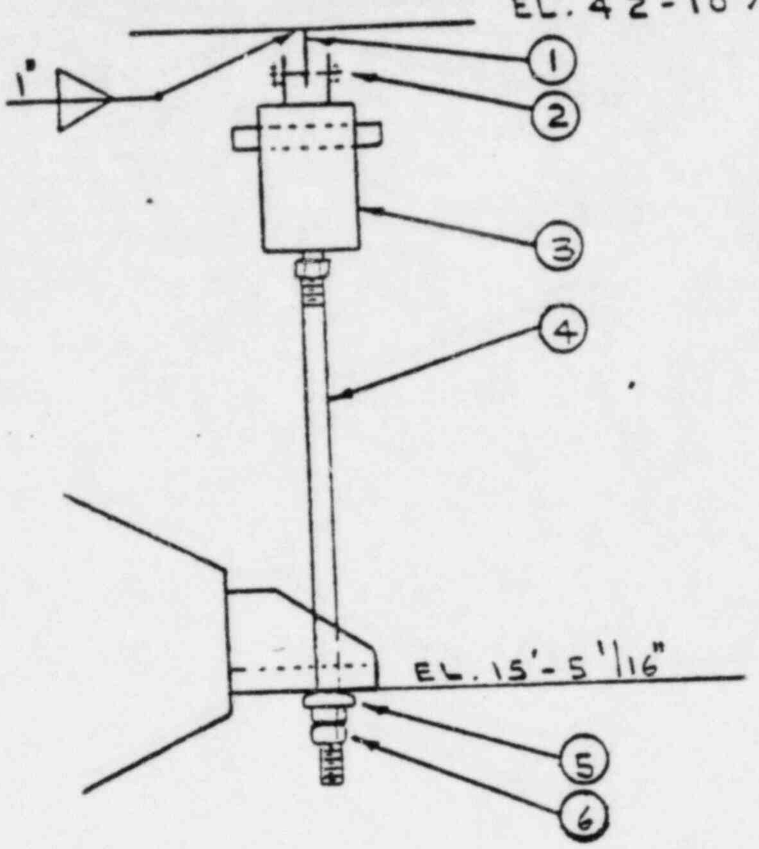


MATERIALS AND OPERATIONS	ITEM NO.
HANGER ASSEMBLIES CONSISTING OF:	
Lugs/Sk. #104	1
Pins/Sk. #103	2
Size #823, Type "C", Pre-engineered Disc Spring Hanger w/Cold Set, HL-53,115#, CL-59,265#, S-2" Mark: (1) W-C-H-1 (1) W-C-H-7 (1) W-C-H-4 (1) W-C-H-10	3
Size #823, Type "C" Pre-engineered Disc Spring Hanger w/Cold Set, HL-58,185#, CL-64,335#, S-2", Mark: (1) W-C-H-2 (1) W-C-H-8 (1) W-C-H-5 (1) W-C-H-11	3
3 3/4" x 26'-8" Fig. 140 w/10" T.O.E. & 16" T.O.E.	4
Rocker Washers/Sk. #102	5
3 3/4" Hex Nuts	6
MARK: (1) W-C-H-1 (1) W-C-H-7	
(1) W-C-H-2 (1) W-C-H-8	
(1) W-C-H-4 (1) W-C-H-10	
(1) W-C-H-5 (1) W-C-H-11	

FIGURE 3-2: RC PUMP SUPPORTS

PRELIMINARY

1. Figure 66 Welded Beam Attachment
2. Figure 82 Type B Size #12 Spring Can
3. Figure 278 Rod
   1" x 2'-11-1/2" LG.
   W/12" THD.
4. Figure 295 10" Pipe Clamp
5. 1" Hex Nut

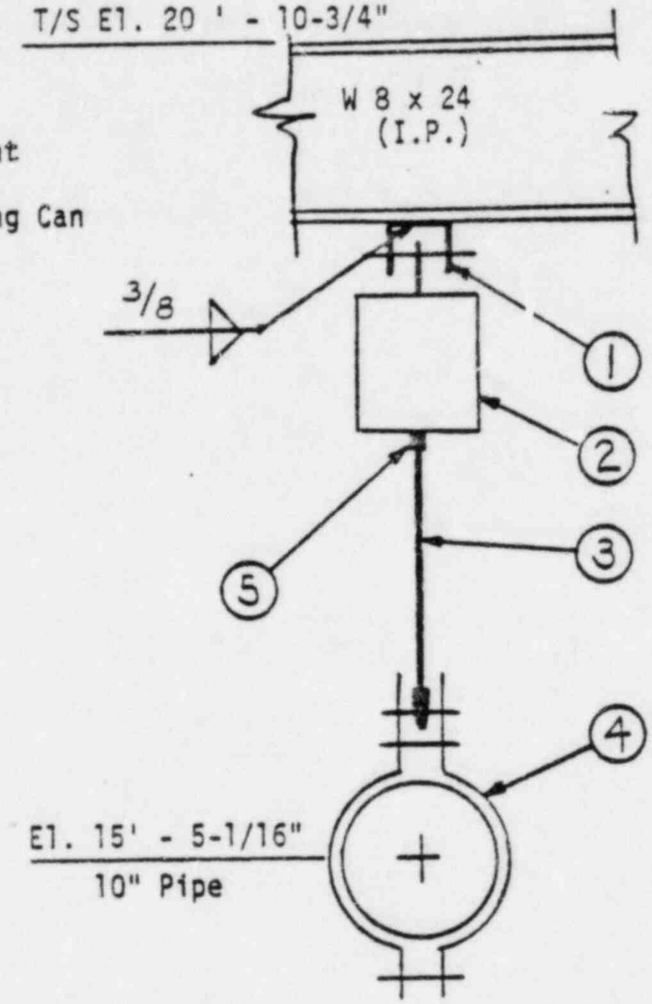
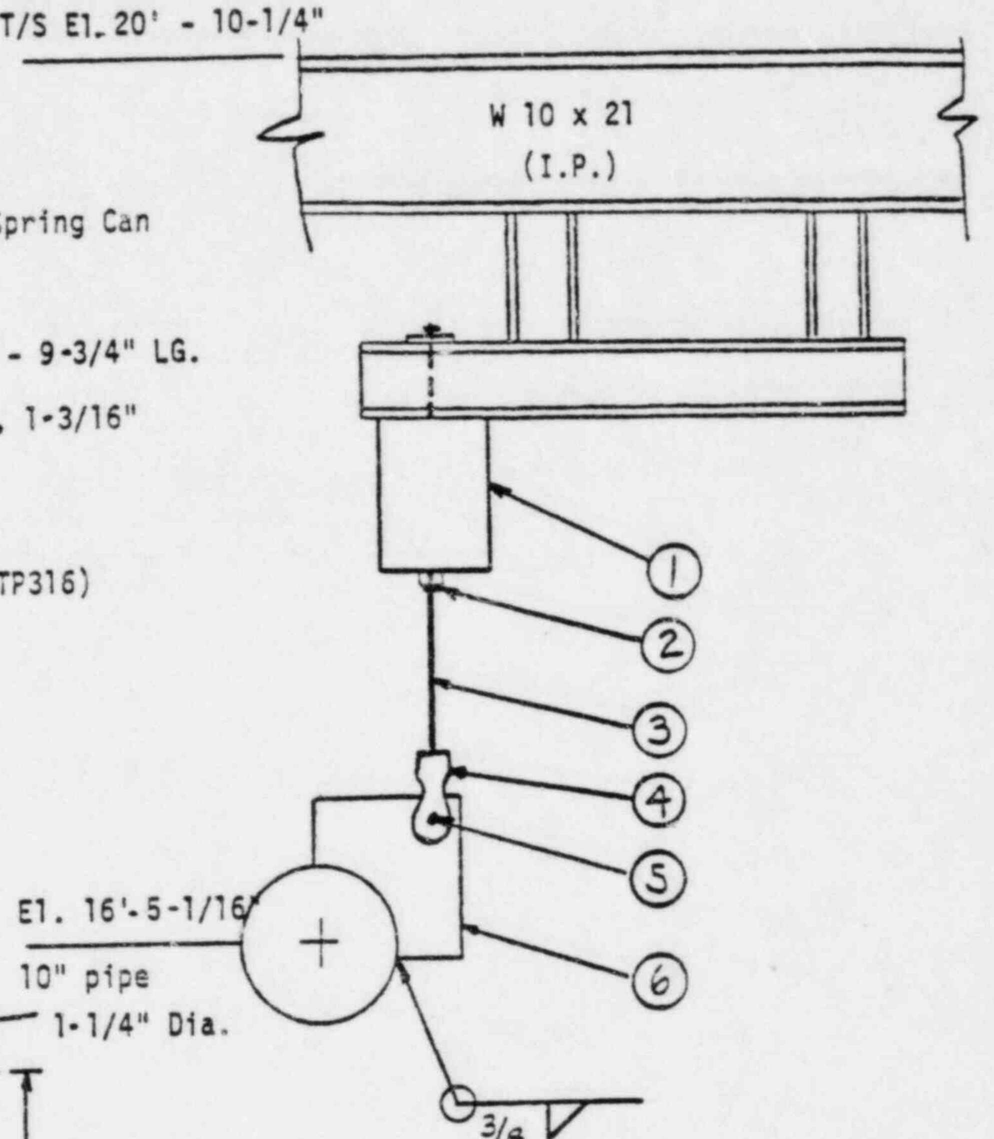


FIGURE 3-3: SURGE LINE SUPPORT RC-H-17

PRELIMINARY

1. Figure 82 Type A Size #14 Spring Can
2. 1-1/8" Hex nuts
3. Figure 140 Rod, 1-1/8" x 0' - 9-3/4" LG.
4. Figure 299 #3 w/1-1/8" Tap, 1-3/16" Pinhole, 1 1/8" Grip
5. HS 68, 1-1/8"
6. Stainless Steel Lug (A312 TP316)



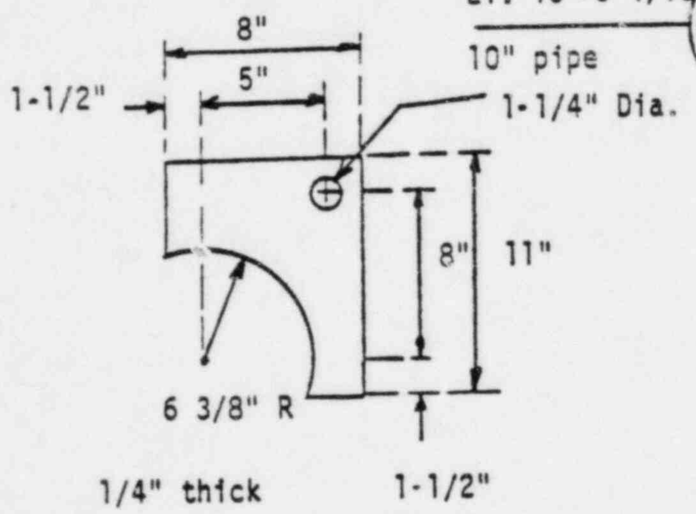


FIGURE 3-4: SURGE LINE SUPPORT RC-H-18

PRELIMINARY

SECTION 4
STRESS EVALUATION RESULTS

The results of the stress evaluation of the RCS supports are summarized in this section. Tables 4-1 through 4-5 present the maximum stresses, allowables, and safety factors for all critical support components.

4-1. REACTOR VESSEL

Linear stress evaluation results are summarized in Table 4-1. All members of the vessel support have been found to be adequate for both normal and faulted conditions.

4-2. STEAM GENERATOR

Most critical among the steam generator support system components are the four upper 3" studs connecting the generator feet to the skirt support. These studs fail to meet a combined tensile and shear stress interaction ratio. All other members of the steam generator support system have been found to be adequate for both normal and faulted conditions. (See Table 4-2.) A possible fix is shown in Section 5-1.

4-3. REACTOR COOLANT PUMP

Analysis of the Reactor Coolant Pump supports shows four of the 12 spring hangers to bottom out due to seismic movement. The analysis uses the cold readings of June 17, 1975 as a basis for analysis and showed that four #824 spring hangers, one on each pump, bottom out. By using a new stiffness value for a "bottomed-out" spring can in the RCS piping analysis, new loads were obtained for the spring after it reaches its lower limit. Evaluation of the spring cans for these loads found acceptable all components of the hanger assembly except the upper attachment lugs and welds to the supporting steel (see Table 4-3). Proposed modifications are shown in Section 5-2.

PRELIMINARY

Preliminary evaluation of the spring hanger support steel members for the "bottomed-out" loads shows the failure of two members: the W18x60 of support RC-H-6 and the W18x60 of RC-H-9. Exact details of any proposed modifications cannot be supplied because of lack of information regarding the member end connection details. An initial proposal calls for the addition of cover plates welded to the existing flanges.

Appendix B gives the reaction loads at the pump spring hanger supporting steel member ends.

4-4. PRESSURIZER

Evaluation of the pressurizer support system shows the guide supports and rod hangers capable of withstanding a faulted condition without any modification. (See Table 4-4.) Members 3, 4, 5 and 6 of the computer plot model of the earthquake truss shown in Figure 4-1 fail to meet compressive stress allowables for the faulted condition. All members of the truss fail the combined stress interaction equation. Proposed modifications to all truss members are shown in Section 5-3.

4-5. SURGE LINE

Spring hanger RC-H-17 will bottom out due to seismic movements. All remaining components of both spring hangers were found to be adequate for the faulted condition (Table 4-5). See proposed modifications in Section 5-4.

The supporting steel for both spring hangers was not evaluated due to lack of information regarding the members and their end conditions.

PRELIMINARY

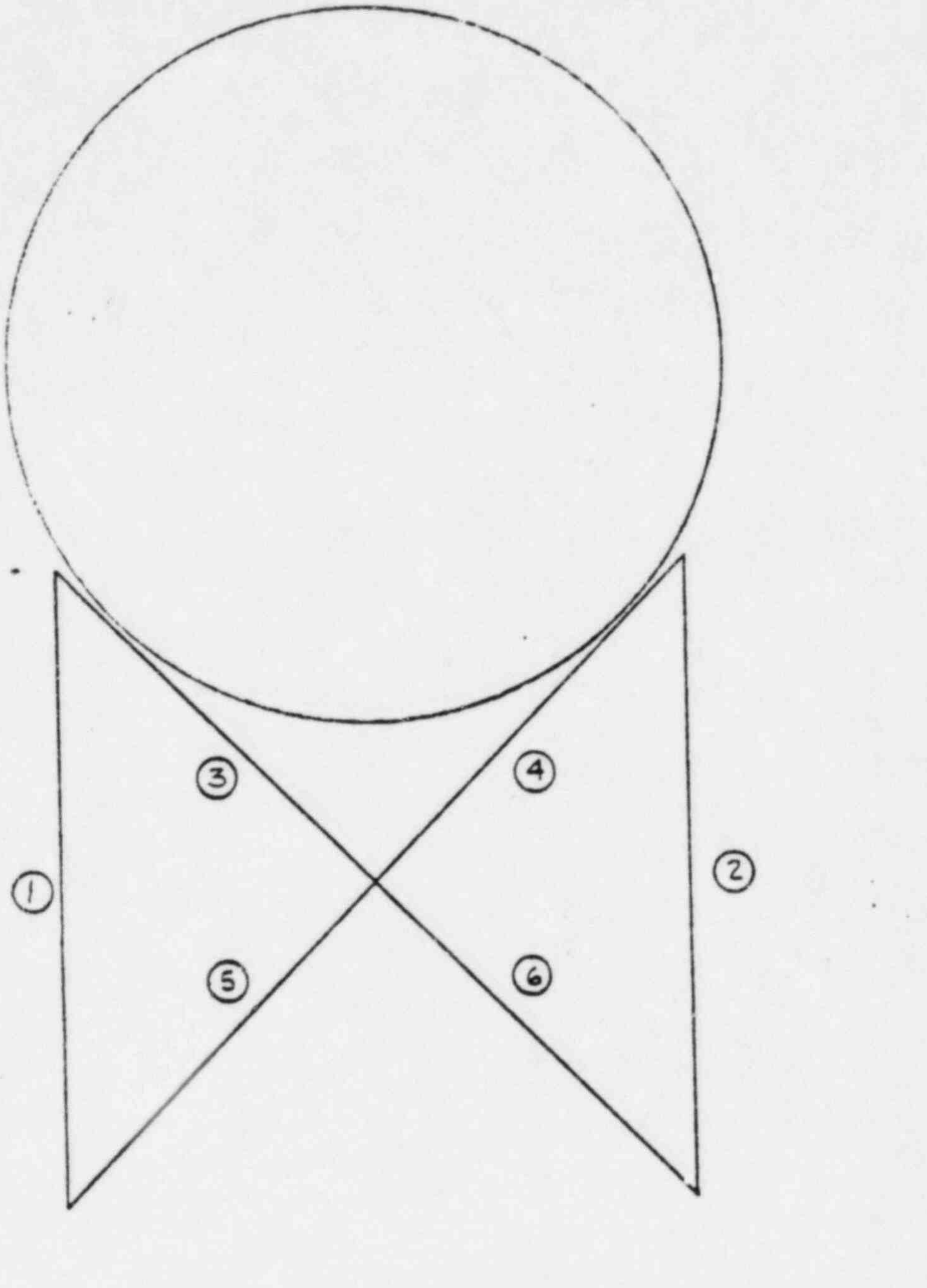


FIGURE 4.1: PRESSURIZER EARTHQUAKE TRUSS MODEL

PRELIMINARY

TABLE 4-1

MAXIMUM REACTOR VESSEL SUPPORT STRESSES (KSI)

RVS	Stress	Normal Condition			Faulted Condition		
Component	Type	Stress	Allowable	S.F.	Stress	Allowable	S.F.
Support Blocks	Shear				9.8	40	4.1
	Bearing				6.4	119	18
Shield Tank	Bending				5.1	44	8.6
	Shear				3.1	40	13

PRELIMINARY

TABLE 4-2
MAXIMUM STEAM GENERATOR SUPPORT STRESSES (KSI)

S.G.S. Component	Stress Type	Normal Condition Stress	Allowable	S.F.	Faulted Condition Stress	Allowable	S.F.
3" Stud	Tensile	7.5	102	14	139	143	1.0*
	Shear	5.2	42	8.1	43	59	1.4*
Sliding Block	Bending	2.8	73	26	13	115	8.8
	Shear	3.1	49	16	14	77	5.5
2 1/2"dia. Steel Ball	Capacity	39K	405K	10	190K	405K	2.1
Support Block	Bending	2.1	73	35	33	115	3.5
	Shear	0.8	49	61	13	77	5.9
Support Block Hold Down Bolt	Tensile	6.7	96	14	92	134	1.5
Shear Key	Shear	4.8	32	6.7	24	49	2.0
Support Stop Block	Bending				20	35	1.8
	Shear				5.8	29	3.3

* Stud fails to meet combined tensile/shear stress interaction ratio (1.47 > 1.0)

PRELIMINARY

TABLE 4-3

MAXIMUM REACTOR COOLANT PUMP SUPPORT LOADS (KIPS)

RCP Support Component		Normal Condition Load	Allowable	S.F.	Faulted Condition Load	Allowable	S.F.
#823 Spring Can		58	74	1.3	68	74	1.1
#824 Spring Can		88	97	1.1	225*	236**	1.1
#104 Lug	4" Hole	58	92	1.6	68	138	2.0
	4 1/4" Hole	88	101	1.1	225*	151	0.7***
#103 Pin	3 3/4" Dia.	58	82	1.4	68	150	2.2
	4" Dia.	88	90	1.0	225*	165	0.7***
Rod	3 3/4" Dia.	58	83	1.4	68	156	2.3
	4" Dia.	88	94	1.1	225*	464	2.1
Hex Nut	3 3/4" Hole	58	83	1.4	68	156	2.3
	4" Hole	88	94	1.1	225*	464	2.1
#102 Washer	3 3/4" Hole	58	83	1.4	68	156	2.3
Washers	4" Hole	88	94	1.1	225*	464	2.1

* Loads based on the stiffness of a "bottomed-out" spring can.

** Capacity of a "bottomed-out" spring can.

*** See proposed modifications in Section 5-1.

PRELIMINARY

TABLE 4-4

MAXIMUM PRESSURIZER SUPPORT STRESSES (KSI)

Pzr. Support Component	Stress Type	Normal Condition Stress	Allowable	S.F.	Faulted Condition Stress	Allowable	S.F.
Backing Ring	Tensile	9.2	20	2.2	14	39	2.8
900 lb Weld Neck Flange	Tensile	9.2	20	2.2	14	39	2.8
Flange Bolts & Nuts	Tensile	8.4	62	7.4	13	87	6.7
Retainer	Tensile	9.2	20	2.2	14	39	2.8
3 1/8" Rod & Nuts	Tensile	18	22	1.2	28	42	1.5
Support Ring Attachment	Bending	8.6	20	2.3	13	39	3.0
	Shear	4.9	13	2.7	7.5	26	3.5
Guide Support	Bending	0.09	18.06	201	1.38	21.67	15.7
	Shear	0.05	12.04	241	0.83	24.08	29.0
Guide Support Anchor Bolt	Tensile	0.41	12.5	30.5	6.31	17.5	2.8
	Shear	0.15	8.2	54.7	2.40	11.5	4.8
Earthquake Truss Member #3	Compres.	2.9	11	3.8	30	21	0.7*
	Bending	3.0	20	6.7	6.0	39	6.5
Truss Anchor Plate Screws	Tensile	7.5	62.5	8.3	74	88	1.2

* See proposed modifications in Section 5-2.

PRELIMINARY

TABLE 4-5

MAXIMUM SURGE LINE SUPPORTS LOADS (KIPS)

S.L. Support	Normal Condition			Faulted Condition		
Component	Load	Allowable	S.F.	Load	Allowable	S.F.
Size #12B Spring Can	2.6	2.9	1.1	3.6	2.9	0.8*
Fig.#66: 1" Beam Attachment	2.6	5.0	1.9	3.6	9.3	2.6
Fig.#278: 1" Rod	2.6	5.0	1.9	3.6	9.3	2.6
Fig.#295: 10" Clamp	2.6	3.2	1.2	3.6	6.1	1.7
1" Hex Nut	2.6	5.0	1.9	3.6	9.3	2.6
Size #14A Spring Can	4.1	5.2	1.3	5.3	5.2	1.0
Fig.#140: 1 1/8" Rod	4.1	6.2	1.5	5.3	12	2.2
Fig.#299: Clevis	4.1	6.2	1.5	5.3	12	2.2
H 68: 1 1/8" Pin	4.1	6.2	1.5	5.3	12	2.2
Stainless Steel Lug	2.0	2.7	1.4	2.7	4.1	1.5
1 1/8" Hex Nuts	4.1	6.2	1.5	5.3	12	2.2

* Spring can bottoms out. See proposed modification in Section 5-3.

PRELIMINARY

SECTION 5
PROPOSED MODIFICATIONS

5-1. STEAM GENERATOR SUPPORT

A possible fix for the combined overstress problem in the 3" stud calls for replacing the stud with a 3 1/2" diameter stud. The interaction ratio will then be reduced from 1.47 to 0.78.

5-2. REACTOR COOLANT PUMP SPRING HANGERS

Using the cold spring hanger readings of June 17, 1975, the four #824 spring hangers were found to bottom out during a faulted condition. The capacity of a bottomed-out spring hanger was found to be governed by the upper clevises on the hanger tube and the single clevis welded to the structural framing. These clevises can be adequately upgraded by welding a 3/4-inch A-572 GR-50 stiffener clevis to each upper lug of the hanger tube and, also by welding two additional 3/4-inch A-572 GR-50 clevises to the pump structural beam (Figure 5-1). The clevis pin will need to be replaced with a longer pin to accommodate these modifications. The spring hanger support steel must be stiffened to take the modified spring hanger loads.

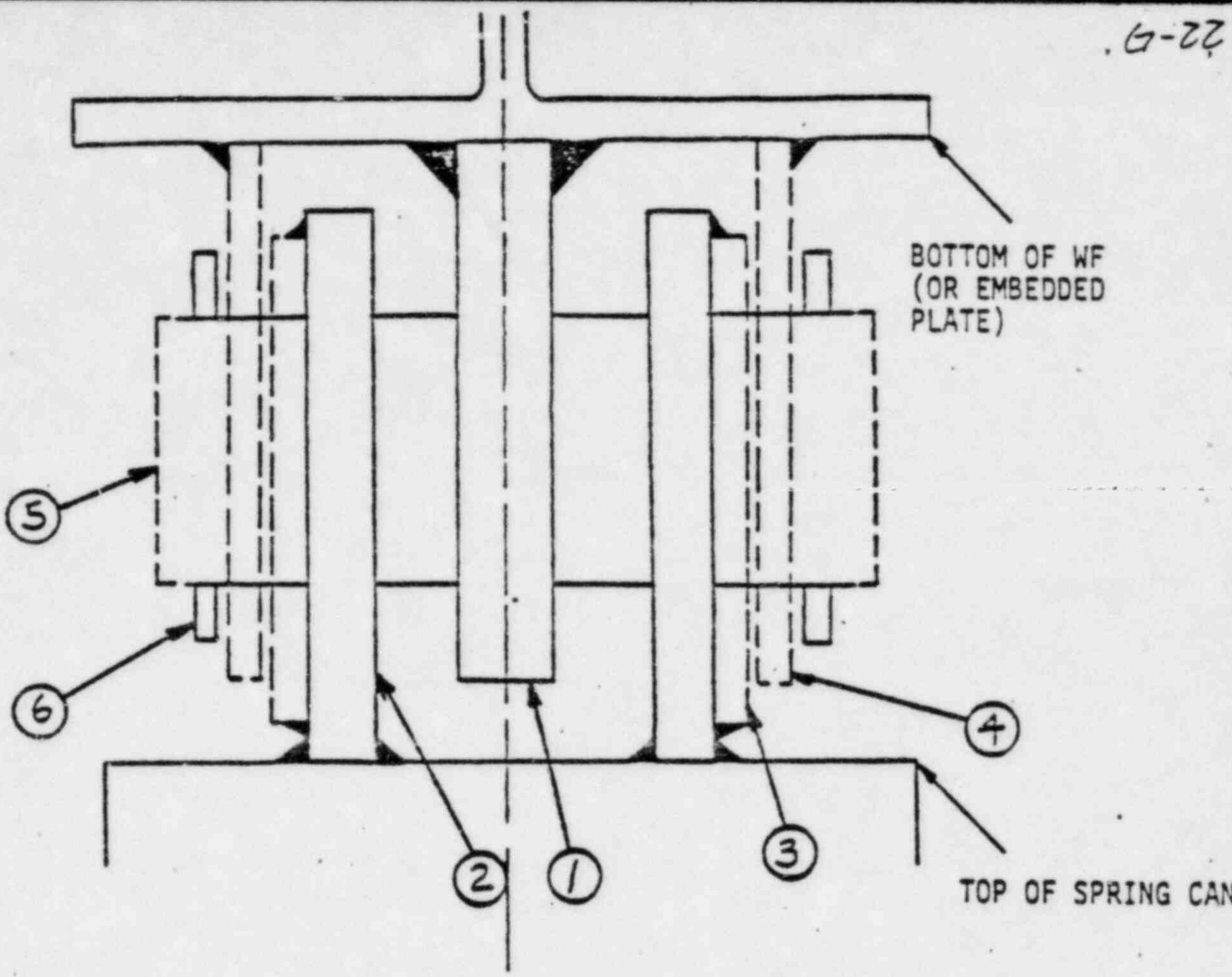
5-3. PRESSURIZER EARTHQUAKE TRUSS

By modifying the entire earthquake truss to the specifications shown in Figure 5-2 those members found to be initially overstressed will now be within allowable stress limits.

5-4. SURGE LINE SUPPORTS

Spring hanger No. RC-H-17 was found to bottom out due to seismic movements. Replacing this spring support with a vertical, rigid support will limit seismic movements, reduce critical stresses at the pressurizer nozzle, and reduce the loads on support No. RC-H-18 to an acceptable level.

PRELIMINARY

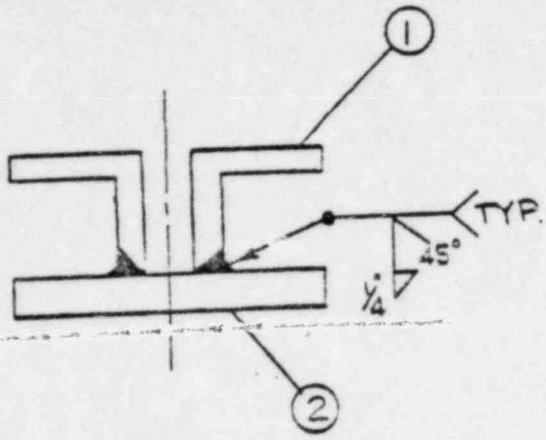


1. Lug #104 (Existing).
2. Top Lugs of #824 Spring Can (Existing).
3. Stiffener Lugs (A-572 G50).
4. Stiffener Lugs (A-572 G50).
5. Pin #103 (C_{min} = 11").
6. Figure 146 Rod, 1/2" x 0'-6" LG, W/Hex Nuts (Existing).

NOTES: A. Modifications are shown as dashed lines.
B. Item 5 has been lengthen to accommodate modifications.
C. Modifications can be performed without removing hanger.

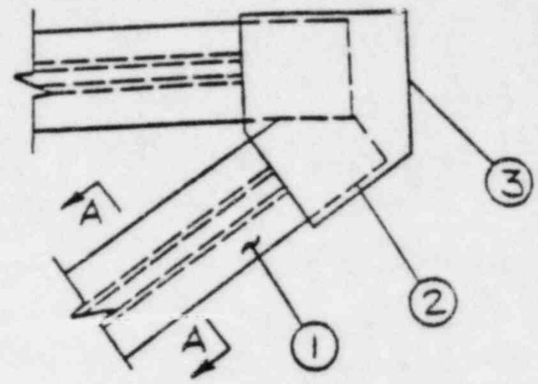
FIGURE 5-1: REACTOR COOLANT SPRING HANGER MODIFICATION

PRELIMINARY



SECTION A-A

(MODIFIED COMPOSITE MEMBER)



PLAN VIEW

(END CONNECTION)

1. L 3" x 3" x 5/16" (A-36), BB= 1/2" (Existing).
2. PL 1/2" x 6½" (A-36).
3. 3" Plate (AISI C-1020) (Existing).

NOTES: A. Approximately 35' of item 2 is required to completely build-up the truss.

PRELIMINARY

FIGURE 5-2: PRESSURIZER EARTHQUAKE TRUSS MODIFICATION

SECTION 6
CONCLUSIONS

The structural evaluation of the Haddam Neck Plant RCS component supports was performed to demonstrate the structural adequacy of the supports under specified loading conditions. By incorporating the proposed modifications the component supports will be structurally qualified for all loading conditions.

PRELIMINARY

APPENDIX A

STONE AND WEBSTER COMPONENT SUPPORT DRAWINGS

PRELIMINARY

Drawing	Revision	Title
10899-FV-4A	5	Reactor Neutron Shield and Support, Sh. 1
10899-FV-4B	2	Reactor Neutron Shield and Support, Sh. 2
10899-FV-4C	2	Reactor Neutron Shield and Support, Sh. 3
10899-FV-4D	4	Reactor Neutron Shield and Support, Sh. 4
10899-FV-33A	3	Steam Generator Supports, Sh. 1
10899-FV-33B	4	Steam Generator Supports, Sh. 2
10899-FV-34A	5	Steam Generator Support Tie Rod, Sh. 1
10899-FV-34B	5	Steam Generator Support Tie Rod, Sh. 2
10899-FV-35A	3	Pressurizer Support Reactor Containment
10899-FS-35B	1	Pressurizer Earthquake Guide Details
10899-FP-14A	9	Reactor Coolant Piping, Sh. 1
10899-FC-38L	6	Interior Concrete Details, Sh. 7, Reactor Containment

PRELIMINARY

APPENDIX B

RESULTANT SUPPORT LOADS TRANSMITTED TO CONTAINMENT CONCRETE, STEEL AND EMBEDMENTS

PRELIMINARY

G-28.

STRUCTURAL ANALYSIS OF THE PRIMARY REACTOR COOLANT LOOP

SYSTEM FOR THE HADDAM NECK NUCLEAR POWER STATION

MAY 1982

PRELIMINARY

A. SCOPE

The purpose of this document is to present the analytical methods and stress criteria which were used for the Connecticut Yankee primary coolant loop system seismic qualification program. The program included static analysis of the primary piping/support system for normal operating thermal. pressure. and deadweight loads along with dynamic system analysis for seismic loads. This report also contains the stress criteria and the stress evaluation for the loop piping.

B. BACKGROUND

PRELIMINARY

In the years since the Connecticut Yankee generating station was designed. seismic analysis methods have become more rigorous and the asme Boiler and Pressure Vessel Code. Section III. Nuclear Power Plant Components, has een published reflecting changes in analysis. design, and quality control techniques. The purpose of this report is to establish requirements for performing the up-grading seismic analyses of the primary coolant loop system with current technology, and to show that the Connecticut Yankee station meets these requirements.

The original design criteria used for analysis of this plant's primary piping system is the ASA B31.1 Code for Pressure Piping. The reactor pressure vessel. steam generator. and reactor coolant pump were designed and analyzed to the rules of the ASME Code Section VIII.

For the purpose of this document. the reactor coolant loop piping shall be considered to consist of the hot legs. cold legs. cross-over legs, and pressurizer surge line. The primary equipment in the system consists of control rod drive mechanisms. reactor vessel internals. reactor pressure vessel. steam generator. reactor coolant pump. and pressurizer. Loads are generated on the supports for the reactor pressure vessel. the steam generator. reactor coolant pump. and pressurizer.

C. LOADING CONDITIONS

The reactor coolant loop piping, supports, and components will be analyzed for the following loading conditions:

1. Normal condition operating pressure, deadweight, and temperature.

2. SSE Condition Seismic- Safe Shutdown Earthquake (SSE) combined with operating pressure and deadweight.

D. STRESS CRITERIA

PRELIMINARY

1. Piping

The piping analysis performed for the Connecticut Yankee evaluation was based on the rules of the ANSI B31.1 -1973 Code, the Summer 1973 Addenda.

The loading combinations and associated stress limits used for the piping systems which are part of the seismic qualification program are given in Table 1. The stress limits used for the SSE condition correspond to faulted condition allowables. The piping stresses were calculated using formulas given in ANSI B31.1 - 1973, 1973 Summer Addenda.

The loads that the primary coolant loop piping transmits to the pressurizer, steam generator, reactor coolant pump, and reactor pressure vessel nozzles and supports are transmitted to those performing those analyses. Separate reports cover the primary equipment supports and the primary component itself.

Since the loop isolation valves are much thicker and stronger than the attached piping, and since valves of this design have no history of gross failure of their pressure boundaries (as long as the stresses of the piping attached to the valve remain within the limits defined in this document) the valve integrity is assured.

2. Supports

The support criteria and evaluation are covered in a separate report. The loads that the primary coolant loop piping transmits to the supports are generated for use by the support analysts.

E. ANALYSIS PROCEDURES

PRELIMINARY

1. General Procedures

The reactor coolant loop piping/support system was evaluated with three-dimensional static or dynamic models, depending on the load requirements. Static analysis of the piping systems was performed using displacement techniques with lumped parameters and stiffness matrix representations of supports. It was assumed that all components and piping behaved in a linear elastic manner. The method used for dynamic analysis was the response spectrum technique.

The primary equipment evaluated as part of this program had dynamic analyses performed in accordance with the procedures outlined in the respective equipment reports. In addition to the detailed models that were developed for the evaluations of the individual components, reduced models were used in the reactor coolant loop system analysis.

Analytical representations of the primary equipment supports were produced for inclusion in the reactor coolant loop system model. The loads generated by the reactor coolant loop system model were used to qualify the component supports.

2. Response Spectrum Analysis Procedures

A response spectrum seismic analysis was performed using a three-dimensional linear dynamic analytical model of the primary coolant loop system. The model includes analytical representations of the components, component supports, and associated piping. The boundaries of the model are defined as the component support to containment concrete interface.

The analysis was performed assuming that the seismic event is initiated with the plant at normal full power condition. The damping values used were for four percent (4%) of critical for the SSE condition. Since the components are supported at different floor evaluations within the containment building, the response spectrum in each direction shall be an envelope of the applicable floor spectra. The spectra used in the analysis are presented in Figures 1, 2, and 3.

The analysis was performed with a simultaneous input of the two horizontal components and one vertical component of the earthquake. The modal response for each item of interest (e.g., force, displacement, stress) was obtained by the square root of the sum of the squares method.

$$R_T = \left[\sum_{i=1}^{3} R_i^2 \right]^{1/2}$$

where: $R_i \quad \left[\sum_{j=i}^{N} R_{ij}^2 \right]^{1/2}$

where: R_T = total combined response at a point

R_i = value of combined response of direction i

R_{ij} = absolute value of response for direction i, mode j

N = total number of modes considered

For systems having modes with closely spaced frequencies, the above method was modified to include the possible effect of these modes. The groups of closely spaced modes were chosen such that the difference between the frequencies of the first mode and the last mode in the group did not exceed ten percent (10%) of the lower frequency. Combined total response for systems which have such closely spaced modal frequencies was obtained in accordance with Regulatory Guide 1.92. Frequency groups are formed starting from the lowest frequency and working toward successively higher frequencies. No frequency was included in more than one group. The resultant unidirectional response for systems having such closely spaced modal frequencies was obtained by the square root of the sum of: (a)the sum of the squares of all modes, and (b) the product of the responses of the modes in various groups of closely spaced modes and associated coupling factors, ϵ . The mathematical expression for this method (with "R" as the item of interest) is:

$$R_i^2 = \sum_{j=1} R_{ij}^2 + 2 \sum_{j=1}^{S} \sum_{K=M_j}^{N_j-1} \sum_{\ell=K+1}^{N_j} R_{iK} R_{i\ell} \epsilon_{K\ell}, \text{ for: } \ell \neq K$$

where: S = number of groups of closely spaced modes

M_j = lowest modal number associated with group j of closely spaced modes

N_j = highest modal number associated with group j of closely spaced modes

$\varepsilon_{K\ell}$ = coupling factor with

$$\varepsilon_{K\ell} = [1 + (\frac{\omega^1_K - \omega^1_\ell}{(\beta^1_K\omega_K + \beta^1_\ell\omega_\ell)})^2]^{-1}$$

and:

$$\omega^1_K = \omega_K [1 - (\beta^1_K)^2]^{1/2}$$

$$\beta^1_K = \beta_K + \frac{2}{(\omega_K t_d)}$$

PRELIMINARY

ω_K = frequency of closely spaced mode K (rad/sec)

β_K = fraction of critical damping in closely spaced mode K

t_d = duration of the earthquake (seconds)

The analyses performed for piping and supports did not include stresses resulting from SSE induced differential motion. These stresses are secondary in nature, based on ASME Code rules for piping (NB-3653, NB-3656, F-1360) and component supports (NF-3231). The SSE being a very low probability single occurrence event, is treated as a faulted condition.

The analysis of the components subjected to seismic loading involved several steps that are similar to those outlined above for the system analysis. A three-dimensional linear elastic analytical representation of the component was developed. The component supports and attached primary coolant loop piping was represented by stiffness matrices. The analysis was performed with the simultaneous input of three response spectrum, two horizontal and one vertical. A damping value of four percent (4%) for SSE was used. The modal combination techniques outlined for the system analysis was also used for the component analysis. The component analyses are contained in separate reports.

When performing a response spectrum analysis, the assumption of a linear system is required. If the system contains nonlinearities, as in the primary equipment supports, different cases may be run in an attempt to bound the results. In the case of the CYW four-loop seismic model, the nonlinearity in the steam generator lower supports is the most evident. Additional nonlinearities also exist in the RCP vertical support and the RPV vertical support. The steam generator support system allows free thermal growth parallel to the hot leg during heatup to normal operating conditions. When the system is at normal operating, further movement of the steam generator away from the reactor vessel is restrained by the support system. For movement of the steam generator toward the reactor vessel, the support system offers the same restraint as during heatup. There are, therefore, two seismic support cases possible for the steam generator. One case for movement away from the reactor vessel and one case for movement toward the reactor vessel.

In the case of the reactor vessel support, a possible nonlinearity exists, if the vessel rocks sufficiently to lift vertically off one quadrant of the support. The vessel supports are preloaded with deadweight and thermal loadings that would have to be overcome before liftoff could occur. A check was performed to see if the seismiic loads exceed the preload on the support. The check indicated no liftoff so the vessel loads were accurate as analyzed.

The reactor coolant pump is supported by three spring hangers. If the deflection of the pump is large enough during a seismic event the springs will bottom out. When a spring can bottoms out, the stiffness of the supporting element increases to the value of the rod attached to the spring can. The increased stiffness will yield a higher support load than the spring can stiffness yields. The new load is calculated using an energy balance on the strain energy of the two systems.

Two loop support configurations were analyzed to bound the actual loop support configuration which varied during the seismic event. The two configurations chosen are shown in Figure 4. The loop response is such that these two cases represent the only unique combinations of supports for a seismic input.

PRELIMINARY

F. MODELING TECHNIQUES

PRELIMINARY

For piping system components, and component supports are represented by an ordered set of data which numerically describes the physical system.

The spatial geometric description of the model is based upon the as-built isometric piping drawings and equipment drawings Node point coordinates and incremental lengths of the members are determined from these drawings. Node point coordinates are input on network cards. Incremental member lengths are input on element cards. The geometrical properties along with the modulus of elasticity, E, the coefficient of thermal expansion, α, the average temperature changes from the ambient temperature, ΔT, and the weight per unit length, ω, are specified for each element. The supports are represented by stiffness matrices which define restraint characteristics of the supports.

A network model is made up of a number of sections, each having an overall transfer relationship formed from its group of elements. The linear elastic properties of the section are used to define the characteristic stiffness m[illegible] for the section. Using the transfer relationship f[illegible] section, the loads requiried to suppress all deflection at the ends of the section arising from the thermal and boundary forces for the section are obtained. These loads are incorporated into the overall load vector.

After all the sections have been defined in this manner, the overall stiffness matrix (K) and associated load vector to suppress the deflection of all the network points is determined. The flexibility matrix is multiplied by the negative of the load vector to determine the network point deflections due to the thermal and boundary force effects. Using the general transfer relationship, the deflections and internal forces are then determined at all node points in the system. The support loads (F) are also computed by multiplying the stiffness matrix (K) by the displacement vector (δ) at the support point.

The models used in the static analyses are modified for use in the dynamic analyses by including the mass characteristics of the piping and equipment.

The lumping of the distributed mass of the piping system is accomplished by locating the total mass at points in the system which approximately represent the response of the distributed system. Effects of the equipment motion are obtained by modelling the mass and the stiffness characteristics of the equipment in the overall system model when required. The supports are again represented by stiffness matrices in the system model for the dynamic analysis.

From the mathematical description of the system, the overall stiffness matrix (K) is developed from the individual element stiffness matrices using the transfer matrix (K) associated with mass degrees-of-freedom only. From the mass matrix and the reduced stiffness matrix, the natural frequencies and the normal modes are determined.

The effect of eccentric masses, such as valves and extended structures, are considered in the seismic piping analyses. These eccentric masses are modeled in the system analysis, and the torsional effects caused by them are evaluated and included in the total system response. The total response must meet the limits of the criteria given in Section D.

The lumped mass model of a typical loop of the 4-loop coupled model is shown in Figure 5. The total model was assembled from variations of this model. The coupled model was used to come up with the loads and stresses in the system.

PRELIMINARY

G. RESULTS

The results of the stress evaluation of the reactor coolant loop piping are summarized in Table 2. The method used to combine the loads to evaluate the adequacy of the piping system is given in Table 1. These results indicate that for the loading conditions considered. the piping is acceptable.

The results of the stress evaluation for the pressurizer surge line piping are also summarized in Table 2. The results indicate that the piping is acceptable for the cases considered.

PRELIMINARY

TABLE 1

LOADING COMBINATIONS AND STRESS LIMITS FOR PIPING

LOADING COMBINATIONS	STRESS LIMITS
1. Normal:	
Design Pressure + Deadweight	<Sh
2. SSE:	
Operating Pressure + Deadweight + Maximum Potential Earthquake Loads (SSE)	<1.8 Sh

where: Sh = allowable stress from USAS B31.1 Code for Pressure Piping.

PRELIMINARY

TABLE 2

PRIMARY PIPING STRESS ANALYSIS SUMMARY

EVALUATION	HOT LEG	X-OVER LEG	COLD LEG	ALLOWABLE
P + DW	6700.	7600.	7700.	16600.
P + DW + SSE	9900.	13000.	16300.	29880.

PRESSURIZER SURGE LINE PIPE STRESS ANALYSIS SUMMARY

		ALLOWABLE
P + DW	8700.	16600.
P + DW + SSE	16900.	29880.

PRELIMINARY

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CONN. YANKEE CONTAINMENT INTERNALS
FREE FIELD SPECTRA (VERT)

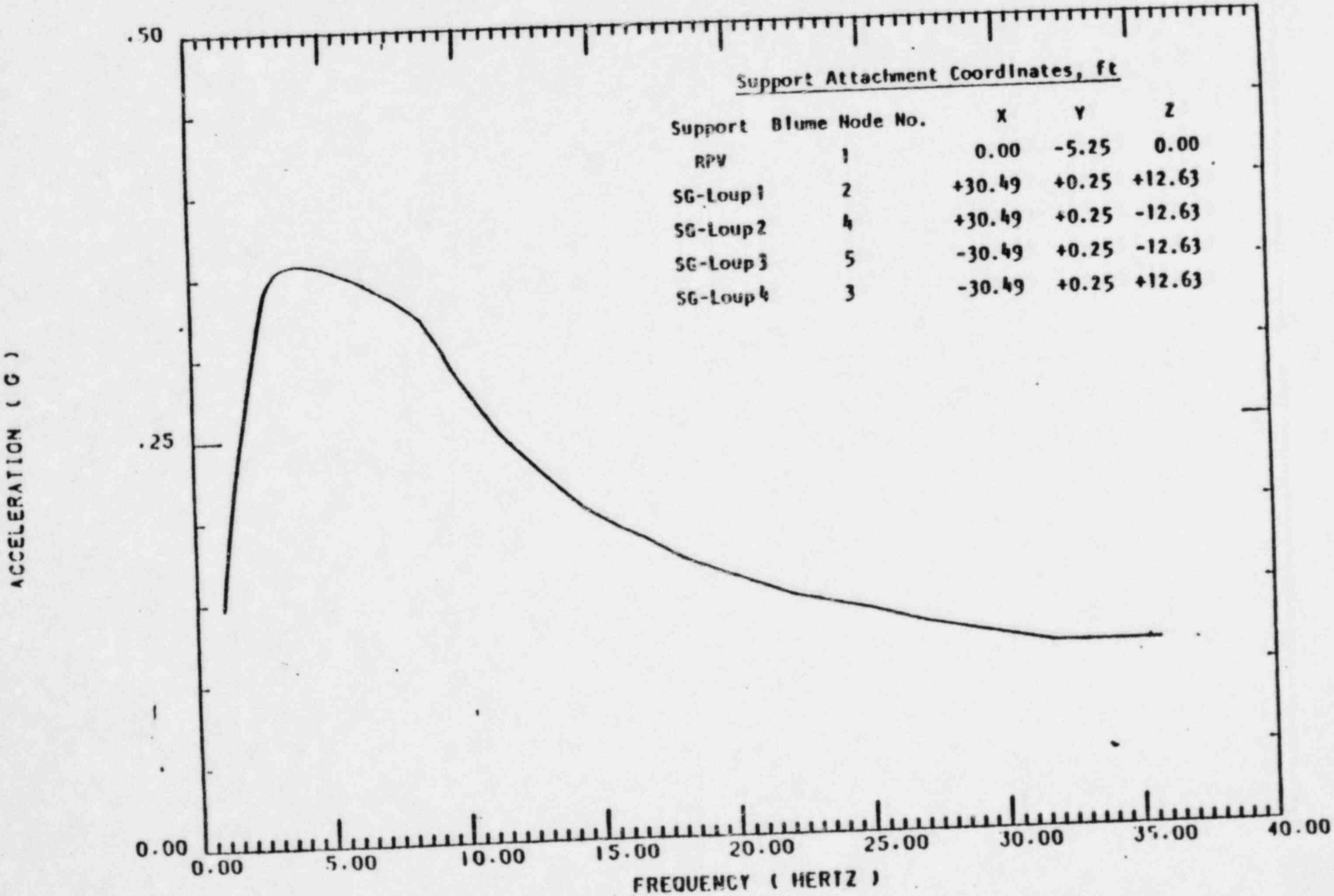

FIGURE 1 REACTOR PRESSURE VESSEL AND STEAM GENERATOR SUPPORTS BELOW EL. 1'-6"

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CONN. YANKEE CONTAINMENT INTERNALS
FREE FIELD SPECTRA (N-S)

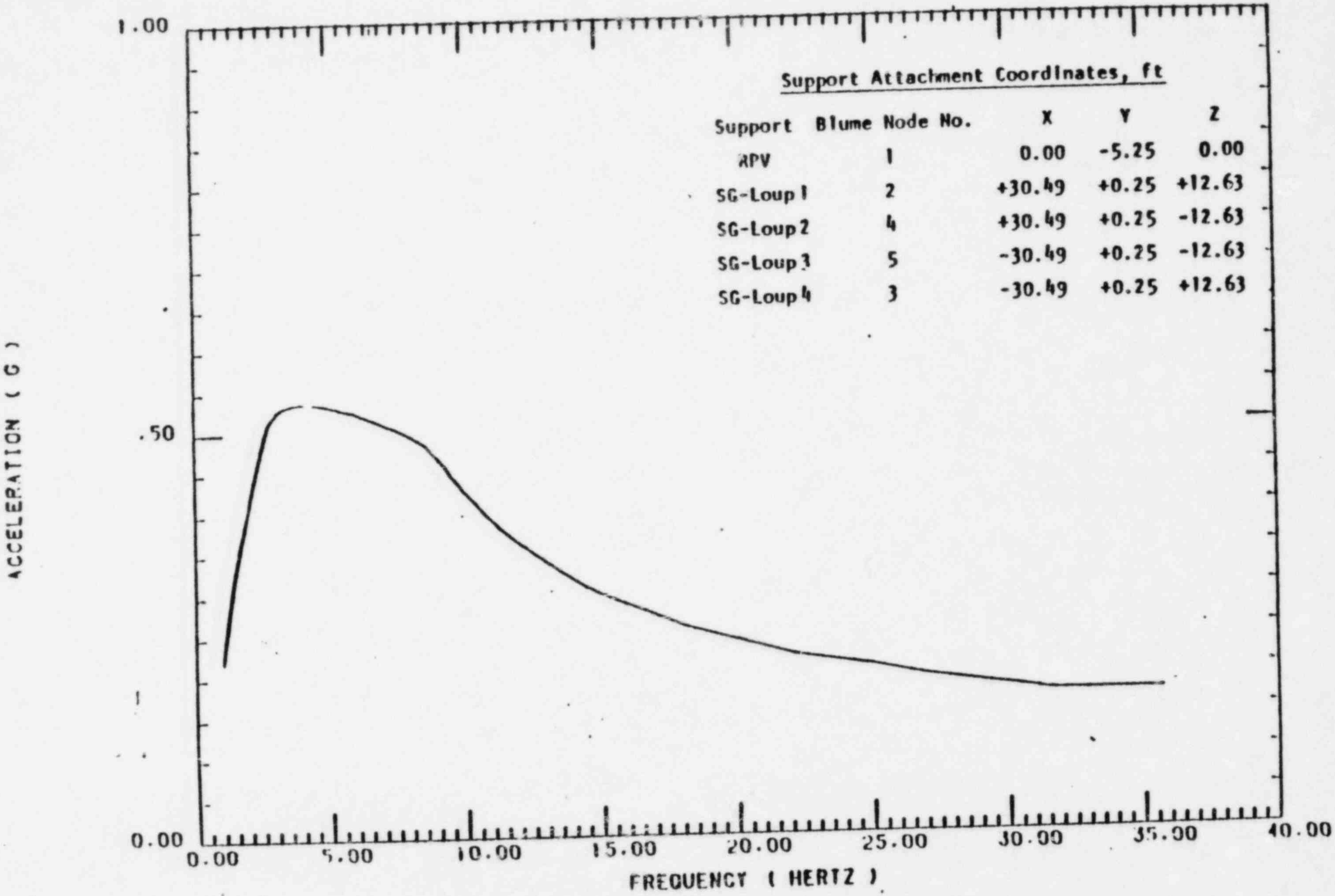

FIGURE 2 REACTOR PRESSURE VESSEL AND STEAM GENERATOR SUPPORTS BELOW EL. 1'-6"

CONN. YANKEE CONTAINMENT INTERNALS
FREE FIELD SPECTRA (E-W)

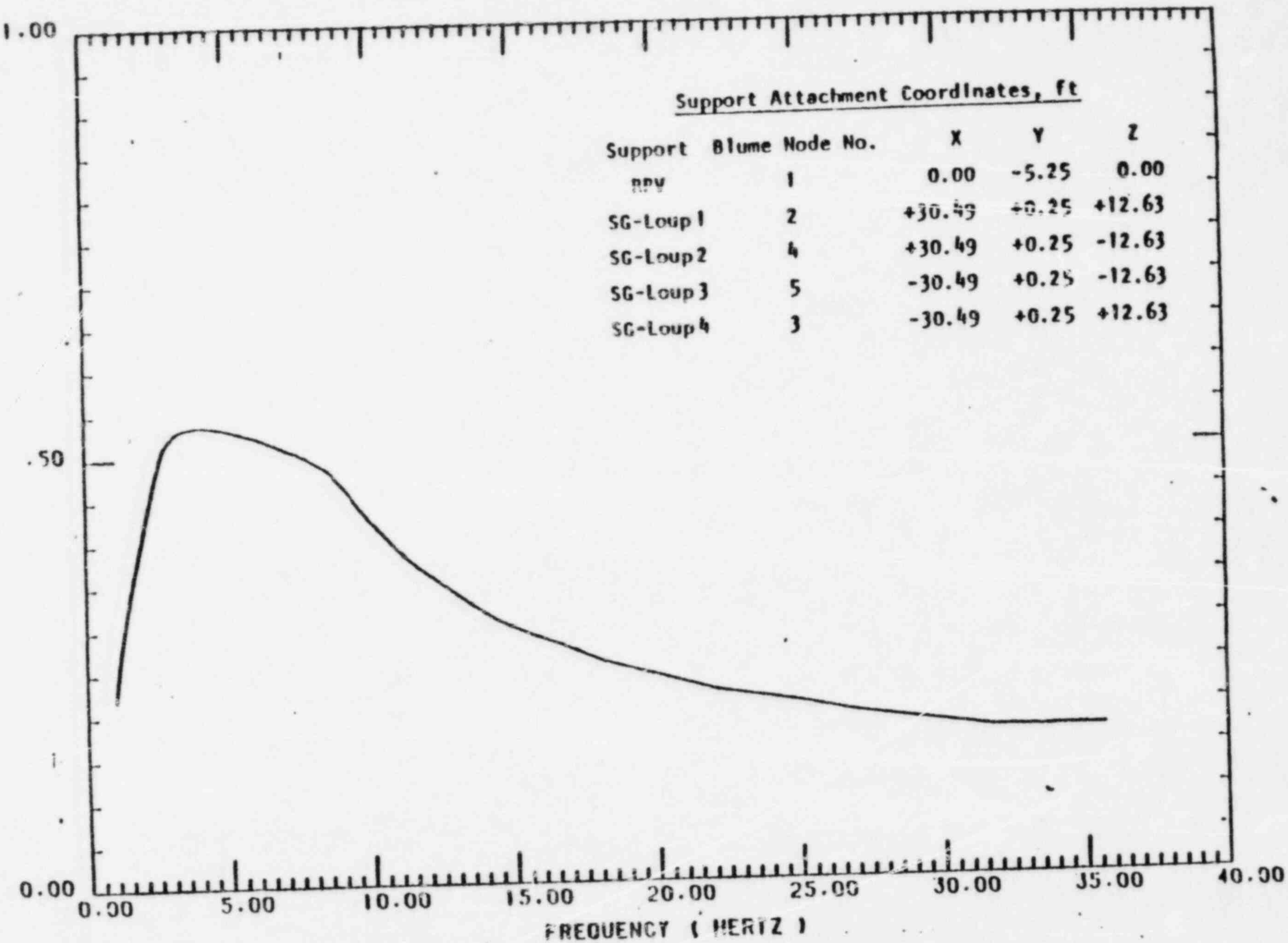

FIGURE 3 REACTOR PRESSURE VESSEL AND STEAM GENERATOR SUPPORTS BELOW EL. 1'-6"

WESTINGHOUSE PROPRIETARY CLASS 2

10868-2

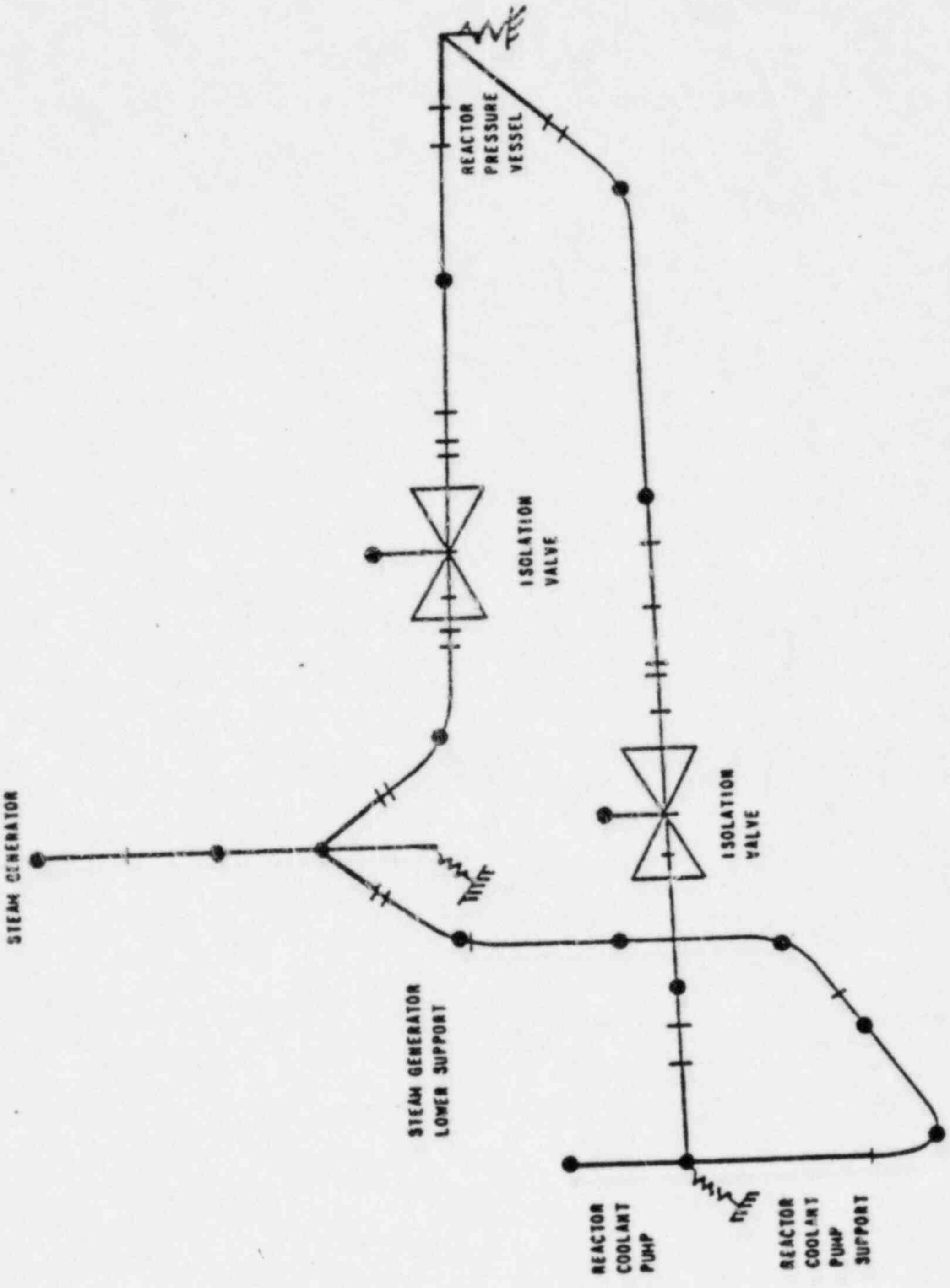

Figure 5: Typical Loop of 4-Loop Model

G-44.

SNTC-PAD-320

PRELIMINARY

CONNECTICUT YANKEE PRESSURIZER
SEISMIC ANALYSIS CALCULATIONS

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CHECKED BY:
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AUGUST, 1982

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CONNECTICUT YANKEE PRESSURIZER
SEISMIC ANALYSIS

1.0 SUMMARY OF RESULTS

The 1300 cubic ft. Connecticut Yankee Pressurizer was evaluated for dynamic seismic response. The evaluation addressed the surge, spray, safety and relief nozzles, shell, support skirt, and internals. All items were within ASME Code, Section III allowable limits and are summarized in Table 1-1.

Applied loads were derived from a response spectrum analysis of the CYW reactor coolant loop dynamic model.

Since earthquakes are oscillatory in nature, the sign on these quantities can be either plus or minus. In this analysis the most conservative combination of signs is used. On the following pages the state of stress in various regions of the steam generator is discussed in greater detail.

The method of analysis followed the procedures outlined in the evaluation of the San Onofre power plant as documented in Reference 5.

2.0 EVALUATION OF PRESSURIZER NOZZLE TO SHELL JUNCTIONS

The local shell stresses in the nozzle to shell junctions for the surge, spray, safety and relief nozzles are evaluated by using the BIJLAARD Method (Reference 2). Stresses in the nozzle to pipe junction are evaluated by using strength of materials equations. The loads were obtained from the response spectrum analysis of the entire loop. Stresses at the pressurizer nozzles were calculated using a load set which envelopes the occurance of faulted conditions

The basic equations used for the evaluation of the nozzle to shell junctions are detailed on the following pages. The governing equations for the BIJLAARD analysis are shown in Reference 2 with the explanation of terms and equations.

The BIJLAARD curves are limited in number and it is necessary to interpolate between the values on different curves. The various BIJLAARD curves (SM and SP series) are constructed for several combinations of nozzles and shells. The parameter T is used to identify the different nozzle geometries ($T = r_m/t$), and there are curves for T = 5, T = 15, and T = 50. The surge nozzle has a value of T = 2.2; the spray nozzle has T = 4.1; the safety and relief nozzles have T = 3.6. Comparisons between the T = 5 curves and T = 15 curves show very little difference in values in the regions of interest; therefore, it is assumed that T = 5 will adequately represent the T<1.0 cases.

For example, $\rho = T/t = .25, \quad U = r_o / . R_m T = 1.2$

	S_m Curves			S_p Curves	
	T = 5	T = 15		T = 5	T = 15
N_x	.035	.026	N_x	.034	.021
M_x	.059	.053	M_x	.041	.038
N_y	.048	.080	N_y	.050	.086
M_y	.017	.019	M_y	.010	.011

Figures 1-1 through 1-4 also contain the work sheets and calculation notes used for the evaluation of the shell to nozzle junctions for the surge, spray, safety, and relief nozzles respectively. From these stress components the principal stresses are obtained and the stress intensities evaluated for comparison to the code allowable.

Primary Membrane

Internal pressure in the shell also contributes to the overall stress state near the nozzles. The nozzle pressure stresses are calculated by using strength of materials equations in the channel head near the respective nozzles. For the surge nozzle (lower head):

$$\sigma_X = \sigma_{Axial} = pr_i/(2t) = 2050(40)/(2)(3.57) = 11485 \text{ psi}$$
$$\sigma_Y = \sigma_{Hoop} = pr_i/(2t) = 11485 \text{ psi}$$
$$\sigma_R = \sigma_{Radial} = -p/2 = -2050/2 = -1025 \text{ psi}$$

The gradient of radial stress through the thickness of the shell is a secondary type stress. Hence, only the average value of radial stress is used in the evaluation of faulted conditions. Since the shear is zero, the maximum membrane stress intensity due to internal pressure is P_m = 12.5 ksi. The allowable limit for primary membrane for faulted conditions is the lesser of 2.4 S_m or .7Su. Since $S_u > 3S_m$ a conservative limit for membrane becomes

$$.7\ (3\ S_m) = 2.1\ S_m$$

For S_m = 18,500 psi the limit becomes 38,850 psi.

Local Membrane Plus Bending

In order to complete the comparison of nozzle stress to code allowables, the pressure stress must be added to the external load stresses.

$$\sigma = \sigma_p + \sigma_e$$

Summation must be performed on a component level prior to stress intensity calculations. Furthermore, the average of the upper and lower surface stresses from the BIJLAARD analysis will be included. Thus, for the spherical shells:

$\sigma_A = pr_i/(2t) + (\sigma_{xU} + \sigma_{xL})/2$
$\sigma_H = pr_i/(2t) + (\sigma_{yU} + \sigma_{yL})/2$
$\tau_{AH} = (\tau_u + \tau_L)/2$, no pressure contribution
$\sigma_R = -p/2$, no nozzle contribution

Utilizing the above equations, the location of the maximum stress intensity on the surge nozzle as shown on the BIJLAARD table in Figure 1-1 is location D. The primary local membrane is evaluated as follows:

$\sigma_A = 11485 + (-1118 + 1430)/2 = 11641$ psi

$\sigma_H = 11485 + (-357 + 629)/2 = 11621$ psi

$\tau_{AH} = (643 + 643)/2 = 643$ psi

$\sigma_R = -2050/2 = -1025$ psi

The bending stress in the nozzle to shell junction which results from the application of external loads is secondary and need not be evaluated for the faulted condition limits. The total stresses result from adding the pressure stresses to the external load local membrane stresses. Thus, the primary local membrane stress intensity due to pressure and nozzle loads is $P_L + P_b = 13.3$ ksi. The allowable limit for membrane plus bending for the faulted condition is 1.5 times the primary membrane limit. That is,

$$P_L + P_B \leq 1.5\ (.7\ S_u) = 3.15\ S_m$$

for $S_u = 3S_m$. For $S_m = 18500$ psi, the limit becomes 58,300 psi.

The same procedure is used for the evaluation of the shell to nozzle junctions for the spray, safety, and relief nozzles. The results of these calculations are summarized in Table 1-1.

It should be noted that the membrane stress in the shell near the nozzle opening as a result of external loads is classified as primary local stress intensity according to the ASME Code.

3.0 EVALUATION OF PRESSURIZER NOZZLE TO PIPE JUNCTIONS

The stresses in the nozzle to pipe junction are evaluated by using basic strength of materials equations. The geometric properties used in the analysis are the pipe cross-sectional area, flexural moment of inertia, and torsional moment of inertia. The geometric properties for the four nozzle/pipe junctions are listed below:

	Surge	Spray	Safety	Relief
$A(in^2)$	23.85	6.62	3.02	3.02
$I(in^4)$	299	13.27	5.03	5.03
$J(in^4)$	599	26.54	10.06	10.06

The resultant shear and bending moment are obtained by calculating the magnitude of the individual vectors.

$$V = (V_1^2 + V_2^2)^{1/2}$$
$$M = (M_1^2 + M_2^2)^{1/2}$$

<u>Membrane</u>

Consider the evaluation of the surge nozzle pipe for the faulted condition. First the primary membrane stress intensity must be evaluated. The axial membrane stress is the result of internal pressure plus the axial force on the nozzle/pipe junction. The hoop stress results only from the internal pressure as does the radial stress. The shear stress is the result of the shear force and the torque on the pipe. The shear stress due to the pipe torque is evaluated at the midsurface of the pipe.

$$\sigma_A = pr_i/(2t) + P/A$$
$$\sigma_H = pr_i/t$$
$$\tau_{AH} = V/A + M_t r_m/J$$
$$\sigma_R = -p/2$$

The pipe loads for the <u>surge nozzle</u> are as follows:

$$P = 1050 \text{ lbs} \qquad M_1 = 251000 \text{ in-lbs}$$
$$V_1 = 4400 \text{ lbs} \qquad M_2 = 194000 \text{ in-lbs}$$
$$V_2 = 4400 \text{ lbs} \qquad M_T = 415000 \text{ in-lbs}$$

$$V = (V_1^2 + V_2^2)^{1/2} = 6223 \text{ lbs}$$
$$M = (M_1^2 + M_2^2)^{1/2} = 317233 \text{ in-lbs}$$

The primary membrane stress components are determined from the previously mentioned equations:

$$\sigma_A = pr_i/(2t) + P/A = 2050(4.615)/(2)(.76) + 1050/23.85$$
$$= 6268 \text{ psi}$$
$$\sigma_H = pr_i/t = 2050(4.615)/.76 = 12448 \text{ psi}$$
$$\tau_{AH} = V/A + M_T r_m/J = 6223/23.85 + 415000\ (4.995)/598.6$$
$$= 3723 \text{ psi}$$
$$\sigma_R = -p/2 = -2050/2 = -1025 \text{ psi}$$

From the above the principal stresses and stress intensities are evaluated, and P_m = 15.2 ksi. The maximum membrane stress intensity is limited by 2.1 S_m for the pipe safe end forging. For S_m = 18,500 the limit becomes 38,850. Hence, the nozzle to pipe stresses for primary membrane are within the allowable limit for the faulted condition.

Membrane Plus Bending

The same procedure is used for the membrane plus bending evaluation with the exception that the stresses are calculated for the outside surface of the pipe. The gradient of radial stress through the thickness of the pipe is a secondary type stress. Hence, only the average value of radial stress is used in the evaluation of faulted conditions.

The primary membrane plus bending stress intensities are evaluated as follows:

$\sigma_A = pr_i/(2t) + P/A + Mr_o/I$
$= 6268 + 317233(5.375)/299.3 = 11965$ psi
$\sigma_H = pr_i/t = 12448$ psi
$\tau_{AH} = V/A + M_T r_o/J = 3723$ psi
$\sigma_R = -p/2 = -1025$ psi

And the principal stresses and stress intensities are thus evaluated as the maximum membrane plus bending stress intensity = $P_L + P_b =$ 17.0 ksi which is within the allowable limit of 3.15 S_m = 58,300 psi.

The same procedure is used for the evaluation of the nozzle to pipe junctions for the spray, safety and relief nozzles. The results of these calculations are contained in Table 1-1.

4.0 EVALUATION OF PRESSURIZER SUPPORT SKIRT

The stresses in the support skirt are evaluated by use of strength of materials equations. The maximum forces and moments that are applied to the support skirt are obtained from the response spectrum analysis of the reactor coolant loop:

F_x = 144000 lbs	M_x = 6344000 in-lbs
F_y = 79000 lbs	M_y = 3443000 in-lbs
F_z = 102000 lbs	M_z = 3619000 in-lbs

The combined external forces and moments in the support skirt are:

P = 144000 lbs
V = 129016 lbs
M = 4995139 in-lbs
M_T = 6344000 in-lbs

The pressurizer deadweight, DW = 234978 lbs.

The support skirt area and sectional inertias are:

Area = 219 in^2
Flexural Inertia = 263200 in^4
Torsional Inertia = 526400 in^4
c = O.D./2 = 41.25 in

The primary membrane stresses are calculated from the following:

$$\sigma_A = (P + DW)/\overset{Z}{A} = 1188 \text{ psi}$$
$$t_A = V/A + \underbrace{TC}_{m_T(c)}/J = 902 \text{ psi}$$

Principal membrane stresses are calculated and the maximum stress intensity is determined to be 2160 psi. The allowable limit for primary membrane for faulted conditions is the lesser of 2.4 S_m or .7 S_u. Since $S_u > 3\ S_m$ a conservative limit for membrane becomes

$$.7(3\ S_m) = 2.1\ S_m$$

For S_m = 18,500 psi the limit becomes 38850 psi.

The membrane plus bending stresses are calculated from the following:

$${}_A = (P + DW)/A + M(c)/I = 1971 \text{ psi}$$

$${}_A = V/A + \underbrace{TC}_{m_T(c)}/J = 902 \text{ psi}$$

Principal stresses are calculated and the maximum stress intensity is determined to be 2671 psi. The allowable limit for membrane plus bending for the faulted condition is 1.5 (.7 S_u). A conservative limit for membrane plus bending becomes

$$1.5(.7)(3\ S_m) = 3.15\ S_m$$

For S_m = 18500 psi, the limit becomes 58275 psi.

5. EVALUATION OF PRESSURIZER SHELL STRESSES

The shell stresses are the result of pressure stresses and a seismic bending moment in the shell. The shell maximum seismic bending moment, as predicted by the response spectrum analysis of the reactor coolant loop is:

$$M = 11005000 \text{ in-lbs}$$

The shell cross-sectional properties are as follows:

$$r_i = 40 \text{ in}$$
$$t = 5.44 \text{ in}$$
$$I = 1340000 \text{ in}^4$$

The primary membrane stresses are calculated as:

$$\sigma_A = Pr_i/(2t) + M(r_i)/I = 7522 + 351 = 7873 \text{ psi}$$
$$\sigma_H = Pr_i/t = 15044 \text{ psi}$$
$$\sigma_R = -P/2 = -1025 \text{ psi}$$

The maximum primary membrane stress intensity is calculated as:

$$P_m = \sigma_H - \sigma_R = 16069 \text{ psi} < 2.1\ S_m = 38850 \text{ psi}$$

The maximum membrane plus bending stress intensity is the same as above:

$$P_L + P_b = 16069 \text{ psi} < 3.15\ S_m = 58275 \text{ psi}$$

6.0 EVALUATION OF HEATER ROD STRESSES

The heater rod stresses are the result of external pressure stresses and a seismic bending moment in the heater rod. Heater rod stress evaluation reported in Reference 5 indicated primary general membrane and membrane plus bending stress intensities of 5.3 ksi, well within the respective allowable limits of 31 ksi and 46.6 ksi. Heater rod stress levels in the Connecticut Yankee Pressurizer can be no greater than those reported in Reference 5 due to the fact that spectral accelerations on the Connecticut Yankee pressurizer, at the natural frequencies of the heater rod as reported in Reference 5, are lower than those presented in Reference 5.

The heater rod stresses are thus within code allowables for the faulted condition.

TABLE 1-1

CYW PRESSURIZER STRESSES[1]

Units in ksi

Description	P_m [2] Stress	P_m [2] Allowable	P_m [2] Surplus Margin	$P_L + P_B$ [2] Stress	$P_L + P_B$ [2] Allowable	$P_L + P_B$ [2] Surplus Margin
Surge Nozzle:						
- Shell Junction	12.5	38.8	3.10	13.3	58.3	4.38
- Pipe Junction	15.2	35.7	2.35	17.0	53.6	3.15
Spray Nozzle:						
- Shell Junction	16.6	38.8	2.34	18.4	58.3	3.17
- Pipe Junction	16.0	35.7	2.23	28.7	53.6	1.87
Safety Nozzle:						
-Shell Junction	16.6	38.8	2.34	17.8	58.3	3.28
-Pipe Junction	21.0	35.7	1.70	22.8	53.6	2.35
Relief Nozzle:						
-Shell Junction	16.6	38.8	2.34	17.7	58.3	3.29
-Pipe Junction	12.7	35.7	2.81	26.0	53.6	2.06
Support Skirt:	2.2	38.8	17.55	2.7	58.3	21.59
Shell:	16.1	38.8	2.41	16.1	58.3	3.62
Heater Rod:	5.3	31.0	5.85	5.3	46.6	8.79

(1) Stresses are for worst case load combinations.

(2) Stress Limits: $P_m \leq .7\ S_u = 2.1\ S_m$

$P_L + P_B \leq 1.5\ (.7\ S_u) = 3.15\ S_m$

LIST OF REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

2. Wichman, K. R., and Hopper, A. G., and Mershon, J. L., "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings," Welding Research Council Bulletin 107, March 1979.

3. Budynas, R. G., Advanced Strength and Applied Stress Analysis, McGraw-Hill Book Company, New York, 1977.

4. Wang. C.T., Applied Elasticity, McGraw-Hill Book Company, New York, 1953.

5. Clinton, K. E., "Time History Analysis fo the Pressurizer for the SCE San Onofre Power Plant", Westinghouse Electric Corporation, Tampa Division, WTD-SM-75-036, January, 1975.

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SNTC-PAD-320

PRELIMINARY

CONNECTICUT YANKEE STEAM GENERATOR
DYNAMIC SEISMIC ANALYSIS
CALCULATIONS

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JULY, 1982

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CONNECTICUT YANKEE STEAM GENERATOR DYNAMIC SEISMIC ANALYSIS

1.0 SUMMARY OF RESULTS

The 27.700 ft.^{2} Connecticut Yankee steam generators were analyzed for dynamic seismic response. The regions evaluated included the primary and secondary nozzles, shells, U-bend tubing, and lower support brackets. All sections were within ASME Code, Section III allowable limits and are summarized in Table 1-1.

Applied loads were derived from a response spectrum analysis of the reactor coolant loop dynamic model. Added conservatism was introduced by utilizing umbrella loads representing the maximum response expected to occur in all four loops.

Since earthquakes are oscillatory in nature, the sign on these quantities can be either plus or minus. On the following pages the state of stress in various regions of the steam generator is discussed in greater detail.

The method of analysis followed the procedures outlined in the evaluation of the San Onofre power plant.

2.0 EVALUATION OF STEAM GENERATOR NOZZLE TO SHELL JUNCTIONS

The local shell stresses in the nozzle to shell junctions for the primary and secondary nozzles are evaluated by using the BIJLAARD Method (Reference 2). Stresses in the nozzle to pipe junction are evaluated by using strength of materials equations.

The maximum forces and moments applied to the nozzles were obtained from the response spectrum analysis of the entire loop. Stresses at the steam generator nozzles were calculated using a load set derived from combination of the maximum loads on all four steam generators. Therefore, the stresses reported envelope those occurring on all four steam generators.

The basic equations used for the evaluation of the nozzle to shell junctions are detailed on the following pages. The governing equations for the BIJLAARD analysis are found in Reference 2 with the explanation of terms and equations.

The BIJLAARD curves are limited in number and it is necessary to interpolate between the values on different curves. The various BIJLAARD curves (SM and SP series) are constructed for several combinations of nozzles and shells. The parameter T is used to identify the different sizes of nozzles, ($T = r_m/t$). There are, for example, curves for T = 5, T = 15, and T = 50. The primary inlet and outlet nozzles have a T of 4.51, and therefore, a value of T = 5 is used. The feedwater nozzle (cylindrical shell junction) has a T of 16.7, and a value of T = 15 will be used for analysis purposes. The steam outlet nozzle has a value of T = 1.1 and, like the primary nozzles, a value of T = 5 is used. Comparisons between the T = 5 curves and T = 15 curves show very little difference in values in the regions of interest; therefore, it is assumed that T = 5 will adequately represent the T = 1 case.

For example. $\rho = T/t = .25$. $U \quad / \sqrt{R_m T} = 1.2$

	S_m Curves			S_p Curves	
	T = 5	T = 15		T = 5	T = 15
N_x	.035	.026	N_x	.034	.021
M_x	.059	.053	M_x	.041	.038
N_y	.048	.080	N_y	.050	.086
M_y	.017	.019	M_y	.010	.011

Figures 1-1 through 1-4 display the BIJLAARD the work sheets and calculation notes used for the evaluation of the shell to nozzle junctions for the primary inlet, primary outlet, feedwater, and steam outlet nozzles respectively. From these stress components the principal stresses are obtained and the stress intensities evaluated for comparison to the code allowables.

Membrane

Internal pressure in the shell also contributes to the overall stress state near the nozzles. The primary nozzle pressure stresses are calculated by using strength of materials equations in the channel head near the inlet and outlet nozzles:

$$\sigma_X = \sigma_{Axial} = pr_i/(2t) = 2050(54.06)/(2)(8.06) = 6875 \text{ psi}$$
$$\sigma_Y = \sigma_{Hoop} = pr_i/(2t) = 6875 \text{ psi}$$
$$\sigma_R = \sigma Radial = -p/2 = -2050/2 = -1025 \text{ psi}$$

The gradient of radial stress through the thickness of the shell is a secondary type stress. Hence, only the average value of radial stress is used in the evaluation of faulted conditions. Since the shear is zero. the maximum membrane stress intensity due to internal pressure is P_m = 7900 psi. The allowable limit for primary membrane for faulted conditions is the lesser of 2.4 S_m or .7Su. Since $S_u > 3S_m$ a conservative limit for membrane becomes

$$.7\ (3\ S_m) = 2.1\ S_m$$

For S_m = 18.500 psi the limit becomes 38.850 psi.

Local Membrane Plus Bending

In order to complete the comparison of nozzle stress to code allowables, the pressure stress must be added to the external load stresses.

$$\sigma = \sigma_p + \sigma_e$$

Summation must be performed on a component level prior to stress intensity calculations. Furthermore, the average of the upper and lower surface stresses from the BIJLAARD analysis will be included. Thus, for the spherical shells:

$\sigma_A = pr_i/(2t) + (\sigma_{xU} + \sigma_{xL})/2$
$\sigma_H = pr_i/(2t) + (\sigma_{yU} + \sigma_{yL})/2$
$\tau_{AH} = (\tau_u + \tau_L)/2$, no pressure contribution
$\sigma_R = -p/2$, no nozzle contribution

Utilizing the above equations, the location of the maximum stress intensity on the inlet nozzle as shown on the BIJLAARD table in Figure 1-1 is location B. The primary local membrane is evaluated as follows:

$$\sigma_A = 6875 + (4057 - 1369)/2 = 8219 \text{ psi}$$

$$\sigma_H = 6875 + (4828 - 1733)/2 = 8423 \text{ psi}$$

$$\tau_{AH} = (31 + 31)/2 = 31 \text{ psi}$$

$$\sigma_R = -2050/2 = -1025 \text{ psi}$$

The bending stress in the nozzle to shell junction which results from the application of external loads is secondary and need not be evaluated for the faulted condition limits. The total stresses result from adding the pressure stresses to the external load local membrane stresses. Thus, the primary local membrane stress intensity due to pressure and nozzle loads is P_L+P_b=9.5 ksi. The allowable limit for membrane plus bending for the faulted condition is 1.5 times the primary membrane limit. That is,

$$P_L + P_B \leq 1.5\ (.7\ S_u) = 3.15\ S_m$$

for $S_u = 3S_m$. For S_m = 18500 psi, the limit becomes 58.300 psi.

The same procedure is used for the evaluation of the shell to nozzle junctions for the primary outlet, feedwater inlet, and steam outlet nozzles. The results of these calculations are summarized in Table 1-1.

It should be noted that the membrane stress in the shell near the nozzle opening as a result of external loads is classified as primary local stress intensity according to the ASME Code.

3.0 EVALUATION OF STEAM GENERATOR NOZZLE TO PIPE JUNCTIONS

The stresses in the nozzle to pipe junction are evaluated by using basic strength of materials equations. The geometric properties used in the analysis are the junction cross-sectional area, flexural moment of inertia, and torsional moment of inertia. The geometric properties for the four nozzle/pipe junctions are listed below:

	Inlet	Outlet	Feedwater	Steam
$A(in^2)$	287	258	31.2	67.9
$I(in^4)$	40472	45027	687	4521
$J(in^4)$	80945	90054	1374	9042

The resultant shear and bending moment are obtained by calculating the magnitude of the individual vectors.

$$V = (V_1^2 + V_2^2)^{1/2}$$
$$M = (M_1^2 + M_2^2)^{1/2}$$

Membrane

Consider the evaluation of the primary inlet pipe for the faulted condition. First the primary membrane stress intensity must be evaluated. The axial membrane stress is the result of internal pressure plus the axial force on the nozzle/pipe junction. The hoop stress results only from the internal pressure as does the radial stress. The shear stress is the result of the shear force and the torque on the pipe. The shear stress due to the pipe torque is evaluated at the midsurface of the pipe.

$$\sigma_A = pr_i/(2t) + P/A$$
$$\sigma_H = pr_i/t$$
$$\tau_{AH} = V/A + M_t r_m/J$$
$$\sigma_R = -p/2$$

The pipe loads for the inlet nozzle are as follows:

$P = 234000$ lbs $\quad M_1 = 1.5 \times 10^6$ in-lbs

$V_1 = 217000$ lbs $\quad M_2 = 4.3 \times 10^6$ in-lbs

$V_2 = 32000$ lbs $\quad M_T = 2.9 \times 10^6$ in-lbs

$$V = (V_1^2 + V_2^2)^{1/2} = 219000 \text{ lbs}$$
$$M = (M_1^2 + M_2^2)^{1/2} = 4.6 \times 10^6 \text{ in-lbs}$$

The primary membrane stress components are determined from the previously mentioned equations:

$$\sigma_A = pr_i/(2t) + P/A = 2050(15.41)/(2)(2.72) + 234000/287$$
$$= 6622 \text{ psi}$$
$$\sigma_H = pr_i/t = 2050(15.41)/2.72 = 11614 \text{ psi}$$
$$\tau_{AH} = V/A + M_T r_m/J = 303289/287 + 2.9 \times 10^6(16.77)/80945$$
$$= 1364 \text{ psi}$$
$$\sigma_R = -p/2 = -2050/2 = -1025 \text{ psi}$$

From the above the principal stresses and stress intensities are evaluated. and $P_m = 13.0$ ksi. The maximum membrane stress intensity is limited by 2.1 S_m for the pipe safe end forging. For $S_m = 18{,}500$ the limit becomes 38,850. Hence, the nozzle to pipe stresses for primary membrane are within the allowable limit for the faulted condition.

Local Membrane Plus Bending

The same procedure is used for the membrane plus bending evaluation with the exception that the stresses are calculated for the outside surface of the pipe. The gradient of radial stress through the thickness of the pipe is a secondary type stress. Hence, only the average value of radial stress is used in the evaluation of faulted conditions.

The primary membrane plus bending stress intensities are evaluated as follows:

$\sigma_A = pr_i/(2t) + P/A + Mr_o/I$
$= 5807 + 857 + 4.6 \times 10^6(18.13)/40472 = 10965$ psi

$\sigma_H = pr_i/t = 11614$ psi

$\tau_{AH} = V/A + M_T r_o/J + 1057 + 2.9 \times 10^6(18.13)/80945 = 1415$ psi

$\sigma_R = -p/2 = -1025$ psi

And the principal stresses and stress intensities are thus evaluated as the maximum membrane plus bending stress intensity = $P_L + P_b = 13.2$ ksi which is within the allowable limit of $3.15\ S_m = 58.300$ psi.

The same procedure is used for the evaluation of the nozzle to pipe junctions for the primary outlet, feedwater inlet, and steam outlet nozzles. The results of these calculations are contained in Table 1-1.

4.0. E.ALUATION OF LOWER SHELL STRESSES

The lower shell stresses are derived from pressure stresses and bending moment in the shell. The primary membrane stress intensity results from internal pressure and bending moment acting on the shell. The shell material is SA-212 Gr B with an S_m value of 19.000 psi.

Membrane

For SSE. the maximum bending moment was 50.9 X 10^6 in-lbs in the lower shell.

$$\sigma_A = pr_i/(2t) + Mc/I = 675(56)/(2)(3.38) + 50.9 \times 10^6(59.5)/(2.24 \times 10^6) = 5592 + 1352 = 6944 \text{ psi}$$

$$\sigma_H = pr_i/t = 675(56)/3.38 = 11184 \text{ psi}$$

$$\sigma_R = -p/2 = -675/2 = -338 \text{ psi}$$

The maximum stress intensity is:

$$P_m = 11184 + 338 = 11522 \text{ psi} < 2.1S_m = 39.900 \text{ psi}$$

Membrane Plus Bending

Membrane plus bending stresses are the same as above.

5.0 EVALUATION OF TRANSITION CONE STRESSES

The state of stress in the cone is the result of both internal pressure and a shell moment. The primary membrane stress intensity is derived from internal pressure and overturning moment. The cone material is SA-212 Gr B with an S_m value of 19,000 psi.

Membrane

For SSE, the maximum bending moment was 47.7×10^6 in-lbs at the bottom of the transition shell.

$$\sigma_A = pr_i/\{2t(\cos\theta)\} + Mc/I$$
$$= 675(60.44)/(2)(3.88)(.98) + 47.7 \times 10^6(64.13)/3.07 \times 10^6$$
$$= 6361 \text{ psi}$$
$$\sigma_H = pr_i/\{t(\cos\theta)\} = 10729 \text{ psi}$$
$$\sigma_R = -p/2 = -675/2 = -338 \text{ psi}$$

The maximum stress intensity is:

$P_m = 11067$ psi $< 2.1\ S_m = 39{,}900$ psi

Membrane Plus Bending

The membrane plus bending stress intensity is the same as the membrane stress intensity, and hence the faulted condition requirements are met.

6.0 EVALUATION OF UPPER SHELL STRESSES

The state of stress in the upper shell is the result of the application of a shell moment due to bending and internal pressure. The primary membrane stress intensity is derived from internal pressure and overturning moment. The shell material is SA-212 Gr B with an S_m value of 19,000. No upper seismic pad is present to contribute to local shell stress.

Membrane

For this evaluation an umbrella analysis will be used for SSE where the maximum bending moment was 27.9 X 10^6 in-lbs in the upper shell assembly.

$$\sigma_A = pr_i/(2t) + Mc/I$$
$$= 675(68)/(2)(4.13) + 27.9 \times 10^6(72)/4.03 \times 10^6 = 6055 \text{ psi}$$
$$\sigma_H = pr_i/t = 675(68)/4.13 = 11114 \text{ psi}$$
$$\sigma_R = -p/2 = -338 \text{ psi}$$

$$P_m = 11114 + 338 = 11452 \text{ psi} < 2.1\ S_m = 39.900 \text{ psi}$$

Membrane Plus Bending

The membrane plus bending stress intensity is the same as above.

$$P_L + P_B = 11452 \text{ psi} < 3.15\ S_m = 59.850 \text{ psi}$$

Hence, the steam generator upper shell is adequate to withstand the faulted condition loadings.

7.0 EVALUATION OF SUPPORT LOADS

The support bracket is a three piece welded assembly consisting of two clevis plates and one support plate. Four bracket assemblies support the steam generator. The stresses in the bracket result from axial and shear loads. Since the brackets are welded to the tubesheet forging, shell stresses do not apply. The assembly layout and dimensions are shown in Drawings 7-1 and 7-2.

The maximum loads were determined from the response of the reactor coolant loop dynamic model:

$$F = 1057 \text{ kips}$$
$$V = 249 \text{ kips}$$

where F is the vertical load (inclusive of deadweight), and V is the tangential load. These loads are derived from the reactor coolant loop model and the steam generator geometry.

The material for the three plates was ASTM-A-212 carbon steel, grade B, with $S_u = 70$ ksi.

Membrane

Classical strength of materials equations were utilized to evaluate the maximum stress intensity for P_M. For the stress due to axial load F:

$$\tau_{xy} = F/A_t = 1057/154.63 = 6.8 \text{ ksi}$$

where A_t is the total weld area on the shell per bracket. For the stress due to shear load V:

$$\tau_{xz} = V/A_t = 249/154.63 = 1.6 \text{ ksi}$$

All other stress components are assumed zero. Solution of the stress tensor produces:

$$P_m = 14.0 \text{ ksi} \quad .7S_u = 49 \text{ ksi}$$

Local Membrane Plus Bending

A composite moment of inertia was calculated for the entire support bracket assembly utilizing classical techniques: G-73

$$Y = \sum y_i A_i / A_t$$
$$I = \sum (I_i + A_i d_i^2)$$

The bending due to axial load F:

$$\sigma_x = Mc/I = Fa(h + t_s - Y)/I_z$$
$$= 1057(9)(10.78)/4745.6 = 21.6 \text{ ksi}$$

and the bending due to shear load V:

$$\sigma_x = Mc/I = Va(w/2)/I_y$$
$$= 249(9)(15/5)/2(4802.5) = 3.6 \text{ ksi}$$

The shear load also causes a torsional load on the bracket due to the centroidal location:

$$T = V(Y - t_s/2)$$
$$= 249(7.47 - 4/2) = 1362 \text{ in-kips}$$

Using an approach outlined by Wang (Reference 4):

$$\tau_{max} = 3T(t_i)/(b_1 t_1^3 + 2b_2 t_2^3)$$
$$= 3(1362)(4)/\{(12.25)(4)^3 + 2(16.25)(3.25)^3\}$$
$$= 8.6 \text{ ksi}$$

τ_{max} could be either τ_{xy} or τ_{xz} depending upon the location on the bracket. The upper corners of the clevis to tubesheet weld are the most conservative locations, and thus

$$\tau_{max} = \tau_{xy} = 8.6 \text{ ksi}$$

Combining and solving the stress components for membrane plus bending produce

$$P_L + P_b = 39.9 \text{ ksi}$$

which is within the limit

$$1.5(.7)S_u = 73.5 \text{ ksi}$$

These numbers are shown in Table 1-1 and indicate that the faulted condition requirements are met.

8.0 EVALUATION OF TUBE BUNDLE

8.1 Introductory Remarks

In order to evaluate the SSE induced stresses in the U-tubes, a response spectrum analysis is performed using a detailed mathematical model of the CYW steam generator. The basis for the model is a 27,000 ft^2 steam generator model developed for the SCE seismic reevaluation presented in Reference 5. The model was modified, however, to account for geometric and weight differences for the CYW steam generators, as outlined in Appendix I. In addition to the geometry and weight differences, the actual CYW support and nozzle stiffnesses were incorporated into the seismic model. The response spectra used in the evaluation are presented in Figure 8-2. Two cases of lower support stiffnesses are evaluated to envelope the possible variations of stiffness at that location.

8.2 Steam Generator Model

A lumped mass model of the CYW steam generator is shown in Figure 8-1. The steam generator is idealized using beam elements and elastic support elements. Beam elements are used to represent the steam generator shell, tube bundle and other internals. Massless elastic support elements represent the stiffness of the lower support system and attached piping.

Nodal point coordinates and beam element data in the form of cross-sectional areas, flexural moments of inertia and outside radii of the cross sections are presented in Appendix I. The global coordinate system employed in the analysis is illustrated in Figure 8-1. The X and Y axes are horizontal and the Z axis is vertical.

The dry weight of the steam generator shell and internals, in addition to the weight of the primary and secondary water, is lumped at the nodal points of the assemblage. Water weights are based on the vessel at the 100% load normal operating condition. Lumped mass data is summarized in Appendix I. The masses are lumped at nodal points of the model for directions corresponding to horizontal and vertical translations of the steam generator. Rotational inertias for torsional vibration of the steam generator are also defined.

Water weights are divided between the nodes of the internals and the nodes of the steam generator shell. The portion of water assigned to the internals represents the hydrodynamic mass of the component.

The steam generator mathematical model is comprised of a shell beam (Elements 1 thru 14), a tube bundle beam (Elements 15 thru 35), and an upper internals beam (Elements 36 thru 41). The shell beam, tube bundle beam, and upper internals beam are located along the longitudinal axis of the steam generator. The horizontal linkages, indicated by dashed lines in Figure 8-1, represent coupling between the steam generator shell beam and the tube bundle and upper internals beams.

8.3 Results of Modal Analysis - Natural Frequencies and Mode Shapes

Modal analyses of the steam generator were performed using the WECAN computer program. The results of these analyses for the two lower support stiffnesses evaluated are summarized in Tables 8-1 and 8-2.

8.4 Results of Dynamic Seismic Analysis - Tube Bundle Response

The maximum combined stresses considering all three shock directions were calculated using the WECAN program post processor COMSPC. The maximum tube bending stress occurred at node 32 and was 3.2 ksi. This value was determined by using the Westinghouse method of combining the results of the 3 shock directions which includes effects of closely spaced modes. The evaluation of the U-tube stresses can be focused on the U-bend region. The U-bend region is selected as the point of interest since the U-bend experiences the highest bending stresses due to earthquake motion. The total stress distribution in the tube is the result of internal pressure in the tube and the bending stress due to earthquake motion. The tube material is SB-163, inconel with an S_m value of 26,700 psi.

Membrane

$\sigma_A = pr_i/2t = (1375)(3.2)/(2)(.055) = 3999$ psi

$\sigma_H = pr_i/t = (1375)(.32)/(.055) = 7998$ psi

$\sigma_R = p/2 = -1375/2 = -688$ psi

$P_m = \sigma_H - \sigma_R = 7998 - (-688) = 8686 \text{ psi} < 2.1\ S_m$
$= 48.930$ psi

Membrane Plus Bending

For the analysis of membrane plus bending, the bending stress in the model tube bundle beam is used. The stress is calculated in the COMSPC run for the Case 1 lower support stiffness.

$\sigma_A = 3999 + 1768 = 5767$ psi

$\sigma_H = 7998$ psi

$\sigma_R = -688$ psi

$P_m + P_b = 7998 - (-688) = 8686 \text{ psi} < 3.15\ S_m = 73.400$ psi

Table 8-3 presents a list of microfiche available in the SNTC files of the computer runs used in the tube bundle stress evaluation.

TABLE 1-1

STEAM GENERATOR STRESSES[1]

Units in ksi

Description	P_m[2] Stress	P_m[2] Allowable	P_m[2] Surplus Margin	$P_L + P_B$[2] Stress	$P_L + P_B$[2] Allowable	$P_L + P_B$[2] Surplus Margin
Primary Inlet:						
- Shell Junction	7.9	38.8	4.91	9.5	58.3	6.14
- Pipe Junction	13.0	33.6	2.58	13.2	50.4	3.82
Primary Outlet:						
- Shell Junction	7.9	38.8	4.91	9.1	58.3	6.41
- Pipe Junction	13.1	33.6	2.56	14.9	50.4	3.38
Steam Nozzle:						
-Shell Junction	11.6	39.9	3.44	12.3	60.0	4.88
-Pipe Junction	8.7	39.9	4.59	8.7	60.0	6.90
Feedwater Nozzle:						
-Shell Junction	11.4	39.9	3.50	12.0	60.0	5.00
-Pipe Junction	6.6	39.9	6.05	6.6	60.0	9.09
Shell Regions:						
-Lower Shell	11.5	39.9	3.47	11.5	59.9	5.21
-Transition Shell	11.1	39.9	3.59	11.1	59.9	5.40
-Upper Shell	11.5	39.9	3.47	11.5	59.9	5.21
Support Bracket:	14.0	49.0	3.50	39.9	73.5	1.84
Tube Bundle:	9.6	48.9	5.09	9.6	73.4	7.65

(1) Stresses are for a combination of the maximum loads on all four steam generators.

(2) Stress Limits: $P_m \leq .7\ S_u = 2.1\ S_m$

$P_L + P_B \leq 1.5\ (.7\ S_u) = 3.15\ S_m$

TABLE 8-1

NORMAL MODES OF THE STEAM GENERATOR

CASE 1 SUPPORT STIFFNESS

Mode	Frequency	Component	Direction	Description
1	6.0	Shell	X	Bending
2	6.2	Shell	Y	Bending
3	9.8	U-Bend	X	Bending
4	21.8	Shell	Y	Bending
5	24.2	U-Bend	Y	Bending

TABLE 8-2

NORMAL MODES OF STEAM GENERATOR

CASE 2 SUPPORT STIFFNESS

Mode	Frequency	Component	Direction	Description
1	6.1	Shell	X	Bending
2	6.4	Shell	Y	Bending
3	9.8	U-Bend	X	Bending
4	24.3	U-Bend	Y	Bending
5	25.9	Shell	X	Bending

TABLE 8-3

MICROFICHE AVAILABLE

Run	Date	Description
TFNECMB	8/14/82	WECAN Modal Analysis - Case 1 Support Stiffness
TFNEC16	8/14/82	WECAN Modal Analysis - Case 2 Support Stiffness
TFNECZR	8/16/82	COMSPC Post Processor - Case 1 Support Stiffness
TFNECZU	8/16/82	COMSPC Post Processor - Case 2 Support Stiffness

APPENDIX I - Steam Generator Dynamic Model Development

Dynamic Model Development

The dynamic model used in this seismic evaluation is developed from the 27,000 ft^2 steam generator model developed for the SCE seismic reevaluation program. The CYW steam generator is also a 27,000 ft^2 model; however, the upper shell sections are larger than those on the SCE steam generator. Therefore, the upper shell portions of the shell beam must be modified to account for the CYW geometry. The shell components remodeled are the transition cone, upper shell, and elliptical head and are discussed below.

1. Transition Cone

Inside radius at small end = 56.12 in.

Outside radius at small end = 60.00 in.

Inside radius at large end = 67.75

Outside radius at large end = 71.63 in.

Dry weight = 26.5 Kip from Dwg. 789D951

Wt. of H_2O = 2.06 Kip from Reference 5

Radius of Gyration = 65.0 in. - Estimated

$I_R = Mr^2 = 28{,}600/386 \times 65^2 = 313$ Kip-sec^2-in.

The cone is represented in the model by two beam elements - elements 59 and 61. The cross-sectional properties of each element are:

Element	R_o, in.	R_i, in.	A, $in.^2$	I, $in.^4$	J, $in.^4$	D_o in.
59	62.91	59.03	1486	2.76×10^6	5.53×10^6	125.82
61	68.72	64.84	1628	3.63×10^6	7.27×10^6	137.44

Transition Cone Lumped Mass Data

Node	Weight, Kip	J_R, Kip-sec^2-in.
7.9	7.2	78.2
8	14.3	156.5

2. Upper Shell

$d_o = 135.5 + 2 \times 4.13 = 143.76$

$d_i = 135.50$

$t = 4.13$

$L = 164$ in.

$A = /4\ (143.76^2 - 135.50^2) = 1811.7\ in.^2$

$I = /64\ (143.76^4 - 135.50^4) = 4.42 \times 10^6$ in.

$J = 2 \times I = 8.84 \times 10^6\ in.^4$

$R_M = (135.5 + 4.13)/2. = 69.82$

Dry Weight = Vol. x .283 = A x L x .283 = 84.1 Kip

Fabricated Weight = 90.84 Kip From Dwg. 671J565

H_2O Weight - From Tech Manual TM-1440-C78

$$\begin{aligned} \text{Total Weight of } H_2O &= 68.3 \text{ Kip} \\ &- 24.8 \text{ Kip} - \text{Primary } H_2O \\ &- 23.1 \text{ Kip} - H_2O \text{ in Lower Shell} \\ &- \ \ 2.8 \text{ Kip} - H_2O \text{ in Transition Cone} \\ \hline &\ \ 17.6 \text{ Kip} - H_2O \text{ in Upper Shell} \end{aligned}$$

$I_R = M R_M^2 = 90.84/386 \times 69.82^2 = 1.15 \times 10^3$ Kip-sec^2-in.

Total Weight = 90.84 + 17.6 - 2.68* = 105.8 Kip

Upper Shell Lumped Mass Data

Node	Weight, Kip	I_R, Kip-sec^2-in.
9	10.6	115
10.11.12.13	21.2	230
14	10.6	115

*Secondary H_2O Added to Feedring

3. Elliptical Head

Weight = 29.52 Kip Head) Dwg. 671J565
4.5 Kip Steam Nozzle)

34.02 Kip

Mass = 34.020/386 = 88.1 lb-sec^2-in.

R_M = 69.78

$I_R = Mr^2/2 = 2.14 \times 10^5$ lb-sec^2-in. = 214 Kip-sec^2-in.

Upper Head Lumped Mass Data

Node	Weight, Kip	J_2, Kip-sec^2-in.^{2}
14	11.33	71.33
15	22.67	142.67

The changes in the model because of the CYW upper shell geometry are summarized in Table I-1.

II. Support Stiffness

The support stiffness used in the steam generator modal response spectrum analysis are derived from the reactor coolant loop model. Because of the difference in co-ordinate systems, the stiffness matrices had to be rotated for use with the CYW steam generator model. Rotating about the loop model Y axis 60° will align the global loop axis with the steam generator horizontal axes. Finally, rotating about the loop model x axis 90° will align the stiffness matrix co-ordinate system with the local steam generator model co-ordinate system. Table I-2 presents the development of the rotated stiffness matrices used in the steam generator detailed beam model.

Table I-1 (a)

Element Data Modeling Changes

Element	Existing Model (SCE)				CYW Model			
	A, in^2	I, $10^6 in^4$	J, $10^6 in^4$	Do, in	A, in^2	I, $10^6 in^4$	J, $10^6 in^4$	Do, in
7	1180.	2.12	4.24	119.	1180.	2.12	4.24	119.
8	1478.	3.19	6.38	124.3	1486.	2.76	5.53	125.8
9	1602.	2.94	5.88	133.2	1628.	3.63	7.27	137.4
10-15	1627.	3.64	7.28	137.5	1812.	4.42	8.84	143.8

Table I-1 (b)

Nodal Mass Point Changes

Node	Existing Model		CYW Model	
	Mass, $lb\text{-}sec^2/in$	Ir, $lb\text{-}sec^2\text{-}in$	Mass, $lb\text{-}sec^2/in$	Ir-$lb\text{-}sec^2\text{-}in$
7	55.	168.800.	56.	183.600.
8	34.	125.000.	37.	156,500.
9	39.	150.000.	46.	193,200.
10-13	44.	175.000.	55.	230.000.
14	39.	150.000	57.	186,330.
15	36.	76,000	59.	142.670.

LIST OF REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

2. Wichman, K. R., and Hopper, A. G., and Mershon, J. L., "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings," Welding Research Council Bulletin 107, March 1979.

3. Budynas, R. G., Advanced Strength and Applied Stress Analysis, McGraw-Hill Book Company, New York, 1977.

4. Wang. C.T., Applied Elasticity, McGraw-Hill Book Company, New York, 1953.

5. Bishop, J. E. and Smith, P. G., "Time History Seismic Analysis of the Steam Generators for the SCE San Onofre Power Plant," Westinghouse-Tampa, WTD-SM-75-021, April 1975.

SUMMARY OF SSE SEISMIC EVALUATION

OF REACTOR COOLANT PUMP MODEL SV-4M-A1

FOR CONNECTICUT YANKEE ATOMIC POWER PLANT UNIT 1

PRELIMINARY

Prepared by: Date:

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INTRODUCTION

The purpose of this report is to present the analytical methods and stress criteria used along with the results and conclusions obtained in SSE seismic evaluation of the Connecticut Yankee Reactor Coolant Pumps (RCP). This evaluation is required as part of the Connecticut Yankee Haddam Neck Plant Systematic Evaluation Program Seismic Reevaluation.

The SSE seismic event is considered a faulted condition. The stress criteria employed as stated in the Connecticut Yankee (CYW) Criteria Document, are obtained from Section III, ASME Boiler and Pressure Vessel Code, Appendix F. For non-pressure containing components of the Motor Stand the stress limits of Westinghouse Equipment Specification G677188 Rev. 4 are used. For the faulted condition the structural integrity of the RCP motor must be assured. The pump/motor must also remain mechanically functional to allow coastdown.

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DESCRIPTION OF COMPONENT

The Haddam Neck Plant Reactor Coolant Loops contain four Westinghouse Model SV-4M-A1 Controlled Leakage Seal Reactor Coolant Pumps. The model SV-4M-A1 RCP is a vertical, single-stage, centrifugal, shaft seal pump designed to pump large volumes of main coolant at high temperatures and pressures. The Connecticut Yankee RCP is designed to produce a differential head of 240 feet while pumping 61900 gpm of main coolant at a temperature of 544 F and a pressure of 2065 psia.

The pump is driven by a vertical, air-cooled, squirrel-cage induction type motor located above the pump on the motor support housing. The entire rotating assembly of the motor/pump is supported vertically by a double Kingsbury type oil lubricated thrust bearing. Lateral support is furnished by two oil bearings in the motor and one water bearing in the pump. See Figure 1 for relative locations of these components.

The entire pump/motor assembly is supported in the Reactor Coolant Loop by the attached loop piping and three spring hangers attached to feet cast integrally with the pump casing.

PRELIMINARY

CONCLUSIONS

The Connecticut Yankee RCP nozzles, casing, feet, motor stand, main flange bolts and shaft and seal housing components meet the faulted condition stress criteria of Appendix F of ASME Boiler and Pressure Vessel Code Section III, Equipment Specification G677188 Rev. 4 Appendix B and the Haddam Neck Plant Seismic Reevalution Program Criteria Document. Results are tabulated in Tables 1, 2 and 3. Structural integrity of the pump/motor is assured.

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ANALYSIS METHOD

Westinghouse Electro-Mechanical Division has performed a detailed time history dynamic analysis and structural evaluation of the Model SV-4M-A1 Reactor Coolant Pump (WEMD Reports #4672 and #4664). Figures 3 and 4 show the math model developed to represent the pump. This analysis was a specific analysis for the Southern California Edison Company San Onofre Plant (SCE) based on a time history Reactor Coolant Loop analysis for SSE with 4% damping. Loads and displacements at points of interest throughout the pump were determined. Stresses were calculated for critical components of the RCP assembly.

A comparison of CYW and SCE RCP technical information and drawings was performed. No major differences were found between the CYW and SCE pumps. For the purpose of seismic analysis these pumps are identical. In addition the method of pump support was compared. The CYW pump and the SCE analysis pump are supported in the same manner by the crossover leg piping at the suction nozzle, by the cold leg at the discharge nozzle and by three spring hangers attached to feet on the pump casing. No lateral supports exist elsewhere on the pumps.

A comparison of the CYW seismic response at the pump was made to the SCE seismic response at the pump. In order to compare the time history analysis to the response spectra analysis, response spectra at the pump were developed. From the SCE time history analysis the response spectra at the pump casing center was developed using the time history output tapes and a post processing computer routine. From the CYW response spectra analysis the response spectra at the pump casing center was developed from the frequencies, mode shapes, and participation factors obtained in a loop analysis using a computer code based on the methodology presented in recent technical papers by M. P. Singh (1975, 1980) utilizing random vibration theory.

Comparison of the two sets of response spectra, Figures 2 A,B,C, show that the SCE pump analysis is conservatively applicable to the CYW analysis for dynamic pump response and evaluation of pump components.

Structural evaluations of the pump casing, nozzles and support feet are performed using the maximum loads obtained in the CYW SSE 4% damping loop analysis at the suction and discharge nozzles and at the support feet. Analysis of the motor standard bolts, main flange bolts and shaft, and seal housing bolts are by direct comparison to the SCE pump analysis. For these components the stress limit allowables have been based on the difference between the faulted condition allowables of Section III ASME Boiler and Pressure Vessel Code, Appendix F and the normal condition allowables.

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TABLE 1

STRESS RESULTS

	PM			PL + PB		
DESCRIPTION	STRESS	ALLOWABLE	SURPLUS MARGIN	STRESS	ALLOWABLE	SURPLUS MARGIN
Suction Nozzle						
@ Casing Junction	8058	39840	4.94	11427	59760	5.23
@ Pipe Junction	8600	39840	4.63	8970	59760	6.66
Discharge Nozle						
@ Casing Junction	9751	39840	4.08	13685	59760	4.36
@ Pipe Junction	30710	39840	1.29	30830	59760	1.94
Support Foot						
@ Casing Junction	3787	23240	6.13	5564	34860	6.27
@ Hanger	26154	39840	1.10			
Motor Stand Shell						
@ Main Flange				44141	54000	1.22
@ Upper End				24519	54000	2.20
Motor Stand Bolts						
@ Main Flange	50377	73500	1.46			
@ Upper End	70862	73500	1.04			
Main Flange						
Bolts	6994	8800	1.26			
Shaft				515	32150	62.4
Seal Housing						
Shell				390	32150	82.4
Bolts	1065	73500	69.0			

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TABLE 2

RELATIVE DISPLACEMENT RESULTS

Location	Peak Displacements, in.	Radial Clearance, in.
Pump Guide Bearing	< .00026	.0105/.0085
Motor Lower Guide Bearing	< .0023	.004 Nominal
Motor Upper Guide Bearing	< .0027	.004/.006
Thrust Bearing	< .0085	.018 Nominal
Motor Upper Oil Pot-to-Rotor	< .014	.0625
Motor Core Centerline	< .017	.125
No. 3 Seal	< .110	.075/.055
Thermal Barrier Bottom Labyrinth	< .0214	.064/.060 + .125 Grooves
Impeller Bottom Labyrinth	< .0443	.030/.025 + .125 Grooves

The No. 3 seal ring could have contact with the shaft during the seismic event; however, the energy involved is negligible. Contact could also occur at the impeller labyrinth causing local deformation of the labyrinth teeth, thereby increasing the clearance. The energy involved is small and coastdown would occur.

TABLE 3

RCP MOTOR RESULTS

DESCRIPTION	CALCULATED VALUE	ALLOWABLE VALUE
Rotor Shaft Bending Stress	12,275 psi	52,500 psi
Rotor Shaft Center Deflection	.0378 in.	.125 in.
Vertical Loading on Rotor Core Assembly	34,104 #	60,088 #
Flywheel Bolt Stress	570 psi	20,500 psi

PRELIMINARY

The No. 3 seal ring could have contact with the shaft during the seismic event; however, the energy involved is negligible. Contact could also occur at the impeller labyrinth causing local deformation of the labyrinth teeth, thereby increasing the clearance. The energy involved is small and coastdown would occur.

PRELIMINARY

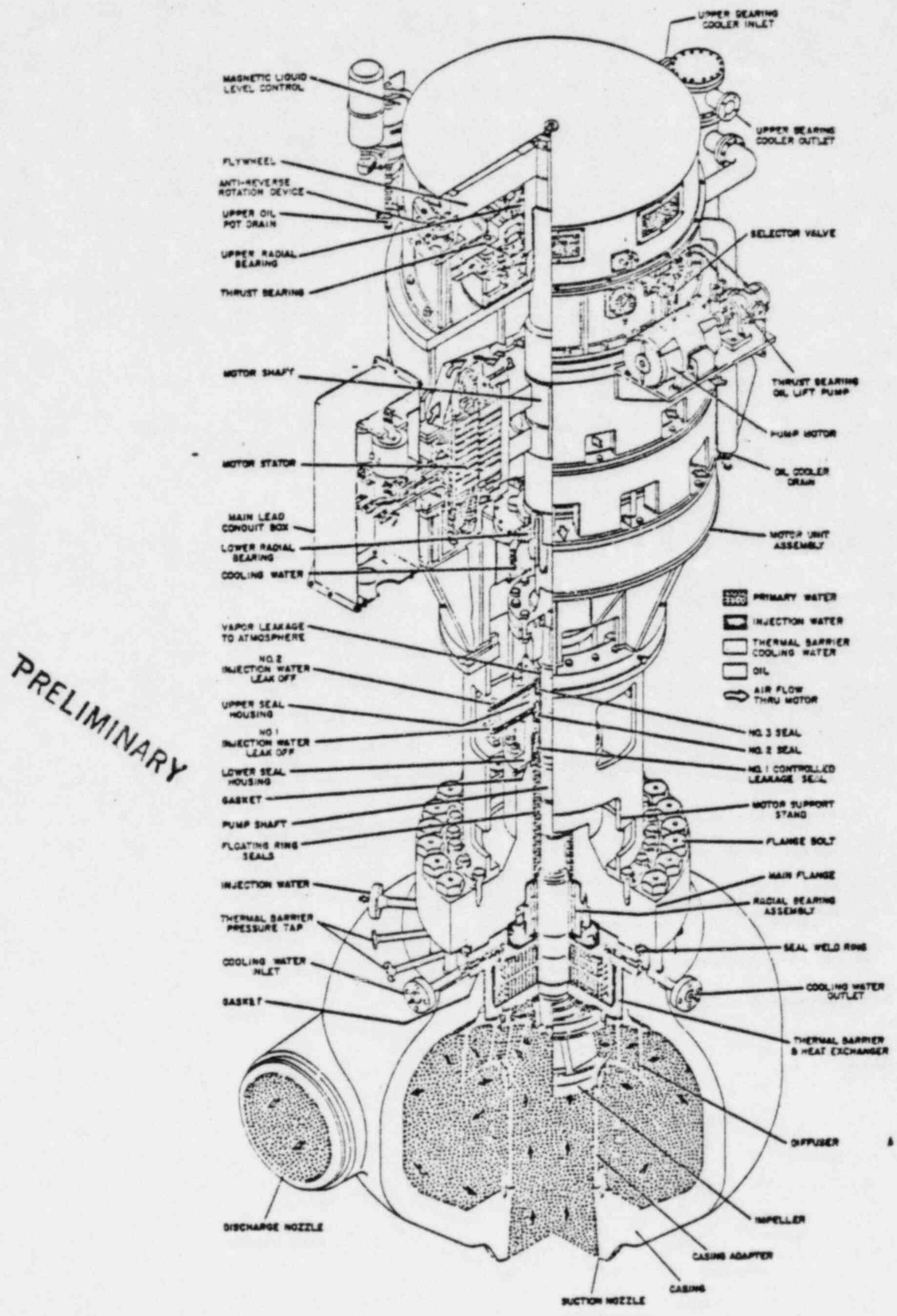


Figure 1: CYW Reactor Coolant Pump

G-100

PRELIMINARY

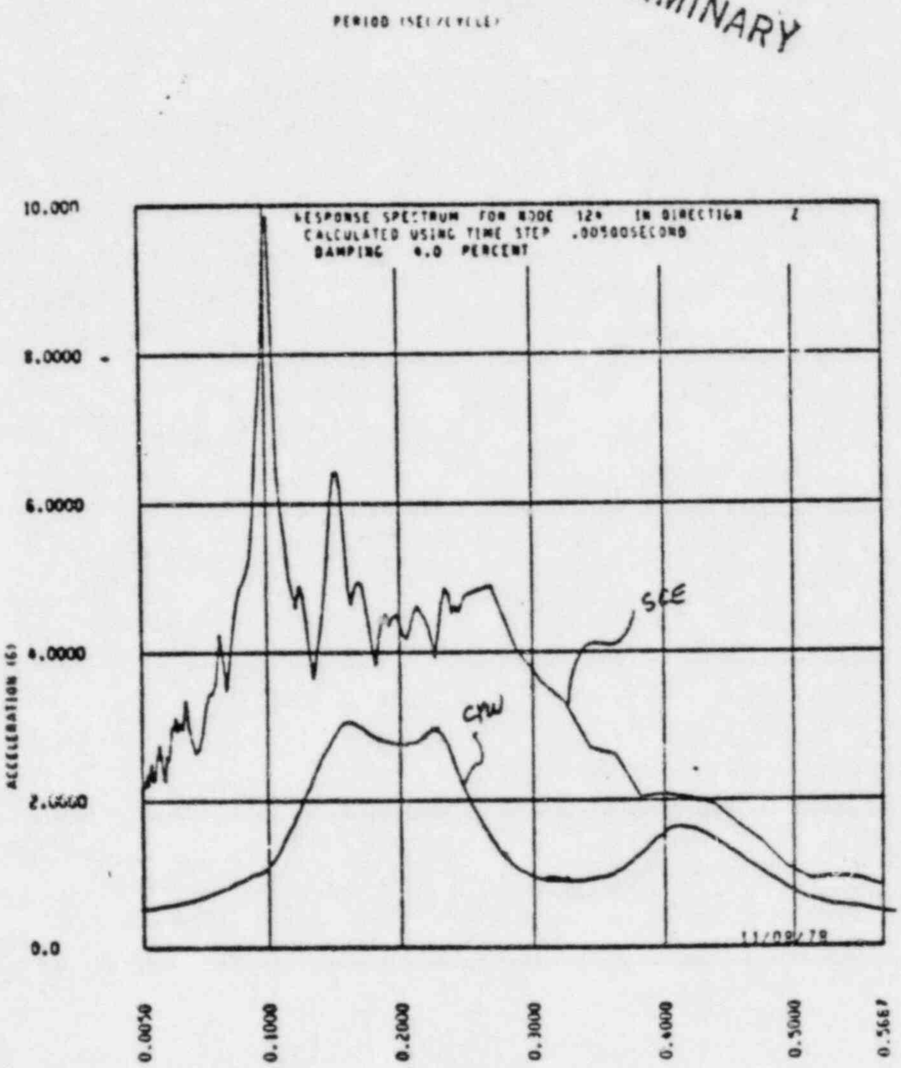


Figure 2A: Comparison of Horizontal Response @ RCP

PRELIMINARY

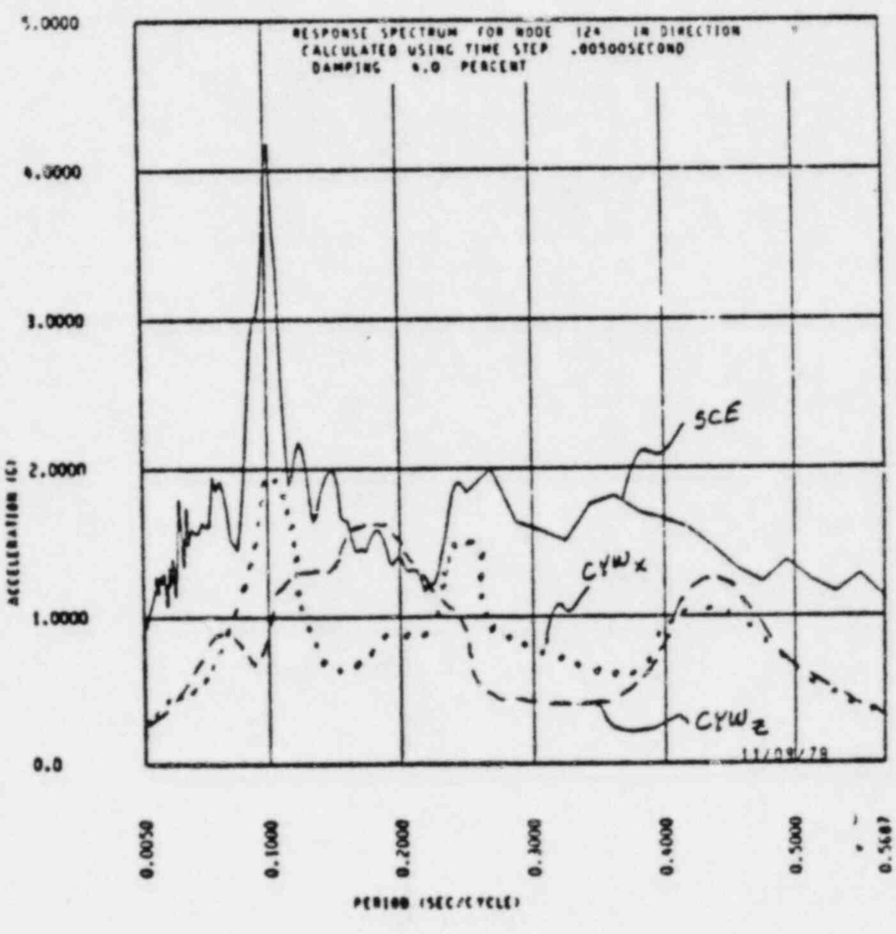


Figure 2B: Comparison of Horizontal Response @ RCP

PRELIMINARY

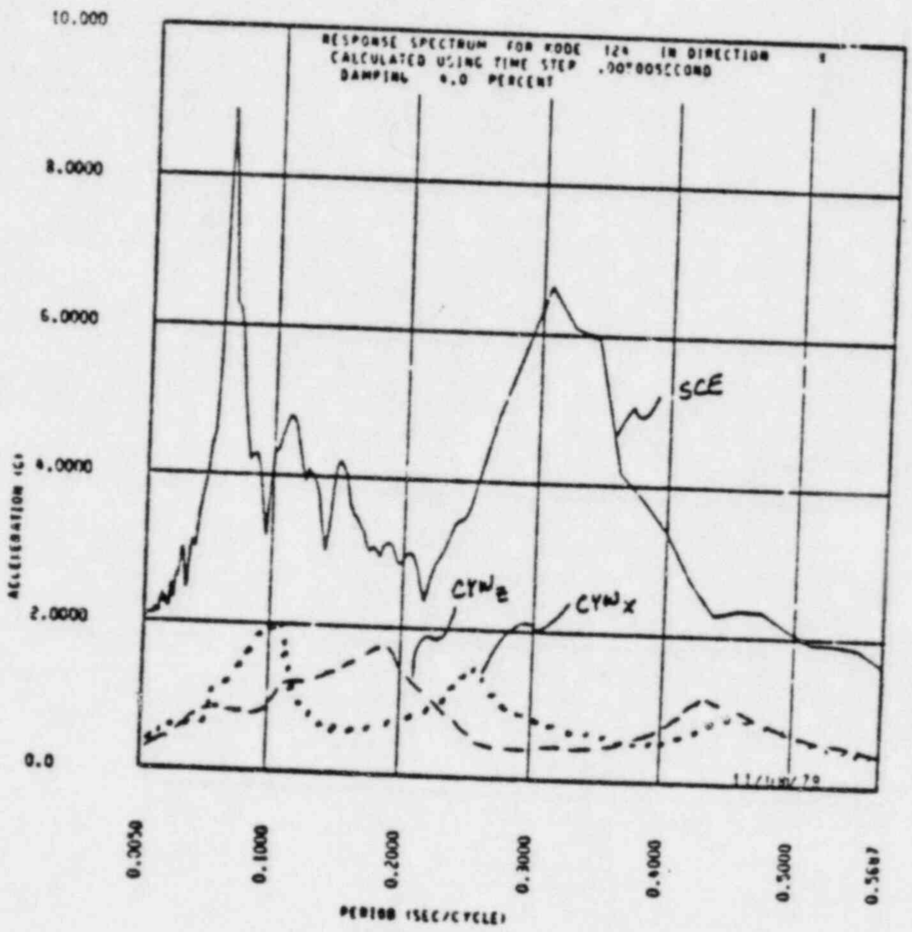


Figure 2C: Comparison of Vertical Response @ RCP

REFERENCES

1. Northeast Utilities Docket No. 50-213, 1/17/80, Haddam Neck Plant Seismic Reevaluation Program Criteria Document.

2. Structural Analysis of the Primary Reactor Coolant Loop System for the Haddam Neck Nuclear Power Station, preliminary May 1982.

3. Westinghouse Electro-Mechanical Division, Engineering Memorandum #4672, March 1975.

4. Westinghouse Electro-Mechanical Division, Engineering Memorandum #4664, March 1975.

5. Section III, ASME Boiler and Pressure Vessel Code, Winter 1972, Appendix F.

6. Westinghouse Equipment Specification G677188, Rev. 4, January 1976.

7. Westinghouse NTD-SEED file CYW-1000.

PRELIMINARY

APPENDIX H

ALLOWABLE STRESS CRITERIA AND DAMPING

Allowable Stress Criteria & Damping Values

Northeast Utilities Service Company - Haddam Neck
SSE Level EQ. 0.17g ZPGA

	Component	A-Active; P-Passive 1 Or 2	Allowable Stress	Damping Values - %	Reference
M-1	ESW Pump	A	$S_{all} \le .8\ S_y$	7	1,2,3
M-2	Diesel Exhaust Duct	P-1	$S_{all} \le S_y$	4	1,2,3
M-3	Diesel Air Start-Up Tanks	P-1	$S_{all} \le S_y$	7	1,2,3
M-4	CVCC Regenerative Heat Exchanger	P-1	$S_{all} \le S_y$	4	1,2,3
M-5	Diesel Generator	A	$S_{all} \le .8\ S_y$	7	1,2,3
M-6	Boric Acid Pump	A	$S_{all} \le .8\ S_y$	7	1,2,3
M-7	High Pressure Safety Injection Pump	A	$S_{all} \le .8\ S_y$	7	1,2,3
M-8	RHR Pump	P-2	$S_{all} \le .9\ S_y$	7	1,2,3
M-9	RHR Heat Exchanger	P-1	$S_{all} \le S_y$	4	1,2,3
M-10	Boric Acid Tank	P-1	$S_{all} \le S_y$	7 - Impulsive 0.5 - Sloshing	1,2,3
M-11	Demineralized Water Storage Tank	P-1	$S_{all} \le S_y$	7 - Impulsive 0.5 - Sloshing	1,2,3

Allowable Stress Criteria & Damping Values Continued

	Component	A-Active; P-Passive 1 Or 2	Allowable Stress	Damping Values - %	Reference
M-12	Refueling Water Storage Tank	P-1	$S_{all} \leq S_y$	7 - Impulsive 0.5 - Sloshing	1,2,3
M-13	Steam Driven Aux. Feedwater Pump	P-2	$S_{all} \leq .9S_y$		1,2,3
M-14	Underground 5,000 Gal. Oil Tank	P-1	$S_{all} \leq S_y$	4 - Impulsive 0.5 - Sloshing	1,2,3
M-15	Clean Diesel Oil Day Tank	P-1	$S_{all} \leq S_y$	7 - Impulsive 0.5 - Sloshing	1,2,3
M-16	Volume Control Tank	P-1	$S_{all} \leq S_y$	4 - Impulsive 0.5 - Sloshing	1,2,3
M-17	Containment Fan Coolers	P-2	$S_{all} \leq .9\ S_y$	7	1,2,3

Allowable Stress Criteria & Damping Values Continued

	Component	A-Active; P-Passive 1 Or 2	Allowable Stress	Damping Values - %	Reference
E-1	Battery Rack	P-1	$S_{all} \leq S_y$	7	1,2,3
E-2	MCC #1	P-1	$S_{all} \leq S_y$	7	1,2,3
E-3	Switch Gear (D.G. Room)	P-1	$S_{all} \leq S_y$	7	1,2,3
E-4	Control Panel (D.G. Room)	P-1	$S_{all} \leq S_y$	7	1,2,3
E-5	Engine Mounted Control Panel (on Diesel Gen)	P-1	$S_{all} \leq S_y$	7	1,2,3
E-6	4160-480 V Switchgear	P-1	$S_{all} \leq S_y$	7	1,2,3
E-7	Transformers (Switchgear Room)	P-1	$S_{all} \leq S_y$	7	1,2,3
E-8	MCC #5 & #6	P-1	$S_{all} \leq S_y$	7	1,2,3
E-9	Battery Charger	P-1	$S_{all} \leq S_y$	7	1,2,3
E-10	MCC #3	P-1	$S_{all} \leq S_y$	7	1,2,3
E-11	Main Control Board	P-1	$S_{all} \leq S_y$	7	1,2,3
E-12	Emergency Power Control Board	P-1	$S_{all} \leq S_y$	7	1,2,3
E-13	MCC #8	P-1	$S_{all} \leq S_y$	7	1,2,3

Allowable Stress Criteria & Damping Values Continued

Component	A-Active; P-Passive 1 Or 2	Allowable Stress	Damping Values - %	Reference
Component Support Structures (3)	P-1	$S_{all} \leq S_y$	As defined for supported component	1,2,3
Bolting				
-Embedded Bolting		$S_{all} \leq 0.7 S_u$		1,2,3
-Expansion Anchors		Ultimate Capacity /4		1,2,3
Welding		$S_{all} \leq 0.4 S_y$		AISC

Notes:

1) Allowable Buckling Load Equal to 2/3 Critical Buckling Load
2) Detailed Stress Analysis of component in accordance with ASME Section III Class 2 requirements.
3) Detailed Stress Analysis of component supports in accordance ASME Section III - NF and Appendex XVI.

References:

1) S & A Proposal
2) EDAC 175-130.01
3) USAEC Docket No. 50-213

APPENDIX I

TYPICAL BALANCE-OF-PLANT EQUIP ENT ANALYSES

82C152.1
0577A

SUMMARY OF HADDAM NECK SEISMIC ANALYSIS OF EQUIPMENT TO DATE

A. Mechanical Equipment

1. Auxiliary Feedwater Steam Driven Pump - Horizontal Pump

   O.K. for both leak tight and structural integrity and operation after E.Q.

2. Boric Acid Tank

   O.K. for leak tight and structural integrity.

3. Refueling Water Storage Tank

   Anchorage System Requires Modification.

B. Distribution System

1. 22 inch dia. Diesel Generator Exhaust Duct

   Duct O.K. supports of muffler may require strengthening.

C. Electrical Equipment

1. MCC-I

   O.K. for structural integrity and supports.

M-2 DIESEL GENERATOR EXHAUST DUCT

Summary of RESULTS

A. MAXIMUM DUCT STRESS

1. Maximum Stress in pipe = 9600 psi
2. MAXIMUM ALLOWABLE = S_y = 36,000 psi

F.S. = 36000/9600 = 3.75

B. FUNDAMENTAL FREQUENCY = 3.14 Hz

C. Supports Still Being Checked — Guided Lateral Restraint on Muffler will probably require reinforcing

M-10 BORIC ACID TANK

Summary of RESULTS

A. Shell Stress – 2845 psi compression F_s = 3.57
– 1783 psi shear
Bolts – 10472 psi tension F_s = 4.34
– 6809 psi shear

B FUNDAMENTAL FREQUENCY 0.38 Hz - sloshing
6.3 Hz - Vertical Frequency Of Tank top
13.6 Hz - 1st Impulsive Mode

C. DESIGN O.K.

M-12 Refueling Water Storage Tank

Summary of RESULTS

A. Shell Stress – 1760 psi Comp. F_s = 1.08
3169 psi Shear

Bolt Stresses - 515 Ksi F.S. = 40.6/515 = .08
40.6 Ksi all.

B. FREQUENCY 0.29 Hz - Sloshing
5.9 Hz - 1st. Impulsive

C. ANCHORAGE SYSTEM REQUIRES MODIFICATION

M-13 STEAM DRIVEN AUX. FEEDWATER PUMP

Summary of Results

A. Shaft Stress 13.98 Ksi Bending S.F. 2.3
BOLTS 4.4 Ksi TENSION S.F. 9.2
1.4 Ksi SHEAR

B. FREQUENCY 6.0 Hz Lateral Rocking
9.3 Hz FRONT to BACK - 1st MODE
11.4 Hz FRONT to BACK - 2nd MODE
13.6 Hz COUPLED HOUSING & SHAFT CANTILEVERED BENDING

C. Relative SHAFT DEFLECTION

0.000006 < 0.005 all. .. OK

D. DESIGN O.K. for both structural & Operational Integrity

E-1 MOTOR CONTROL CENTER MCC #1

Summary of Results

A (1) Maximum Tubing Stress 1126 psi All. 32,000 S.F. =
(2) Welded Connection Stress 989 psi All. 21,000 S.F. =
(3) Bolts shear 561 psi All. 16,780 S.F. =
(4) [illegible]

(5) Plate Buckling Comp. 24 psi All. 273 F.S.=

B. Frequency

19.7 Hz	1st	Bending	Anal.
20.8 Hz	2d	Bending	Anal.
17.2 Hz	1st		Test
23.6 Hz	2nd		Test

C. Design O.K. for both Cabinet Frame + Anchorage

SYSTEM Mechanical - #2

COMPONENT NAME Diesel Exhaust Duct COMPONENT №

LOCATION Diesel Generator Room ELEVATION From El. 35'-0" => El. 47'-2"

COMPONENT SAFETY FUNCTION: ACTIVE ☐ PASSIVE 1 ☒ 2 ☐

S-LIST PAGE № 125

METHOD OF ANALYSIS: 3 Dimensional Pipe Element Model. Supports are then analyzed by applying forces obtained from the computer run.

SPECTRAL CURVES USED: Response Spectra of Roof of the Diesel Generator Building

DAMPING VALUE ASSUMED: 4%

ACCEPTANCE BEHAVIOR CRITERIA USED: $S_{all} \leq S_{yield}$, Equation 9 of Subsection NC-3652 of ASME Section III, Division I, ASME Appendix XVII & Appendix A

COMPUTER CODE USED: FINITE ELEMENT MULTI-PURPOSE PROGRAM DYNAFLEX

REMARKS: Transition piece from diesel exhaust nozzle (rectangular) to duct (circular), will be analyzed separately using a multi-purpose finite element analysis program.

S&A	CALCULATION COVER SHEET	NORTHEAST UTILITIES - HADDAM NECK	REV.№	0		
			BY	PNP		
			DATE	4-1-82		
			CHK'D	FAT		
			DATE	4/15/82		
			APPR.			
			DATE			

1. INTRODUCTION

The Diesel Exhaust Duct is a 22" diameter pipe, fabricated from plate, that runs from the top of the Diesel Generator, through an expansion joint and a muffler, and then out to atmosphere through the roof of the diesel generator building.

A total of six supports exist on the line. Three are adjustable roll rod type supports which provide vertical restraint in one direction. Another vertical support, fabricated from plate, is located on the roof. The other two supports are found on the muffler, one of which is an anchor. The anchor consists of plates welded to the outside of the muffler and to a WT section, which fits between the plates. The WT section is then bolted and welded to an existing beam. The other muffler support is similar to the anchor with the exception that a horizontal pin is placed in a slotted hole through the plates and the WT section. Thus, this support as well as the anchor provide both vertical and lateral restraint. Details of the supports are indicated in Figures 1-1 through 1-5.

An isometric drawing is constructed which models the duct and its supports and is illustrated in Figure 1- 6. The analysis was made using Dynaflex, a piping program with static and dynamic capabilities applicable to the Nuclear Industry, so as to determine the capability of the duct system to withstand a defined seismic event.

2. SUMMARY OF RESULTS

The analysis of the duct indicates a maximum primary stress of 9,600 psi, which occurs at an elbow, node point 11n, for the load combination of dead weight and seismic inertial forces. This is acceptable when compared to the allowable stress limit of 36,000 psi, which is defined in Section 4 of this report.

The piping model was constructed with the masses automatically lumped at the specified node points and at maximum intervals of 8'-0" by using the computer aided option of mass lumping. This assures there is at least one lumped mass between support points. The fundamental frequency of the duct system is equal to 3.14 hz. The second mode is found at 5.08 hz and the third through the fifth modes are 11.84, 14.87 and 30.70, respectively.

Two computer runs were necessary for the duct system. For a dead weight analysis the program assumes the weight acts in a Y direction. However, the pipe at node point 8 actually lifted off the support for an earthquake because the inertial force was greater than the dead weight force. Thus, the actual loads on the adjacent supports did not experience this increase in load and another run was made. For this case the support at node point 8 was removed. The loads were then redistributed during the earthquake with the adjacent vertical supports reflecting this change with no other line segments lifting off the supports.

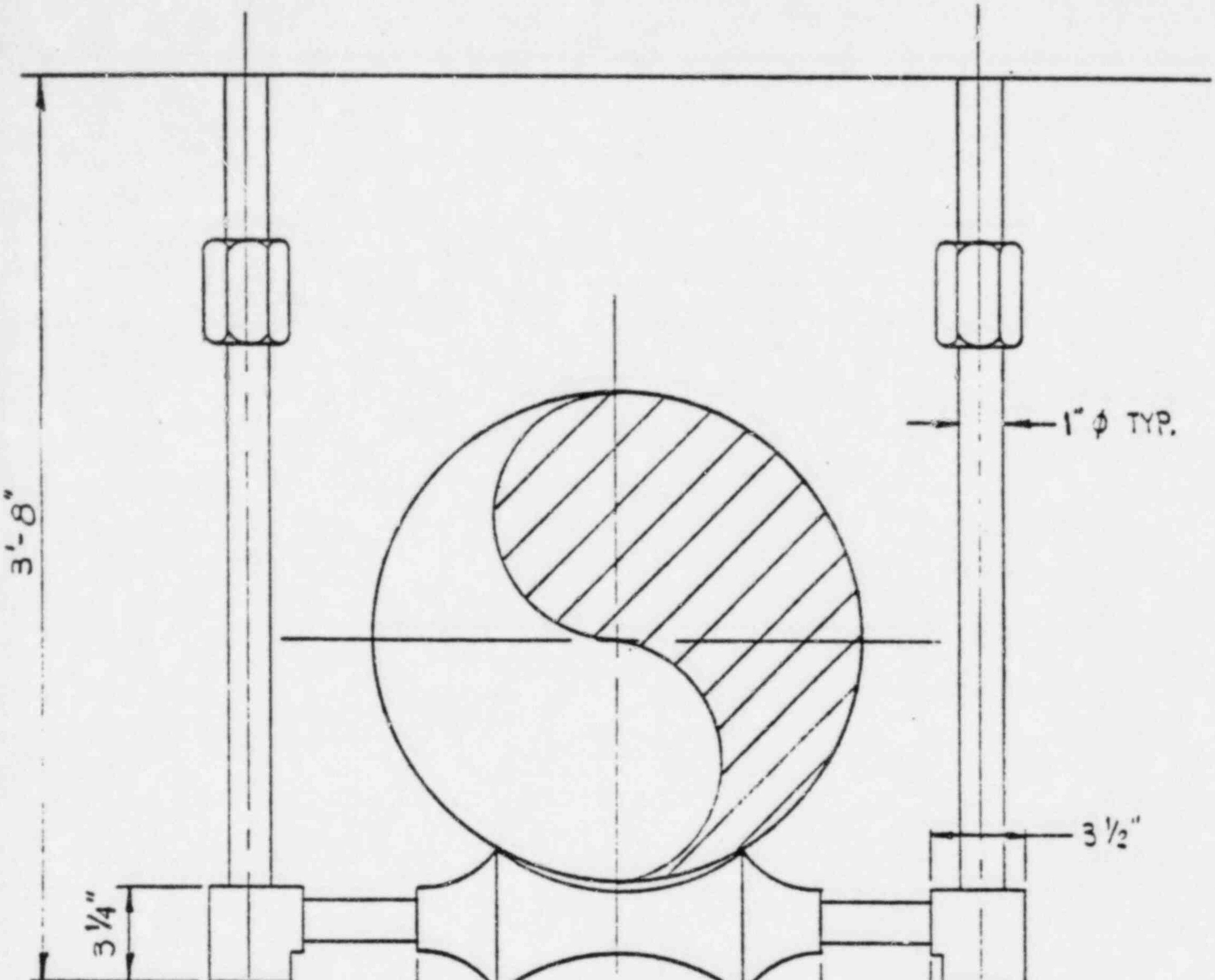


FIG 1-1 ADJUSTABLE ROLL ROD-TYPE SUPPORT

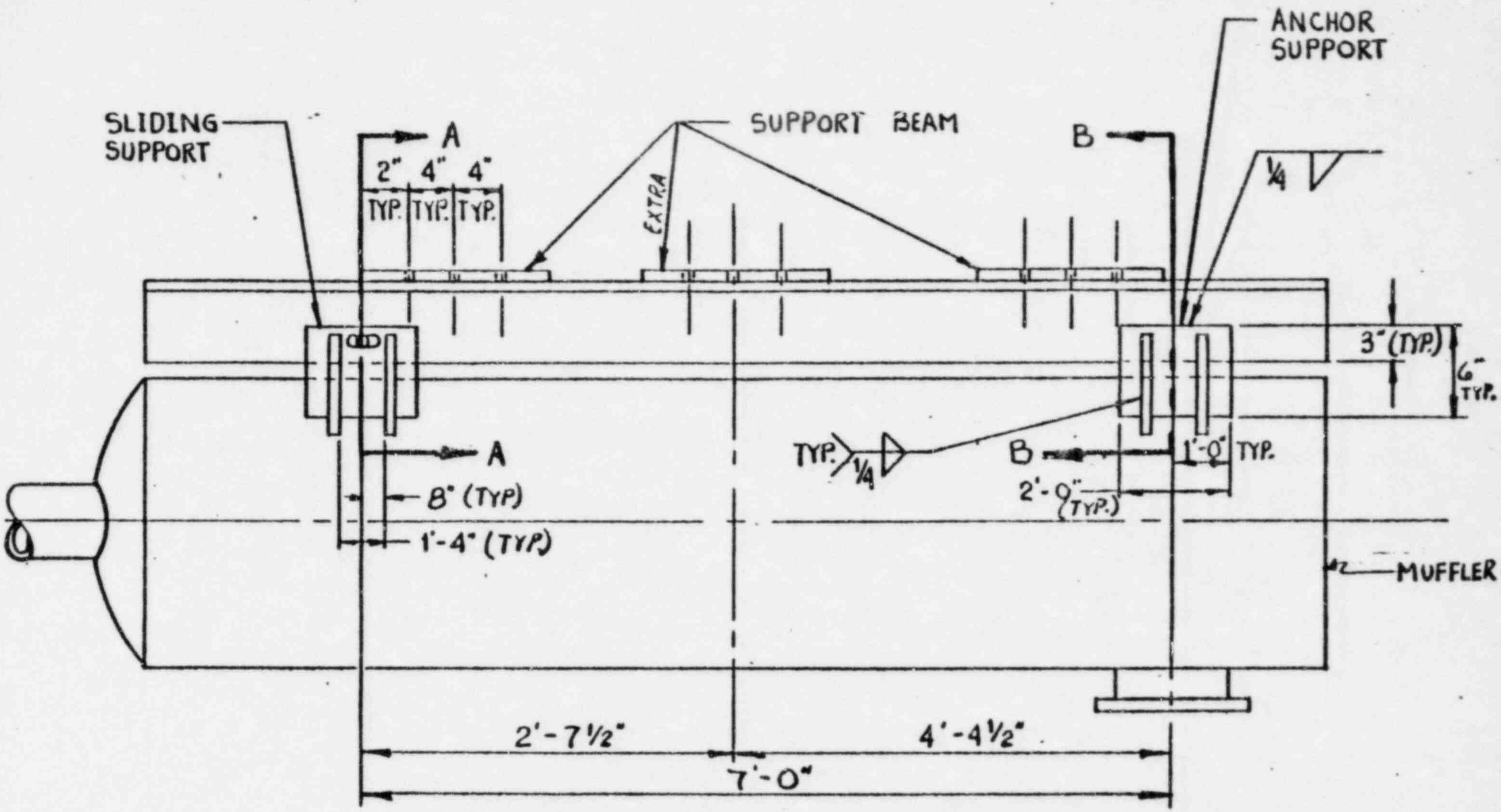


FIG 1-2 PLAN OF MUFFLER AND SUPPORTS

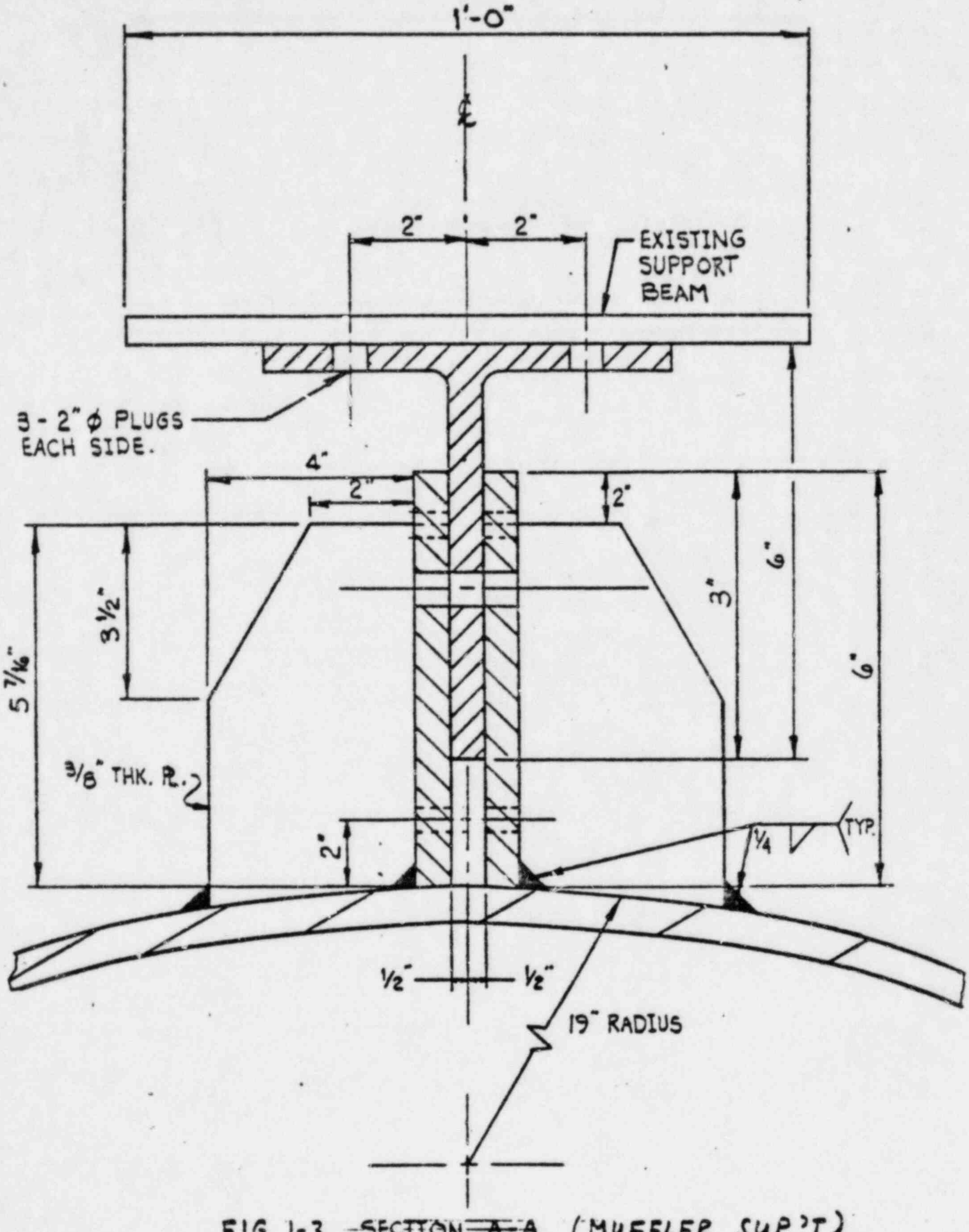


FIG 1-3 ~~SECTION A-A~~ (MUFFLER SUP'T)

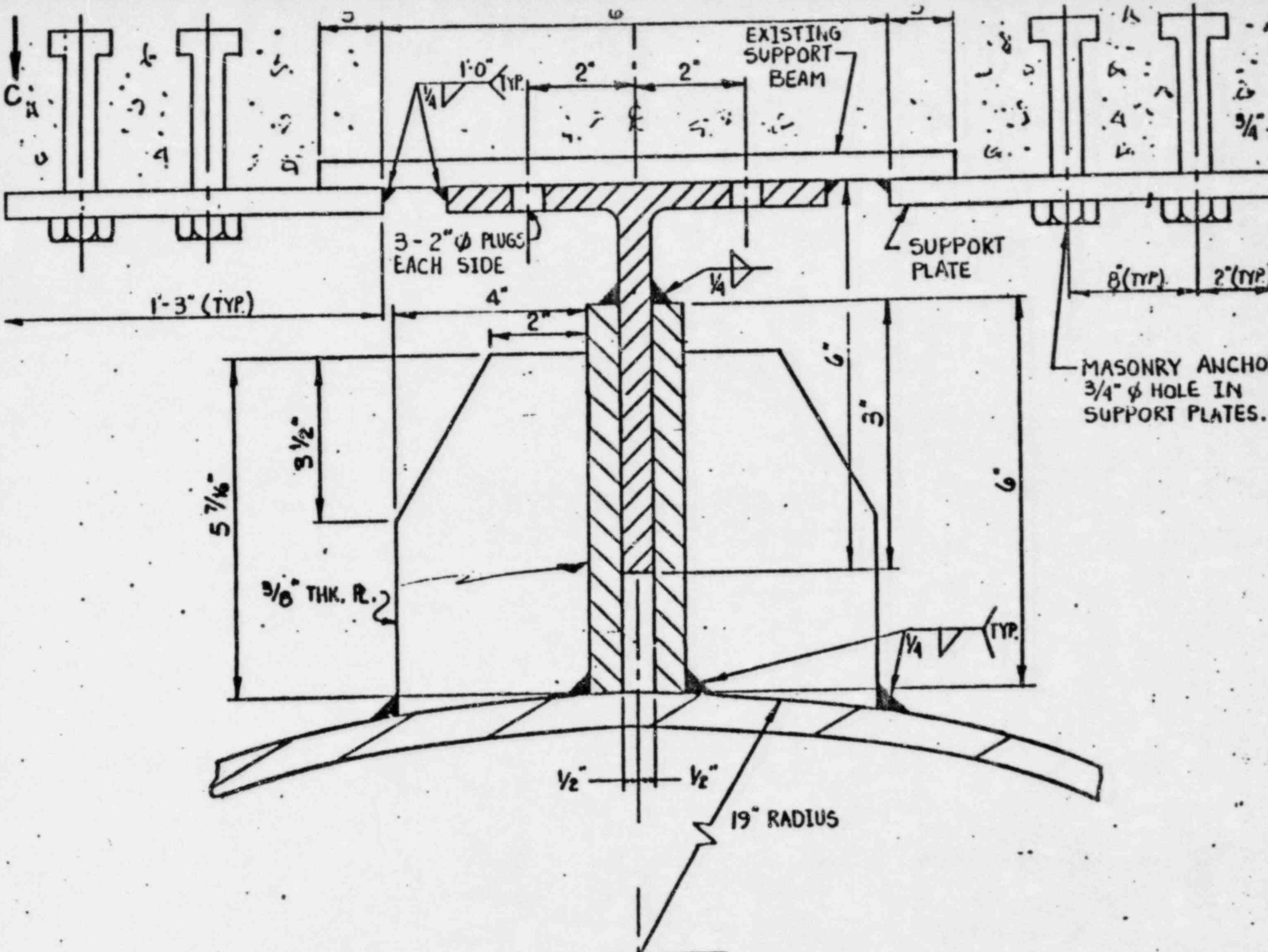


FIG 1-4 SECTION B-B (Muffler Sup't)

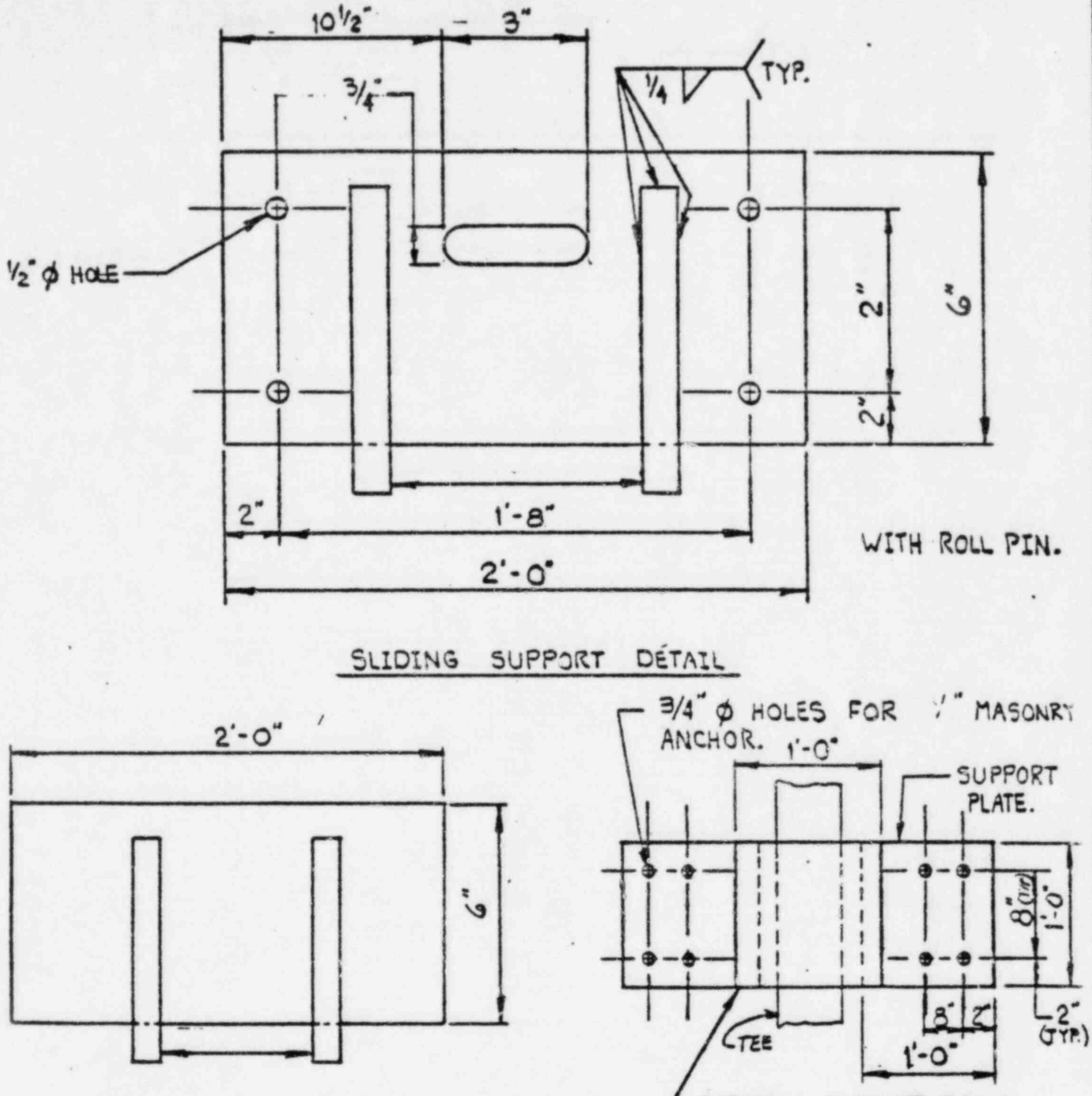


FIG 1-5 (MUFFLER ASSEMBLY)

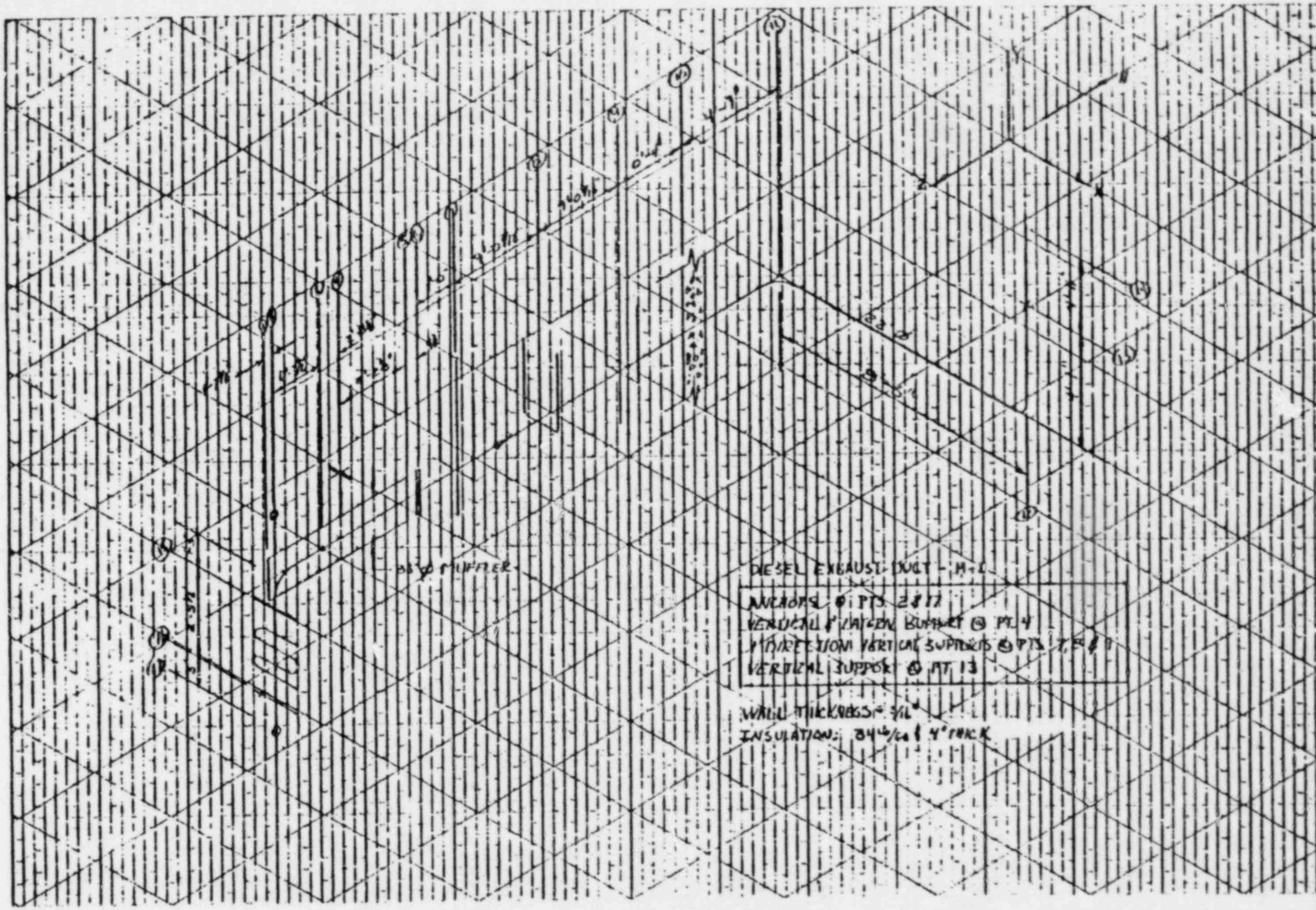


FIG. 1-6 ISOMETRIC OF DIESEL EXHAUST SYSTEM

Several assumptions have been made to facilitate the development and implementation of the model for the analysis:

- The transition piece from the diesel generator to the duct will be included in the analysis of the diesel generator.

- The wall thickness of the duct is 3/16", which is the same thickness as indicated for the transition piece.

- The insulation thickness is 4" and weighs 34 lb/ft.

- A Pathway expansion bellows with 10 convulutions and a 50 PSIG working pressure has been assumed in the analysis. The stiffness properties are 430 lb/in - axial, 5300 lb/in - lateral, and 500 in-lb/deg - rotational.

- The connection of the exhaust duct to the muffler was modelled as an unreinforced fabricated tee.

Preliminary analysis of the support system indicate the web of the non-anchor muffler support is overstressed and many require stiffening.

3. LOAD CRITERIA

The piping system and its supports are identified as passive, P-1 components. No secondary stresses resulting from either live loads or the defined earthquake are considered. In addition, no applicable live load in the steady state or transient operation exists for this system. Thus, the load combination considered in the seismic design adequacy is;

$$U = D + E$$

where:

U = Load capacity of the component.

D = Dead load resulting from the pipe weight and insulation weight.

E = Load from the defined seismic event (as defined in Figures 3-1, and 3-2) which is representative of the roof of the diesel generator building for 4% damping.

4. STRESS DEFORMATION - STABILITY CRITERIA

The duct system, including its supports, is analyzed for the effect of the loads resulting from the earthquake. For service conditions the combined dead weight and seismic inertial loads must satisfy Equation 9 of Subsection NC-3652 for Class 2 pipe of ASME Section III, Division I except that allowable stress is reduced from 3S to 1.5 S or S_y

The allowable stress limit for the duct system is defined as

$$S_{all} \leq S_y;$$

which was determined from the "Allowable Stress Criteria for the Haddam Neck Plant" attached hereto as Appendix A.

5. METHODS OF ANALYSIS

Dynaflex was the computer program utilized in the analysis of the Diesel Exhaust Duct System. A static analysis and a dynamic analysis using the response spectrum option were performed. The resulting loads from the dead weight and the earthquake were then combined by absolute summation and applied to the supports to determine stresses. Stresses in the pipe at the node points were calculated directly by the program.

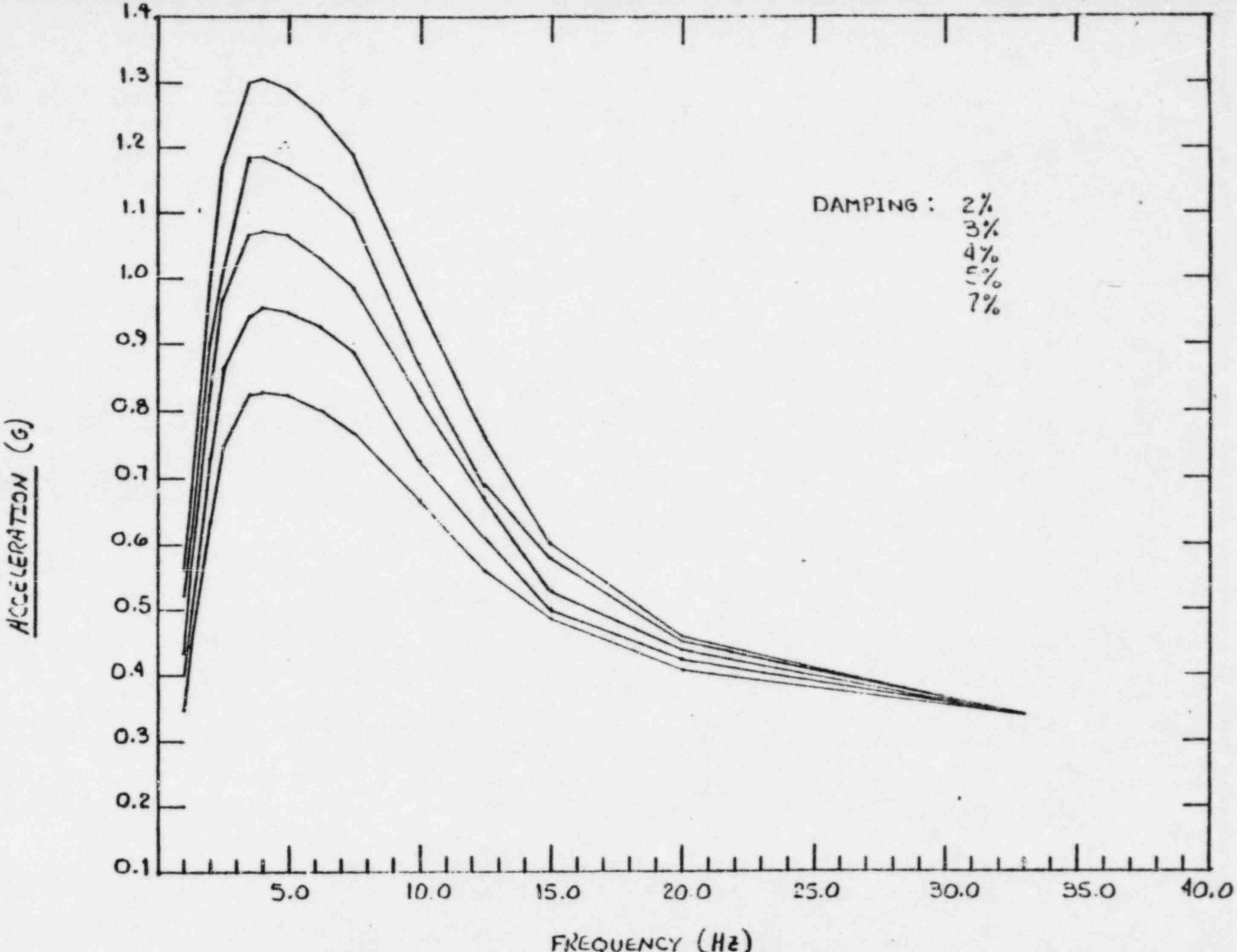


FIG 3-1 HORIZONTAL RESPONSE SPECTRUM AT DIESEL GENERATOR BLDG. AT ROOF.
EQUAL TO 2x GROUND RESPONSE AT ROCK SURFACE.

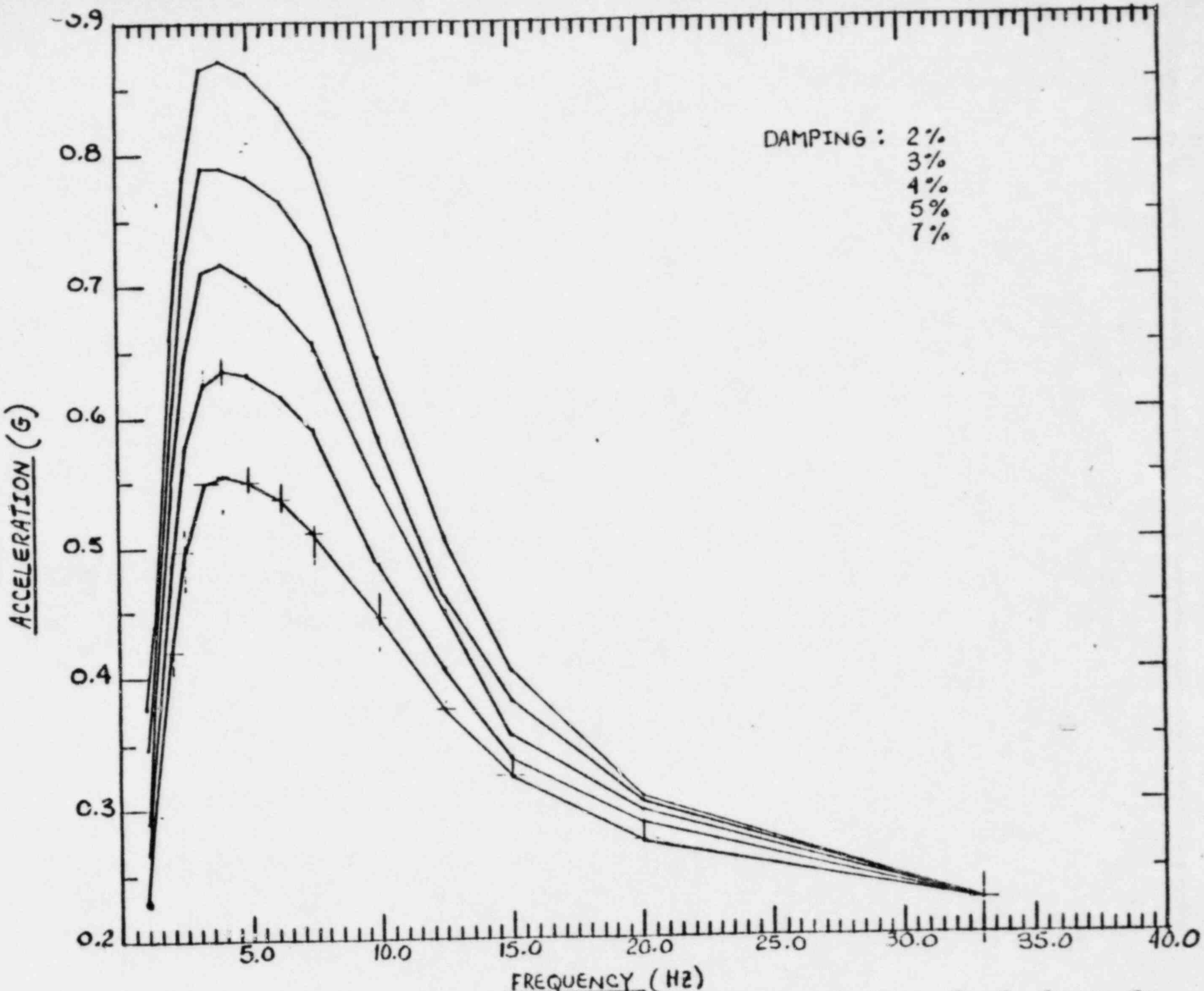


FIG 3-2 VERTICAL RESPONSE SPECTRUM AT DIESEL GENERATOR BLDG. ROOF. EQUAL TO 2x GROUND RESPONSE AT ROCK SURFACE.

SYSTEM Mechanical - #10

COMPONENT NAME Boric Acid Tank COMPONENT № TK- 2-1A

LOCATION Prim. Aux. Bldg. ELEVATION 35'-6"

COMPONENT SAFETY FUNCTION: ACTIVE ☐ PASSIVE 1 ☒ 2 ☐

S-LIST PAGE № 46

METHOD OF ANALYSIS: One Dimensional Finite Element (Beam) Multidegree of Freedom Dynamic Model. Sloshing effect done by TID - 7024

SPECTRAL CURVES USED: FRS of PAB ; Figures 19, 20, & 25

DAMPING VALUE ASSUMED: 2% (Impulsive) , 0.5% (Sloshing)

ACCEPTANCE BEHAVIOR CRITERIA USED:

S_{all} & S_{yield}

COMPUTER CODE USED: FINITE ELEMENT MULTI-PURPOSE , SAP IV

REMARKS: REFERENCES

TID - 7024

"Aboveground Vertical Tanks," R.P. Kennedy, Oct. 1981

"Basis of Seismic Design Provisions for Welded Steel Oil Storage Tanks," Wozniak & Mitchell

Reg. Guide 1.60

Roark , "FORMULAS FOR STRESS & STRAIN" 3rd Edition

S&A	CALCULATION COVER SHEET	NORTHEAST UTILITIES - HADDAM NECK	REV. №	0		
			BY	PNP		
			DATE	4-1-82		
			CHK'D	FAT		
			DATE	4/15/82		
			APPR.			
			DATE			

1. INTRODUCTION

The Boric Acid Storage Tank is located at floor elevation 35'-6" in the Primary Auxiliary Building as shown in Figure 1. The tank is 17 feet in diameter and has a hemispherical bottom. The tank roof is flat and made out of lap welded stainless steel plate and structural framing which support an electric motor and mixer. Support for the tank is provided by a stiffened ring at about mid-height on the shell. This ring is bolted to the floor by 8-2 inch diameter anchor bolts. The tank shell and roof plates are 3/16" thick stainless plate and the hemispherical head is 3/8" stainless steel plate.

These calculations were made to evaluate the seismic design adequacy of the Boric Acid Storage Tank for the seismic floor response spectra input defined in Figures 2, 3 and 4.

2. RESULTS

The results of this analysis show that the Boric Acid Storage Tank will maintain its structural and leak tight integrity during the prescribed seismic disturbance defined by Figures 2, 3 and 4. Satisfactory performance for this component is defined, in the P-1 or passive one category of "Allowable Stress Criteria & Damping Values for NUSCO-Haddam Neck", as shown in Appendix A to this report.

Three load cases were investigated.

1. Tank full; no sloshing of fluid considered, entire mass acts impulsively.

2. Fluid height at base of support ring with all fluid and tank motion assumed impulsive.

3. Fluid height at base of support ring with all fluid assumed sloshing. Fluid and empty tank response are combined by SRSS.

The calculated natural frequencies for this tank are the following:

6.262	Hz	Motor bouncing on tank roof. All load cases.
13.02	Hz	1st lateral shell mode, load case 1.
13.60	Hz	1st lateral shell mode, load case 2.
26.29	Hz	2nd mode of motor bouncing on roof, all load cases.
33.95	Hz	1st lateral shell mode, load case 3.
.3779	Hz	1st mode sloshing in half-full tank, load case 3.

For each load case stresses were computed in the shell and anchor bolts. Factors of safety were computed for combined bending and shear in the shell and combined shear and tension. These are shown in Table 1.

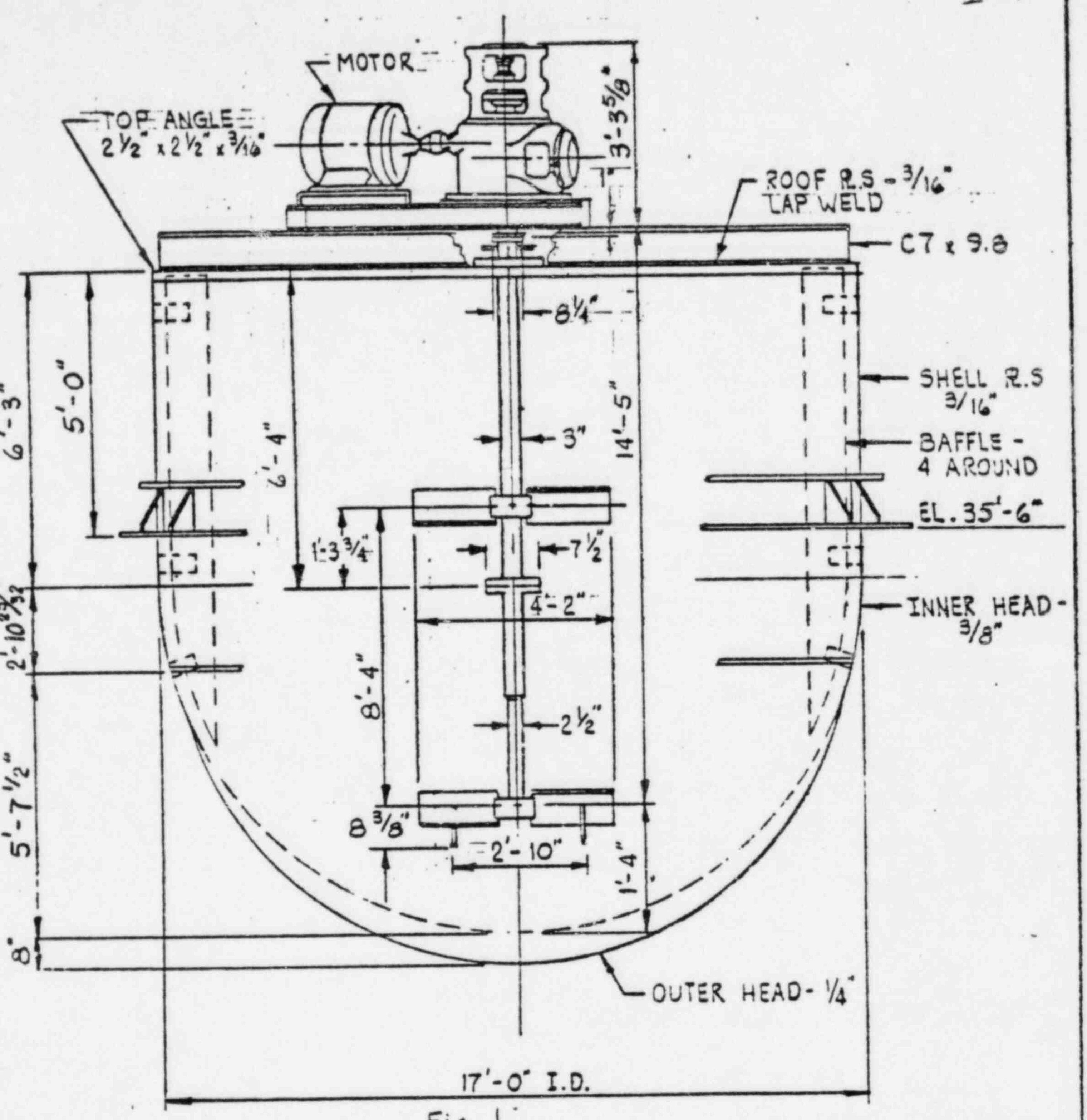


Fig. 1

MATERIAL :

SHELL + HEADS - A 240 TYPE 304 SS
SUPPORT RING - A 283 C.
STEAM JACKET - A 285 C.
STRUCTURAL - A 36

MOTOR :

10 H.P. , 1800/900 RPM
WEIGHT - 265 lbs.
MIXER WT. (LESS MOTOR) - 1778 lbs.
IMPELLER RPM - 56/28
4-BLADES

ALL MIXER PARTS IN CONTACT WITH TANK CONTENTS ARE 316 SS.

WT. FULL = 244267 lbs.
WT. EMPTY = 17307 lbs.

Table 1

	Actual Stress				Allowable Stress[1]				Factors of Safety	
	Shell		Bolts		Shell		Bolts		Shell	Bolts
Load Case	Shear	Bending	Shear	Tension	Shear	Bending	Shear	Tension	Combined Bending & Shear	Combined Shear & Tension
1	1867	1162	4467	6822	7528	12924	16780	40600	6.6	10.08
2	2845	1783	6809	10472	7528	12924	16780	40600	3.57	4.34
3	843	555	2017	3256	7528	12924	16780	40600	17.96	47.7

All stresses in pounds/in^2

[1] Note: Allowable stresses are determined by buckling calculations.

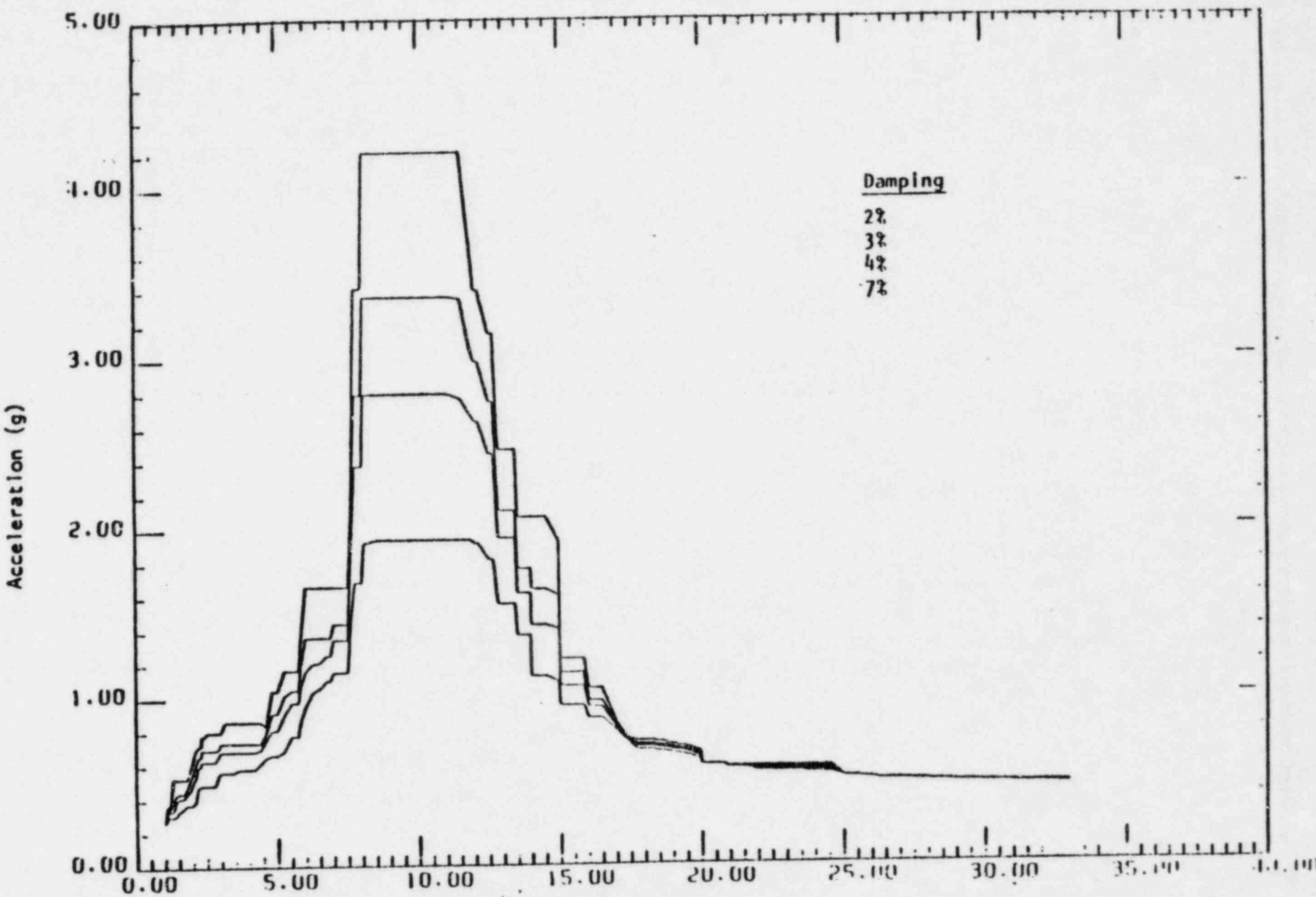


Figure 2: North-South Floor Response Spectra For Elevation 35'-6", Primary Auxiliary Building, Connecticut Yankee Atomic Power Plant

12-I

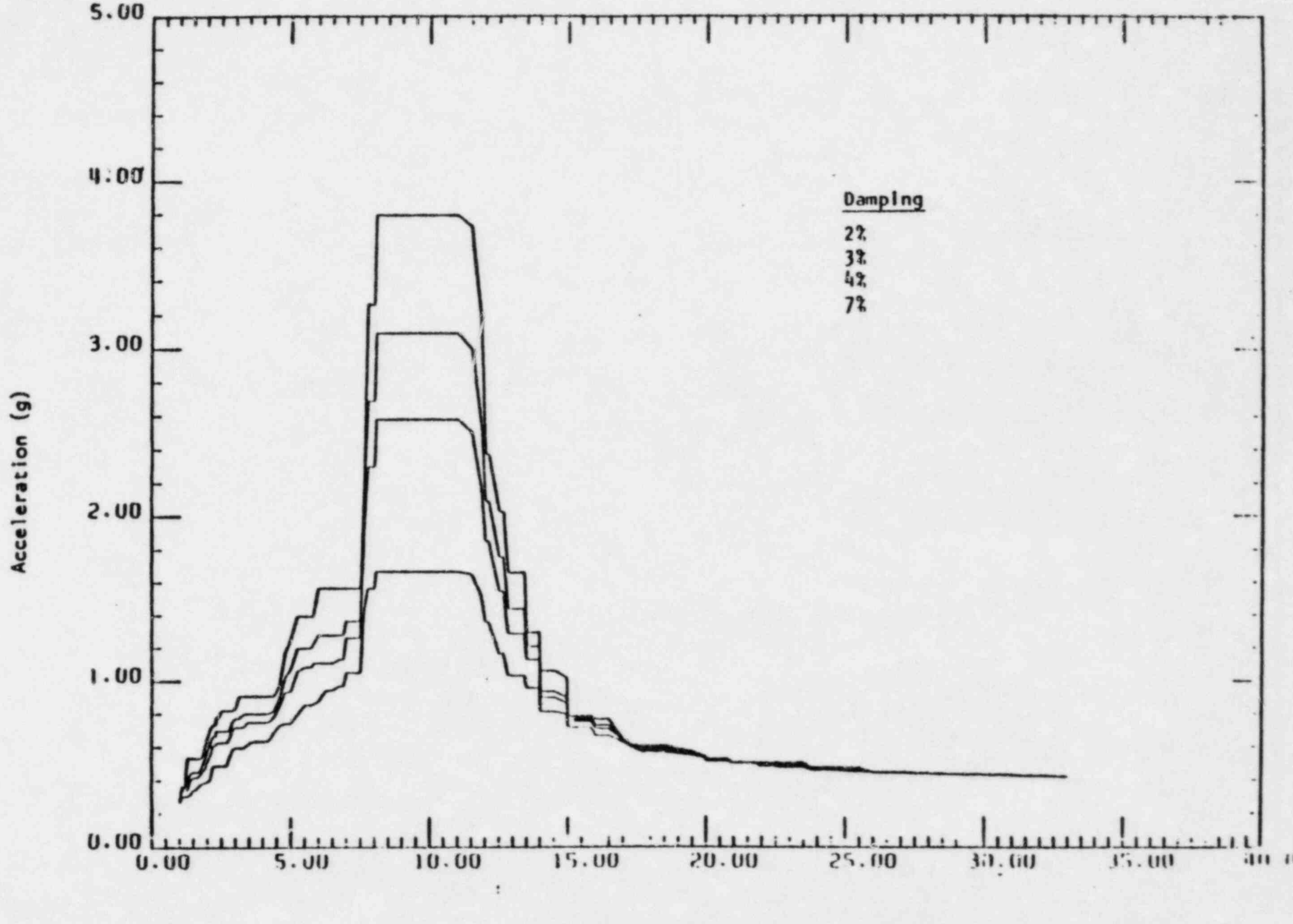


Figure 3: East-West Floor Response Spectra For Elevation 35'-6", Primary A—iliary Building, Connecticut Yankee Atomic Power —nt

22-I

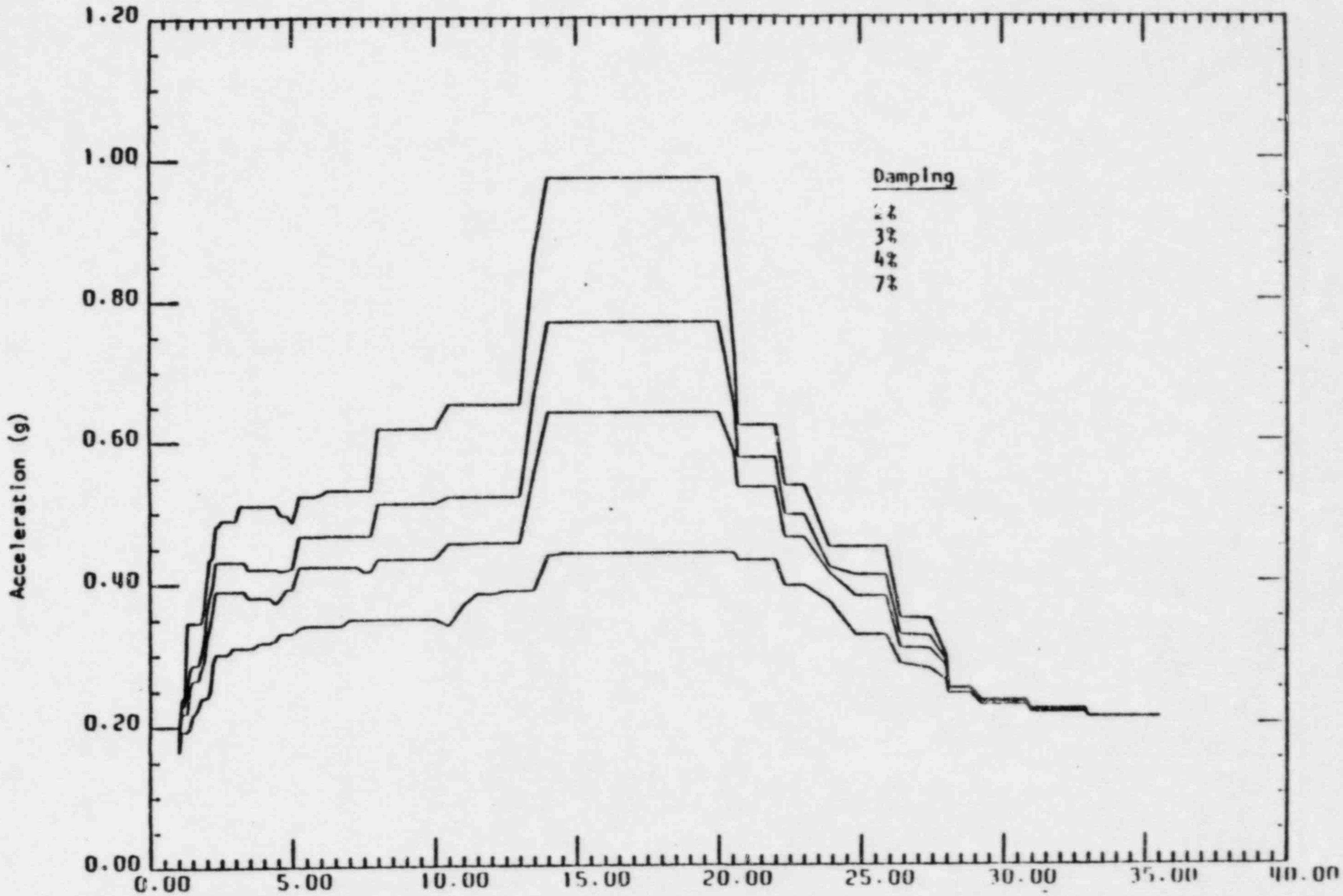


Figure 4 : Vertical Floor Response Spectra For Elevation 35'-6"
At Location 12'-6" West of Column Line J and 16'-9" South of Column Line 13½
Primary Auxiliary Bui 'ng, Connecticut Yankee Atomic Power Plant

Safety factors for shear and bending in the shell plates were computed from the interaction equation:

$$F.S. = \frac{1}{\left(\frac{f_b}{F_b}\right) + \left(\frac{f_t}{F_t}\right)^2}$$

Safety factors for bolting followed from:

$$F.S. = \frac{1}{\left(\frac{f_v}{F_v}\right)^2 + \left(\frac{f_t}{F_t}\right)^2}$$

The factor of safety for lateral-torsional buckling of the main framing members on the roof is

$$F.S. = \frac{F_b}{f_b} = \frac{22.66}{11.777} = 1.924$$

The maximum dynamic deflection of the roof is $\pm$ .09 inches.

3. LOAD CRITERIA AND FAILURE MODE ASSUMPTIONS

Because of the lack of available research information on tanks with hemispherical bottoms, the three load previously mentioned were considered.

Case 1. Tank full; no sloshing of fluid considered, entire mass acts impulsively.

This case was investigated because an overflow nozzle is located at 8" below the roof and the slosh height of the fluid is 21.8", which is greater than 2 times the freeboard. For this case the fluid must be treated impulsively.

Case 2. Fluid height at base of support ring with all fluid motion assumed impulsive, along with tank.

Since the fluid could be drained down at least to the base of the ring, this case was investigated because all fluid mass was on one side of the support ring. The fluid was treated as being impulsive because the exact motion of the fluid is not easily determined.

Case 3. Fluid height at base of support ring with all fluid assumed sloshing. Tank motion is impulsive and fluid and empty tank response are combined by SRSS.

In this case, the empty tank response was computed by machine and combined by hand with the sloshing fluid response by SRSS. The entire fluid mass is assumed to slosh because the exact motion of the fluid is not easily determined as stated in Case 2 also. It should be noted that the true response of the fluid is a combination of Cases 2 and 3 which at this time is difficult to determine.

No nozzle loads were considered on the Boric Acid Storage Tank. Nozzles less than 4" diameter do not produce loads of significant magnitude and no nozzle greater than 3" diameter enters the tank.

The response spectra used were provided by Northeast Utilities. The curves for 7% damping were used for all directions of input spectra associated with impulsive response. One-half percent damping was used for sloshing response mode. This amount of damping had negligible effect on forces produced from sloshing of the boric acid. Slight smoothing of the response spectra was done to reduce computer input time.

4. STRESS CRITERIA

The stress criteria are as shown in Appendix A with these exceptions:

Allowable stress in bending for the tank shell is governed by buckling criteria, S_{all} = 12.9 ksi.

Allowable stress in shear for the tank shell is governed by buckling criteria, S_{all} = 7.528 ksi.

5. METHODS OF ANALYSIS

The SAP IV computer program was used for the analysis of the impulsive motion of tank and fluid and the sloshing motion was computed and combined with the computer analysis by hand. The boundary conditions for the computer model were input as being rigid.

The tank was modelled as a cylindrical beam stick with fluid mass lumped at points along the length. A drawing of the computer stick model is shown in Figure 5 along with the values for the section properties and lumped masses. The computation of these values is described in Section 6.

SYSTEM: Mechanical - #12

COMPONENT NAME: Refuel Wtr Stor Tk. COMPONENT №: Tk-4-1A

LOCATION: Outside ELEVATION: 23'-0 5/16"

COMPONENT SAFETY FUNCTION: ACTIVE ☐ PASSIVE 1 ☒ 2 ☐

S-LIST PAGE №: 46

METHOD OF ANALYSIS: One Dimensional Finite Element (Beam) Multidegree of Freedom Dynamic Model. Sloshing effect done by TID-7024.

Beam model w/ soil springs

SPECTRAL CURVES USED: RESPONSE AT SOIL SURFACE

DAMPING VALUE ASSUMED: 7% (Impulsive), 0.5% (Sloshing)

ACCEPTANCE BEHAVIOR CRITERIA USED:

Sall & Syield

COMPUTER CODE USED: MULTI-PURPOSE FINITE ELEMENT

REMARKS:

Ref: TID-7024

"Aboveground Vertical Tanks," R.P. Kennedy, Oct. 1981

"Basis of Seismic Design Provisions for Welded Steel Oil Storage Tanks," Wozniak & Mitchell

Reg. Guide 1.60

ROARK, "FORMULAS FOR STRESS AND STRAIN" 5th Ed.

S&A

CALCULATION COVER SHEET

NORTHEAST UTILITIES - HADDAM NECK

REV. №	0		
BY	PNP		
DATE	4-1-82		
CHK'D	FA		
DATE	4/15/82		
APPR.			
DATE			

1. INTRODUCTION

The Refueling Cavity Water Storage Tank TK4-1A, is located in the yard, northeast of the containment, at grade elevation. An elevation of the RCWS tank is shown in Figure 1. The tank is 35'-0 in diameter and 36'-0 from grade to top roof support angle. The tank is covered by a dome roof of 35'-0 radius. The plate material for this tank is 5052-F aluminum. Anchorage consists of 8-1" diameter 36 ksi threaded rods embedded in the slab foundation and bolted to aluminum anchor bolt chairs as shown in Figure 2. The maximum liquid height is 35'-0 from the bottom.

The foundation consists of a 4'-10" thick concrete octagonal slab which sits on a backfill of clean sand above rock.

These calculations were made to evaluate the seismic design adequacy of the Refueling Cavity Water Storage Tank for the seismic response spectra input defined in Figures 3 and 4.

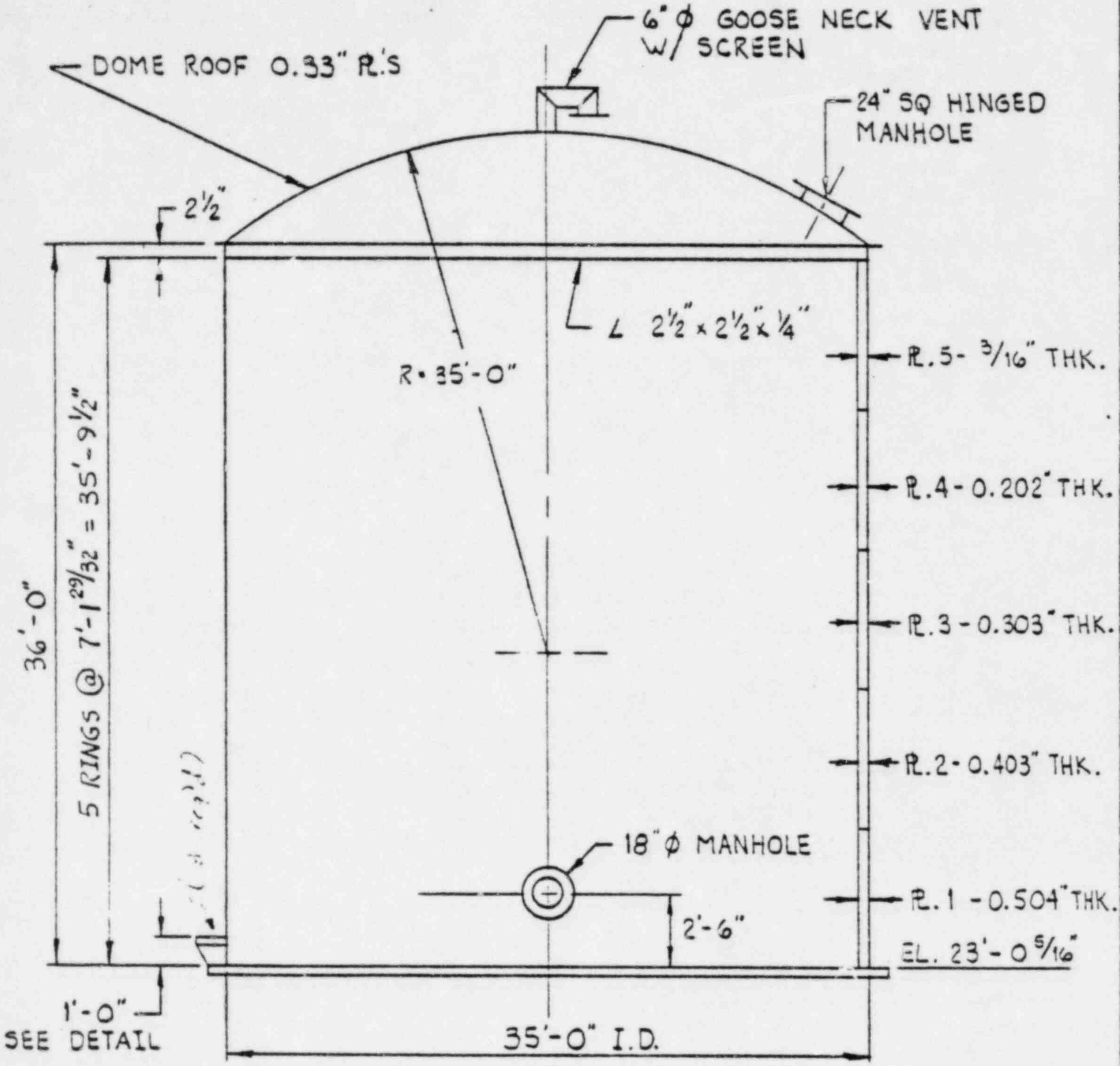
2. RESULTS

The results of this analysis show that the anchorage of the Refueling Cavity Water Storage Tanks is not adequate to resist the forces and moments produced by the prescribed seismic disturbance defined by Figures 3 and 4. The anchor bolts will fail in tension from tank and fluid overturning effects. It should be noted that failure of the anchorage does not necessarily mean that this tank will not maintain structural and leak tight integrity, as defined in category P-1 a passive one of "Allowable Stress Criteria & Damping Values for NUSCO Haddam Neck." The probability of the tanks leaking or the shell buckling, however, is increased due to this anchorage failure. After this failure the tank must be treated as being unanchored and uplift of the tank bottom from the overturning effects may occur. Seismic tests and tank response observed as the result of earthquakes have shown that greatly increased compressive stresses appear in the tank shell on the sides opposite to the uplift. In addition the welds between the tank bottom plate and wall cylinder would be required to transfer very large bending moments not considered in design as the result of lift off.

The tank was analyzed treating some of the fluid as impulsive and moving with the tank and some of the fluid as sloshing and oscillating at a much lower natural frequency.

The calculated natural frequencies for this tank are:

0.29	Hz	1st mode sloshing of fluid in tank
5.88	Hz	1st mode of horizontal impulsive fluid and tank motion
15.59	Hz	2nd mode of horizontal impulsive fluid and tank motion
25.93	Hz	3rd mode of horizontal impulsive fluid and tank motion
35.84	Hz	4th mode of horizontal impulsive fluid and tank motion

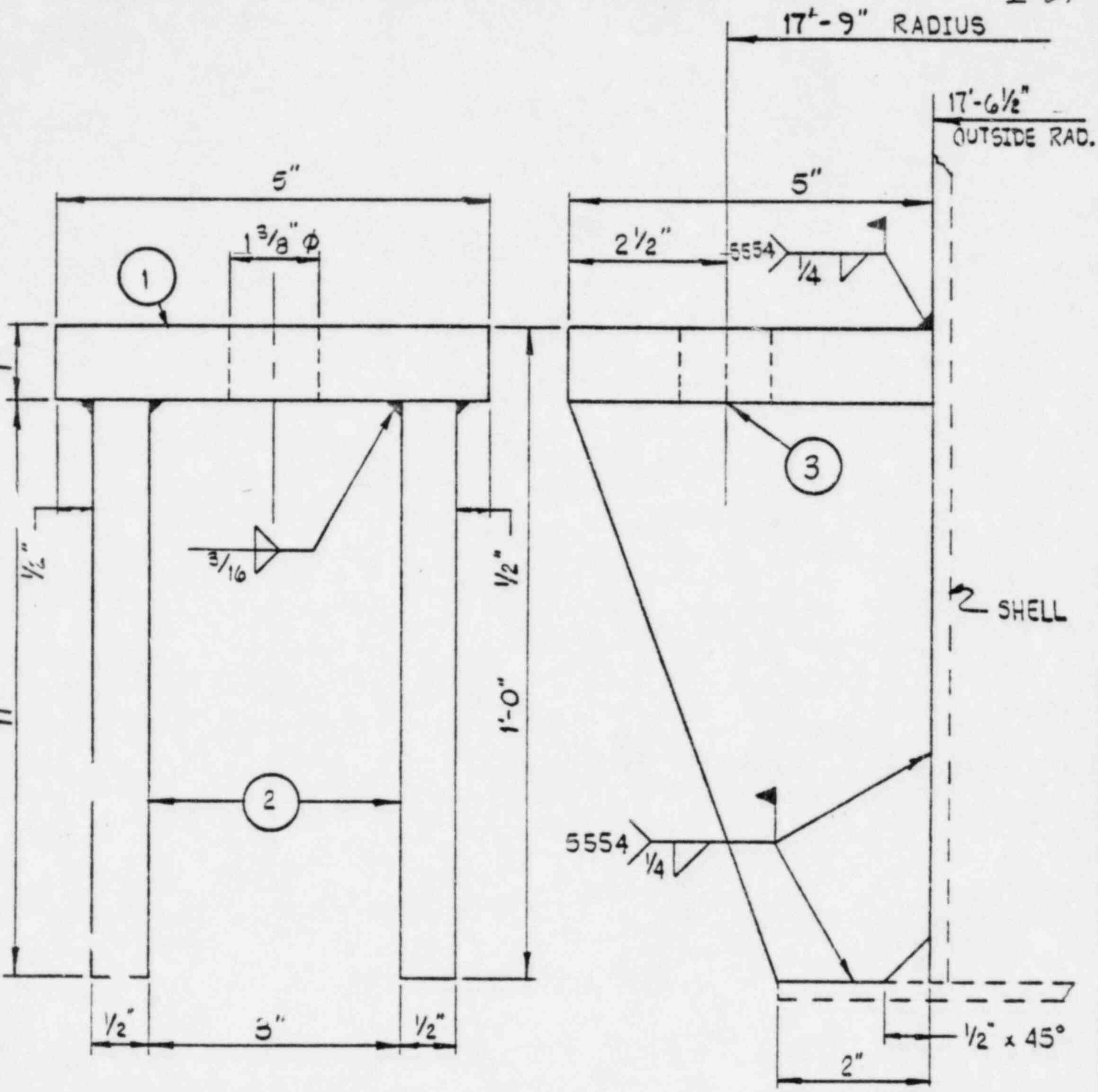


MATERIAL

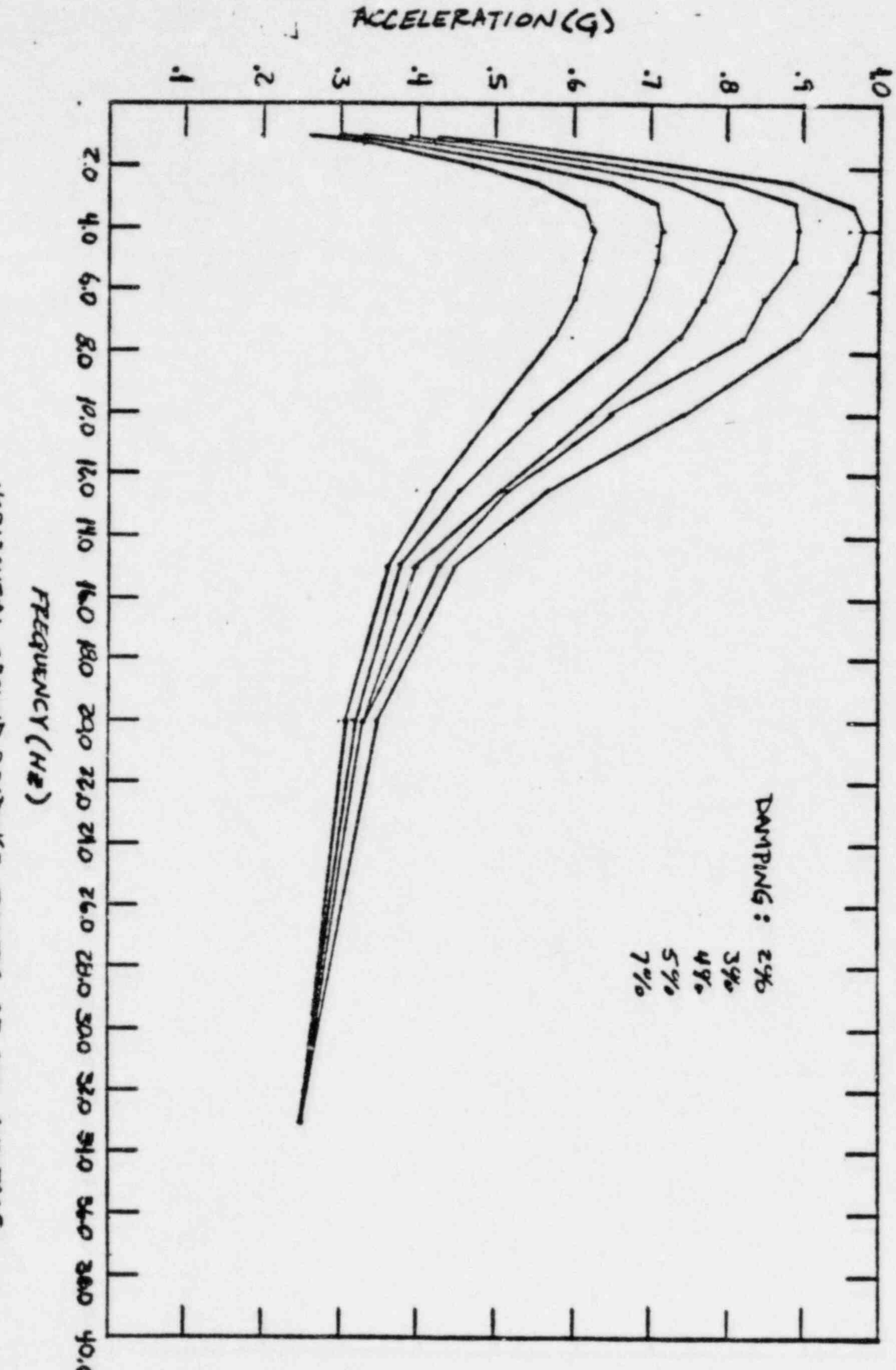
PLATES - AL 5052 - F
STRUCTURAL - 6061 - T6

WEIGHT

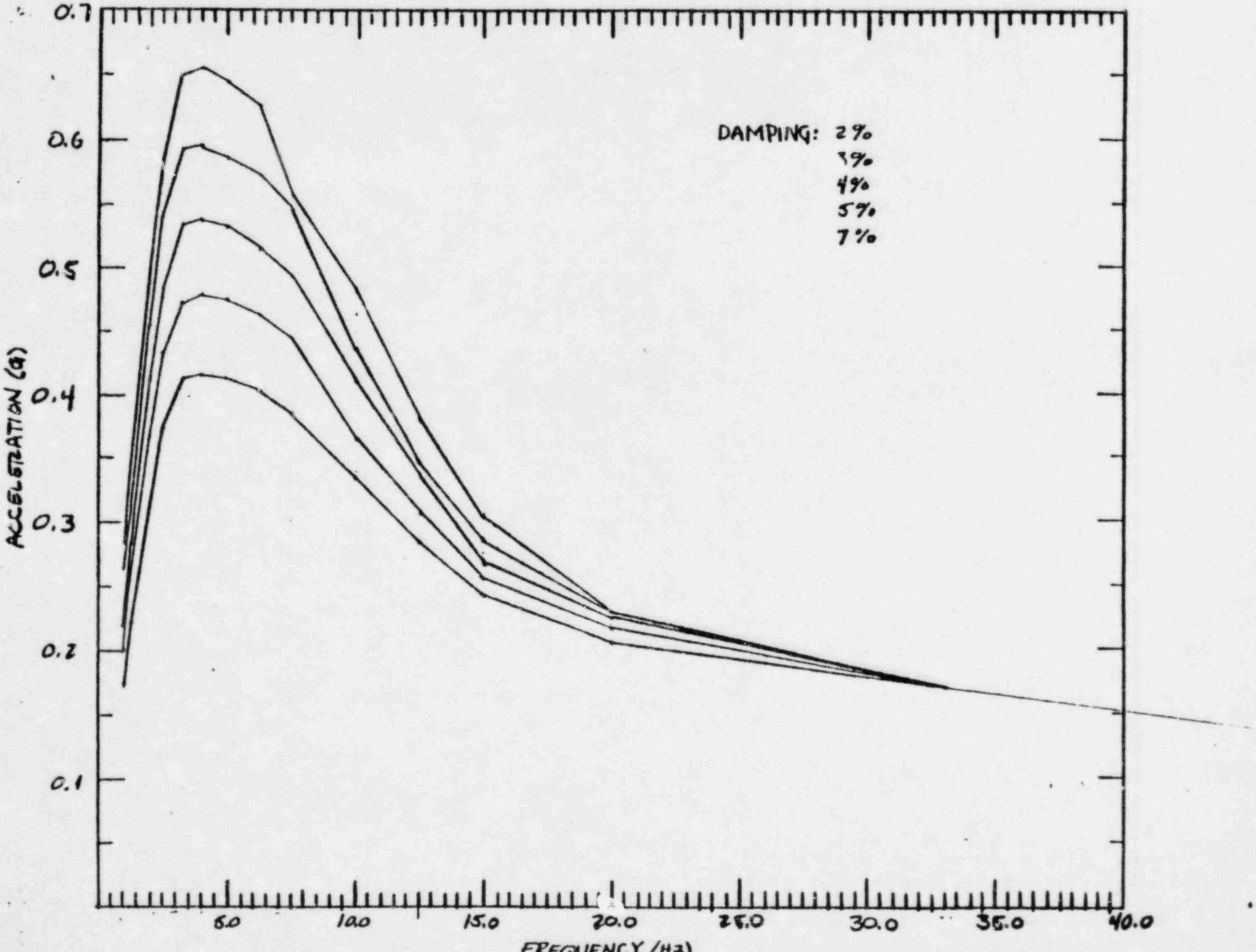
EMPTY - 26,053 lbs.
FULL - 2,127,307 lbs.



No.	DESCRIPTION & DIMENSION		
1.	5083 . JMINUM R. 5"x5"x1". 8 REQ'D.		
2.	5052 ALUMINUM R. SK x ½". 16 REQ'D.		
3.	1" φ ANCHOR BOLTS, NUTS & WASHERS.		



HORIZONTAL GROUND RESPONSE SPECTRA AT SOIL SURFACE



VERTICAL GROUND RESPONSE SPECTRUM AT SOIL SURFACE

Total overturning moment at base of tank

353483 k-in

Total resisting moment at base of tank

370790 k-in

No rigid body rotation of the tank will occur but there may be uplift after failure of the anchor bolts.

The tank shell was checked for buckling due to shear and bending with the following results in Table 1.

Actual Stress		Allowable Stress (1)		Facter of Safety
Shear	Bending	Shear	Bending	Combined Shear & Bending
3169	1760	3203	3148	1.075

TABLE 1

(1) allowable stresses are computed from buckling criteria

$$FS = \frac{1}{\left(\frac{f_b}{F_b}\right) + \left(\frac{f_v}{F_v}\right)^2}$$

The anchor bolts and anchor bolt chairs were checked for tension. The anchor bolts are assumed to be 36 ksi yield stress material and the weld metal for the anchor bolt ~~chains~~ chairs is assumed to be 40 ksi yield stress. Stresses are shown in Table 2 for the bolts and chairs in Figure 5.

	Computer Stress		Alowable Stress		Factoer of Safety	
	Bolt	Bolt Chair	Bolts	Bolt Chairs	Bolts	Bolt Chairs
A	515000	78760	40600	24000	FAILURE	FAILURE
B	364000	55969	40600	24000	FAILURE	FAILURE

TABLE 2

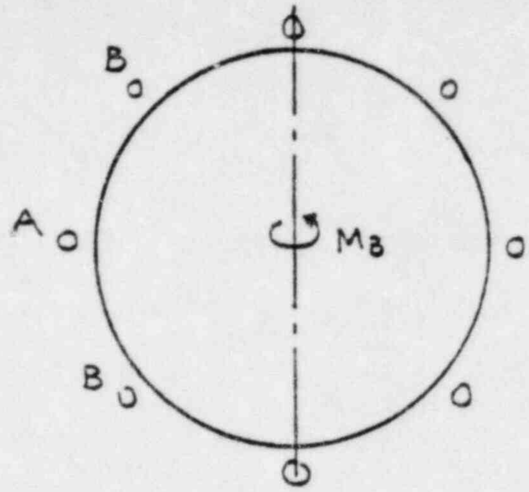


FIGURE 5

The total hydrodynamic and hydrostatic pressure on the tank shell was computed and hoop stresses were calculated at the bottom of each tank shell ring as shown in Table 3.

	PLATE THICKNESS INCHES	DEPTH IN FLUID INCHES	P_T, PSI	HOOP STRESS KSI	ALLOWABLE STRESS KSI
RING 5	.1875"	76.375"	4.02	4.505	10
RING 4	.202"	162.28125	7.275	7.563	10
RING 3	.303"	248.1875"	11.074	7.674	10
RING 2	.403"	334.09375	15.24	7.941	10
RING 1	.504"	420	19.17	7.988	10

TABLE 3

$$\text{Hoop Stress} = \frac{P_T \cdot r}{t}$$

$$P_T = (P_1 + P_2 + P_v)^{1/2} + P_{STATIC}$$

where P_1 = impulsive motion hydrodynamic pressure
P_2 = sloshing motion hydrodynamic pressure
P_v = vertical hydrodynamic pressure

At this time, recommendations for correcting the potential problems of the Refueling Cavity Water Storage Tank are uncertain because of space limitations in the area of the tank.

3. LOAD CRITERIA AND FAILURE MODE ASSUMPTION

Failure of the tank anchorage because of the methods used in the original design was considered the area of greatest analytical concern.

A modified ground response spectra was used in the analysis of the tank. The basic ground response spectra is considered to be representative of accelerations at the rock surface, which is 6 feet below the foundation of the tank. The basic rock ground response spectra were multiplied by a factor of 1.5 to produce modified ground response spectra at the soil surface to be used as input to the analysis of the tank. Curves for seven percent damping were used with impulsive mode response. One-half percent damping was used for the sloshing mode of response. This amount of damping had negligible effect on forces produced from sloshing of the water.

Limiting nozzle loads on the tank were assumed to cause stresses in the nozzles of $S_{all}/3$. These loads were compared with the loads on the tank anchorage system from the earthquake. The largest nozzle induced load was 2.5% of the overturning movement on the base of the tank determined from the earthquake. The computations of these nozzle loads are found in Section 6. Because of the small values determined the effect of nozzle loads on tank supports were not combined with earthquake loads.

4. STRESS CRITERIA

The stress criteria are as shown in Appendix A with these exceptions:

Allowable stress in bending for the tank shell is governed by buckling criteria, $S_{all} = 3.148$ ksi.

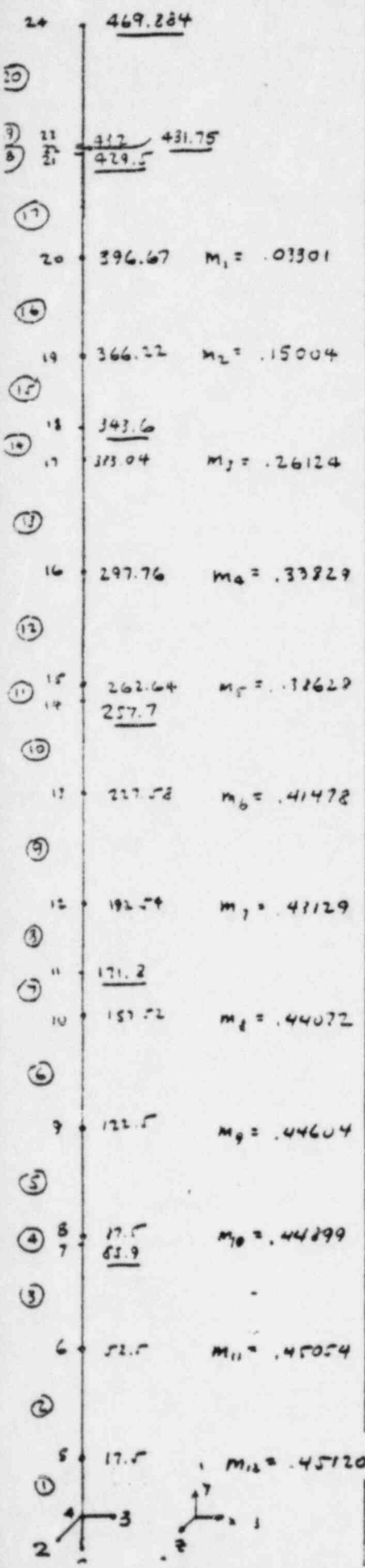
Allowable stress in shear for the tank shell is governed by buckling criteria, $S_{all} = 5.203$ ksi.

5. METHODS OF ANALYSIS

Since the tank is supported on compacted backfill the effect of soil structure interaction was considered in the modeling of the tank. Equivalent soil springs were computed for vertical, lateral and rocking motions of the tank. The torsional spring was considered very stiff and the largest value for spring constants in the computer program was used. The equivalent soil springs computed were also quite stiff and had a minimal effect on the tanks fundamental frequency of vibration. This effect placed the fundamental frequency near the peak of the response spectrum and produced the large loads on the anchorage.

The SAP IV computer program was used for the analysis of the impulsive motion of the tank and fluid and the sloshing motion was computed and combined with the computer analysis by hand.

The boundary elements for the computer model were input with the approximate soil spring stiffness in each direction. The tank was modelled as a cylindrical beam stick with fluid mass lumped at points along the length. A sketch of the computer stick model is shown in Figure 6 along with the values for the section properties and lumped masses. The computation of these values is described in Section 6 along with computations for the equivalent soil spring constants.



Element	A in^2	A_{vz}	A_{vx}	I_y	I_z	I_x
20	1294.83	1294.83	1294.83	$56.86\cdot10^6$	$28.42\cdot10^6$	$28.42\cdot10^6$
19	3318.31	1659.15	1659.15	$148.09\cdot10^6$	$74.04\cdot10^6$	$74.04\cdot10^6$
18	330.06	165.032	165.032	$14.57\cdot10^6$	$7.29\cdot10^6$	$7.29\cdot10^6$
17, 16, 15	247.511	123.755	123.755	$10.93\cdot10^6$	$5.46\cdot10^6$	$5.46\cdot10^6$
14, 13, 12, 11	266.661	133.33	133.33	$11.77\cdot10^6$	$5.88\cdot10^6$	$5.88\cdot10^6$
10, 9, 8	400.088	200.044	200.044	$17.67\cdot10^6$	$8.83\cdot10^6$	$8.83\cdot10^6$
7, 6, 5, 4	532.256	266.128	266.128	$23.52\cdot10^6$	$11.76\cdot10^6$	$11.76\cdot10^6$
3, 2, 1	665.81	332.905	332.905	$29.43\cdot10^6$	$14.72\cdot10^6$	$14.72\cdot10^6$

FORCE: K
MASS: $\frac{K\text{-}sec^2}{in}$
LENGTH: in
Time: sec
PRESSURE: K/in^2
Density: $\frac{K\text{-}sec^2}{in^4}$
g: in/sec^2

TANK #12 DYNAMIC MODEL

S 352.1 I-36

CS 2? +

SYSTEM Mechanical - #13

COMPONENT NAME Stm. Driven Aux. Feedwater Pump COMPONENT № P-32-1A, 1B

LOCATION Outside ELEVATION 21'-0"

COMPONENT SAFETY FUNCTION: ACTIVE ☐ PASSIVE 1 ☐ 2 ☒

S-LIST PAGE № 135

METHOD OF ANALYSIS: FINITE ELEMENT MDOF MODEL (one dimensional)

SPECTRAL CURVES USED: Ground Response Curves

DAMPING VALUE ASSUMED: 7%

ACCEPTANCE BEHAVIOR CRITERIA USED:

$S_{all} \leq 0.9 S_{yield}$

COMPUTER CODE USED: MULTI-PURPOSE FINITE ELEMENT

REMARKS:

S&A	CALCULATION COVER SHEET	NORTHEAST UTILITIES - HADDAM NECK	REV. №	0		
			BY	PNP	CAM	
			DATE	4-?-82		
			CHK'D	EAT		
			DATE	4/15/82		
			APPR.			
			DATE			

1. INTRODUCTION

The Horizontal Steam Driven Auxiliary Feedwater Pump is a single stage, Worthington model, centrifugal diffuser pump located at grade (elev. 21'-0") outside of containment. The pump is operated by a turbine driver and joined to the turbine by a mechanical coupling. The pump and turbine assembly are mounted on a common bedplate, consisting of channel sections fabricated from plates. Two additional end plates and another plate, which serves a a drainage collector and also provides stiffness, are part of the bedplate. Poured in place anchor bolts then fasten the bedplate to the concrete floor. Detailed information regarding the pump and turbine assembly are found in Figures 1-1 through 1-4.

The horizontal pump identified by equipment tag number, P-32, is classified as a passive component II. Pump Number P-32 is not required to move or change state during a seismic event, but, may be required to do so after the event, and must retain its structural and leak tight integrity. Thus, the intent of this report is to evaluate by analysis the anchorage system and the pump internals necessary for the pump P-32, to withstand a Safe Shutdown Earthquake Seismic Event and its ability to operate following the seismic event.

The analytical model of P-32 is illustrated in Figures 1-5 and 1-6, with SAP IV being the general purpose, finite element computer program used in the analyses.

2. SUMMARY OF RESULTS

Based on the results of the analyses, the Horizontal Steam Driven Auxiliary Feedwater Pump, P-32, will maintain its structural and leak tight integrity and be capable of operating following the defined seismic event.

A finite element model of the pump and turbine was prepared which considered the pump and turbine casings as rigid members. The shafts of the pump and turbine and bedplates were modelled with their respective structural stiffness properties. The bolts, in turn, were modelled as linear springs. This resulted in frequencies for the first eight modes to be less than 30 hz. They are as follows:

Mode Number	Frequency Hz	Dominant Direction
1	6.01	X
2	9.25	Y
3	11.40	Y
4	13.62	X
5	14.60	Z
6	22.47	Z
7	24.63	X
8	27.70	X
9	55.02	X

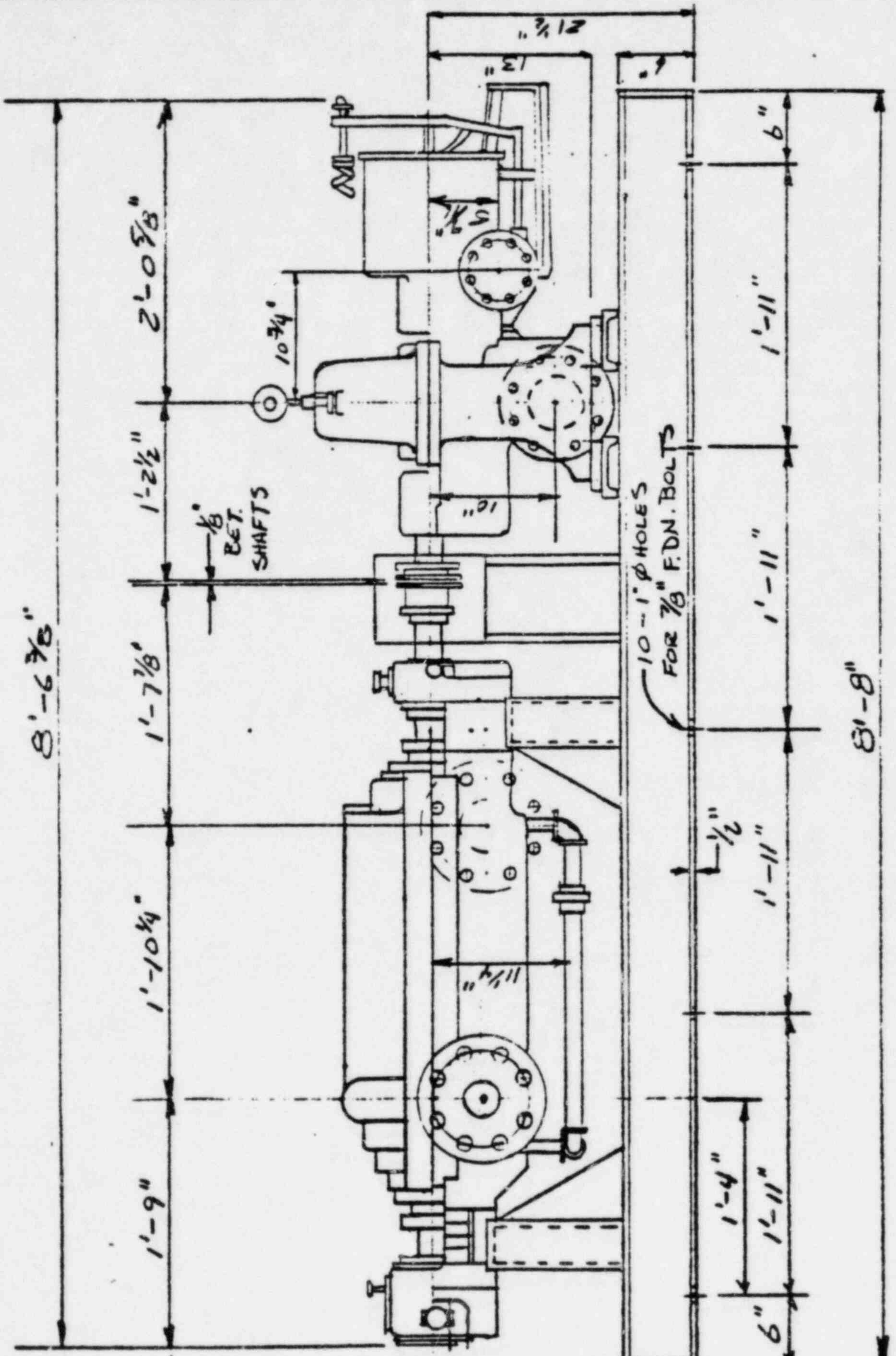
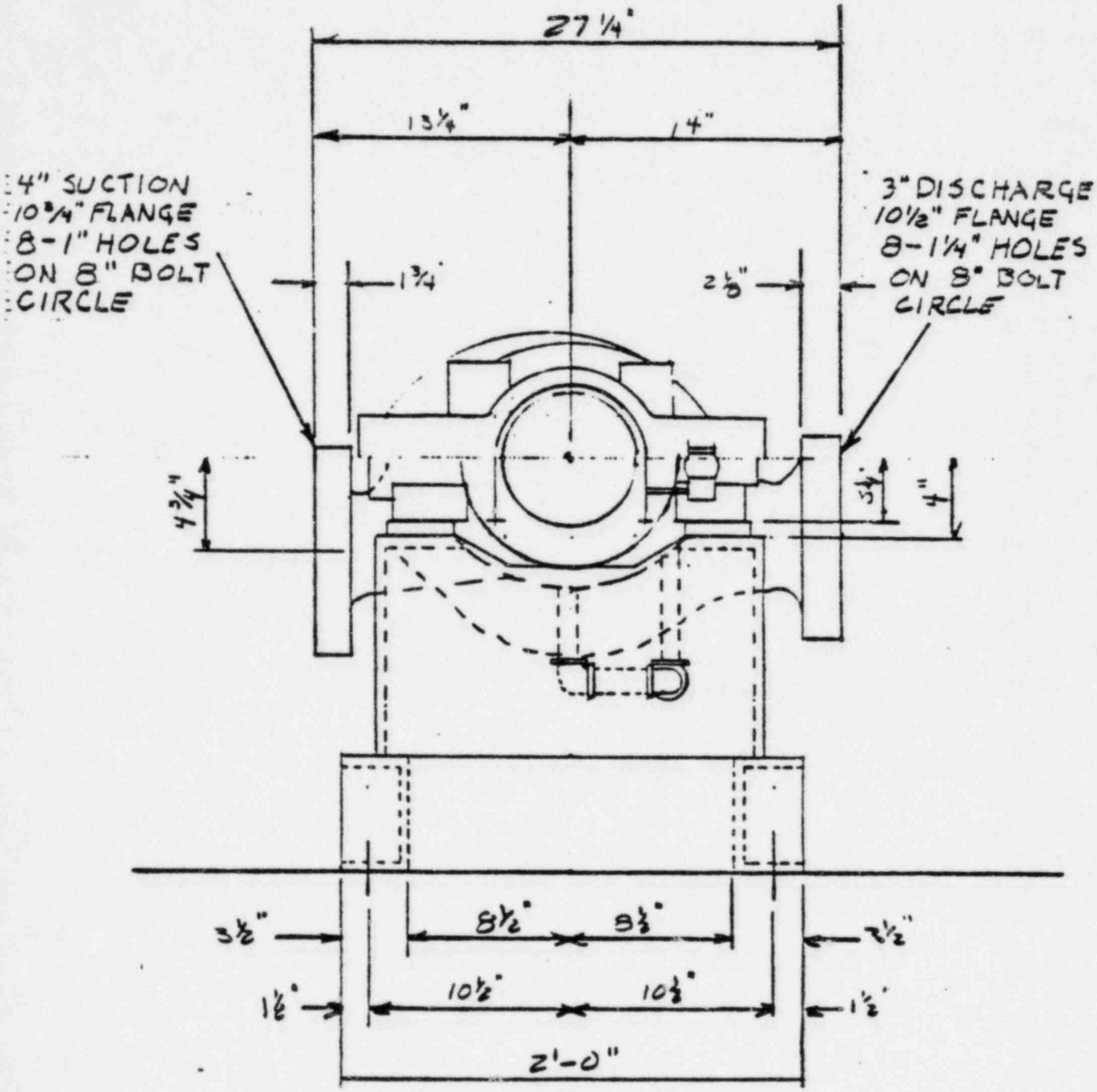


FIG. I-1

SCALE: 1" = 1'-0"

WEIGHTS

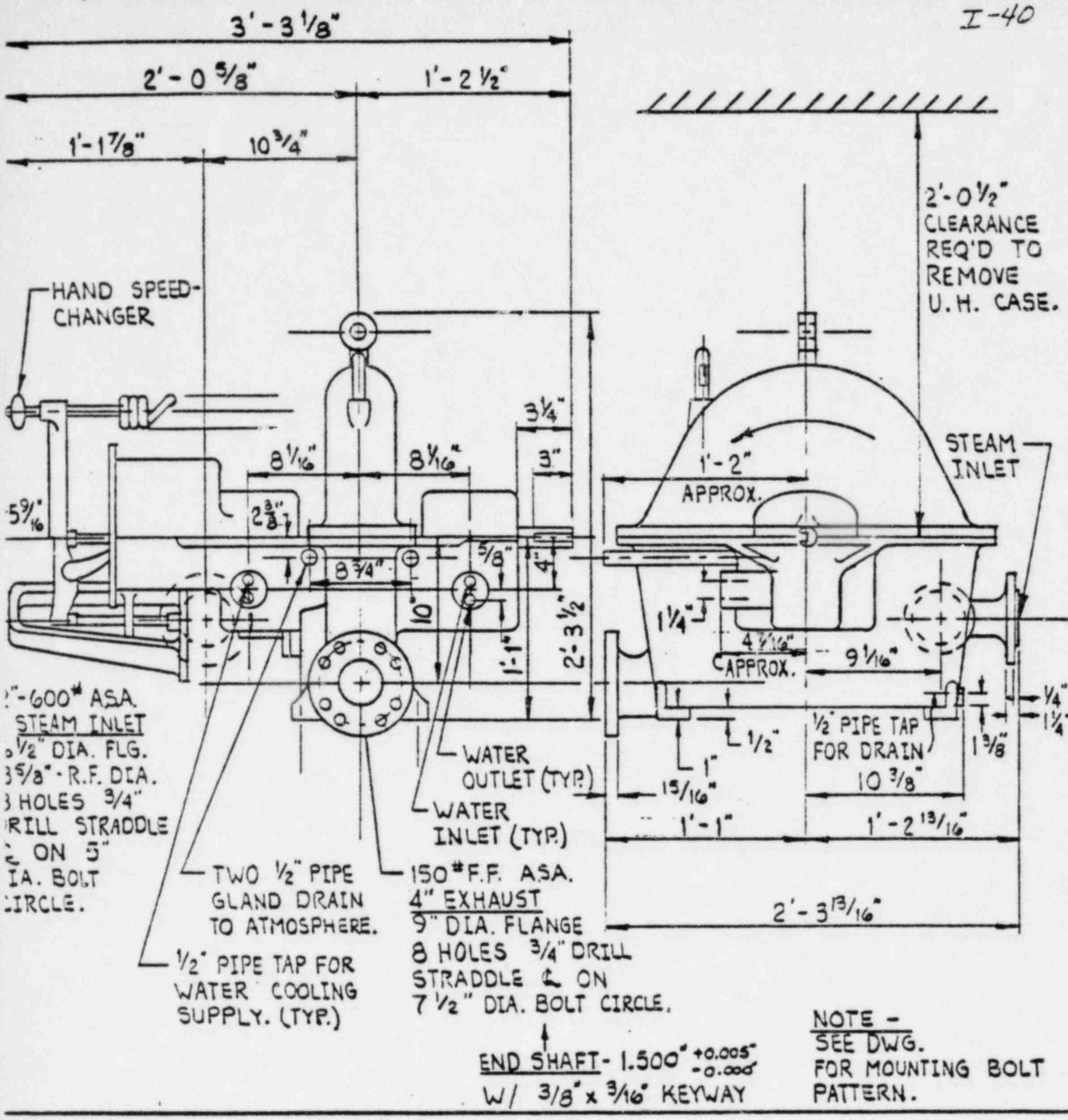
PUMP & COMMON BASEPLATE	2900 Lbs
TURBINE DRIVER	700 Lbs
TOTAL	3600 Lbs



SECTION "A-A"

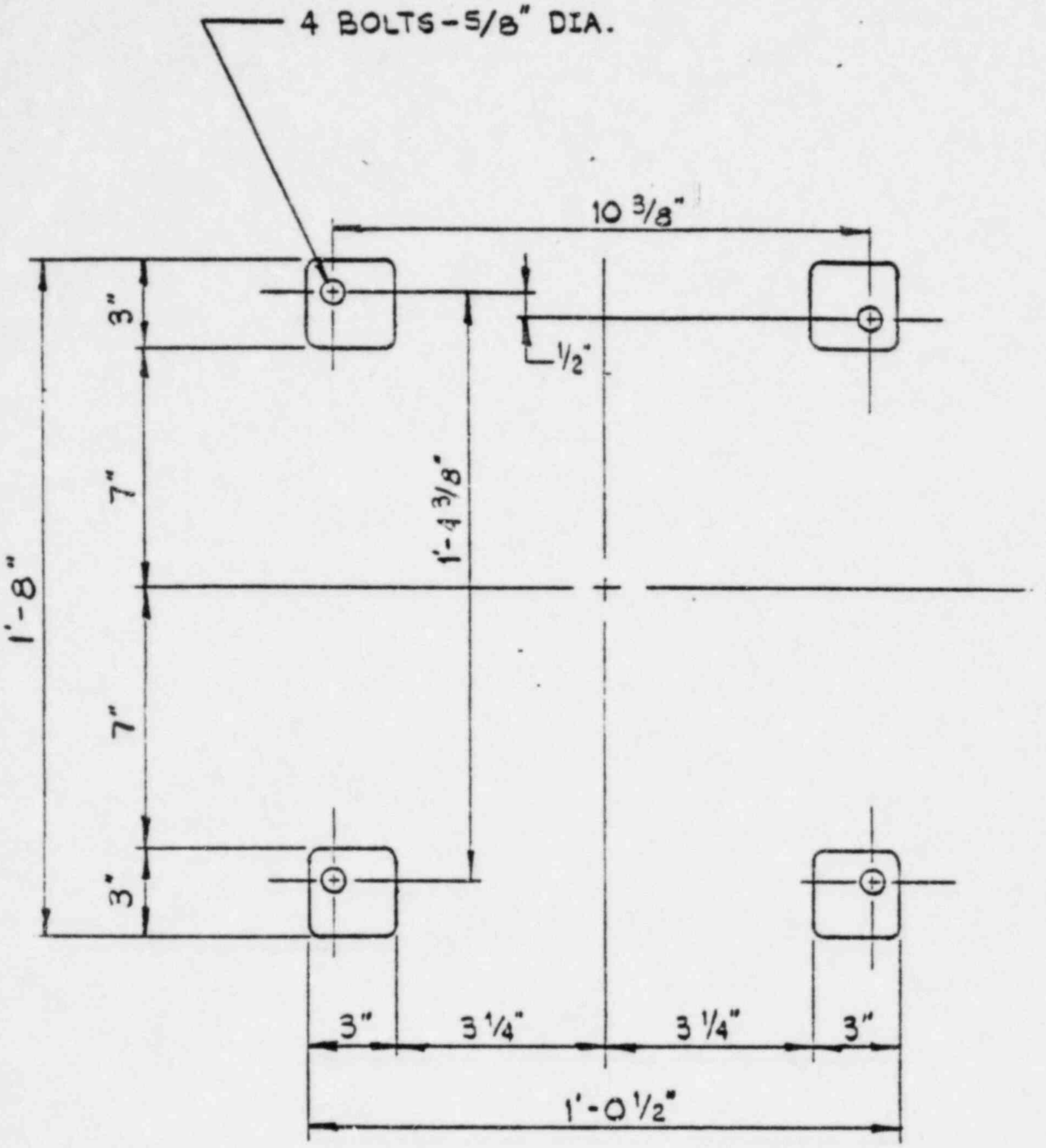
SCALE 1 1/2" = 1'-0"

FIG. 1-2



NET WT. 700 lbs.
HEAVIEST PIECE 370 lbs.

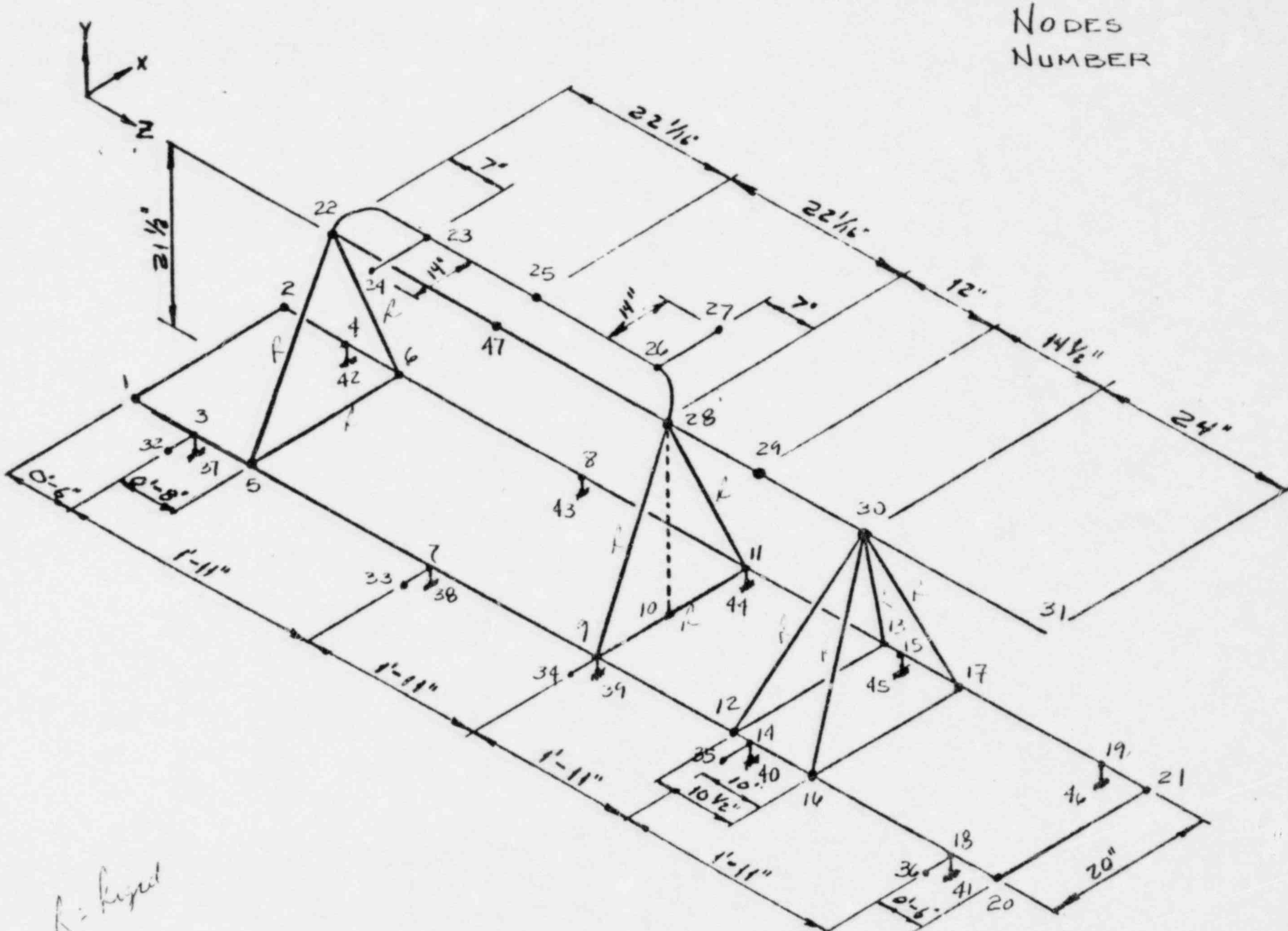
FIG 1.3



DIMENSIONS ARE APPROXIMATE.

PLAN VIEW OF FEET

FIG. I-4


NODES
NUMBER
Y
X
Z
21 1/2"
22 1/16"
22 1/16"
12"
14 1/2"
24"
7"
7"
14"
14"
0'-6"
0'-8"
1'-11"
1'-11"
1'-11"
1'-11"
10"
10 1/2"
0'-6"
20"
1
2
3
4
5
6
7
8
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10
11
12
13
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15
16
17
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28
29
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f = Rigid

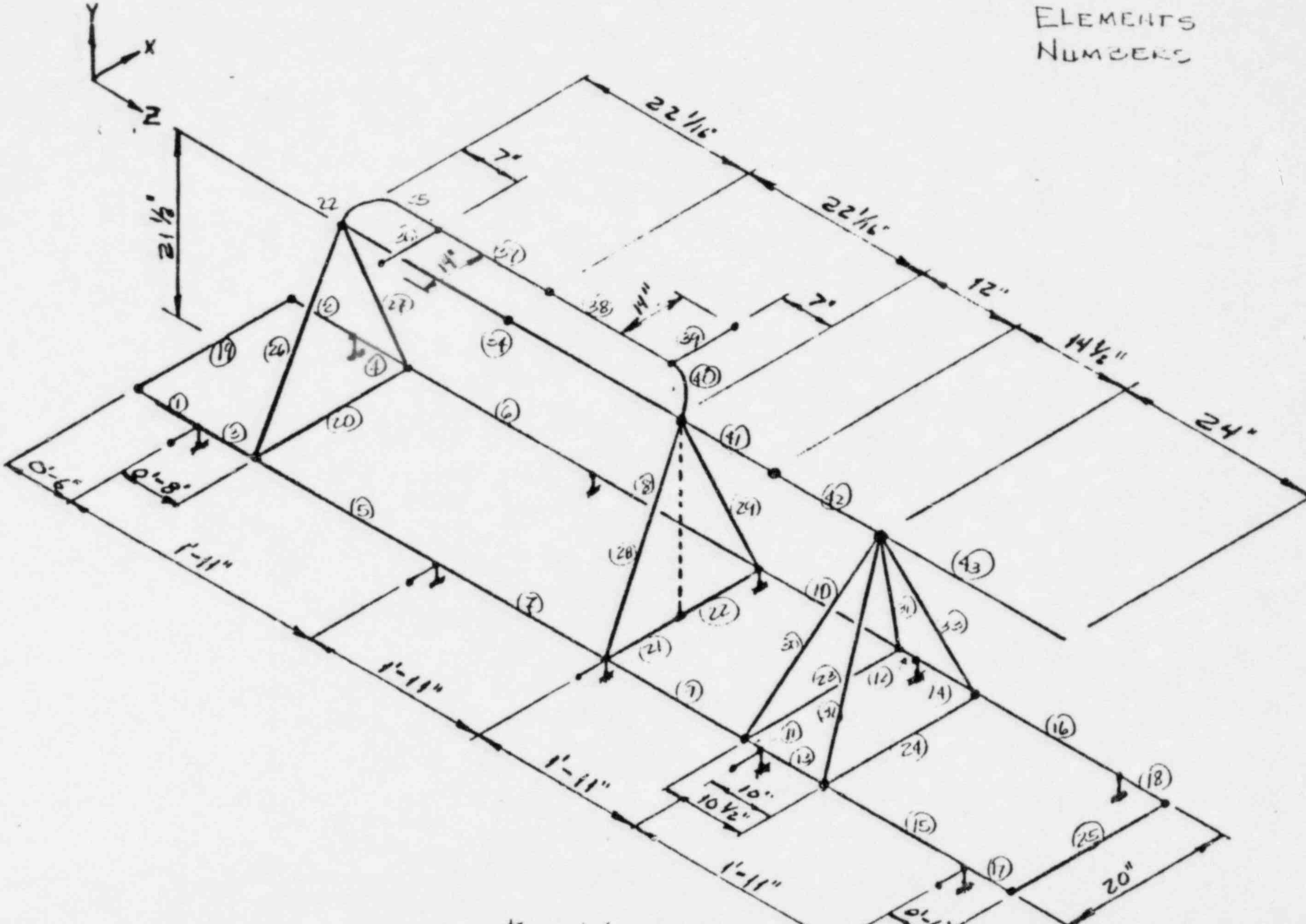

ELEMENTS
NUMBERS
Y
X
Z
21 1/2"
22 1/16"
22 1/16"
12"
14 1/2"
24"
7"
7"
14"
14"
0'-6"
0'-8"
1'-11"
1'-11"
1'-11"
1'-11"
10"
10 1/2"
0'-6"
20"

Fig. I-6

The first mode obtained was a coupled sliding and rocking mode, with the sliding in the horizontal X-direction, and the rocking about the Z axis. The second and third modes were rocking modes about the horizontal X axis (transverse axis) with the shaft and pump housing participation in a vertial motion. The fourth mode was predomiently a coupled shaft and housing bending mode.

The allowable stress on the bolts were as follows:

Tension $F_{tb} = \frac{0.5}{2} S_u \times 1.4 = 40.6$ ksi

Shear $F_{vb} = \frac{0.62}{3} S_u \times 1.4 = 16.78$ ksi

Combined Shear and Tension $$\frac{f_t^2}{F_{tb}^2} + \frac{f_v^2}{F_{vb}^2} \leq 1$$

The bolts stresses and safety factors were as follows:

Tensile	$f_t = 4.4$ ksi,	S.F. = 9.2
Shear	$f_v = 1.38$ ksi,	S.F. = 12.2

Combined, using the interaction of Appendix XVII of the ASME Code.

$0.0186 \leq 1$ S.F. = 53.76

The shaft stresses

Tensile	$f_t = 0.345$ ksi	S.F. = 94
Shear	$f_v = 0.11$ ksi	S.F. = 294
Torsion	$f_{tor} = 0.19$ ksi	S.F. = 170
Bending	$f_b = 13.98$ ksi	S.F. = 2.3

The shaft and pump casing is relative differential displacement was 6×10^{-6} in which is well within the allowable of 5×10^{-3} in.

No evaluation of bearing loads has been made but, given the low stresses and deformations in the shaft, bearing behavior should not be a limiting condition.

3. LOAD CRITERIA

The analysis was performed by considering both static and dynamic load cases, and then combining forces and moments by absolute summation method to determine stresses. The static load case was performed on a structural model of the pump, motor, and bedplate and applying the nozzle loads as external static loads. These external nozzle loads were obtained from API Standard 610 for horizontal centrifugal pumps, having 4" discharge nozzles or smaller. This allowed the external piping loads to be transmitted through the pump to the anchor bolts.

The dynamic load case considered the mass as well as stiffness characteristics of the pump. It incorporated the ground response spectra for two horizontal orthogonal directions oriented parrallel and perpendicular to the pump shaft as well as in the vertical direction. A damping value of 7% was used in the analysis. See Figures 3-1 and 3-2 for the response spectra curves used.

4. STRESS DEFORMATION - STABILITY CRITERIA

The allowable stress criteria for the passive, P-2, component is

$$S_{all} \leq 0.9\ S_y$$

and for the poured in place bolting

$$S_{all} \leq 0.7\ S_u$$

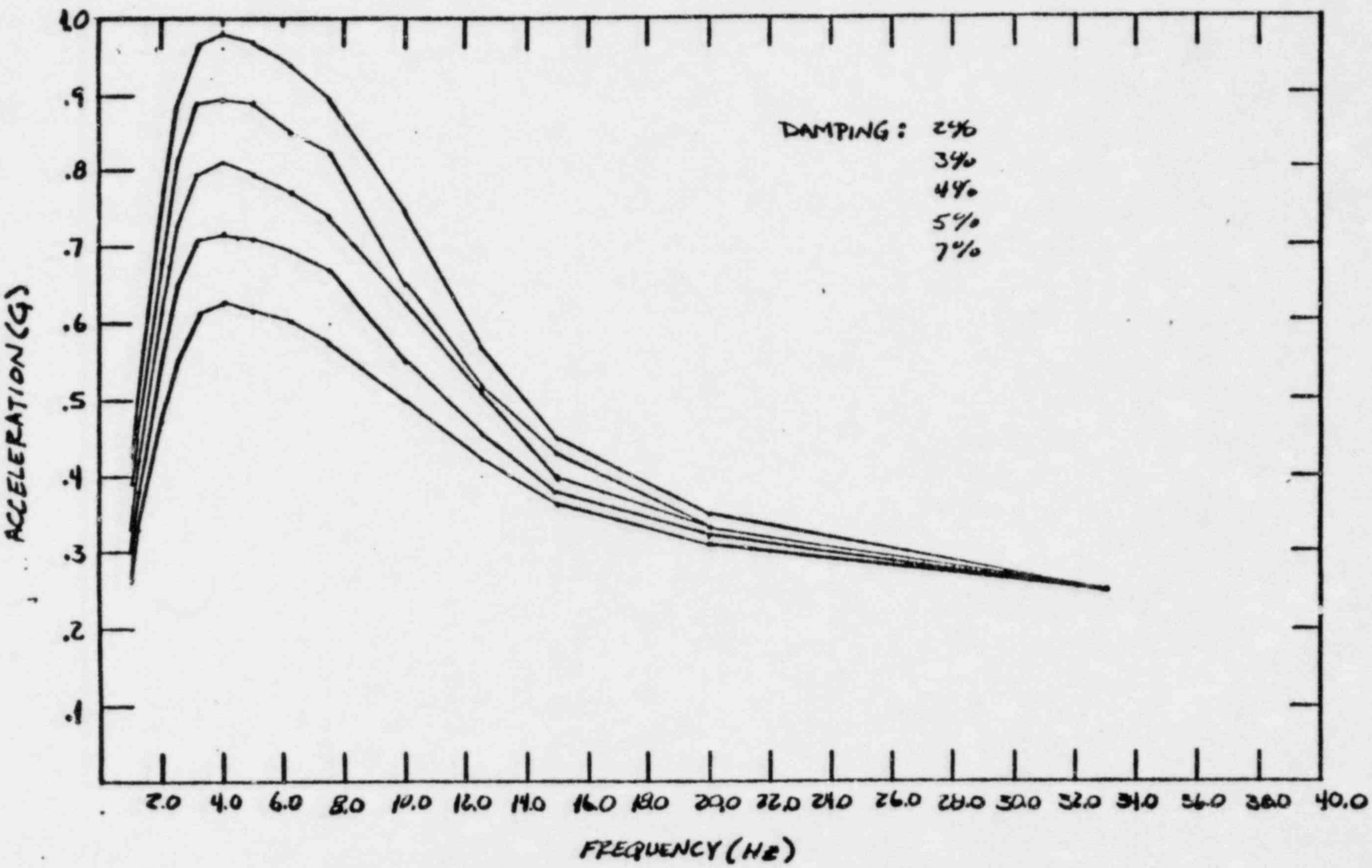
These stresses are established in the "Allowable Stress Criteria For the Haddem Neck Plant" and are attached to this report as Appendix A.

The requirement of API 610 for limiting the shaft deflection to a maximum of 0.005 inches has also been considered in this analysis.

5. METHOD OF ANAYLSIS

The steam driven auxiliary feedwater pump was analyzed by the response spectrum option of SAP IV, which is a general purpose, finite element program having both static and dynamic capabilities.

The analytical model, which will adequately predict the behavior of the mechanical component under seismic loads, was developed using the principles developed in a paper by C. K. McDonald entitled "Seismic Qualification by Analysis of Nuclear Power Plant Mechanical Components." Normally, McDonald recommends using ten lump masses to represent the pump assembly. These masses would then be located along the shaft centerline to represent the pump casing, impeller, coupling, motor casing, and motor rotor. Because the center of mass of the unit is several inches below the shaft centerline, applying the masses at the shaft centerline is then conservative with respect to the system overturning.



HORIZONTAL GROUND RESPONSE SPECTRA AT SOIL SURFACE

FIG 3-1

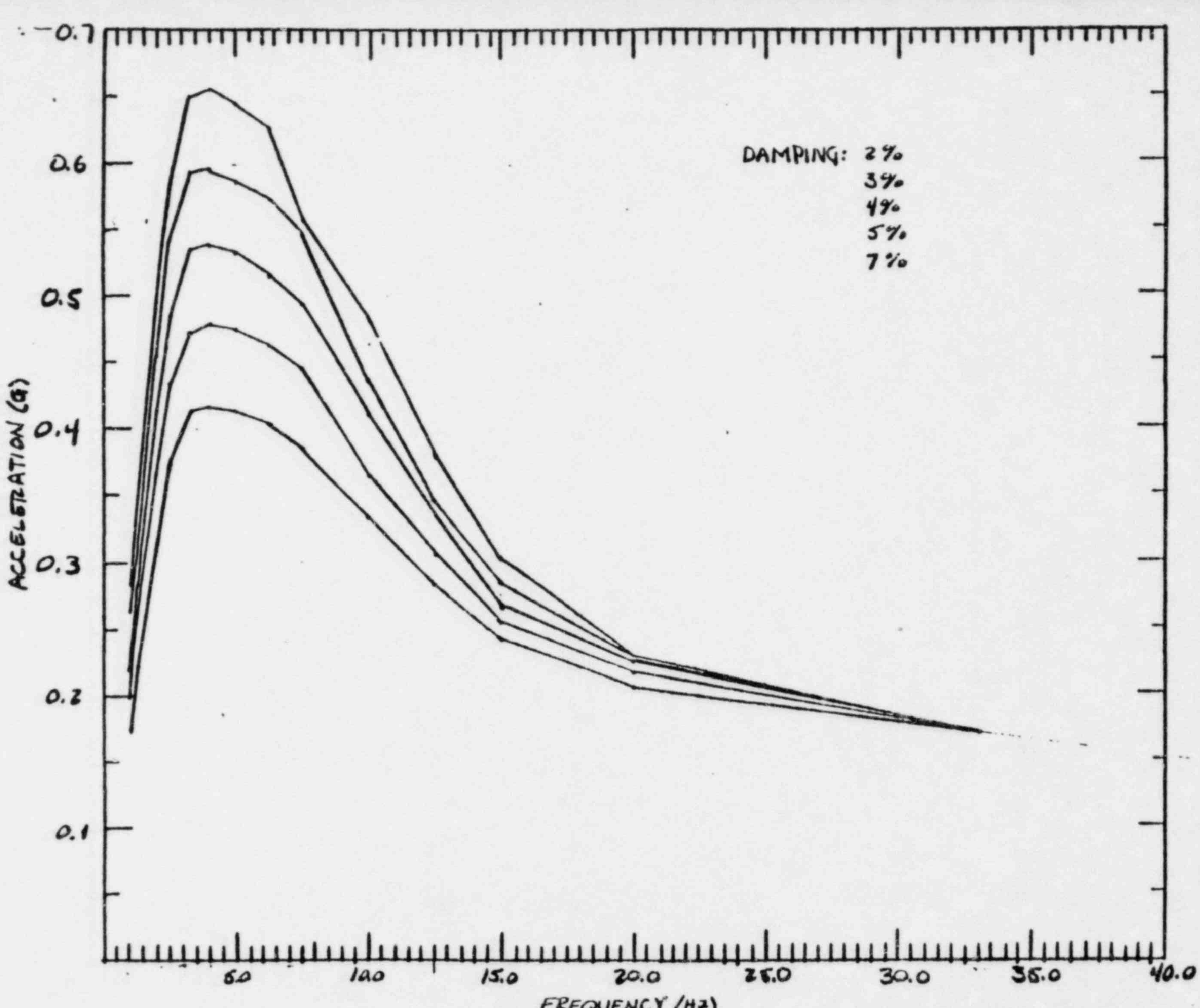


FIG. -3-2 VERTICAL - GROUND RESPONSE SPECTRUM AT SOIL SURFACE

The bedplate is supported only at the anchor bolts. This is true for upward motion but for downward motion the bedplate is continuously supported. Therefore, the model will predict lower frequencies for the system than what actually exists. Since the predicted frequencies are usually higher than the resonance frequency where the pump is located, and the actual frequencies are even higher, the support assumptions are conservative.

Basically, two analyses were performed to predict the behavior of the piece of equipment.

1. A static analysis with nozzle loads

2. Dynamic analysis with a response spectra loading

Both resulting loads were then combined by direction summation and stresses were determined and shaft deflections were calculated.

SYSTEM Electrical - # 2

COMPONENT NAME Motor Control Center COMPONENT № MCC #1

LOCATION Screenwell House ELEVATION 21'-6"

COMPONENT SAFETY FUNCTION: ACTIVE ☐ PASSIVE 1 ☒ 2 ☐

S-LIST PAGE № 150

METHOD OF ANALYSIS: A finite element model of beam and plate elements is constructed and a response spectra analysis is then done.

SPECTRAL CURVES USED: FRS of Screenwell House; Figures 4, 5, 6

DAMPING VALUE ASSUMED: 7%

ACCEPTANCE BEHAVIOR CRITERIA USED: ASME Appendix XVII + Appx A, S_{all} & S_{yield} for component and component support structure. Bolting and welding as indicated in "Allowable Stress Criteria"

COMPUTER CODE USED: SAP IV

REMARKS: Response Spectrum Analysis is conducted after frequencies and mode shapes of the model are in agreement with those determined in a low level insitu test for MCC1.

S&A	CALCULATION COVER SHEET	NORTHEAST UTILITIES - HADDAM NECK	REV.№ 0 BY PNP DATE 6-1-92 CHK'D [illegible] DATE 6/1/92 APPR. DATE

1. INTRODUCTION

Motor Control Center Number 1, MCC1, is located at floor elevation 21'-6" in the screenwell house. It is a box-shaped structure composed of three individual cabinets, bolted together, whose overall dimensions are 7'-0" x 1'-3" x 7'-6" high. These cabinets consist of unistrut framing members enclosed by metal plates, which are joined to the framing members by screw fasteners. Electrical devices contained within the cabinet are then attached directly to the unistrut framing of the cabinet.

Revised seismic supports have already been added to MCC1. As shown in Figures 1-1 through 1-4, an angle is bolted at the top along the full length of the cabinet. Structural tubing is then used to connect the angle to an existing channel, which spans between building columns. On the bottom, two clip angles are attached to the cabinet by welded plates, and the angles are then expansion anchored to the floor.

The analytical model of MCC1, including the additional seismic supports, is illustrated in Figures 1-5 through 1-8. The analyses was made using the general purpose, finite element program, SAP IV, to determine the structural adequacy of the MCC1 cabinet, its supports and anchorage system to withstand the defined seismic event.

2. SUMMARY OF RESULTS

Based upon the results of the analyses, the MCC1 will maintain structural integrity during and after the defined seismic event.

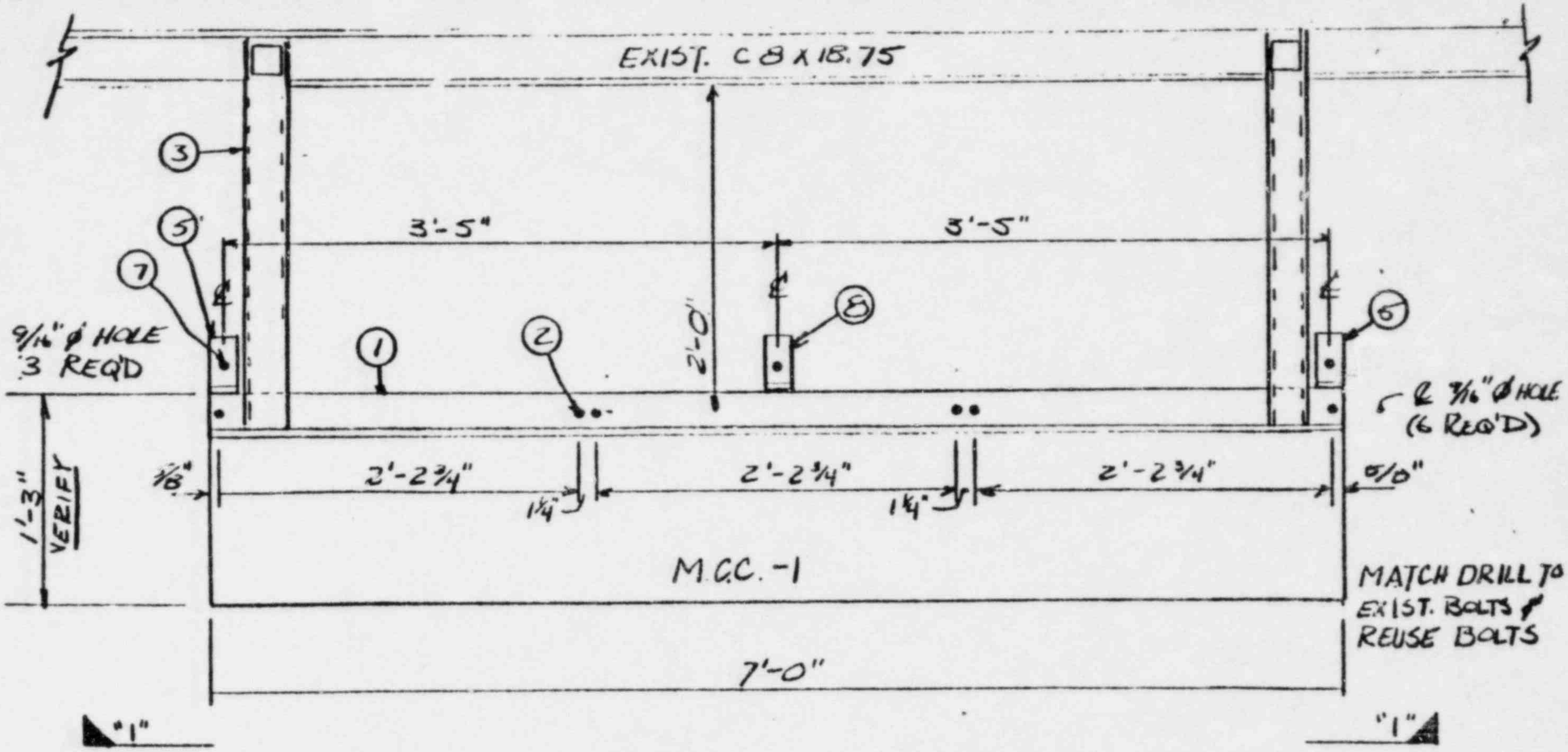
A finite element model was constructed of beam and plate elements and produced frequencies and mode shapes that were in close proximity to those that were previously established by a low impedance, insitu test of the MCC1. The frequencies for the first two modes are 17.2 hz and 23.6 hz for the test and 19.7 hz and 20.8 hz for the analytical model. Figures 2-1 and 2-2 for the test and Figures 2-3 and 2-4 for the model indicate the first two mode shapes of the cabinet with the relative displacements along the transverse (z) direction. Thus, the model was justified, and a response spectrum analysis was then conducted.

All of the seismic supports exhibit stresses that are well below the allowables for both the structural members and the connections. Considering only the limiting sections, the following results are obtained:

(1) Structural Member:

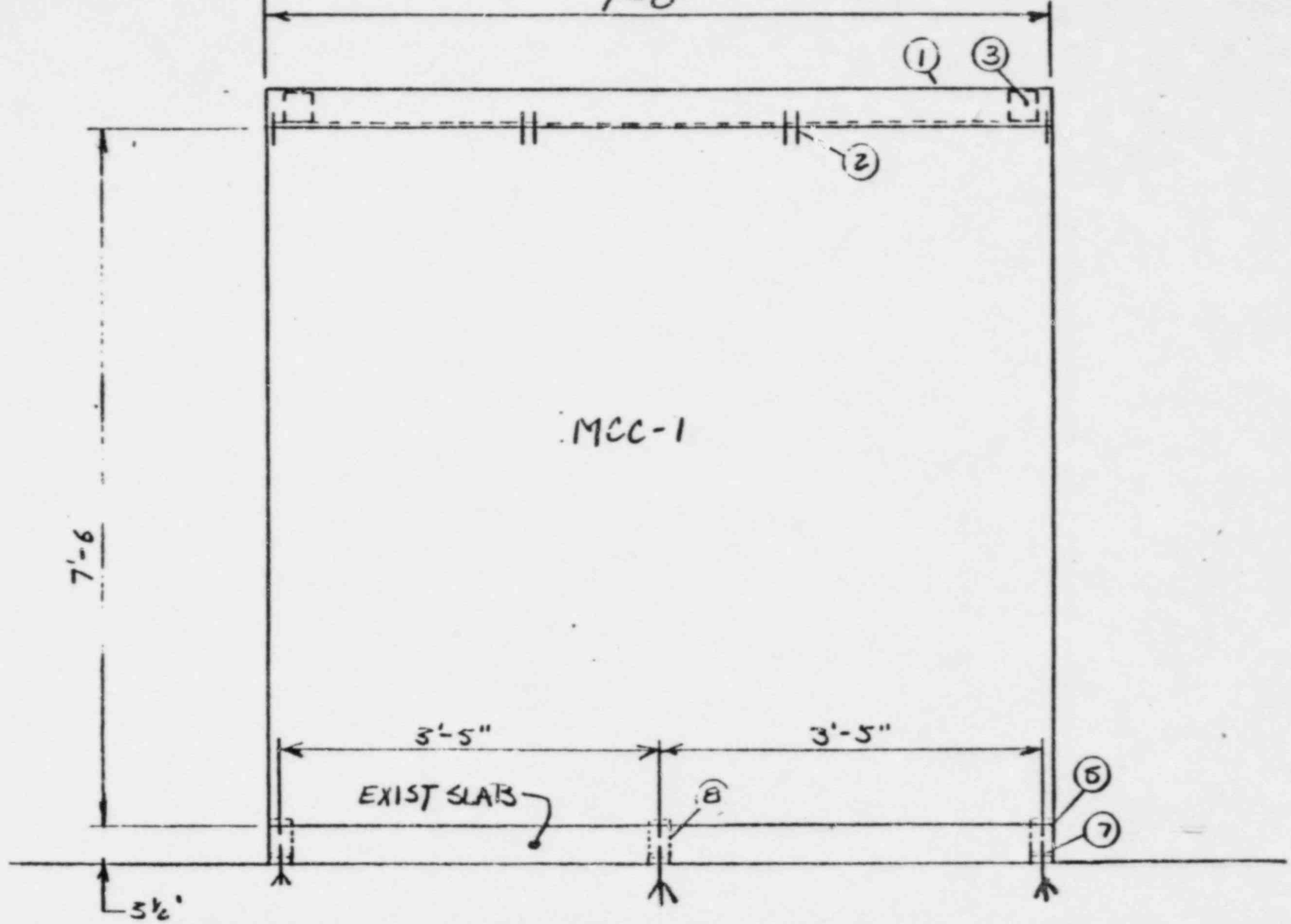
- structural tubing (see item 3 on Fig. 1-1)

TS 3 x 3 x 1/4 f_b = 1044 psi; F_b = ~~24~~ 32,000 psi
f_a = 82 psi; F_a = 19,940 psi



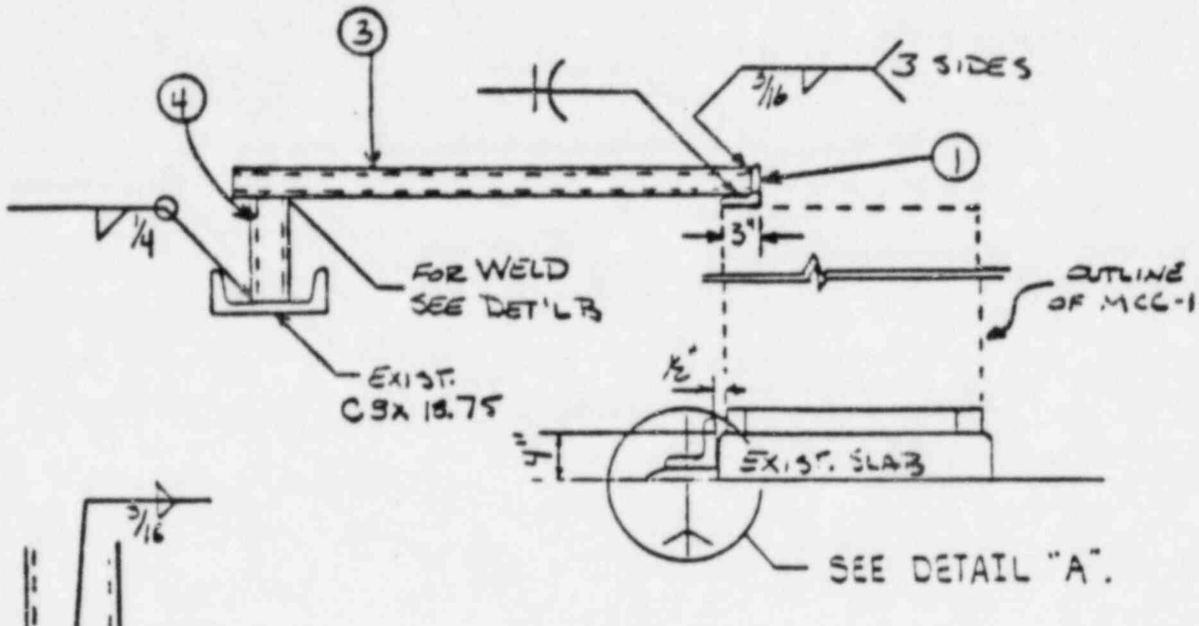
PLAN
SCALE 1"=1'-0"

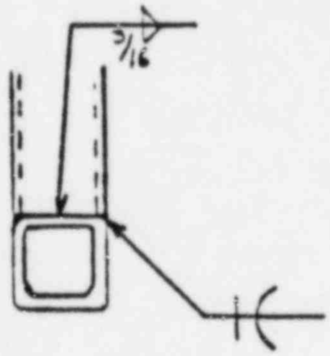
Figure 1-1



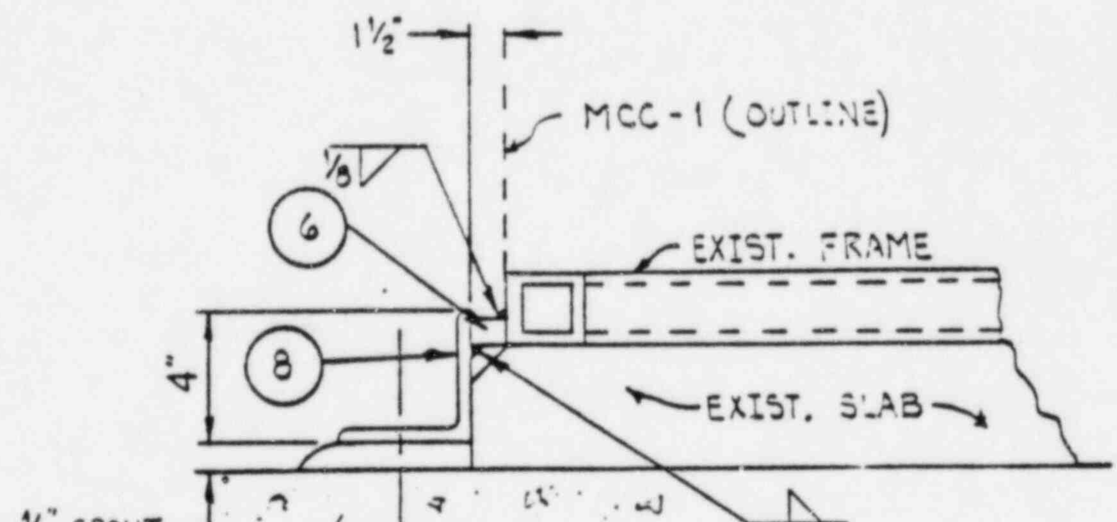
"1-1"
SCALE 3/4" = 1'-0"

Figure 1-2





DETAIL B



DETAIL "A"

Figure 1-3

LIST OF MATERIALS – SEISMIC SUPPORT FOR MCC-1

ITEM No	No REQ'D	DESCRIPTION	REMARKS
1	1	L 4 X 3 X 1/4 X 7'-0" LNG.	A-36
2	8	1/2" ∅ X 2" LNG HEX HEAD BOLT W/NUTS	A-307
3	2	TS 3 X 3 X 1/4 X 2'-8 3/8" LNG	
4	2	TS 3 X 3 X 1/4 X 0'-10 1/2" LNG	
5	2	L 4 X 3 X 3/8 X 0'-2" LNG	A-36
6	3	BAR 3/8 X 1 1/2 X 2	A-36
7	3	1/2" ∅ X 5 1/2' LNG HILTI KWIK-BOLT	2 1/4' MIN. EMB'NT
8	3	L 3 X 4 X 3/8 X 0'-2" LNG	A-36

Figure 1-4

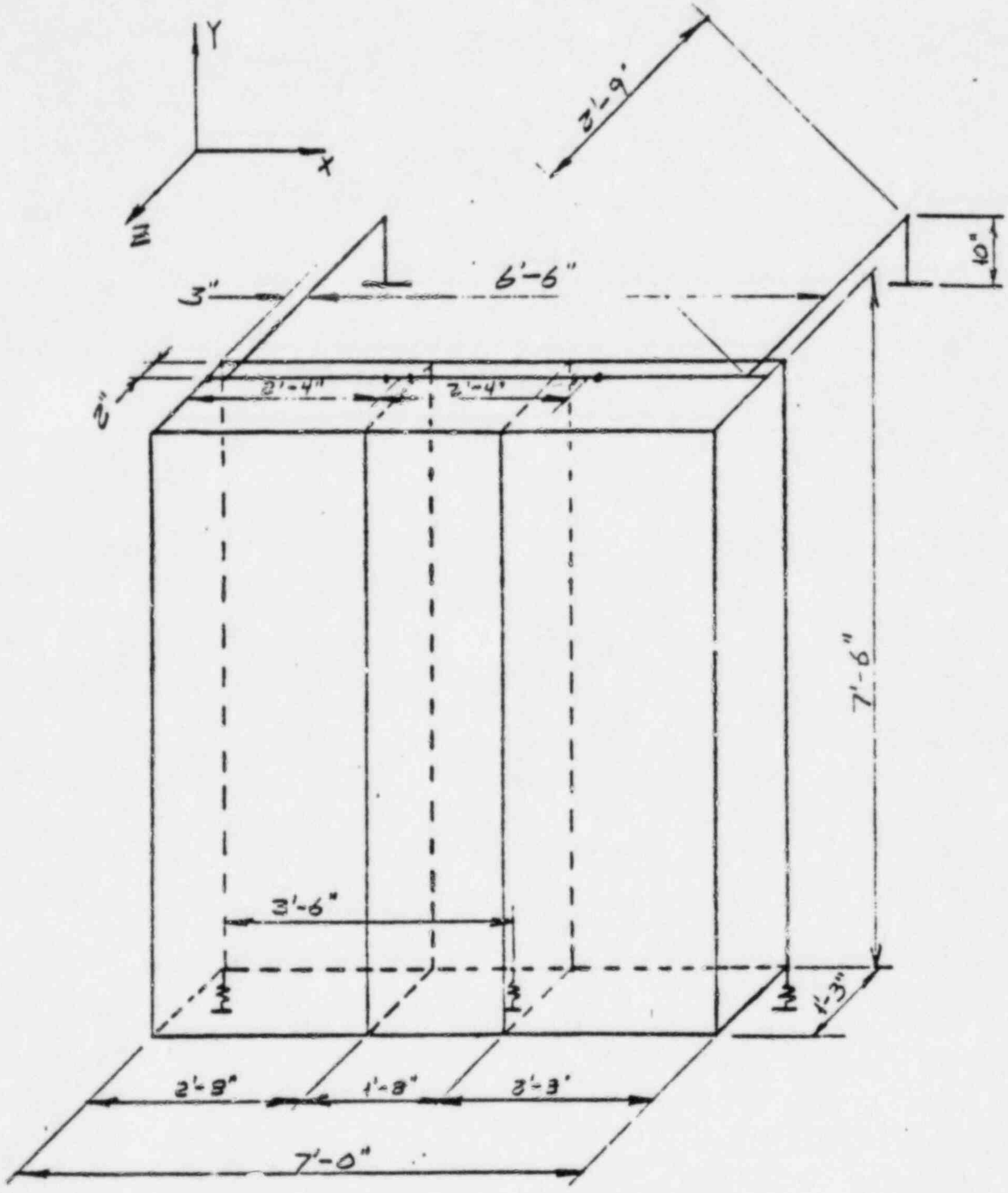
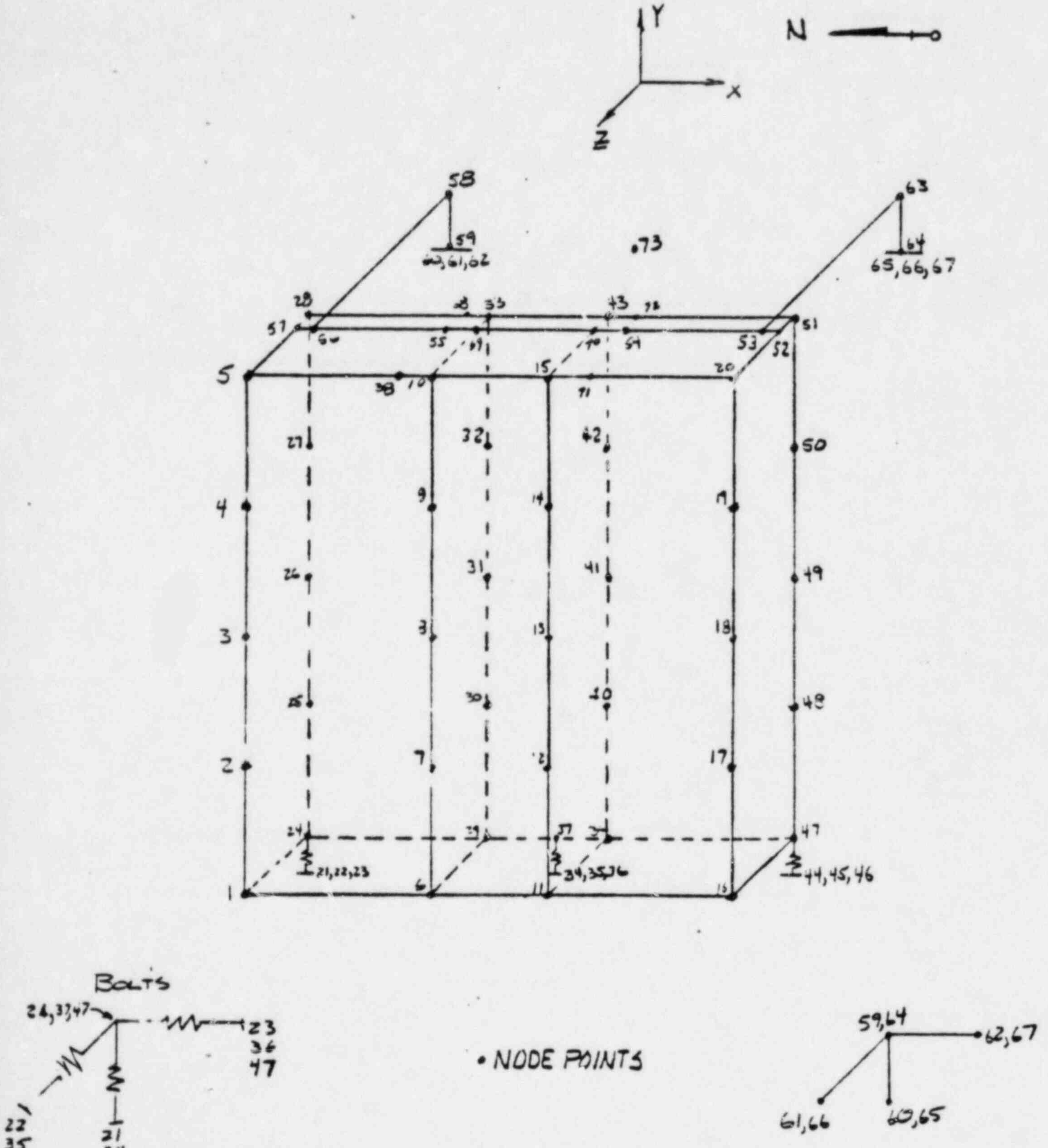
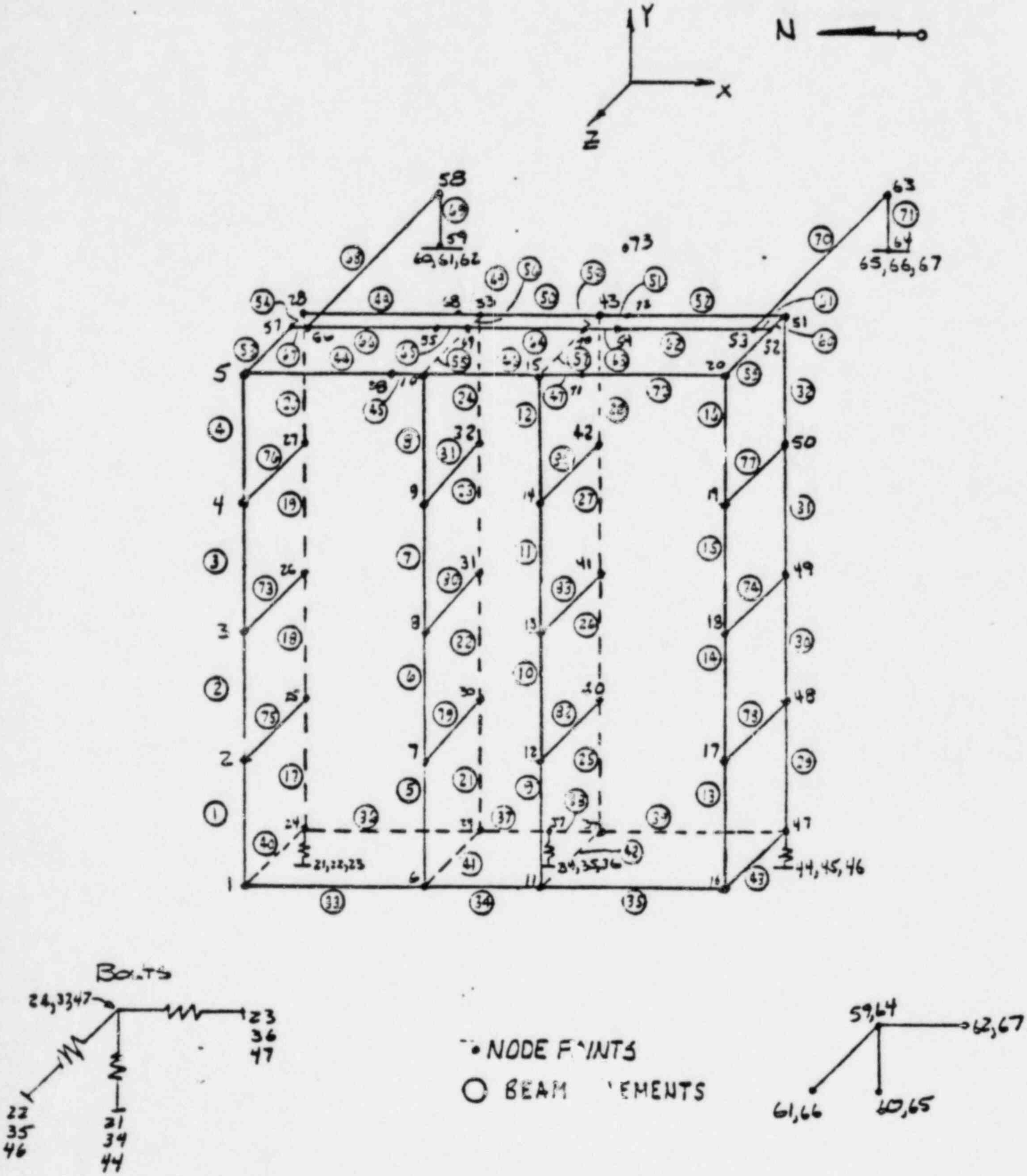


Figure 1-5



Y
X
Z
N
58
59
60,61,62
.73
63
64
65,66,67
BOLTS
24,37,47
23
36
47
22
35
46
21
34
44
21,22,23
34,35,36
44,45,46
• NODE POINTS
59,64
62,67
61,66
60,65

Figure 1-6


Y
X
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N
58
59
60,61,62
63
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65,66,67
21,22,23
34,35,36
44,45,46
BOLTS
24,33,47
23
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59,64
62,67
61,66
60,65

Figure 1-7

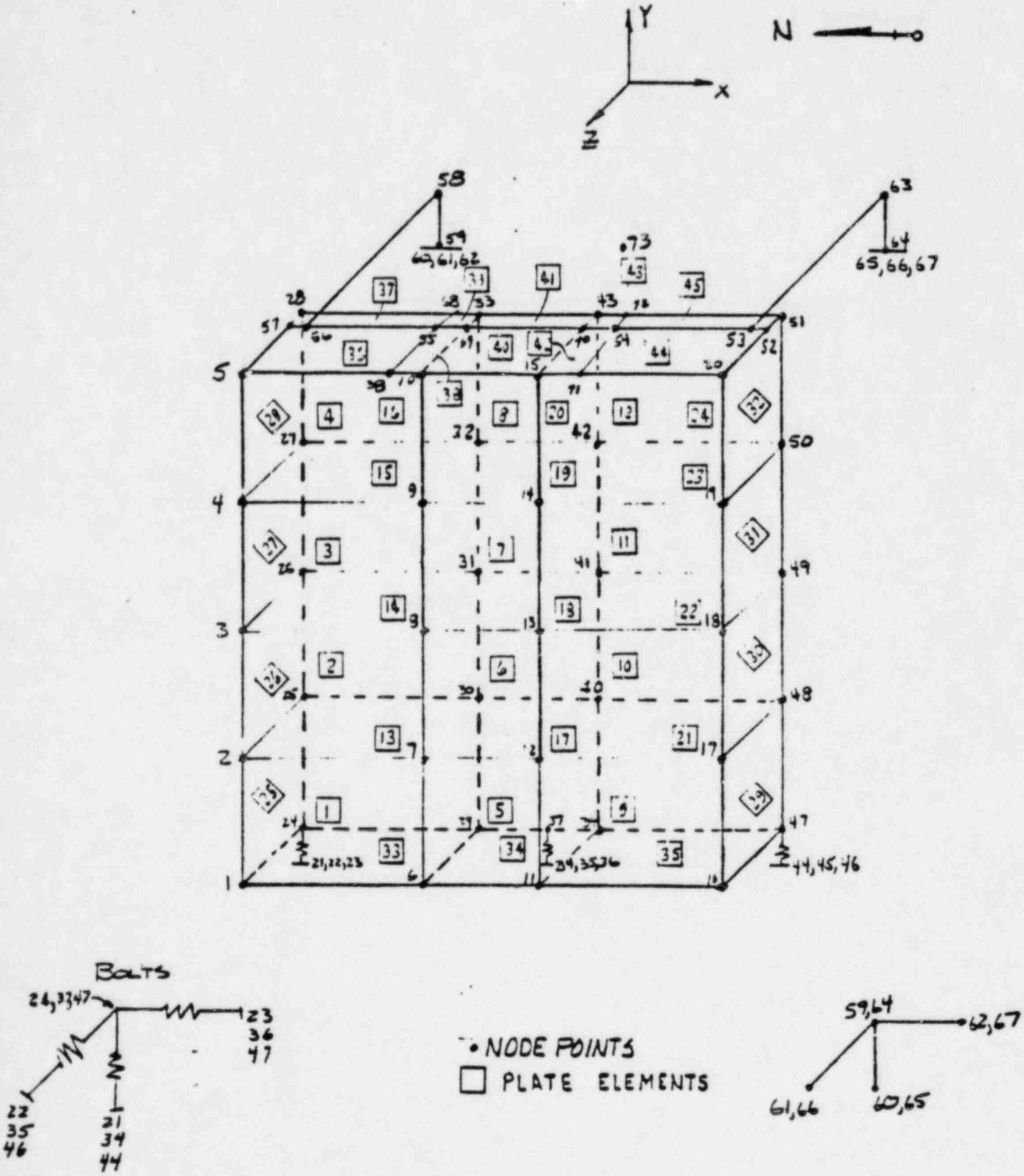


Figure 1-8

(TEST) MODE SHAPE 1

FREQ.= 17.15 Hz.

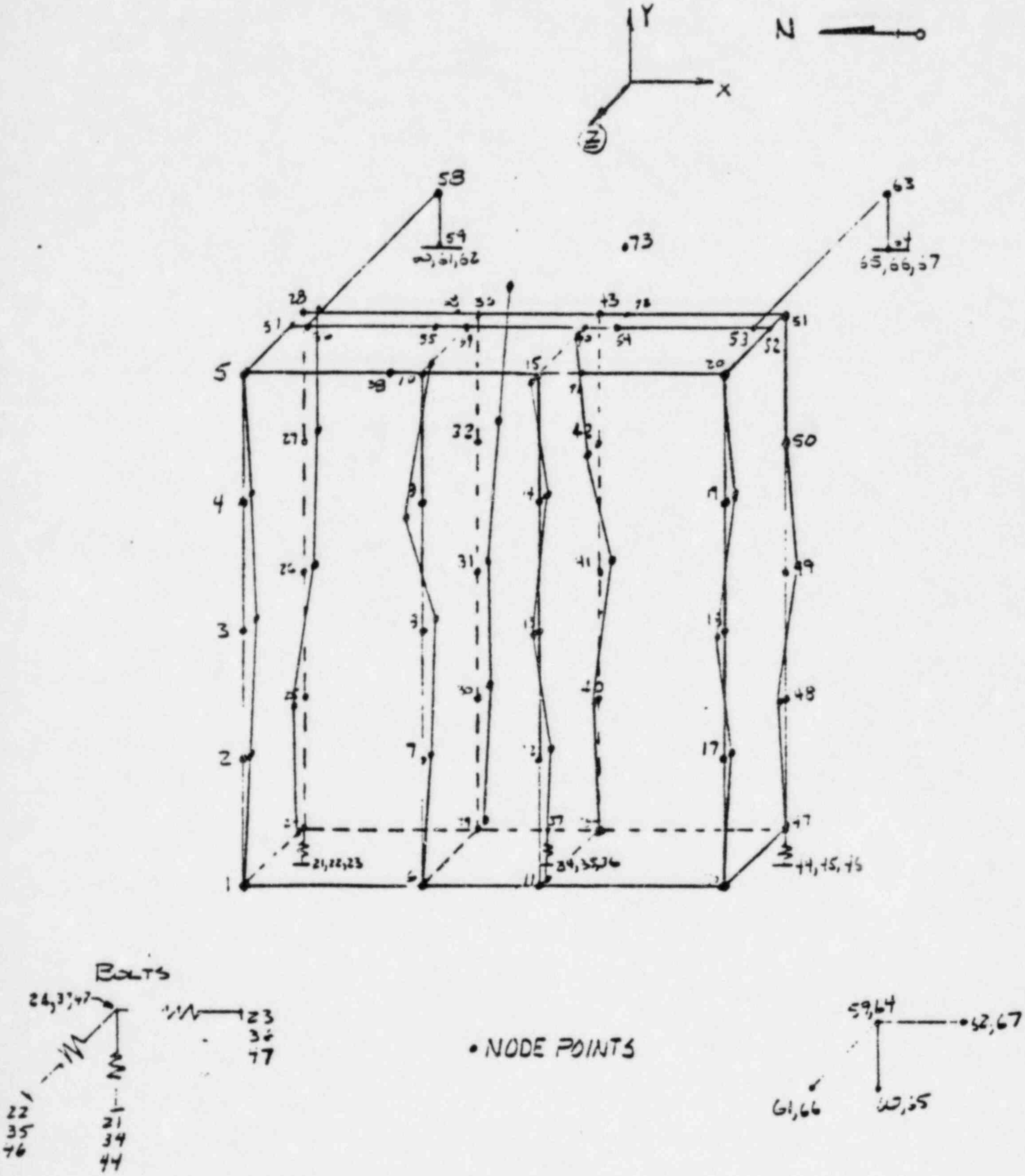


Figure 2-1

(TEST) MODE SHAPE 2

FREQ.= 23.6 Hz.

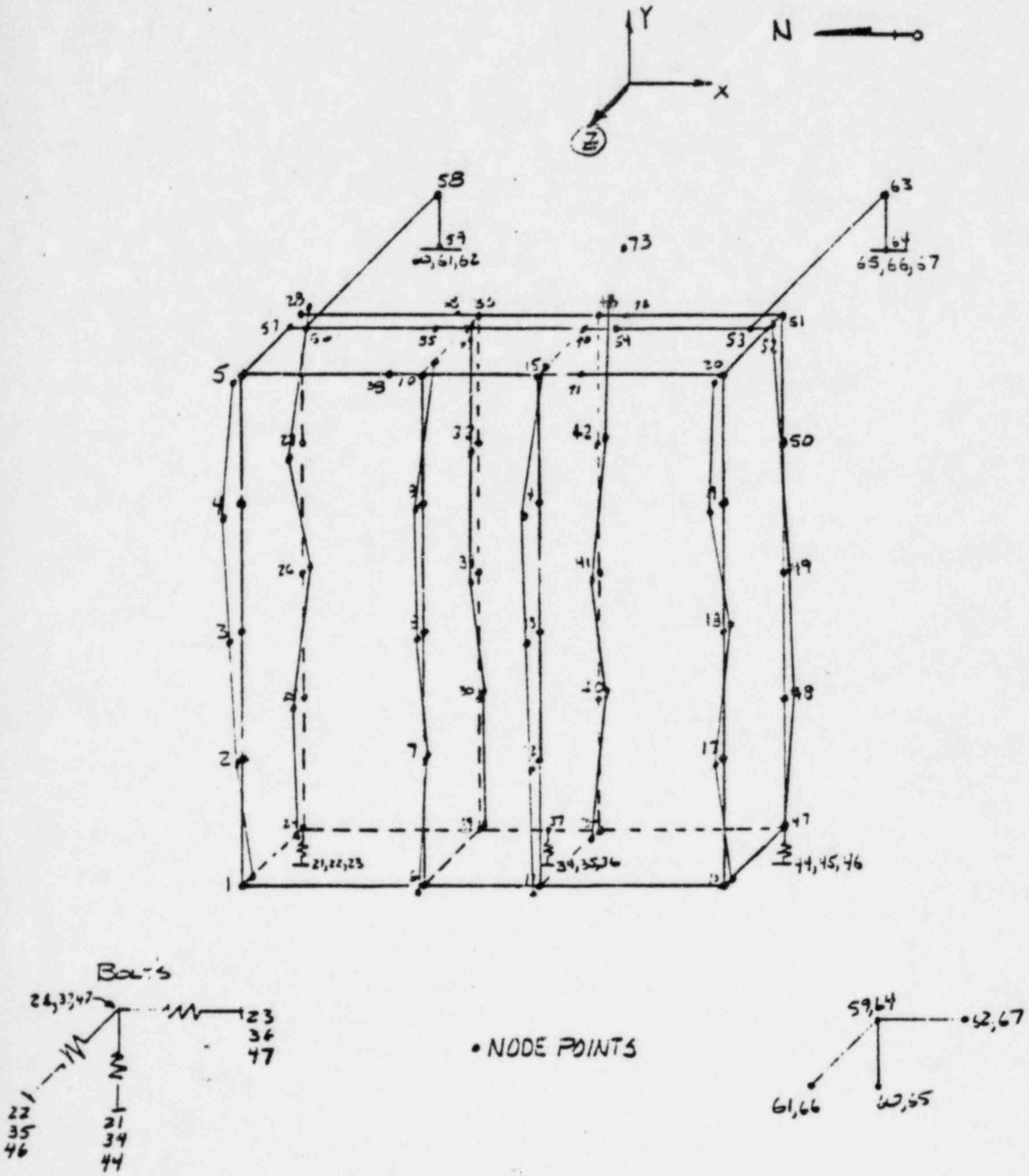


Figure 2-2

(MODEL) MODE SHAPE 1

FREQ.= 19.75 Hz.

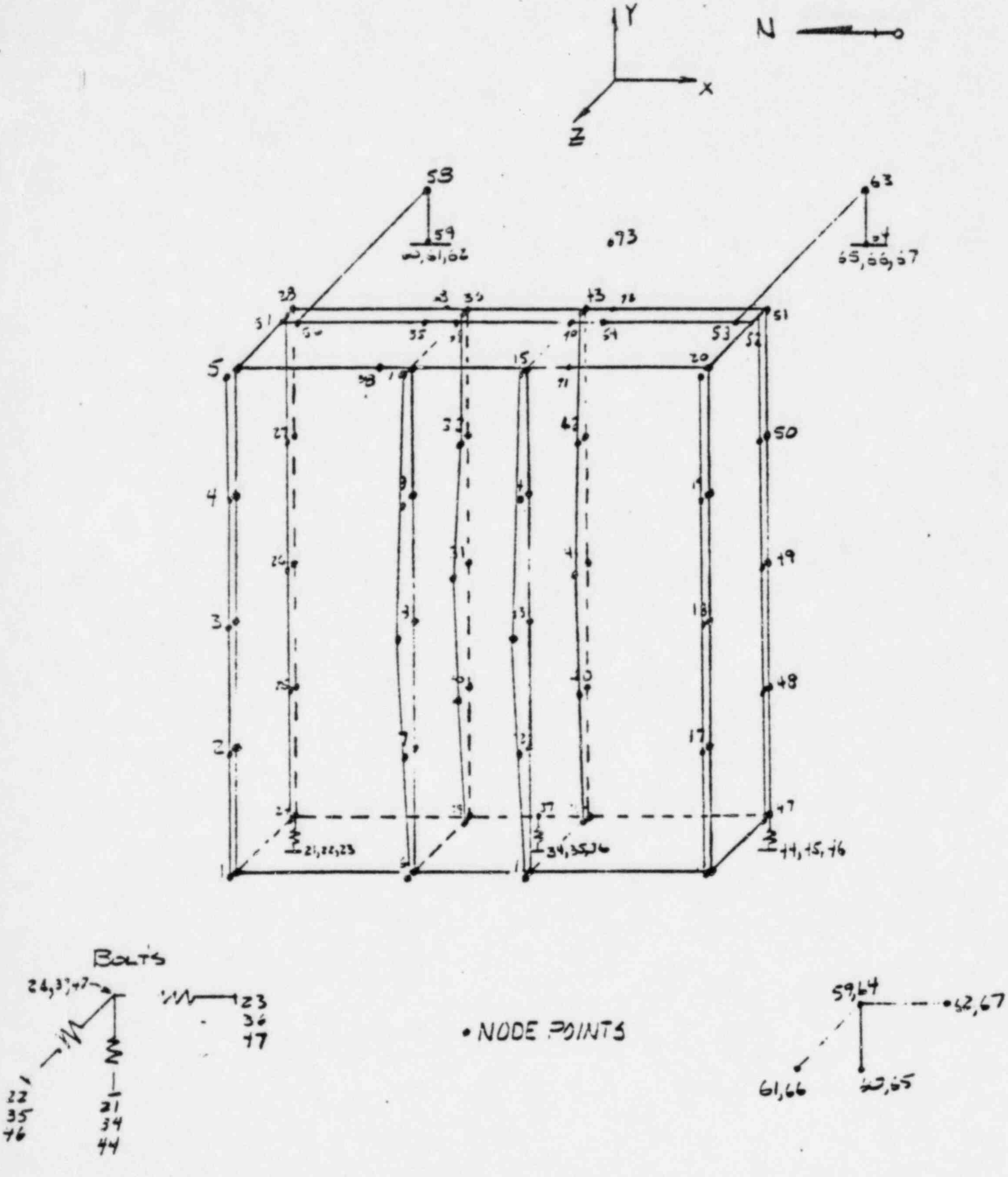


Figure 2-3

(MODEL) MODE SHAPE 2

FREQ. = 20.86 Hz.

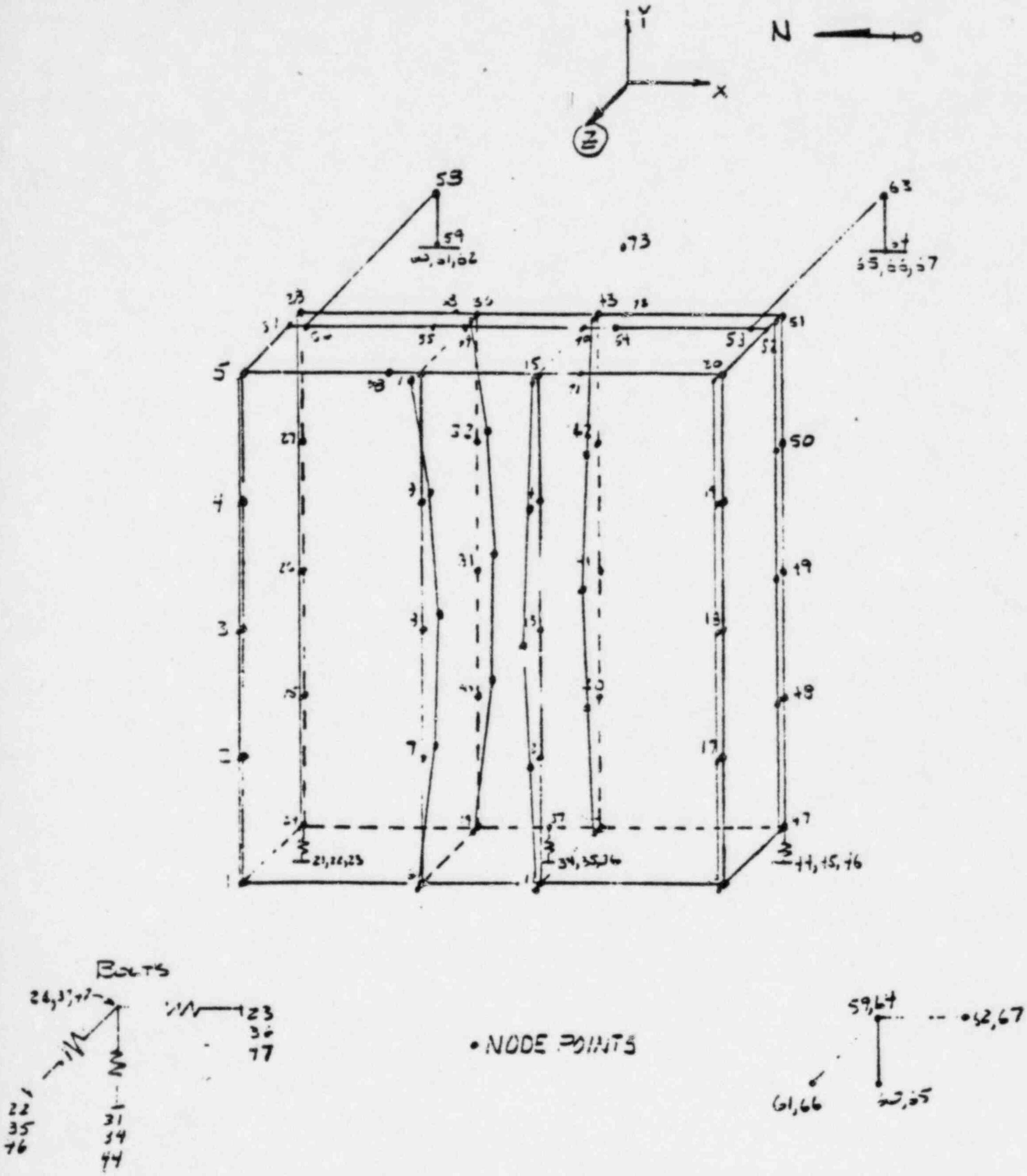


Figure 2-4

Applying the interaction formula from the ASME Code, Appendix XVII, for biaxial bending and axial compressive stresses yields $0.1 \leq 1.0$, and the safety factor equals 10.

(2) Welded Connection:

3/16" fillet weld between the horizontal TS 3 x 3 x 1/4 and the vertical TS 3 x 3 x 1/4

f_r = 989 psi; F_r = 21,000 psi

where:

f_r = resultant stress due to combined shear, bending, and torsion

F_r = allowable resultant stress taken as .3 F_u for E70xx electrodes

The safety factor is then equal to 21.

(3) Bolted Connection:

bolting of the ∠4 x 3 x 1/4, identified as item 1 in Figure 1-1, to MCC1

Shear f_v = 561 psi; $F_v = \frac{.62}{3} S_u \times 1.4$ = 16,780 psi

Tension f_t = 141 psi; $F_T = S_u/2 \times 1.4$ = 40,600 psi

Applying the interaction elipse formula found in Appendix XVII of the ASME Code yields $.001 \ll 1.0$ and the safety factor is approximately 1,000.

(4) Expansion Anchor:

1/2" ϕ Hilti expansion anchor bolts for 4,000 psi concrete with a minimum embedment of 2-1/4"

Shear f_v = 106 lbs F_v = 8316 lbs /4 = 2079

Tension f_t = 179 lbs F_t = 5510 lbs /4 = 1379

Using the interaction formula

$$IC = \frac{f_t \times f_p}{F_t/4} + \frac{f_v}{F_v/4} \times \frac{1}{f_r} \leq 1.0$$

with:

f_r = capacity reduction factor which accounts for bolt spacing and the distance to the free edge of concrete and is equal to 1.0

f_p = prying factor and is equal to 1.5

IC = $0.25 \leq 1.0$

Thus, an additional safety factor of 4 is provided.

While seismic loads and structural capacity of the MCC1 cabinet are determined primarily by the adequacy of the unistrut frame, the cabinet side plate tends to act as a shear beam having a major stiffening effect on the frame. This significantly increases the frequency of the cabinet and, thereby, reduces the applied seismic inertia loads.

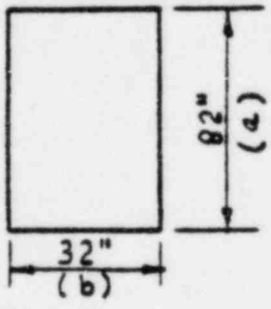
For this reason the potential for the side plates of the MCC1 cabinet to buckle were investigated. Should such buckling occur the fundamental frequency of the cabinet would be dramatically reduced and resultant seismi inertial loads increased substantially.

The following assumptions were made:

- The plates are fastened to the frame by screw fasteners located at the various node points
- The plates are 1/16" thick, rectangular, perfectly flat, and isotropic
- The plates are simply supported and subjected to loads in its plane

It was determined that the most limiting section was along the front and back portions of the cabinet because that was the longest unsupported length.

The USS Steel Design Manual, 1968 edition, was used to evaluate the elastic buckling stresses for the plate in the following manner:



- For pure shear:

$$\tau_{cr} = \frac{k_s \pi^2 E}{12(1-\mu^2)(b/t)^2}$$

where:

k_s = a nondimensional plate buckling coefficient depending on the type of edge support (simply supported) and the length to width ($^a/b$) ratio.

From Figure 4.6, page 80, k_s = 6 and τ_{cr} = 614 psi.

- For uniform compression;

$$\sigma_{cr} = \frac{k_c \pi^2 E}{12(1-\mu^2)(^b/t)^2}$$

where:

k_c = a nondimensional plate buckling coefficient which depends on the type of edge support (simply supported) and the $^a/b$ ratio.

From Figure 4.2, page 73, k_c = 4 and σ_{cr} = 409 psi.

The largest stresses for the individual plate elements, 32" x 22", found in this portion were then applied to the overall dimensions of the plate, 82" x 32". This was a conservative approach and resulted in:

τ_{max} = 25 psi; τ_{all} = 2/3 τ_{cr} = 409 psi

σ_c = 24 psi; σ_{all} = 2/3 σ_{cr} = 273 psi

- For compression along one axis with shear

$$\frac{\tau}{\tau_{cr}} = \frac{25}{614} = .04$$

$$\frac{\sigma_c}{\sigma_{cr}} = \frac{24}{409} = .06$$

Referring to Figure 4.7, page 81, of the USS Design Manual the results are well within the interaction curve acceptance, hence, are acceptable.

3. LOAD CRITERIA

After the model was verified by comparing the frequencies and mode shapes to those that were established by the low impedance, insitu test, a response spectrum analysis was performed using the floor response spectral curves, shown in Figures 3-1, 3-2, and 3-3, located at elevation 21'-6" of the screenwell house for 7% damping.

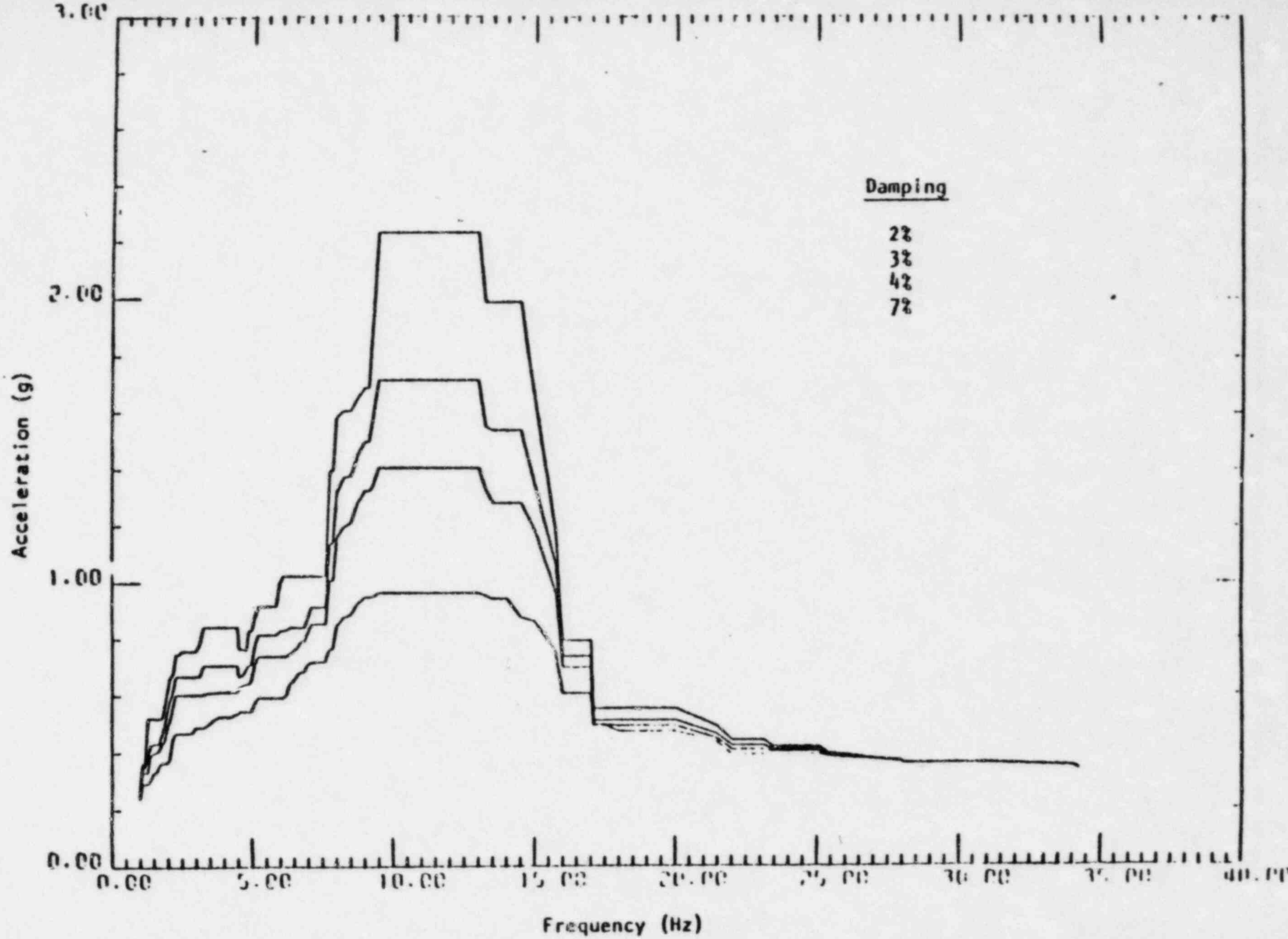


Figure 3-1 North-South Floor Response Spectra For Elevation 21'-6", Screenwell House, Connecticut Yankee Atomic Power Plant

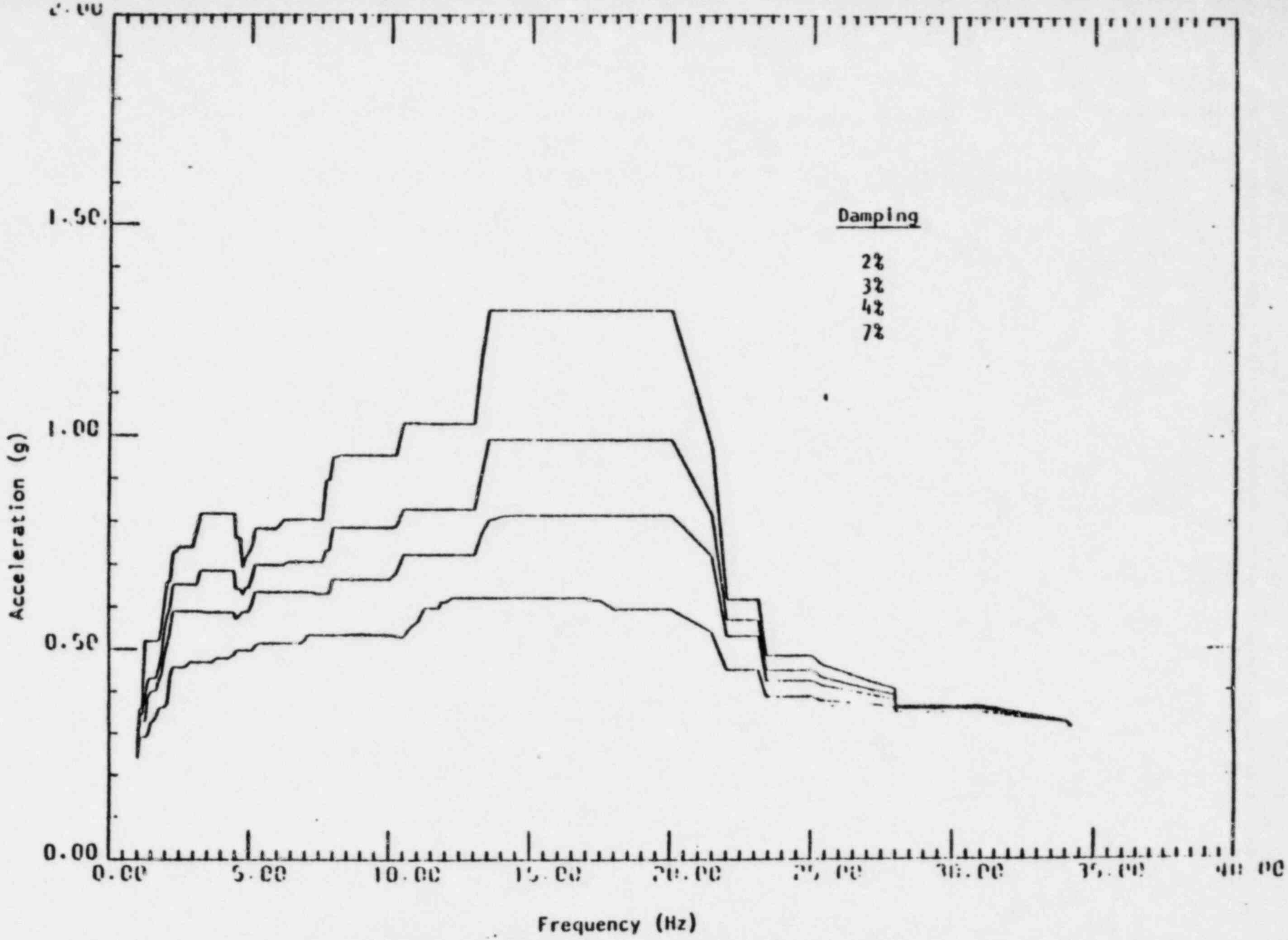


Figure 3-2 East-West Floor Response Spectra For Elevation 21'-6"
Screenwell House, Connecticut Yankee Atomic Power Plant

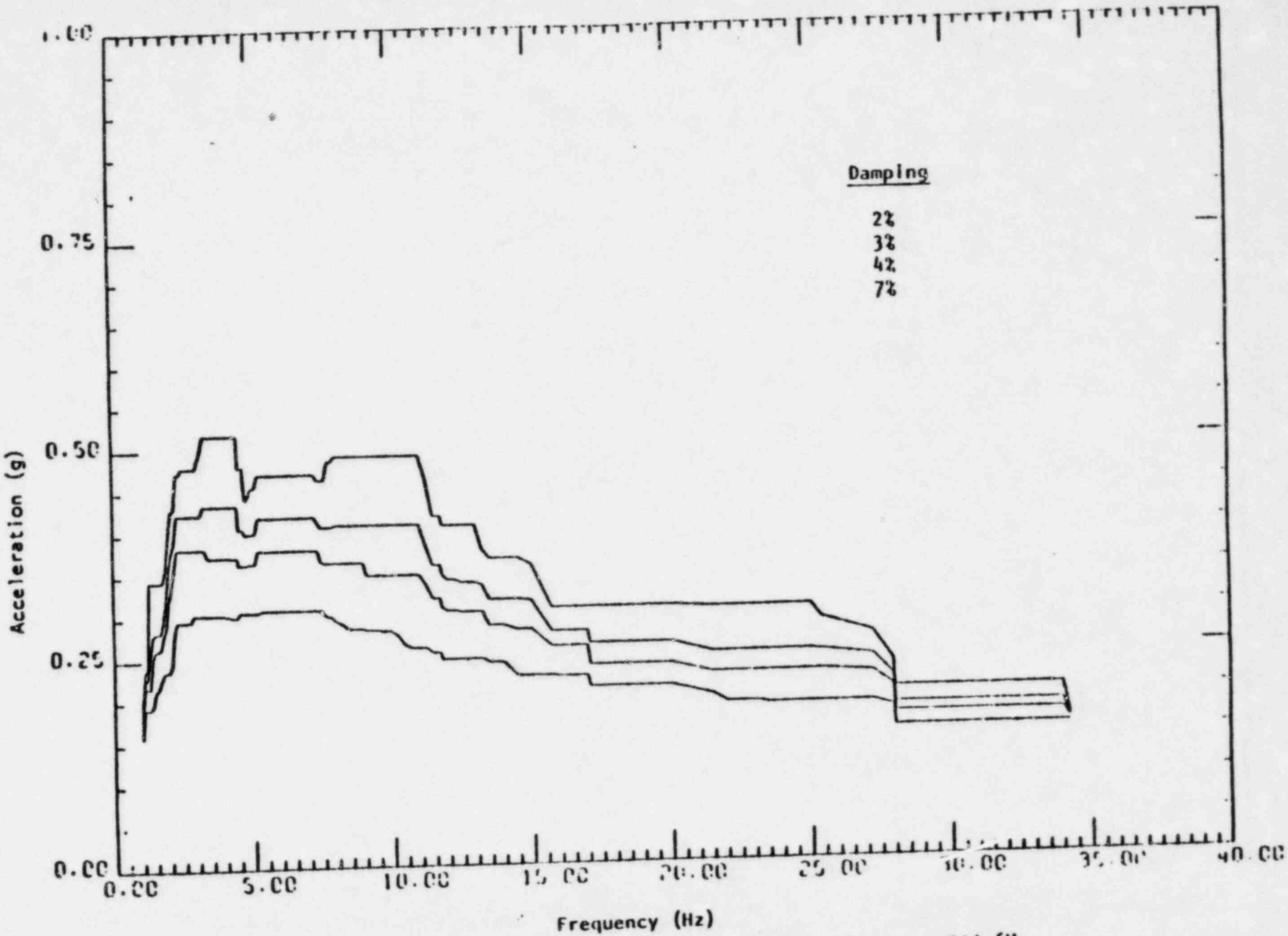


Figure 3-3 Vertical Floor Response Spectra For Elevation 21'-6", Screenwell House, Connecticut Yankee Atomic Power Plant

4. STRESS DEFORMATION - STABILITY CRITERIA

The MCC1 is a passive, P-1, component whose stress limit is defined as

$$S_{all} \leq S_y;$$

which was determined from the "Allowable Stress Criteria for the Haddam Neck Plant" attached hereto as Appendix A.

5. METHODS OF ANALYSIS

The MCC1 was analyzed using the response spectrum option of SAP IV. SAP IV is a general purpose, finite element program that includes both static and dynamic capabilities for a broad range of elements and has been used extensively in the nuclear power industry for analyzing structures and mechanical component responses to dynamic loads. The function of the computer program is to assemble the stiffness and mass matrices, reduce them according to the choice of dynamic degrees of freedom to a set of homogeneous equations, and extract their eigenvalues and eigenvectors. Then, the forcing function is applied and the response spectra analysis is performed.

Several assumptions have been made to facilitate the development and implementation of the model for analysis:

- The beam and plate members are modelled together with the plates joining the beam structure at every frame node and contribute to the stiffness of the structure.
- The additional mass of the cabinet has been lumped at the eight center frame nodes.
- The expansion anchor bolts are modelled as linear springs having both tension and compression capability as installed. This is an added conservatism as the floor and not just the bolts carry the compressive bearing loads. The effective anchor bolt length for the determination of bolt stiffness consists of the free length plus 5-bolt diameters depth into the floor.
- Where the structural tubing is welded to the existing channel section, it is considered to be a rigid connection.