ATTACHMENT 3

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August 1982

TECHNICAL EVALUATION OF THE YANKEE ROWE POWER STATION SEISMIC DESIGN

Applied Mechanics Branch
Engineering Analysis Division

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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. Yankee Atomic Electric Company (YAEC) personnel and their consultants analyzed the Yankee Rowe 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 YAEC and their consultants were generally acceptable with the exception of minor suggestions, comments, and questions. Some items were not completed and remain open items. The results were obtained through working level meetings with YAEC 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 Yankee Rowe Power Station 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 Yankee Atomic Electric Company (YAEC) 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 YAEC 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 meetings. The review team developed an acceptance criteria for guidance in evaluating these analyses.

This report was divided into individual sections covering the piping, electrical equipment, mechanical equipment, and component supports. These sections contain procedures utilized by YAEC 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 YAEC 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 Yankee Rowe Power Station, operated by the Yankee Atomic Electric Company (YAEC), is included with the Group II plants. A reanalysis was performed to demonstrate that the structural integrity of the safety related piping systems, mechanical equipment, electrical equipment, and component supports would not be impaired when subjected to a Safe Shutdown Earthquake (SSE) combined with other normal design loadings.

The Yankee Rowe Power Station is a pressurized light water moderated and cooled system. The plant initially produced 485 MW of heat and 136 MW of gross electric power. The plant was designed to produce 110 MW of gross electric power and 392 MW of heat. The reactor vessel and equipment were analyzed to the ASME Code, Section VIII. 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, Yankee Atomic Electric Company (YAEC) 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 for some systems. 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 YAEC 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 o 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 Yankee Rowe 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 other systems directly connected to RCS up to and including isolation valves
e) Control Rod Drives
f) Auxiliary Feedwater System
g) Shutdown Cooling System
h) Portions of the Chemical and Volume Control System
i) Portions of the Component Cooling Water System
j) Portions of the Service Water System
k) Portions of the Pressure Control and Relief
l) Accumulators
m) High Pressure Safety Injection System
n) Low Pressure Safety Injection System

o) Containment Air Recirculation (Post Accident) Fans

p) 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 piping, electrical equipment, mechanical equipment, and component supports. Each section explains in detail YAEC'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 YAEC or their consultants. The review team's conclusions were based upon the presentations and documents provided by YAEC and their consultants.

PIPING SYSTEMS

Licensee Evaluations

YAEC's consultant performed the analyses required for the safety related piping systems of the Yankee Rowe Power Station. The analyses were performed in accordance with their piping stress analysis procedures (Appendix B). The safety related piping systems (defined in Appendix B) analyzed for SEP were:

1. Main Steam
2. Feed Water
3. Reactor (Main) Coolant
4. Pressure Control & Relief
5. Charging & Volume Control
6. Safety Injection
7. Shut Down Coolant
8. Sample and Drain System
9. Primary Plant Purification
10. Emergency Feedwater System
11. Vapor Containment Heating System
12. Control Rod Drive
13. Emergency Core Cooling
14. Component Cooling System
15. Service Water System.

The piping stress analysis procedure provided a guide for modeling wall and floor penetrations, valves, flanged joints, branches, and anchors.

YAEC's approach to modeling single acting restraints required the analyst to examine each of these supports individually. When the uplift due to seismic load (plus the thermal load if it is upward) is larger than 90% of the weight load, the single acting restraint was assumed to be ineffective. The single acting restraints that were considered ineffective were not included in the seismic analyses.

Wall and floor penetrations were examined on an individual basis. If the penetration was not grouted, a deflection of 1/4 in. was considered acceptable and the penetration was not treated as a support. If the combined thermal and seismic deflections were greater than 1/4 in., further review as to whether or not this penetration should be considered a restraint was required. Grouted penetrations were considered either a bilateral restraint or a trilateral restraint. The penetration was considered to have axial restraint if a welded collar imbedded into the concrete was attached to the pipe.

Each problem shall be considered from anchor to anchor. YAEC 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 25
3. An anchor; i.e., a six way restraint.

If the anchor to anchor problem exceeded program limitations, the following modeling approaches were considered:

1. Overlapping such that negligible migration of loads from one problem to another existed
2. Bracketing results of multiple computer runs to assess boundary conditions or loading conditions.

YAEC assumed the thickness of their valve bodies were twice the connecting pipe wall thickness. Motor and air operated valves were modeled with their eccentric masses concentrated at a point 1/3 the distance between the centerline of the operator and valve body. Manually operated valves were modeled with their mass concentrated at the centerline of the pipe.

The stress int nsification factors for nonstandard fittings were considered as pe e Code requirements of ANSI B31.1 - Power Piping, 1980 Edition.

If the moment of inertia of a run line was 25 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. The main line deflections and rotations were input as anchor movements for the branch line analysis.

Other modeling techniques found in the piping stress analysis procedure were also utilized. These methods were also reviewed. However, the piping stress analysis procedure did not indicate the modeling tecnniques utilized for rigid restraints, spring hangers, and one-way restraints.

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

1. Design pressure plus weight plus temperature
2. Safe Shutdown Earthquake (SSE) plus maximum operating pressure plus weight.

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, 1977 Edition; U.S. Nuclear Regulatory Guides 1.60, 1.61, 1.92, 1.122; the ASME Boiler and Pressure Vessel Code, 1971 Edition; and U.S. Nuclear Regulatory Guide, NUREG/CR-0098, May 1978. The loading combination and stress limits utilized by YAEC are summarized in Appendix B.

Most of the seismic analyses were performed using lumped mass dynamic models with the appropriate floor response spectra. The percent of critical damping utilized was in accordance with Regulatory Guide 1.61. 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. Seismic inertia analyses and seismic anchor movement analyses were performed for the SSE. Modal responses for each component of the earthquake were combined by considering closely spaced modes in accordance with Regulatory Guide 1.92. A cut-off frequency of 33 cps and no less than ten modes were considered in the analyses. The computer codes utilized for the piping system analyses are listed in Appendix B.

For small piping systems (2 in. and less), YAEC felt a simplified stress analysis was sufficient where engineering judgement deemed necessary. This method analyzed each span of piping in the system as a simply supported beam. Thermal, weight and seismic analyses were performed on the span. Details of this simplified stress analysis are contained in Appendix B.

YAEC's piping system analysis results are presented in the tables of Appendix B. YAEC was required to present a final report of the piping results and modifications required to the piping systems to the NRC.

Review Team Evaluations

The Acceptance Criteria for Piping provided by the NRC review team is contained in Appendix C. 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 applied by YAEC in their piping reanalyses are acceptable. The modeling techniques utilized by YAEC provide a complete and practical representation of the piping system. Uplift on the supports was considered; however, support impact loading was not evaluated. The percent of damping utilized by YAEC for the response spectra was in accordance with Regulatory Guide 1.61. The masspoint spacing utilized by YAEC was considered acceptable.

Several YAEC piping system analyses were reviewed in detail at one of the working level meetings. The lines reviewed were the pressure control and relief line, Problem 41B; the main coolant piping, Problem 102; and the main steam line, Problem 2. All of these lines were analyzed using the response spectra method and the review team considered these analyses acceptable. The pressure control and relief line and the main coolant line did not require modifications. However, the main steam line requires modifications to bring this line to a level acceptable for the SEP. A complete report of the piping system analyses results including the required modifications should be submitted to the NRC.

YAEC performed two sets of analyses for the safety related piping systems. Seismic (SSE) analyses were performed using both the Yankee composite spectra and the NRC spectra. Both sets of analyses were performed using Class 2 procedures for all of their piping. An allowable stress of 1.8 S_h was used when evaluating piping that utilized the Yankee composite spectra. This is consistent with the SEP reevaluation guideline criteria for Class 1 piping. An allowable stress of 2.4 S_h was used when evaluating the piping that utilized the NRC spectra. This is consistent with SEP requirements for Class 2 piping only. The results of Class 1 piping analyses using NRC spectra were reviewed and it was determined that calculated stresses in excess of 1.8 S_h occur for only one component. The component was a reducer on the main coolant line near the reactor pressure vessel. The calculated stress for this reducer exceeds 1.8 S_h by only 6%. This calculated stress was based on a conservative computer code default stress intensification factor. A more realistic value would easily put the calculated stress within the Class 1 allowable of 1.8 S_h. Therefore, it was concluded that the Yankee Rowe Plant's Class 1 piping results are within the allowables specified in the SEP reevaluation guideline (Appendix C).

ELECTRICAL EQUIPMENT

At the working level meetings, reevaluation of electrical equipment was presented by the licensee only with regard to anchorage of that equipment. Their program was outlined but no analysis has been done to date. Their plan was to use existing earthquake data and similarity of equipment to that not damaged in previous earthquakes as being sufficient to demonstrate acceptability of the Yankee installed equipment. That plan is insufficient. Although these arguments may be used as support information, additional justification is required. In addition to demonstrating that the equipment is adequately supported, the licensee must also demonstrate the structural integrity of the equipment. After some discussion, the licensee agreed to perform some analysis in development of their conclusions. These plans have not been finalized by the licensee so the entire area of reevaluation of electrical equipment has been inadequately addressed to date. This remains a major open item for the Yankee Plant.

MECHANICAL EQUIPMENT

At the working level meetings, an overall screening criteria for the acceptance of mechanical equipment was presented by the licensee. This plan was based on a linear interaction formula with nozzle reaction shear and moment effects included. However, the possible axial effect due to pressure loading was not included. The licensee's consultant will investigate the need for doing this. Equipment to be reevaluated includes the reactor pressure vessel (RPV), steam generators, main coolant pumps, and pressurizer. Analyses of the RPV, steam generators, and pressurizer addressed only the nozzle loads and support structures for these components. The components themselves were not reanalyzed. Details of the analyses procedures were not presented. For the components, an in-service "g" (acceleration) level will be determined and compared to allowables established by the vendor. Preliminary results were presented for the Yankee Composite Spectra (YCS) and the NRC Spectra. Nozzle load ratios were over the proposed interaction formula allowables for several cases. However, these analyses are preliminary so acceptability of equipment reevaluation for the SEP is an open item pending final results and review of reports pertaining to the reevaluation plan. In addition to analyzing components for nozzle loads, structural integrity of the components must also be demonstrated. The licensee's plan for implementing this requirement has not been addressed to date. Therefore, this area remains an open item for the mechanical equipment reevaluation.

COMPONENT SUPPORTS

Licensee's Evaluation

The licensee's consultant performed the reevaluation analyses for the following hot shutdown mechanical equipment supports: the reactor vessel ring support, steam generator supports, and pressurizer supports. The objective for performing these analyses was to demonstrate structural adequacy of these component supports when subjected to combined normal design loads plus SSE loading. These analyses were performed utilizing two and three dimensional, lumped mass finite element computer models to represent the mechanical components and their support structures. The seismic input acceleration spectra used to perform these analyses were the NRC spectra. None of these analyses were completed; however, some preliminary results are contained in Appendix D.

The reactor ring support (RRS) analysis was performed by first developing a three dimensional plate element model of the RRS. This model was used to determine the stiffnesses and natural frequencies of the RRS. A two dimensional model was then developed representing the reactor vessel and contents, the neutron shield tank and contents, the RRS, and the reactor coolant system piping stiffness and mass effects at the reactor vessel nozzles. Using this model, reaction loads were determined for normal design loading (static) and SSE loading (dynamic). The dynamic portion of this analysis was performed by the response spectra method. The reactor loads were used to determine if sliding or rocking of the reactor supports occurs. In addition, these reaction loads were used to evaluate the RRS using the three dimensional model of the RRS. From this analysis it was concluded that neither sliding nor rocking of the reactor or RRS would occur and that the RRS can withstand seismic loads as specified by the NRC spectra.

An analysis of the steam generator and steam generator support frame will be performed using a three dimensional, lumped mass finite element model representation. This analysis will be performed using the computer code ANSYS. This analysis is not complete. Preliminary results indicated that the support anchorage bolts cannot adequately withstand the effects of SSE (NRC spectra) loading in combination with normal design loadings.

Seismic reevaluation analyses will be performed for all hot shutdown piping supports by the licensee's consultant. At the latest working level meeting for Yankee Rowe Power Station, a presentation on the seismic pipe support reevaluation was given. Appendix D contains copies of overhead slides used for that presentation. These analyses of the pipe supports will be performed for normal design loading (including thermal anchor movement effects) and normal design loading combined with SSE loading (including seismic anchor movement effects). The allowable stress criteria utilized to evaluate the pipe supports was generally based upon the requirements of the ASME Code, Subsection NF. The analyses performed addressed the evaluation of adequate support stiffnesses for rigid supports, the evaluation of support gaps, and the evaluation of spring hanger travel.

Review Team Evaluation

The licensee's component support seismic reevaluation program for mechanical equipment is incomplete. Based on the licensee's presentation at the latest working level meeting, the licensee primarily addressed anchorage of the mechanical equipment being reevaluated. The presentation did not include a discussion of evaluation of support structures. The computer models of the equipment and support structures appeared to provide an adequate mathematical representation of the equipment and support structures. The reevaluation seismic analyses were performed utilizing the proper loading of normal design in combination with SSE (NRC spectra) loadings.

Since final analyses for the mechanical equipment component supports were not available for detailed review, it is recommended that these analyses be further reviewed when they become available. The following open items can then be resolved.

1. What computer codes were used to perform these analyses?

2. What value of damping was used?

3. What was the allowable stress criteria for the components, component support structures and the anchorage system?

4. Where did the maximum stresses occur and what were their magnitudes?

5. What effect do the equipment support reactions have on the mechanical equipment?

In general, the piping support seismic reevaluations and the allowable stress criteria for the Yankee Rowe Power Station were acceptable. The load combinations evaluated were consistent with SEP requirements in that both normal and faulted conditions were evaluated. Impact loading as a result of uplift during a postulated SSE event was not addressed for one-way supports. Evaluation of this loading condition is required. The allowable stress criteria generally follows the requirements of Subsection NF of the ASME Code. Criteria for anchorage and buckling were also adequately addressed and consistent with the SEP allowable stress criteria contained in Appendix C. There were, however, some questions concerning the allowable stress criteria presented at the latest working level audit meeting. It appeared that the allowable stress criteria contained a typographical error for the allowable shear strength of weld metal. The allowable shear strength for fillet weld metal should be a factor of 0.30 times the nominal tensile strength of the weld metal. For full and partial penetration groove welds, the allowable tensile stress should be expressed in terms of both the nominal tensile strength of weld metal and the yield strength of the base metal. Care must be taken in expressing the allowable strength of catalog items for faulted conditions in terms of catalog values. In some cases, the catalog values may be listed for faulted conditions and should not be increased. This evaluation of the piping support reevaluation analyses was based solely on the latest working level meeting presentations. Detailed review of final support analyses were not performed. It is recommended that a detailed review of typical pipe supports analyses be performed when they become available.

CONCLUSIONS

Piping Systems

YAEC 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 YAEC's overall analysis techniques and piping criteria are reasonable. However, the review team's acceptance criteria was modified since the meeting with YAEC. YAEC should submit all remaining analyses results to the NRC.

Electrical Equipment

The program plan for reevaluation of electrical equipment was inadequately addressed in the working meeting presentations. The licensee has not finalized their reevaluation plan for electrical equipment so acceptability of the electrical equipment reevaluation for the SEP is an open item pending final results and review of reports pertaining to this equipment.

Mechanical Equipment

Because the presentations for mechanical equipment were based on preliminary analyses with no reports available for audit checks, acceptability of the mechanical equipment reevaluation for the SEP is an open item pending final results and review of reports pertaining to these pieces of equipment. The screening criteria for selecting equipment to be reanalyzed in detail is acceptable; however, the final analysis of components must also address the structural adequacy of the various components, not just the acceptability of the nozzle loads on the components.

Component Supports

Seismic reevaluation of the piping supports for the Yankee Rowe Power Station hot shutdown piping systems as presented by the licensee's consultant at the latest working level audit meeting, in general, appear to be adequate.

The piping supports were evaluated using the proper load combinations, however, one-way pipe supports were not evaluated for impact loading resulting from uplift during a seismic event. The allowable stress criteria was adequate in that they are generally consistent with the allowable stress criteria developed for the SEP Group II Plants contained in Appendix C. A couple of concerns with the allowable stress criteria were expressed in the component support review team evaluation section of this report. A detailed review of the final pipe support analyses should be performed when they become available.

The mechanical equipment component support analyses performed for the Yankee Rowe Power Station appear to be incomplete based on the presentation at the latest working level audit meeting. Seismic reevaluation of these component supports appear to be only addressing anchorage of the equipment. Evaluation of the entire component support is required. This includes the effects of the support reaction loads on the components. Since these component support analyses were not finalized at the audit meetings, a detailed review of these analyses was not performed. When these analyses are completed, it is recommended that they be submitted to NRC for a detailed review.

APPENDIX A

YANKEE ROWE AUDIT PLAN FOR SEP SEISMIC QUALIFICATION OF PIPING, MECHANICAL, AND ELECTRICAL EQUIPMENT

YANKEE ROWE 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 Yankee Rowe 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 Yankee Rowe 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. <u>General Criteria and References</u>

The criteria contained in the following documents will be the bases used to evaluate the SEP seismic re-evaluation of Yankee Rowe 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 has been performed. A summary of this review is contained in Appendix A. Contained at the end of this summary are concerns which require response by the licensee. 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 B.

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.

B. Audit Meeting Agenda

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.

---

a. Required at initial meeting only.

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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 Yankee Rowe nuclear power 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
Tommie R. Thompson

VI. Review Schedule

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

1. First working meeting — Week of 05-24-82

2. Plant visit — Not Scheduled

3. Final working meeting — Not Scheduled

4. Complete TER — 09-30-82

APPENDIX A. REVIEW OF LICENSEE SEP PROGRAM PLAN

Each item in this checklist is given two reviews. The first one is an "acceptance" review to check if that particular item has been addressed. The second is an "adequacy" review to judge the acceptability of the item for reevaluation purposes. Numbers in parentheses refer to comments listed at the end of this checklist.

I. Analysis Audit Format (Piping)	Addressed?	Adequate?
1. What computer codes were used in the analyses?	No (1)	--
a. How were the above computer codes verified?	No	--
2. Is the proper input forcing function being utilized?	Yes	No (2)
a. If response spectra method is used:		
(1) Is correct spectra and damping utilized?	No (2)	--
(2) Have sufficient modes been used to adequately describe system response?	Yes	Yes
(3) Do system frequencies straddle any peaks?	No (3)	--

		Addressed?	Adequate?
b.	If time history method is used:		
	(1) Is proper damping utilized?	N/A (4)	
c.	If static equivalent method is used:	N/A	--
	(1) Is justification provided for performing a static equivalent analysis?	N/A	--
	(2) How was required level of input determined?	N/A	--
3.	Has the piping system been properly modeled?	Yes	No (5)
a.	Have valves been properly modeled including any eccentricity?	Yes	No (6)
b.	Has adequate mass point spacing been utilized?	Yes	No (7)
c.	Are adjacent element length ratios reasonable?	No	--
d.	Have all significant branch piping systems been included?	Yes	Yes
e.	Have all supports been specified with correct imposed loads (if any), direction and stiffness?	Yes	No (8)
f.	Have supports with significant nonlinear characteristics been properly handled?	Yes	No (9)

	Addressed?	Adequate?
g. Have correct pipe sizes, geometry, thicknesses, and uniform weights been specified?	Yes	No (5)
h. Have correct design and operating pressure and temperature data been specified?	Yes	No (5)
i. If chart methods were used, were they used correctly?	Yes	No (10)
4. Has the piping system been evaluated against proper criteria?	Yes	No (11)
a. Have proper stress intensification factors been utilized?	Yes	No (12)
b. Have proper load combinations been analyzed?	Yes	Yes
c. Have proper allowable stress limits been selected in order to assure the required operation of the piping?	Yes	No (13)
d. Were standard or nonstandard components used?	No	--
e. What criteria were used in evaluating adequacy of supports?	No	--
II. Analysis Audit Format (Mechanical Equipment)	No (14)	--
1. Is the equipment rigid or flexible?	No	
a. How were the natural frequencies determined?	No	

	Addressed?	Adequate?
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?	No	
2. What type of analysis was performed?	No	
a. Static g level		
(1) How was required level of input determined?	No	
b. If response spectra method is used:	No	
(1) Is correct spectra and damping utilized?	No	
(2) Is sufficient system response achieved?	No	
(3) Do system frequencies straddle any peaks?	No	
(4) How were directional components of input applied (combined)?	No	
c. If time history method is used:	No	
(1) Is proper damping utilized?	No	
(2) How were directional components of input applied (combined)?	No	

	Addressed?	Adequate?
d. If testing was used for requalification:		
(1) What type of test was performed?	No	
(2) What justification is provided for the type of test used?	No	
(3) How were system natural frequencies determined?	No	
(4) How was the required response spectra (RRS) determined?	No	
(5) How does the test response spectra (TRS) compare to the RRS?	No	
(6) What g level was used in the test?	No	
(7) Were support and boundary conditions, including anchor bolts, properly simulated in the test?	No	
(8) How was functional operability verified during the test?	No	
(9) What criteria were used in evaluating the adequacy of the test results?	No	
3. What computer codes were used in the analyses?	No	
a. How were the above computer codes verified?	No	

	Addressed?	Adequate?
4. Has the system been properly modeled?		
a. Has adequate mass point spacing and distribution been used?	No	
b. Have all supports and boundary conditions, including anchor bolts, been properly modeled?	No	
c. Have significant nonlinear effects been properly handled?	No	
5. Has the system been evaluated against proper criteria?	No	
a. Have the proper load combinations been analyzed?	No	
b. Have proper stress intensities been evaluated?	No	
c. Have proper allowable stress limits been selected for all load carrying elements (anchor bolts, equipment supports, equipment housing and internal elements important to maintaining structural integrity)?	No	
d. How were computer output responses combined (directional and modal)?	No	
III. Analysis Audit Format (Electrical Equipment)	No (14)	--
1. Is the equipment rigid or flexible?		
a. How were the natural frequencies determined?		

	Addressed?	Adequate?
b. If flexible, is its response single-directional or multi-directional?	No	
c. If flexible, is its response at one predominant frequency or at several frequencies?	No	
2. What type of analysis was performed?	No	
a. Static g level	No	
(1) How was required level of input determined?	No	
b. If response spectra method is used:	No	
(1) Is correct spectra and damping utilized?	No	
(2) Is sufficient system response achieved?	No	
(3) Do system frequencies straddle any peaks?	No	
(4) How were directional components of input applied (combined)?	No	
c. If time history method is used:	No	
(1) Is proper damping utilized?	No	
(2) How were directional components of input applied (combined)?	No	

	Addressed?	Adequate?
d. If testing was used for requalification:	No	
(1) What type of test was performed?	No	
(2) What justification is provided for the type of test used?	No	
(3) How were system natural frequencies determined?	No	
(4) How was the required response spectra (RRS) determined?	No	
(5) How does the test response spectra (TRS) compare to the RRS?	No	
(6) What g level was used in the test?	No	
(7) Were support and boundary conditions, including anchor bolts, properly simulated in the test?	No	
(8) How was functional operability verified during the test?	No	
(9) What criteria were used in evaluating the adequacy of the test results?	No	
3. What computer codes were used in the analyses?	No	

	Addressed?	Adequate?
a. How were the above computer codes verified?	No	
4. Has the system been properly modeled?	No	
a. Has adequate mass point spacing and distribution been used?	No	
b. Have all supports and boundary conditions, including anchor bolts, been properly modeled?	No	
c. Have significant nonlinear effects been properly handled?	No	
5. Has the system been evaluated against proper criteria?	No	
a. Have the proper load combinations been analyzed?	No	
b. Have proper stress intensities been evaluated?	No	
c. Have proper allowable stress limits been selected for all load carrying elements (anchor bolts, equipment supports, equipment housing and internal elements important to maintaining structural integrity?	No	
d. How were computer output responses combined (directional and modal)?	No	

(1) Appendix E was missing from the Seismic Reevaluation Criteria (Reference a). No computer code information was provided.

(2) The proper forcing function depends on the individual piping systems. The proper input should be considered for each piping analysis. Is 1% to 2% damping utilized or is 2% to 3% damping used?

(3) The analyst is cautioned to check if frequencies straddle response spectra peaks. The most conservative analysis may not result if the peaks are straddled.

(4) N/A

(5) Individual piping analysis required to answer this question.

(6) Justify mass positions for manually operated valves and check valves. Do actual valve drawings exist to locate centers of gravity rather than assuming a distance of 1/3 of the "stem length?"

(7) What procedure was used to determine mass point spacing? How are system masses lumped (LLNL question 6, Ref. b)?

(8) Non linear supports were mentioned, but how are the rest of the supports modeled? What stiffnesses are utilized? Are support directions accurate?

(9) If the rod hanger was included in this analysis, was impact loading considered? Was rod hanger buckling considered? How were pipes resting on beams handled? How were potential higher stresses treated at support points when supports were removed? (LLNL questions 10 and 12 and EG&G San Ramon, Reference c, question F are also directed at this subject.)

(10) Method used for simplified stress analysis for small piping was not provided. (EG&G San Ramon question J)

(11) Piping support criteria was not provided.

(12) When a branch line was excluded from the analysis, was the proper stress intensification factor provided on the run at that point? Was the proper stress intensification factor provided on the branch analysis at the anchor?

(13) S_h must be taken at the design temperature.

(14) Proposed mechanical and electrical criteria were not contained in the program plan.

REFERENCES

a. Seismic Reevaluation Criteria for Yankee Nuclear Power Station Rowe, Massachusetts, Job No. 80023, Document No. DC-1, December 1980.

b. Lawrence Livermore National Laboratory letter to Mr. William, T. Russell, Program Plan Review for Yankee Rowe, December 2, 1981.

c. EG&G Energy Measurements Group, San Ramon Operations letter to Mr. T. A. Nelson, April 1, 1981.

APPENDIX B. 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 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 prope 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?

APPENDIX B
YAEC PIPING STRESS ANALYSIS PROCEDURES

Feb. 4, 1981

P.Y. Chen
B-1

Job No.: 80023
Doc. No.: DC-1
Revision: 0

SEISMIC REEVALUATION CRITERIA

For
Yankee [redacted] Nuclear Power Station
Rowe [redacted] Massachusetts
Prepared for
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

Prepared By
Earthquake Engineering Systems, Inc.
141 Battery Street, Suite 400
San Francisco, California 94111

Prepared by: Wong Chun Kau
Chun K. Wong

Reviewed by: J. Vallenas
~~Jose~~ Vallenas

Reviewed by: _______________
Herman Piryoutomo

Approved by: Eric van Stijgeren
Eric Van Stijgeren

December, 1980

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1.0 INTRODUCTION

Yankee Nuclear Power Station was designed before the current technology and codes had fully evolved. In the last decade, the state-of-the-art of earthquake engineering has progressed considerably. During this same period, new codes and regulations governing the design of nuclear power plants have been developed and have undergone significant changes. This evolution, while not resulting in a change in the basic design concepts, has yielded more detailed information concerning the behavior of structures, systems and equipment during earthquakes.

Yankee Atomic Electric Company has requested Earthquake Engineering Systems, Inc. (EES) to perform a seismic evaluation of the plant's critical structures and piping systems in accordance with the NRC Systematic Evaluation Program (SEP).

This document establishes the seismic criteria and the seismic evaluation approaches to be used in the investigation.

The present criteria have been specifically developed for linear elastic analysis. If necessary, additional criteria will be developed for non-linear analysis.

2.0 SCOPE

The purpose of this document is to establish the methodology and the criteria to be used for the seismic evaluation of piping systems and structures for the Yankee Nuclear Power Station.

Within the scope of this program, Earthquake Engineering Systems, Inc. (EES) will:

(a) Perform static analyses for thermal, dead weight, anchor movement and pressure loads, and dynamic analyses for seismic inertia loads. These analyses will be based on the as-built geometry of the piping systems and structures.

(b) Perform an evaluation of the critical piping systems and structures to withstand the loading conditions specified herein.

The piping systems and the structures included in the scope of this effort are summarized in Appendices A and B respectively.

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3.0 CODES AND STANDARDS

The following codes and standards shall be applicable to the appropriate sections of this document (except where noted otherwise).

(a) American National Standard Code for Pressure Piping - ANSI B31.1, 1980.

(b) Nuclear Regulatory Guides - 1.60 Rev. 1, 1.61 Rev. 0, 1.92 Rev. 1 and 1.122, Rev. 1.

(c) American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 8th Edition.

(d) American Concrete Institute (ACI) "Building Code Requirements for Reinforced Concrete" (ACI 318-77), including 1977 commentary.

(e) American Iron and Steel Institute (AISI), "Specification for the Design of Cold Formed Steel Structural Members," 1968 Edition with 1970 commentary and 1971 supplement.

(f) American Welding Society (AWS), "Structural Welding Code," D1.1-75.

(g) American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel Code", 1971 Edition including Code Case 1607.

(h) U. S. Nuclear Regulatory Commission, (NRC), "Development of Criteria for Seismic Review of selected Nuclear Power Plants", NUREG/CR-0098 May 1978.

(i) International Conference of Building Officials, "Uniform Building Code", 1979 Edition.

(j) U.S. Nuclear Regulatory Commission, (NRC), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-75/087, Section 3.7, Washington, D.C. - Office of Nuclear Reactor Regulation, September, 1975.

(k) American Concrete Institute (ACI), "Code Requirements for Nuclear Safety Related Concrete Structures" (ACI 349-76), including supplements.

(l) American Concrete Institute (ACI), "Code for Concrete Reactor Vessels and Containments" (ACI 359-77), including 1977 commentary.

(m) N.M. Newmark, W.J. Hall, R.P. Kennedy, J.D. Stevenson, and F.J. Tokarz, Seismic Review of Dresden Nuclear Power Station--Unit 2 for the Systematic Evaluation Program U.S. Nuclear Regulatory Commission, NUREG/CR-0891 (19[illegible]).

(n) ANSI B16.10, Face-to-Face and End-to-End Dimensions of Ferrous Values, 1973.

4.0 REFERENCE DOCUMENTS

The following reference documents shall be used in carrying out the piping stress and structural analysis effort:

4.1 Documents

(a) Yankee Atomic Electric Company, "Final Hazard Summary Report, Yankee Nuclear Power Station, Rowe, Massachusetts".

Do we have this document?

(b) Specification for Piping. YS-497 (S & W, JO-9699) July 15, 1959. Yankee Atomic Electric Company, Yankee Nuclear Power Station, Rowe, Massachusetts.

(c) Hot Service Thermal Insulation for Yankee Atomic Electric Plant. YS-2304 (S & W, JO-9699) June 1, 1959. Yankee Nuclear Power Station, Rowe, Massachusetts.

(d) Piping flow Diagrams, Yankee Nuclear Power Station, Rowe, Massachusetts. (Drawing Nos. E - S).

(e) Piping Drawings (Drawing No. 9699-FP-1 through 77).

(f) Stone and Webster Contract Drawings for Yankee Nuclear Power Station, 1958, 1959.

(g) Earthquake Engineering Systems, Inc., "Seismic Analysis and Stress Report for the Steel Vapor Container Structure of Yankee Nuclear Power Station," E-Y-YR-80061, 4-3-79, Rev. 1.

(h) Earthquake Engineering Systems, Inc., "Preliminary Seismic Evaluation, Concrete Reactor Support Structure for Yankee Nuclear Power Station", E-Y-YR-80064, 8-10-79, Rev. 1.

(i) Weston Geophysical Corporation, "Geology and Seismology, Yankee Rowe Nuclear Power Plant", January 29, 1979.

(j) Wiegel, R.L., Earthquake Engineering, Prentice-Hall, Inc. (Englewood Cliffs, N.J.), 1970, 518p.

(k) Housner, G.W. (January, 1957), "Dynamic Pressures on Accelerated Fluid Containers", Bulletin, Seismic Society of America, 47(1).

(l) U.S. Atomic Energy Commission (1963). Nuclear Reactors and Earthquakes, TID-7024, Washington, D.C. - Office of Technical Services.

5.0 STRUCTURAL PERFORMANCE CRITERIA

This section describes the criteria to be used in the analysis and evaluation of the structures listed in Appendix B.

5.1 Material Properties

The following material specifications govern unless superseded by field tests.

5.1.1 Concrete

The concrete properties ~~for~~ each building are summarized in Table 5.1. These values are obtained from References 4.1(f) & (h).

5.1.2 Steel

The steel properties for the different structures are summarized in Table 5.1.

5.1.3 Masonry

(later)

5.1.4 Soils

Bearing capacity for the soil underneath all footings shall be assumed to be 8 ksf if wind or earthquake loads are not considered and 10.6 ksf if they are. Compacted backfills shall be assumed to have a bearing capacity of 4 ksf. For reference, see drawing no. 9699-FC-59B.

Provide the geotechnical properties of the surrounding soil? e.g. Vs, β, water Table, etc.

5.2 LOADS DESCRIPTION

5.2.1 Dead Loads

Dead loads and their related internal moments and forces, including fixed equipment loads, will be included in the analysis. Equipment weights less than 500 lbs. will be considered as distributed loads and equipment weights more than 500 lbs. will be applied as concentrated loads.

5.2.2 Live Loads

How these live loads were taken into account in the analysis?

Live loads and their related internal moments and forces, including any moveable equipment loads, will be included in the analysis.

5.2.3 Earth Pressure and Groundwater Table

Loads due to earth pressure will be included in the analysis. Hydrostatic loads due to groundwater table will be included in the analysis.

5.2.4 Fluid Loads

Fluid loads will be treated as hydrostatic loads except under seismic conditions. For this case, the fluid loads will be computed using the Housner method. For references see 4.1(j), (k) & (l).

More update ref. should be used to consider the fluid loads [illegible]

5.2.5 Seismic Loads

All structures shall be evaluated for the Safe Shutdown Earthquake (SSE).

5.3 Analysis Methodology

This section outlines the methodology to be utilized in order to achieve the objectives of this evaluation.

5.3.1 Analysis Procedure

Figure 5-1 shows the general structural analysis steps. These steps are described in the following paragraphs:

The SSE ground response spectra will be used to generate artificial time histories using EES' SIMQUAKE program. These time histories will then be verified by checking their generated response spectra (plotted using INSPEC program) against the recommendations of Reg. Guide 1.122.

In a parallel effort the structural models are developed.

The scope of the present evaluation involves linear elastic analysis only. The basic analysis technique will be the response spectrum modal superposition method of dynamic analysis. Soil-structure interaction effects will be neglected, since studies performed previously have shown these effects to be negligible. The inter-connected buildings will be studied on a case-by-case basis. If coupling exists between the buildings they will be considered connected and will be analyzed as one unit. With a few possible exceptions (where the floor slabs have substantial openings) the floor diaphragms will be treated as rigid in plane. Three-dimensional beam elements will be used to describe columns and other beam type components. The models will describe the stiffness and mass relationship in three-dimensional space. Torsional effects due to asymmetric characteristics are automatically considered in this procedure. In buildings where the potential for further accidental torsion is considered to be likely, accidental torsion considerations as per NUREG/CR-0098 will be included. The superstructure in the reactor concrete pedestal is very stiff and consequently it responds dynamically in a rigid body type motion. The majority of the structural deformations will take place in the base columns and especially at their connections with the superstructure and foundation. Based on this behavior, a simplified model will be developed respresenting the superstructure as a vertical cantilever with multiple lumped masses. This cantilever will be connected to the base columns with a rigid beam system. The superstructure will be subsequently analyzed for the effects generated from the above analysis in addition to its own loads.

The steel vapor container will be modeled using shell elements to represent the sphere, and beam elements to represent the columns. In developing the spherical model, care will be taken to generate a finer mesh at the locations of high stress. The base of all columns will be fixed. The thin shell elements will model the actual thickness of the plate used in the structure which ranges from 7/8 to 3 inches. Additional masses will be applied to selected node points in the model to account for concentrated loads such as hatches and platforms.

The structural dynamic properties obtained from the dynamic analyses will be used to perform a modal superposition analysis using the program MOST. Amplified time histories will be generated at designated locations in the structures. These amplified time histories will be used to generate the Amplified Response Spectra (ARS) using the program INSPEC. The ARS will be broadened in conformance with Reg. Guide 1.122. These ARS will be used as input for the piping and equipment analyses.

The resulting stresses and deformations will be obtained from the modal responses using the SRSS method, except for closely spaced modes where Regulatory Guide 1.61 method will be used.

The stresses and deformations will be evaluated for their compliance with section 5.4. The relative displacements of neighboring structures will be checked for the possibility of impact. The piping between buildings will be reviewed for these displacements.

5.4 ACCEPTANCE CRITERIA

5.4.1 Load Combination

The analyses will be performed assuming that the seismic event is initiated with the plant at normal full power condition. The following load combination will be considered in evaluating the structure:

$$U = D + L + T_o + R_o + P_o + E$$

where

U = Total load to be resisted.

D = Dead loads or their related internal moments and forces, including any permanent equipment loads, hydrostatic loads, and lateral soil pressures. It also includes operating static and dynamic heads and fluid flow effects.

L = Live loads or their related internal moments and forces, including any moveable equipment loads and other loads which vary in intensity and occurrence, such as wind. For equipment supports, it also includes loads due to vibration and any support movement effects.

T_o = Thermal effects and loads during startup, normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

R_o = Pipe reactions during the startup, normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

P_o = Pressure equivalent static load within or across a compartment generated by normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

E = Loads generated by the safe shutdown earthquake. Three earthquake directions will be considered as per NUREG/CR-0098 except for special condition as discussed in NUREG/CR-0891.

5.4.2 Allowable Stresses

This section is specifically developed for linear elastic dynamic analysis. Additional criteria will be developed for non-linear analysis if required. The allowable stresses for reinforced concrete portions of the structures will be per ACI code 318-77. In lieu of the code load factors, the factors shown in section 5.4.1 will be used.

The stresses for steel structures will be checked against Part 1 of AISC Specifications, 1980 edition. Stresses up to 0.95 of yield or buckling will be allowed. The stress limits of the vapor container steel shell elements will be based on those currently allowed by the ASME Boiler and Pressure Vessel Code for faulted conditions including membrane and bending effects.

5.4.3 Allowable Deformations

The deformations will be limited according to the existing clearances so as to prevent impact of adjacent structures.

5.4.4 Damping

Damping values for different types of structures will be based on the stress levels generated in each structure. The values to be used will be in accordance with NUREG/CR-0098 (See Table 5-2).

5.4.5 Alternate Criteria

In cases where stresses exceed the allowables given in section 5.4.2, modifications may be recommended or alternate methods of analysis may be used, taking into consideration the non-linear behavior of the structure. Special acceptance criteria for these cases will be developed on a case-by-case basis.

6.0 PIPING ANALYSIS CRITERIA

This section describes the criteria to be used in the stress analysis of the piping systems listed in Appendix A. These criteria are applicable to pipings with nominal outside diameter larger than 2".

6.1 Load Description

The following load cases shall be considered for the piping stress analysis, in addition, local stress concentration due to integral support shall be evaluated.

6.1.1 Thermal Load

Loads due to steady state temperature effect, including thermal anchor movements.

6.1.2 Weight Load

Loads due to pipe, content and insulation.

6.1.3 Pressure Load

Loads due to steady state internal pressure.

6.1.4 Seismic (SSE) Load

Loads due to earthquake excitations which include both seismic inertia effect and seismic anchor movements.

6.2 ANALYSIS METHODOLOGY

6.2.1 Geometry and Computer Modeling:

For the purpose of computer analysis, pipig system will be idealized by three dimensional linear elastic model with finite numbers of structural members interconneced at finite numbers of nodal points. All supports and anchors are assumed to be rigid. The direct stiffness method is to be used in the solution of the problem.

(a) Each problem shall be considered from anchor to anchor. If ah anchor to anchor problem exceeds program limitations, the following approach shall be considered in modeling:

- Overlapping such that there is negligible migration of loads from one problem to another.
- Bracketing results of multiple computer runs to assess boundary conditions or loading conditions.

(b) The geometry and restraint conditions shall be modeled in accordance with Isometrics based on as-built conditions.

(c) The pipe material properties and analysis conditions shall be considered as per YAEC's approved information such as Yankee Piping Specification (YS-497), YAEC flow diagrams, Yankee Insulation Specifications (YS-2304) and Grinnel catalog data.

(d) Branch connections with a moment of Inertia ratio >25:1 (main line/branch line) may be decoupled for analysis purpose assuming the main line node point as an anchor for the branch line. The main line deflections and rotations shall be input as anchor movements for the branch line analysis.

(e) Equipment nozzles and penetrations shall be considered as anchor points in the analysis. All equipment are assumed to be properly supported. Loading shall be summarized and compared to allowables when available. When allowable loads are not available, the analysis loads shall be submitted to YAEC for their review. Thermal anchor movements at nozzles and penetrations shall be indicated on the "As-Built" Isometrics. Or, if necessary, they shall be calculated by conventional methods based on system design temperature.

(f) Valves shall be modeled as follows:

- Thickness of the valve body shall be assumed as twice the connecting pipe wall thickness.

0-2

- Manually operated valves and check valves shall be modeled with the mass of the valve concentrated at the centerline of the pipe at the valve node points.

- Motor and air operated valves shall be modeled as eccentric mass points. The total weight of the valve shall be concentrated at a point one-third (1/3) the distance between the centerline of the operator and valve assembly (one-third of the "stem length" measurements as noted on the valve data form).

- If not available, body length of the valve shall be as per ANSI B16.10.

- Seismic accelerations of the valves will not be summarized.

(g) Flanges shall be considered as additional lumped weights. Flange thickness shall be assumed to be the same as that of pipe for purposes of modeling stiffness.

(h) Stress intensification factors for tees, reducers, flanges, elbows and couplings (half and full) shall be considered as per code requirements, ANSI B31.1 - Power Piping, 1980 edition.

(i) For the purpose of analysis, penetrations shall be treated as follows:

- Grouted penetrations: A bilateral restrain condition shall be assumed to exist on either side of the penetration for all load cases. Axial restraint of the pipe shall not be considered unless a welded collar is indicated on the pipe and embedded in the penetration.

- Ungrouted penetrations: At ungrouted penetrations, deflection of the pipe ≤ 1/4" shall be considered acceptable. Where deflections exceed 1/4", further review of actual penetration clearances shall be initiated. Deflections shall be based on the combined thermal and seismic conditions.

(j) The Cold modulus of elasticity E_c at (70°F) room temperature shall be used. The moduli of elasticity for ferrous and non-ferrous materials shall be taken from Appendix C Tables C-1 and C-2 of the ANSI B31.1 code.

(k) The Poisson's ratio shall be taken as 0.3 for all metals at all temperature.

6.2.2 <u>Weight Analysis</u>

The following considerations shall be made for dead weight analysis:

Weight analysis shall be performed considering weight of the pipe, content, insulation and concentrated masses (such as pipes supported off pipe, flanges and valves).

6.2.3 Thermal Analysis

Thermal analysis of the piping system shall be performed based on the maximum design temperatures as designated on YAEC flow diagrams or stress isometric drawings. Effects of thermal movements from equipment nozzles, anchors, penetrations and connecting piping shall be analyzed. The Thermal Anchor Movement stress (TAM) shall be added to thermal expansion stress to obtain the total thermal stress.

6.2.4 Seismic Analysis

(a) The basic analysis technique will be the Response Spectrum, Modal Superposition method of dynamic analysis. Lumped mass models will be employed.

For rod hanger type of supports, when the uplift due to seismic load (include Thermal Load if it is upward) is larger than 90% of weight load, the rod hanger support shall be assumed noneffective. Consequently the particular rod hanger support will not be included in the computer modeling.

Seismic Inertia analysis and Seismic Anchor Movement analysis shall be performed for the Safe Shutdown Earthquake (SSE).

The spectra for the SSE is in the process of development and will be incorporated into this document when it becomes available.

(b) Application of Spectra:

For each earthquake condition, three directions of earthquake will be considered. (Two horizontal components and one vertical component). The total response due to each of the three (3) components of earthquake shall be calculated first. These responses shall then be combined by the SRSS method (Square Root of the Sum of Squares). The procedures to be used in combining the modal responses and responses due to spatial components of earthquake shall be as follows:

1. The modal responses for each component of earthquake shall be combined by taking into consideration the modes with closely spaced frequencies in accordance with NRC Regulatory Guide 1.92 Rev. 1, Feb. 1976. Subsections 1.2.1, 1.2.2, or 1.2.3.

2. The total systems responses due to the three (3) spatial components of earthquake are then combined by the SRSS method.

The responses of the Yankee Site Specific load case shall be used to evaluate the piping system and its support. For piping systems spanning several floors or with pipe supports connected to support structures attached to different floors, the response spectra for the analysis of the piping system shall be the envelope of the floor response spectra of all the floors involved.

(c) Cut-off frequency and minimum number of modes:

A cut-off frequency of 33 cps and with no less than 10 modes shall be considered in the analysis. An equivalent Static-Seismic analysis based on a constant acceleration from the spectra at 33 cps cut-off frequency shall be performed when the contributions of higher modes (>33 cps) are significant.

(d) Damping values:

For the seismic SSE condition, a damping value of two percent (2%) of critical damping shall be used for piping with outside diameter less than or equal to 12" and a damping value of three percent (3%) of critical shall be used for piping with outside diameter larger than 12".

6.2.5 Seismic Anchor Movement Analysis (SAM)

The SSE Seismic Anchor Movement load condition shall be considered for both stress and support load evaluations.

6.2.6 Pressure Effect

The effect of internal pressure shall be considered in computing longitudinal stresses.

6.3 Acceptance Criteria

6.3.1 Stresses in the piping system must not exceed the allowable stress limits of the ANSI B31.1 - Power Piping Code, 1980. The Acceptance Criteria shall be considered satisfied when the requirements of the following equations are met.

(a) The effects of pressure, weight, and other sustained loads must meet the following requirements:

$$\frac{PDo}{4tn} + \frac{0.75i}{Z} M_A \leq K S_h \quad \text{(Eq. 6.3.1-A)}$$

Where:

K	=	1.0 for Dead Weight Loading
P	=	Internal Design Pressure, psi
D_o	=	Outside Diameter of Pipe, in.
t_n	=	Nominal wall thickness of components, in.

M_A = Resultant moment loading on cross section of the pipe due to weight and other sustained loads, in-pounds.

Z = Section modulus of the pipe, in^3.

S_h = Basic material allowable stress at maximum temperature from allowable stress tables, psi.

i = Stress intensification factor. The product of 0.75i shall never be taken as less than 1.0.

Stress Intensification Factors, "i" shall be as per ANSI B31.1 code 1980 edition.

(b) The effects of pressure, weight, other sustained loads and occasional loads including earthquake must meet the following requirements:

$$\frac{PDo}{4tn} + \frac{0.75i\ M_A}{Z} + \frac{0.75i\ M_B}{Z} < K\ S_h \quad \text{(Eq. 6.3.1-B)}$$

Where:

K = 1.8 for Safe Shutdown Earthquake (SSE).

M_B = Resultant moment loading on cross section due to occasional loads such as earthquake. For earthquake use only one-half the earthquake moment range. Other terms same as 6.3.1 - A.

(c) Thermal Expansion Stress (S_E):

$$S_E = \frac{i\ Mc}{Z} < S_A \quad \text{(Eq. 6.3.1-C)}$$

Where:

M_C = Range of resultant moments due to thermal expansion. Also include moment effects of anchor displacement due to earthquake if anchor displacement effects were omitted from Eq. 6.3.1-B

S_A = Allowable stress range for expansion stress.

= $f\,(1.25\ S_c + 0.25\ S_h)$

Where:

S_c = Allowable stress of the specific material at 70 degrees F. (Psi)

S_h = Allowable stress of the specific material at maximum temperature in degrees Fahrenheit (Psi)

(d) <u>Sustained Plus thermal Expansion Stresses:</u>

The effects of pressure, weight, other sustained loads and thermal expansion must meet the requirements of the equation 6.3.1-D

$$\frac{PD_o}{4t_n} + \frac{0.75 i M_A}{Z} + \frac{i M_c}{Z} \leq (S_h + S_A) \quad \text{(Eq. 6.3.1-D)}$$

Terms as previously described.

(e) The requirements of either Equation 6.3.1-C or Equation 6.3.1-D must be met.

(f) Even though only the Response Spectrum Analysis method is considered in this criteria, we do not preclude the possibility of using time history analysis method, if the situation warrants its application. Specific criteria for time history analysis will be provided when the need arises.

6.3.2 Allowable Stresses

Allowable stress values to be used for power piping systems are given in Appendix A of ANSI B31.1 power piping code. Those values shall be used for piping stress analyses.

For material allowable stress values not available in Appendix A of ANSI B31.1, reference should be made to ASME Boiler and Pressure Vessel Code Section III, Division 1. The appropriate allowable stress values shall be taken from tables contained in Appendix I.

6.4 Small Pipe Stress Analysis

This section applies to piping with nominal outside diameter of 2" or smaller.

6.4.1 Detailed Stress Analysis

For detailed stress analysis the same procedures and methods as those for large pipe stress analysis shall be followed. (Sections 6.1 through 6.3). In addition:

- All pipe bend shall be considered to have a bend radius of five (5) times the pipe diameter.

- Connections at Elbow, Tee, Reducer, Coupling and nozzle shall be considered as socket welded.

6.4.2 Simplified Stress Analysis

This is an alternative method to the Detailed Stress Analysis method. Each span of a piping system (spans are generally separated by guides) is evaluated by simpliefied thermal, seismic and weight stress analyses. Span lengths and support locations are investigated to ensure the requirements of piping flexibility and high natural frequency are met.

(a) Weight stress - weight stress is kept to predetermined level by using specified support spacings. Span length tables, based on a bending stress of 1,500 psi shall be used for pipe with uniform weight. When concentrated loads such as value or risers exist, a hanger should be placed within 6 inches of the concentrated weight or the weight span spacing should be modified.
(Applicable Gravity Span tables will be provided later).

(b) Thermal stress - thermal stress shall be kept to an acceptable level by providing a minimum offset to absorb thermal movement. Offset is defined as the length of

piping in a plane perpendicular to the direction of movement. The offset piping shall be unrestrainted in the direction of movement.
(Applicable Offset tables will be provided later).

(c) Seismic stress - Seismic pipe spans shall be generated by simplified analysis method so that the actual stress will be less than the predetermined max. stress. These seismic pipe spans and restraint loads are defined as a function of unique spectra curves and pipe sizes. The basic approach is to keep the seismic acceleration of the system low and to keep the natural frequencies in the "Rigid Range". The seismic spans shall generally be separated by guides at each change of direction, at all extended massses and at each tee. (Applicable Seismic Span tables will be provided later).

(d) Pressure stress - longitudinal pressure stress shall be computed as per ANSI B31.1 code requirement. The pressure stress shall be compared with a pre-specified value.

(e) Acceptance requirements - the piping system is considered to have met the stress acceptance requirements if each span satisfies the span length, offset and pressure stress requirements mentioned above. Span length shall be adjusted to account for the effect of stress intensification factor applicable to the component under consideration. If any of the above requirements cannot be satisfied, a detailed stress analysis shall be performed for the portion of piping involved.

APPENDICES

Appendix A

A. The following piping systems are included in the scope of this evaluation.

1. Main Steam ✓
2. Feed Water ✓
3. Reactor (Main) Coolant ✓
4. Pressure Control & Relief ✓
5. Charging & Volume Control ✓
6. Safety Injection ✓
7. Shut Down Coolant ✓
8. Sample and Drain System --
9. Primary Plant Purification --
10. Fuel Transfer
11. Vapor Containment Heating System

APPENDIX B

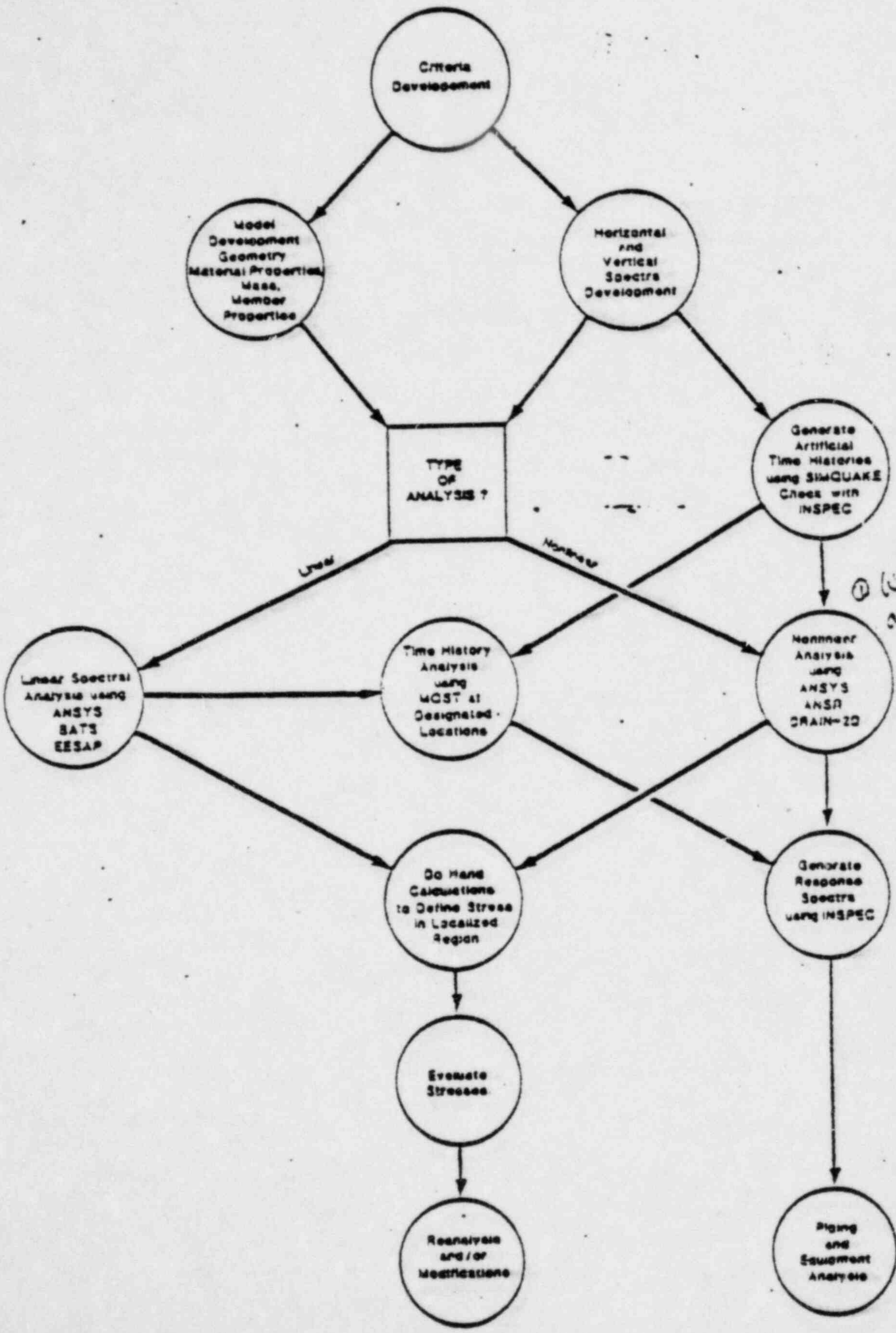
B. The following structures are included in the scope of this evaluation.

1. Concrete Reactor Support Structure
2. Vapor Container Structure
3. Diesel Generator Building and Accumulator Enclosure
4. Turbine Building and Turbine Pedestal
5. Ion Exchanger Building
6. Primary Auxiliary Building and Radioactive Tunnel
7. Screen Well and Pump House
8. Spent Fuel Pool and Spent Fuel Chute

Appendix C

FIGURE 5-1

STRUCTURAL ANALYSIS FLOWCHART

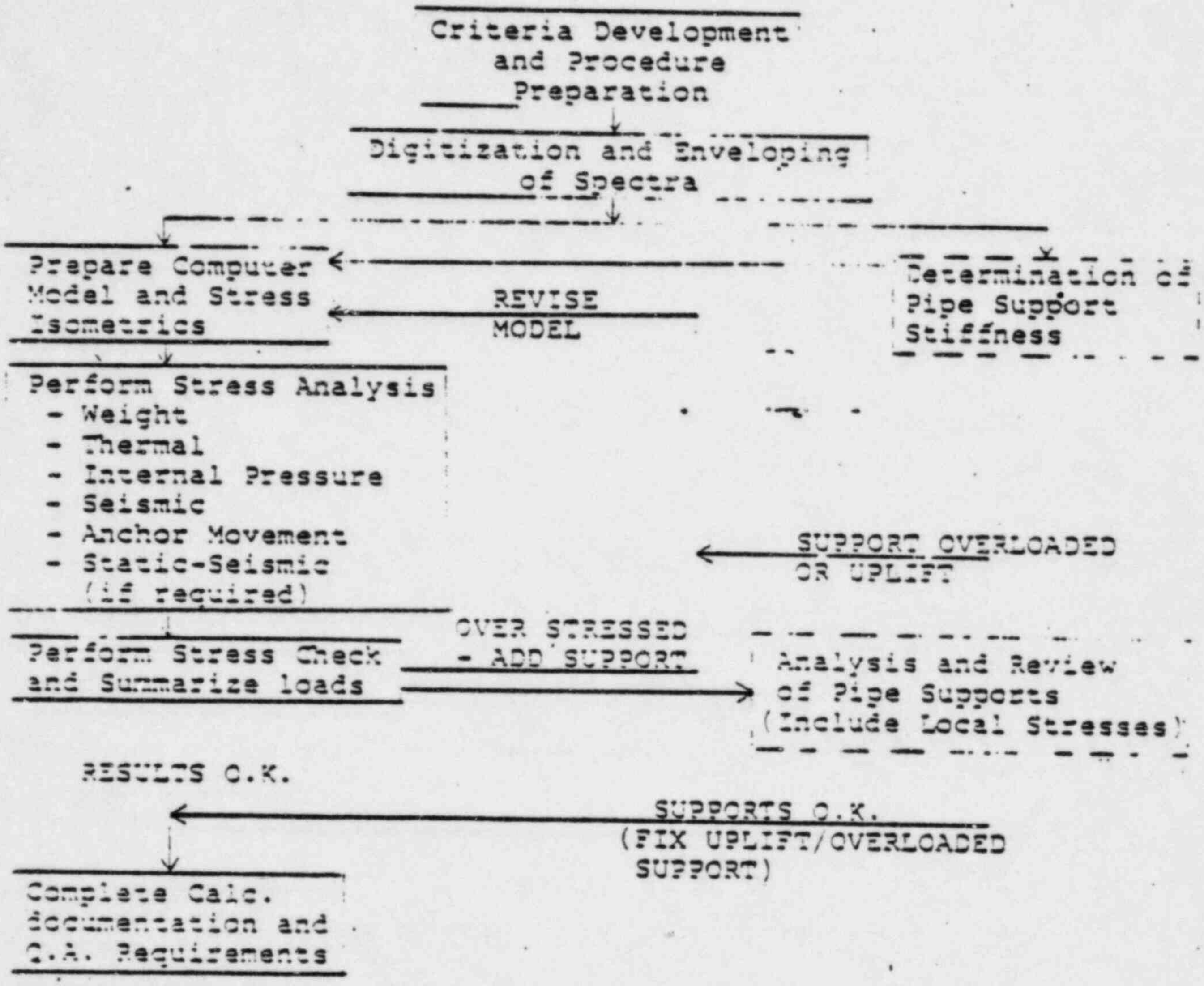


① What kind of nonlinear analyses? ②

Appendix C

Figure 6-1

PIPING STRESS ANALYSIS FLOWCHART



Appendix D

TABLE 5-1

MATERIAL PROPERTIES

BUILDING DESCRIPTION	MATERIAL		
	STRUCTURAL STEEL	CONCRETE	REINFORCING STEEL
1. DIESEL GEN. BLDG. ACCUM. TANK [illegible]	ASTM A7 (Fy = 33 ksi)	fc' = 2,000 psi	ASTM A 305, INT. GR. (Fy = 40 ksi)
[illegible] TURBINE BLDG. CONTROL ROOM	ASTM A7 (FY = 33 ksi)	a) Footings, fc' = 2,500 psi. GRADE BMS. b) ALL OTHER CAST-IN PLACE fc' = 3000 psi c) PRECAST BMS & WALL SHIELD fc' = [illegible] psi d) TURBINE SUPPORT MAT. fc' = 3000 psi.	ASTM A305, INT. GR. (Fy = 40 ksi)
3. SPENT FUEL POOL SPENT FUEL CHUTE	ASTM A7 (FY = 33 ksi)	fc' = 3000 psi	ASTM A 305, INT. GR. (Fy = 40 ksi
4. [illegible] WELL [illegible]	ASTM A7 (FY = 33 ksi)	fc' = 3000 psi	ASTM A 305, INT. GR. (Fy = 40 ksi)
5. STEEL VAPOR CONTAINER	A 201 B to F_t = 60 ksi A 300 Fy = 33 ksi	PRESSURE = 31 psi fc' = 3000 psi (Pedestals & Footings)	ASTM A 305, INT. GR. (Fy = 40 ksi)
6. CONCRETE CONTAINMENT & SUPPORT STRUCTURES	A201 B to A300 Ft = 60 ksi Fy = 33 ksi	a) FOOTINGS & GRADE BMS, fc' = 3000 psi b) PEDESTALS, COLS, WALLS, ALL OTHERS fc' = 4000 psi	ASTM A 305, INT. GRADE (Fy = 40 ksi)
[illegible]	ASTM A7 Fy = 33 ksi	fc' = 3000 psi	ASTM A 305, INT. GR. (Fy = 40 ksi)

* Source: NUREG/CR-0098

Where are these numbers obtained from?

Appendix D

TABLE 5-2

RECOMMENDED DAMPING VALUES*

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

* Source: NUREG/CR-0098

CYGNA ENERGY SERVICES

Rev, 0

NRC
PIPING PRESENTATION

YANKEE ROWE POWER STATION

August 3 & 4, 1982

PRELIMINARY

D-7-

PIPING PRESENTATION SLIDE INDEX

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1) Program
2) Total System Scope
3) Problem No. Identification (17 pages)
4) Spectra Considered to Date
5) Load Cases
6) Computer Programs
7) Criteria Used
8) Load Combinations
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10) Flow Chart for Analysis Effort

B) RESULTS FOR HOT SHUTDOWN SYSTEMS

1) Stress Tables/Plots for Equations 11, 12, 13, and 14 for YCS
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5) Clarification of Stress Results ECCS/AFW to IDBS
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SCOPE DEFINITION

- NON-ISOLABLE PIPING SYSTEMS WHICH COULD CAUSE A PRIMARY COOLANT PRESSURE BOUNDARY FAILURE
- NON-ISOLABLE PIPING SYSTEMS WHICH COULD CAUSE A SECONDARY COOLANT BOUNDARY FAILURE
- PIPING SYSTEMS NECESSARY TO MAINTAIN A QUALIFIED HEAT REMOVAL FLOW PATH REQUIRED TO DEMONSTRATE SAFE SHUTDOWN CONDITION

ANALYSIS PROGRAM

- TOTAL NO. CALCULATIONS 126
- HOT SHUTDOWN CALCULATIONS 30
- COLD SHUTDOWN CALCULATIONS 96

TOTAL SYSTEM SCOPE

- MAIN COOLANT LOOP
- MAIN STEAM
- BOILER FEED DISCHARGE
- PRESSURE CONTROL AND RELIEF
- CHARGING
- CHEMICAL SHUTDOWN
- SHUTDOWN COOLING
- PRIMARY PLANT PURIFICATION
- SAMPLE AND DRAIN
- CONTAINMENT VENTILATION
- SAFETY INJECTION
- FUEL CHUTE TRANSFER
- COMPONENT COOLING
- SERVICE WATER
- V.C. VENTILATION AND PURGE

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
001	Main Steam	HS	From nozzle at steam generator E-7-1 to anchor at P.B. of V.C. EL. 1070'-11"
002	Main Steam	HS	From nozzle at steam generator E-7-2 to EL. 1122'-2" to anchor at P.B. of V.C. EL. 1070'-11"
003	Main Steam	HS	From nozzle at steam generator E-7-3 EL. 1122'-2" to anchor at P.B. of V.C. EL. 1070'-2"
004	Main Steam	HS	From nozzle at steam generator E-7-4 EL. 1122'-2" to anchor at P.B. of V.C. EL. 1070'-11"
005	Main Steam	HS	From anchor SHP-A50 on 24" SHP-601-5 to Anchor at P.B. of V.C. EL. 1070'-11"
006	Main Steam	HS	From anchor SHP-A50 and the pin anchor to anchor at P.B. of V.C. EL. 1070'-11"
007	Main Steam	HS	From anchor SHP-A50 and the pin anchor to anchor at P.B. of V.C. EL. 1070'-11"

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
008	Main Steam	HS	From anchor SHP-A50 on 24" SHP-601-5 to anchor at P.B. of V.C. EL. 1070'-11"
009	Main Steam	HS	From anchor SHP-A50 on line 24" SHP-601-5 to 4 nozzles on turbin unit
10A	Main Steam	HS	From anchor SA-A-1 at EL. 1041'-11" to connection on 24"-SHP-601-5
10B	Main Steam	HS	From anchor SA-A-1 and end at anchor A-1 El. 1032'-7", to nozzle at boiler E-43-2, nozzle at main condenser E-5 and 2 nozzles at aux. priming ejectors EJ-2 and EJ-3
021	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. El. 1070'-10" to nozzle at steam generator E-7-1
022	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to nozzle at steam generator E-7-2

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
023	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to nozzle at steam generator E-7-3
024	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to nozzle at steam generator E-7-4
025	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to anchor WCBD-A102 on 14"-WCBD-601-6
026	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to anchor WCBD-A102 on 14"-WCBD-601-6
027	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to anchor WCBD-A102 on 14"-WCBD-601-6
028	Boiler Feed Discharge	HS	From anchor at P.B. of V.C. EL. 1070'-10" to anchor WCBD-A102 on 14"-WCBD-601-6
41A	Pressure Control and Relief	HS	From anchor BRL-A-10 on 6" BRL-302-6 to nozzle at pressurizer E-22 at EL. 1115'-5"

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
41B	Pressure Control and Relief	HS	From anchor BRL-A-10 on 6" BRL-302-6 to nozzle at pressurizer E-22 EL. 1115'-5"
030	Boiler Feed Discharge	CS	From anchor at EL. 1028'-11-3/4" to anchor at primary auxilliary building conc. FL. EL. 1022'-8" and x, z rest. at 1027'-7-3/4"
41C	Pressure Control and Relief	HS	From Anchor BRL-A-10 on 6" BRL-302-6 to nozzle at pressurizer E-22 EL. 1115'-6"
101	Main Coolant Piping	HS	From nozzle at R.P.V. to nozzle at R.P.V. and A branch to nozzle at pressurizer and 2 nozzles at steam generator E-7-1
102	Main Coolant Piping	HS	From 2 nozzles at R.P.V. to 2 nozzles at steam generator E-7-2
103	Main Coolant Piping	HS	From 2 nozzles at R.P.V. to 2 nozzles at steam generator E-7-3

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
104	Main Coolant Piping	HS	From 2 nozzles at R.P.V. to 2 nozzles at steam generator E-7-4
121	Shutdown Cooling Piping	HS	From anchor at P.B. of V.C. EL. 1058'-5" to connection on 20"-CRM-2504-10
122	Shutdown Cooling Piping	HS	From anchor at P.B. of V.C. EL. 1057'-8" to connection on 20"-CRM-24-2504-12
201	Safety Injection	HS	From anchor at P.B. of V.C. EL. 1053'-10" to connection on 8" PRSH-2502-1 EL. 1065'-5"
207	Safety Injection	HS	From 2 anchor at P.B. of V.C. at EL. 1055'-0" and 1060'-5". 4 connection to lines 20" CRM-2504-9 20" CRM-2504-3 20" CRM-2504-6 20" CRM-2504-12
011	Main Steam Piping	CS	From anchor A-1 to anchor 75-A-4 and in-between 2 nozzles, one at boiler E-43-1 and one at emergency boiler feed pump.

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
029	Boiler Feed Discharge	CS	From anchor at floor aux. boiler room conc. EL. 1022'-8" and proposed x and z rest. at EL. 1027'-4-1/2" to nozzle (tank #1) EL. 1024'-2", nozzle at emergency boiler feed pump EL. 1023'-5", anchor EL. 1028'-5", nozzle at cond. return pump (P-64), nozzle at cond. receiver tank (TK-63), nozzle at aux. boiler feed pump (P-60-1) and nozzle at aux. boiler feed pump (P-60-2)
061	Charging Piping	CS	From anchor at P.B. of S.V.C. on 3"-CRBH-2502-4 to the cap at EL. 1065'-4"
062	Charging Piping	CS	From anchor at P.B. of S.V.C. on 3" CRBH-2502-4 to the nozzle at low pressure surge tank TK-23

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
63A 63B 63C 63D	Charging Piping	CS	From anchor at P.B. of S.V.C. to 4 proposed anchors, 3 in P.A.B. and 1 on 2" CRCH-2502-5. Also bounded by charge pumps P-15-1, P-15-2 and P-15-3. 3 branches on sht. 2/2 end with caps, but only one (1) of them connects to 2" CRCL-152-3
064	Charging Piping	CS	From charge pumps P-15-1, P-15-2 and P-15-3 to 2 proposed anchors in P.A.B. and connected to lines 8" PRSL-302A-3 and 8" CRT-302-1
081 082 083 084	Chemical Shutdown Piping	CS	From one end it is bounded by 4" CRCL-152-1 and the floor drain at EL. 1021'-8" to the other end by 2 proposed anchors in P.A.B.
123	Shutdown Cooling	CS	From anchor at P.B. of S.V.C. to low pressure surge tank TK-23, low pressure tank cooling pumps P-19 and P-23. Also 1 proposed anchor in P.A.B. and a connection to 6" line on problem 125.

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
124	Shutdown Cooling	CS	From anchor at P.B. of S.V.C. to low pressure surge tank TK-23, low pressure surge tank cooler (E-10) and shut-down cooling heat exchanger (E-9). Also a proposed anchor in P.A.B.
125	Shutdown Cooling Pipin	CS	From low pressure surge tank cooler E-10 to low pressure surge tank cooling pumps P-23 & P-19. Also shutdown cooling heat exchanger E-9.
042	Pressure Control and Relief Piping	CS	From anchor at P.B. of S.V.C. to anchors at boundary of P.A.B.
43A	Pressure Control and Relief Piping	CS	From anchor BRL-A-1 on 12" BRL-154-11 to 3 nozzles at low pressure surge tank (TK-23).
43B	Pressure Control and Relief Piping	CS	From Anchor BRL-A-1 on 12" BRL-154-11 to 3 nozzles at low pressure surge tank (TK-23)

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
044	Pressure Control and Relief Piping	CS	From anchor BRL-A-1 on 12" BRL-154-11 to flanged rupture disc on 20"-BRL-154-11.
141	Primary Plant Purification	CS	From nozzle at low pressure surge tank TK-23 to proposed anchor in P.A.B..
142	Primary Plant Purification	CS	From nozzle at low pressure surge tank (TK-23) to purification drain and cooling pumps P-16-1 and P-16-2.
143	Primary Plant Purification	CS	From S.V.C. anchor at EL. 1044'-6" to branch connection to PI-1212 Sht. 1/4. Also 2 connections to 2-1/2" CRP-152-1.
161 162 163	Sample and Drain Piping	CS	From 2 anchors at P.B. of S.V.C. EL. 1046'-0" and 1046'-7-1/2". North side bounded by 2 nozzle connections at drain for V.C. col. seals. In south connected to nozzle at component cooling water surge tank (TK-2) and nozzle at VAPEC container drain tank (TK-25).

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
181	Containment Ventilation	CS	From a proposed anchor on 6"-CV-121-1 at anchor 7L-A1 on 6" CV-151-1.
182 183 184	Containment Ventilation	CS	From anchor 7L-A-1 to guide 7L-G-1. Four branches coming out of the manifold all end. To anchor at P.B. of S.V.C. EL. 1068'-8" (4 S.V.C. anchors.)
185	Containment Ventilation	CS	From anchor 7L-A-2 to proposed anchor on 6" line in P.A.B.
186 187 188	Containment Ventilation	CS	From anchor 7L-A-2 to N.S. #2. The four branches end up to anchors at P.B. of V.C. at EL. 1068'-7".
202 208	Safety Injection Piping	CS	From 2 anchors at P.B. of V.C. to 10" PRSL-152-1, LPSI accumulator tank, Disch. LPSI-1, LPSI-2, LPSI-3 and 6" PRSL-302-1.
205	Safety Injection Piping	CS	From nozzle at LPSI accumulation tank to 3 vents to atmosphere at EL. 1064'-0".

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
206	Safety Injection Piping	CS	From nozzles at LPSI accumulation tank and expansion tank at nozzles at nitrogen storage tank #1 through tank #18.
203	Safety Injection Piping	CS	From anchor at P.B. of V.C. at EL. 1060'-4-7/16" to anchor A-1 on line 4" HPSI-902-8
209	Safety Injection Piping	CS	From anchor H23 on line 4"-HPSI-902-8 to nozzles at disch. HP-1, HP-2 and HP-3.
204	Safety Injection Piping	CS	From anchor and anchor H30 on line 10" PRSL-152-1.
210	Safety Injection Piping	CS	From anchor H30 on 10" PRSL-152-1 to suction nozzles on HP-1, HP-2 and HP-3. Also at suction nozzles on LPSI-1, LPSI-2 & LPSI-3.

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
211	Safety Injection Piping	CS	On one side bounded by anchor A-1 and H23 on 4" HPSI-902-8. Other ends are bounded by a seal plate anchor and lines 8" PRSL-302-9, 4" PRSL-302-9, 4" PRSL-302-15, 4" PRSL-302-14, 2" HPSI-902-8, 2" HPSI-902-12, and 2" HPSI-902-1.
261	Fuel Chute Tranfer Piping	CS	From anchor at P.B. at V.C. to fuel chute dewatering pump nozzle P-34.
262	Fuel Chute Tranfer Piping	CS	From connection point on a 12" line to fuel chute dewatering pump P-34. A branch goes to pipe nipple for LS-232.
300	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1064' to main coolant pump P14-1.
301	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1064'-7" to main coolant pump P14-2.
302	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1065'-3" to main coolant pump P14-3.

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
303	Componnt Cooling	CS	From anchor at P.B. of V.C. EL. 1063'-11" to main coolant pump P14-4.
305	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1058'-7" to connection on line 6"-K-151-12.
306	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1059'-4" to connection on line 6"-K-151-12.
307	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1059'-11" to connection on line 6"-K-151-12.
308	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1058'-9" to anchor on 6"-K-151-12.
314	Component Cooling	CS	From anchor at P.B. of V.C. EL 1058'-9" to nozzle on tank TK-2. Also from connection on line 6"-K-151-25 to proposed anchor in P.A.B..

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
315	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1059'-4" to K-151-25 and a proposed anchor in P.A.B.
316 317	Component Cooling	CS	From 2 anchors at P.B. of V.C. EL. 1059' to the anchor on 6"-K-151-25 and an anchor on line 4"-BL-151-1.
350	Component Cooling	CS	From proposed anchor on 10"-K-151-7 to nozzles at component cooling exchangers E-9, E-10 and E-12.
318	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1058'-9" to the anchor on line 6"-K-151-12.
319	Component Cooling	CS	From anchor at P.B. of V.C. EL. 1059' to anchor on line 6"-K-151-25.
320	Component Cooling	CS	From proposed anchor on 10"-K-151-7 to anchor on 3"-K-151-21, proposed anchor on 6"-K-151-12 and component cooling exchanger E-11-1 and E-11-2.

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
321	Component Cooling	CS	From component cooling exchanger E-12 to proposed anchor on 10" line at EL. 1028'-0" to component cooling exchangers E-9 and E-10.
322	Component Cooling	CS	From 2 nozzles at E-11-1, E-11-2 to 2 nozzles at P-20-1, P-20-2.
400	Service Water	CS	From 3 nozzles P-6-1, P-6-2, P-6-3 to 4 anchors, which are 2 steel rings embedded in walls and 2 proposed anchors on 6"-WS-151-7 and 6"-WS-151-6.
401 402	Service Water Lines	CS	Burried pipes from a turbine room to screenwell and pump house.
403 404	Service Water	CS	From turbine oil coolers E-29-1 and E-29-2 to 4 anchors at slabs 2 at EL. 1023'-8" and 2 at EL. 1022'-8". 2 proposed anchors below valves SW-V-681 and SW-V-682. 1 proposed anchor on 6" WS-151-47 and another proposed anchor at EL. 1043'-7-5/8".

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
351	Component Cooling	CS	From proposed anchor at EL. 1028' to 2 nozzles at component cooling pumps P-20-1 and P-20-2. Also 2 proposed anchors. One on 3"-K-151-34 and one on 6"-K-151-25.
405	Service Water Line	CS	Buried pipe from turbine building to P.A.B..
406	Service Water Line	CS	Buried pipe from turbine building to P.A.B..
407 408	Service Water Line	CS	From heat exchangers E-11-1 and E-11-2 to proposed anchors on slabs at ground elevation. Also 2 proposed anchors on lines 4"-WS-151-97 and 4"-WS-151-98 and one proposed anchor south of line 3"-WS-152-85.
352	Component Cooling	CS	From anchor on slab at EL. 1022'-8" to analytical anchor on 10"-K-151-2 and nozzle at component cooling water surge tank TK-2.

PROBLEM NO.	SYSTEM DESCRIPTION	SCOPE	SYSTEM BOUNDARY
410 412 418	Service Water	CS	From 2 anchors at P.B. of V.C. at EL. 1065'-5" to anchor A-4 on line 6" WS-151-110.
409 411 413	Service Water	CS	From 2 anchors at P.B. of V.C. at EL. 1064'-7" to anchor 7L-A-3 on line 6"-WS-151-99.
414	Service Water	CS	From anchor at P.B. of V.C. at EL. 1065'-5" to connection on line 6" WS-151-110.
409 415 417	Service Water	CS	From nozzles at booster pumps P-51-1, P-51-2 to anchor 7L-A-3 on line 6"-WS-151-99 to 2 anchors at P.B. of V.C. EL. 1065'-5".
416 418	Service Water	CS	From between anchor A-4 and proposed anchor on slab. EL. 1039'-6" to anchor at P.B. of V.C. EL. 1065'-5".
500 501	V.C. Ventilation and Purge	CS	From anchor at P.B. of V.C. EL. 1079'-2" to anchor at P.B. of V.C. at EL. 1079'-2".
502	V.C. Ventilation and Purge	CS	From anchor at P.B. of V.C. EL. 1079'-2" to support "C".

SPECTRA CONSIDERED TO DATE
FOR HOT SHUTDOWN SCOPE

- IDBS - Interim Design Basis Spectra
- YCS Yankee Composite Spectra
- NRC - NRC Specified Spectra

LOAD CASES CONSIDERED

- WEIGHT
- PRESSURE
- THERMAL
- SEISMIC
- SEISMIC ANCHOR MOVEMENTS
- THERMAL ANCHOR MOVEMENTS

COMPUTER PROGRAMS USED

- ANSYS, REV. 3, UPDATE 67L
- ADL PIPE, REV. 3C, FEBRUARY 1977
- PROGRAMS SUPPORTED AND VERIFIED

CRITERIA USED

- ANSI B 31.1 - POWER PIPING CODE, 1977
- U.S. NUCLEAR REGULATORY GUIDES
  1.60, REV. 1; 1.61, REV. 0
  1.92, REV. 1; AND 1.122, REV. 1
- ASME BOILER & PRESSURE VESSEL CODE, 1971
  INCLUDING CODE CASE 1607 FOR PEAK BROADENING
- U.S. NRC REG. GUIDE, NUREG/CR-0098, MAY 1978
- YAEC DESIGN CRITERIA, DC-1, REV. 2
  JOB NO. 80023

LOAD COMBINATIONS

EQ. 11: PRESSURE + WEIGHT < S_H

HOT SHUTDOWN SCOPE

EQ. 12: PRESSURE + WEIGHT + SEISMIC INERTIA < KS_H

FOR SEISMIC INERTIA DEFINED BY YCS,
"CODE ALLOWABLE" LIMIT K = 1.8

FOR SEISMIC INERTIA DEFINED BY NRC,
"FUNCTION" EVALUATION CONSIDERED K = 2.4

REMAINING SYSTEMS

EQ. 12: PRESSURE + WEIGHT + SEISMIC INERTIA < KS_H

WITH SEISMIC INERTIA DEFINED BY YCS,
"FUNCTION" EVALUATION CONSIDERED K = 2.4

EQ. 13: THERMAL EXPANSION + TAM + SAM < S_A

EQ. 14: PRESSURE + WEIGHT + THERMAL EXPANSION + TAM + SAM < S_A + S_H

ARTICLE F-1000
RULES FOR EVALUATION OF SERVICE LOADINGS WITH LEVEL D SERVICE LIMITS

F-1100 INTRODUCTION

F-1110 SCOPE

In this Appendix are given rules which may be used by Owners and Manufacturers with respect to evaluation of those Service Loadings for which Level D Service Limits are specified by the Design Specifications (NCA-3250).

F-1120 DEFINITION OF LEVEL D SERVICE LIMITS

Level D Service Limits are those sets of limits which must be satisfied for all loadings identified in the Design Specifications for which these Service Limits are designated. These sets of limits permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of the component from service. These sets of limits are permitted for combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that conditions of public health and safety are involved. Therefore, the selection of this limit shall be reviewed by the Owner for compatibility with established system safety criteria (NCA-2141).

F-1130 APPLICABILITY

This Appendix shall be considered for Class 1 components and supports, and Class CS core support structures if Level D Service Limits (NB-3225) are given in the Design Specification (NCA-2144).

F-1200 INTENT OF LEVEL D SERVICE LIMITS CONSIDERATION

F-1210 LIMITS OF CODE CONSIDERATIONS

Components are designed to provide a pressure containing barrier or to act as a pressure retaining member in the system or to act as core support structures.

F-1220 LIMITS OF DESIGN PROCEDURES

(a) The Level D Service Limits design procedures contained in F-1300 are provided for limiting the consequences of the specified event. They are intended (NCA-1130) to ensure that violation of the pressure retaining boundary will not occur in components or supports which are in compliance with these procedures. These procedures are not intended to ensure the safe operability or reoperability of the system either during or following the postulated event.

(b) In addition, the procedures specifically identified for core support structures (F-1380) limit the consequences of the postulated event with respect to failure modes other than leakage.

(c) The procedures of F-1300 need not be applied to any portion of a component or support in which a failure has been postulated in defining the Level D Service Limits.

F-1300 DESIGN PROCEDURES FOR SERVICE LOADINGS WITH LEVEL D SERVICE LIMITS

F-1310 GENERAL

(a) These design procedures are provided to limit the consequences of the Service Loadings for which Level D Service Limits are included in the Design Specifications.

(b) The contents of F-1320 provide general procedures which are applicable to all components. Specific procedures, which may limit the applicability of the procedures of F-1320, are provided as follows:

(1) F-1330 Vessels
(2) F-1340 Pumps
(3) F-1350 Valves
(4) F-1360 Piping
(5) F-1370 Component Supports
(6) F-1380 Core Support Structures
(7) F-1390 Class MC Vessels

(c) Only limits on primary stresses are prescribed. Thermal stresses need not be considered.

(d) When compressive stresses are present the stability of the component must be ensured (F-1325).

(e) Potential for unstable crack growth should also be considered basing assumed defects on the inspection techniques employed.

F-1320 DESIGN BY ANALYSIS

F-1321 Terms Related to Analysis

In addition to the terms related to stress analyses, the following terms are defined.

F-1321.1 Plastic Analysis

(a) Plastic analysis is that method which computes the structural behavior under given loads considering the strain hardening characteristics of the materials, strain rate effects, permanent deformations, and stress redistributions occurring in the structure. A plastic analysis is primarily distinguished from a limit analysis (NB-3213.21) because the actual strain hardening characteristics of the material are considered in performing a plastic analysis.

(b) The true stress-strain curve shall be adjusted to correspond to the tabulated value at the appropriate temperature in Table I-2.1 or I-2.2 and shall be included and justified in the Stress Report. However, strain rate effects on the flow curve may also be considered.

(c) The yield criteria and associated flow rule used in performing a plastic analysis may be either those associated with the maximum shear stress or the distortion energy method.

(d) A plastic analysis may be used to determine the collapse load for a given combination of loads on a given structure. The collapse load shall be taken as the one at which the distortion is two times the value at the calculated initial departure from linearity. In evaluating the analysis, the computations should be interpreted in a manner consistent with that used in an experiment (II-1430). When interpreted in this manner, the collapse load which results from the plastic analysis shall be limited in accordance with procedures applied to limit analyses. The symbol applied to the collapse load is P_C.

(e) A plastic analysis may be used to determine the plastic instability load for a given combination of loads on a given structure. The plastic instability load is taken as the one at which the deformation increases without bound or the relation of force and deformation has a horizontal tangent. The symbol used to designate this plastic instability load is P_I.

(f) A plastic analysis may be used to determine the load or combination of loads which result in a particular strain within the structure. When a limit is placed upon a strain, the load associated with the strain limit will be designated by the symbol P_S.

F-1321.2 Stress Ratio Method

(a) The stress ratio method is a pseudo-elastic analysis method which may be used as an approximate plastic analysis when the required interaction equations or curves are available. Such information is included in A-9000.

(b) The equations and curves of the stress ratio method are developed for specific configurations, for specific loading combinations, and for specific materials, considering the strain hardening characteristics of the material. The method may be used for statically or dynamically applied loads.

(c) The stress ratio method may be used to determine the maximum loads which may be carried by the structure without exceeding an assigned apparent stress. The symbol used to designate this load is P_R.

F-1321.3 Experimental Method

(a) Experimental investigations may be performed in accordance with II-1220 and interpreted in accordance with II-1430 to determine the collapse load. The symbol applied to the collapse load is P_C.

(b) Experimental investigations may be performed to determine the plastic instability load for a structure. If failure occurs before plastic instability is experienced, the failure load shall be used. The symbol used to designate this plastic instability load is P_I.

(b) certain thermal stresses which may cause fatigue but not distortion;

(c) the stress at a local structural discontinuity;

(d) surface stresses produced by thermal shock.

NB-3213.12 Load Controlled Stresses. Load controlled stresses are the stresses resulting from application of a loading, such as internal pressure, inertial loads, or the effects of gravity, whose magnitude is not reduced as a result of displacement.

NB-3213.13 Thermal Stress. Thermal stress is a self-balancing stress produced by a nonuniform distribution of temperature or by differing thermal coefficients of expansion. Thermal stress is developed in a solid body whenever a volume of material is prevented from assuming the size and shape that it normally should under a change in temperature. For the purpose of establishing allowable stresses, two types of thermal stress are recognized, depending on the volume or area in which distortion takes place, as described in (a) and (b) below.

(a) General thermal stress is associated with distortion of the structure in which it occurs. If a stress of this type, neglecting stress concentrations, exceeds twice the yield strength of the material, the elastic analysis may be invalid and successive thermal cycles may produce incremental distortion. Therefore this type is classified as secondary stress in Table NB-3217-1. Examples of general thermal stresses are:

(1) stress produced by an axial temperature distribution in a cylindrical shell;

(2) stress produced by the temperature difference between a nozzle and the shell to which it is attached;

(3) the equivalent linear stress[2] produced by the radial temperature distribution in a cylindrical shell.

(b) Local thermal stress is associated with almost complete suppression of the differential expansion and thus produces no significant distortion. Such stresses shall be considered only from the fatigue standpoint and are therefore classified as local stresses in Table NB-3217-1. In evaluating local thermal stresses the procedures of NB-3228.1(c) shall be used. Examples of local thermal stresses are:

(1) the stress in a small hot spot in a vessel wall;

(2) the difference between the actual stress and the equivalent linear stress resulting from a radial temperature distribution in a cylindrical shell;

(3) the thermal stress in a cladding material which has a coefficient of expansion different from that of the base metal.

[2] *Equivalent linear stress* is defined as the linear stress distribution which has the same net bending moment as the actual stress distribution.

NB-3213.14 Total Stress. Total stress is the sum of the primary, secondary, and peak stress contributions. Recognition of each of the individual contributions is essential to establishment of appropriate stress limitations.

NB-3213.15 Operational Cycle. Operational cycle is defined as the initiation and establishment of new conditions followed by a return to the conditions which prevailed at the beginning of the cycle. The types of operating conditions which may occur are further defined in NB-3113.

NB-3213.16 Stress Cycle. Stress cycle is a condition in which the alternating stress difference [NB-3222.4(e)] goes from an initial value through an algebraic maximum value and an algebraic minimum value and then returns to the initial value. A single operational cycle may result in one or more stress cycles. Dynamic effects shall also be considered as stress cycles.

NB-3213.17 Fatigue Strength Reduction Factor. Fatigue strength reduction factor is a stress intensification factor which accounts for the effect of a local structural discontinuity (stress concentration) on the fatigue strength. Values for some specific cases, based on experiment, are given in NB-3338 and NB-3339. In the absence of experimental data, the theoretical stress concentration factor may be used.

NB-3213.18 Free End Displacement. Free end displacement consists of the relative motions that would occur between a fixed attachment and connected piping if the two members were separated and permitted to move.

NB-3213.19 Expansion Stresses. Expansion stresses are those stresses resulting from restraint of free end displacement of the piping system.

NB-3213.20 Deformation. Deformation of a component part is an alteration of its shape or size.

NB-3213.21 Inelasticity. Inelasticity is a general characteristic of material behavior in which the material does not return to its original shape and size after removal of all applied loads. Plasticity and creep are special cases of inelasticity.

NB-3213.22 Creep. Creep is the special case of inelasticity that relates to the stress-induced, time-dependent deformation under load. Small time-dependent deformations may occur after the removal of all applied loads.

bution through a fractional part of the wall thickness. The stress distribution associated with a local discontinuity causes only very localized types of deformation or strain and has no significant effect on the shell type discontinuity deformations. Examples are small fillet radii, small attachments, and partial penetration welds.

NB-3213.4 Normal Stress. Normal stress is the component of stress normal to the plane of reference. This is also referred to as direct stress. Usually the distribution of normal stress is not uniform through the thickness of a part, so this stress is considered to be made up in turn of two components, one of which is uniformly distributed and equal to the average value of stress across the thickness under consideration, and the other of which varies from this average value with the location across the thickness.

NB-3213.5 Shear Stress. Shear stress is the component of stress tangent to the plane of reference.

NB-3213.6 Membrane Stress. Membrane stress is the component of normal stress which is uniformly distributed and equal to the average value of stress across the thickness of the section under consideration.

NB-3213.7 Bending Stress. Bending stress is the variable component of normal stress described in NB-3213.4. The variation may or may not be linear across the thickness.

NB-3213.8 Primary Stress. Primary stress is any normal stress or a shear stress developed by an imposed loading which is necessary to satisfy the laws of equilibrium of external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. Primary stresses which considerably exceed the yield strength will result in failure or, at least, in gross distortion. A thermal stress is not classified as a primary stress. Primary membrane stress is divided into general and local categories. A general primary membrane stress is one which is so distributed in the structure that no redistribution of load occurs as a result of yielding. Examples of primary stresses are:

(a) general membrane stress in a circular cylindrical or a spherical shell due to internal pressure or to distributed live loads;

(b) bending stress in the central portion of a flat head due to pressure.

NB-3213.9 Secondary Stress. Secondary stress is a normal stress or a shear stress developed by the constraint of adjacent material or by self-constraint of the structure. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the conditions which cause the stress to occur and failure from one application of the stress is not to be expected. Examples of secondary stresses are:

(a) general thermal stress [NB-3213.13(a)];

(b) bending stress at a gross structural discontinuity.

NB-3213.10 Local Primary Membrane Stress. Cases arise in which a membrane stress produced by pressure or other mechanical loading and associated with a primary or a discontinuity effect produces excessive distortion in the transfer of load to other portions of the structure. Conservatism requires that such a stress be classified as a local primary membrane stress even though it has some characteristics of a secondary stress. A stressed region may be considered local if the distance over which the membrane stress intensity exceeds $1.1S_m$ does not extend in the meridional direction more than $1.0\sqrt{Rt}$, where R is the minimum midsurface radius of curvature and t is the minimum thickness in the region considered. Regions of local primary stress intensity involving axisymmetric membrane stress distributions which exceed $1.1S_m$ shall not be closer in the meridional direction than $2.5\sqrt{Rt}$, where R is defined as $(R_1 + R_2)/2$ and t is defined as $(t_1 + t_2)/2$ (where t_1 and t_2 are the minimum thicknesses at each of the regions considered, and R_1 and R_2 are the minimum midsurface radii of curvature at these regions where the membrane stress intensity exceeds $1.1S_m$). Discrete regions of local primary membrane stress intensity, such as those resulting from concentrated loads acting on brackets, where the membrane stress intensity exceeds $1.1S_m$, shall be spaced so that there is no overlapping of the areas in which the membrane stress intensity exceeds $1.1S_m$.

NB-3213.11 Peak Stress. Peak stress is that increment of stress which is additive to the primary plus secondary stresses by reason of local discontinuities or local thermal stress [NB-3213.13(b)] including the effects, if any, of stress concentrations. The basic characteristic of a peak stress is that it does not cause any noticeable distortion and is objectionable only as a possible source of a fatigue crack or a brittle fracture. A stress which is not highly localized falls into this category if it is of a type which cannot cause noticeable distortion. Examples of peak stresses are:

(a) the thermal stress in the austenitic steel cladding of a carbon steel component;

TABLE NB-3217-2
CLASSIFICATION OF STRESS INTENSITY IN PIPING, TYPICAL CASES

Piping Component	Locations	Origin of Stress	Classification	Discontinuities Considered: Gross	Discontinuities Considered: Local
Pipe or tube, elbows, and reducers. Intersections and branch connections except in crotch regions	Any, except crotch regions of intersections	Internal pressure	P_m	No	No
			P_L and Q	Yes	No
			F	Yes	Yes
		Sustained mechanical loads, including weight	P_b	No	No
			P_L and Q	Yes	No
			F	Yes	Yes
		Expansion	P_e	Yes	No
			F	Yes	Yes
		Axial thermal gradient	Q	Yes	No
			F	Yes	Yes
Intersections, including tees and branch connections	In crotch region	Internal pressure, sustained mechanical loads, and expansion	P_L and Q [Note (1)]	Yes	No
			F	Yes	Yes
		Axial thermal gradient	Q	Yes	No
			F	Yes	Yes
Bolts and flanges	Any	Internal pressure, gasket compression, and bolt load	P_m	No	No
			Q	Yes	No
			F	Yes	Yes
		Thermal gradient	Q	Yes	No
			F	Yes	Yes
		Expansion	P_e	Yes	No
			F	Yes	Yes
Any	Any	Nonlinear radial thermal gradient	F	Yes	Yes
		Linear radial thermal gradient	F	Yes	No
		Anchor point motions, including those resulting from earthquake	Q	Yes	No

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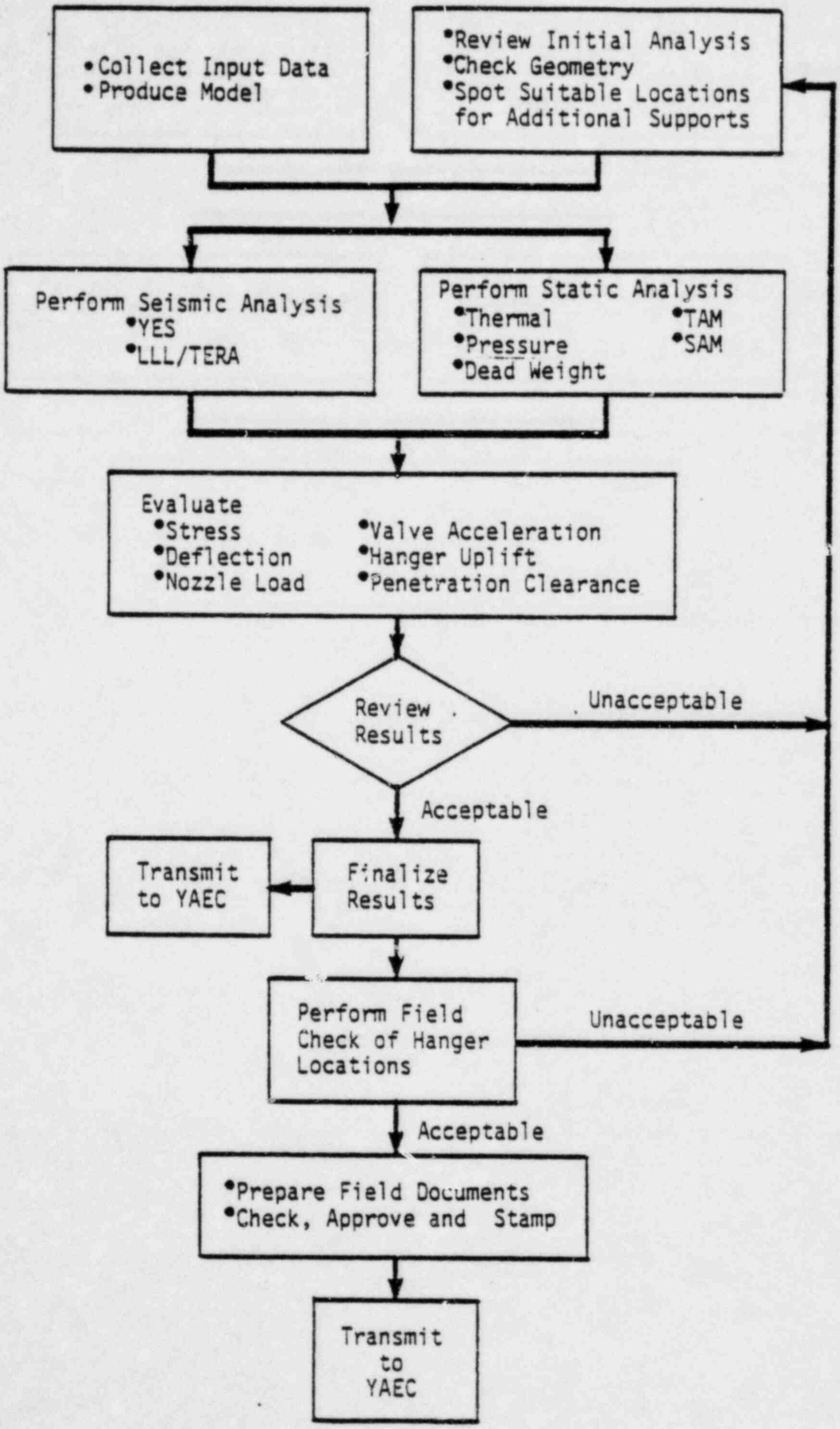
NOTE:
(1) Analysis is not required when reinforced in accordance with NB-3643.

solid rectangular sections, a value of α times the limit established in NB-3221.1 may be used, where the factor α is defined as the ratio of the load set producing a fully plastic section to the load set producing initial yielding in the extreme fibers of the section. In the evaluation of the initial yield and fully plastic section capacities, the ratios of each individual load in the respective load set to each other load in that load set shall be the same as the respective ratios of the individual loads in the specified design load set. The value of α shall not exceed the value calculated for bending only ($P_m = 0$). In no case shall the value of α exceed 1.5. The propensity for buckling of the part of the section that is in compression shall be investigated. The α factor is not permitted for Level D Service Limits when inelastic component analysis is used as permitted in Appendix F.

NB-3221.4 External Pressure. The provisions of NB-3133 apply.



FLOWCHART FOR ANALYSIS EFFORT

CONVENTIONAL SEISMIC ANALYSIS USING YCS
BEFORE FIXES WERE IMPLEMENTED

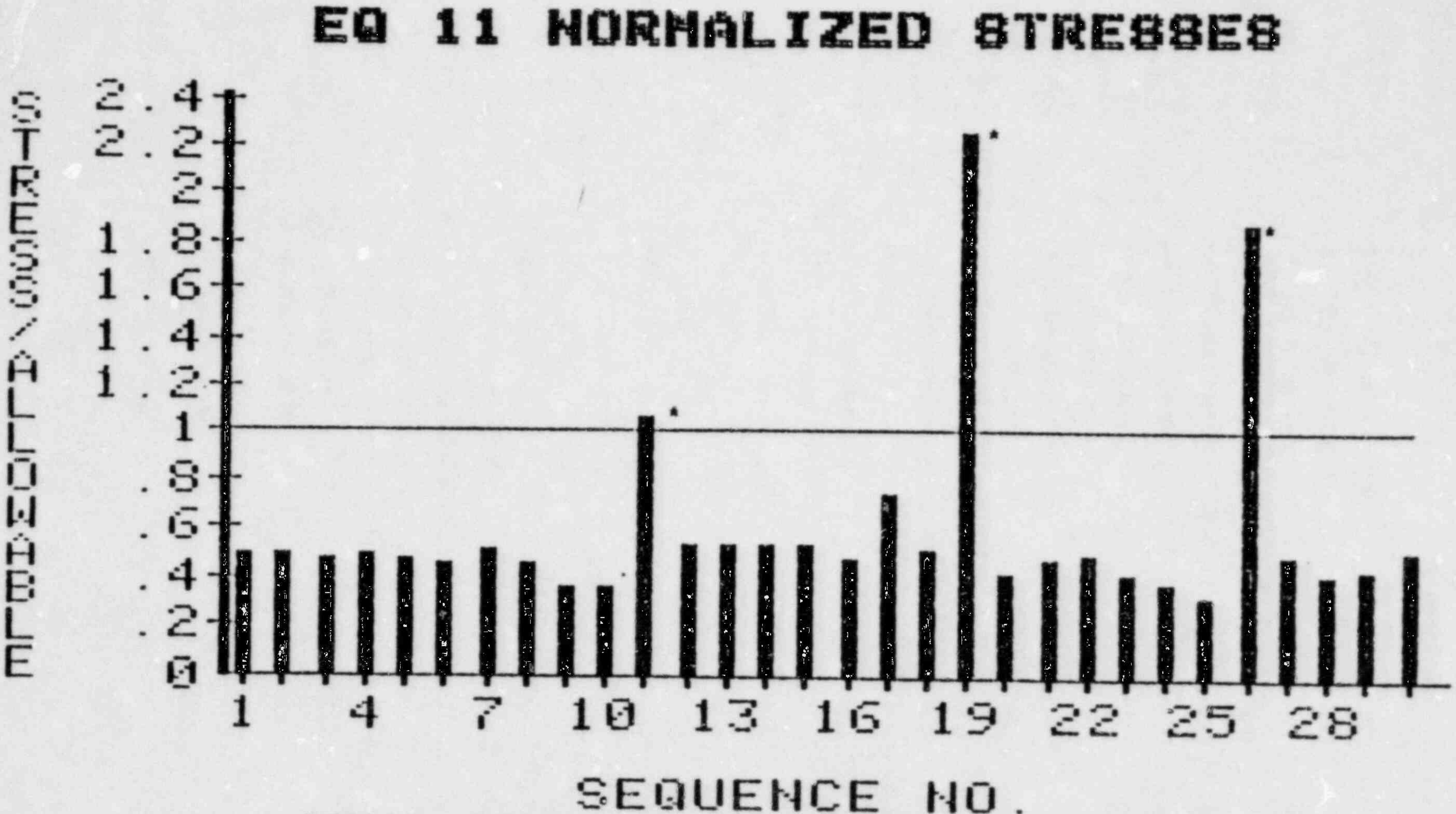
PROB. NO	SEQUENCE	SDMAX	TDMAX	NORM 11	NORM 12	NORM 13	NORM 14
1	1	8.371	1.09	0.49	1.26	0.46	0.43
2	2	7.635	1.09	0.49	1.22	0.47	0.32
3	3	4.446	.84	0.48	1.13	0.56	0.32
4	4	6.76	-1.22	0.49	0.90	0.45	0.31
21	5	9.8	.55	0.47	1.57	0.28	0.32
22	6	11.429	.64	0.45	1.79	0.30	0.31
23	7	14.42	.57	0.51	2.55	0.29	0.31
24	8	10.016	.63	0.44	1.59	0.30	0.32
41A	9	.087	-.38	0.36	0.20	0.42	0.32
41B	10	.11	-.37	0.35	0.20	0.34	0.29
41C	11	.11	-.32	1.06*	1.16	0.35	0.61
101	12	.279	-1.51	0.53	0.50	1.08	0.85
102	13	.417	1.58	0.53	0.56	0.55	0.54
103	14	.435	1.4	0.53	0.53	1.03	0.84
104	15	.397	-1.36	0.54	0.58	0.85	0.73
121	16	6.482	-.97	0.46	1.89	0.30	0.29
122	17	8.05	-.92	0.74	1.45	0.32	0.37
201	18	.0659	.07	0.51	0.36	2.68	1.88
207	19	2.78	2.9	2.25*	3.68	2.41	2.21
5	20	5.458	2.811	0.40	0.81	0.45	0.40
6	21	6.52	-5.37	0.47	1.28	0.99	0.74
7	22	6.59	-5.39	0.48	1.55	1.01	0.75
8	23	6.3	2.94	0.42	0.87	0.37	0.35
9	24	2.23	-3.03	0.38	1.22	1.15	0.83
10A :+:	25	10.13	-1.913	0.32	1.48	0.39	0.34
10B :+:	26	5.004	2.21	1.87*	2.55	3.11	1.89
25	27	13.00+	2.1	0.49	1.61	0.11	0.22
26	28	10.00+	-.99	0.42	1.69	0.09	0.20
27	29	9.00+	1.52	0.44	1.54	0.12	0.22
28	30	12.00+	1.24	0.50	1.70	0.12	0.20

HOT SHUTDOWN SYSTEMS

CYGNA

* FURTHURE FIELD INVESTIGATION REQUIRED
\+ CONSIDERING SLEEVE CLEARANCE
:+:SYSTEM DESIGN CHANGE PENDING

PRELIMINARY



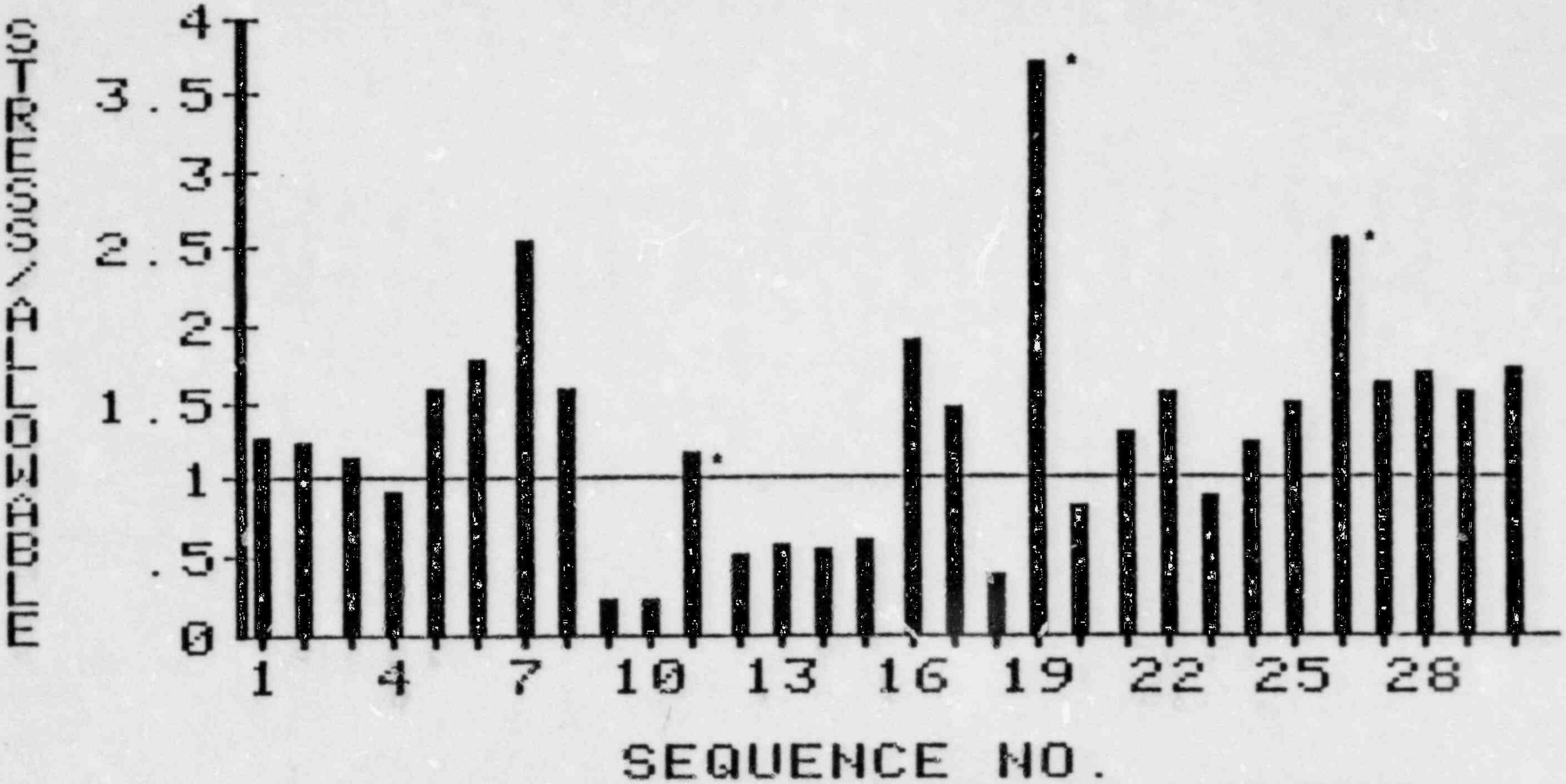
YCS ANALYSIS BEFORE FIXING HOT SHUTDOWN SYSTEMS

CYGNA

* SEE NOTES ON "HOT SHUTDOWN SYSTEM" STRESS AND DEFLECTION SUMMARY

PRELIMINA

EQ 12 NORMALIZED STRESSES



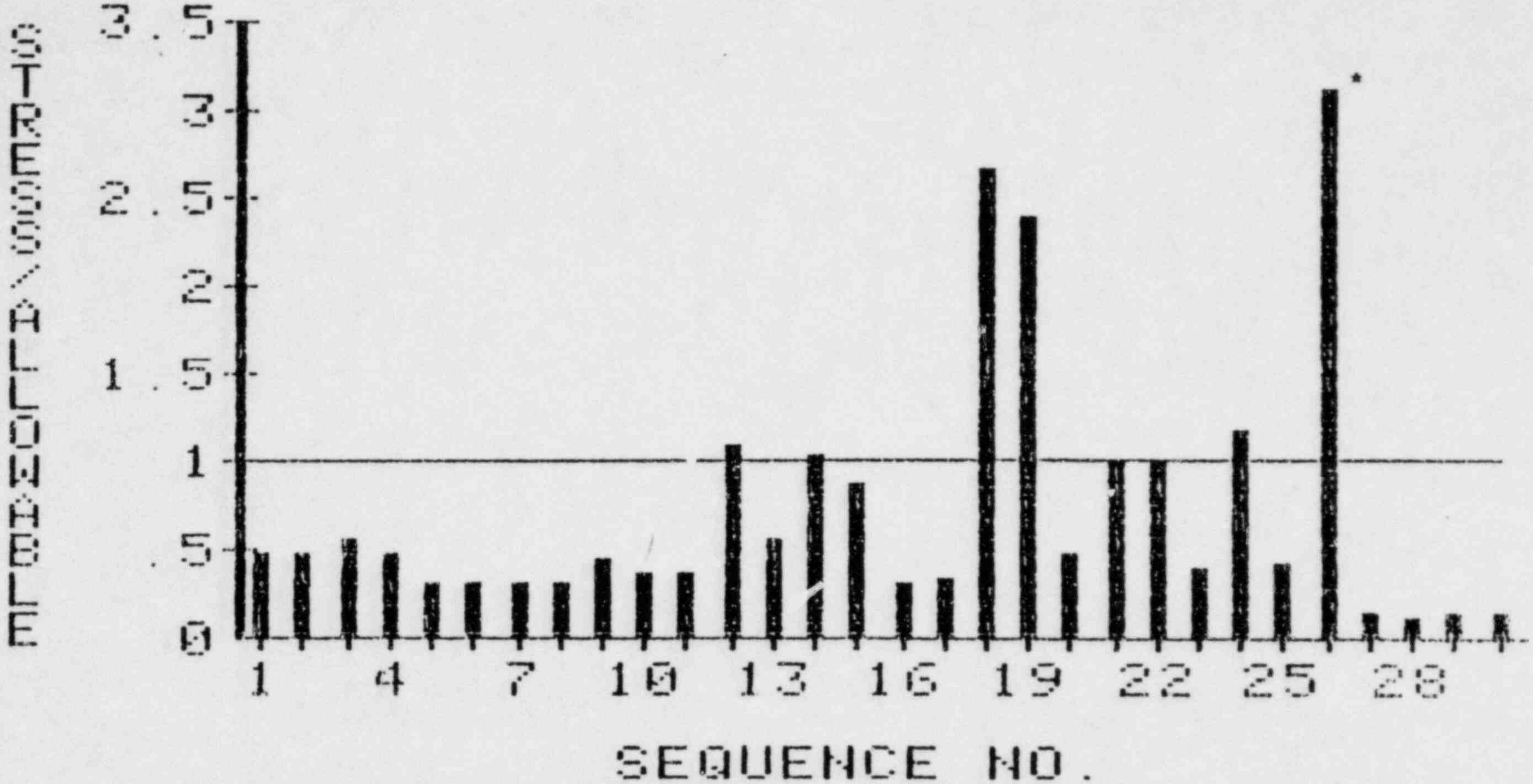
YCS ANALYSIS BEFORE FIXING HOT SHUTDOWN SYSTEMS

CYGNA

* SEE NOTES ON "HOT SHUTDOWN SYSTEM" STRESS AND DEFLECTION SUMMARY

PRELIMINARY

EQ 13 NORMALIZED STRESSES



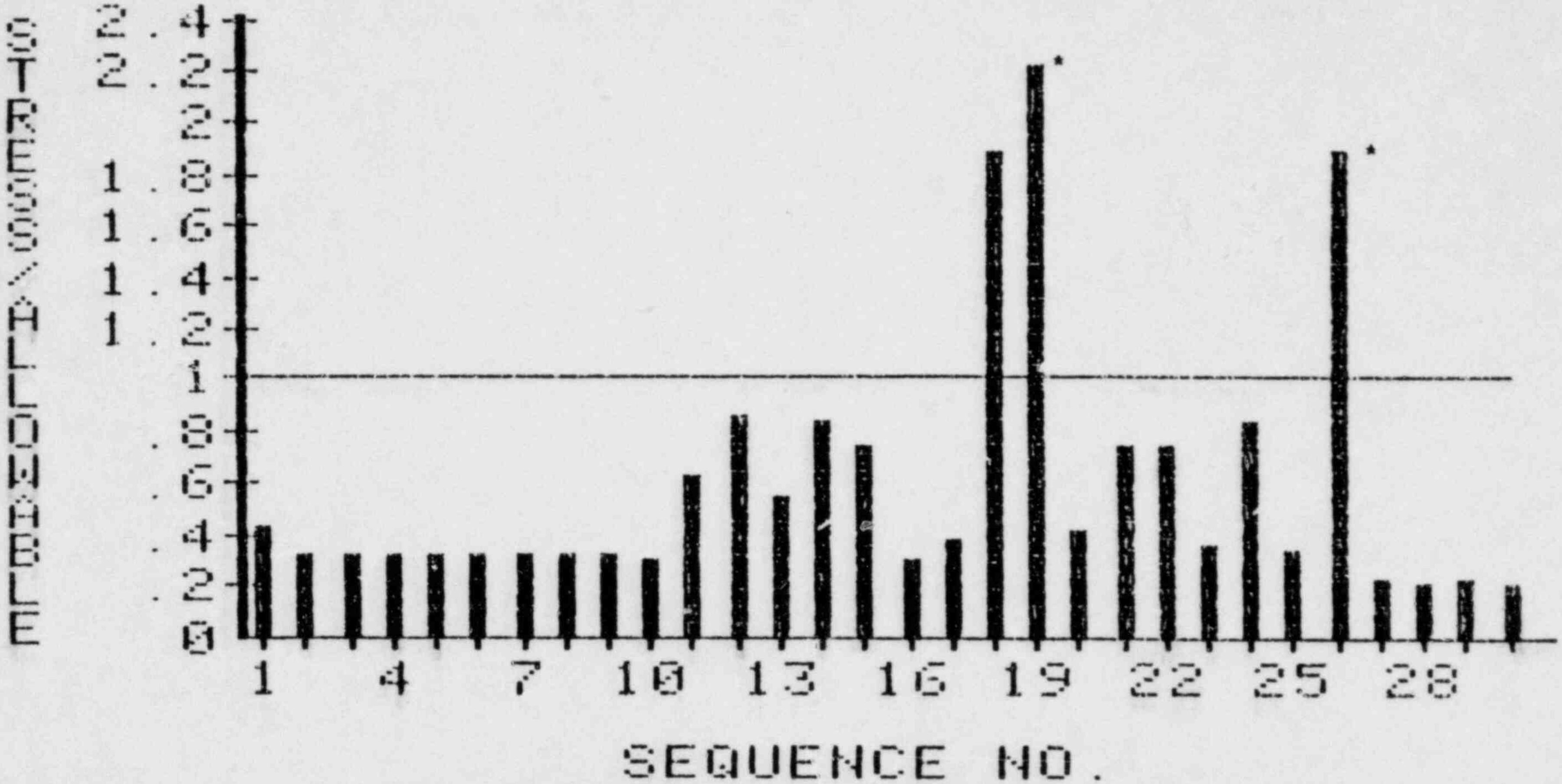
YCS ANALYSIS BEFORE FIXING HOT SHUTDOWN SYSTEMS

CYGNA

* SEE NOTES ON "HOT SHUTDOWN SYSTEM" STRESS AND DEFLECTION SUMMARY

PRELIMINARY

EQ 14 NORMALIZED STRESSES



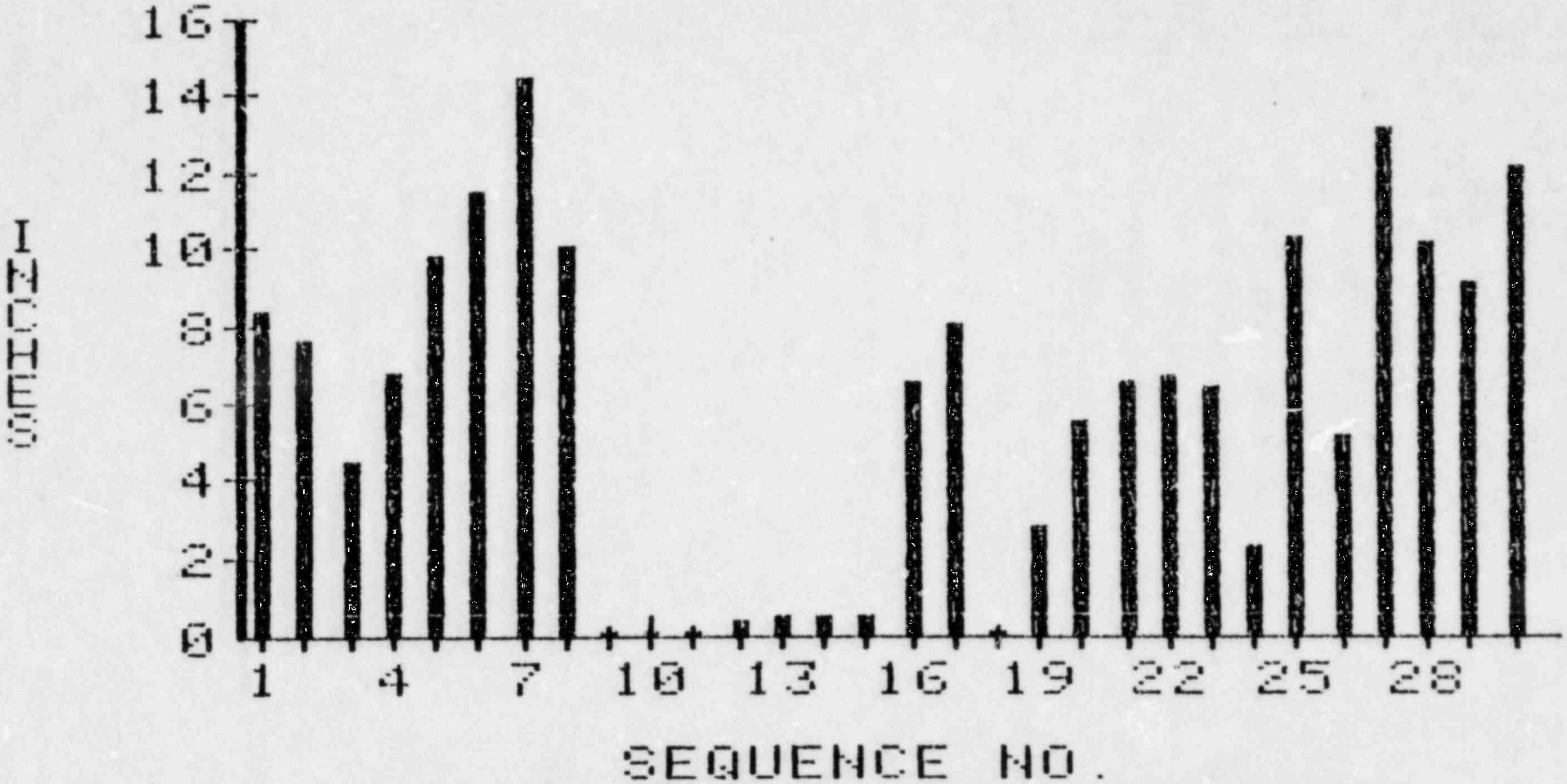
YCS ANALYSIS BEFORE FIXING HOT SHUTDOWN SYSTEMS

CYGNA

* SEE NOTES ON "HOT SHUTDOWN SYSTEM" STRESS AND DEFLECTION SUMMARY

PRELIMINARY

SEISMIC DEFLECTIONS

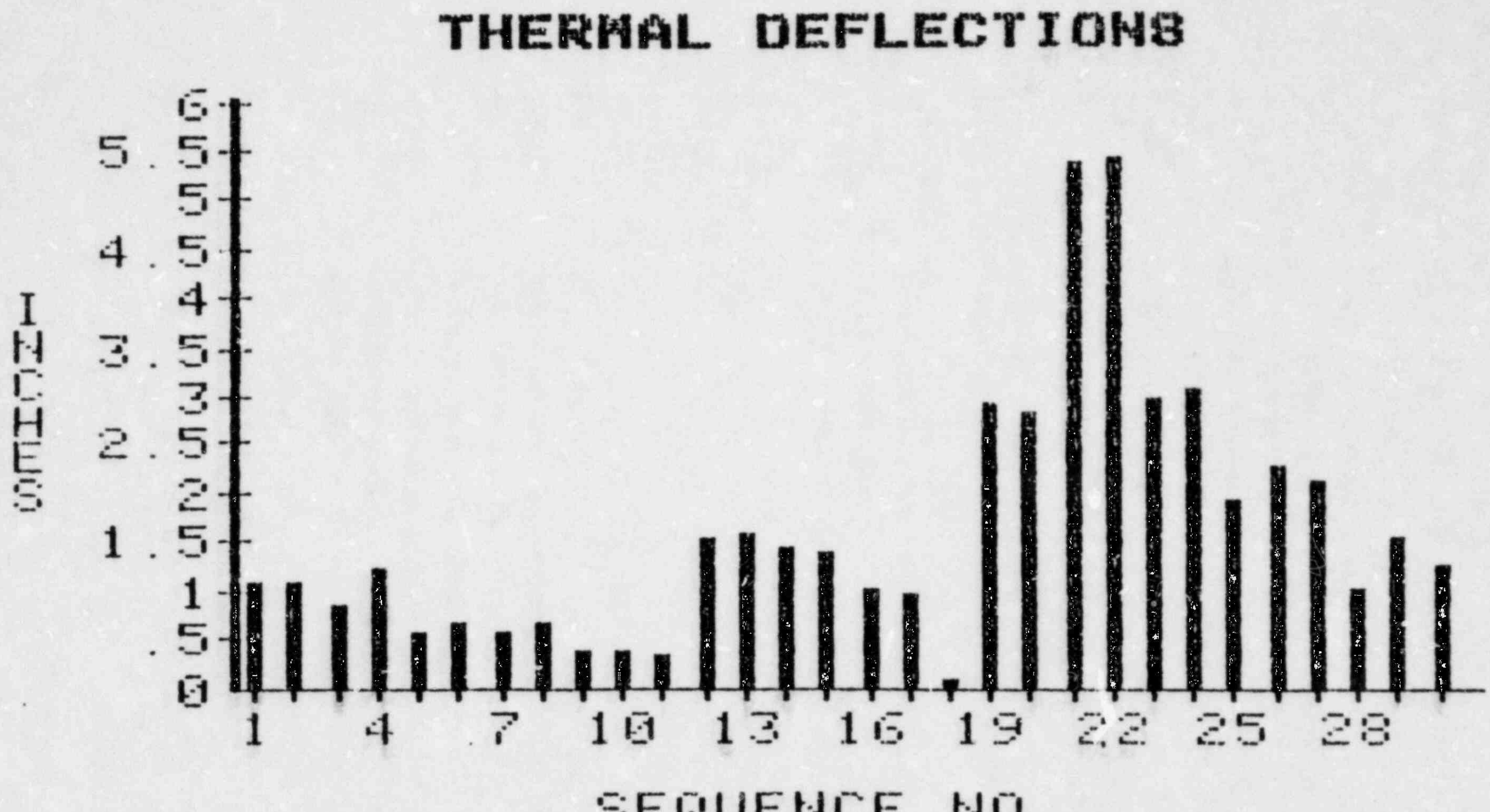


YCS ANALYSIS BEFORE FIXING HOT SHUTDOWN SYSTEMS

CYGNA

* SEE NOTES ON "HOT SHUTDOWN SYSTEM" STRESS AND DEFLECT[illegible]RY

PRELIMINARY



THERMAL ANALYSIS BEFORE FIXING HOT SHUTDOWN SYSTEMS

CYGNA

PRELIMINARY

SEISMIC ANALYSIS USING YCS
AFTER FIXES WERE IMPLEMENTED

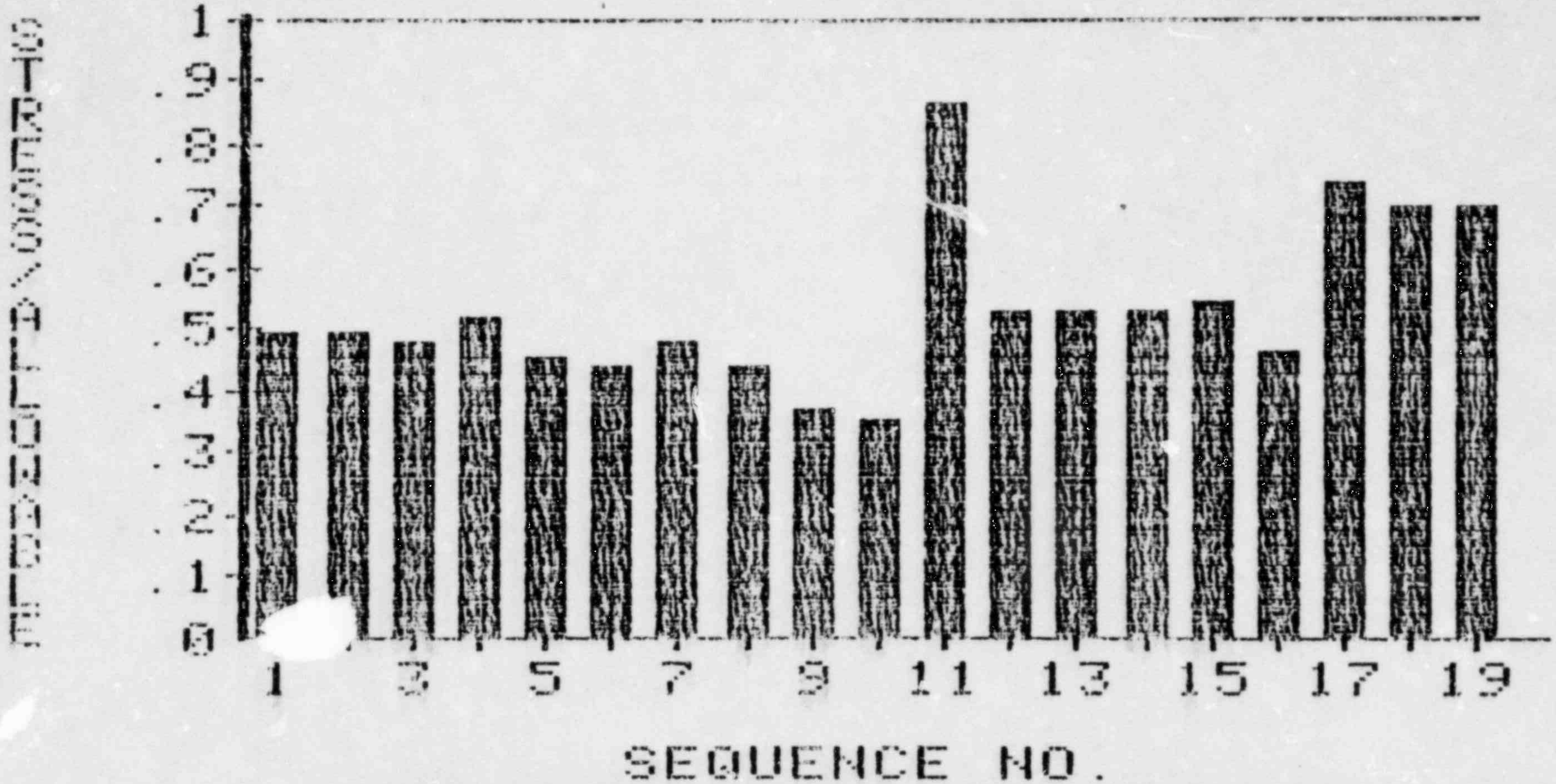
PROB. NO	SEQUENCE	SDMAX	TDMAX	NORM 11	NORM 12	NORM 13	NORM 14
1	1	1.20	2.04	0.49	0.53	0.60	0.56
2	2	1.22	2.04	0.49	0.53	0.60	0.56
3	3	2.63	2.20	0.48	0.82	0.80	0.67
4	4	1.06	2.08	0.51	0.51	0.64	0.59
21	5	1.00	2.06	0.45	0.68	1.29	0.95
22	6	0.30	2.06	0.44	0.35	1.33	0.93
23	7	0.50	2.06	0.48	0.43	1.00	0.73
24	8	1.03	2.05	0.43	0.71	1.16	0.83
41A	9	0.09	0.56	0.36	0.27	0.70	0.55
41B	10	0.11	0.60	0.35	0.26	0.70	0.60
41C	11	1.70	0.71	0.86	0.89	0.46	0.61
101	12	0.34	2.46	0.52	0.50	1.19	0.94
102	13	0.42	2.13	0.53	0.80	1.22	0.96
103	14	0.46	1.98	0.53	0.56	0.84	0.72
104	15	0.40	1.92	0.54	0.65	0.92	0.78
121	16	0.17	1.10	0.46	0.55	1.24	0.85
122	17	0.36	1.63	0.73	0.51	1.15	0.82
201	18	0.30	2.13	0.69	0.62	1.62	1.00
207	19	0.30	2.13	0.69	0.62	1.62	1.00

HOT SHUTDOWN INSIDE VC

PRELIMINARY

B-80

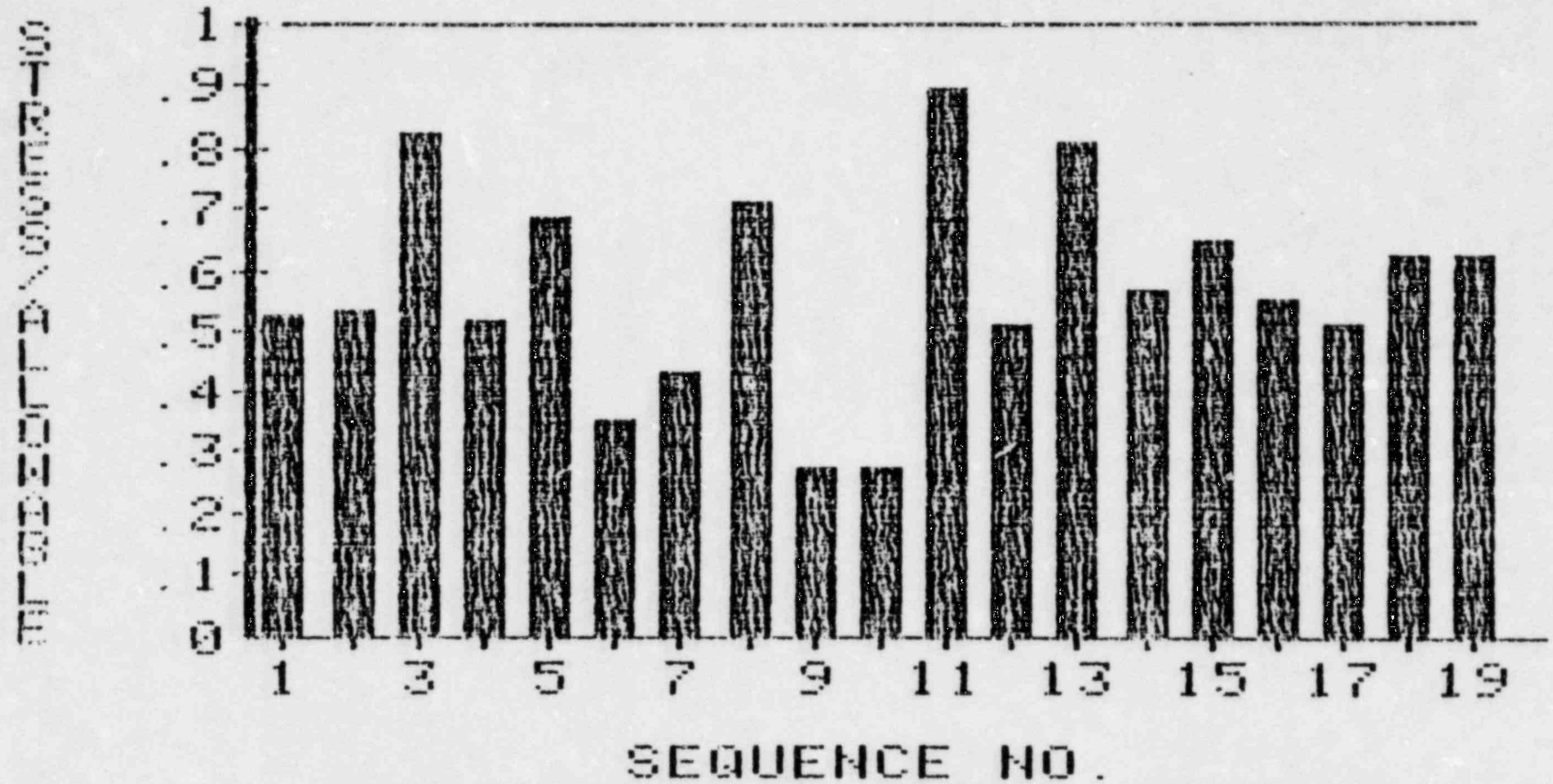
EQ 11 NORMALIZED STRESSES



YCS ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

PRELIMINARY

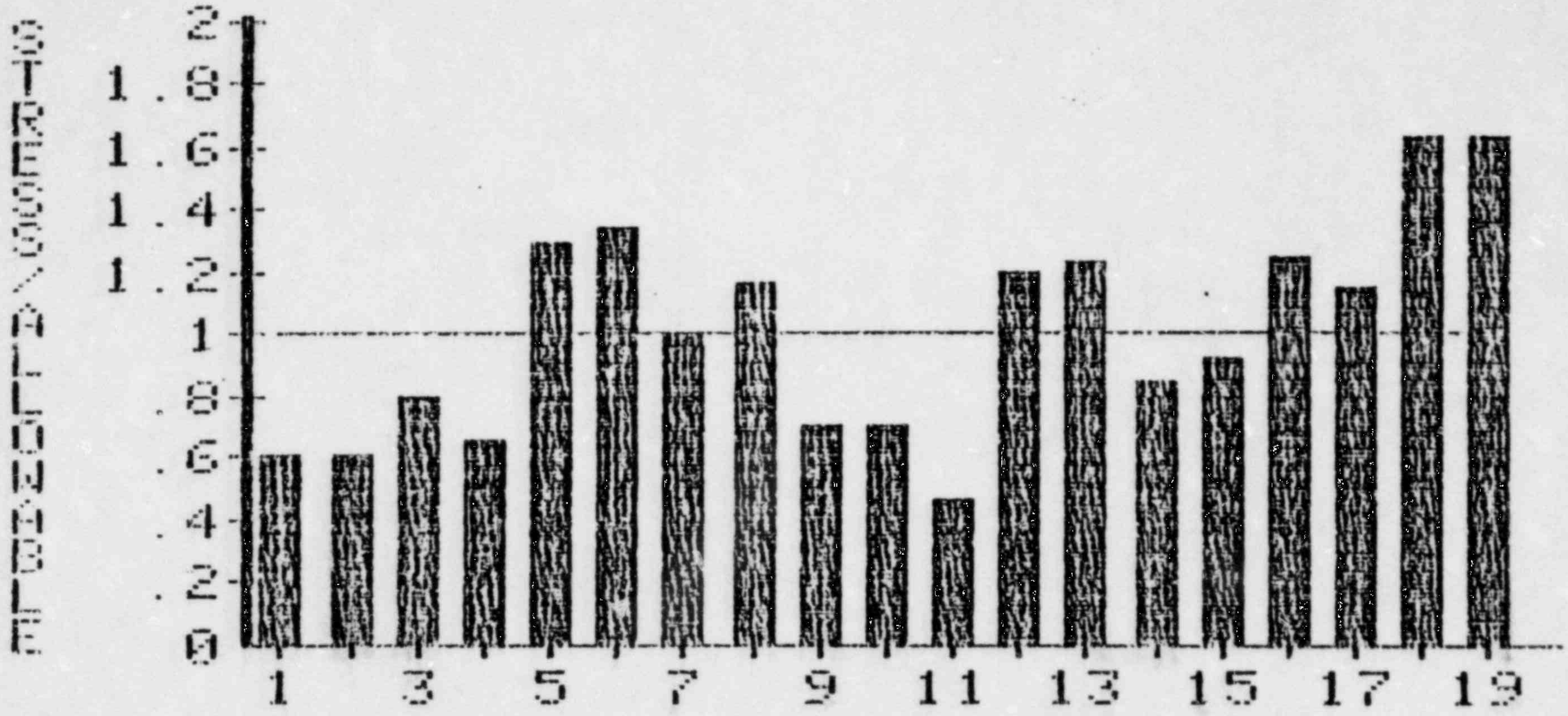
EQ 12 NORMALIZED STRESSES



YCS ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

PRELIMINARY

EQ 13 NORMALIZED STRESSES

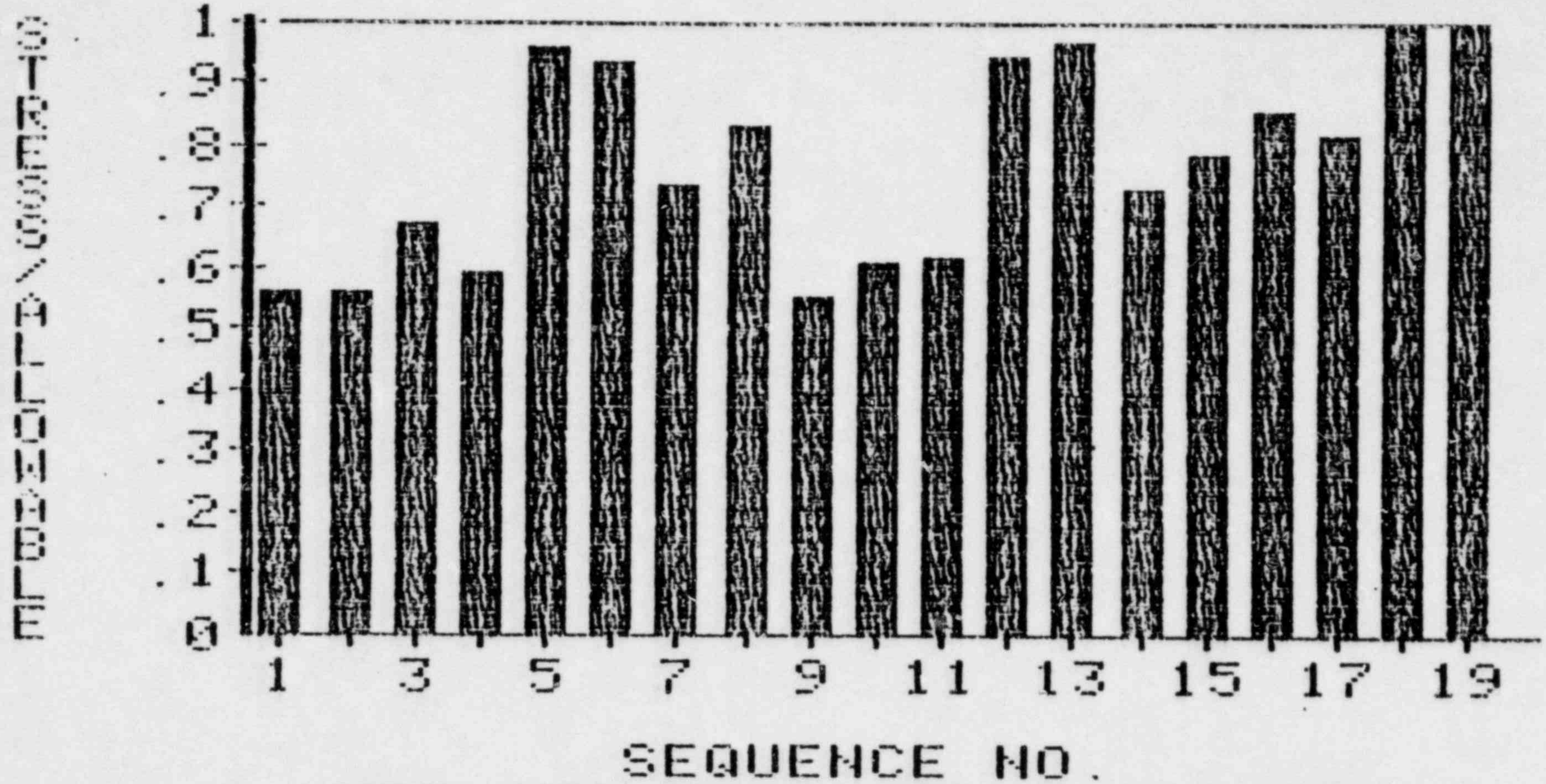


SEQUENCE NO.

YCB ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

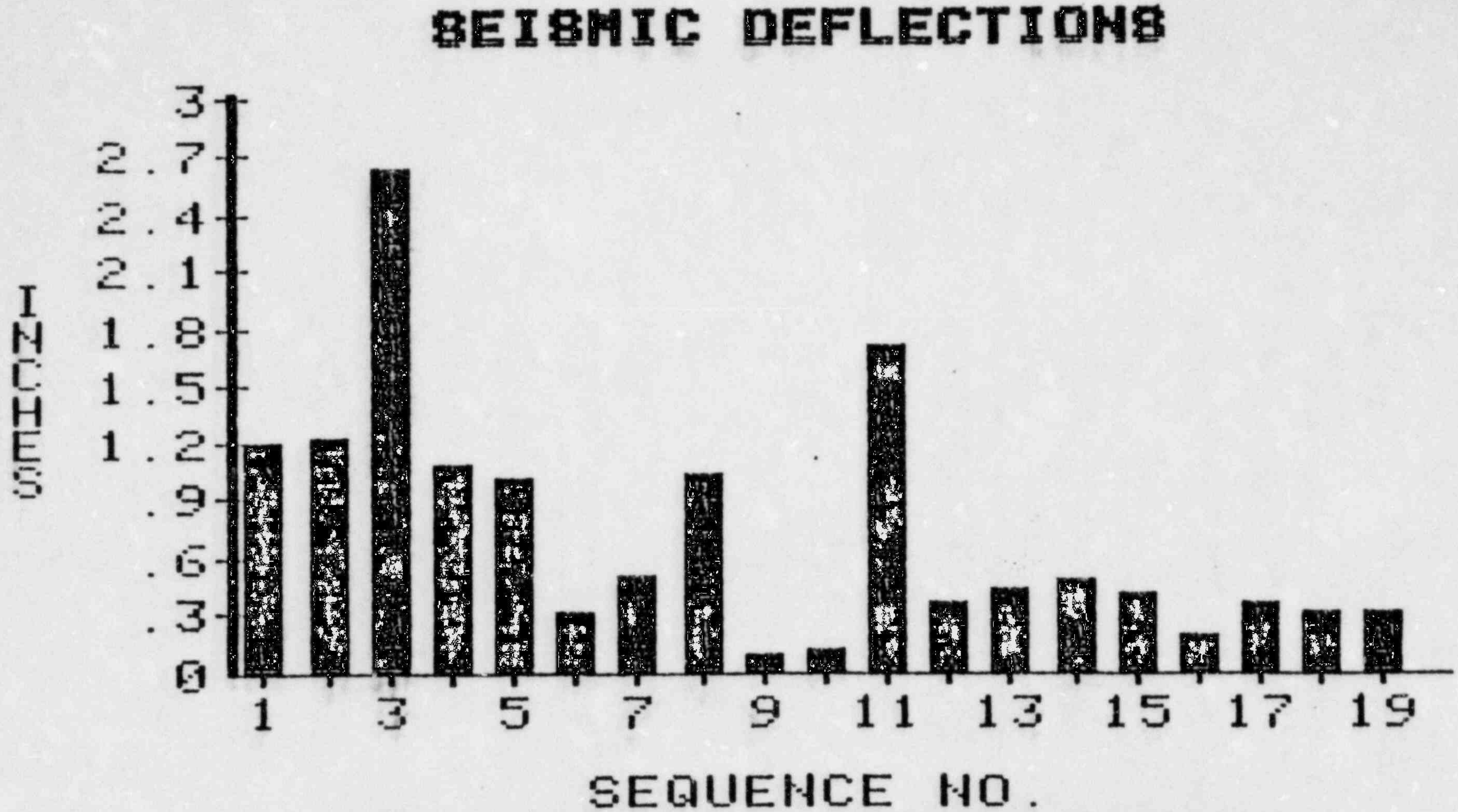
PRELIMINAR

EQ 14 NORMALIZED STRESSES



YCS ANALYSIS AFTER FIXING HOT SHUTDOWN INSIDE VC

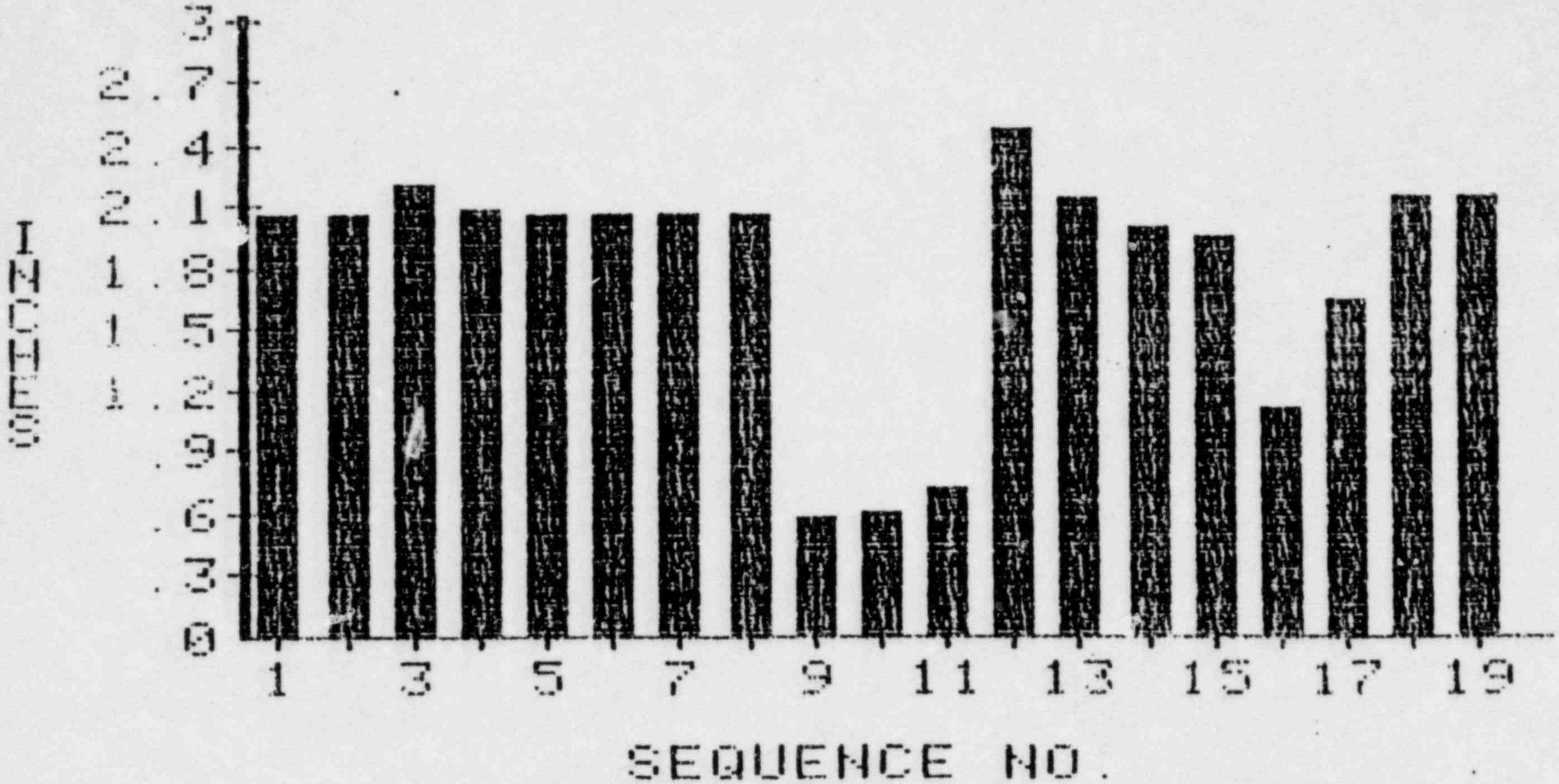
PRELIMINARY



YCS ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

PRELIMINARY

THERMAL DEFLECTIONS



YCS ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

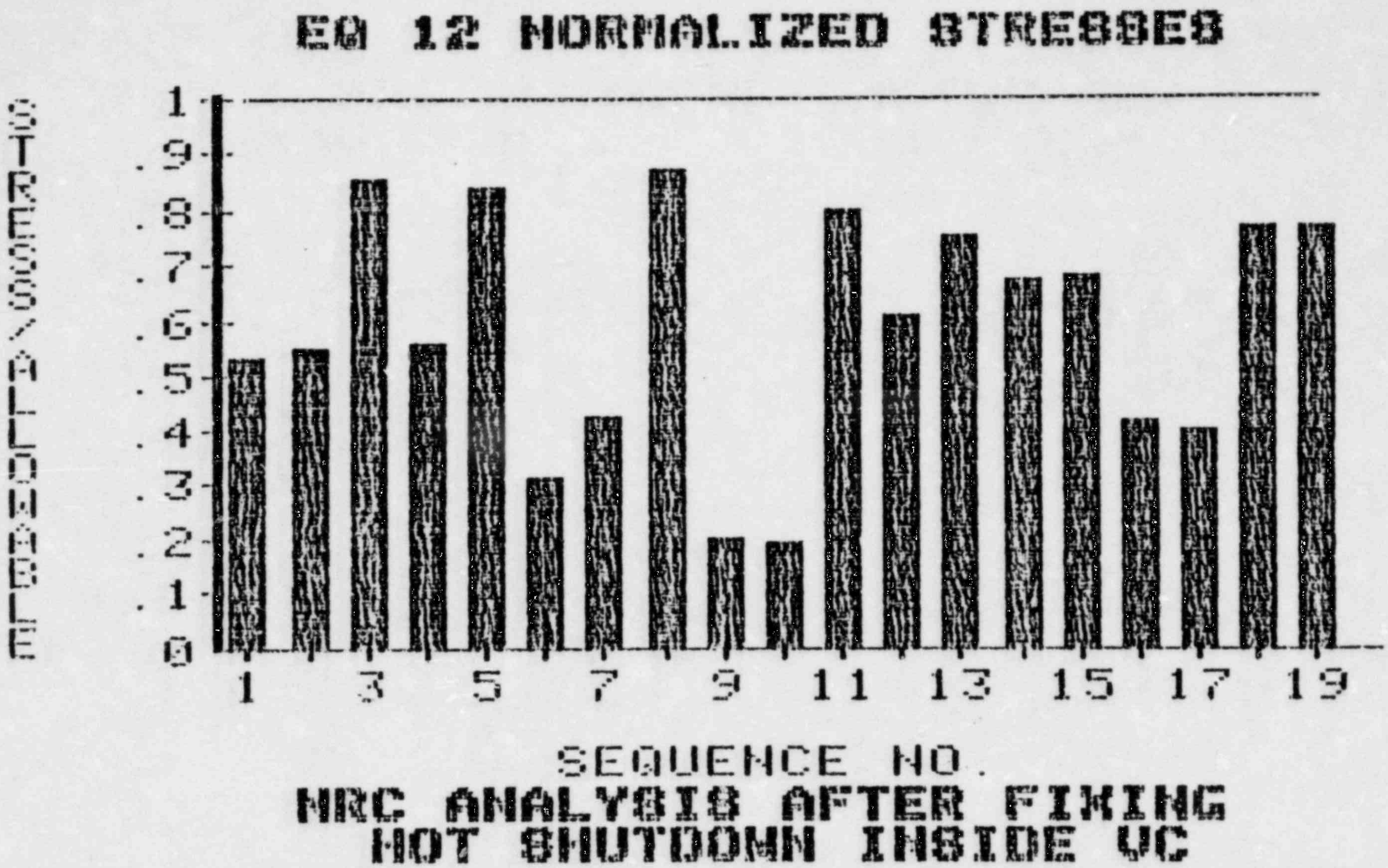
PRELIMINAR

SEISMIC ANALYSIS USING NRC
AFTER FIXES WERE IMPLEMENTED

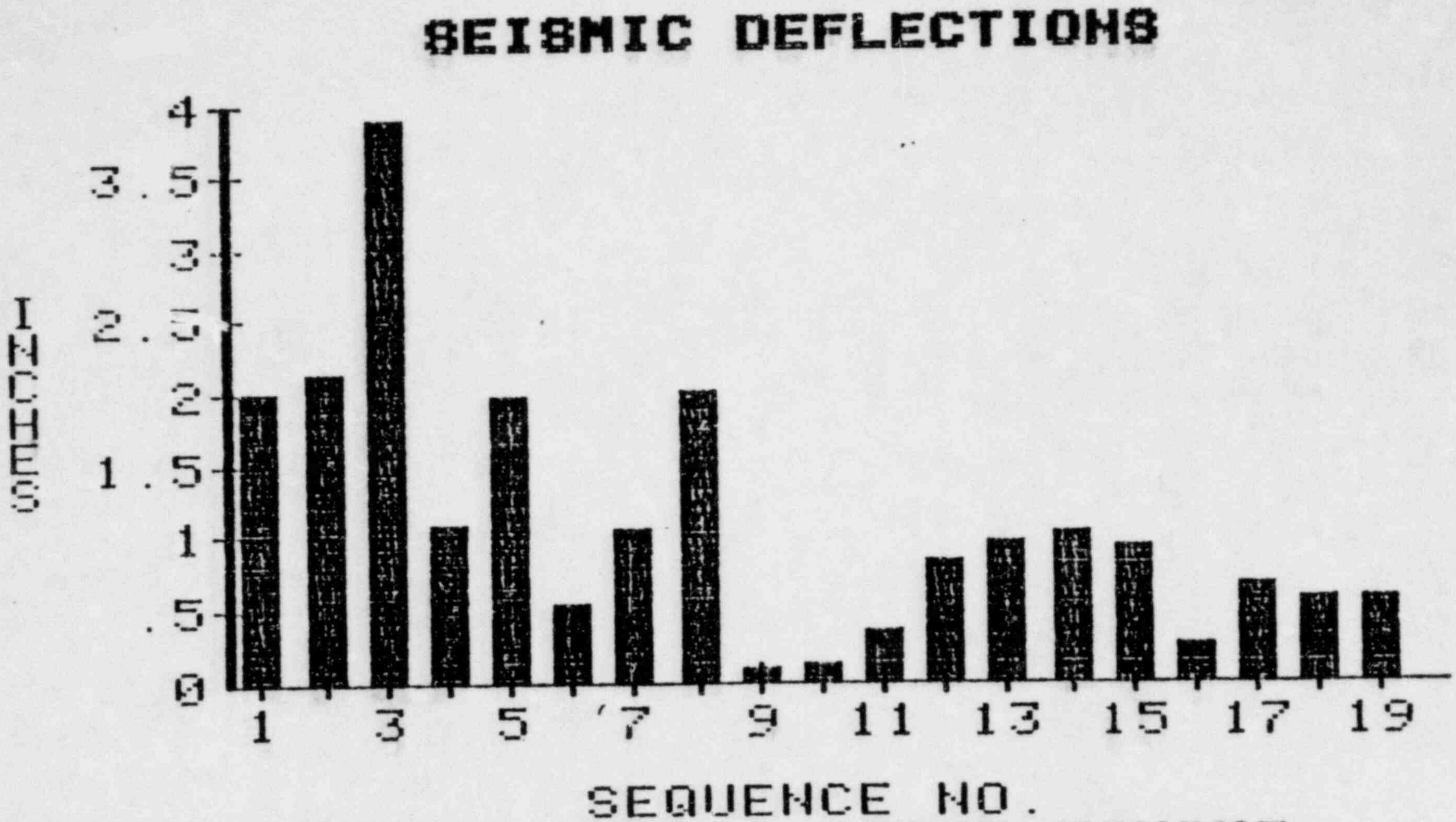
PROB. NO	SEQUENCE	SDMAX	NORM 12
1	1	2.00	0.53
2	2	2.12	0.54
3	3	3.90	0.85
4	4	1.06	0.55
21	5	1.96	0.84
22	6	0.54	0.31
23	7	1.03	0.42
24	8	1.99	0.87
41A	9	0.08	0.19
41B	10	0.10	0.19
41C	11	0.34	0.80
101	12	0.81	0.60
102	13	0.96	0.75
103	14	1.02	0.67
104	15	0.90	0.67
121	16	0.25	0.41
122	17	0.67	0.39
201	18	0.56	0.77
207	19	0.56	0.77

HOT SHUTDOWN INSIDE VC

PRELIMINAR


EQ 12 NORMALIZED STRESSES
STRESS/ALLOWABLE
1
.9
.8
.7
.6
.5
.4
.3
.2
.1
0
1
3
5
7
9
11
13
15
17
19
SEQUENCE NO.
NRC ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

PRELIMINAR



NRC ANALYSIS AFTER FIXING HOT SHUTDOWN INSIDE VC

PRELIMINAR

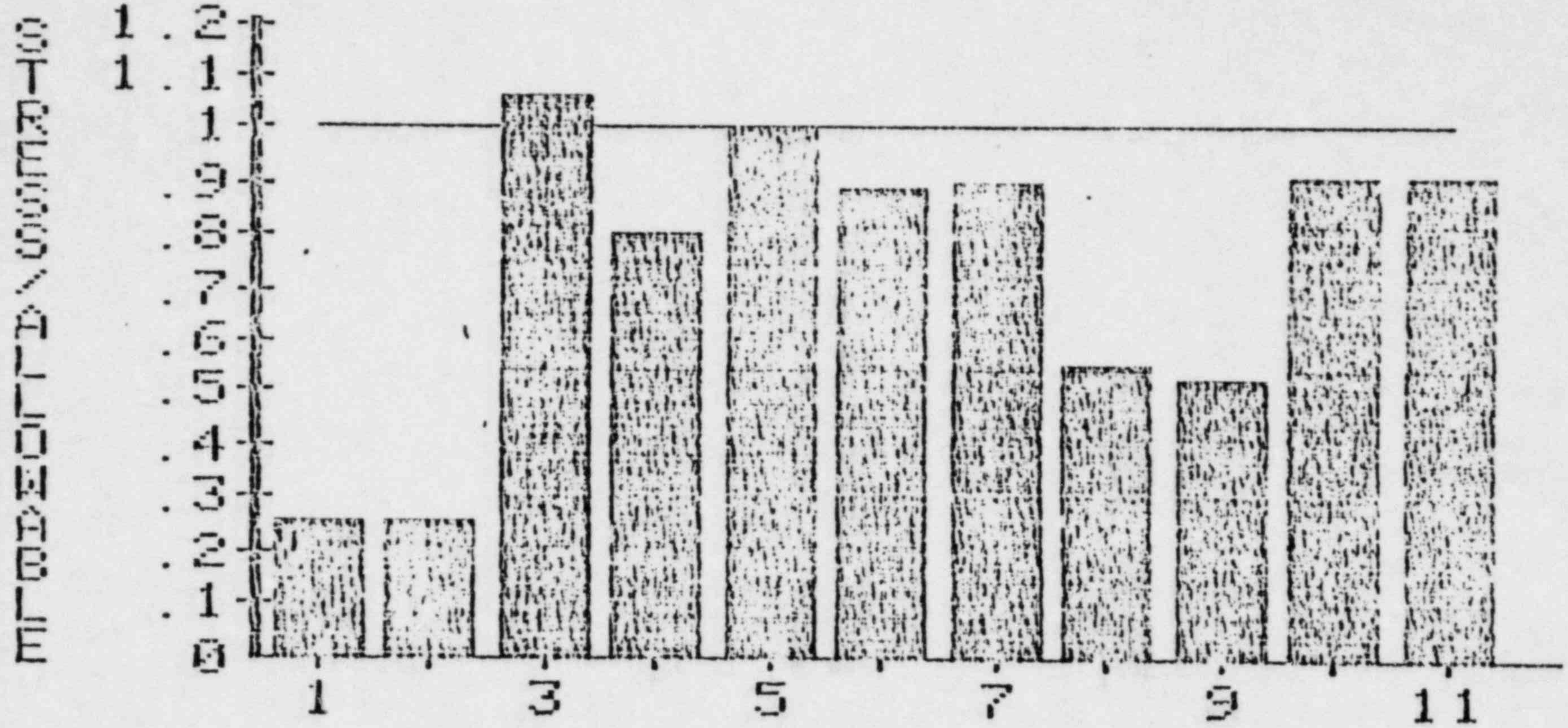
SEISMIC ANALYSIS
USING NRC
NORMALIZED TO 1.8 SH

PROB. NO	SEQUENCE	NORM 12
41A	1	0.26
41B	2	0.26
41C	3	1.06
101	4	0.81
102	5	1.00
103	6	0.89
104	7	0.90
121	8	0.55
122	9	0.53
201	10	0.90
207	11	0.90

CLASS 1 PORTIONS OF
HOT SHUTDOWN INSIDE VC

PRELIMINARY

EG 12 NORMALIZED STRESSES



NRC ANALYSIS AFTER FIXING
CLASS 1 PORTIONS OF HOT SHUTDOWN

PRELIMINARY

CLARIFICATION OF THE STRESS ANALYSIS RESULTS FOR REMAINING SYSTEMS TO YCS

The stress results summarized in the following tables and graphs are based on as-built piping geometries excluding the potential restraining effects of thru wall penetrations. Thus, many of the large reported seismic deflections and stresses (equation 12) will not occur because of the restraint of the wall on the pipe. For example, in problem 161, 162, 163 (see slide C4), three areas of high seismic stress (dark shaded areas) occur near the maximum lateral deflections (light shaded areas). A field review indicates that the pipe is passing through walls and will not deflect the 10 to 12 inches predicted by the conservative method, but only about 1/4 of these values. Since inertia stresses are proportional to acceleration and, therefore, deflection, accounting for the restraining effect will reduce the ratio of equation 12 to allowable from the reported 4.65 to the order of 1.5. Based on a preliminary inspection, Cygna is confident that a substantial number of spikes beyond 1.0 on the equation 12 slide can be reduced.

SEISMIC ANALYSIS USING YCS

PROB. NO.	SEQUENCE	SDMAX	NORM 11	NORM 12	NORM 13	NORM 14
61	1	3.786	0.55	0.39	0.02	0.21
30	2	4.21	0.17	0.28	0.03	0.07
62	3	9.221	0.43	1.66	0.56	0.39
63A	4	.9947	0.49	0.86	1.43	1.05
63B	5	.9947	0.49	0.86	1.43	1.05
63C	6	.9947	0.49	0.86	1.43	1.05
63D	7	.9947	0.49	0.86	1.43	1.05
64	8	3.12	0.62	0.64	3.69	2.48
81	9	.969	0.42	0.28	0.70	0.47
82	10	.969	0.42	0.28	0.70	0.47
83	11	.969	0.42	0.28	0.70	0.47
84	12	.969	0.42	0.28	0.70	0.47
123	13	6.53	0.33	1.21	1.44	0.96
124	14	6.44	0.26	0.99	1.47	0.96
125	15	.3096	0.21	0.37	0.67	0.46
11	16	5.188	0.16	2.76	1.22	0.80
29	17	.101	0.28	0.23	1.20	0.76
181	18	.64	0.17	0.19	0.34	0.23
262	19	.863	0.25	0.13	0.26	0.25
209	20	.1909	0.36	0.23	0.72	0.48
182	21	13.817	0.45	5.25	0.27	0.24
183	22	13.817	0.45	5.25	0.27	0.24
184	23	13.817	0.45	5.25	0.27	0.24
44	24	1.0588	0.21	0.46	0.77	0.50
161	25	12.7573	0.54	4.65	0.94	0.60
162	26	12.7573	0.54	4.65	0.94	0.60
163	27	12.7573	0.54	4.65	0.94	0.60
206	28	5.5827	0.94	9.01	3.49	2.47
205	29	.616	0.50	0.34	0.86	0.72

REMAINING SYSTEMS

PRELIMINARY

SEISMIC ANALYSIS USING YCS

PROB. NO.	SEQUENCE	SDMAX	NORM 11	NORM 12	NORM 13	NORM 14
43B	30	.0294	0.15	0.09	0.46	0.33
211	31	8.8689	0.71	2.30	0.55	0.61
204	32	.0905	0.10	0.09	0.39	0.26
42	33	10.85678	0.27	1.59	0.36	0.32
185	34	.2972	0.20	0.19	0.64	0.41
261	35	.469	0.21	0.33	1.15	0.78
43A	36	.018	0.13	0.07	0.53	0.38
202	37	12.3486	0.35	1.72	1.33	0.86
208	38	12.3486	0.35	1.72	1.33	0.86
210	39	.1073	0.04	0.07	1.30	0.79
203	40	9.894	0.22	1.12	0.11	0.12
143	41	16.378	0.83	3.50	0.67	0.59
141	42	.3438	0.29	0.20	0.86	0.64
142	43	2.212	0.38	0.51	0.12	0.22
186	44	4.8112	0.40	1.73	0.40	0.32
187	45	4.8112	0.40	1.73	0.40	0.32
188	46	4.8112	0.40	1.73	0.40	0.32
300	47	13.97	0.32	3.15	0.54	0.45
301	48	8.67	0.13	2.25	0.56	0.39
302	49	6.77	0.26	1.86	0.34	0.31
303	50	13.22	0.45	4.21	0.37	0.40
305	51	.82	0.35	0.40	0.24	0.22
306	52	1.91	0.47	0.56	0.21	0.26
307	53	2.07	0.29	0.46	0.36	0.27
308	54	9.5	0.26	1.33	0.17	0.18
314	55	.74	0.24	0.51	0.88	0.57
315	56	1.318	0.23	0.60	0.87	0.58
316	57	1.598	0.26	0.44	1.03	0.71
317	58	1.598	0.26	0.44	1.03	0.71

REMAINING SYSTEMS

PRELIMINARY

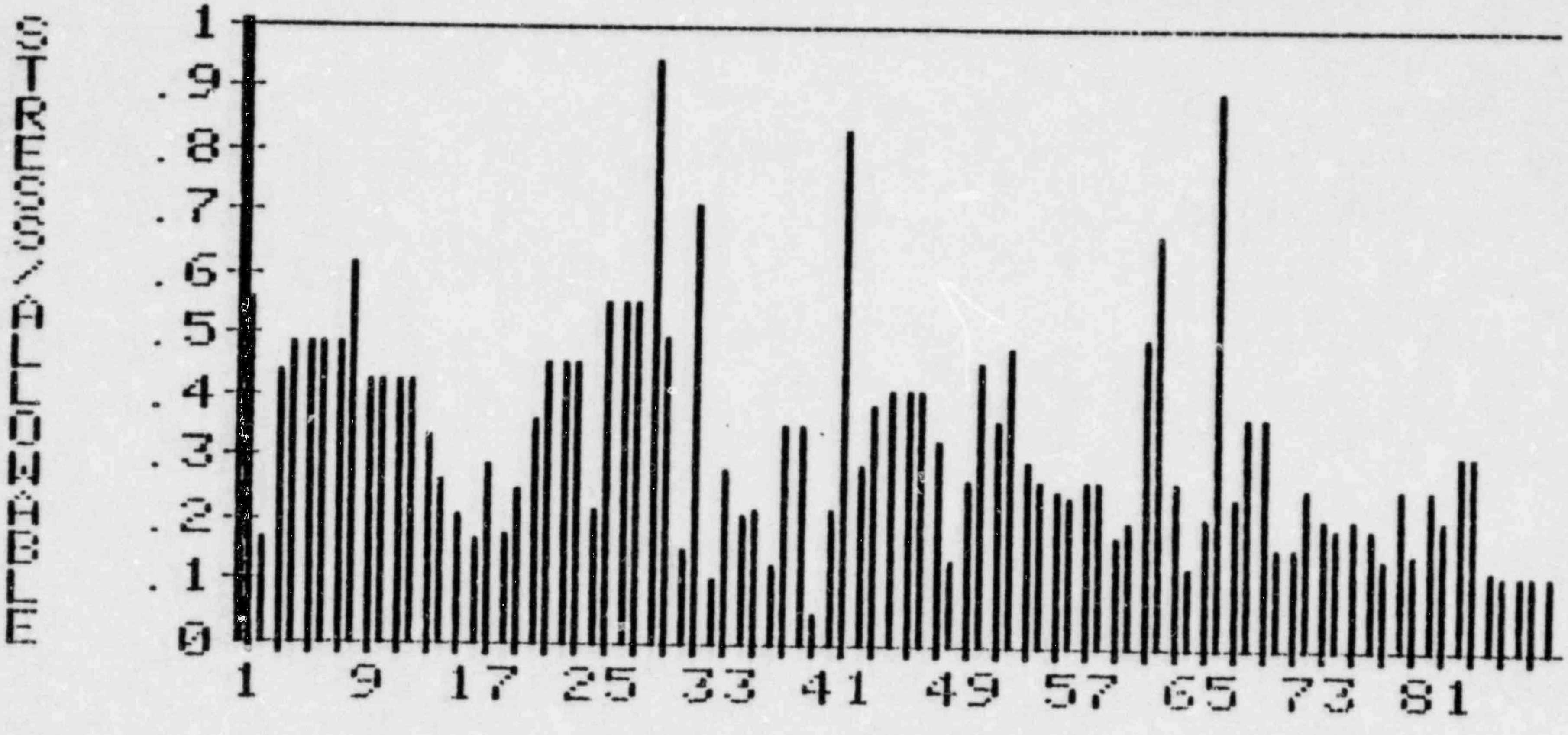
SEISMIC ANALYSIS USING YCS

PROB. NO.	SEQUENCE	SDMAX	NORM 11	NORM 12	NORM 13	NORM 14
318	59	1.13	0.17	0.34	0.34	0.22
319	60	4.29	0.19	0.99	0.45	0.30
320	61	.83	0.49	0.50	0.78	0.67
321	62	.94	0.66	0.94	7.93	4.85
322	63	.68	0.26	0.67	0.52	0.35
350	64	1.19	0.12	0.39	0.51	0.35
351	65	.91	0.21	0.39	3.31	2.05
352	66	2.68	0.89	1.63	4.71	2.87
400	67	.99	0.24	0.30	1.35	0.85
403	68	5.73	0.36	2.89	0.41	0.28
404	69	5.73	0.36	2.89	0.41	0.28
407	70	.48	0.16	0.25	1.54	0.95
408	71	.48	0.16	0.25	1.54	0.95
409	72	7.33	0.25	1.71	0.35	0.24
410	73	6.32	0.21	1.21	0.33	0.22
411	74	5.86	0.19	1.71	0.28	0.20
412	75	6.32	0.21	1.21	0.33	0.22
413	76	5.86	0.19	1.71	0.28	0.20
414	77	6.6	0.14	1.60	0.33	0.22
415	78	7.33	0.25	1.42	0.35	0.24
416	79	8.45	0.15	2.25	0.78	0.49
417	80	7.33	0.25	1.42	0.35	0.24
418	81	8.45	0.21	2.25	0.78	0.49
500	82	.1	0.31	0.16	1.22	0.78
501	83	.1	0.31	0.16	1.22	0.78
502	84	.048	0.12	0.06	0.00	0.05
401	85	.02	0.12	0.08	1.47	0.93
402	86	.02	0.12	0.08	1.47	0.93
405	87	.085	0.12	0.14	0.86	0.55
406	88	.075	0.12	0.13	0.78	0.54

REMAINING SYSTEMS

PRELIMINARY

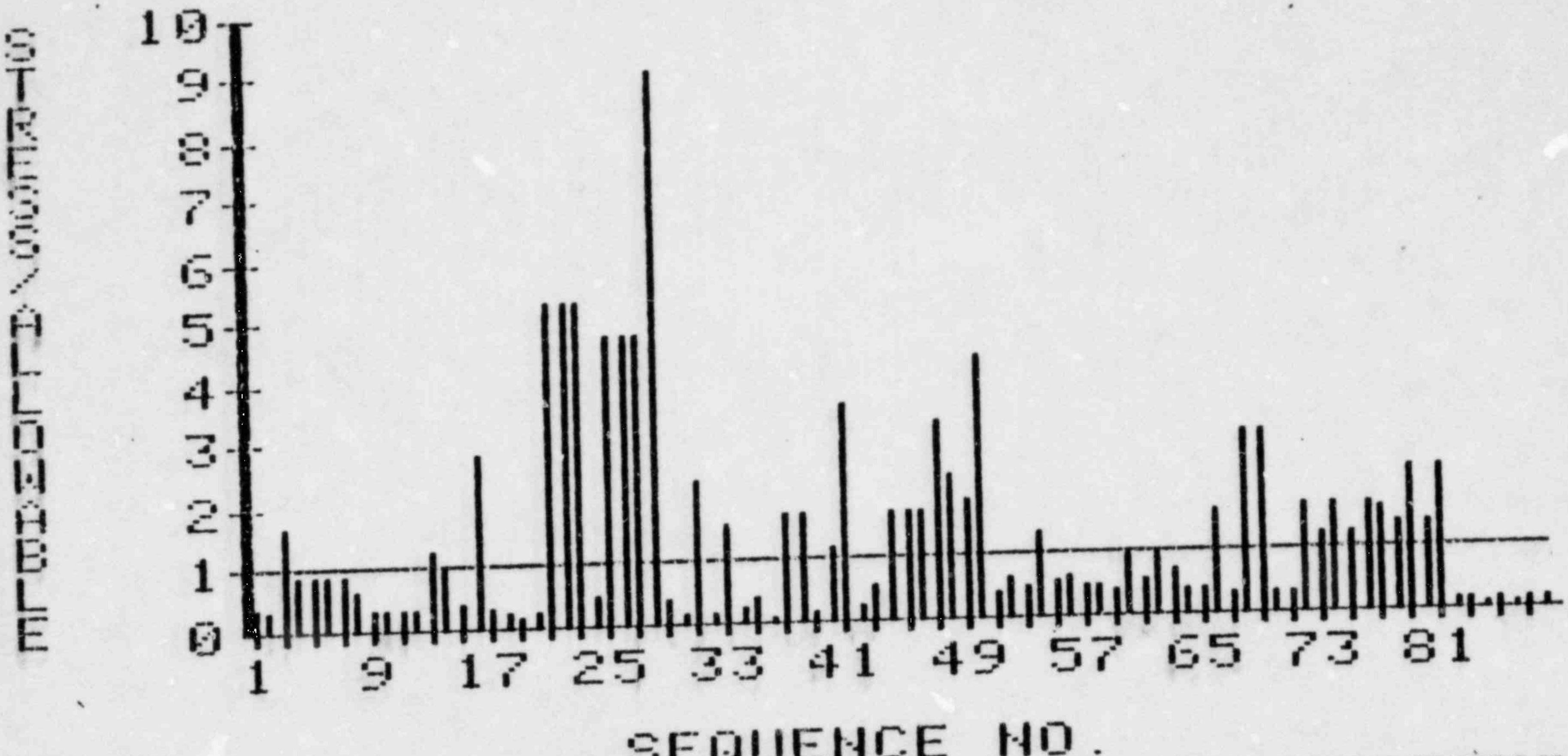
EQ 11 NORMALIZED STRESSES



YCS ANALYSIS REMAINING SYSTEMS

PRELIMINARY

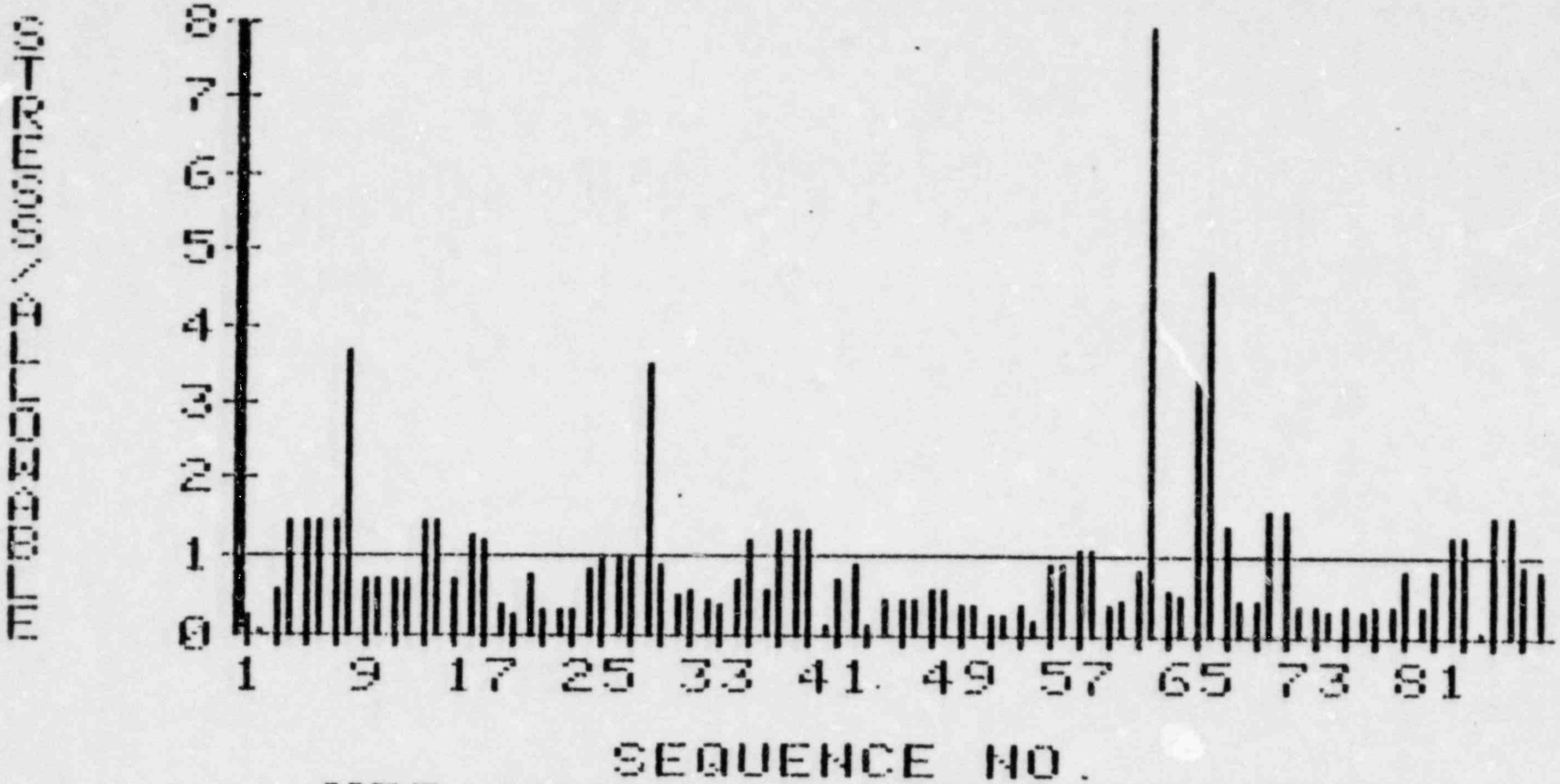
EA 12 NORMALIZED STRESSES



YCS ANALYSIS REMAINING SYSTEMS

PRELIMINARY

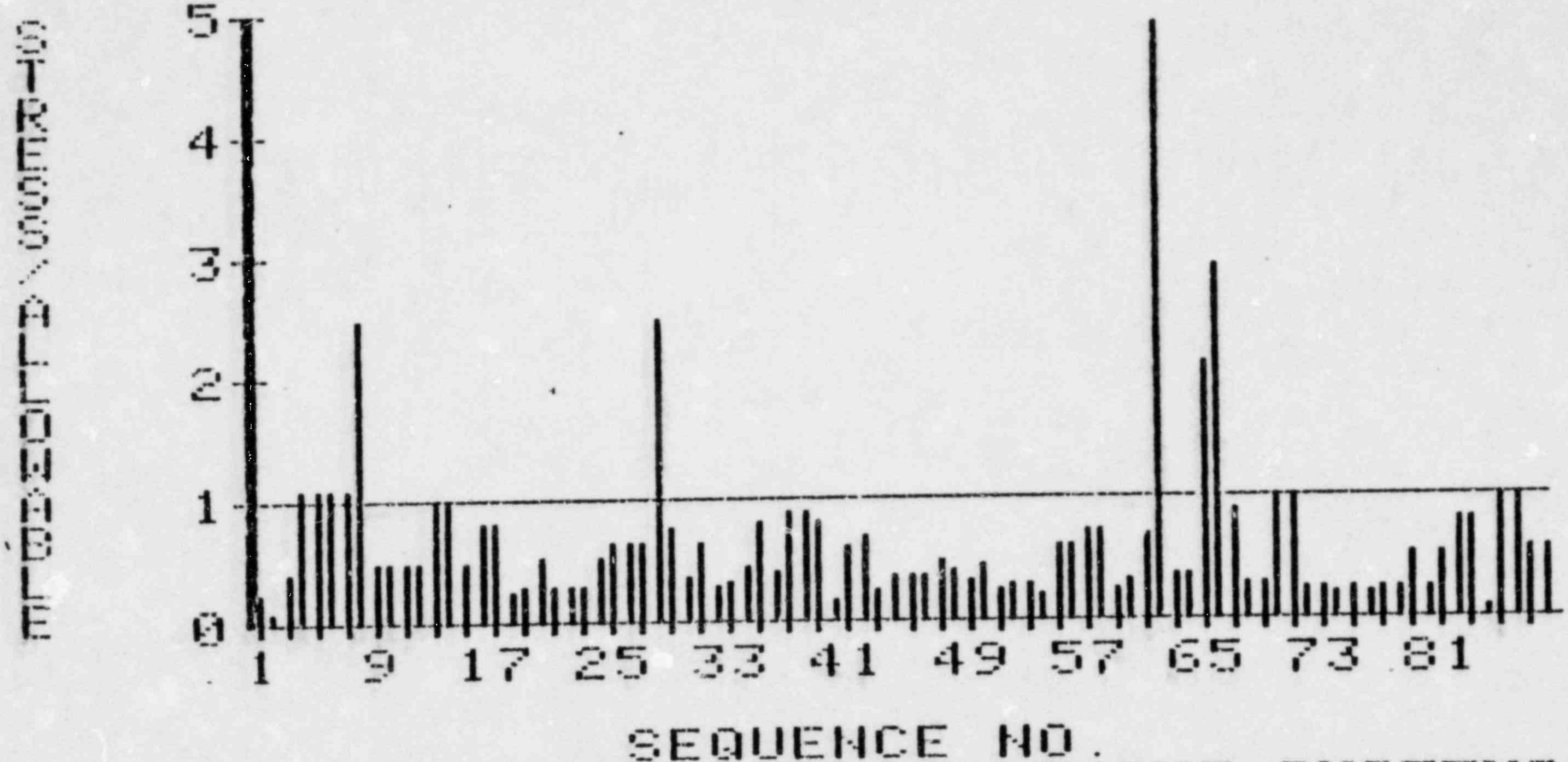
EQ 13 NORMALIZED STRESSES



YCS ANALYSIS REMAINING SYSTEMS

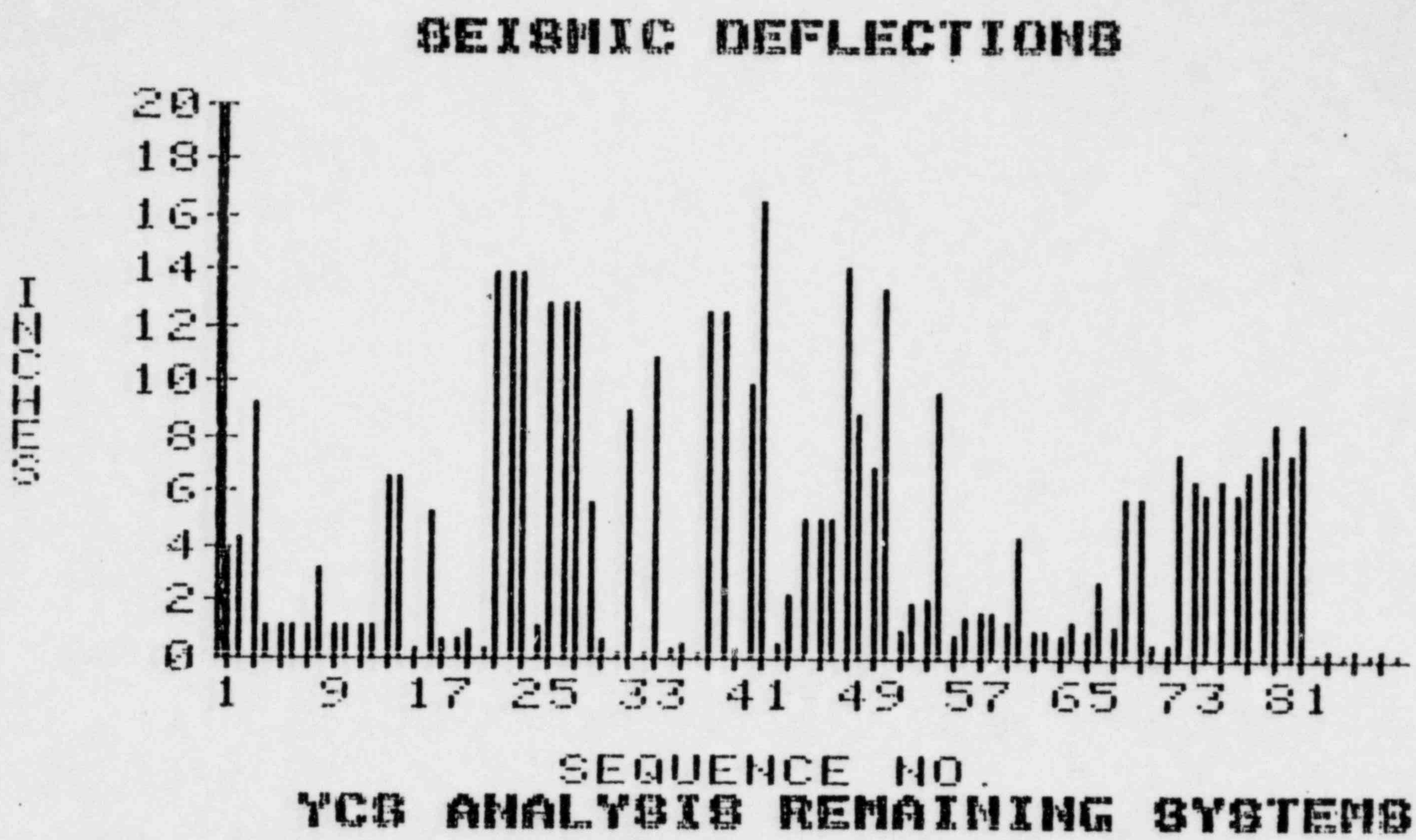
PRELIMINARY

EQ 14 NORMALIZED STRESSES



YCS ANALYSIS REMAINING SYSTEMS

PRELIMINARY



PRELIMINARY

CONSERVATIVE ASSUMPTIONS IN ANALYSIS

1) NEGLECTED RESTRAINING EFFECT OF THE PENETRATIONS ON THE PIPING

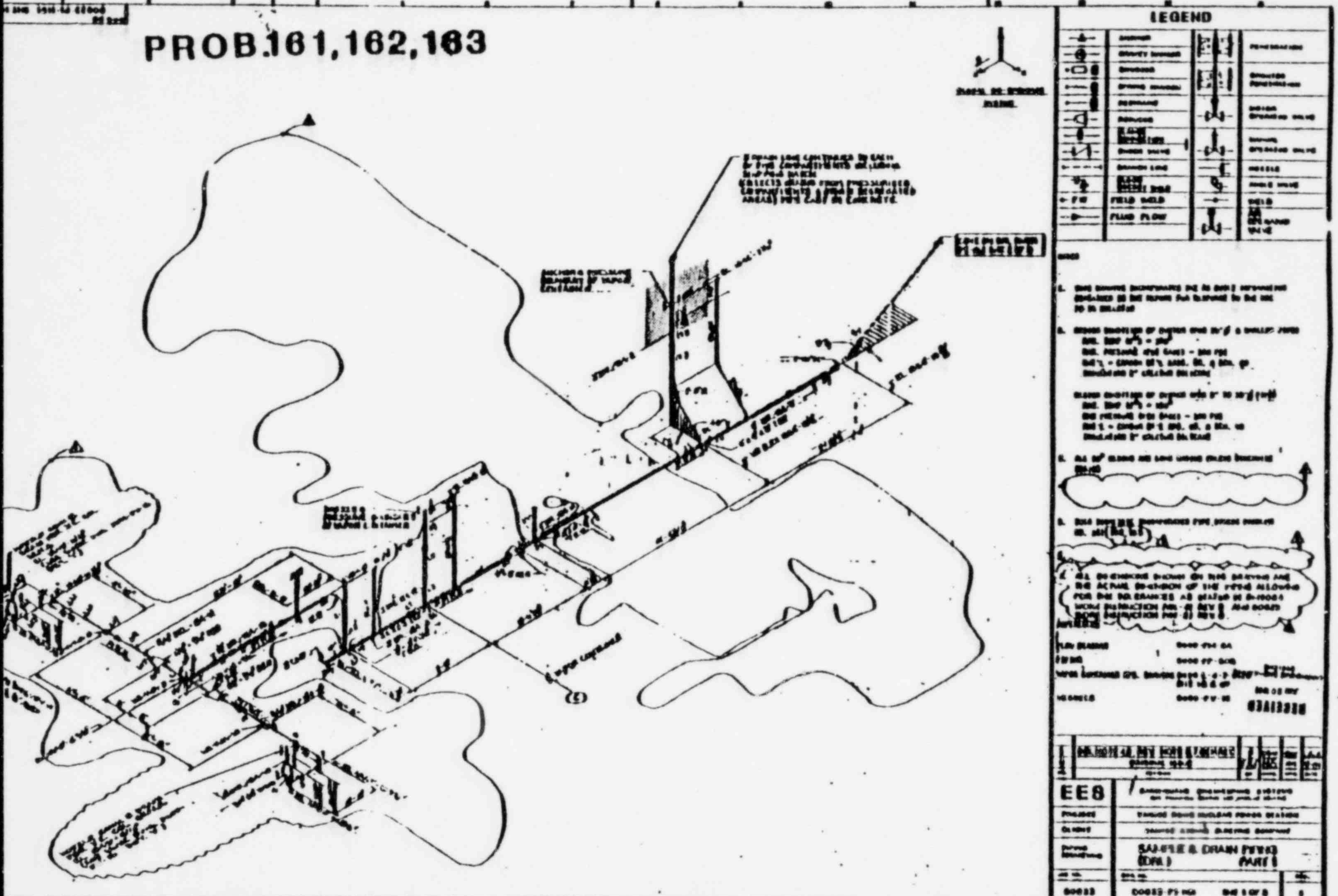
2) SPECTRA FROM VARIOUS STRUCTURES ENVELOPED

3) PIPING MATERIALS ARE DUCTILE, NOT BRITTLE, AND YIELD RATHER THAN FRACTURE

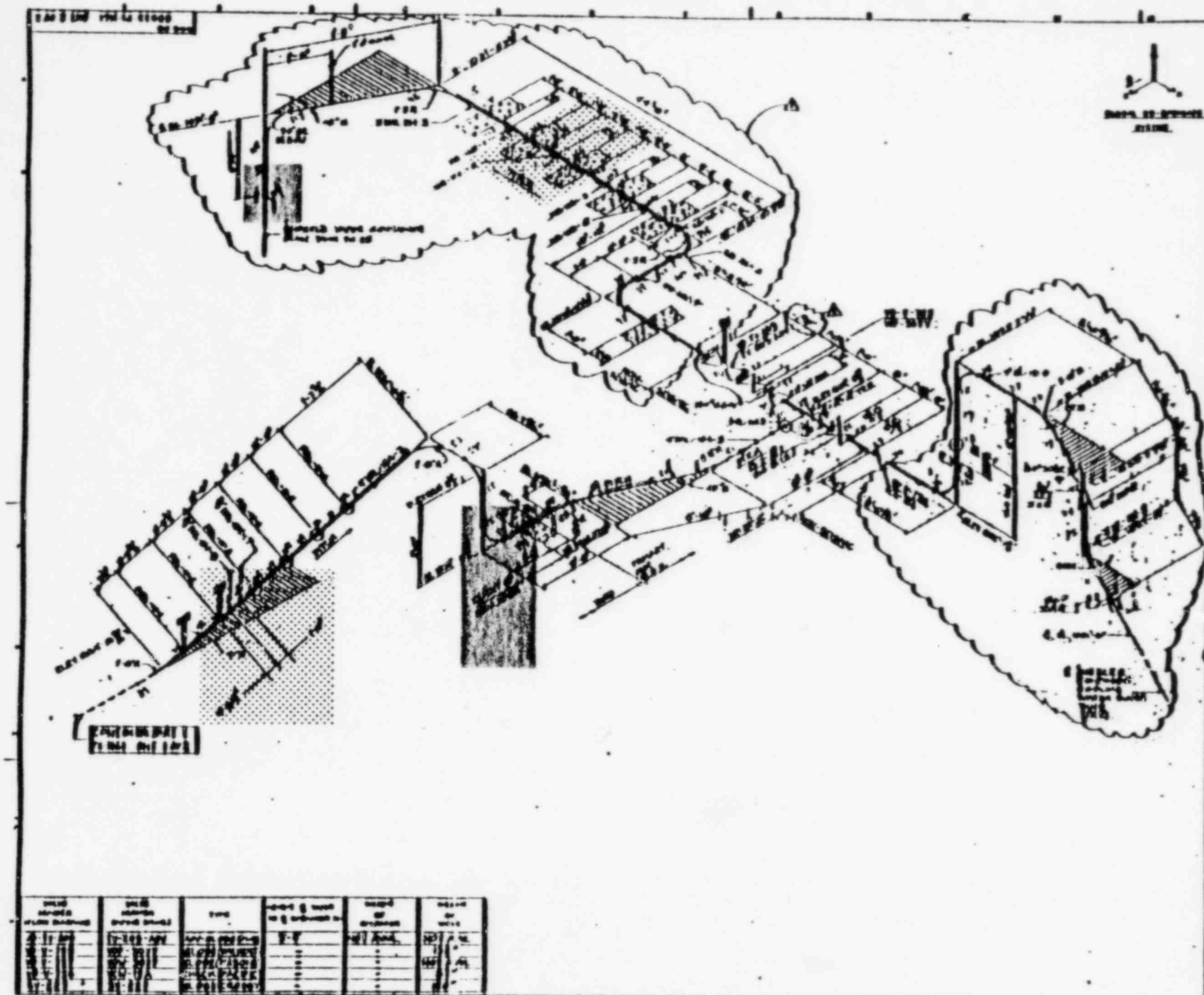
SAMPLE OF HIGH DEFLECTIONS/STRESSES
WHERE RESTRAINING EFFECT OF PENETRATION IS NEGLECTED

PROBLEM	SEQUENCE	EQ 12	ΔX	NODE	ΔZ	NODE
161,162,163	25,26,27	4.65	12.8	44	10.0	106

PROB.161,162,163



B-102



LEGEND

RECEIVED

EES

SAMPLE & DRAIN PIPING (DRL) PART 1

CLARIFICATION OF STRESS RESULTS FOR ECCS/AUX FW TO IDBS

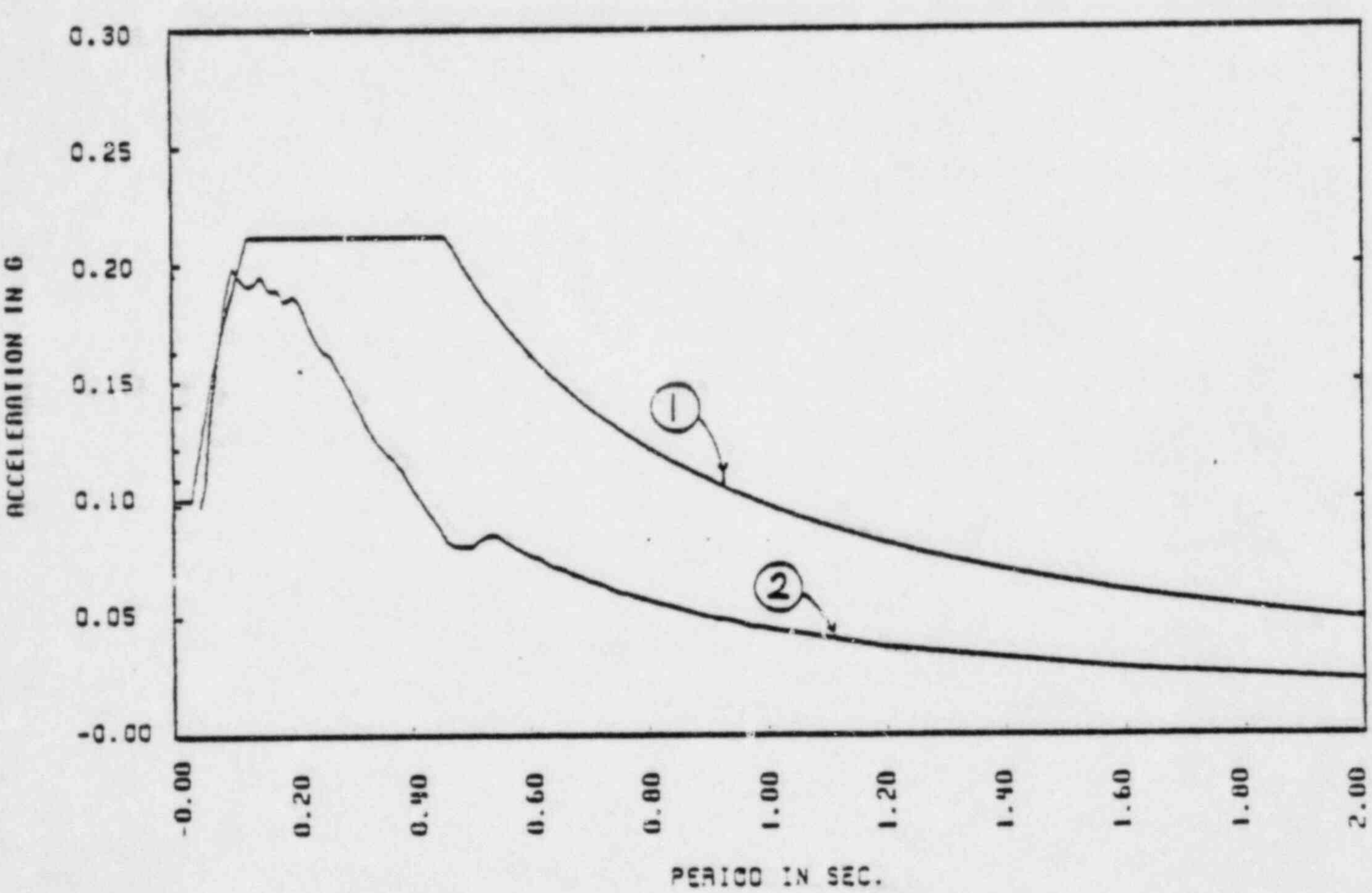
This section indicates that a large portion of the ECCS/Aux FW pipe stress results are well within acceptable limits. It should be further noted that:

- Only four out of twenty-six problems exceed the "normalized to one" limit.
- The restraining effect of the penetrations, as elaborated on earlier, is not considered.

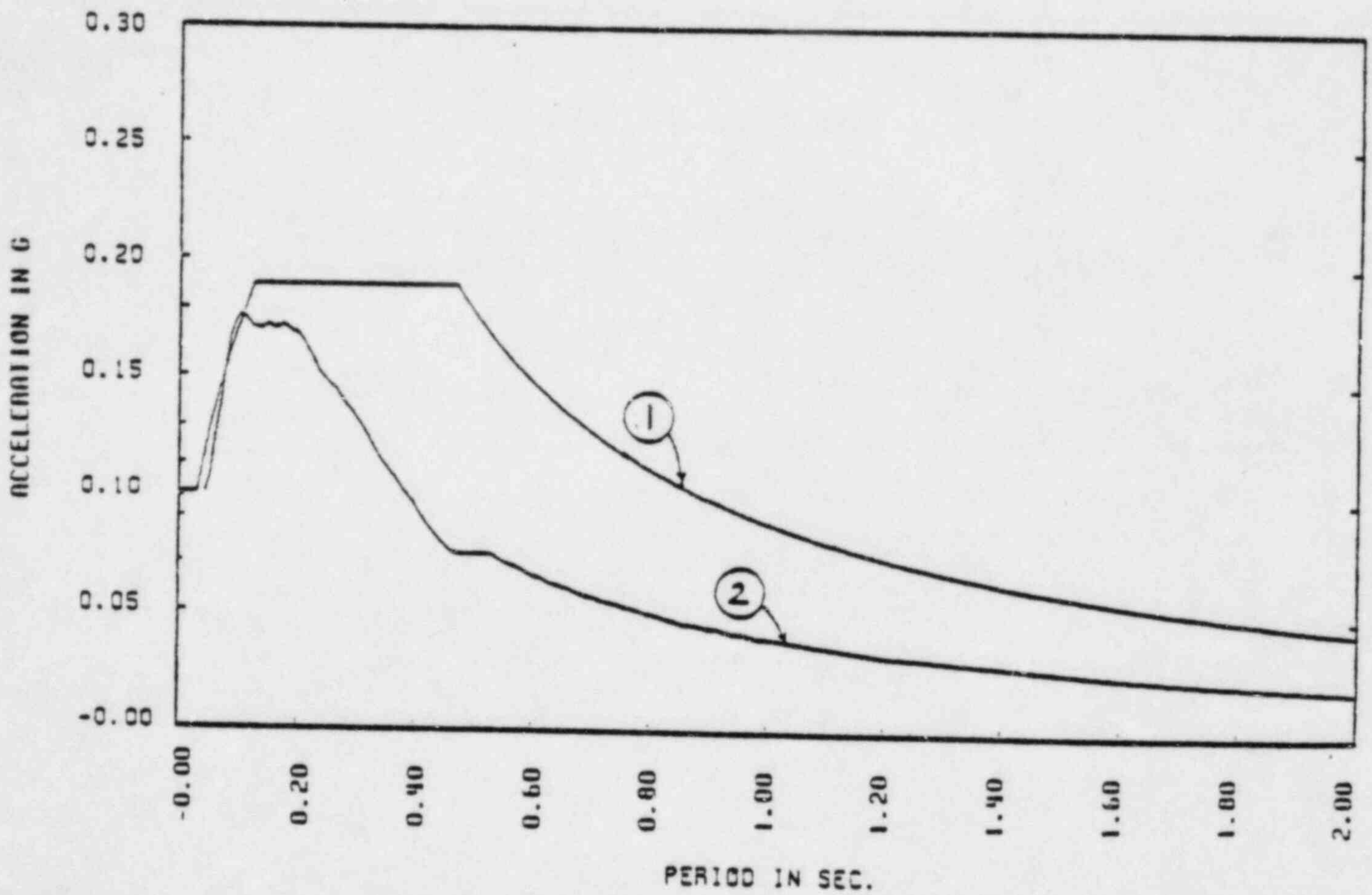
PIPE STRESS

- SUBSECTION OF REMAINING SYSTEMS EVALUATED TO IDBS
- COMPARISON OF IDBS VS. YCS GROUNDMOTIONS
- RESULTS OF EVALUATION

1) YANKEE ROWE FINAL DESIGN SPECTRA PGA=0.1G DAMPING = 0.05
2) YANKEE INTERIM DESIGN BASIS SPECTRA (IDBS) DAMPING = 0.05



1) YANKEE ROWE FINAL DESIGN SPECTRA PGA=0.1G DAMPING = 0.07
2) YANKEE INTERIM DESIGN BASIS SPECTRA (IDBS) DAMPING = 0.07



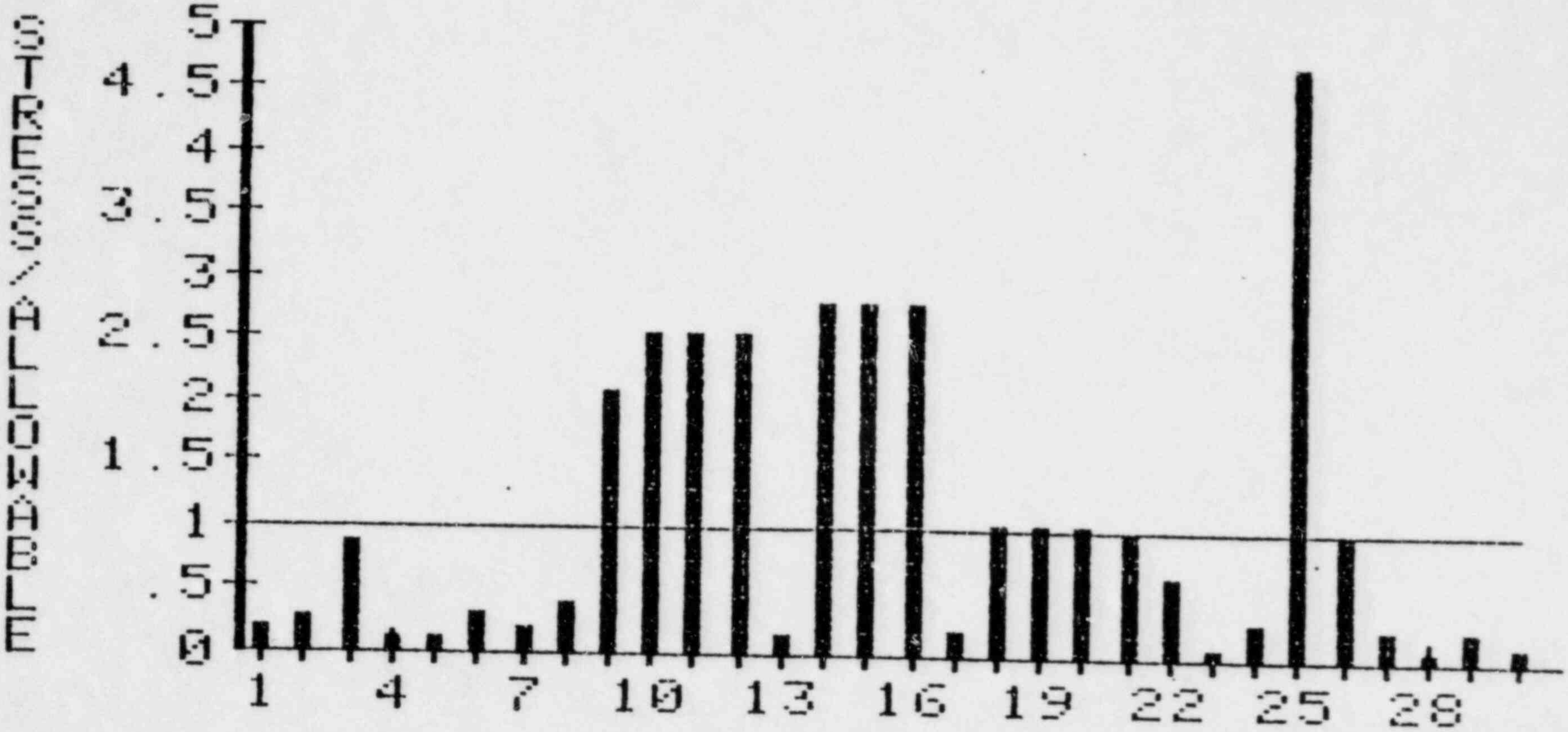
SEISMIC EVALUATION USING IDBS

PROB. NO.	SEQUENCE	NORM 12
29	1	0.17
30	2	0.27
42	3	0.87
43A	4	0.08
43B	5	0.09
44	6	0.32
141	7	0.18
142	8	0.36
143	9	2.07
161	10	2.52
162	11	2.52
163	12	2.52
181	13	0.14
182	14	2.77
183	15	2.77
184	16	2.77
185	17	0.17
186	18	1.02
187	19	1.02
188	20	1.02
202	21	0.99
203	22	0.63
204	23	0.07
205	24	0.28
206	25	4.69
208	26	0.99
209	27	0.20
210	28	0.05
261	29	0.21
262	30	0.11

ECCS/AUX FEED SYSTEMS

PRELIMINARY

EQ 12 NORMALIZED STRESS



IDBS EVALUATION
ECCS/AUX FEED SYSTEMS

PRELIMINARY

APPENDIX C

REEVALUATION GUIDELINES 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 1
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.

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), NC-3653.1], the following stresses are not to be exceeded for the specified piping: 1

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 the code which renumbered the equations differently from past and present 1
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: $\|W\|$ or $\|W + T\|$	Code Subsection NF Design, Level A, and Level B Limits	$P_m \leq 1.0\ S_m$ $P_\ell + P_b \leq 1.5\ S_m$ 1
The higher of: $\|W\| + \|SSE\| + \|AM\|$ or $\|W + T\| + \|SSE\| + \|AM\|$	Code Subsection NF Level D Limits	$P_m \leq 1.5$ Sm or 1.2 Sy[c] not to exceed 0.7 Su 1 $P_\ell + P_b$ 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	Acceptance Criteria[a] Plate and Shell
The higher of: $\lvert W \rvert$ or $\lvert W + T \rvert$	Code Subsection NF Design, Level A, and Level B Limits	$\sigma_\ell \leq 1.0\ S$ $\sigma_\ell + \sigma_b \leq 1.5\ S$
The higher of: $\lvert W \rvert + \lvert SSE \rvert + \lvert AM \rvert$ or $\lvert W + T \rvert + \lvert SSE \rvert + \lvert AM \rvert$	- - - - - - - - - - Code Subsection NF Level D Limits	$\sigma_\ell \leq 1.5\ S$ or $0.4\ S_u$ (b) $\sigma_\ell + \sigma_b \leq 2.25\ S$ or $0.6\ S_u$ (b)

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

C-7

ACCEPTANCE CRITERIA FOR CLASS 1 MECHANICAL EQUIPMENT

Component	Loading Combination[b]	Criteria[d] [g]
Pressure vessels and heat-exchangers	$\|W + P_D\| + \|SSE\| + \|\text{Nozzle Loads}\|$	$P_m \leq 2.4\ S_m$ 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]	$\|W + P_D\| + \|SSE\| + \|\text{Nozzle Loads}\|$	$P_m \leq 1.2\ S_m$ 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	$\|W + P_D\| + \|SSE\| + \|\text{Nozzle Loads}\|$	$P_m \leq 2.4\ S_m$ 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]	$\|W + P_D\| + \|SSE\| + \|\text{Nozzle Loads}\|$	$P_m \leq 1.2\ S_m$ 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]	$\|W + P_D\| + \|SSE\| + \|\text{Nozzle Loads}\|$	$P_m \leq 2.4\ S_m$ 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.

i-10'

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	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_\ell) + \sigma_b \leq 1.8\ S$
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 D

YAEC COMPONENT SUPPORT CRITERIA

CYGNA ENERGY SERVICES

Rev. 0

NRC
EQUIPMENT QUALIFICATION
YANKEE ROWE POWER STATION

August 3 & 4, 1982

PRELIMINARY

D-1

D-2

EQUIPMENT QUALIFICATION

(1) INTRODUCTION

NOZZLE REACTIONS

(2) SCOPE

(3) GUIDELINES

(4) TABLES AND GRAPHS OF YCS AND NRC VALVE ACCELERATIONS

VALVE ACCELERATIONS

(5) SCOPE

(6) TABLES AND GRAPHS HOT SHUTDOWN INSIDE VC (YCS AND NRC)

(7) TABLES AND GRAPHS ECCS/AUX FEED (YCS)

EQUIPMENT ANCHORAGE EVALUATION

(8) SCOPE AND OBJECTIVES

(9) DRAWINGS OF SVC SHOWING EQUIPMENT LOCATIONS

(10) DRAWINGS OF RPV AND SUPPORT RING

(11) REACTOR RING SUPPORT FLOW CHART

(12) REACTOR RING SUPPORT FINITE ELEMENT MODEL

(13) MASS AND STIFFNESS OF RCS PIPING

(14) FREQUENCIES OF MODEL

(15) RING SUPPORT EVALUATION

(16) DRAWING OF STEAM GENERATOR AND SUPPORTS

(17) STEAM GENERATOR ANCHORAGE FLOWCHART

(18) STEAM GENERATOR MODEL AND DETAILS

(19) FORCES ON STEAM GENERATOR SUPPORTS

(20) DRAWINGS OF PRESSURIZER AND SUPPORTS

(21) PRESSURIZER ANCHORAGE FLOWCHART

(22) PRESSURIZER MODEL

(23) PRESSURIZER ANCHORAGE FORCES

(24) PRESSURIZER ANCHORAGE EVALUATION

(25) DRAWINGS OF FIRE TANK AND ANCHORAGE

(26) PHOTOGRAPHS OF FIRE TANK AREA

(27) DESIGN REQUIREMENTS FOR FIRE TANK AND COMPARISON

(28) CONCLUSIONS FROM EQUIPMENT ANCHORAGE EVALUATION

EQUIPMENT QUALIFICATION

INTRODUCTION

- NOZZLE REACTIONS
- VALVE ACCELERATIONS
- EQUIPMENT ANCHORAGES

<u>EQUIPMENT QUALIFICATION</u>
<u>NOZZLE REACTIONS</u>

SCOPE

- Reactor Pressure Vessel (RPV)
- Steam Generator
- Main Coolant Pump
- Pressurizer

D-4

EQUIPMENT QUALIFICATION
NOZZLE REACTIONS

needs axial load due to pressure

GUIDELINES

$$\frac{F_R}{F} + \frac{M_R}{M} < 1$$

WHERE

F_R = COMPUTED RESULTANT FORCE ON NOZZLES (LBS)

M_R = COMPUTED RESULTANT MOMENTS ON NOZZLES (IN-LBS)

F = aAs/10 WHERE a = 1.8 EMERGENCY CONDITION (YCS)
2.4 FAULTED (NRC)

A = PIPE METAL AREA (IN^2)

S = 8000, FOR PIPES WITH DIAMETER < 4"
6000, FOR PIPES 6"- 8" IN DIAMETER
4000, FOR PIPES WITH DIAMETER > 10"

M = aZS WHERE Z = PIPE SECTION MODULUS (IN^3)

over: 15/35 = .43   over: 18/35 = .51

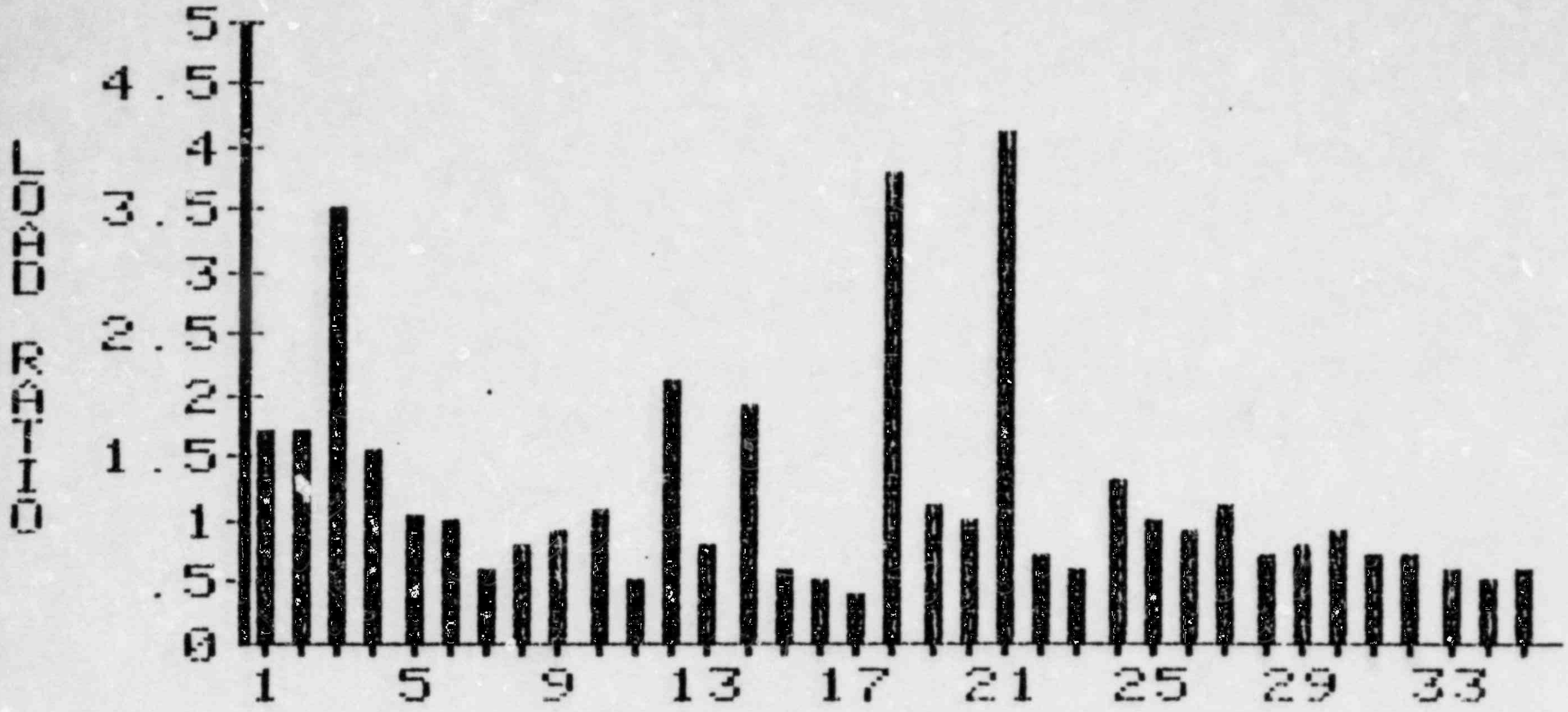
NOZZLE LOAD RATIOS ON MAJOR EQUIPMENT

PROB	SEQUENCE	EQUIP	SIZE	YCS	NRC
1	1	SG	14	1.71	1.92
2	2	SG	14	1.71	1.95
3	3	SG	14	3.49	2.62
4	4	SG	14	1.56	1.67
21	5	SG	8	1.03	1.05
22	6	SG	8	0.98	0.91
23	7	SG	8	0.59	0.55
24	8	SG	8	0.79	0.75
41A	9	PR	2	0.91	0.93
41B	10	PR	2	1.05	1.36
41C	11	PR	2	0.50	0.52
101	12	RPV	20	2.10	1.60
	13	RPV	20	0.80	0.90
	14	SG	20	1.90	1.50
	15	SG	24	0.60	0.80
	16	RCP	24	0.50	0.60
	17	RCP	20	0.40	0.50
102	18	RPV	20	3.80	5.40
	19	RPV	20	1.10	1.30
	20	SG	20	1.00	1.20
	21	SG	24	4.10	6.00
	22	RCP	24	0.70	0.90
	23	RCP	20	0.60	0.70
103	24	RPV	20	1.30	1.50
	25	RPV	20	1.00	1.10
	26	SG	20	0.90	1.00
	27	SG	24	1.10	1.30
	28	RCP	24	0.70	0.80
	29	RCP	20	0.80	0.90
104	30	RPV	20	0.90	1.00
	31	RPV	20	0.70	1.10
	32	SG	20	0.70	0.90
	33	SG	24	0.60	0.80
	34	RCP	24	0.50	0.50
	35	RCP	20	0.60	0.60

PRELIMINARY

HOT SHUTDOWN INSIDE VC

EQUIPMENT NOZZLE LOADS

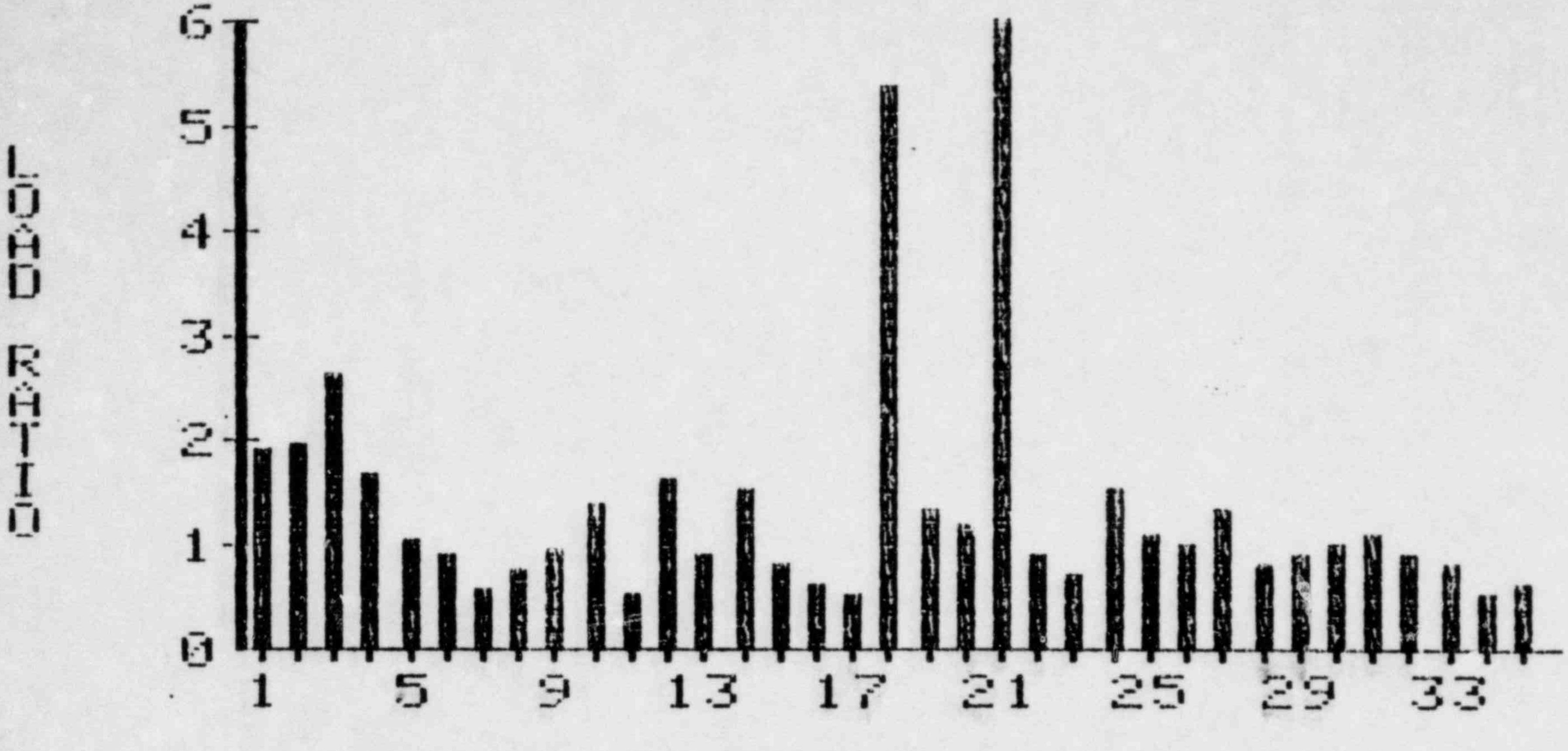


YCS ANALYSIS AFTER FIXING HOT SHUTDOWN INSIDE VC

PRELIMINARY

D-7

EQUIPMENT NOZZLE LOADS



NRC ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

PRELIMINARY

D-

EQUIPMENT QUALIFICATION

VALVE ACCELERATIONS

SCOPE

- HOT SHUTDOWN INSIDE VC (FIXED)
- ECCS/AUX FEEDWATER DUE TO YCS

PRELIMINARY

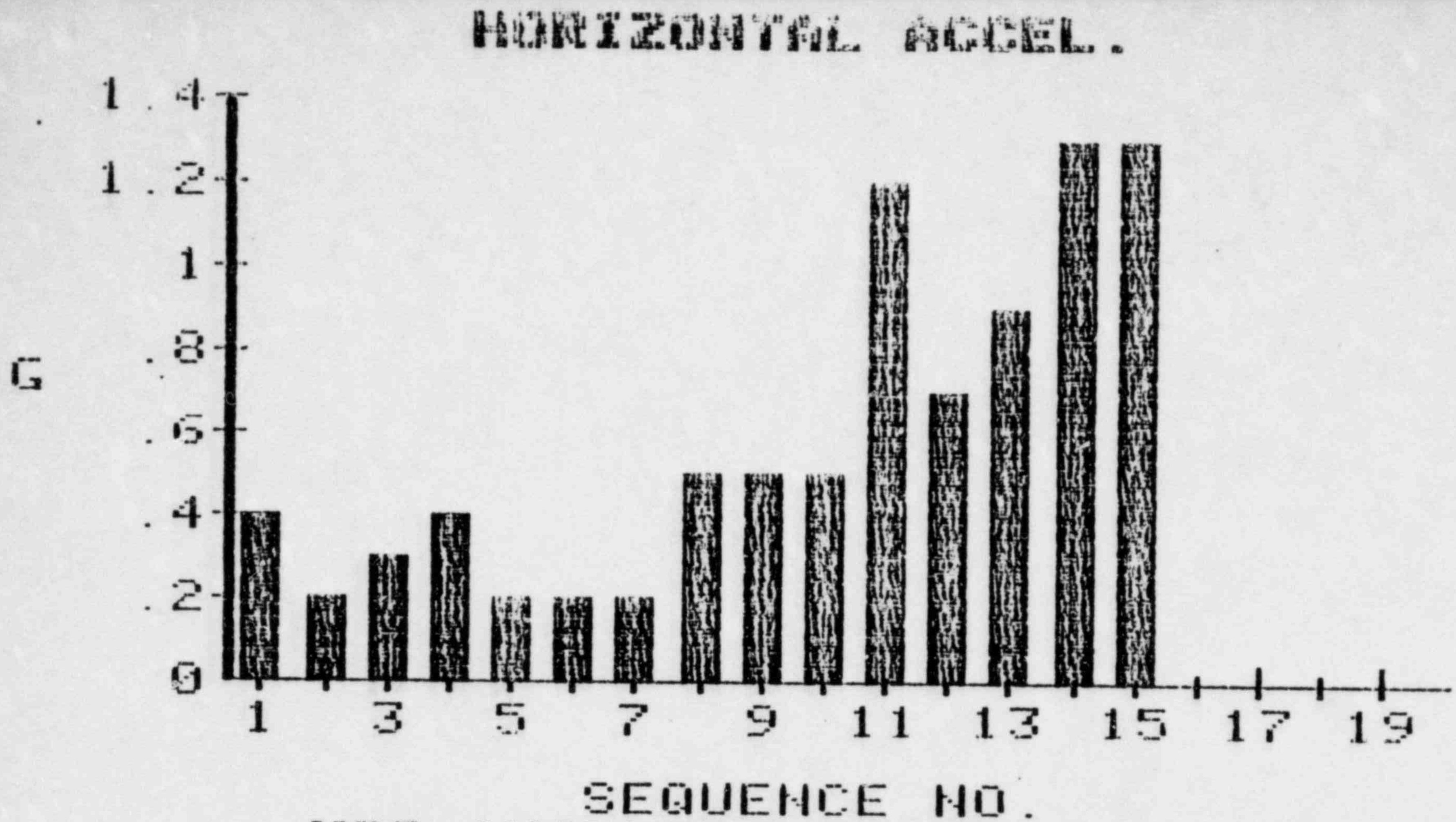
Using nozzle load. = 3g
valve no.

SEISMIC ANALYSIS AFTER FIXING
VALVE ACCELERATIONS

SYSTEM	SEQUENCE	#VALVES	YCS H MAX	YCS V MAX	NRC H MAX	NRC V MAX
21	1	1	0.40	0.40	0.50	0.60
22	2	1	0.20	0.20	0.30	0.10
23	3	1	0.30	0.60	0.50	0.80
24	4	1	0.40	0.30	0.70	0.50
41A	5	1	0.20	0.10	0.30	0.10
41B	6	1	0.20	0.10	0.30	0.10
41C	7	2	0.20	0.10	0.60	0.20
101	8	4	0.50	0.60	0.90	0.90
102	9	4	0.50	0.40	0.80	0.80
103	10	4	0.50	0.50	0.80	0.90
104	11	4	1.20	0.70	0.80	0.80
121	12	2	0.70	1.00	1.10	1.00
122	13	2	0.90	0.60	1.10	0.70
201	14	2	1.30	0.70	2.20	1.10
207	15	25	1.30	0.70	2.20	1.10
1	16	0	0.00	0.00	0.00	0.00
2	17	0	0.00	0.00	0.00	0.00
3	18	0	0.00	0.00	0.00	0.00
4	19	0	0.00	0.00	0.00	0.00

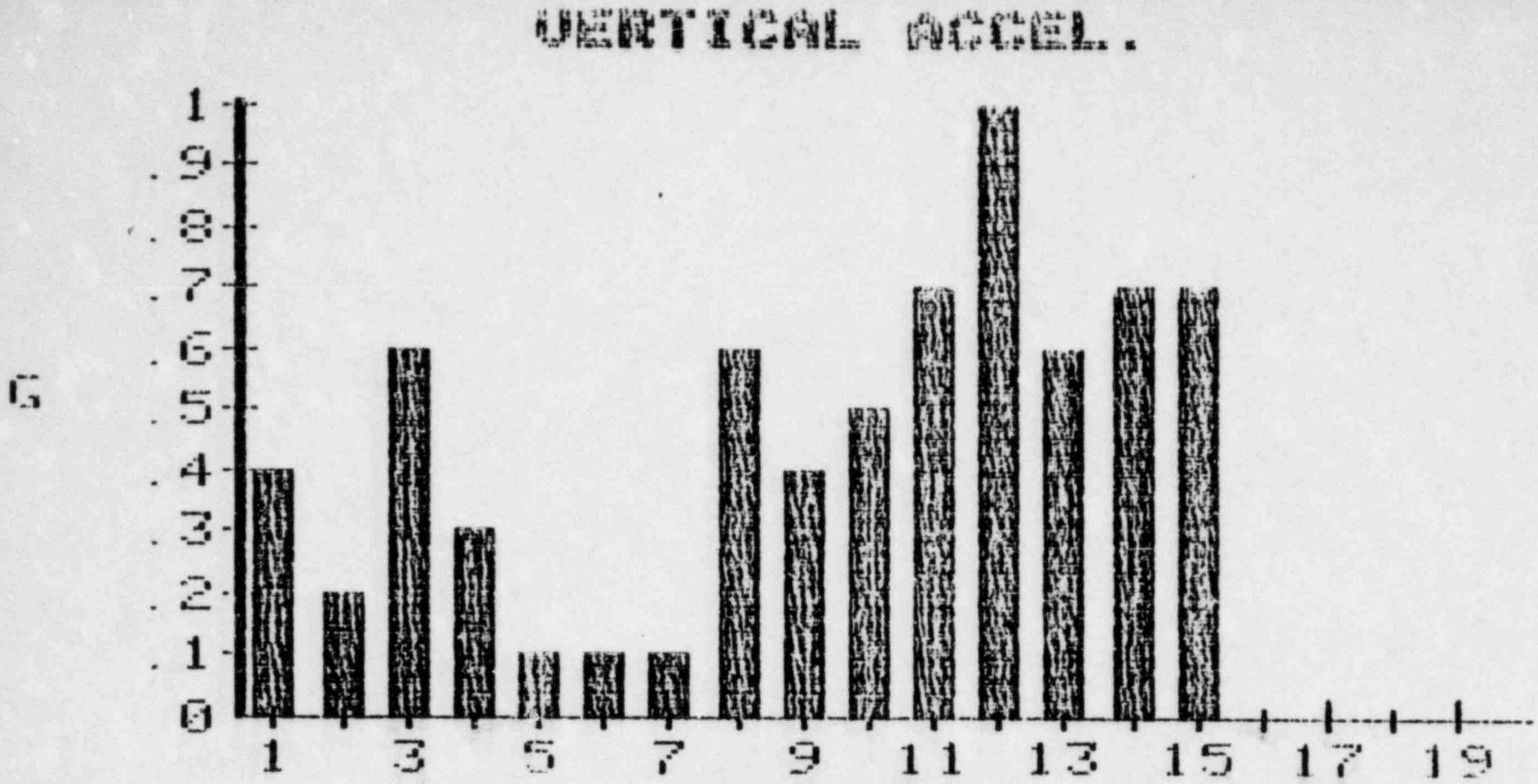
HOT SHUTDOWN INSIDE VC

PRELIMINARY



YCB ANALYSIS AFTER FIXING HOT SHUTDOWN INSIDE VC

PRELIMINARY

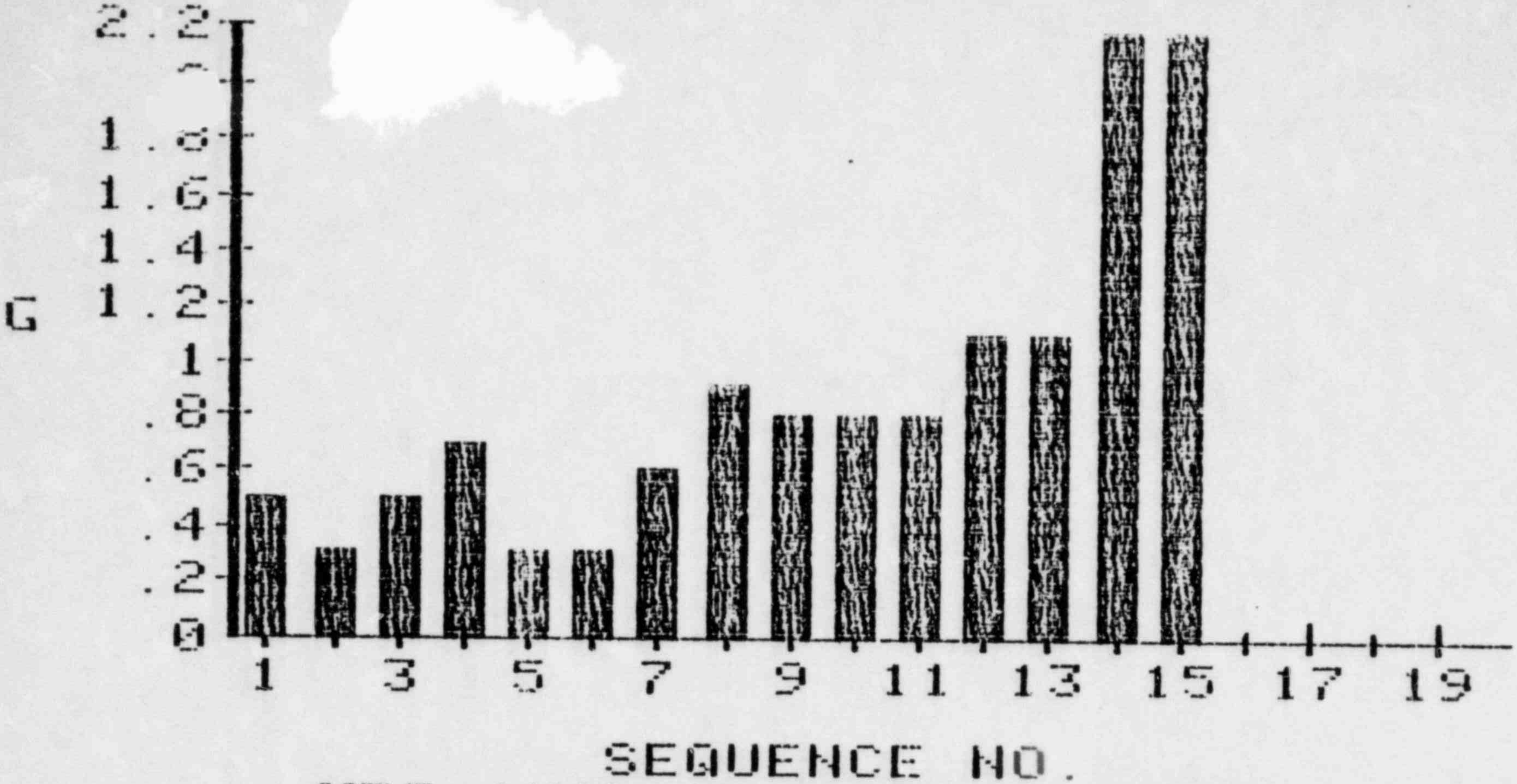


**YCS ANALYSIS AFTER FIRING
HOT SHUTDOWN INSIDE VC**

PRELIMINARY

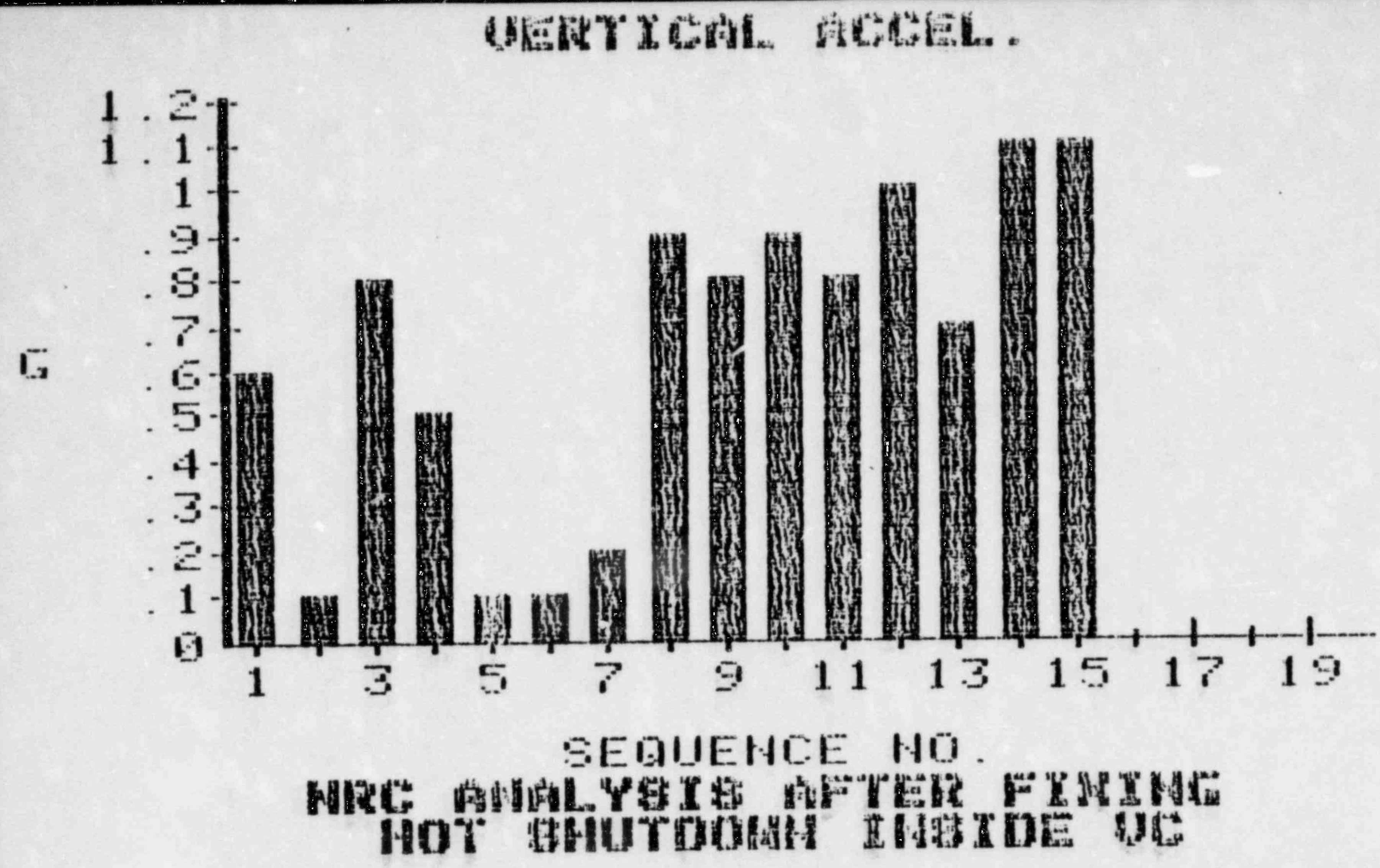
D-1

HORIZONTAL ACCEL.



NRC ANALYSIS AFTER FIXING
HOT SHUTDOWN INSIDE VC

PRELIMINARY



PRELIMINARY

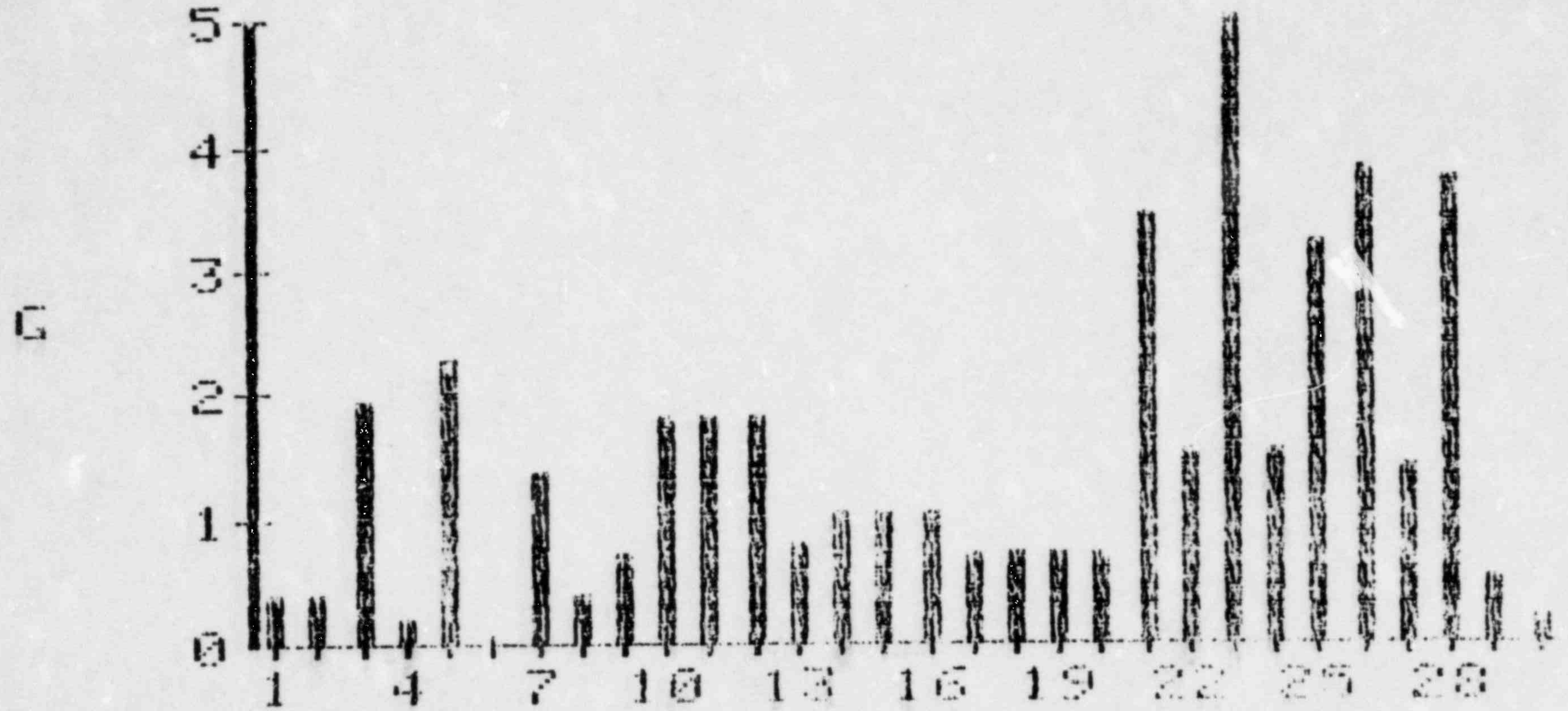
SEISMIC ANALYSIS USING YCS
VALVE ACCELERATIONS

SYSTEM	SEQUENCE	#VALVES	H MAX	V MAX
29	1	12	0.40	0.20
30	2	2	0.40	0.20
42	3	1	1.90	0.40
43A	4	3	0.20	0.20
43B	5	3	2.30	1.00
44	6	0	0.00	0.00
141	7	2	1.34	0.90
142	8	3	0.40	0.30
143	9	5	0.70	0.30
161	10	5	1.80	0.30
162	11	5	1.80	0.30
163	12	5	1.80	0.30
181	13	2	0.80	0.60
182	14	8	1.03	0.53
183	15	8	1.03	0.53
184	16	8	1.03	0.53
185	17	1	0.70	0.10
186	18	8	0.70	0.30
187	19	8	0.70	0.30
188	20	8	0.70	0.30
202	21	12	3.40	0.90
203	22	2	1.50	0.20
204	23	2	5.00	1.40
205	24	6	1.50	0.80
206	25	21	3.20	1.00
208	26	4	3.80	1.20
209	27	6	1.40	0.80
210	28	7	3.70	1.20
261	29	3	0.50	0.50
262	30	2	0.20	0.20

ECCS/AUX FEED SYSTEMS

PRELIMINARY

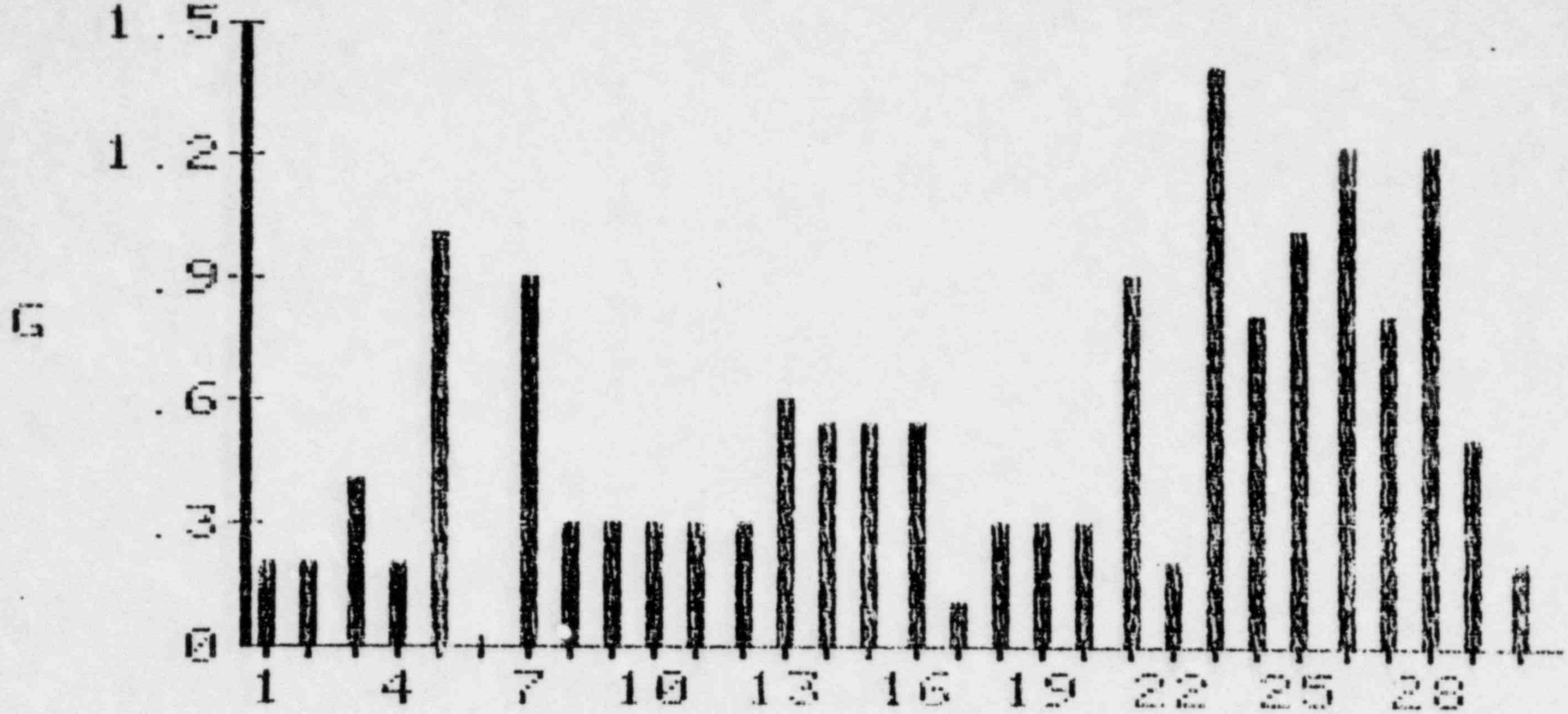
HORIZONTAL ACCEL.



YCS ANALYSIS
ECCS/AUX FEED SYSTEMS

PRELIMINARY

VERTICAL ACCEL.



SEQUENCE NO.
VCS ANALYSIS
ECCS/AUX FEED SYSTEMS

PRELIMINARY

EQUIPMENT ANCHORAGE EVALUATION

SEISMIC INPUT: NRC SPECTRUM WITH 7% CRITICAL DAMPING

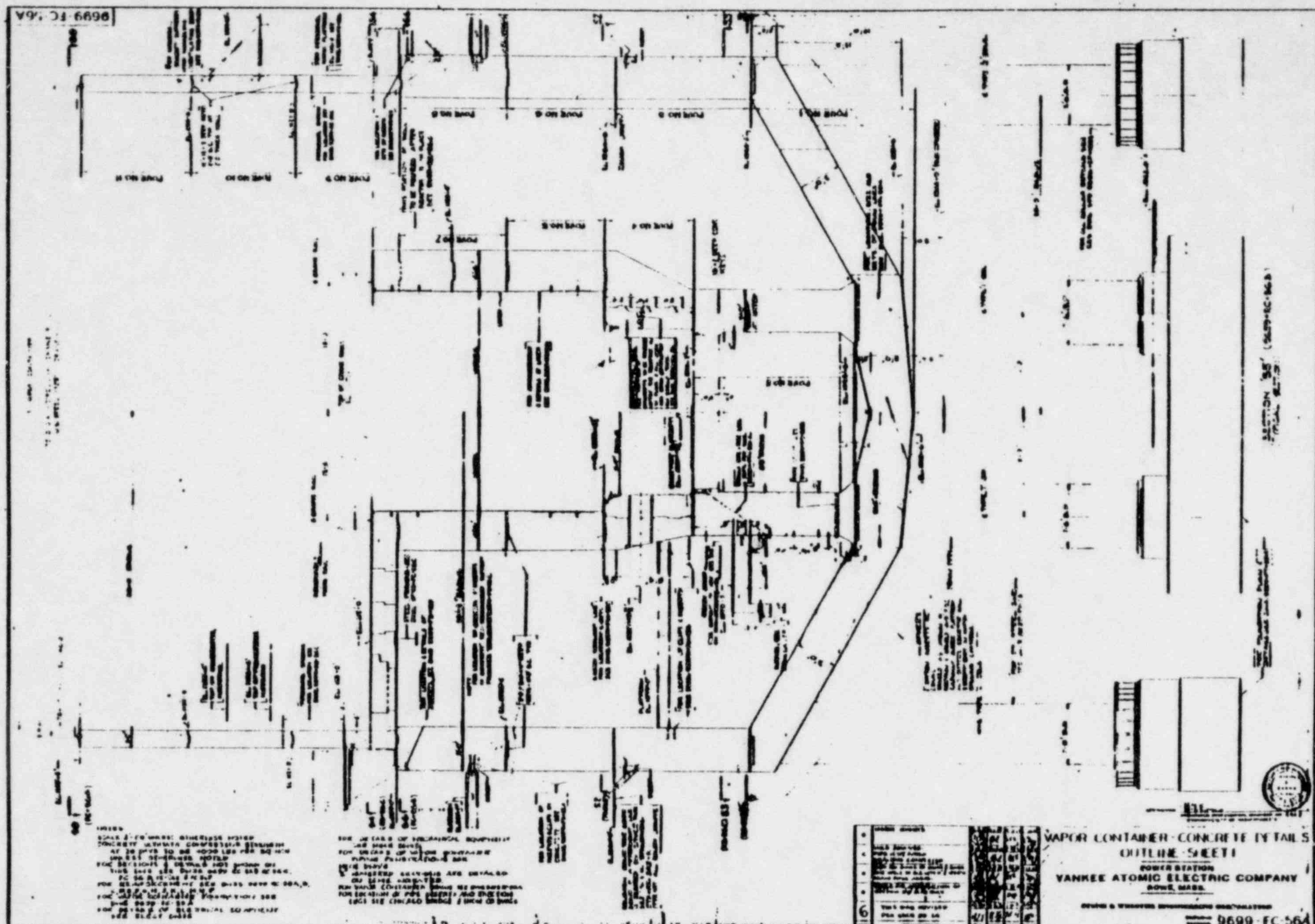
MAJOR EQUIPMENTS IN SCOPE:

- REACTOR PRESSURE VESSEL
- STEAM GENERATOR
- PRESSURIZER
- FIRE TANK

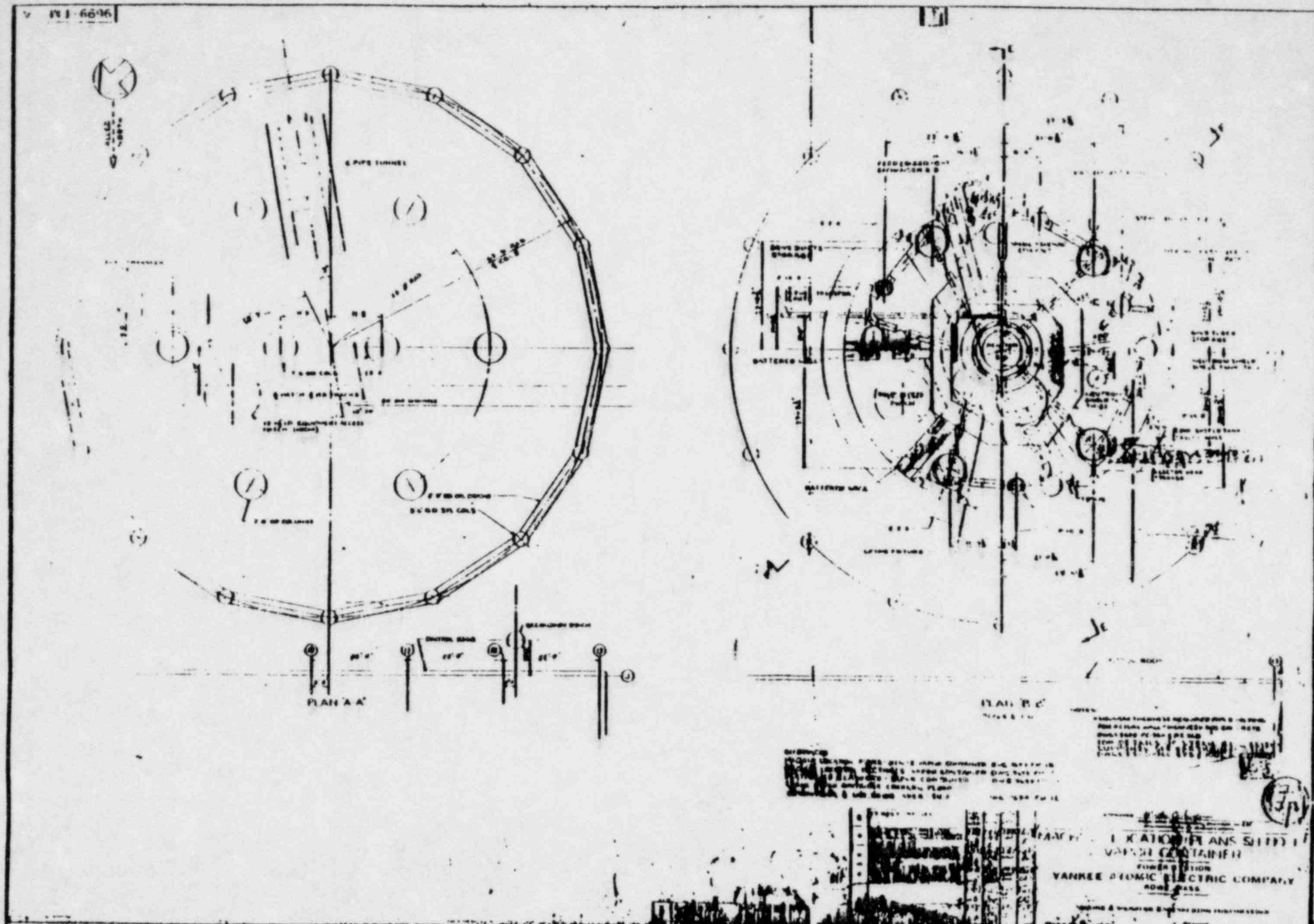
OBJECTIVE:

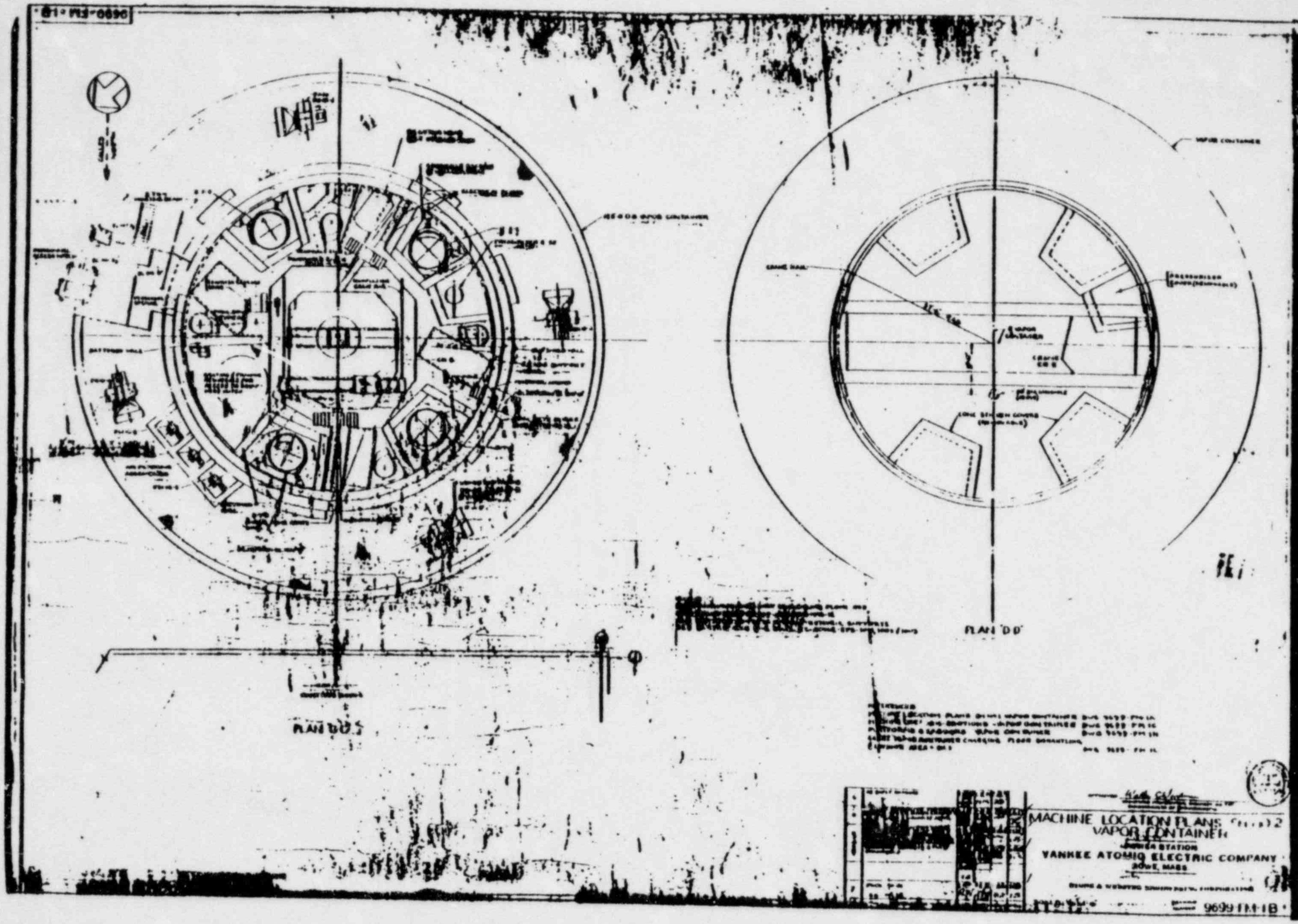
EVALUATE STRUCTURAL ADEQUACY OF ANCHORAGE TO SUPPORTING STRUCTURES OF MAJOR EQUIPMENT WHEN SUBJECTED TO SEISMIC FORCES BY THE NRC SPECTRUM.

D-19

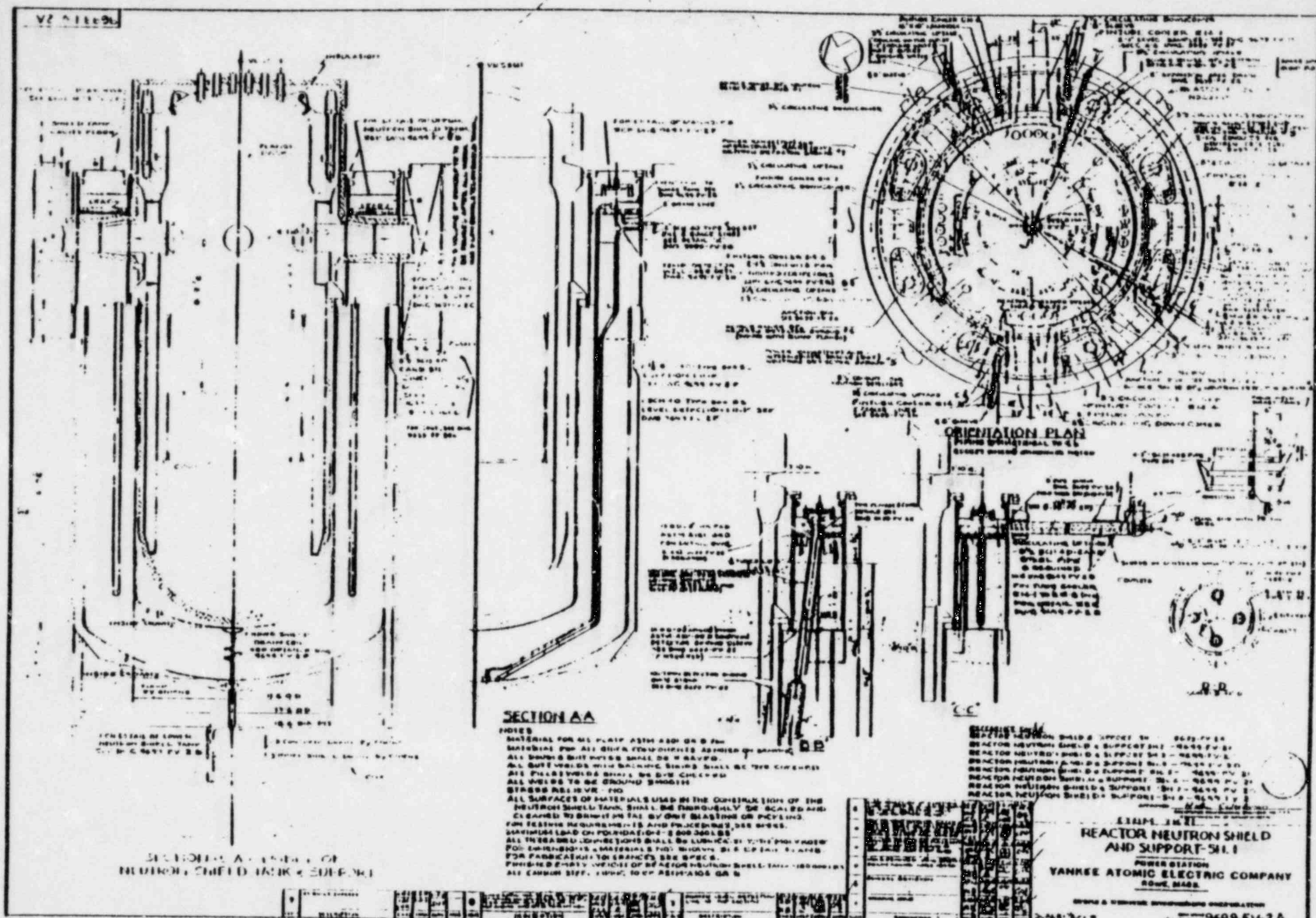


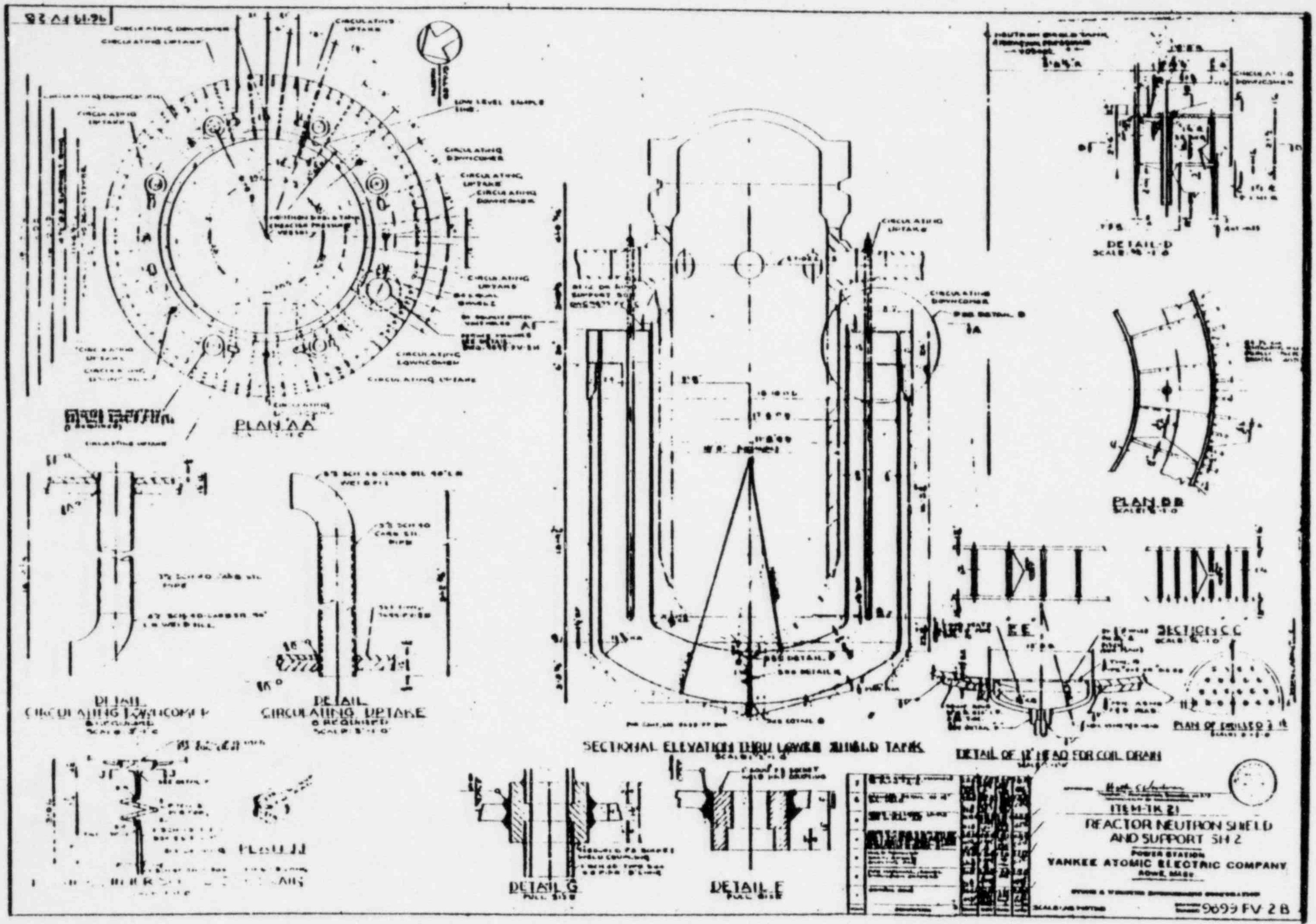
SECTION "SS"
VAPOR CONTAINER - CONCRETE DETAILS
OUTLINE SHEET 1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
9699-FC-56A

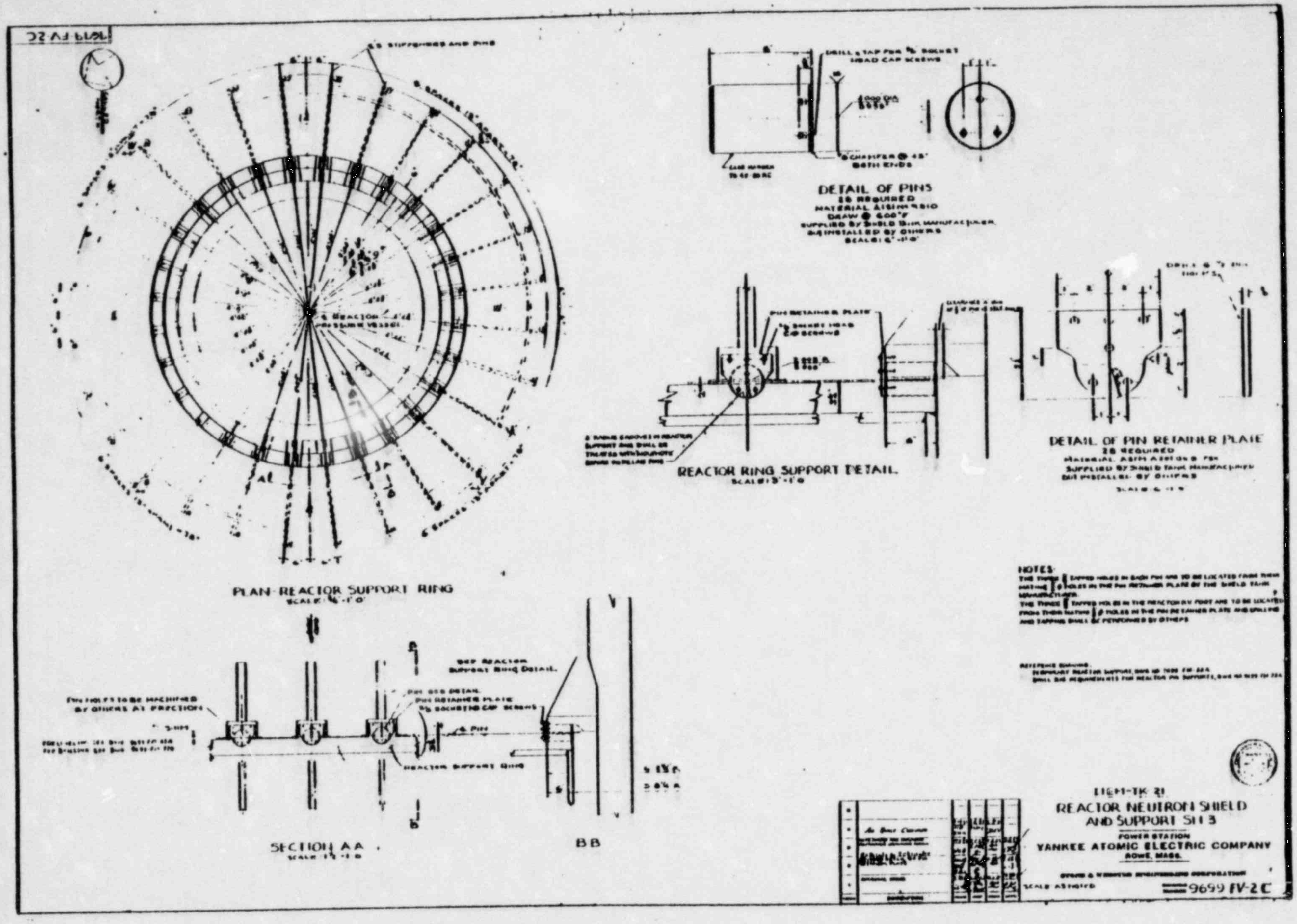

PLAN 'A-A'


PLAN 'DD'
MACHINE LOCATION PLANS
VAPOR CONTAINER
YANKEE ATOMIC ELECTRIC COMPANY
9699 FM-1B

D-22


SECTION AA
ORIENTATION PLAN
"C-C"
"D-D"
B-B
REACTOR NEUTRON SHIELD
AND SUPPORT-SH.1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
9699-FV-2A

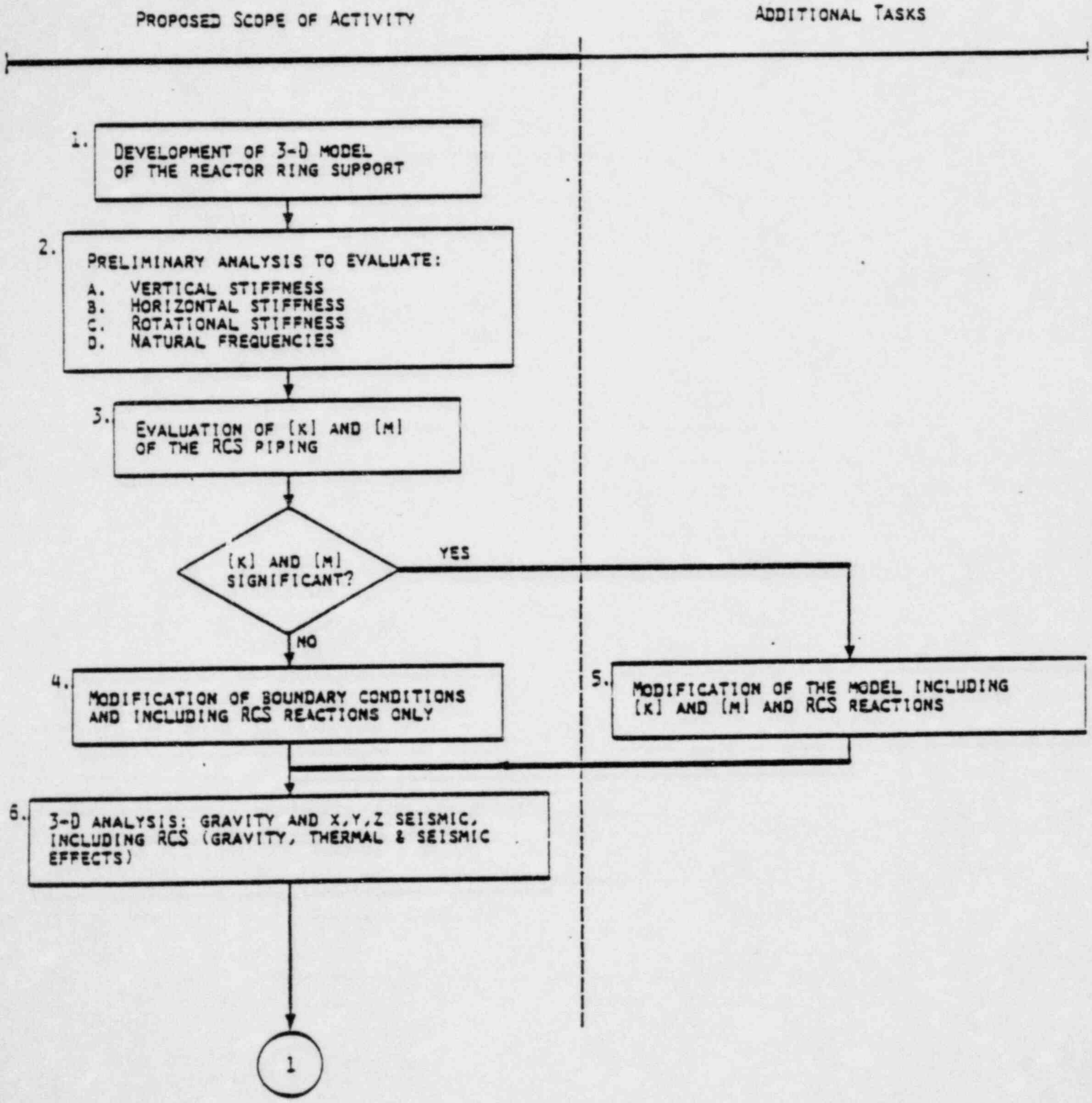

PLAN AA
CIRCULATING UPTAKE
CIRCULATING DOWNCOMER
LOW LEVEL SAMPLE LINE
DETAIL CIRCULATING DOWNCOMER
DETAIL CIRCULATING UPTAKE
SECTIONAL ELEVATION THRU LOWER SHIELD TANK
DETAIL D
PLAN BB
SECTION CC
PLAN OF DRILLED 3/4 R
DETAIL OF 12" HEAD FOR COIL DRAIN
DETAIL G
DETAIL E
ITEM-TK 21
REACTOR NEUTRON SHIELD
AND SUPPORT SH 2
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
9699 FV 2 B


DETAIL OF PINS
28 REQUIRED
MATERIAL AISI 410
DRAW @ 600°F
PLAN-REACTOR SUPPORT RING
REACTOR RING SUPPORT DETAIL
DETAIL OF PIN RETAINER PLATE
28 REQUIRED
SUPPLIED BY SHIELD TANK MANUFACTURER
BUT INSTALLED BY OTHERS
NOTES:
SECTION AA
BB
REACTOR SUPPORT RING
LIGHT-TK-21
REACTOR NEUTRON SHIELD
AND SUPPORT SH 3
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
9699 FV-2C

D-24

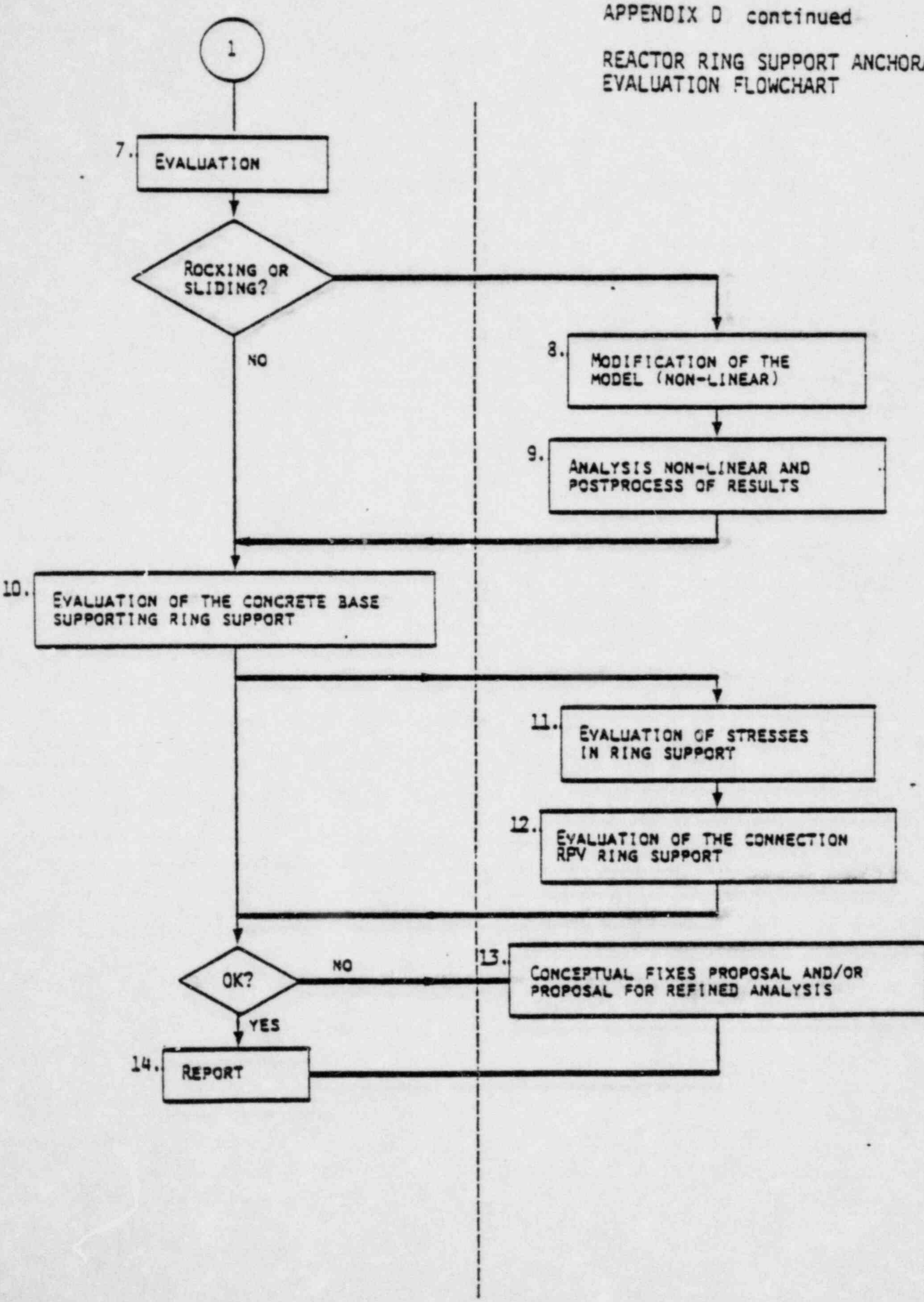
APPENDIX D

REACTOR RING SUPPORT ANCHORAGE EVALUATION FLOWCHART

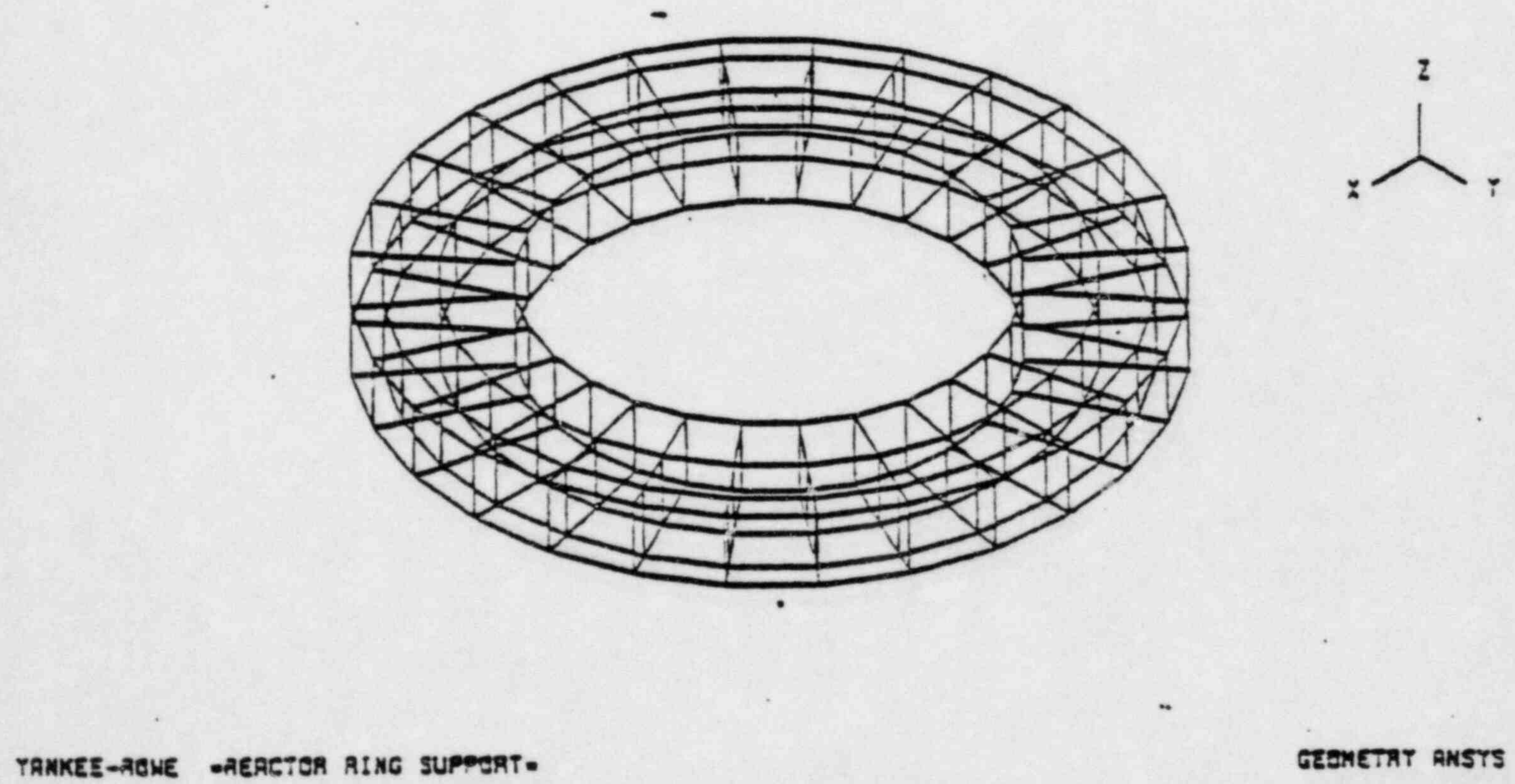


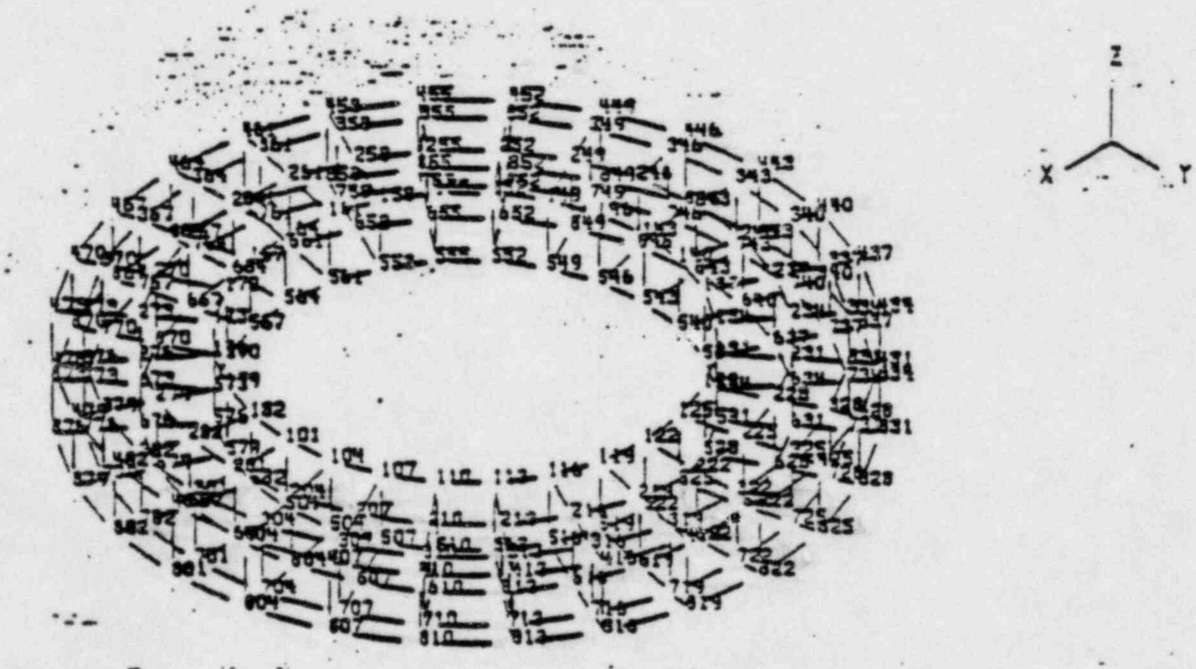
APPENDIX D continued

REACTOR RING SUPPORT ANCHORAGE EVALUATION FLOWCHART



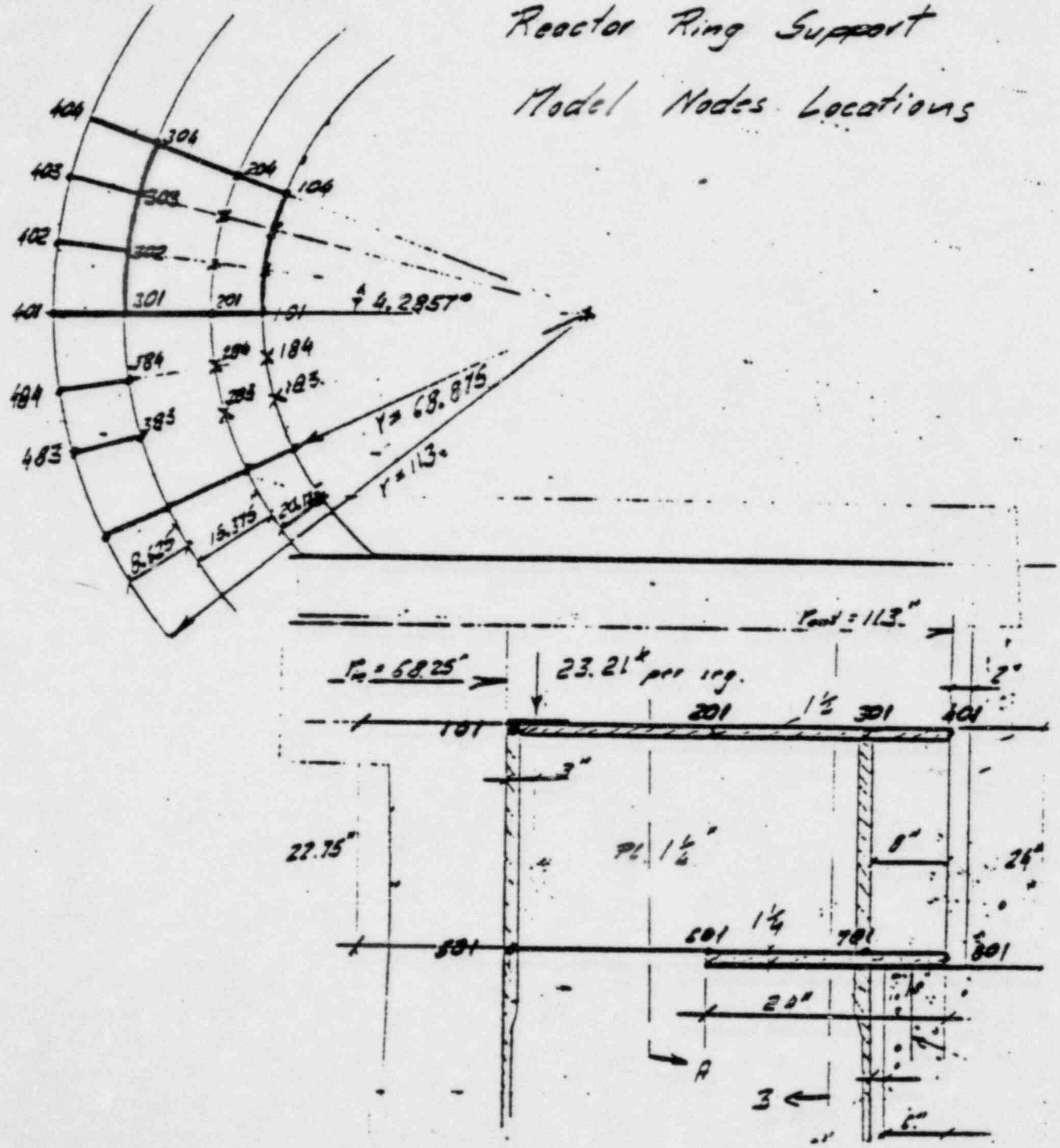
1.) 3-D Model of reactor ring support plots



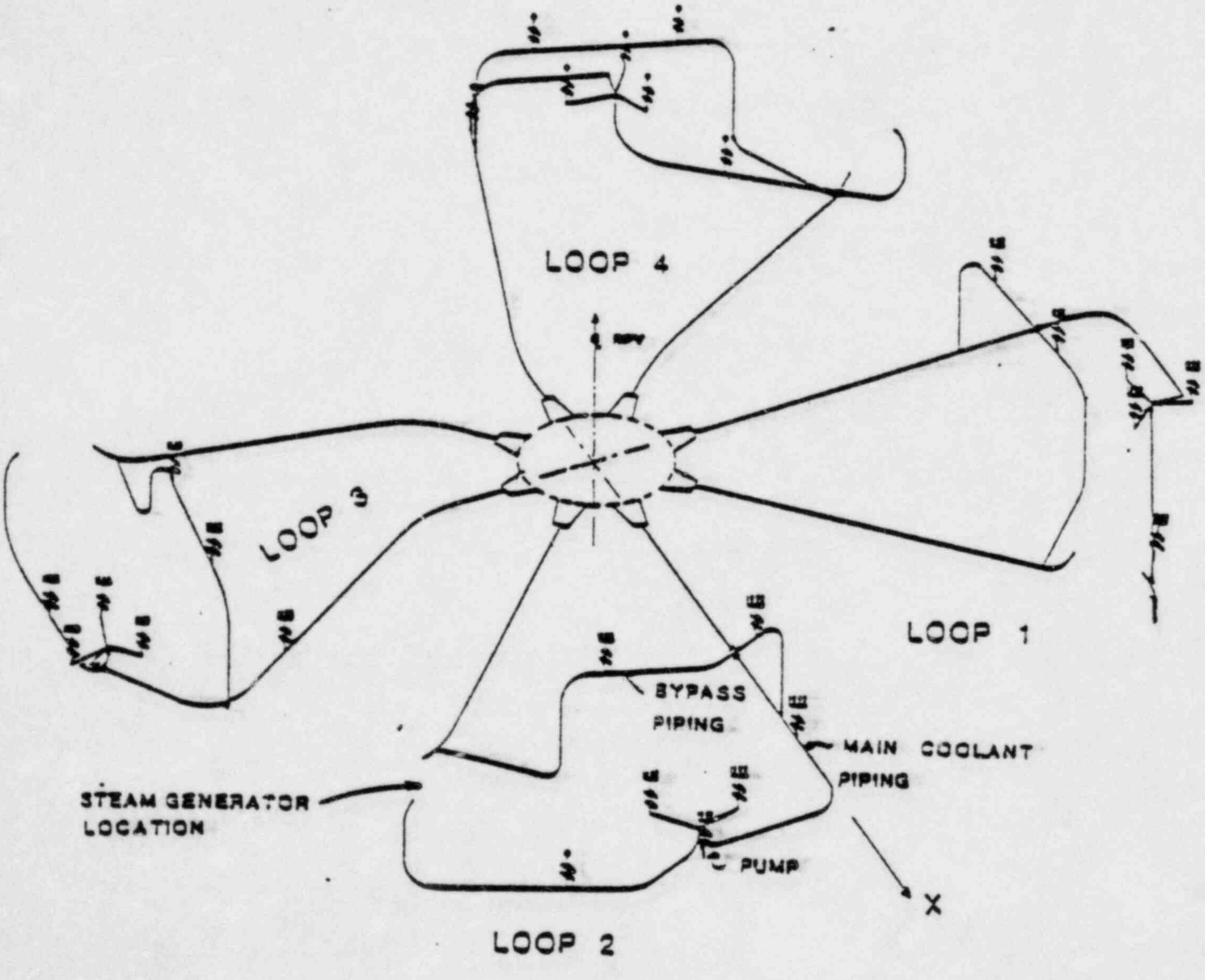

Z
X
Y

Reactor Ring Support

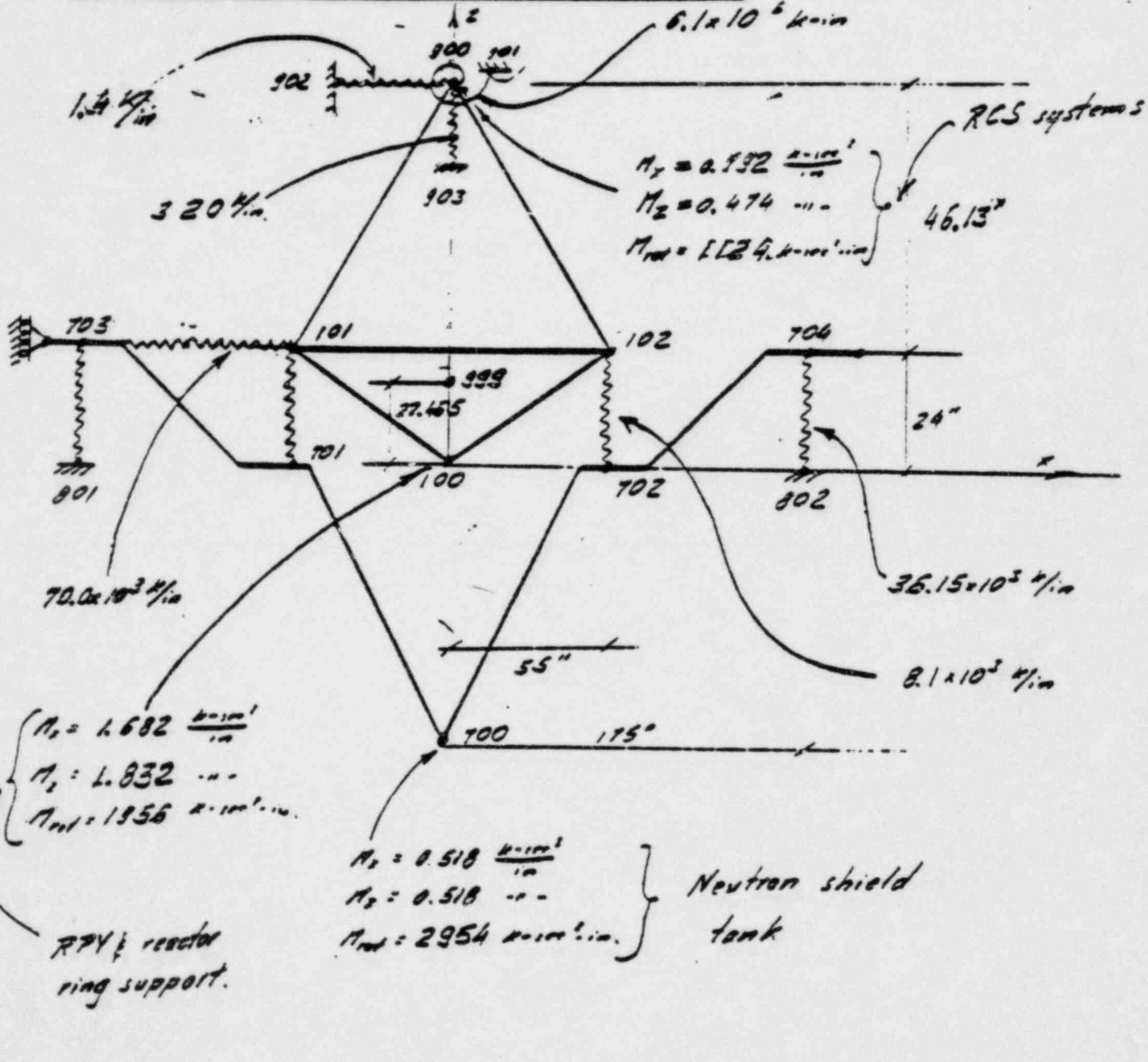
Model Nodes Locations



7. Mass & stiffness effects of RCS piping system:



9. Final 2-D model including RCS effects



10 SUMMARY OF MODEL FREQUENCIES

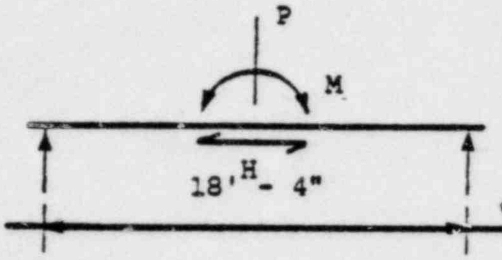
Model	Case	Freq. of vert. mode	Freq. of rotat. mode	Output file
3-D	1) f = ∞	19.2	--	MODA16
3-D	2) f = 0	--	--	---
2-D vert.	1) f = ∞	18.9	--	MOD16
2-D vert.	2) f = 0	11.9	--	MOD26
2-D vert.-rot.	1) f = ∞	18.9	32.1	SIM16
2-D vert.-rot.	2) f = 0	11.9	20.4	SIM26
2-D vert.-rot.	1) f = ∞			} includ. RCS [K] & [M]
2-D vert.-rot.	2) f = 0	11.0	15.4	FIN6

f = friction coefficient

PRELIMINARY

REACTOR RING SUPPORT
SUMMARY OF EVALUATION

A. LOADS



LOADS	$P_{(K)}$	$H_{(K)}$	$M_{(K-IN)}$
GRAVITY	1290	---	---
PRESSURE IN RCS	-234	---	---
THERMAL DUE TO RPV EXPANSION	268	---	---
NRC SPECTRUM:			
F_V = 11.0 Hz	±594	---	---
F_H = 27.8 Hz	---	273	---
F_R = 15.4 Hz	---	---	±823

B. FACTOR OF SAFETY AGAINST ROCKING, F. S. R.:

$$F.S.R. = \frac{M_R\ (MIN)}{M_{O.T.}\ (MAX)} = 27.5\ (\text{Reactor})$$

$$= 72.0\ (\text{Ring Support})$$

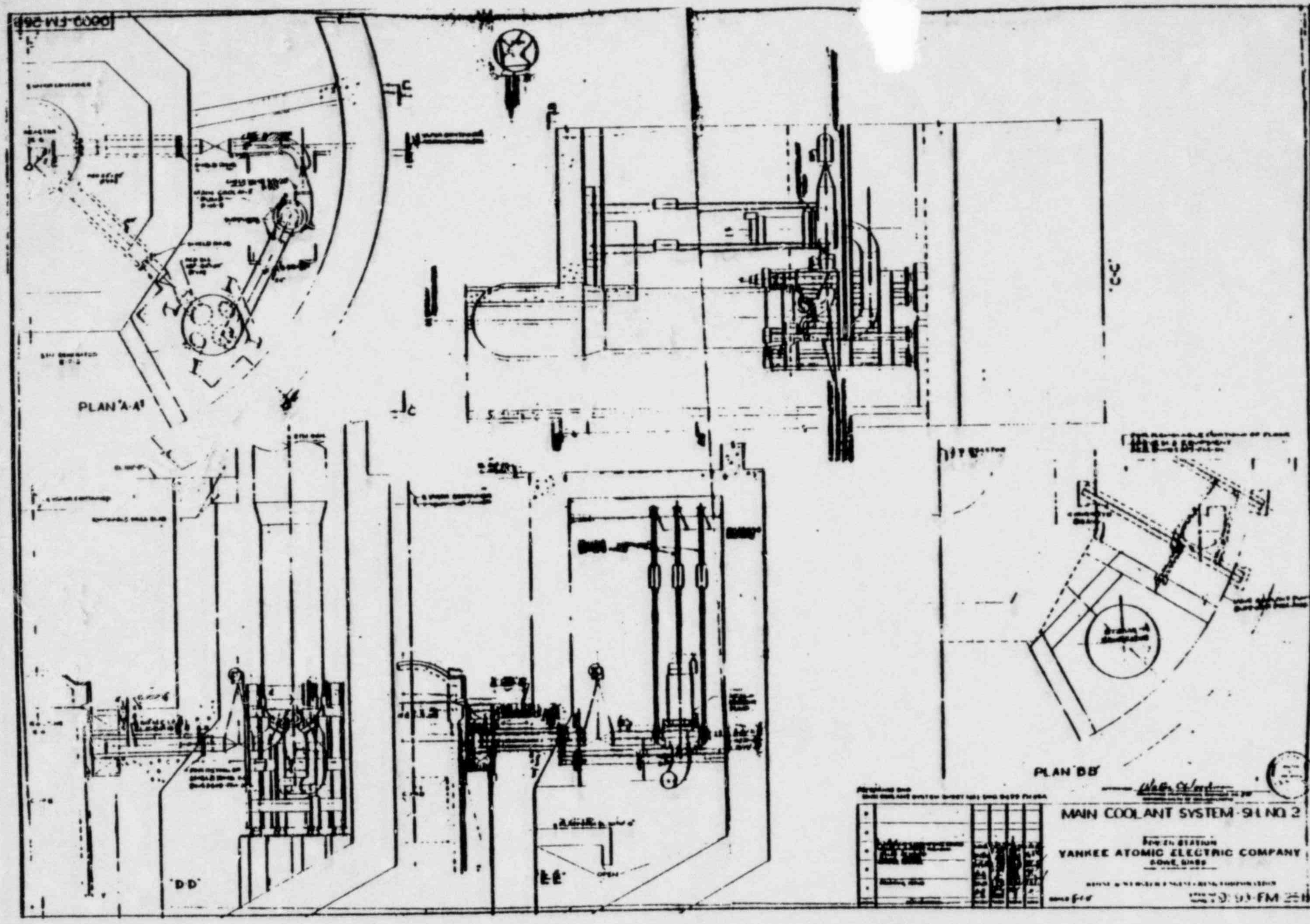
C. ACTUAL FRICTION COEFFICIENT:

$$\int \text{ACTUAL} = \frac{F_X\ (MAX)}{F_V\ (MIN)} = 0.392 = 0.40\ \text{OK}$$

.57

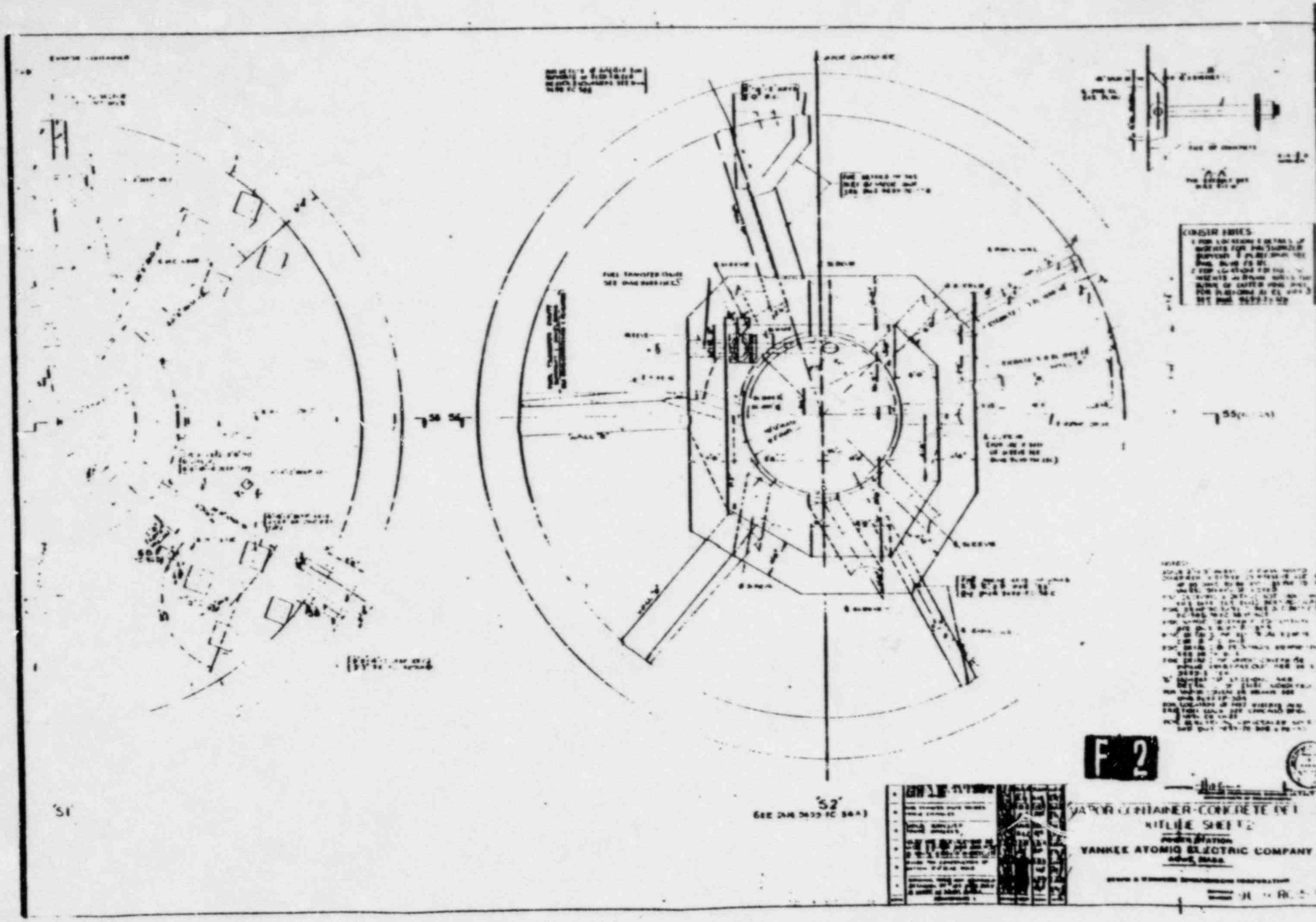
D. BEARING STRESS IN CONC. = 1041 PSI. < 1,780 PSI OK (0.85 ϕ FC')

PRELIMINARY

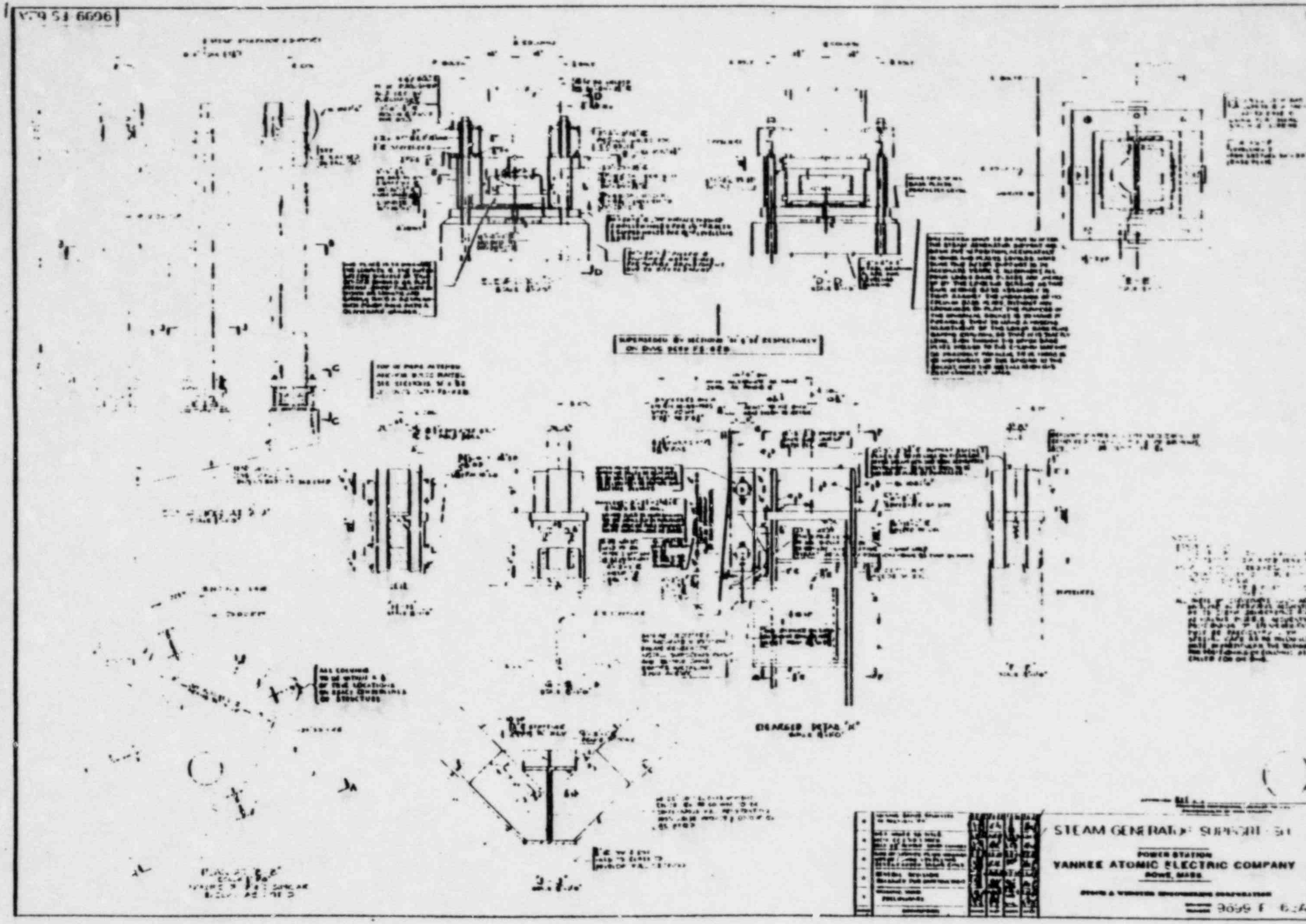


PLAN 'A-A'
'C-C'
PLAN 'BB'
'DD'
'EE'
MAIN COOLANT SYSTEM - SH NO. 2
YANKEE ATOMIC ELECTRIC COMPANY

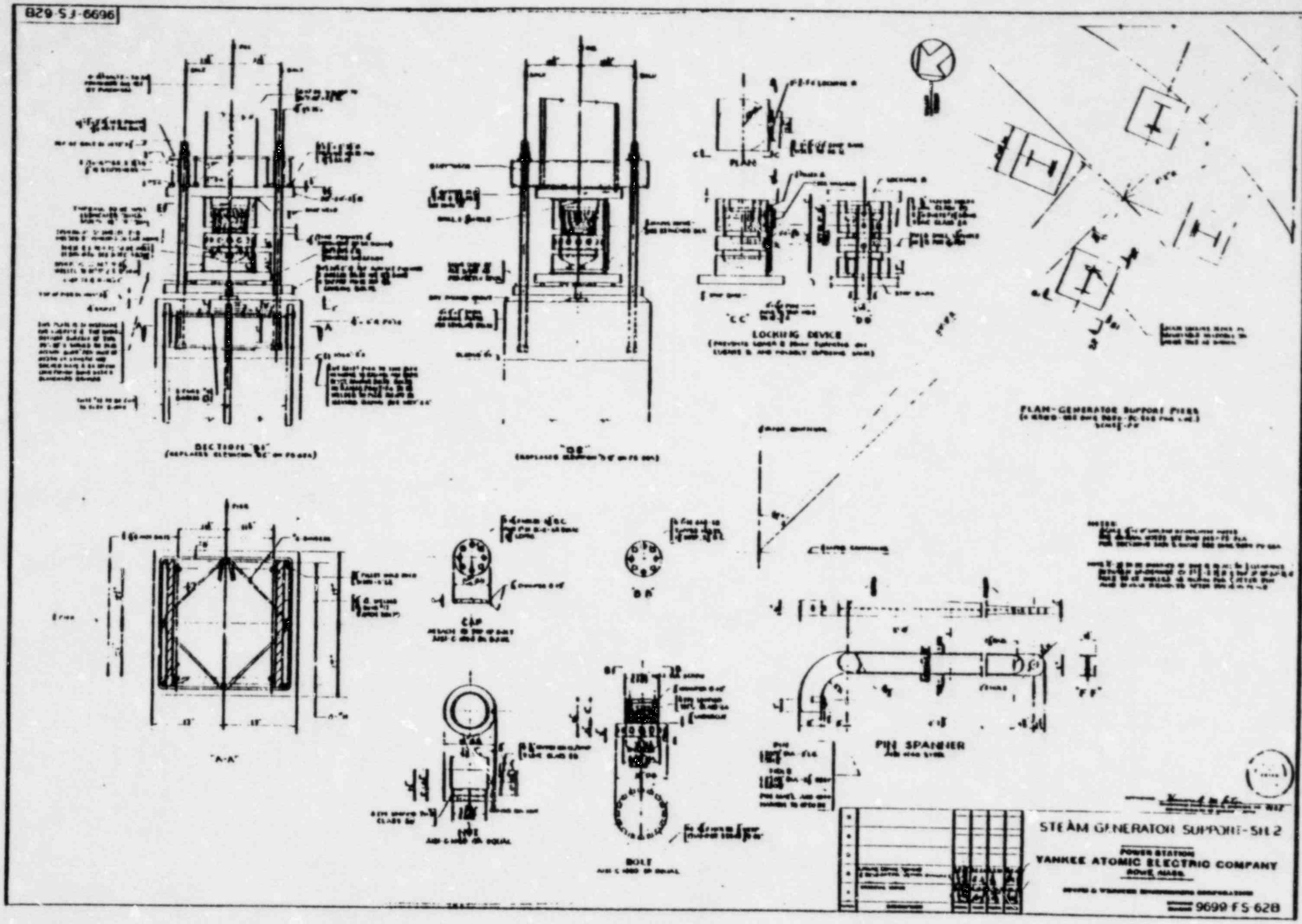
D-34


CONSTR NOTES
FUEL TRANSFER CHUTE
"S1"
"S2"
F 2
YANKEE ATOMIC ELECTRIC COMPANY

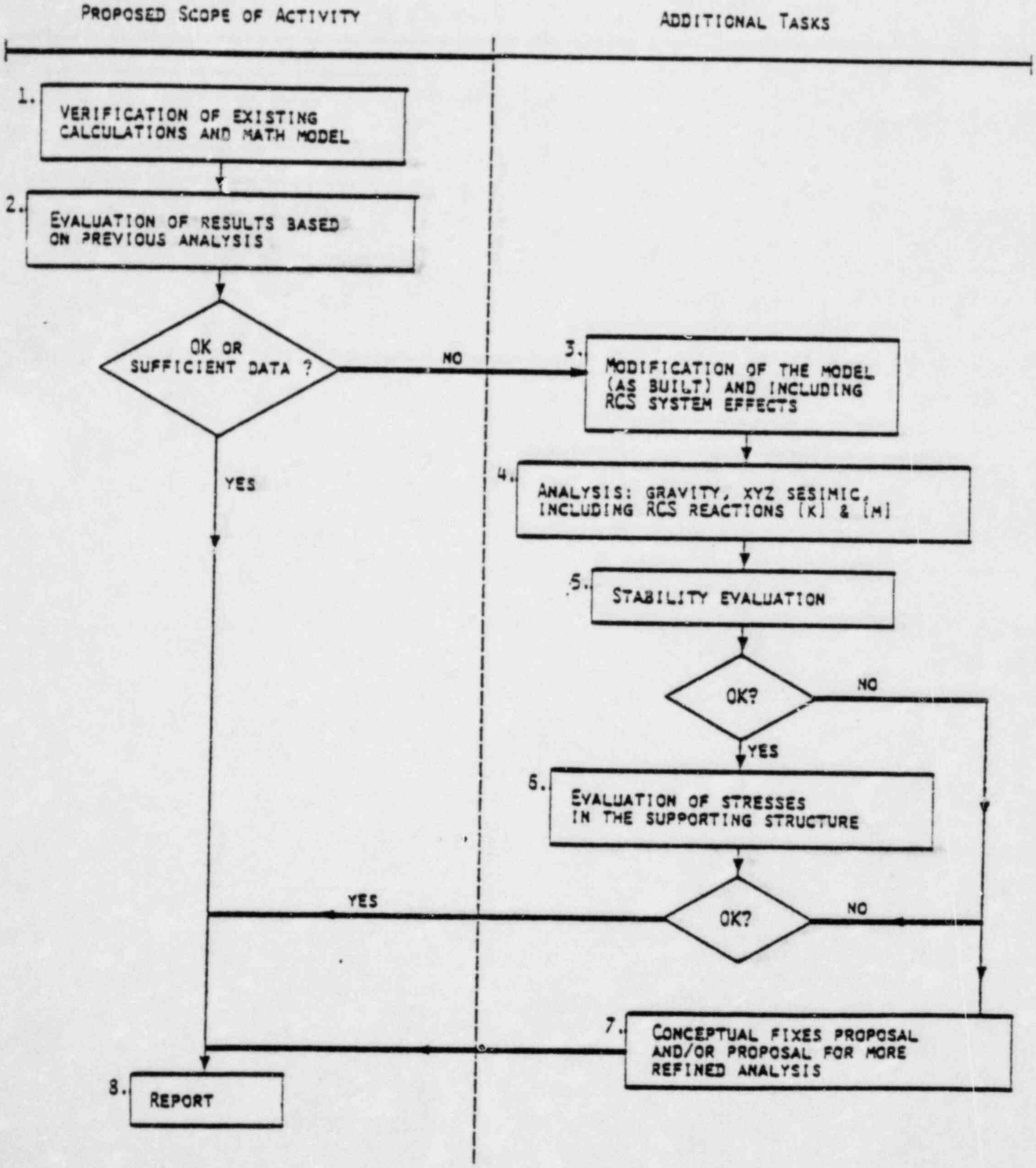
D-36



D-37


9699-FS-628
SECTION "BB"
"DB"
LOCKING DEVICE
"CC"
"DD"
PLAN
PLAN - GENERATOR SUPPORT PIERS
"A-A"
CAP
"B-B"
PIN SPANNER
PIN
BOLT
STEAM GENERATOR SUPPORT - SH. 2
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
9699 FS 628

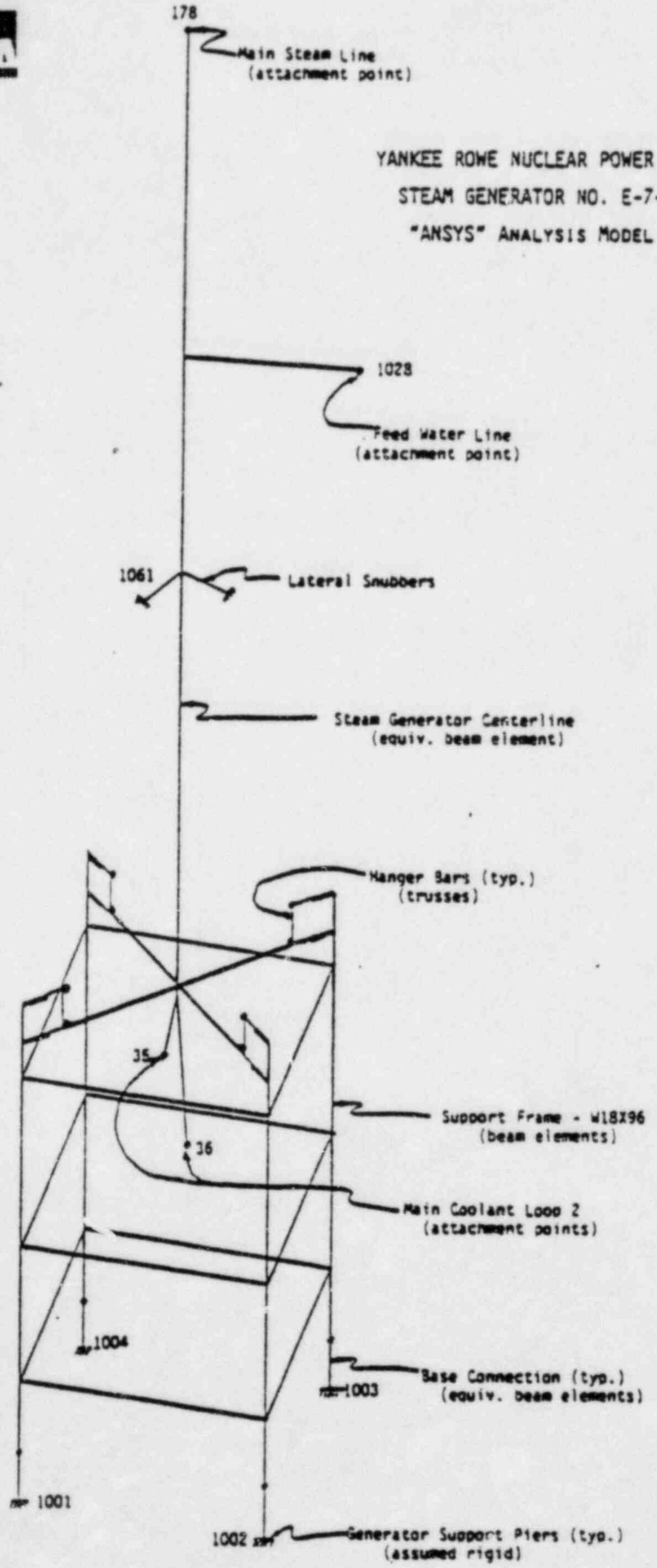
STEAM GENERATOR ANCHORAGE EVALUATION FLOWCHART



YANKEE ROWE NUCLEAR POWER STATION

STEAM GENERATOR NO. E-7-2

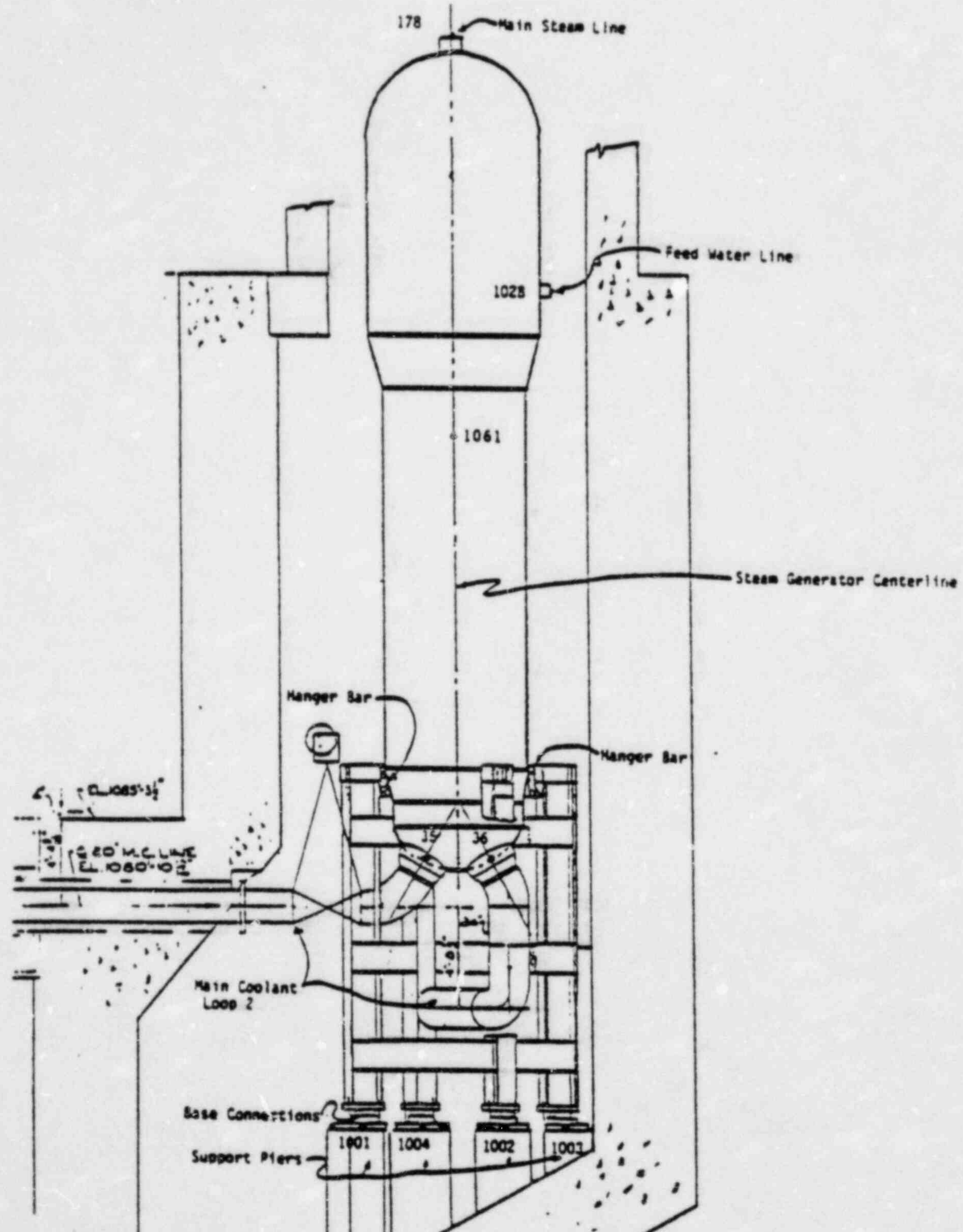
"ANSYS" ANALYSIS MODEL



0-40

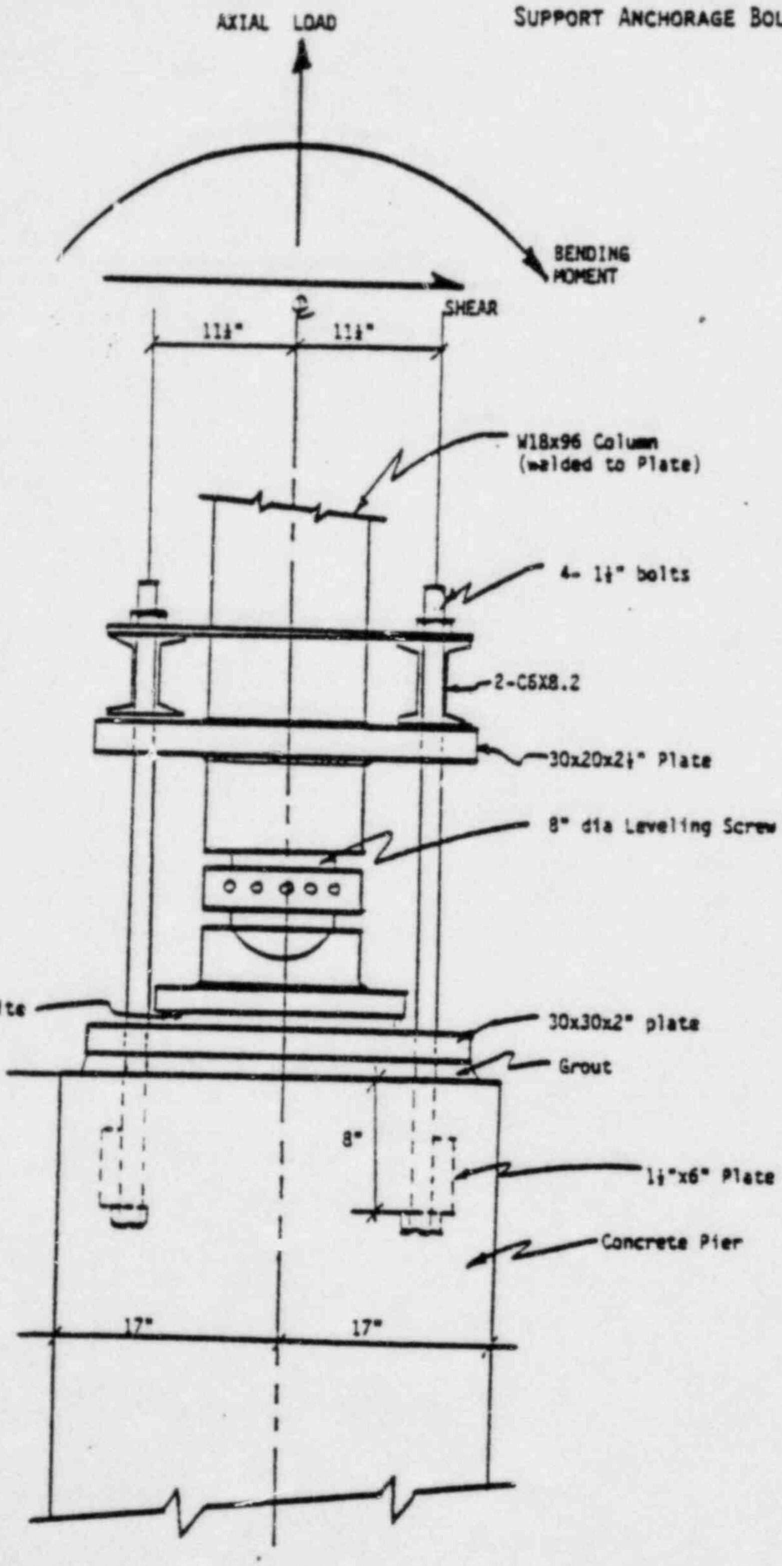
CYGNA

YANKEE ROWE NUCLEAR POWER STATION
STEAM GENERATOR NO. E-7-2
ELEVATION DETAIL



CYGNA

YANKEE ROWE NUCLEAR POWER STATION
STEAM GENERATOR NO. E-7-2
SUPPORT ANCHORAGE BOLT DETAIL



FORCES IN ANCHOR BOLTS

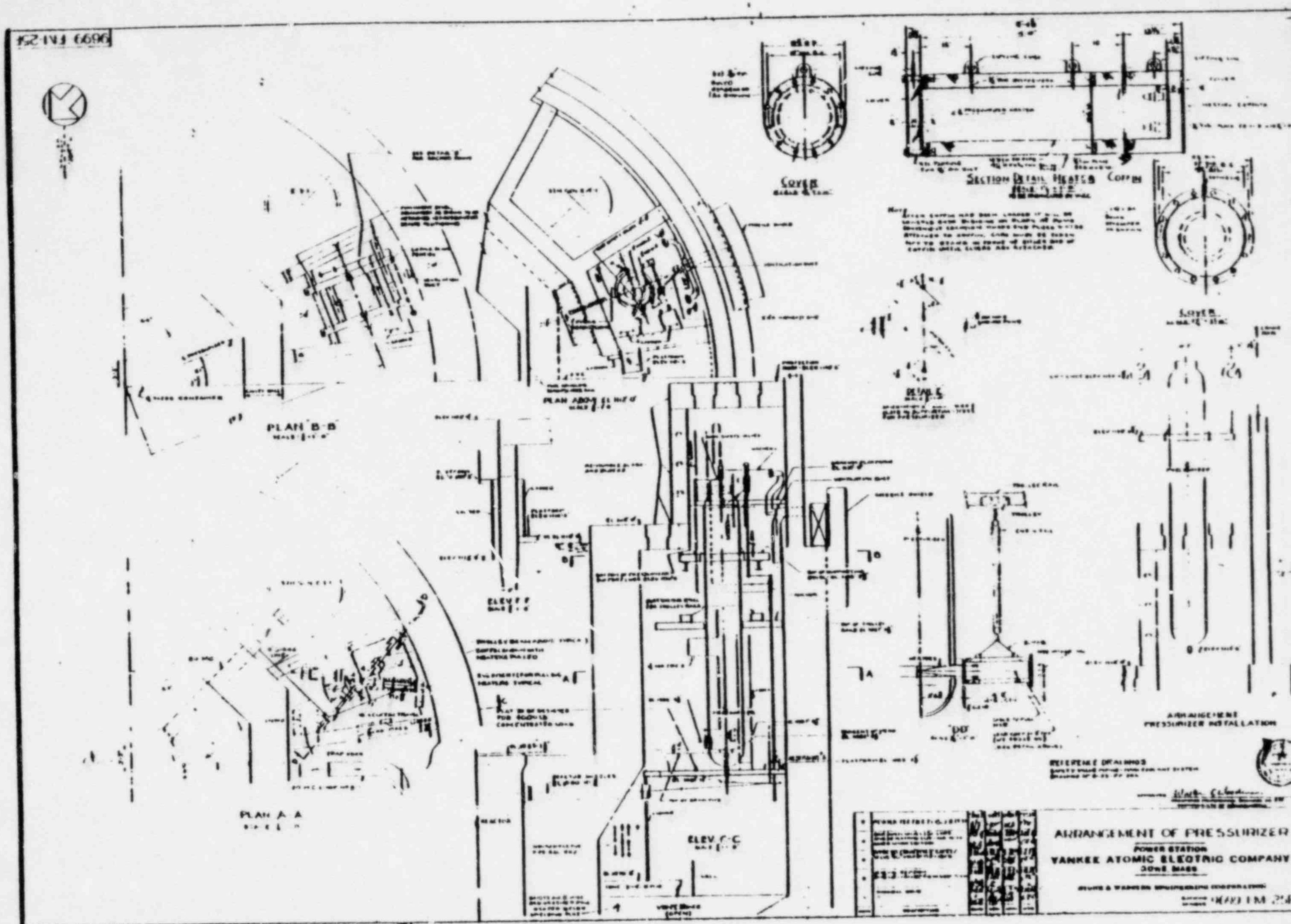
LOAD CASE	LOADING TYPE	SHEAR F_Y, F_Z, (LBS.)	AXIAL F_X (LBS)	FLEXURE M_Y, M_Z (IN-LBS)
1	GRAVITY	195.	-66,000.	5600.
2	TAM	-106.	9,100.	-2400.
3	NRC Spectrum (Structure)	4900.	21,100.	96,000.
4	SAM (Piping)	156,500.	137,500.	7,230,000.
5	1+2+3+4	161,500.	101,700.	7,243,000.
6	1-2-3-4	-161,100.	----	-7,318,000.
CAPACITIES		Bolts: 34,500 Friction: 44,000	2 A.B.: 75,000 4 A.B.: 150,000	35,200.

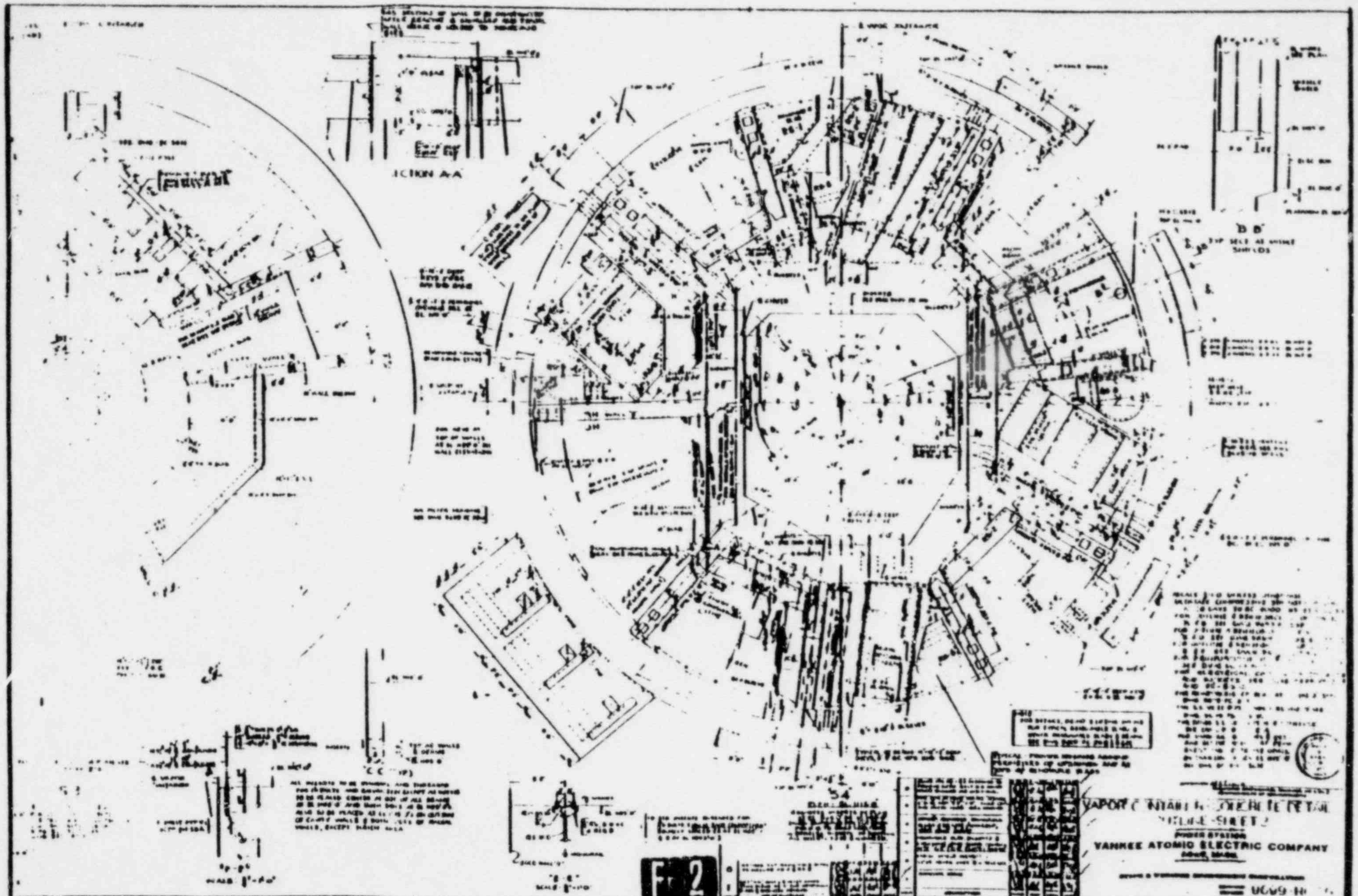
PRELIMINARY

STEAM GENERATOR

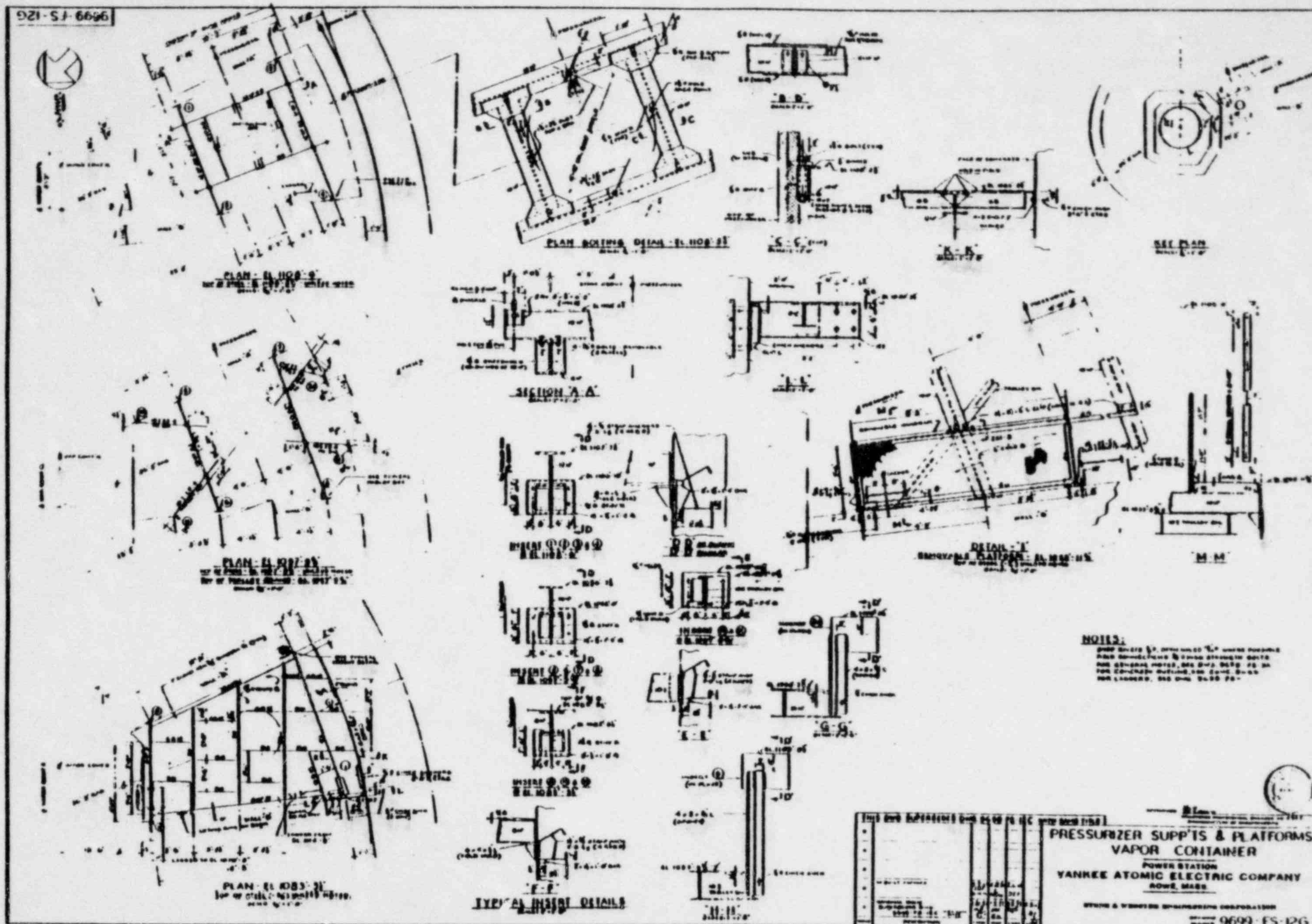
LOAD CASE	LOAD TYPE	FORCES IN SNUBBERS (PSA-100) AXIAL FORCE X-DIRECTION (LBS)	Z-DIRECTION (LBS)
A	NRC Spectrum (Piping)	7,300.	13,050.
B	SAM (Piping)	101,000.	104,330.
C	NRC Spectrum (Structure)	39,550.	40,130.
D	A + B + C	147,850.	157,510.

PRELIMINARY

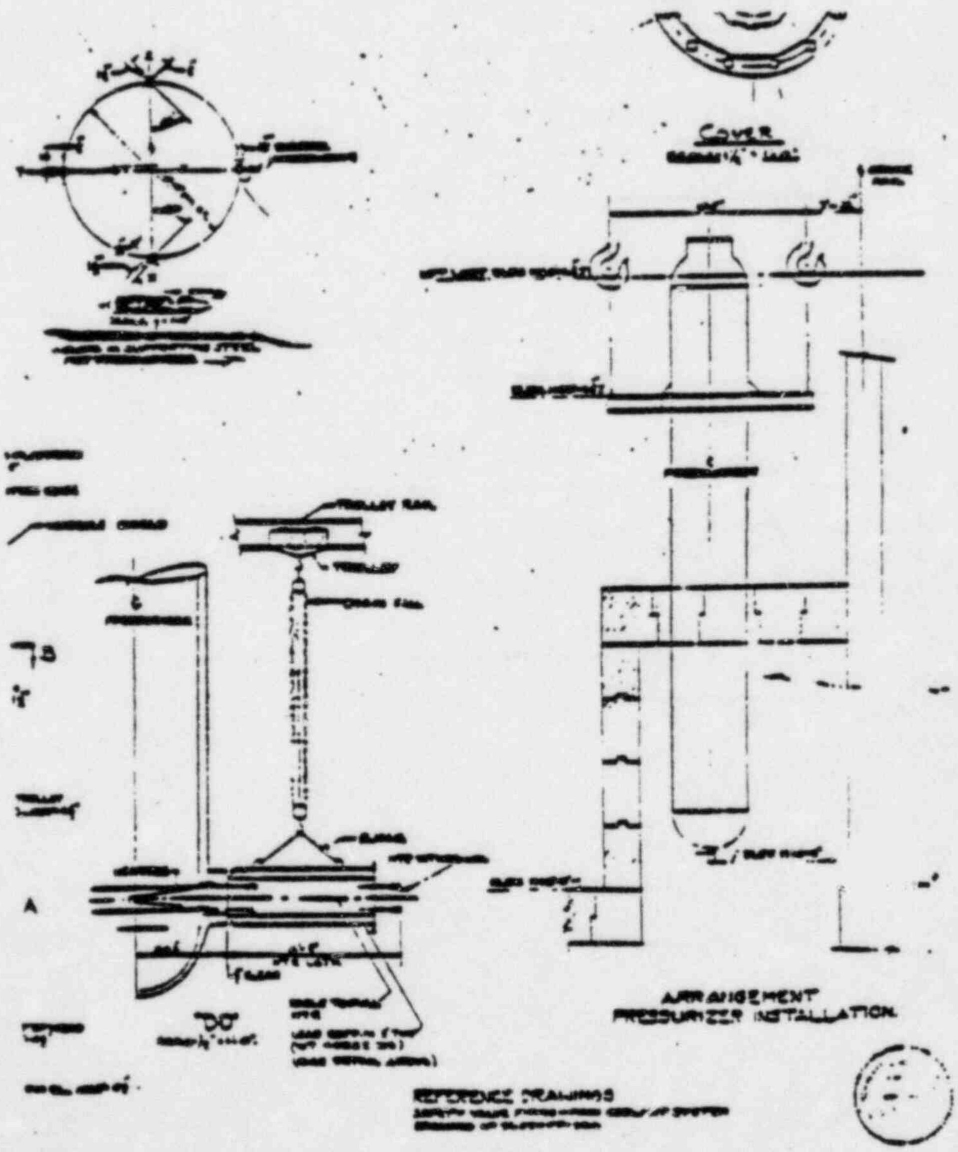

9699 FM 24
COVER
SECTION DETAIL HEATER COVER
COVER
PLAN B-B
PLAN A-A
ELEV F-F
ELEV C-C
ARRANGEMENT PRESSURIZER INSTALLATION
REFERENCE DRAWINGS
ARRANGEMENT OF PRESSURIZER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
9699 FM 24



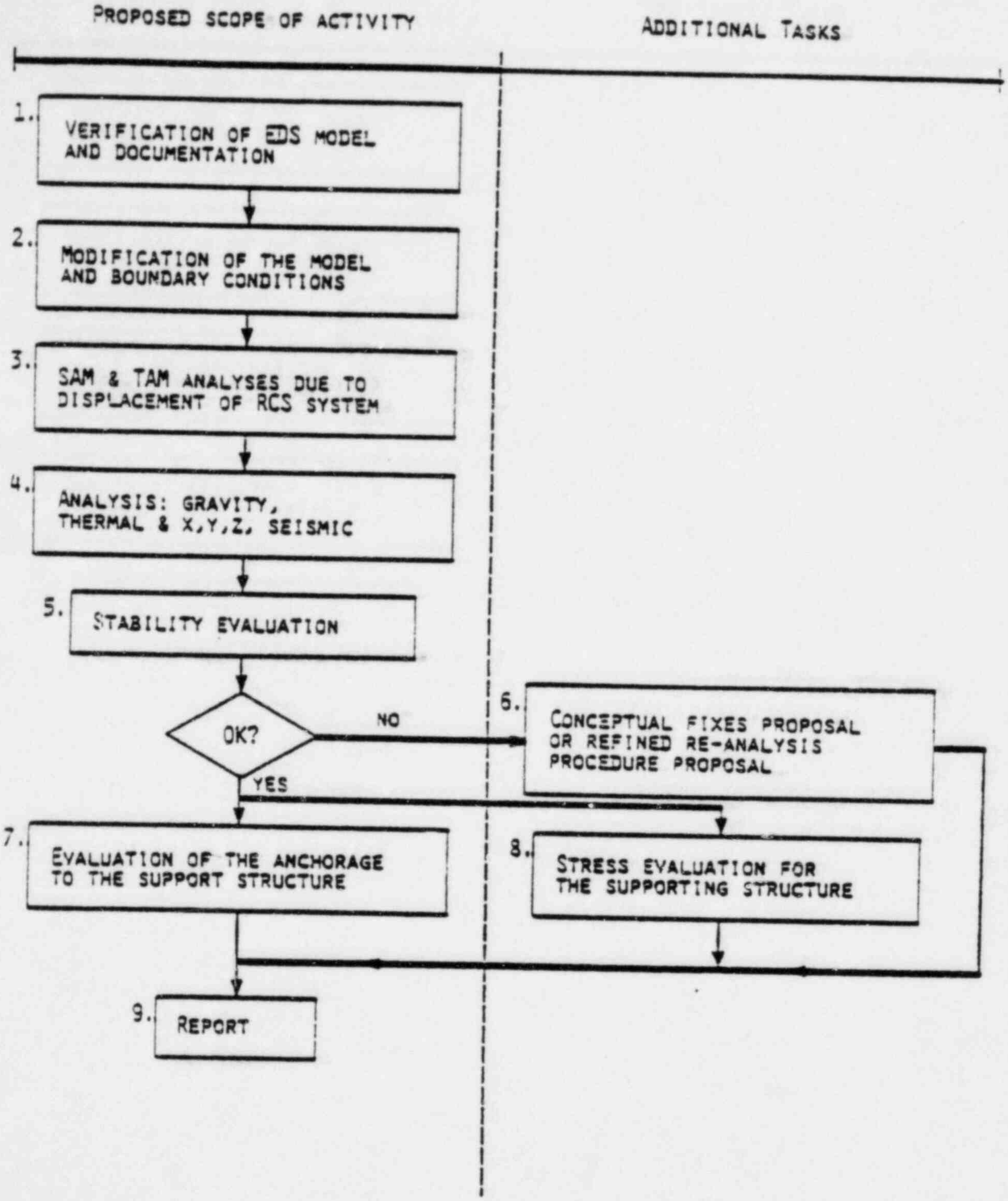
D-45

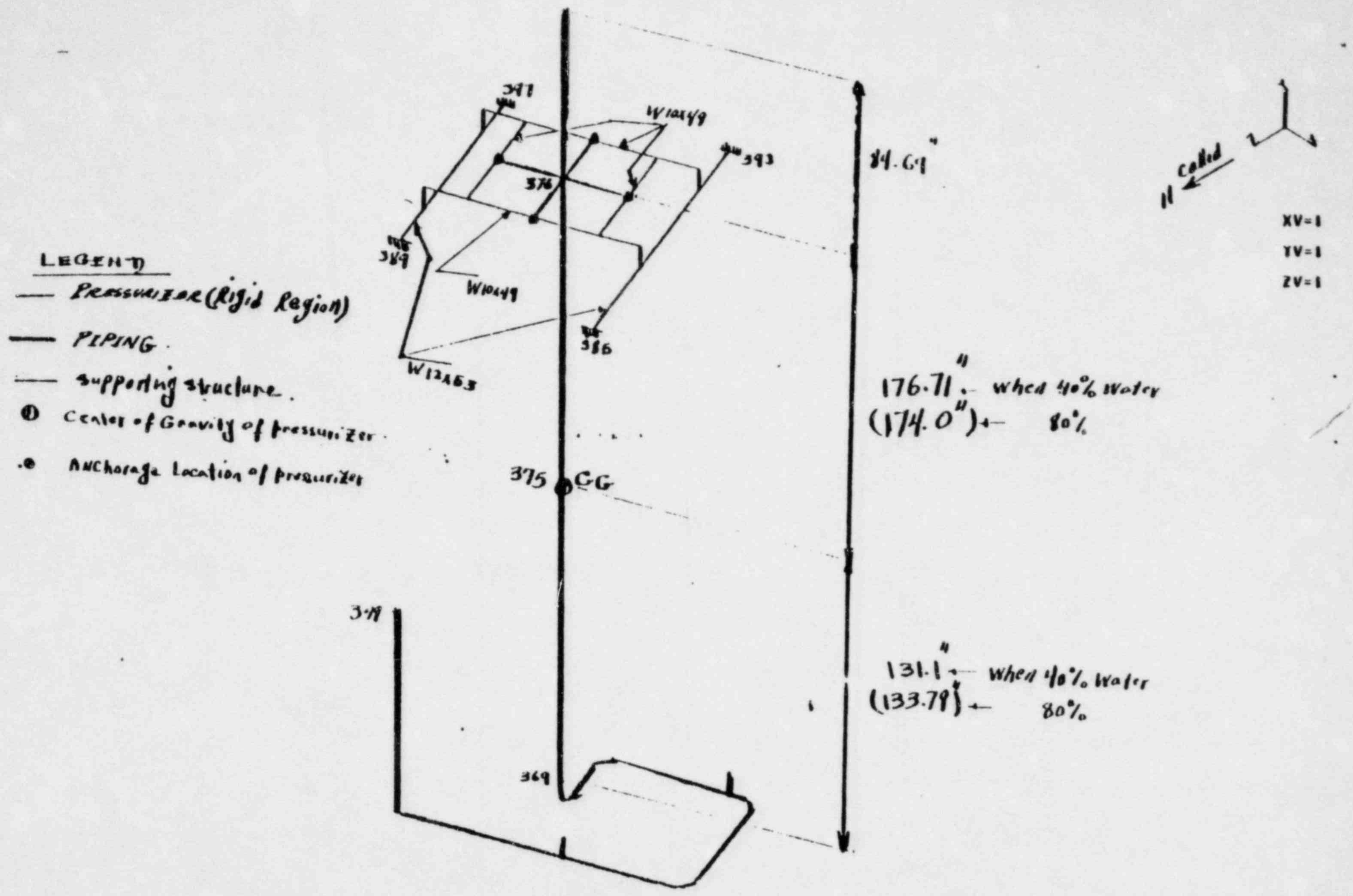

9699-FS-12G
PLAN - EL 1108'-0"
PLAN BOLTING DETAIL - EL 1108'-0"
"B-B"
"C-C"
"K-K"
SEC PLAN
SECTION "A-A"
PLAN - EL 1097'-8½"
DETAIL "I"
"M-M"
PLAN - EL 1083'-3½"
TYPICAL INSERT DETAILS
"F-F"
"E-E"
"G-G"
"H-H"
NOTES:
PRESSURIZER SUPP'TS & PLATFORMS
VAPOR CONTAINER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
9699-FS-12G

PRESSURIZER ANCHORAGE



PRESSURIZER ANCHORAGE EVALUATION FLOWCHART



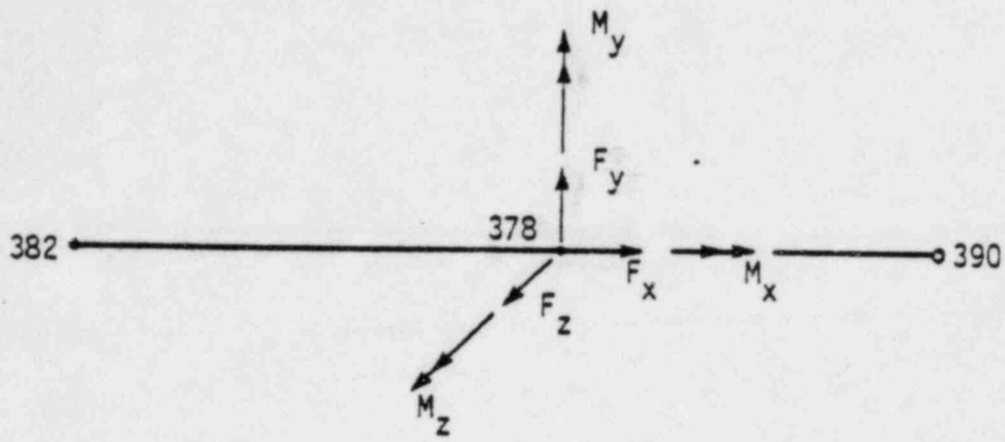


YANKEE ROWE PRESSURIZER ANCHORAGE EVALUTION

GEOMETRY ANSYS 1

MAXIMUM ANCHOR FORCES

From Load Case 3 (Page 61)



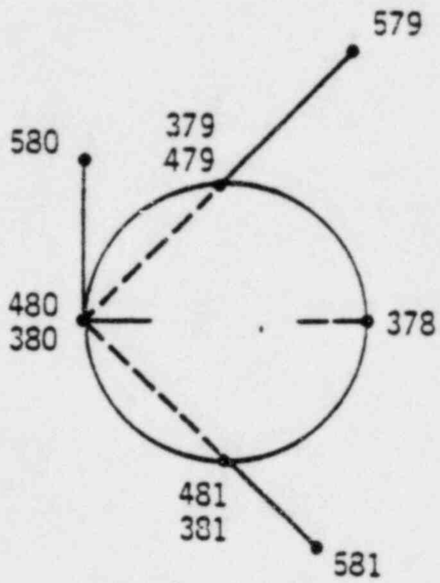
Maximum Anchor Forces

MODEL	F_x (k)	F_y (k)	F_z (k)	M_z (k-in)	M_y (k-in)	M_z (k-in)
40% Water	9.6	28.6	7.1	1.4	92.6	887.3
80% Water	11.6	35.5	8.5	1.9	112.5	1100.6

PRELIMINARY

MAXIMUM CONNECTION FORCES

From Load Case 3 (Page 61)



Maximum Connection Forces

MODEL	HORIZONTAL FORCES (K)			VERTICAL FORCES (K)		
	NODE 479-579	NODE 480-580	NODE 481-581	NODE 479-379	NODE 480-380	NODE 481-381
Model 1 (40%)	9.4	2.7	9.3	27.7	27.8	24.4
Model 2 (80%)	12.2	2.9	12.0	33.4	37.6	29.5

PRELIMINARY

EVALUATION OF STRESSES:

FROM DRAWING 9699-FM-25F

1-3/4" ΦBOLT $A = 2.4058\ IN^2$

FROM TABLE 5-1 (ATTACHMENT PAGE 68)

$F_T = 20$ KSI   ($F_Y = 32$ KSI

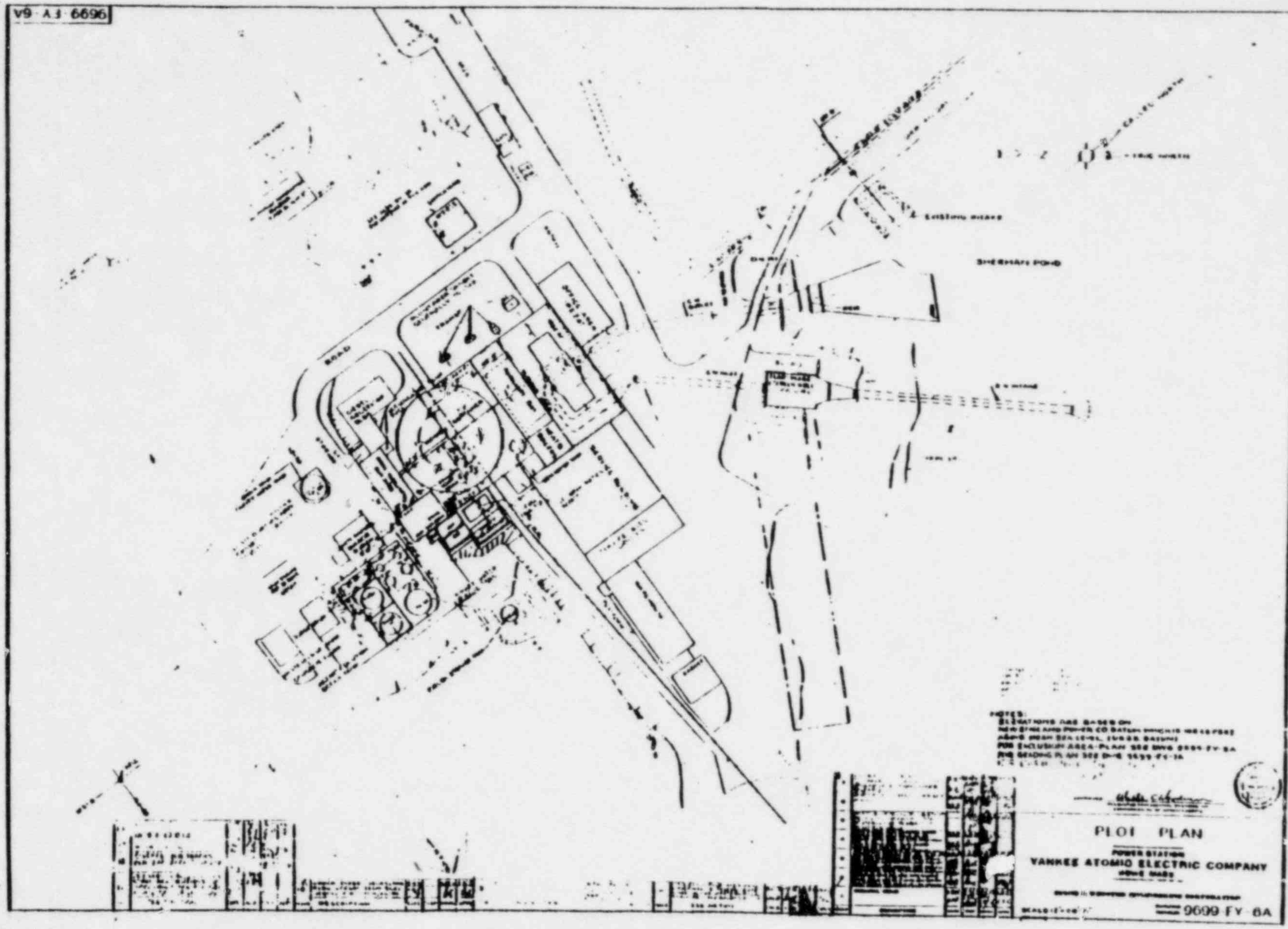
ALLOWABLE $F_U = 10$ KSI (S.S.)

$F_B = 25$ KSI (D.S.)

FEASILE STRESS $\sigma = \frac{27.8\ K}{2.4058} = 11.6$ KSI   SHEAR STRESS $\tau = \frac{9.4}{2.4058} = 3.9$

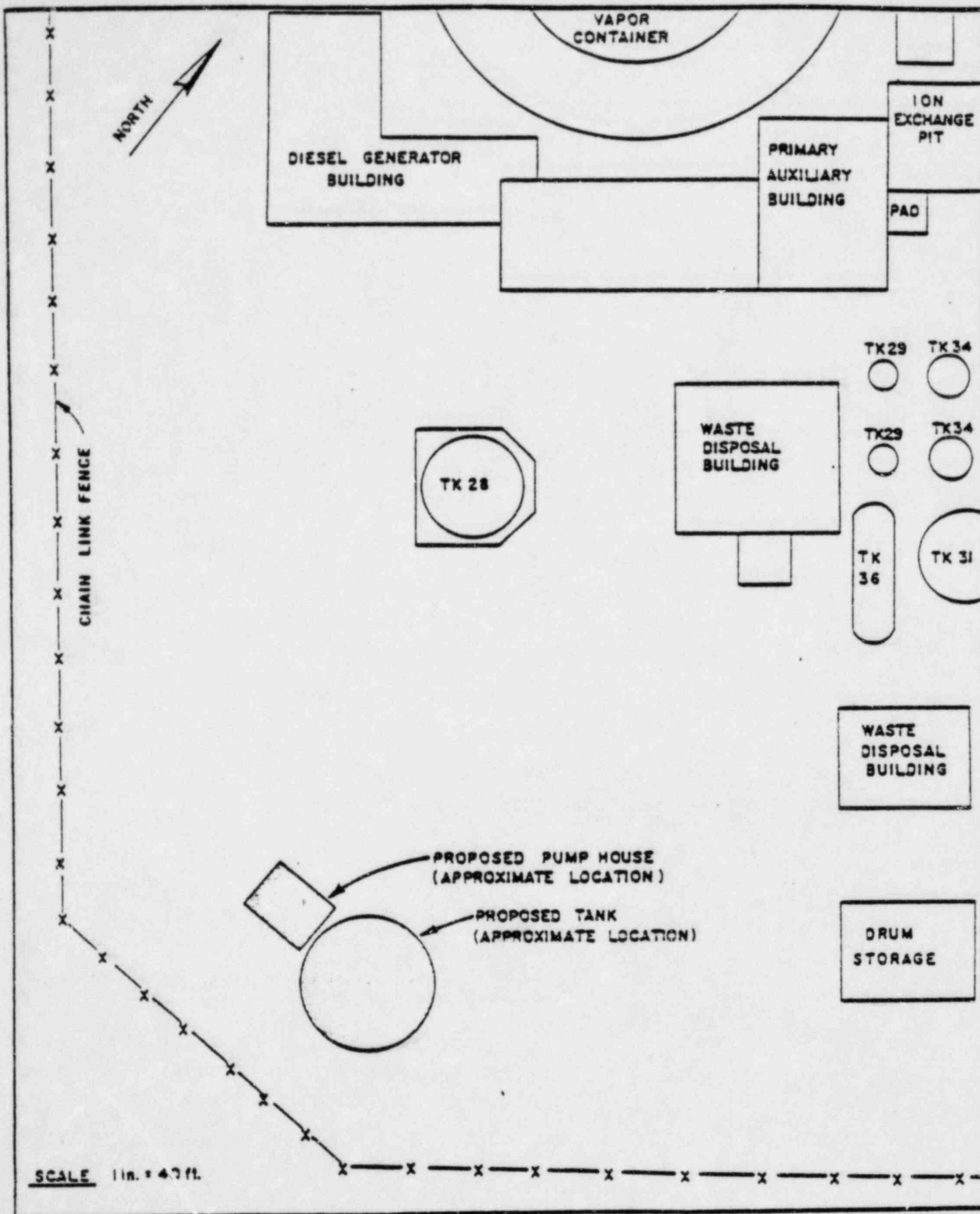
MODEL	TENSILE STRESS, σ (KSI)	SHEAR STRESS, τ (KSI)	ALLOWABLE STRESS		EVALUATION STATUS
			F_T (KSI)	F_V (KSI)	
MODEL 1 (40%)	11.6	3.9	20.	10.	OK
MODEL 2 (80%)	$\frac{37.6}{2.4058} = 15.6$	$\frac{12.2}{2.4058} = 5.1$	20.	10.	OK

PRELIMINARY


PLOT PLAN
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
9699-FY-6A
9699-FY-6A
SHERMAN POND
NOTES:

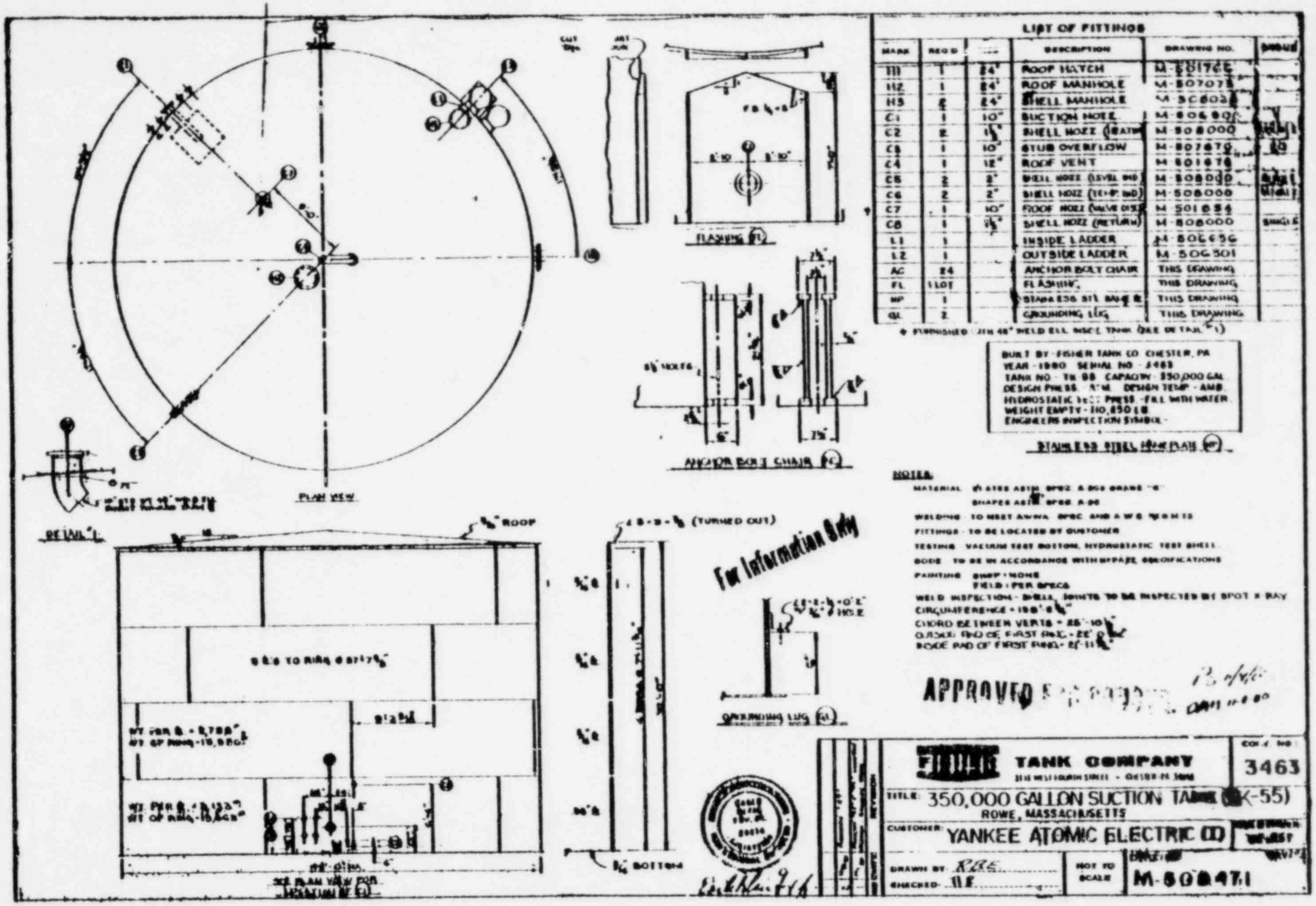
D-53

. D-54

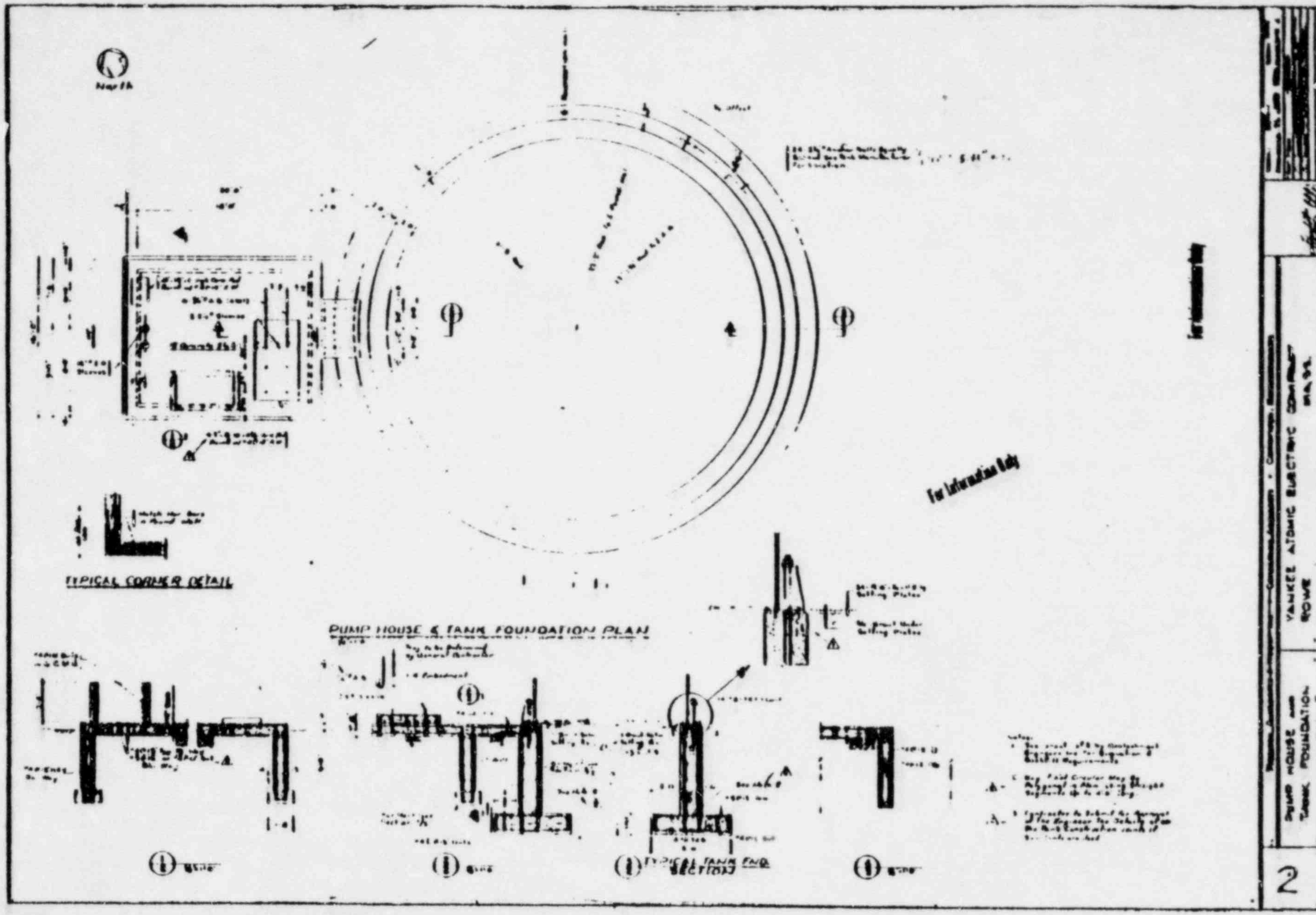


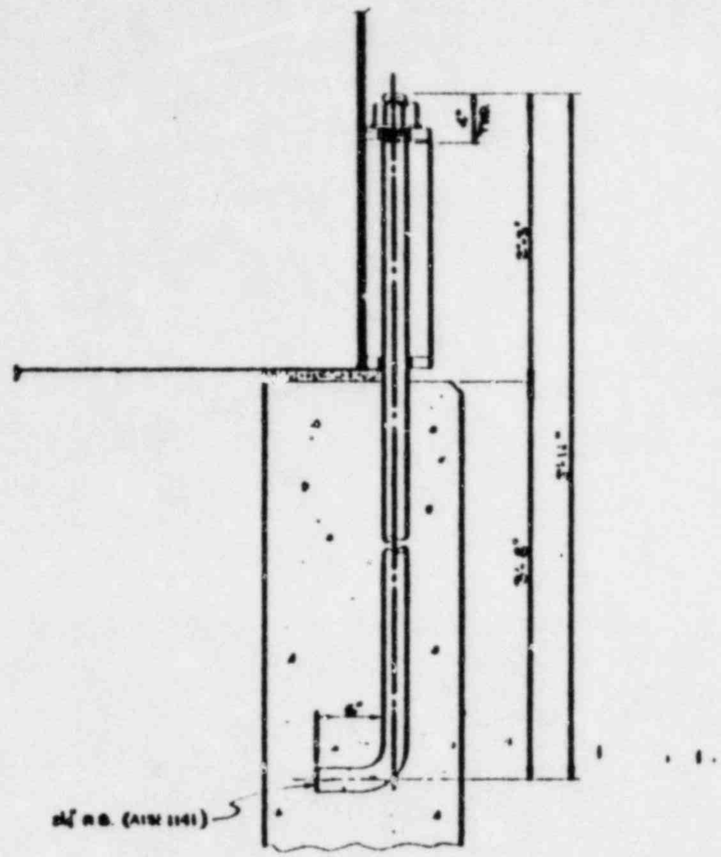
Yankee Atomic Electric Co. Westborough, Massachusetts	Fire Water Tank Rowe Atomic Plant	LOCATION PLAN
GEOTECHNICAL ENGINEERS INC WINCHESTER • MASSACHUSETTS	Project 79617	January 3, 1980 Fig. 2

D-55

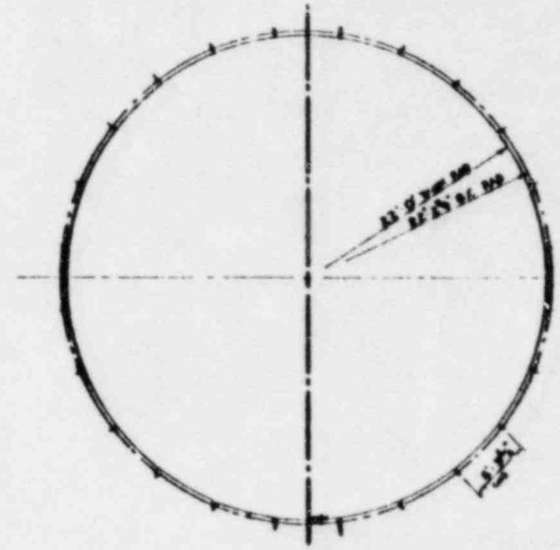

LIST OF FITTINGS
MARK | REQ'D | SIZE | DESCRIPTION | DRAWING NO.
H1 | 1 | 24" | ROOF HATCH | M-501766
H2 | 1 | 24" | ROOF MANHOLE | M-507073
H3 | 2 | 24" | SHELL MANHOLE | M-506032
C1 | 1 | 10" | SUCTION NOZZ. | M-506808
C2 | 2 | 1½" | SHELL NOZZ (DRAIN) | M-508000
C3 | 1 | 10" | STUB OVERFLOW | M-507670
C4 | 1 | 12" | ROOF VENT | M-501676
C5 | 2 | 2" | SHELL NOZZ (LEVEL IND) | M-508000
C6 | 2 | 2" | SHELL NOZZ (TEMP IND) | M-508000
C7 | 1 | 10" | ROOF NOZZ (VALVE DISC) | M-501654
C8 | 1 | 1½" | SHELL NOZZ (RETURN) | M-508000
L1 | 1 | | INSIDE LADDER | M-506456
L2 | 1 | | OUTSIDE LADDER | M-506501
AC | 24 | | ANCHOR BOLT CHAIR | THIS DRAWING
FL | 1 LOT | | FLASHING | THIS DRAWING
NP | 1 | | STAINLESS STL NAME PL. | THIS DRAWING
GL | 2 | | GROUNDING LUG | THIS DRAWING
BUILT BY - FISHER TANK CO. CHESTER, PA
YEAR - 1980 SERIAL NO. - 3463
TANK NO. - TK 55 CAPACITY - 350,000 GAL.
DESIGN PRESS. - ATM. DESIGN TEMP. - AMB.
HYDROSTATIC TEST PRESS. - FILL WITH WATER
WEIGHT EMPTY - 110,850 LB.
ENGINEERS INSPECTION SYMBOL -
STAINLESS STEEL NAME PLATE
NOTES:
WELDING - TO MEET AWWA SPEC. AND A.W.S.
FITTINGS - TO BE LOCATED BY CUSTOMER
TESTING - VACUUM TEST BOTTOM, HYDROSTATIC TEST SHELL
PAINTING SHOP: NONE
FIELD: PER SPECS
WELD INSPECTION - SHELL JOINTS TO BE INSPECTED BY SPOT X-RAY
PLAN VIEW
DETAIL "1"
FLASHING
ANCHOR BOLT CHAIR
GROUNDING LUG
ROOF
(TURNED OUT)
BOTTOM
For Information Only
APPROVED
FISHER TANK COMPANY
TITLE: 350,000 GALLON SUCTION TANK (TK-55)
ROWE, MASSACHUSETTS
CUSTOMER: YANKEE ATOMIC ELECTRIC CO
DRAWN BY: RBE
CHECKED: JLF
NOT TO SCALE
M-508471
3463

D-5


North
For Information Only
TYPICAL CORNER DETAIL
PUMP HOUSE & TANK FOUNDATION PLAN
TYPICAL TANK FND. SECTION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE MASS.
PUMP HOUSE AND TANK FOUNDATION
2



ANCHOR BOLT DETAIL



ANCHOR BOLT LAYOUT

For Information Only

NOTE:
ANCHOR BOLTS TO BE TIGHTENED & SECURED BY DISTORTING THREADS

GPS 10/9/
CDN 11-9

APPROVED FOR CONSTR.

TITLE:	ANCHOR BOLTS
	FISHER TANK COMPANY, CHESTER PA & LEXINGTON S.C.
DRAWN BY	E.K.E.
CHECKED	S.I.W.
NOT TO SCALE	
DWG. NO.	M-509544
REV.	

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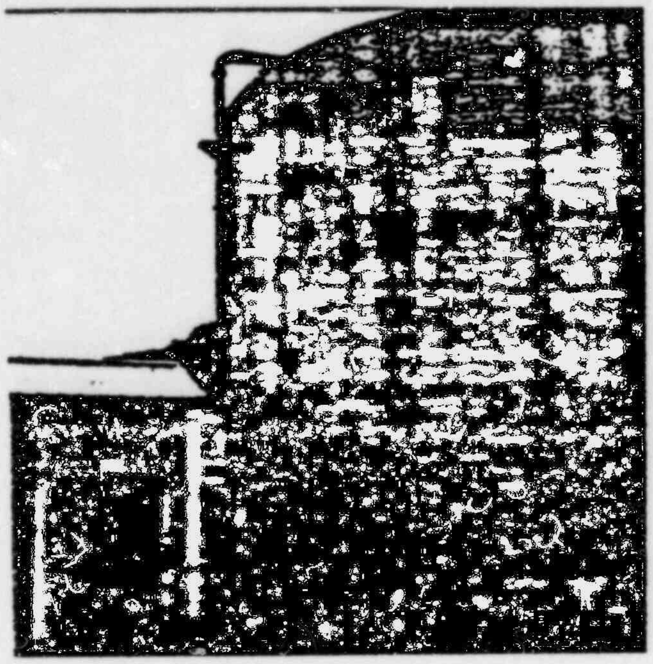
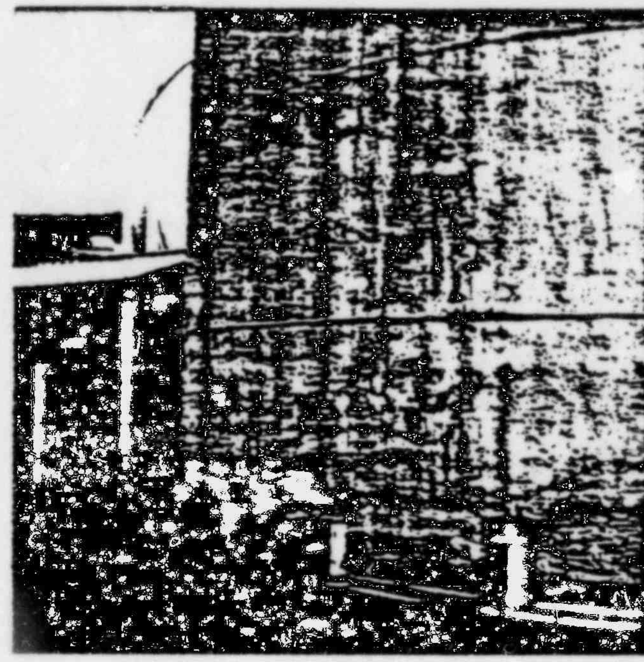
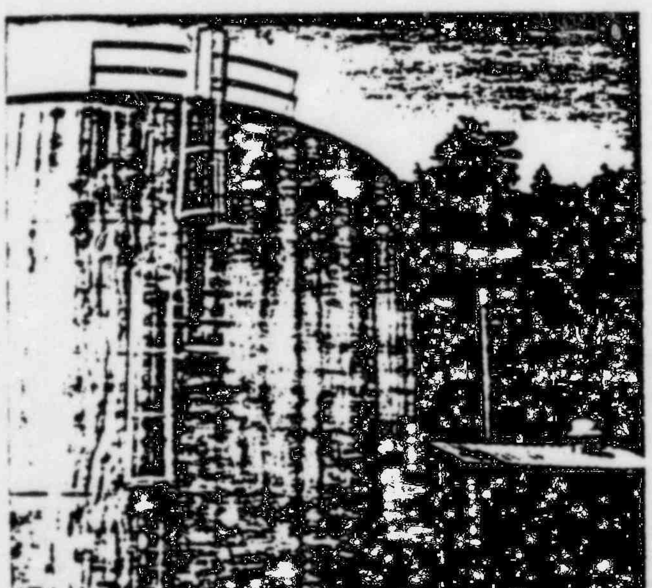


(a) Looking East



(b) Looking Northeast



(c) Looking West

(d) Looking East

FIRE TANK ELEVATION VIEW AND PUMP HOUSE

CERTIFICATION

Commission No. 80134

Structural Design of Foundations for
Water Tank and Pump House
Yankee Rowe, Rowe, Massachusetts
Yankee Atomic Electric Company

Client:

Geotechnical Engineers Inc.
Purchase order No. 1650
Client Project No. 79617

The earthquake design specifications are summarized below.

1. Lateral forces and overturning moments due to earthquake horizontal accelerations are based on Appendix E, "Seismic Desig.. of Storage Tanks," of API Standard 650, "Welded Steel Storage Tanks for Oil Storage," Sixth Edition, Revision 3, October 15, 1979. The following coefficients appearing in formulas in Appendix E are used:

    (a) Essential facilities factor I = 1.5
    (b) Zone coefficient Z = 1.0
    (c) Soil profile coefficient S = 1.0

2. Vertical earthquake forces are computed for a peak ground vertical acceleration of 0.08 g, amplified in accordance with Fig. 2 of the Nuclear Regulatory Guide 1.60. The curve for 2% damping is used.

3. Horizontal earthquake forces in one direction and vertical earthquake forces are applied simultaneously. The horizontal force may come from any direction.

4. The provisions of the Massachusetts State Building Code are used, except as modified by items 1, 2, and 3 above. The zone coefficient is taken as 1.0, rather than the one-third incorporated in the Massachusetts Code.

F. Strength

All concrete shall achieve a minimum 28-day compressive strength of 4,000 psi, unless noted otherwise on the structural drawings.

G. Density

All concrete shall be normal weight concrete, unless noted otherwise on the structural drawings.

H. Reinforcement

1. Reinforcement shall be bars of sizes called for on the Drawings.

2. Reinforcing bars shall be deformed and comply with the requirements of the "Standard Specifications for Deformed and Plain Billet Steel Bars for Concrete Reinforcement": ASTM A615, Grade 60, except that stirrups and #3 bars may be Grade 40.

I. Other Materials

1. Polyethylene film vapor barrier .006", Visqueen, "Cerpak" by Monsanto, "Zendel" by Union Carbide, Premold Membrane Vapor Seal by W. R. Meadows, or equal.

2. Kraft paper: Waterproof, reinforced, meeting ASTM C171.

3. Admixtures: air-entraining – ASTM C260; water reducing – ASTM C494. Use only approved admixtures by the same manufacturer, recommended by him to produce the specified air content and w/c ratio. Do not use calcium chloride or any other admixture.

YANKEE ROWE FIRE TANK
DESIGN CRITERIA COMPARISON

	GEOTECH. ENGIN.	CYGNA
SEISMIC MODEL	API	HOUSNER NRC SPECT.
SEISMIC COEFF. (HORIZ.)	0.24 g	0.19 g
SEISMIC COEFF. (VERT.)	REGUL. GUIDE 1.60, $\xi = 2\%$	2/3 x HORIZ.
PERIOD (HYD. MASS)	3.8 sec.	3.9 sec.
COEFF. OF HYD. MASS	0.32 M @ H = 21.4'	0.26 M @ H = 25.1
FORCE ON ANCHOR (#18)	43.3^k	(T) 30.0^k (V) 18.8^k
f_t	13.3 ksi	9.2 ksi
f_v	3.9 ksi	4.7 ksi
f_{BOND}	134. psi	93. psi

PRELIMINARY

EQUIPMENT ANCHORAGE EVALUATION

CONCLUSIONS

- THE REACTOR RING SUPPORT CAN WITHSTAND SEISMIC LOADS AS SPECIFIED BY THE NRC SPECTRUM.
- STEAM GENERATOR ANCHORS - EVALUATION NOT COMPLETED.
- THE ANCHORAGE OF THE PRESSURIZER CAN WITHSTAND SEISMIC LOADS AS SPECIFIED BY THE NRC SPECTRA.
- FIRE TANK - EVALUATION IS IN PROGRESS.

CYGNA ENERGY SERVICES

Rev. 0

NRC
PIPE SUPPORT PRESENTATION
YANKEE ROWE POWER STATION

August 3 & 4, 1982

PRELIMINARY

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INDEX PIPE SUPPORT PRESENTATION

PIPE SUPPORTS

INTRODUCTION

- Scope
- Criteria
- Approaches
- Conceptual Designs
- Schedules

PIPE SUPPORTS

SCOPE

- Inside Vapor Container
- Evaluation of Existing Supports
- Design of New Supports

PIPE SUPPORTS
SUPPORT SUMMARY

SYSTEM	EXISTING SUPPORTS	SUPPORTS TO BE ADDED			REMARKS
		SPRINGS	SNUBBERS	RIGIDS	
1	2	--	1	1	
2	2	--	--	2	
3	2	--	1	1	
4	1	--	--	2	
021	2	--	--	3	
022	2	--	--	4	
023	2	--	--	4	
024	2	--	--	3	
41A	7	--	--	--	4 EXISTING REQUIRE REPLACEMENT
41B	9	--	--	--	
41C	5	1	--	--	
101	10	--	--	--	
102	7	--	--	--	
103	7	--	--	--	
104	7	--	--	--	
121	1	--	1	1	
122	1	--	1	1	
201 207	20	*	*	*	

* FINALIZATION OF DESIGN IN PROGRESS

PIPE SUPPORTS

CRITERIA

- Frequency > 33 Hz in support direction
- Stress
- Anchor bolt loads
- Buckling

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PIPE SUPPORTS

Allowables

Stress		SERVICE LEVEL: NORMAL (Value)	SERVICE LEVEL: EMERGENCY/FAULTED (Value)
Tension		$0.6\ F_y$	$0.9\ F_y$
Shear		$0.4\ F_y$	$0.6\ F_y$
Web Crippling		$0.75\ F_y$	$0.9\ F_y$
Compression		F_a	Smaller of $1.5\ F_a$ or $2/3\ F_{cr}$
Bending		$0.6\ F_y$	$0.9\ F_y$
Bearing		$0.9\ F_y$	N/A
Bolts 307	Tension	Allowable Tension per AISC	1.5 X (Allowable Tension per AISC)
	Shear	Allowable Shear per AISC	1.5 X (Allowable Shear per AISC)
Anchor Bolt		From "Hilti" catalog With a Safety Factor of 4	
Welds (Fillet)	Shear	$0.3\ F_v$ (Weld Metal); $.4\ F_y$ (Basemetal)	$.45\ F_v$ (Weld Metal); $.6\ F_y$ (Basemetal)
(Full or Partial Penetration)	Tension	$0.6\ F_y$ (Base Metal)	$0.9\ F_y$ (Base Metal)
Combined Stress		Per AISC	Per AISC
Catalog Items		Catalog Values	1.5 X (Catalog Values)

PIPE SUPPORTS
LOADING CRITERIA

NORMAL

- DEAD LOAD (D)
- THERMAL EXPANSION (TH)
- THERMAL ANCHOR MOVEMENTS (TAM)
- FRICTION (FL)

EMERGENCY/FAULTED

- SSE (YCS OR NRC)
- SAM (YCS OR NRC)

LOAD COMBINATIONS

NORMAL	(D) OR (D + TH + TAM) + FL
EMERGENCY/FAULTED	(D) OR (D + TH + TAM) + SSE_{YCS} + SAM_{YCS}
EMERGENCY/FAULTED	(D) OR (D + TH + TAM) + SSE_{NRC} + SAM_{NRC}

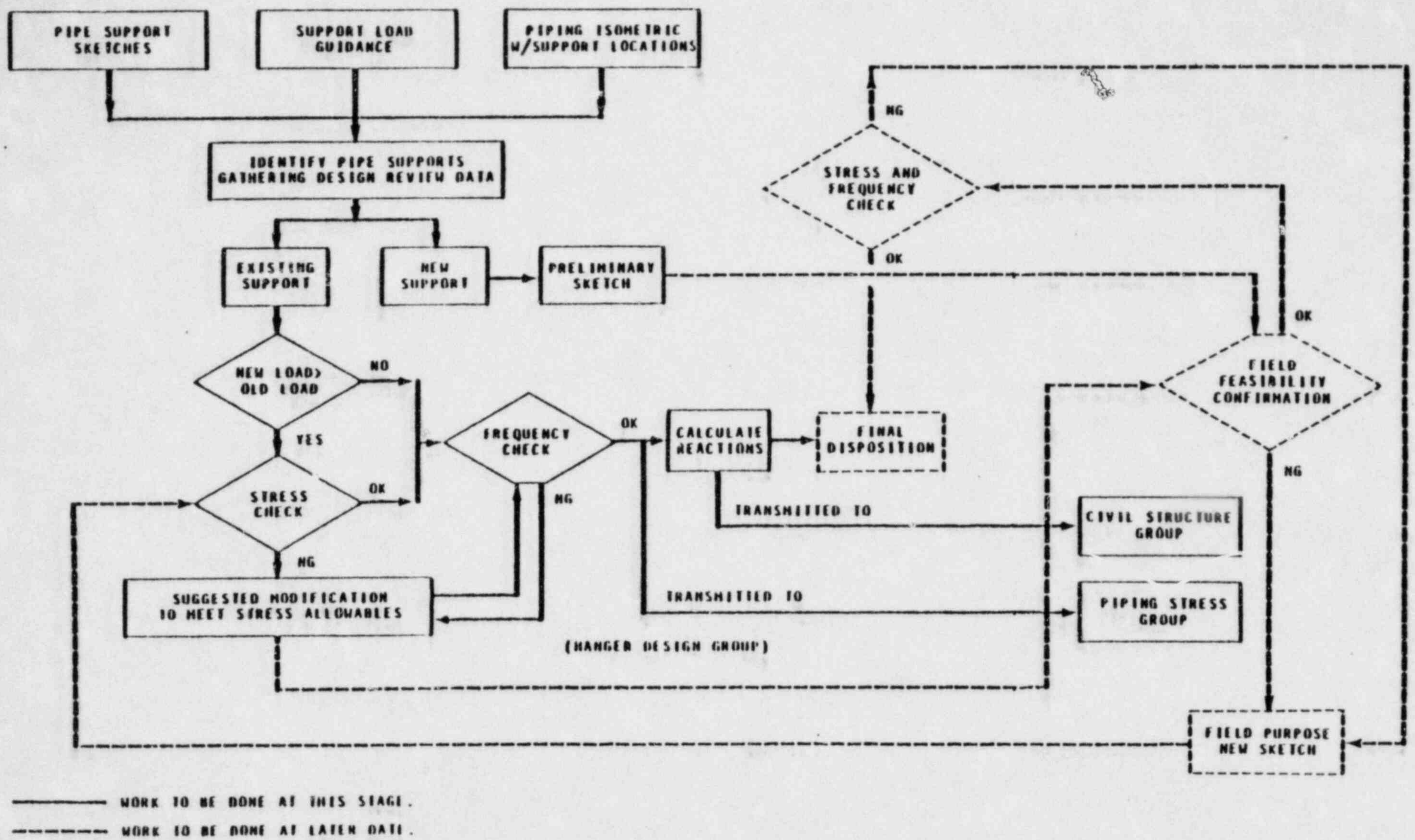
PIPE SUPPORTS

ADDITIONAL CONSIDERATIONS

- Tributary Mass
- Gaps for Frames
- Spring Hanger Hot Load/Cold Load
- Anchors
- Pipe Travel for Springs

PIPE SUPPORTS

FLOWCHART



PIPE SUPPORT DESIGN REVIEW FLOWCHART

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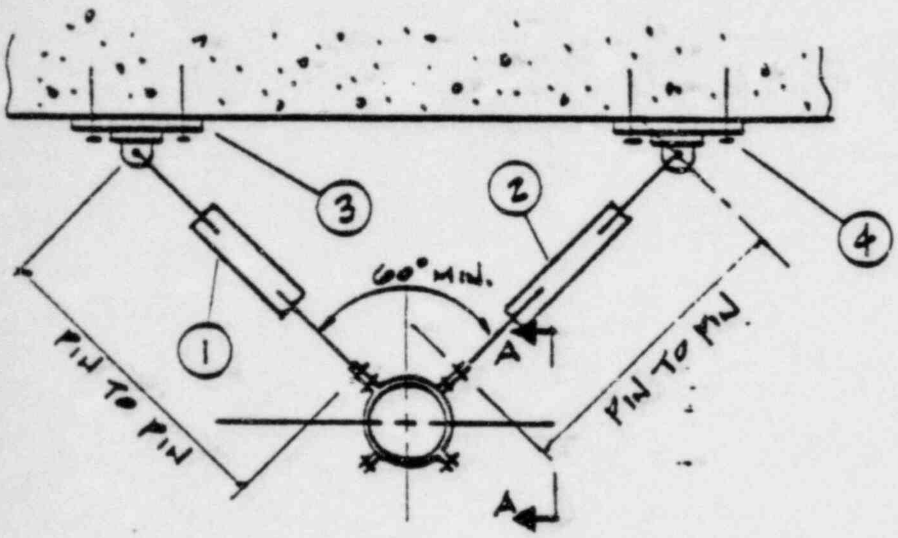
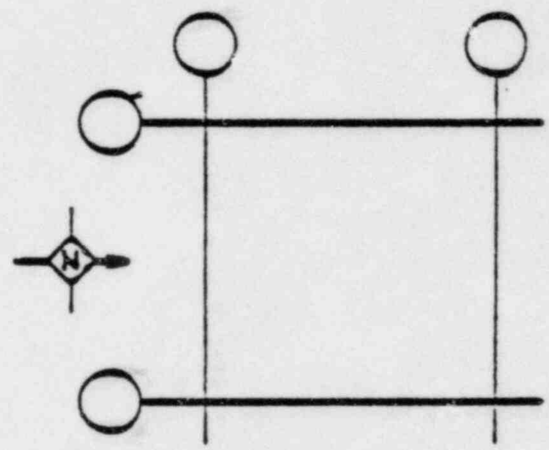
PIPE SUPPORTS

CONCEPTUAL DESIGNS





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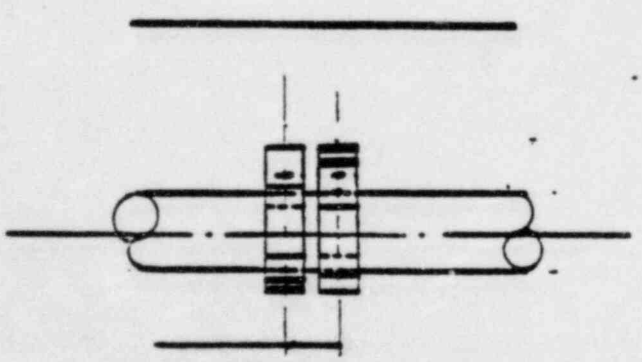

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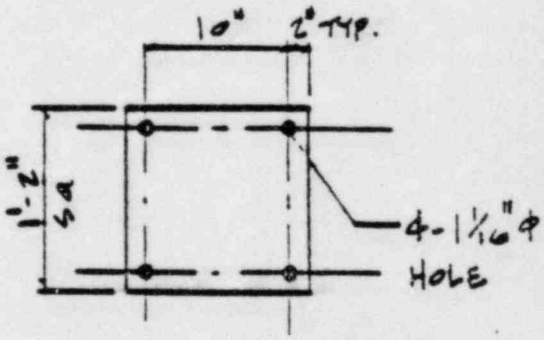

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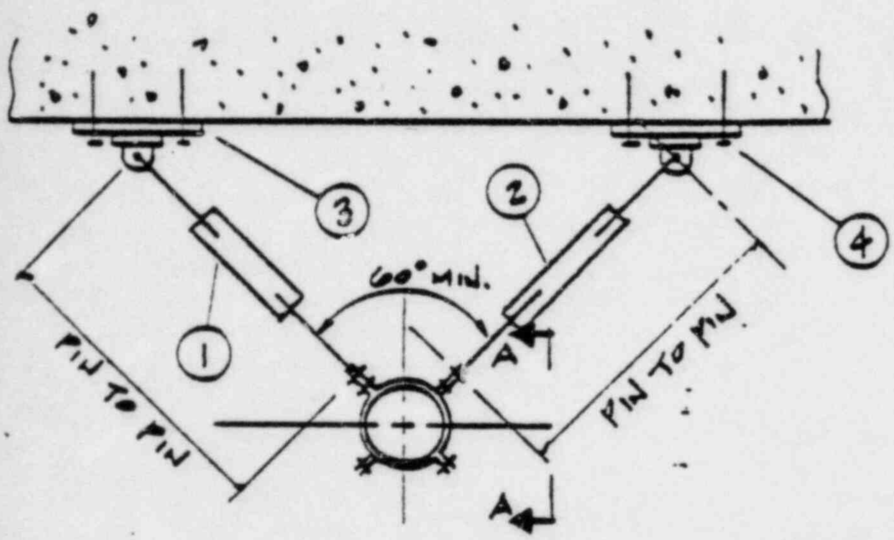

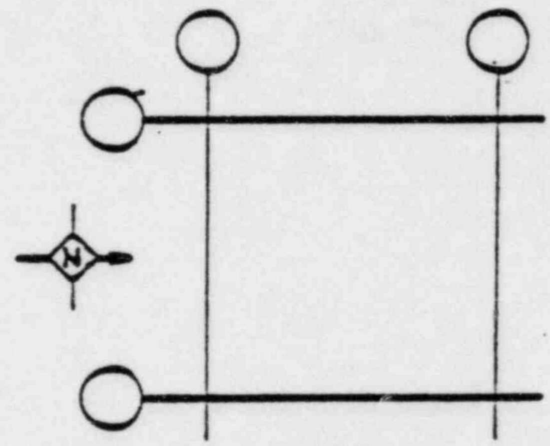

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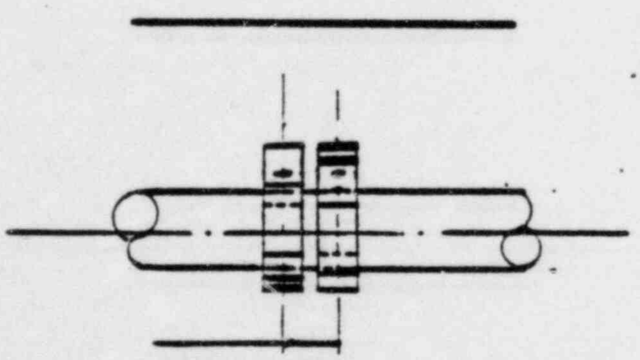

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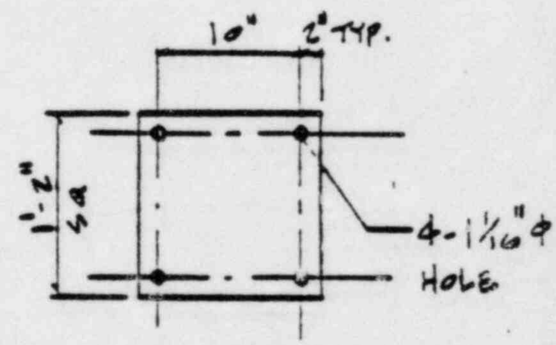

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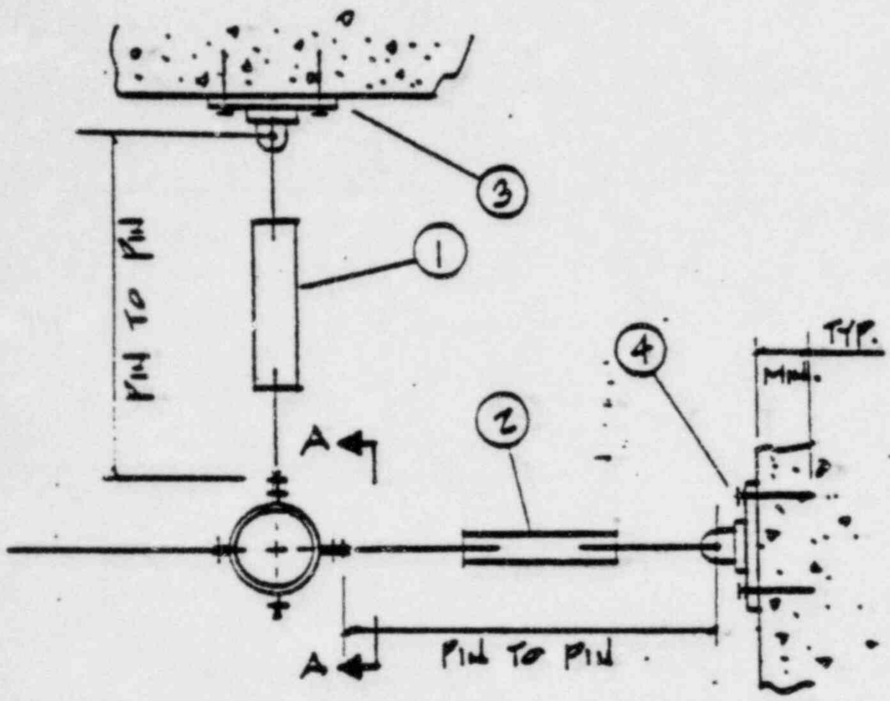

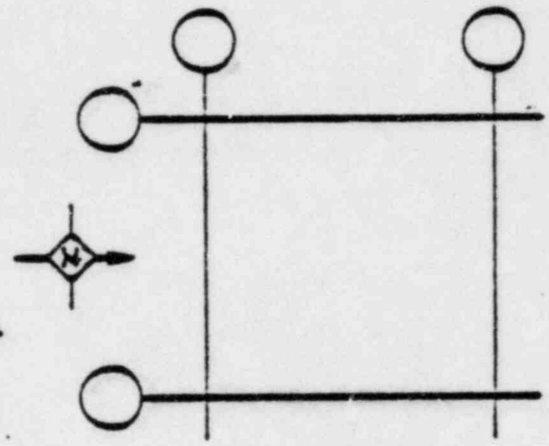

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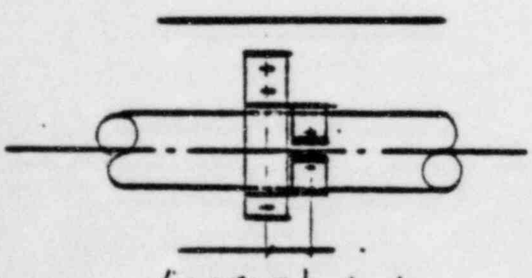

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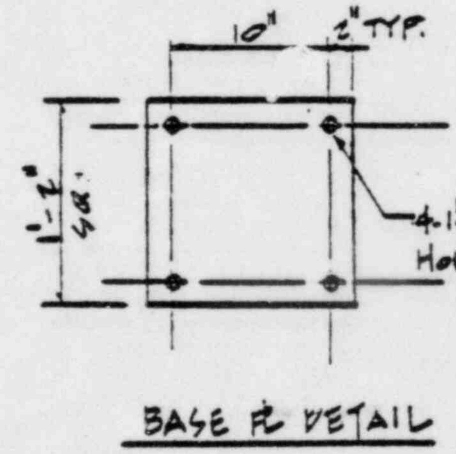

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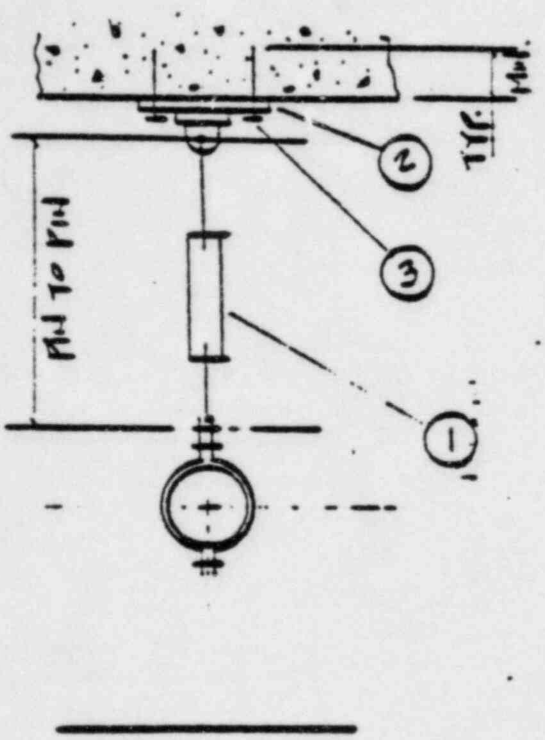

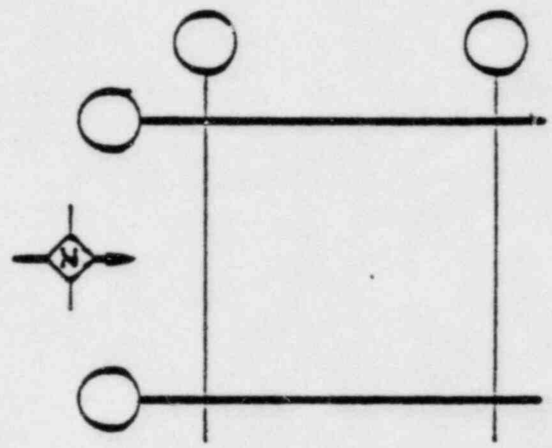

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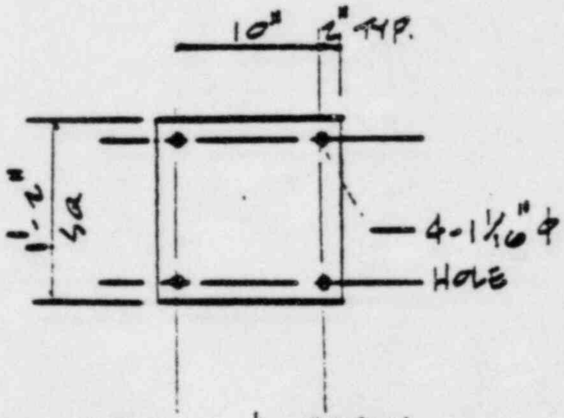

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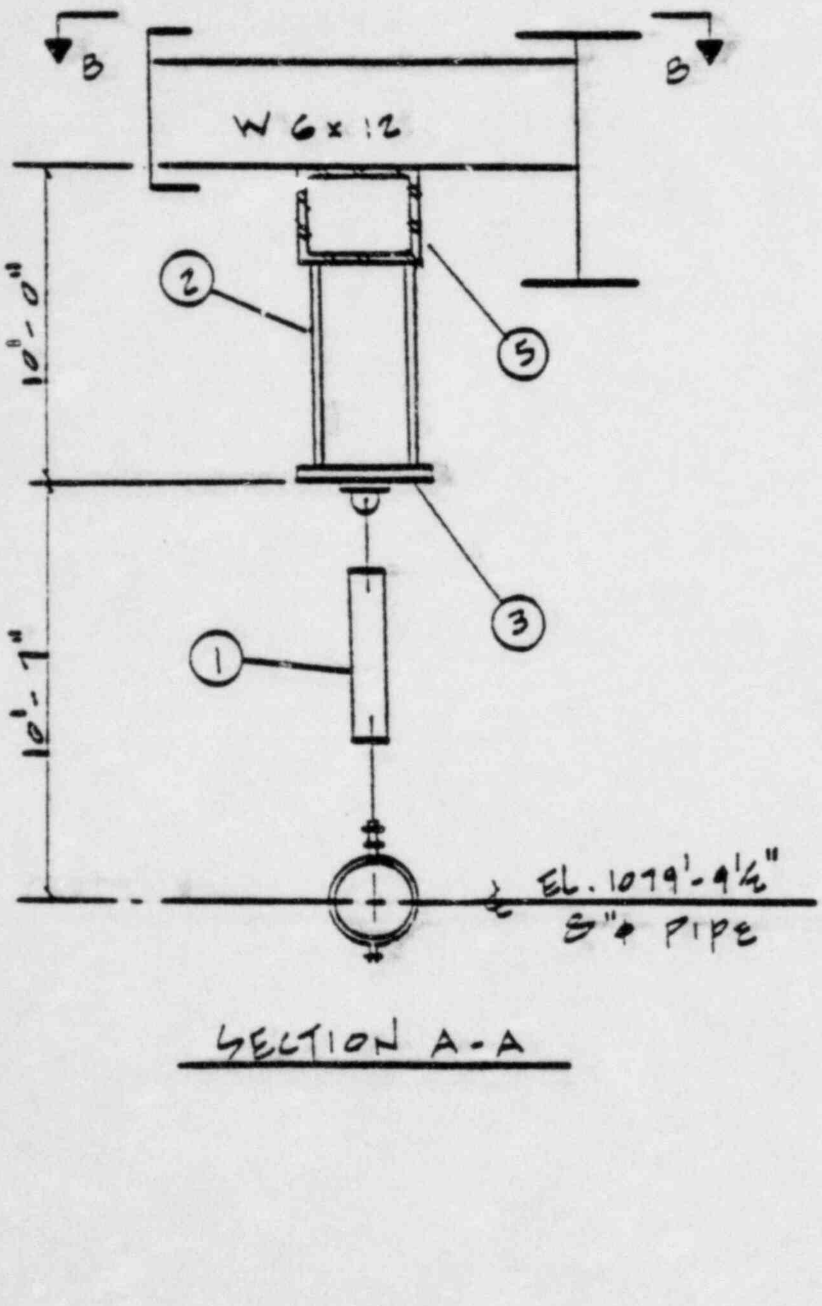

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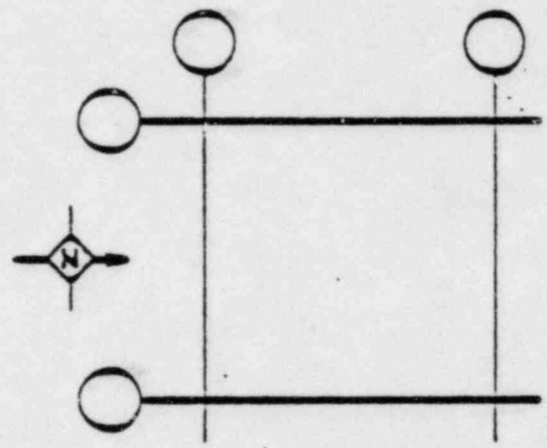

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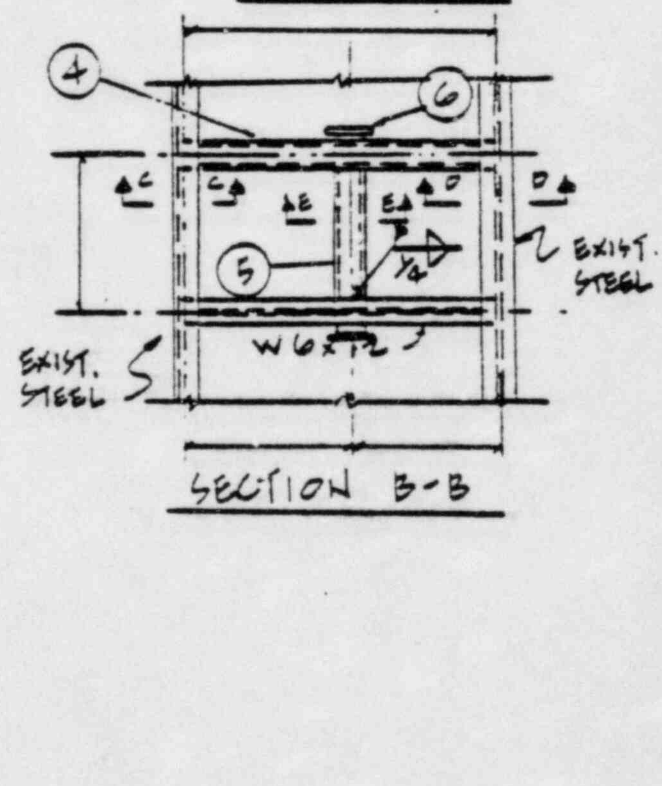

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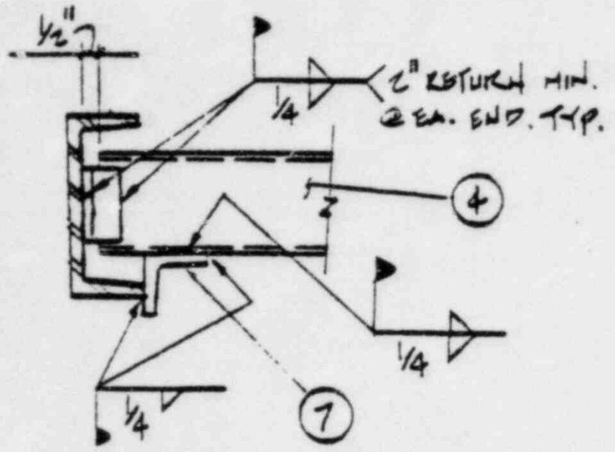

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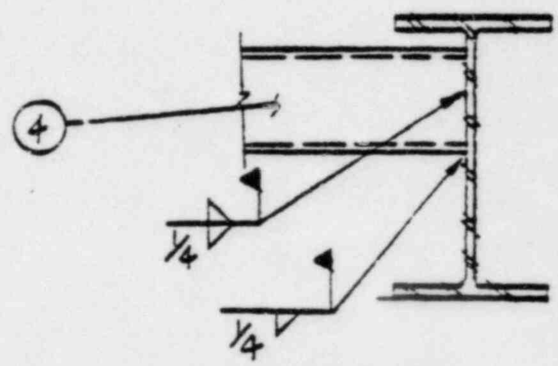

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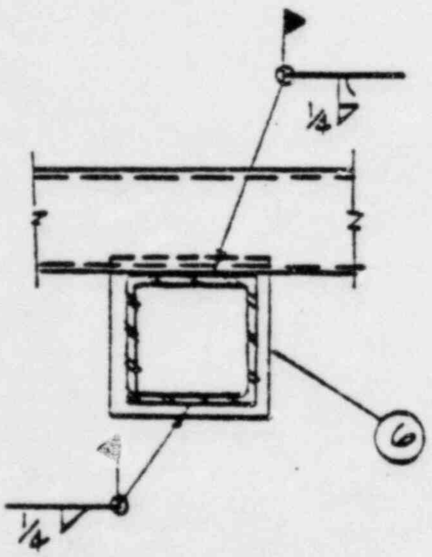

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