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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
ASYMMETRIC LOCA LOADS
NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2
DOCKET NO. 50-336

1.0 INTRODUCTION

On May 7, 1975, the Nuclear Regulatory Commission (NRC) was informed that asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered in the original design of the reactor vessel support for North Anna Units 1 and 2. It had been identified that in the event of a postulated, instantaneous, double-ended offset shear pipe break at the vessel nozzle, asymmetric loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the development of more detailed analytical models, it became apparent that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports and other components, thereby possibly affecting their integrity. Although this potential safety concern was first identified during the review of the North Anna facilities, it was determined to have generic implications for all pressurized water reactors.

In October of 1975, the NRC staff notified each operating Pressurized Water Reactor (PWR) licensee of a potential safety problem concerning the design of their reactor pressure vessel support system. From this survey it was discovered that these asymmetric loads had not been considered in the design of any PWR primary system. In June 1976, the NRC requested all operating PWR licensees to evaluate the adequacy of reactor system components and supports at their facilities, with respect to these newly identified loads. Licensee and vendor responses to this request were proposals to augment inservice inspection and/or probability studies that supported no analyses due to the low probability of the pipe breaks at a particular location. Although the NRC recognized some merit in these proposals, they determined that the more fundamental questions still remained unanswered. Therefore, licensees of PWR plants were notified by letter dated January 20, 1978 that the evaluation of their primary systems for asymmetric LOCA loads would be required.

Although the NRC Staff's original emphasis and concerns were focused primarily on the integrity of the reactor vessel support system with respect to postulated breaks inside the reactor cavity (i.e., at a nozzle), it became apparent that significant asymmetric forces could also be generated by postulated pipe breaks outside the cavity and that the scope of the problem was not limited to the vessel support system, itself. The staff, after reviewing this problem, determined that a reevaluation of the primary system integrity of all PWR plants to withstand these loads was necessary. Therefore in January of 1978, the NRC staff requested each PWR licensee to submit additional information in accordance with the expanded scope of the problem. Those letters outlined the present scope of the problem specifying a minimum number of pipe break locations to be addressed and the reactor system components to be evaluated.

The asymmetric loading on reactor vessel supports resulting from a postualted coolant pipe rupture at a specific location that was identified in Virginia Electric Power Company's (VEPCO) May 7, 1975 letter to NRC was later determined by NRC to have generic implications for all PWRs. This was formerly identified in Task Action Plan A-2 (Unresolved Safety Issue (USI), "Asymmetric Blowdown Loads on Reactor Primary Coolant System," as published in NUREG-0371, "Task Action Plans for Generic Activities (Category A)," USNRC, November 1978. Since the identification of the asymmetric load problem in May 1975, EG&G Idaho, Inc. has performed a number of independent audit analyses to verify licensee submittals on this problem. A total of six analyses have been completed (one linear elastic and one nonlinear-inelastic analysis of a reactor coolant loop (RCL) for each of the three major reactor vendors). Based on these analyses and additional NRC staff investigations, criteria and guidance for conducting an evaluation of asymmetric loss of coolant accident (LOCA) loads were developed. USI A-2 was resolved in January 1981 with the publication of NUREG-0609 (Reference 3). This document provided an acceptable basis for performing and reviewing plant analyses for asymmetric LOCA loads and affected all operating and future PWRs.

During the course of the work on USI A-2, it was demonstrated that there were only a very limited number of break locations which could give rise to significant loads. Subsequently, after the development of substantial new technical work it was demonstrated that the new techniques for the analysis of piping failures assured adequate protection against failures in primary system piping in Pressurized Water Reactors. This was reflected in a revision to GDC-4 published in the Federal Register in final form on April 11, 1986, and in a further revision to GDC-4 published in the Federal Register on July 23, 1986. In addition, it has also been satisfactorily demonstrated in the course of the A-2 effort that there is a very low likelihood of simultaneous pipe loading with both LOCA and SSE loads.

For Combustion Engineering plants of the pre-CESSAR vintage without the SSE-LOCA load combination, the loads on primary system piping would not result in pipe breaks which could lead to significant loads on the core structure. Accordingly, for these facilities the staff had concluded that the potential for asymmetric loading on the core structure resulting from primary system piping LOCA, need not be considered in the design of the core structure.

In June of 1980, Combustion Engineering (CE), on behalf of the Northeast Utilities, a member of the CE Owners Group, submitted a final asymmetric LOCA loads evaluation report (Reference 1), applicable to the Millstone Unit 2 nuclear power plant. This material submitted in response to the January 1978 letter from NRC, was reviewed by the NRC staff and its consultants. Upon review of the submital, it was determined that addi-tional information was required to satisfy the established guidelines and acceptance criteria. On February 23, 1981, the NRC staff notified CE of the additional requests, and the response (Reference 2) was submitted in August of 1981.

The Millstone Unit 2 final submittal and supplement represent the limiting cases for the asymmetric LOCA loads evaluation and have been reviewed in conjunction with the criteria outlined in NUREG-0609 (Reference 3). Subsequent sections of this safety evaluation report summarize the evaluations performed by the licensee for subcooled blowdown loads, cavity pressurization, and structural response. Following this is the staff's evaluation of the licensee's analyses. The staff's evaluation includes the assessment of the licensee's compliance with acceptance criteria.

2.0 DISCUSSION

The licensee's analysis procedure including analytical models, computer methods and analytical results are discussed in the following paragraphs.

The analytical methodology primarily consists of development of (a) Thermal hydraulic loads for the reactor coolant system (RCS) structural analysis, (b) Calculation of the steam generator and reactor cavity pressures, and (c) Calculation of the loads and stresses on the various components and supports of the RCS which include the vessel and steam generator supports, vessel internals, fuel assemblies, control element drive mechanisms (CEDM) and emergency core cooling system (ECCS) piping.

2.1 Thermal-Hydraulic Loads Analysis

The CEFLASH-4B computer code (References 4 and 5) was used to predict the transient hydraulic response of the reactor primary coolant system to the most critical postulated pipe breaks. These were the guillotine pipe breaks with a break opening area of 0.10A[a] (or 135 in.^{2}) in 0.20 sec at the reactor vessel outlet nozzle and the 2.0A (or 1414 in.^{2}) break in 0.23 sec at the reactor vessel inlet nozzle. The CEFLASH-4B analysis is based on the volume-flow path concept. This involves simultaneously solving the conservation equations of mass, momentum, and energy, and the fluid pressures, densities and enthalpies. All of these parameters are assumed to exist in a state of thermo-dynamic equilibrium. The CEFLASH-4B code also assumes the

a. A break opening area will be referred to as a multiple of the cross-sectional flow area (A) of the pipe at the specified location. A full guillotine off-set break is defined as 2A.

fluid boundaries to be rigid and at rest, thereby excluding fluid-structure interaction effects of the core barrel-reactor vessel relative motion on the downcomer pressure transients in the subcooled loads hydraulic analysis. The results of the thermal-hydraulic analysis provide the time history forcing functions applied to the reactor vessel and internals in the reactor coolant system (RCS) structural analysis.

2.2 Cavity Pressurization Analysis

The subcompartment pressurization analyses of the reactor cavity and steam generator subcompartment were performed in a two-step procedure. First, the blowdown mass flow rates and energy release rates were calculated for the RCS. Then cavity pressures were computed using these release rates to determined component support and compartment wall loading transients.

Mass and energy release rates were calculated using the modified CEFLASH-4 computer code based on four design basis postulated pipe ruptures; which were determined to be the most limiting breaks.

1. 2.0A (or 1414 in.^{2}) break in 0.023 sec at the reactor vessel inlet nozzle.

2. 0.10A (or 135 in.^{2}) break in 0.020 sec at the reactor vessel outlet nozzle.

3. 0.70A (or 1000 $in.^2$) break in 0.024 sec at the steam generator inlet nozzle.

4. 2.0A (or 1414 $in.^2$) break in 0.020 sec at the steam generator outlet nozzle.

The CEFLASH-4 code was modified to incorporate a critical flow correlation subroutine that maximizes the blowdown rates. This was achieved by utilizing a combined Henry/Fauske and Moody critical flow correlation with a flow multiplier of 0.7 throughout the blowdown transient. The subcooled and low quality fluid conditions used the Henry/Fauske correlation, and the Moody correlation was used for the remainder of the saturated regime.

Calculation of reactor pressure vessel (RPV) cavity pressures was performed using the RELAP4-MOD6 computer code. Pressures were determined with the mass and energy releases from the design breaks at the reactor vessel inlet and outlet nozzles.

The reactor cavity has a net free volume of about 9000 ft^3. The input model contains 36 volume-nodes, determined from sensitivity studies to be detailed enough to provide a convergent solution. Resultant force and moment time histories on the reactor vessel are shown in Figures 1 and 2, respectively, for the hypothesized hot leg break of 0.10A. The 2.0A double-ended cold leg break results are provided in Figures 3 and 4.

Calculation of steam generator compartment pressure was performed using the DDIFF1-MOD7 computer code. Pressure time histories were determined with the mass and energy releases from the design breaks at the steam generator inlet and outlet nozzles. The steam generator cavity is illustrated in Figures 5 and 6 using an elevation view and several plan views. Each plan view in Figure 6 is keyed to an elevation shown in the elevation view of Figure 5. The nodalization shown in the figures refers to the input model of the cavity. Based on the geometry of the cavity and sensitivity studies, 36 volume-nodes were determined to be adequate.

2.3 Structural Analysis

The Licensee's structural analysis was performed utilizing two primary finite element models and several component and support detailed finite element models. The subsystem models were used to develop input to the primary models and to calculate component and support loads and stresses for detailed evaluations. The mathematical models to which asymmetric LOCA loads were applied are described in the following subsections. The general plant layout is shown by the illustration of Figure 7.

2.3.1 Reactor Coolant System Analysis

As shown in Figures 8 and 9, the mathematical model consists of the reactor vessel, simplified reactor internal components, reactor coolant pumps, steam generators, and the interconnecting piping. The RPV support characteristics were determined from an individual detailed model (described in Section 2.3.3). Responses of the modeled system components were calculated using the STRUDL, DAGS, and FORCE computer codes for the four design basis pipe breaks. STRUDL developed the system stiffness matrix which was supplied to DAGs, a nonlinear structural code. Along with other pertinent structural parameters and the applied LOCA forcing functions, DAGS determined the RCS time dependent motions. Applied loads consisted of cavity pressurization and pipe tension release forces, and for pipe breaks within the RPV cavity internal blowdown loads were also included. The nonlinear analysis contained gapped support definition and included the effects of hydrodynamic mass. The DAGS time history motions were supplied to the post-processor code FORCE, which calculated maximum pipe nozzle loads and support loads. For the breaks at the steam generator (SG) inlet and outlet nozzles, the SG was modeled in greater detail to better understand its response. Included in the modeling were the SG internals and support non-linearities as shown in Figure 10. The resulting RCS analysis response time histories were utilized in the subsystem evaluations (Section 2.3.3) of the vessel supports, vessel internals, fuel assemblies, control element drive mechanism (CEDM) and emergency core cooling system (ECCS) piping to determine final components and support qualifications.

2.3.2 Primary Shield Wall Analysis

The ability of the primary shield wall to sustain the worst case pipe rupture loads was determined from a linear elastic, three dimensional model of the wall using the NASTRAN computer code. The mode, shown in Figure 11, represents the shield wall using plate elements and includes the hot and cold leg penetrations. Loads on the model consisted of reactor vessel support reaction loads and reactor cavity pressurization loads. The loads were applied as static loads.

2.3.3 Subsystems Analysis

Numerous smaller, more detailed mathematical models were used in the LOCA analysis to provide representative and meaningful responses to the applied loadings. One model was developed to determine the component support stiffness to be used in the system analyses as well as to qualify the supports once system responses were obtained. The MARC computer program was used for the analyses of the reactor vessel supports, located below the two cold leg nozzles of Loop 1 and below the hot leg nozzle of Loop 2. Figure 12 illustrates the supports, and Figure 13 represents the mathematical model using elastic-plasitic three-dimensional elements. The resulting cold leg and hot leg support load deflection relationships supplied to the system model of Section 2.3.1 are shown in Figures 14 and 15, respectively.

Several other models were utilized for detailed qualifications of particular components. Applied loadings and/or motions to these models were responses from the system dynamic analysis. The detailed models are as follows:

1. The reactor vessel internals were evaluated for guillotine pipe breaks at the reactor vessel inlet and outlet nozzles, employing lateral and axial mathematical models. Nonlinear analyses were performed in accordance with established procedures (Reference 6) using beam elements, gap elements, and linear and nonlinear spring elements. Hydrodynamic coupling effects were included in the horizontal model. Both models were subjected to a combination of applied forces and excitations. The time history forces applied to vessel internals resulted from the LOCA blowdown analysis described in Section 2.1, and the time history motions of the reactor vessel resulted from the RCS analysis, described in Section 2.3.1. The lateral and axial models are shown in Figures 16 and 17 respectively. Results of the analyses were time dependent member loads and nodal displacements, velocities, and accelerations. In addition to the horizontal and vertical responses of the vessel internals, vibration and stability analyses were performed on the CSB to determine possible contributing barrel stresses.

The shell mode response of the barrel due to LOCA pressure loads applied to the barrel from the break at the RPV inlet nozzle were analyzed. Axial type loadings on the CSB from a pipe break at the RPV outlet nozzle were investigated with the aid of the SAMMSOR/DYNASOR computer code and the buckling potential of the barrel was determined. Figures 18 and 19 show the axisymmetric vibration and stability models of the barrel.

2. The fuel was analyzed with the CESHOCK computer code using the mathematical model shown in Figure 20. Displacement time histories of the fuel alignment plate, core shroud, and core support plate from the detailed internals analysis, above, provided the input motions to this nonlinear core model. Stability and any additional bending stress in the fuel assemblies was determined from a dynamic beam-column analysis using the finite element code ANSYS. The model shown in Figure 21 was subjected to concurrent lateral and axial LOCA loadings. Effects of adjacent structures were included in the modeling.

3. The CEDMs were evaluated with an elastic-plastic finite element model using the MARC computer code. Time history motions of the reactor vessel head, determined from the RCS analysis, were applied to the base of the CEDMs. The controlling section of the component is the CEDM nozzle, near the interface with the RPV head.

4. The integrity of the ECCS piping was evaluated for asymmetric LOCA loadings by performing an elastic analysis using the STRUDL and DAGS computer codes. Input excitation to the analysis was provided by the time history motions of the ECCS nozzle, resulting from the system LOCA response. The motions were directly computed at the appropriate location on the reactor coolant pump (RCP) discharge leg. Two lines were analyzed, and the mathematical models are shown in Figures 22 and 23.

2.4 Summary of Licensee's Analytical Results

The basic criteria for acceptability of the plant for the postulated faulted condition is to provide high assurance that the reactor can be brought safely to a cold shutdonw condition. The licensee concluded that overall acceptability of the plant for the postulated LOCA was met. This was demonstrated by the following component and structure evaluations believed by the licensee to be the worst or limiting cases. A summary of load and stress results from the LOCA analyses is presented in Table 1.

2.4.1 Reactor Vessel Supports

The primary supporting system for the reactor vessel consists of three nozzle supports: beneath two cold leg nozzles and one hot leg nozzle. Using the system model previously described in Section 2.3.1, the greatest loads on the supports were horizontal forces, resulting from a postulated break at the reactor vessel inlet nozzle. The criterion for the supports in based on an instability analysis as described in Section 2.3.3, and according to the ASME Code, Section III, Appendix F, loads should not exceed 70% of the plastic instability load. As it can be seen from Table 1, this criterion is exceeded by 59%; however,, a comparison of the resultant load to the instability curve of Figure 14 demonstrates that a considerable amount of additional strain capacity still exists. The RPV supports are considered acceptable for LOCA loadings by the licensee.

2.4.2 Steam Generator Supports

The SG supports were evaluated using the RCS model with the SG described in Section 2.3.1. Subcompartment pressurization considering pipe breaks at the SG inlet and outlet nozzles providing the primary source of applied loadings in the analysis. Support components evaluated consist of the lower pads, lower stop, lower and upper keys, holddown bolts, and snubbers. Acceptability of the support components was based on a comparison of the results to design loads criteria. The design loads are greater for all support components as shown in Table 1. The design loads are less than or equal to 90% of yield values, except for the bolts and snubbers which are compared to yield and actual test values, respectively.

2.4.3 Reactor Coolant Piping

The primary coolant piping was expected to be most highly stressed at component nozzles. Considering the four design basis pipe breaks, resultant loads on the RPV nozzles, SG nozzles, and RCP nozzles were determined from the RCS analysis, described in Section 2.3.1. All loads result in stresses which meet the faulted limits set forth by the ASME Code, Section III, Appendix F, except the moment on the RCP discharge nozzle due to a RPV inlet nozzle guillotine break. Since this result exceeded the the allowable by only 2%, the licensee considered the pressure retaining integrity and geometric stability of the piping not to be impared. Table 1 indicates the acceptability of the primary piping.

2.4.4 Reactor Internals

The three major parts of the internals consist of the core support barrel, the lower core support structure, and the upper guide structure. These components were evaluated with the mathematical models described in Section 2.3.3, and the results are shown in Table 1, as percent margins only. The shell vibration response was combined with the axial and lateral barrel beam response, and barrel stability was investigated and found to be too far removed from stability considerations. The internals responses to the asymmetric LOCA loadings are shown to be acceptable compared to the ASME code, Appendix F allowables: 2.4 S_m for membrane stress intensity and 3.6 S_m for membrane plus bending stress intensity.

2.4.5 Fuel Assemblies

The reactor core was dynamically analyzed for veritcal and horizontal responses, utilizing the axial and lateral models described in Section 2.3.3. A beam-column model was also developed to determine additional bending stresses and check stability, but the beam-column effects did not significantly increase the maximum fuel bundle stresses. Table 1 expresses the critical results of the analysis. Maximum spacer grid impact loads in the peripheral assemblies are shown to exceed the grid strengths determined from test results. In an effort to demonstrate core coolability an analysis was performed on the peripheral assemblies assuming reduced area coolant channels. Single channel flow area was reduced up to 35% around the hottest rod in the peripheral assembly, effectively reducing the net assembly flow area by 10%. The linear heat generation rate of the hottest rod in the peripheral assembly was calculated to be less than that of the hottest rod in the core, indicating acceptable ECCS performance.

2.4.5 Control Element Drive Mechanism

The evaluation of the CEDM is based on an elastic-plastic instability analysis (Section 2.3.3). The maximum moment at any section, due to the RPV motions during the transient, occurs at the base of the CEDM nozzle. From Table 1 it can be seen that this moment is less than 70% of the plastic instability load; therefore, the component is acceptable.

2.4.7 Emergency Core Cooling System Piping

The ECCS piping was evaluated based on the results of an elastic dynamic analysis using the substructural models described in Section 2.3.3. Motions from the RCS dynamic analysis provided the input excitations to the models, and the calculated maximum piping moments were shown to meet the acceptance criteria of 70% of the instability load. The results are given in Table 1, for the pipe break at the RPV outlet nozzle.

2.4.8 Primary Shield Wall

The reactor cavity wall was evaluated for cavity pressurization loadings and vessel support reaction loads resulting from pipe breaks at the vessel inlet and outlet nozzles. A static analysis was performed with the structural model described in Section 2.3.2. Worst case reaction loads were applied to each support, and peak cavity loads for each model element were applied as a continuous static pressure. The results are represented on the interaction diagrams of Figure 23 and are shown to be within the capacity of the shield wall, indicating that the integrity of the wall is maintained for all loading conditions.

3.0 STAFF EVALUATION

The licensee's analysis procedure including analytical models, computer methods, and acceptance criteria have been evaluated by the staff for asymmetric LOCA loads. The staff evaluation was accomplished by reviewing the licensee's submittal and using the independent audit calculations performed by the staff or their consultants. In general, the staff has concluded that the licensee's assessment of the asymmetric LOCA loads problem is acceptable. The staff evaluation of each specific analysis phase is addressed in subsequent paragraphs, following the guidelines set forth by NUREG-0609.

3.1 Thermal Hydraulic Blowdown Loads

The thermal hydraulic blowdown calculation portion of the Millstone Unit 2 asymmetric LOCA load submittal has been reviewed and is considered to be acceptable to the staff. The basis of this acceptance is the staff's review and approval of the CEFLASH-4B computer code used for the internal hydraulic loads calculations. Independent audit calculations for CE 2570 MW plant by the staff's consultant aided in approval of CEFLASH-4B application to subcooled blowdown (Reference 7). The code does not consider fluid-structure interaction, and the structural boundaries are assumed rigid and at rest. Such conditions normally give rise to conservative pressures and loads. A significant number and location of postulated pipe breaks were analyzed to determine worst case loadings on the primary coolant system. Size and length of break openings consisted of reasonable and realistic values. Nodalization and modeling were also developed in a manner that provided reasonable representation of the existing system.

3.2 Cavity Pressurization Analysis

The licensee's reactor cavity pressurization analysis of the Millstone Unit 2 plant for postulated breaks at the reactor vessel inlet and outlet nozzles has been reviewed and is considered to be acceptable by the staff. The basis of this acceptance is the staff's review and approval of the CEFLASH-4 and RELAP4-MOD6 computer codes used for calculating LOCA mass and energy release rates and cavity pressure loadings, respectively. Although the licensee used RELAP4-MOD6 instead of RELAP4-MOD5, the code and its application were concluded to be acceptable. The licensee used a flow multiplier of 0.7 instead of the recommended value of 1.0 in the CEFLASH-4 cal-culations. The value 0.7, was justified by comparison of test data with the critical flow correlations. The nodalization of the input model is acceptable based on the staff's review of input data and sensitivity studies performed by the licensee.

The SG subcompartment pressurization analysis of the Millstone Unit 2 plant for postulated breaks at the SG inlet and outlet nozzles has been reviewed and is considered acceptable. Acceptance is based on the staff's review of the data provided by the licensee and its previous review and approval of the DDIFF1-MOD7 code for calculating LOCA cavity pressure loadings. The nodalization of the input model is acceptable based on review of the input data and sensitivity studies performed by the licensee.

3.3 Structural Evaluation

3.3.1 Evaluation of Methods and Models

The structural computer codes cited in the licensee's report are found to be acceptable to the staff. The codes (STRUDL, DAGS, NASTRAN, MARC, CESCHOCK, ASHSD, SSMSOR/DYNASOR, and ANSYS) utilized in the LOCA analyses have been bench marked in a satisfactory manner to the staff. The methods used in performing the requ[illegible] structural analyses are acceptable to the staff in as muc[illegible] they conform to the accepted state-of-the-art, standards, and regulatory codes. Based on the submittal (References 1 and 2) reviews, the detail employed in the system and subsystem structural finite el[illegible] models is considered acceptable by the NRC staff for pred[illegible]ng the mechanical response.

The staff evaluation in this report has considered the need to combine LOCA and safe shutdown earthquake (SSE) loads in the design of the RCL piping. The staff believes that there is sufficient technical evidence (Reference 9) which demonstrates that the SSE and LOCA for the main loop piping in PWR plants may be considered as independent events in determining the appropriate combination of the effects of accident conditions and natural phenomena as required by GDC 2.

In its load combination program, as a part of generic issue B-6, Lawrence Livermore National Laboratory (LLNL) conducted a program to estimate the probability of a double ended guillotine break (DEGB) in the reactor coolant loop piping of PWRs.

The results of the LLNL investigations indicate that the probability of a direct seismically induced DEGB is extremely small. The best estimate probabilities of direct DEGB using the medians of the distribution of the modeling uncertainties ranges from 5×10^{-14} to 7×10^{-12} per plant year for both Westinghouse and Combustion Engineering plants. From the uncertainty analysis, considering the whole range of modeling uncertainty, it is concluded that a direct DEGB probability of 3×10^{-10} per plant year can be considered as the absolute upper bound for Westinghouse and CE plants.

Indirectly induced DEGB in the reactor coolant loop piping (defined as a DEGB in the reactor coolant loop piping as a result of an impact with a large component or structure, e.g., a falling polar crane) is a more likely event compared to direct DEGB; however, the probability of indirect DEGB is also very small. For the lowest seismic capacity Westinghouse plant, the median probability of DEGB is 3.3×10^{-6} per plant year. The corresponding indirect DEGB probability at the 90th percentile is 2.3×10^{-5}. Even for this lowest capacity plant, these probability values are still very small. For all 46 Westinghouse units east of the Rockies as a whole, the median probability is more than one order of magnitude lower. The probability values for the Combustion Engineering plants are also very low. The upper bound probability values for the Combustion Engineering plants are compatible with those of the Westinghouse plants. Based on the results of these probability studies, staff evaluations discussed in Section 3.3.2 and results of independent confirmatory analysis under various loading conditions, the staff has concluded that the dynamic loading on the reactor coolant piping for Millstone Unit 2 calculated by the licensee without the SSE-LOCA load combination, is acceptable.

The instability approach in the analyses of the RCS supports, CEDM, and ECCS piping is acceptable since it complies with the ASME Code, Section III, Appendix F guidelines.

The determination of fuel deformation and spacer grid impact loads is accepted as the appropriate internals motion (upper and lower grid plate and core shroud) is adequately incorporated as the fuel assembly forcing functions. The acceptability of the rule analysis is also based on audit calculations performed by the staff's consultant, EG&G Idaho, Inc. (Reference 8) The audit determined that the CE modeling scheme utilizes a dual load path for the spacer grids and, therefore, provides an adequate response of the fuel assemblies.

Determination of the total stresses in the core barrel resulting from the asymmetric downcomer depressurization using decoupled beam and shell modes is acceptable since this procedure has been shown to be mathematically exact for linear analyses.

Analysis of the ECCS piping is acceptable based on the bounding analysis performed by the licensee. This analysis consisted of a dynamic analysis of the most highly stressed ECCS lines for motion of the ECCS nozzles on the cold leg piping as determined from the RCS dynamic system analysis.

Acceptability of the shield wall analysis is based on the conservatism employed in the structural model and the applied loads.

3.3.2 Compliance with Acceptance Criteria

The licensee's stress and/or load evaluation of the reactor system components is acceptable to the staff. The criteria used in the evaluation are, in general, in agreement with industry standards and meet the acceptance criteria outline in NUREG-0609. Although some exceptions to the outlined criteria occur, functionality of each analyzed reactor system component is demonstrated.

The reactor vessel supports exceed the ASME Code, Appendix F criteria based on 70% of the instability load. However, a comparison of the resulting support load to the instability curve indicates sufficient strain capacity to exist. Therefore, the component is considered acceptable.

The licensee's stress and/or load evaluation of the reactor vessel internals, primary piping, CEDMs, and ECCS piping is acceptable since ASME Code, Appendix F criteria are met. The only exception to this is the resultant moment on the RCP discharge nozzle, which exceeded the elastic limit by 2%. But due to the conservatism of the analysis and the allowable being exceeded by only 2%, the component is acceptable.

Acceptability of the steam generator support evaluation is based on the comparison of the calculated support loads to design loads, yield capacities, and test loads. The LOCA results for the supports are well within their allowable limits.

Acceptance of the shield wall stress evaluation is based on the use of standard industry practices for determining load criteria and the use of conservative material properties.

Two principal acceptance criteria apply for the LOCA, which includes the asymmetric effects: (a) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (b) the 10 CFR 50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisfied if the calculated loads on the fuel rods and components other than grids remain below designated allowable values. The second criterion is shown to be satisfied by an ECCS analysis.

Stresses are calculated in accordance with the previously approved methods documented in Reference 6. Maximum stress levels associated with the fuel rods and fuel assembly components other than grids are determined by the licensee to be below ASME Code, Appendix F, allowable values. Fuel rod fragmentation will, therefore, not occur.

Although a small number of spacer grids are predicted to experience some permanent deformation following a LCOA, including those with asymmetric effects, the effect of this grid distortion was conservatively incorportated into an appropriate ECCS analysis. The peak clad temperature predicted for this distorted geometry is 2036°F and the peak local clad oxidation is 14.5%. Both values are below their respective allowable limits of 2200°F and 17%. The computed values are based on an analysis using an assumed maximum reduced channel flow area of 35%, a slightly greater value than the maximum expected 34% obtained from tests reported by the licensee.

Control rod insertability is not required for a large break LOCA, but based on the fuel system analysis, control rod insertion should not be significantly impaired.

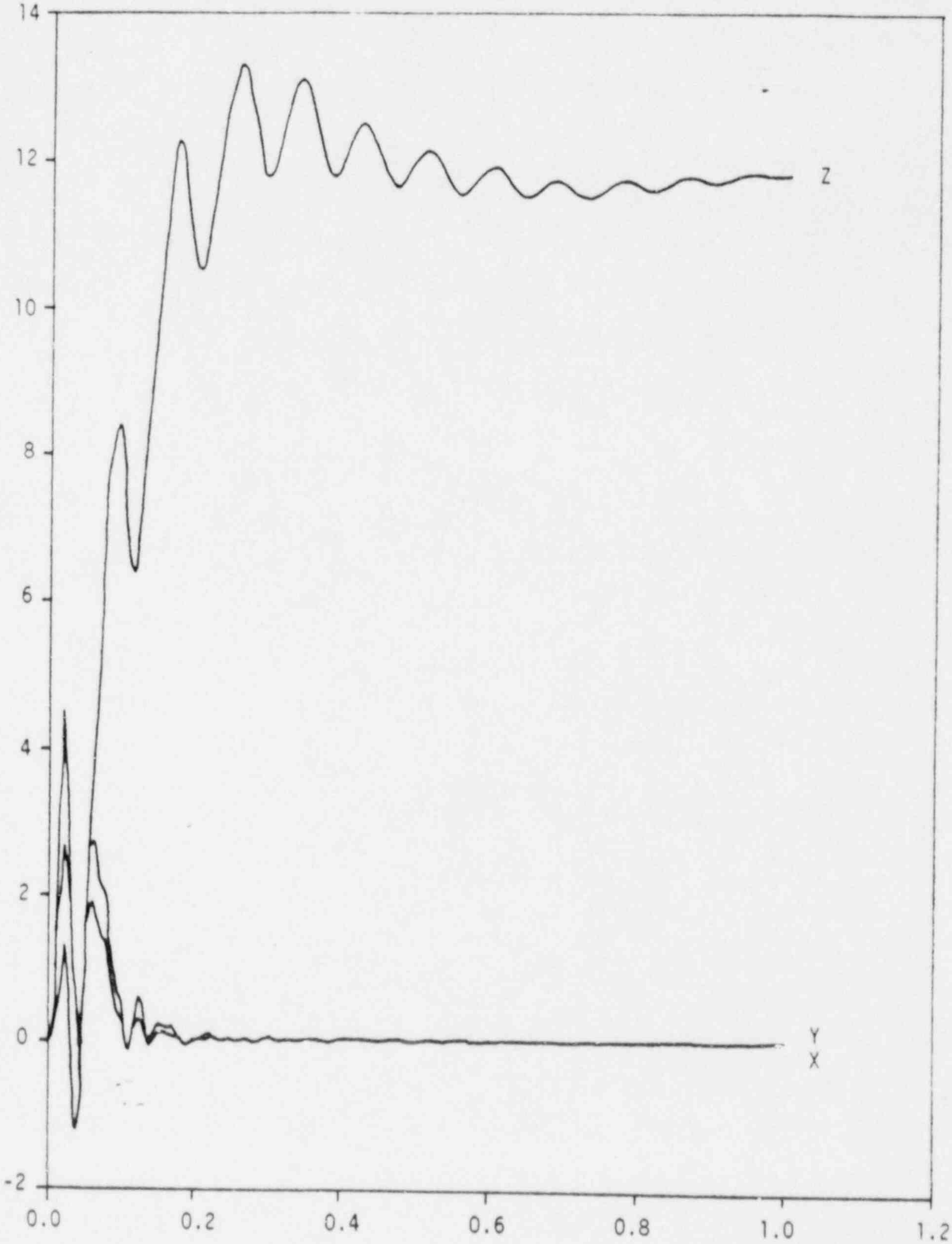

Figure 1. Hot leg break force time history.

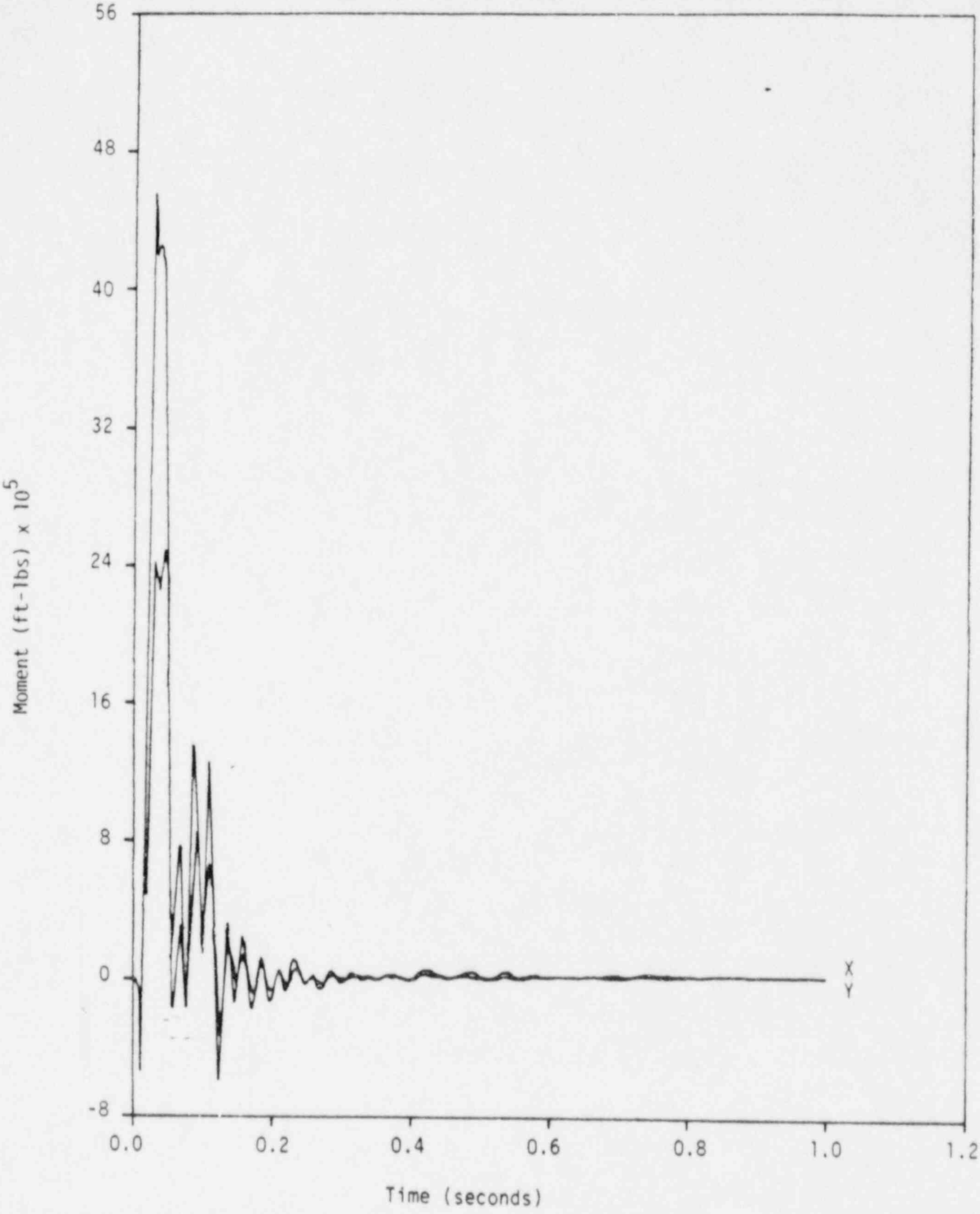

Figure 2. Hot leg break moment time history.

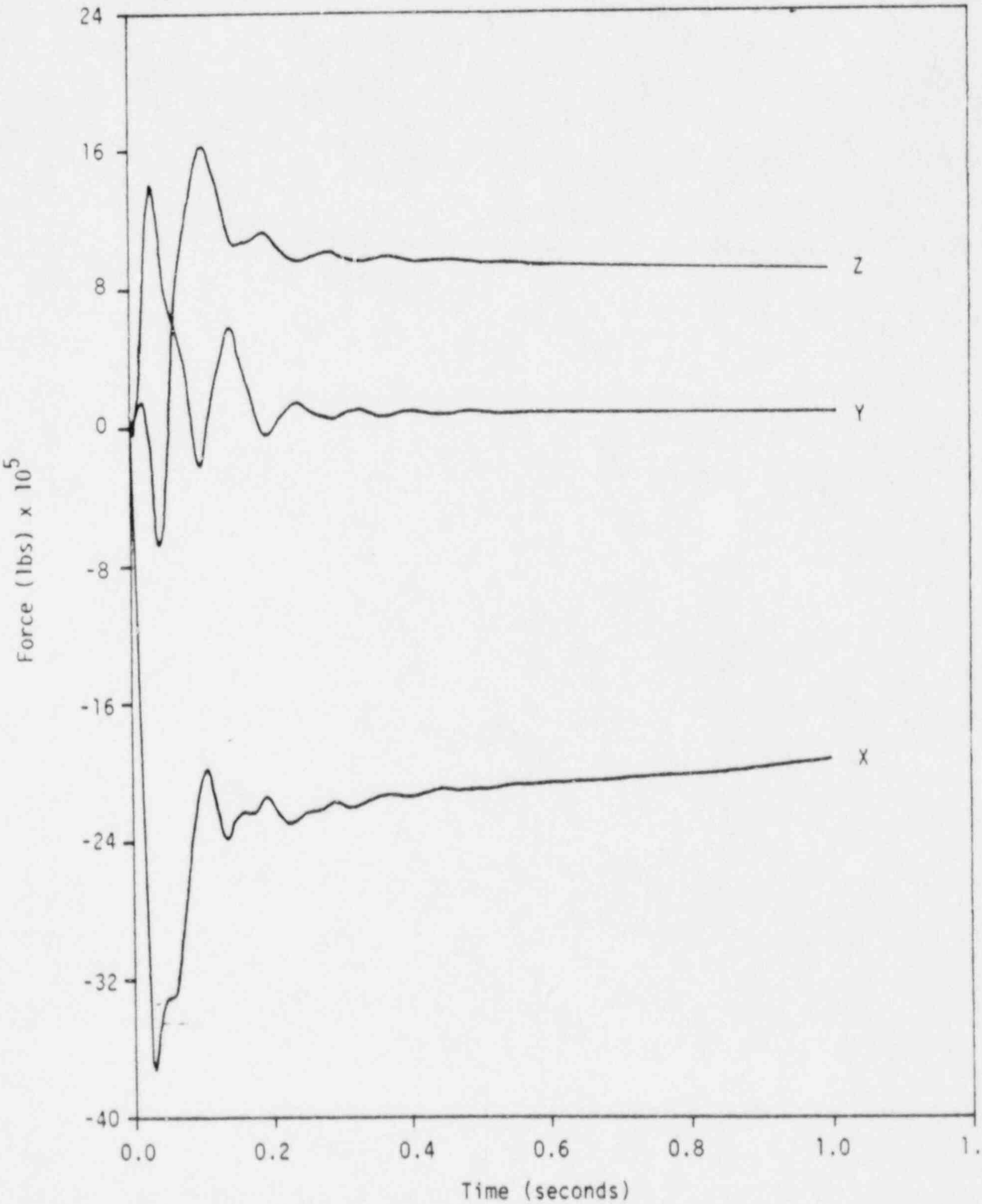

Figure 3. Cold leg break force time history.

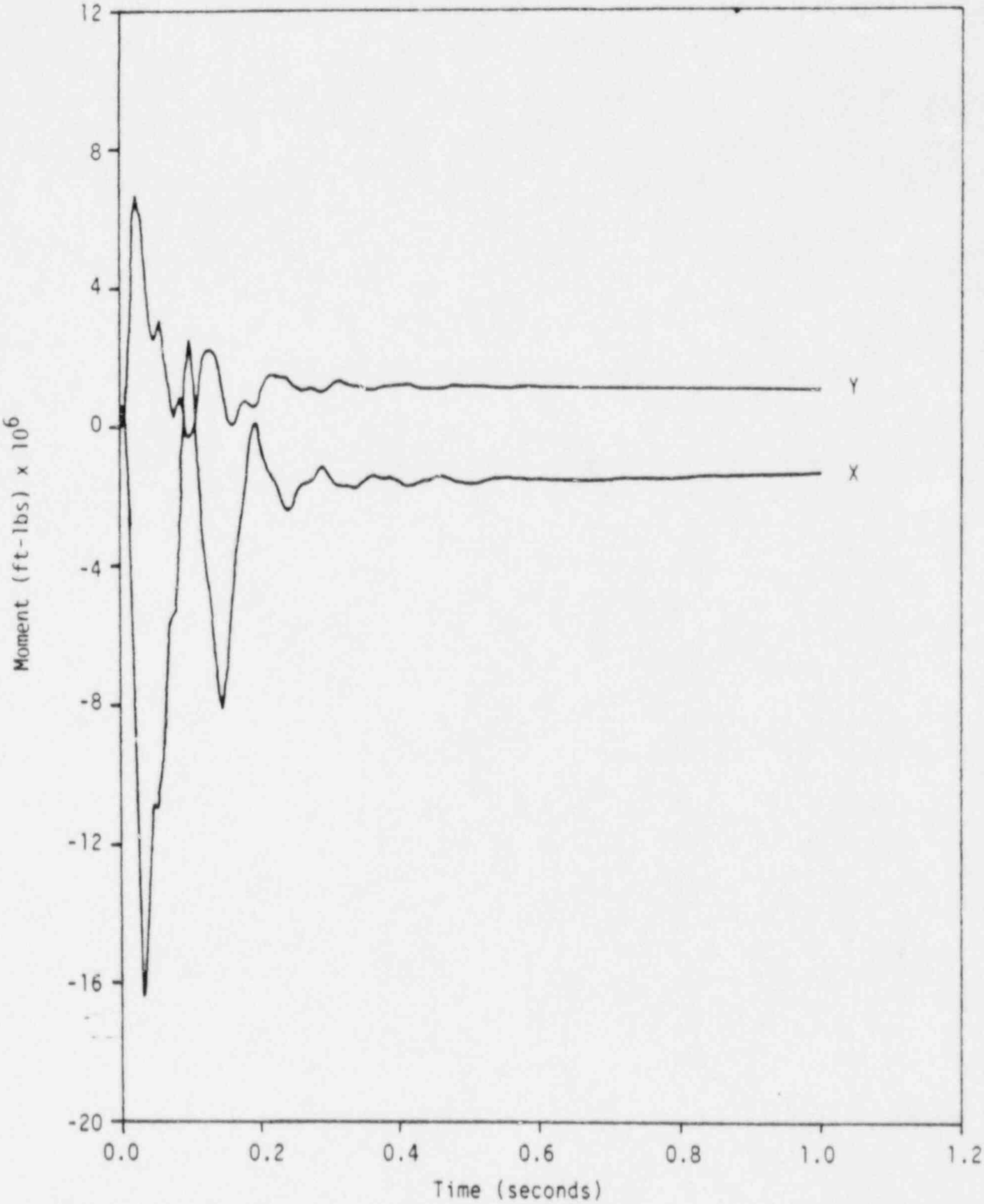

Figure 4. Cold leg break moment time history.

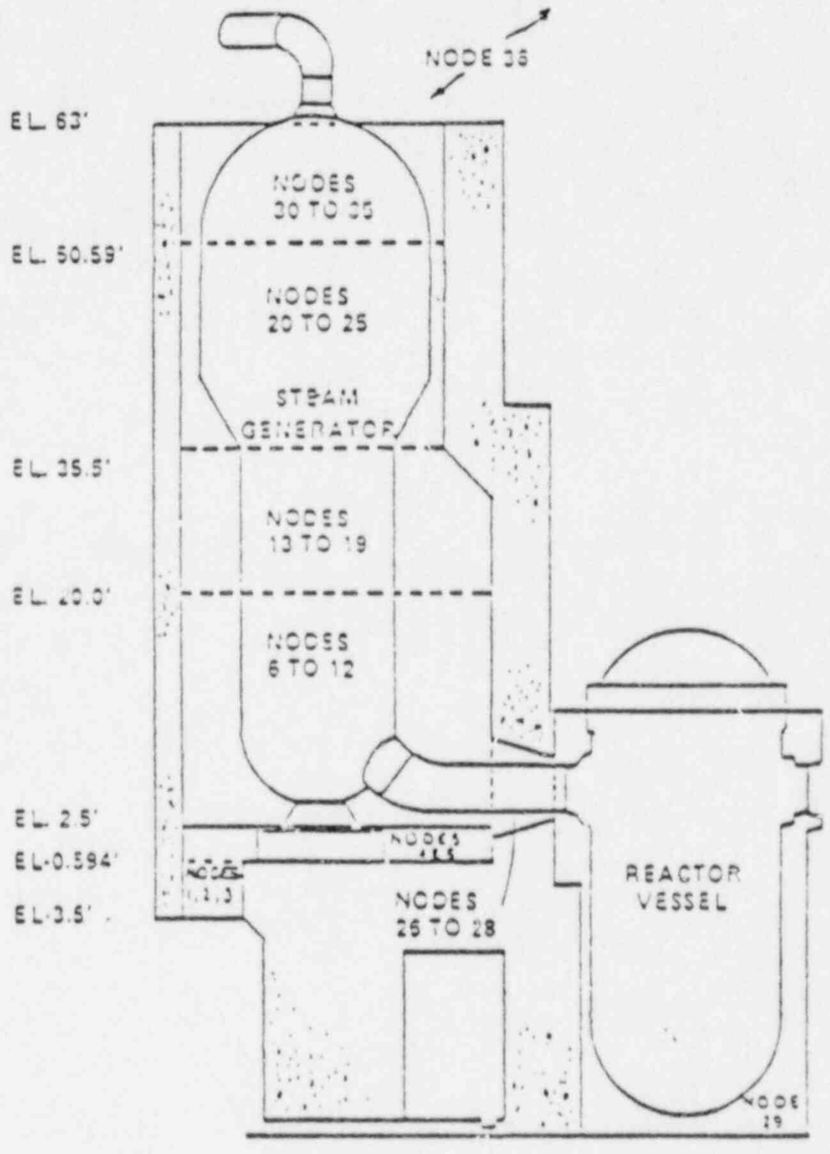

Figure 5. Steam generator compartment, elevation view.

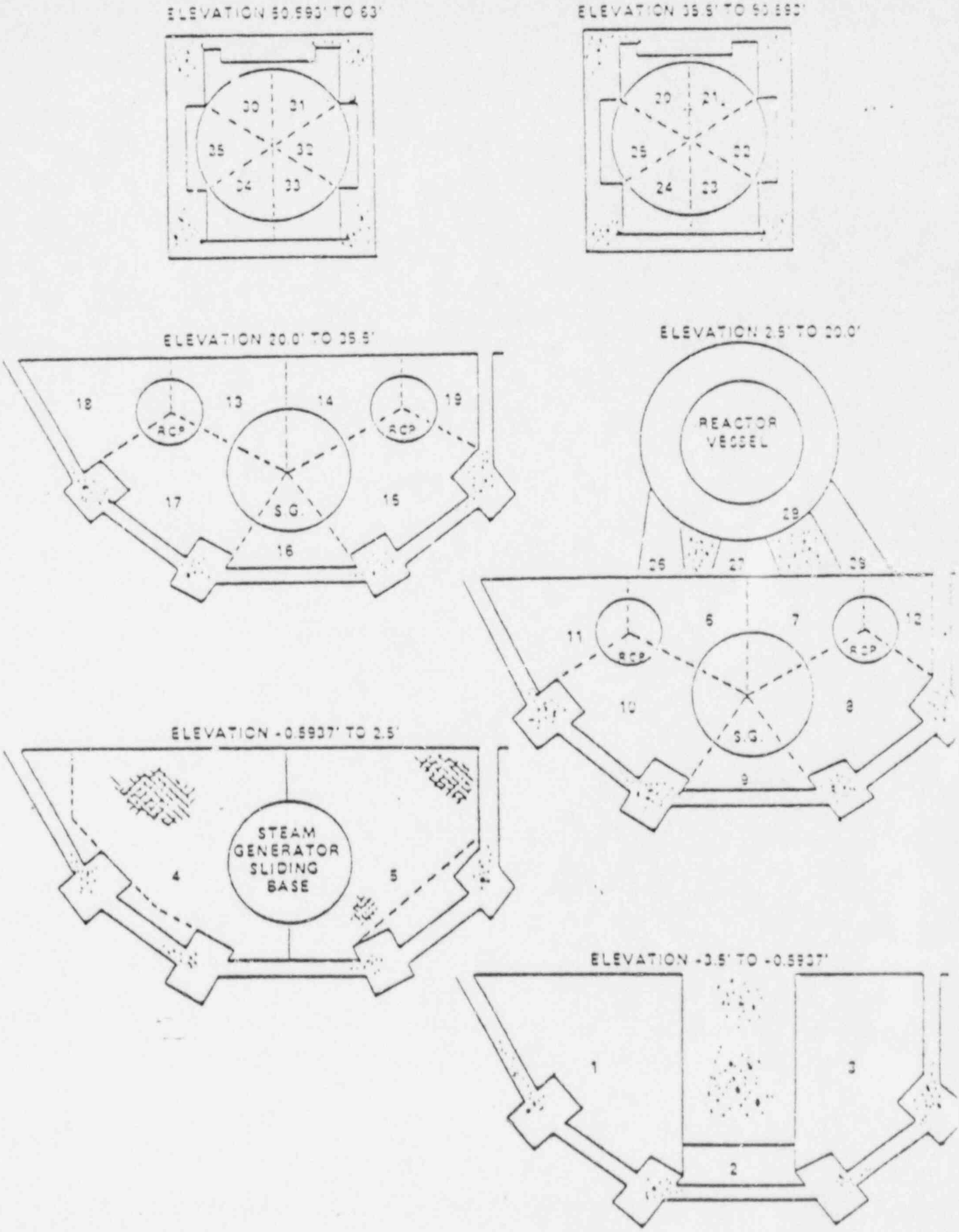

Figure 6. Steam generator compartment, plan views.

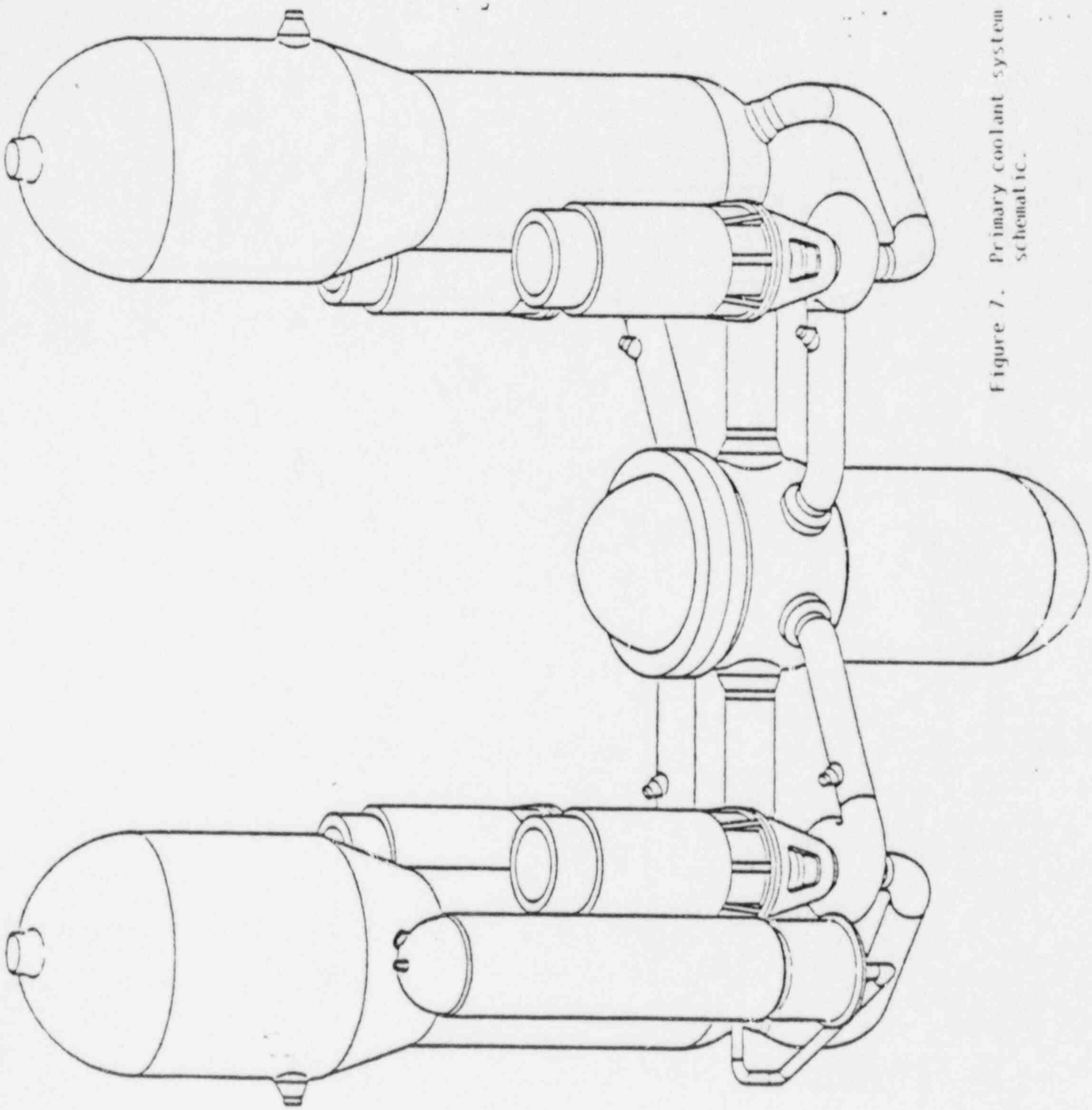

Figure 7. Primary coolant system schematic.

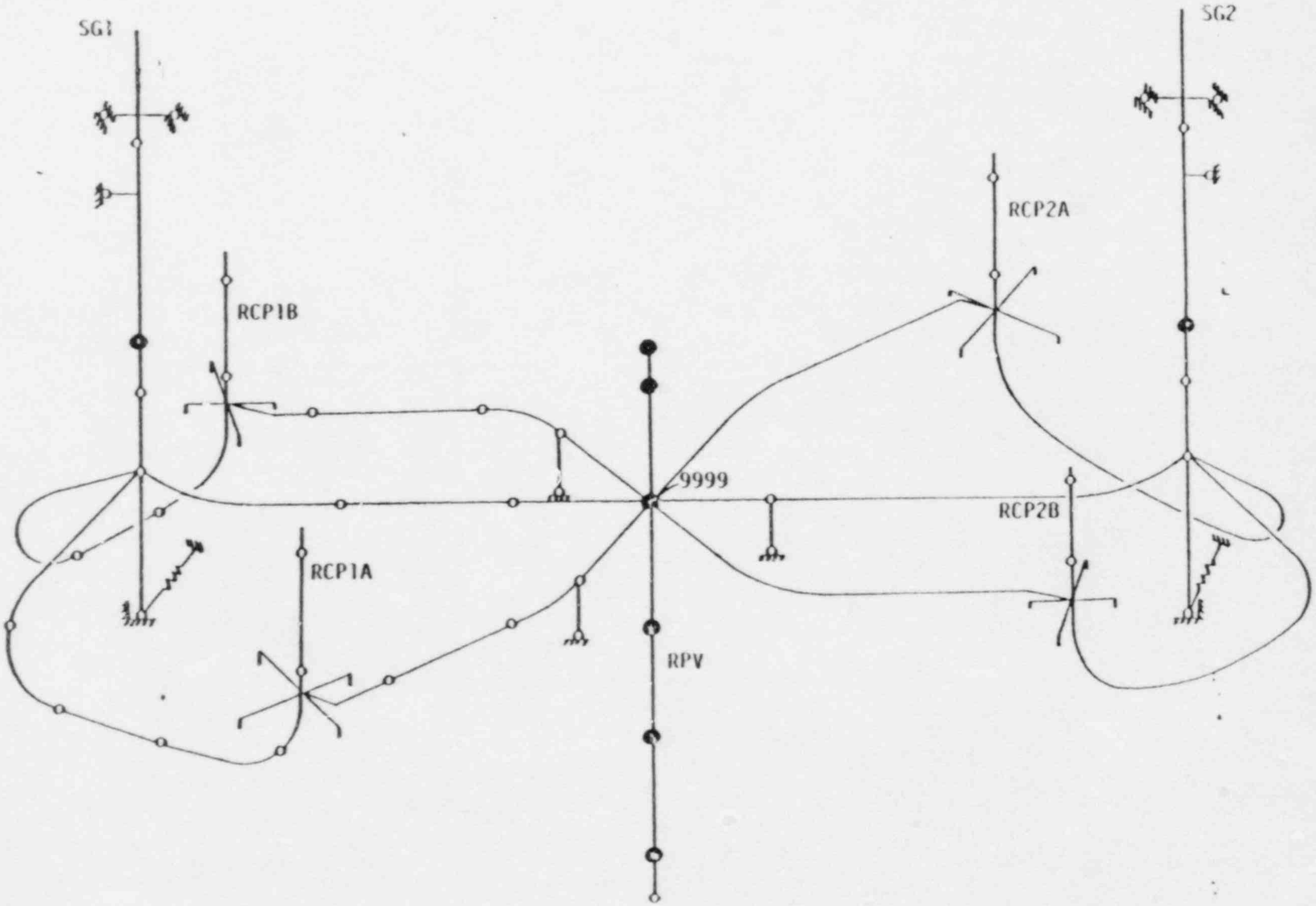

Figure 8. Reactor coolant system model.

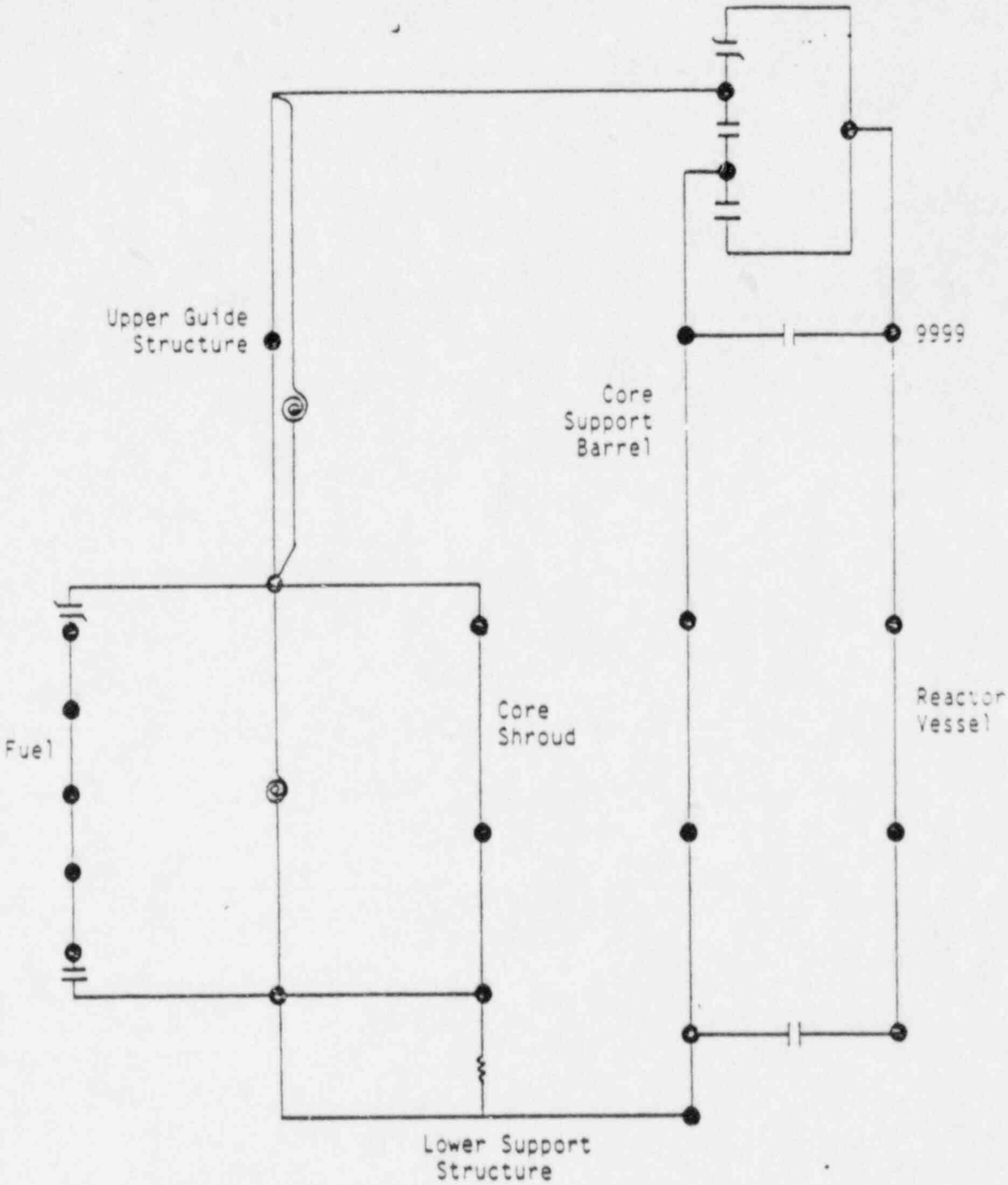

Figure 9. RCS reduced internals model.

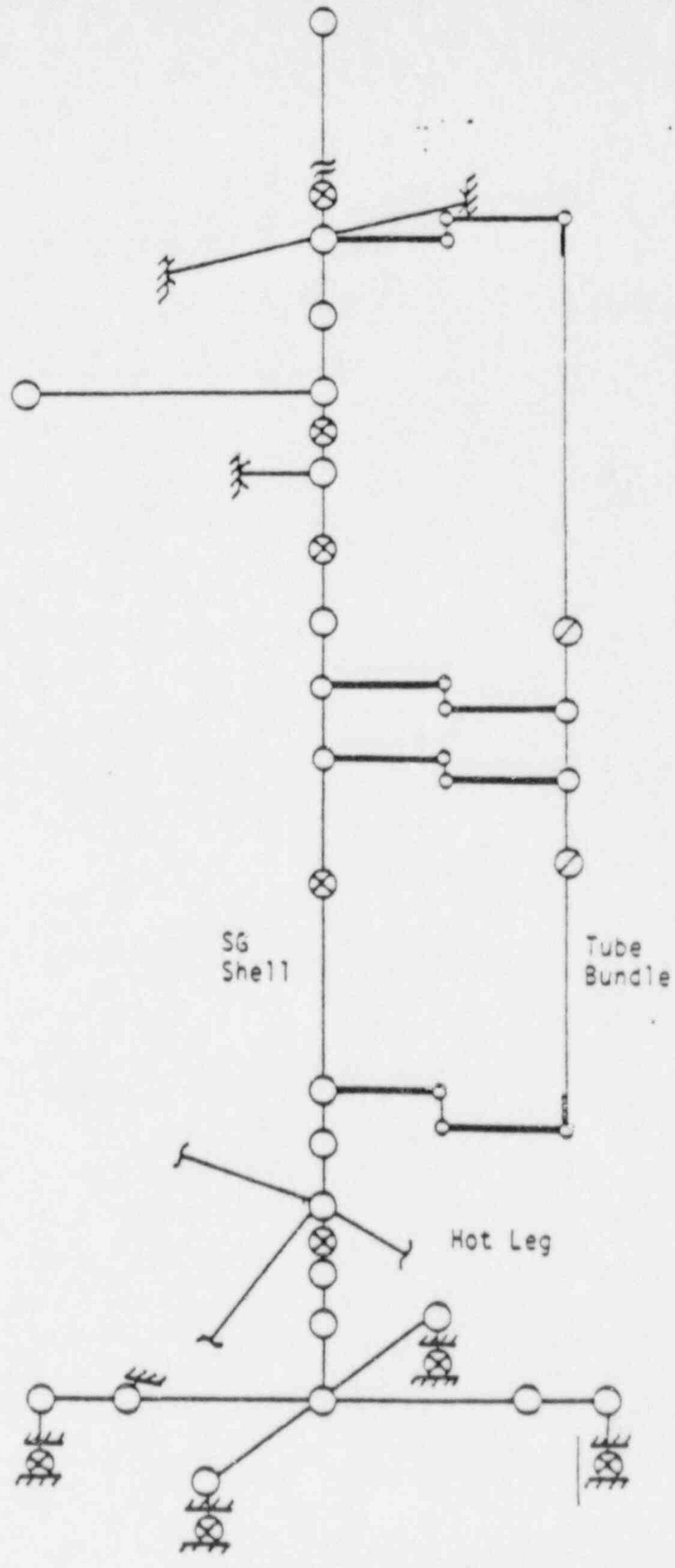

Figure 10. Detailed steam generator model.

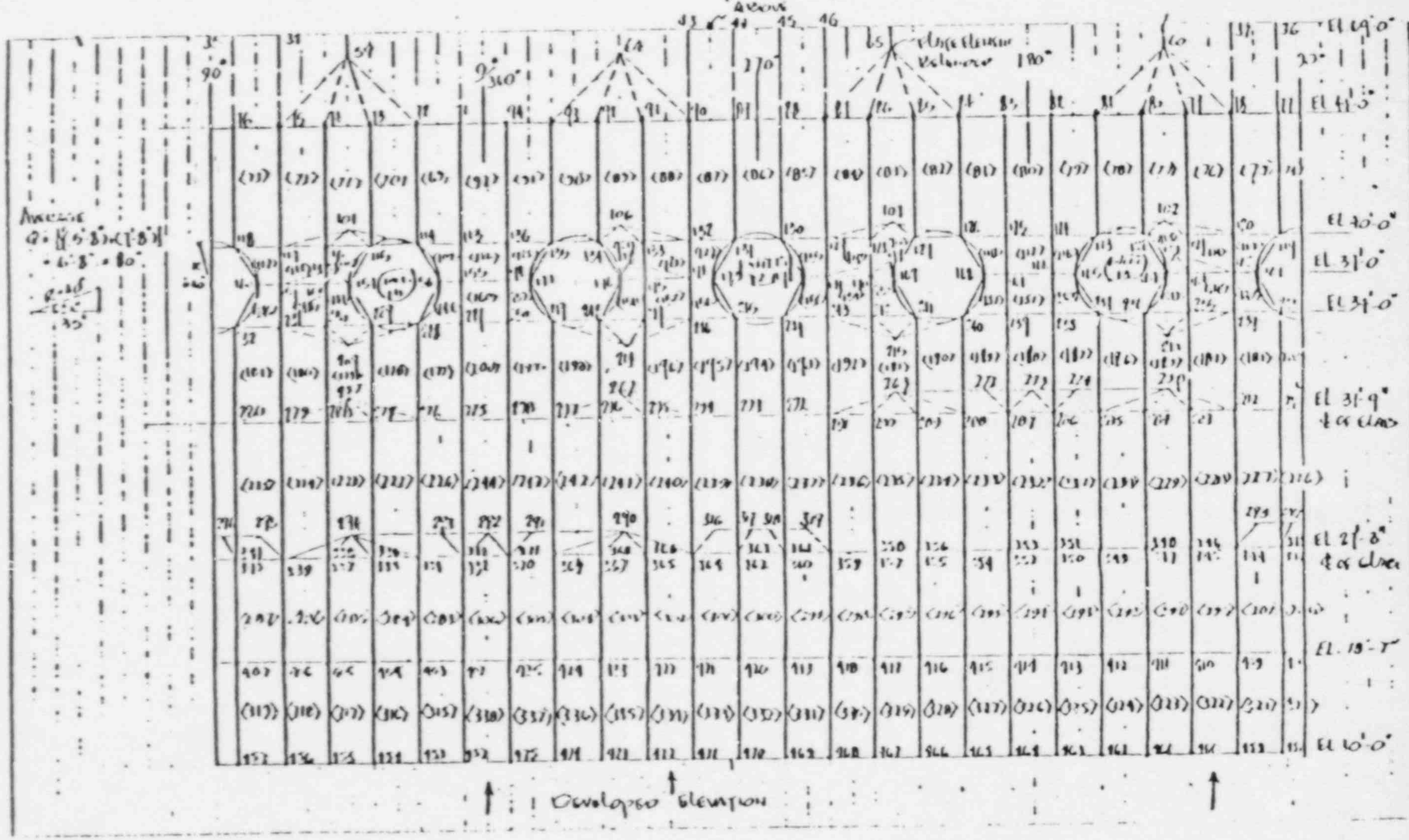

Figure 11. Cavity wall model.

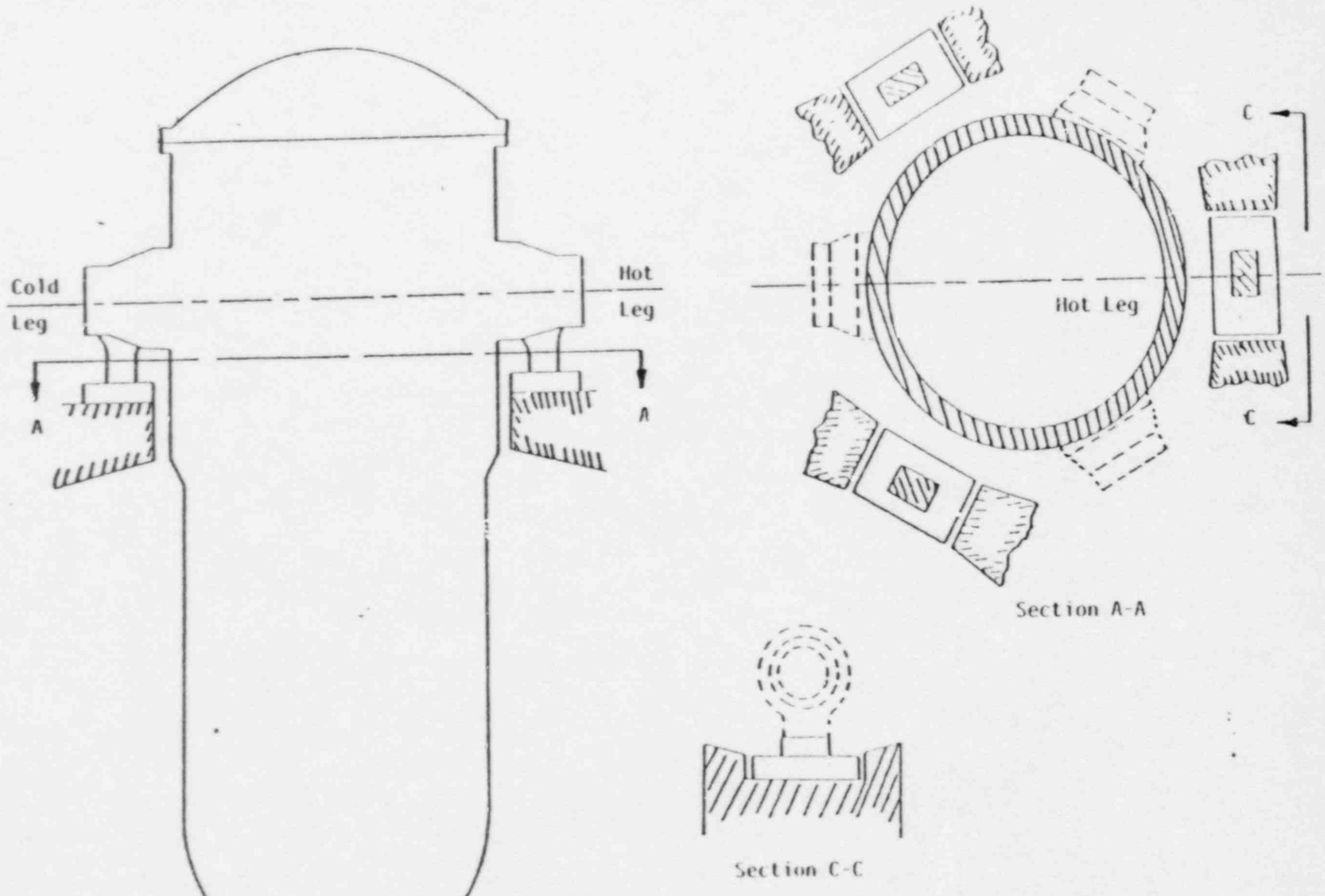

Figure 12. Reactor vessel supports.

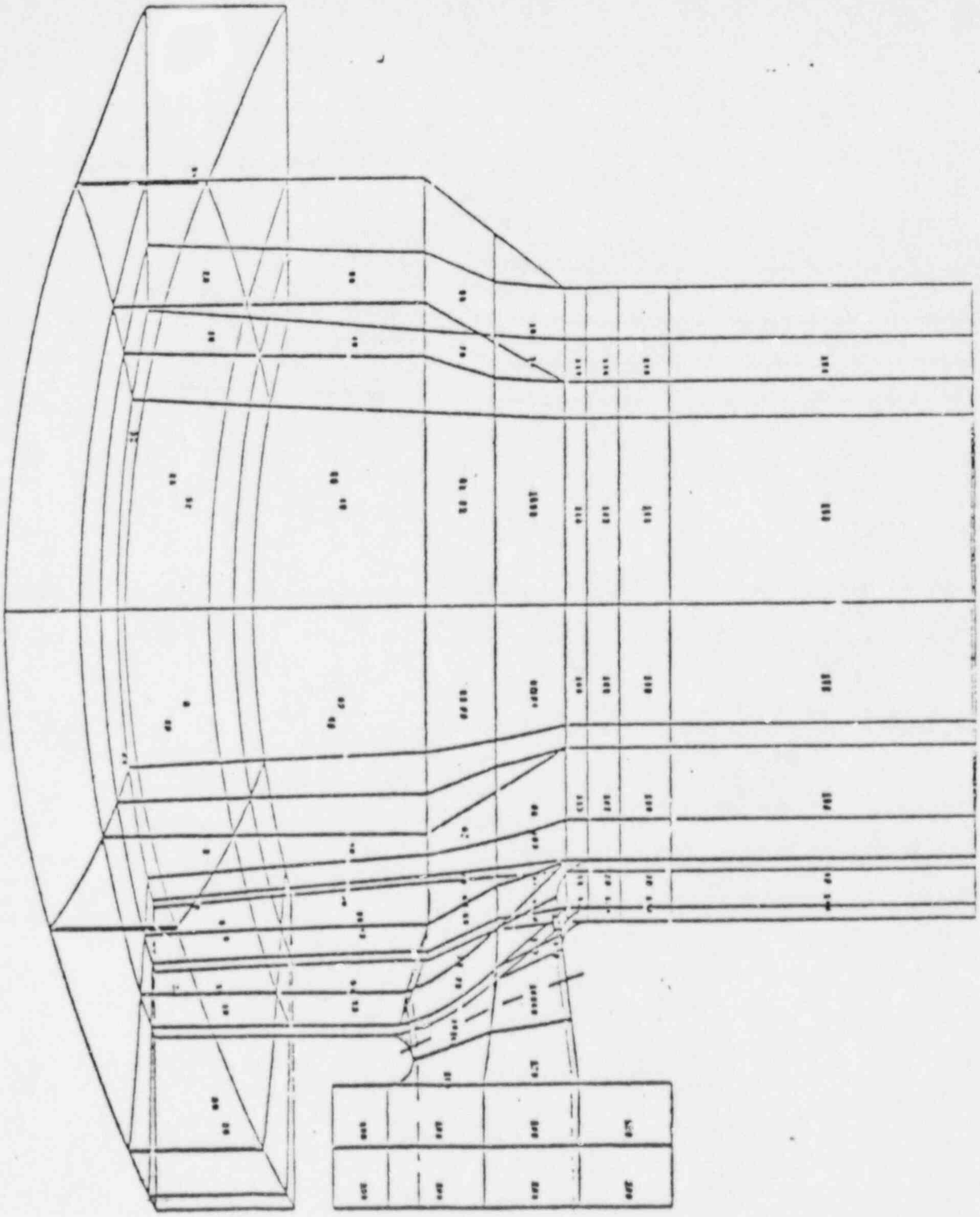

Figure 13. Reactor vessel support model.

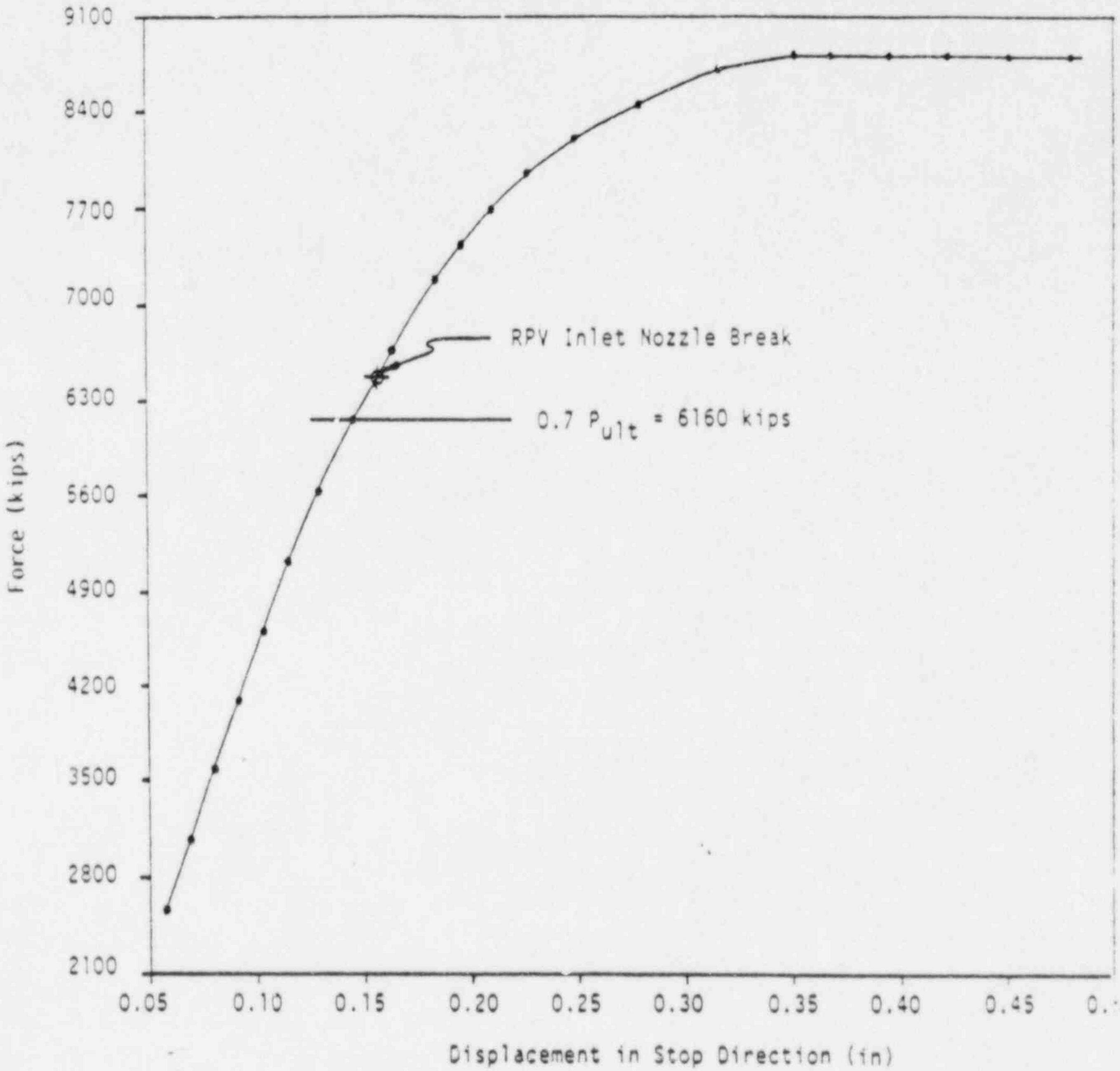

Figure 14. Cold leg support foot load-deflection relationship.

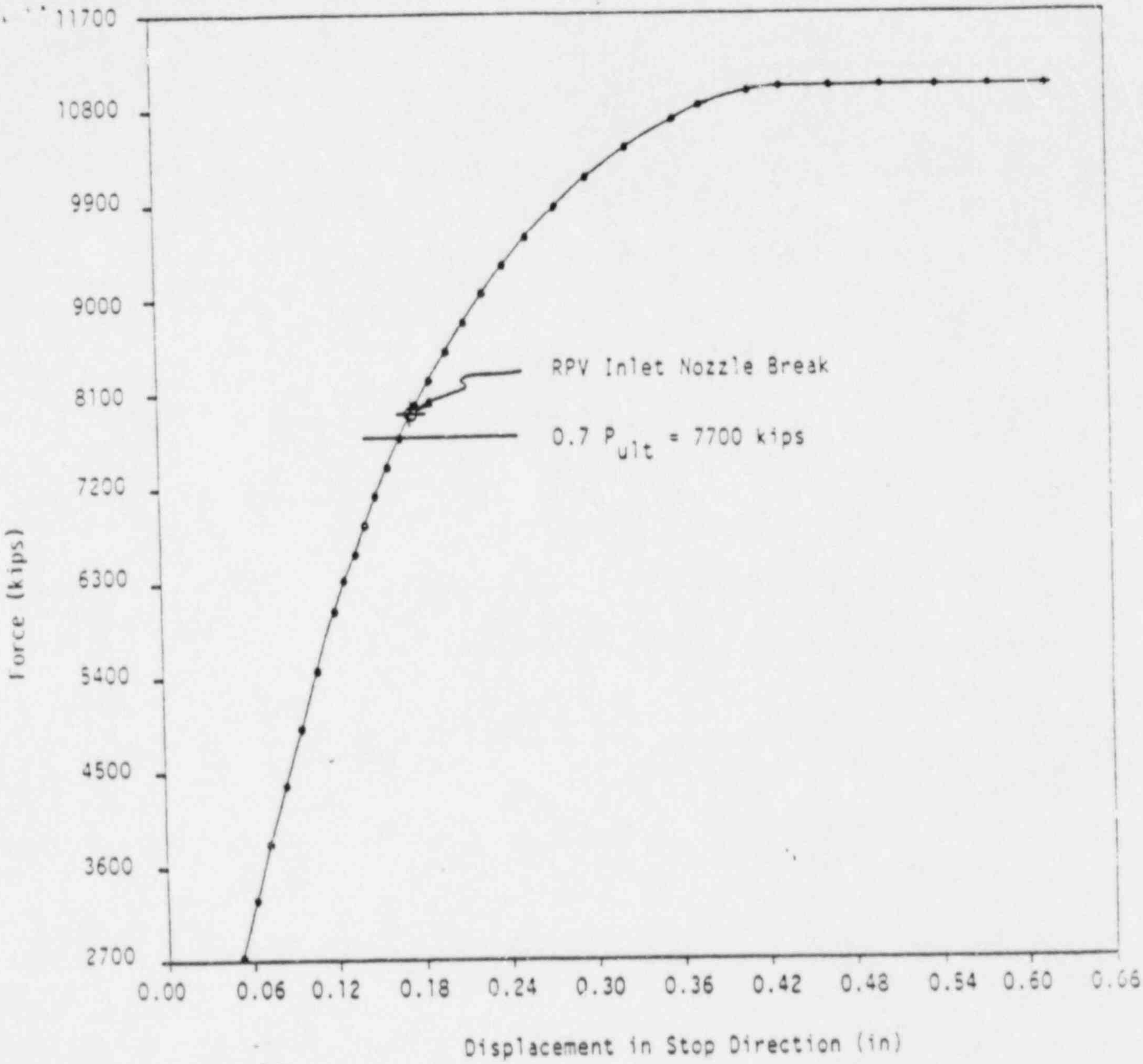

Figure 15. Hot leg support foot load-deflection relationship.

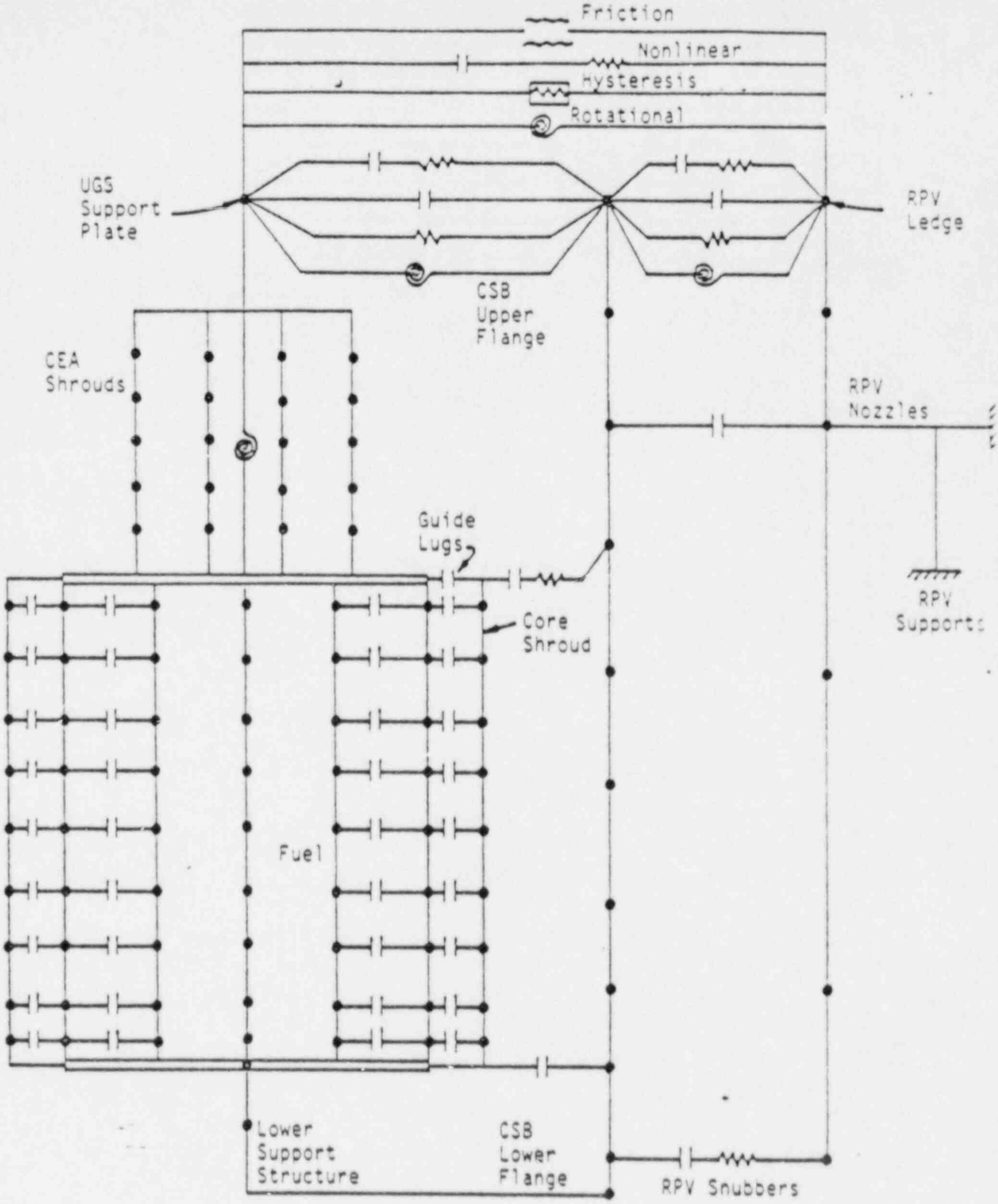

Figure 16. Lateral internals model.

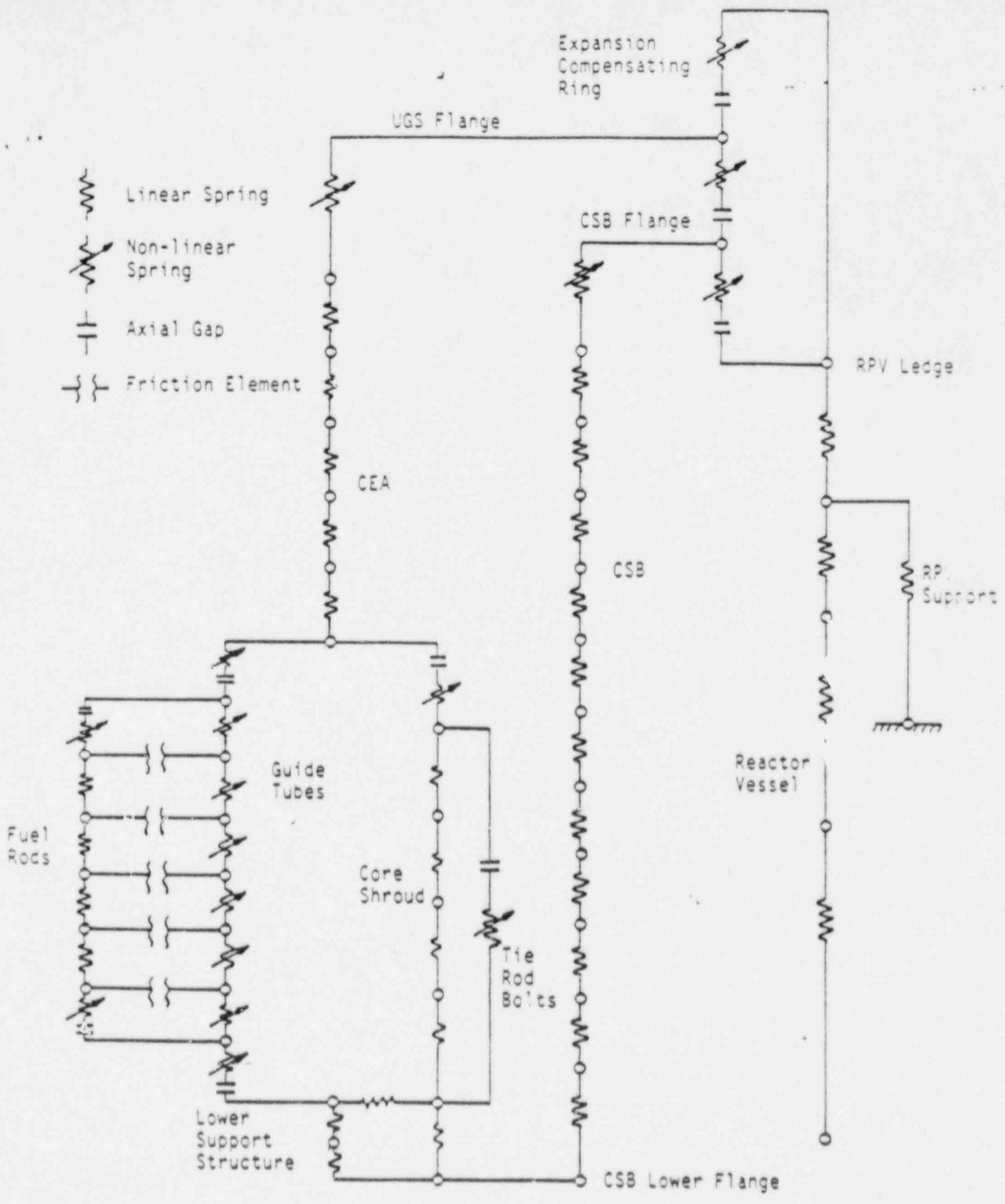

Figure 17. Axial internals model.

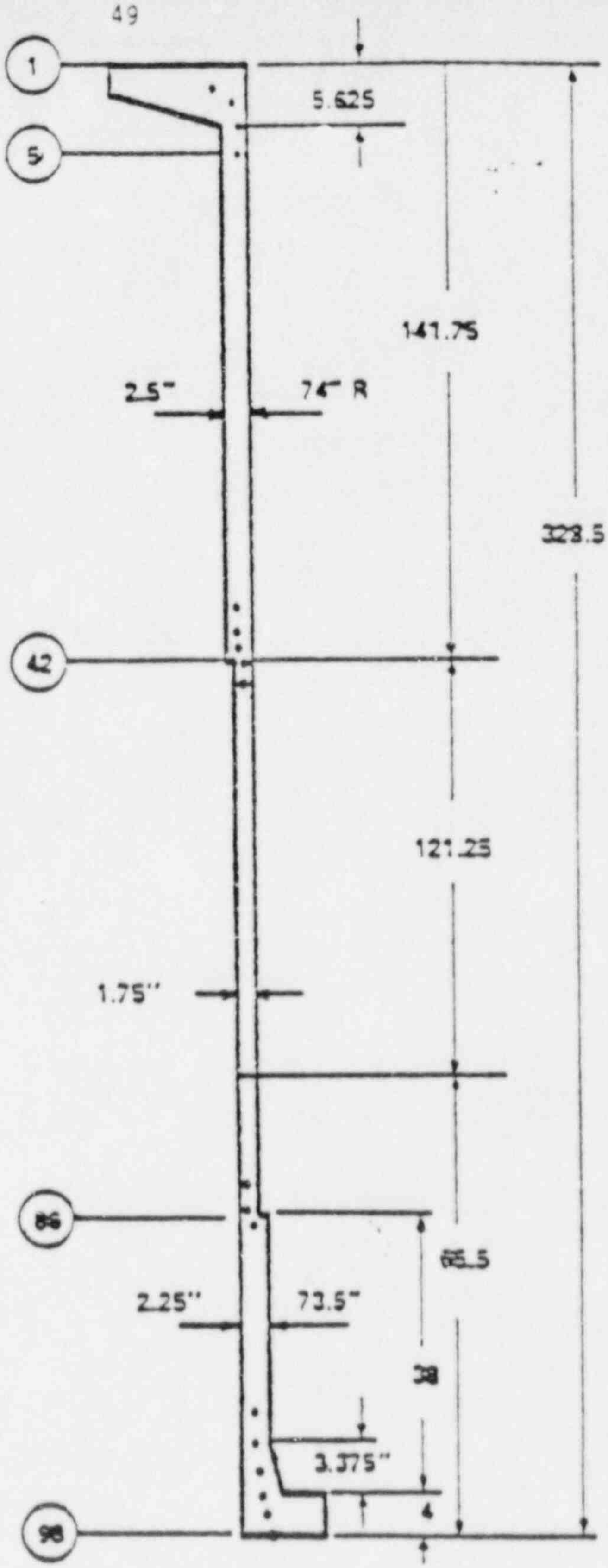

Figure 18. CSB axisymmetric vibration model.

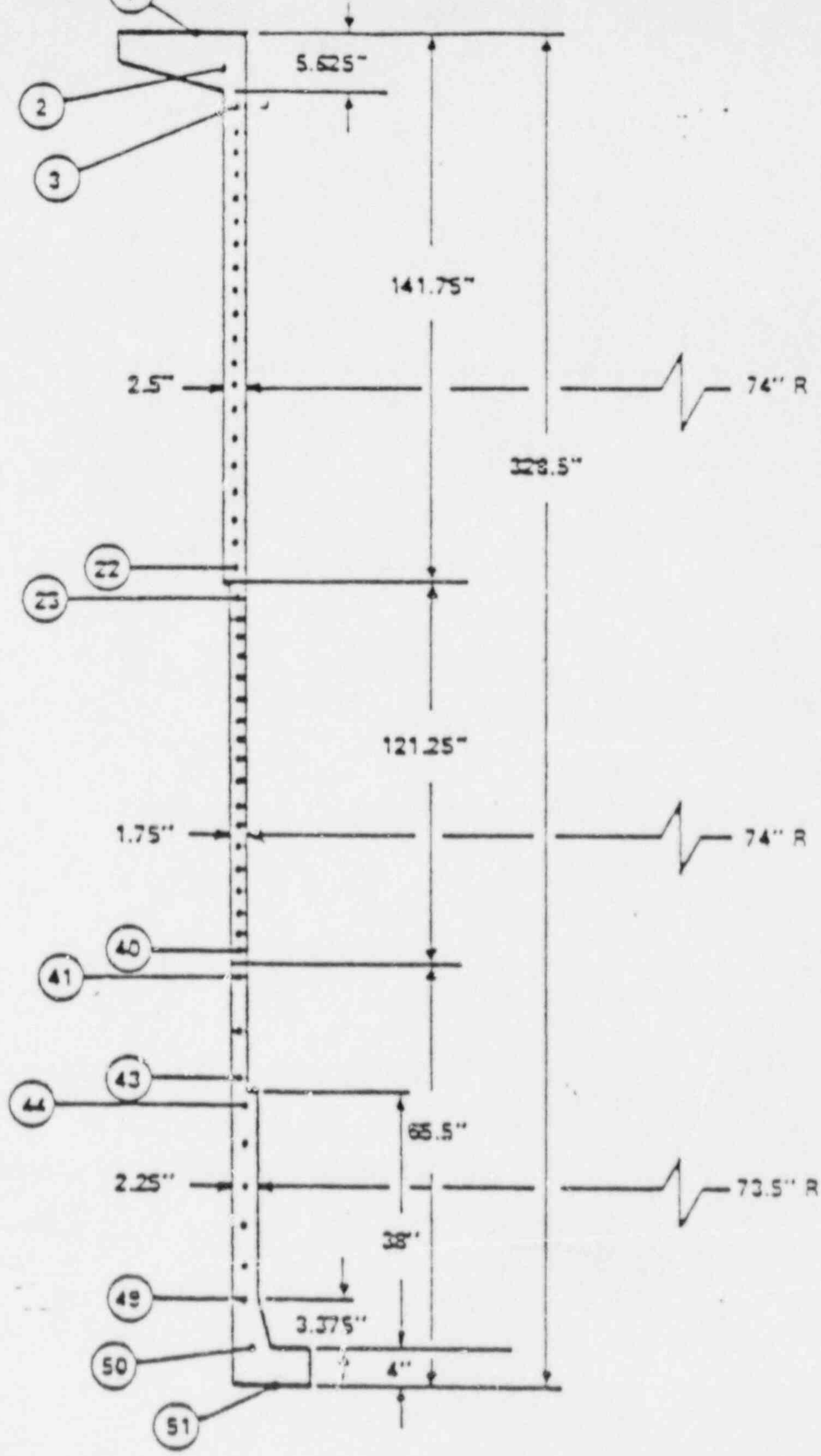

Figure 19. CSB axisymmetric stability model.

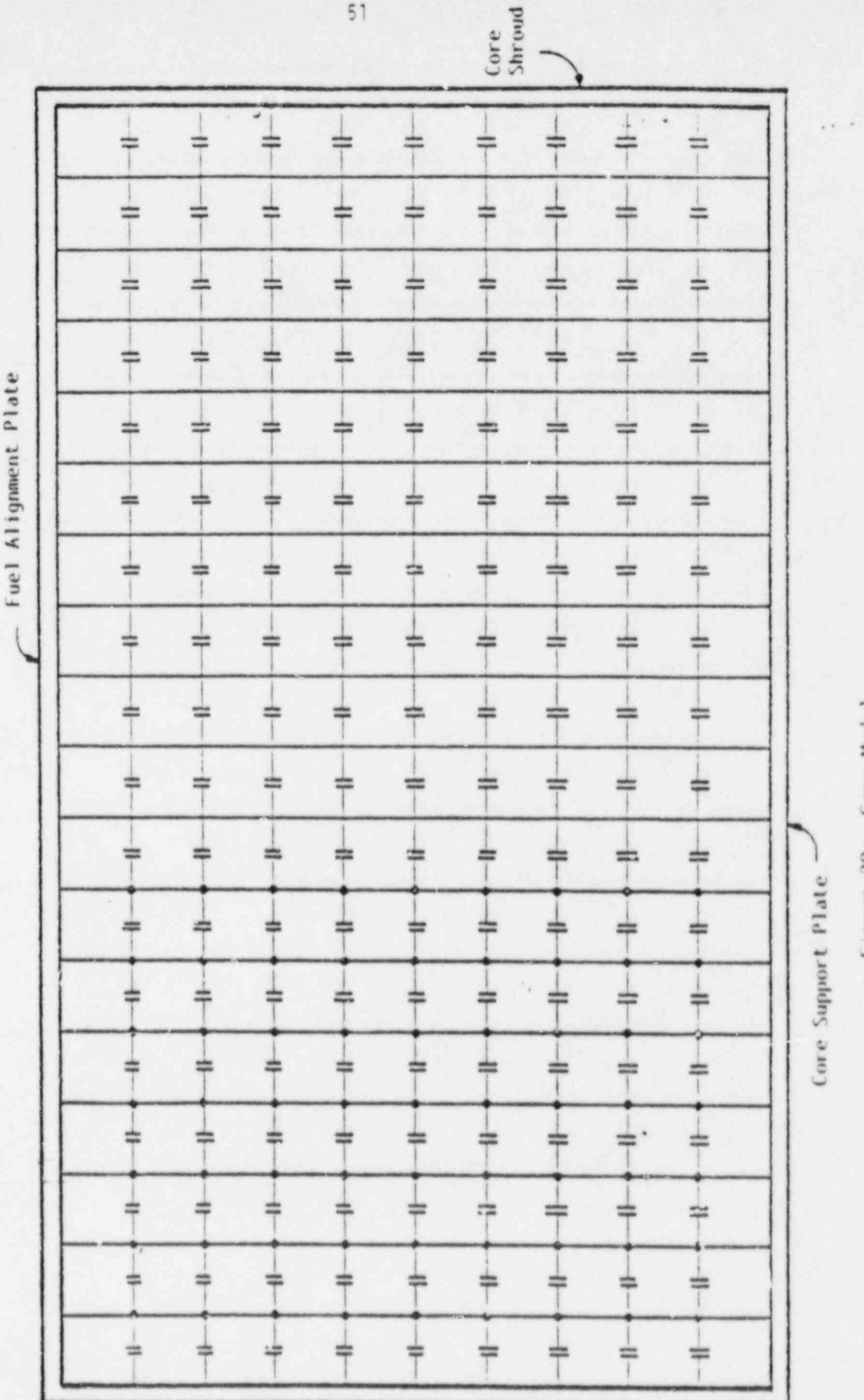

Figure 20. Core Model

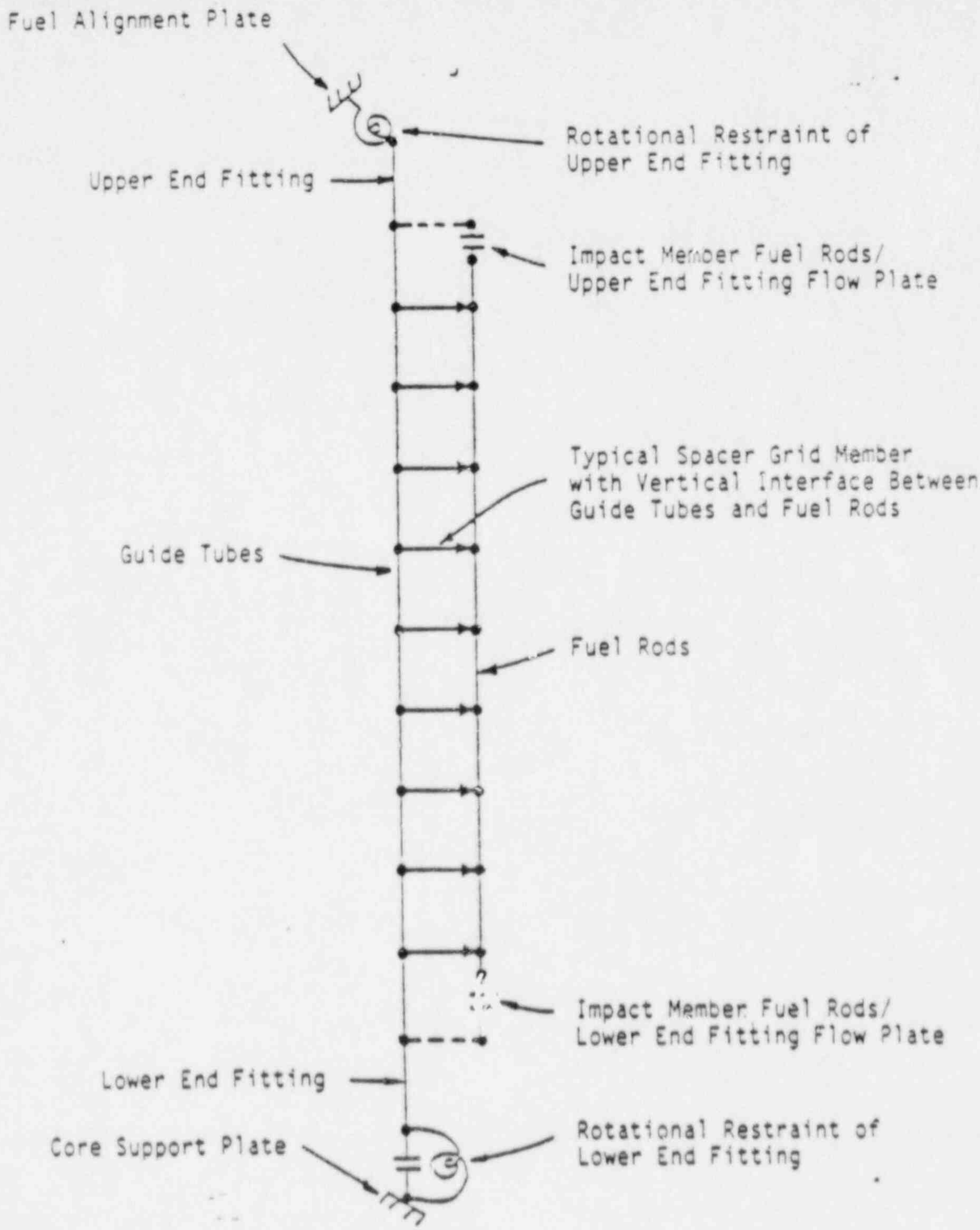

Figure 21. Fuel assembly beam-column model.

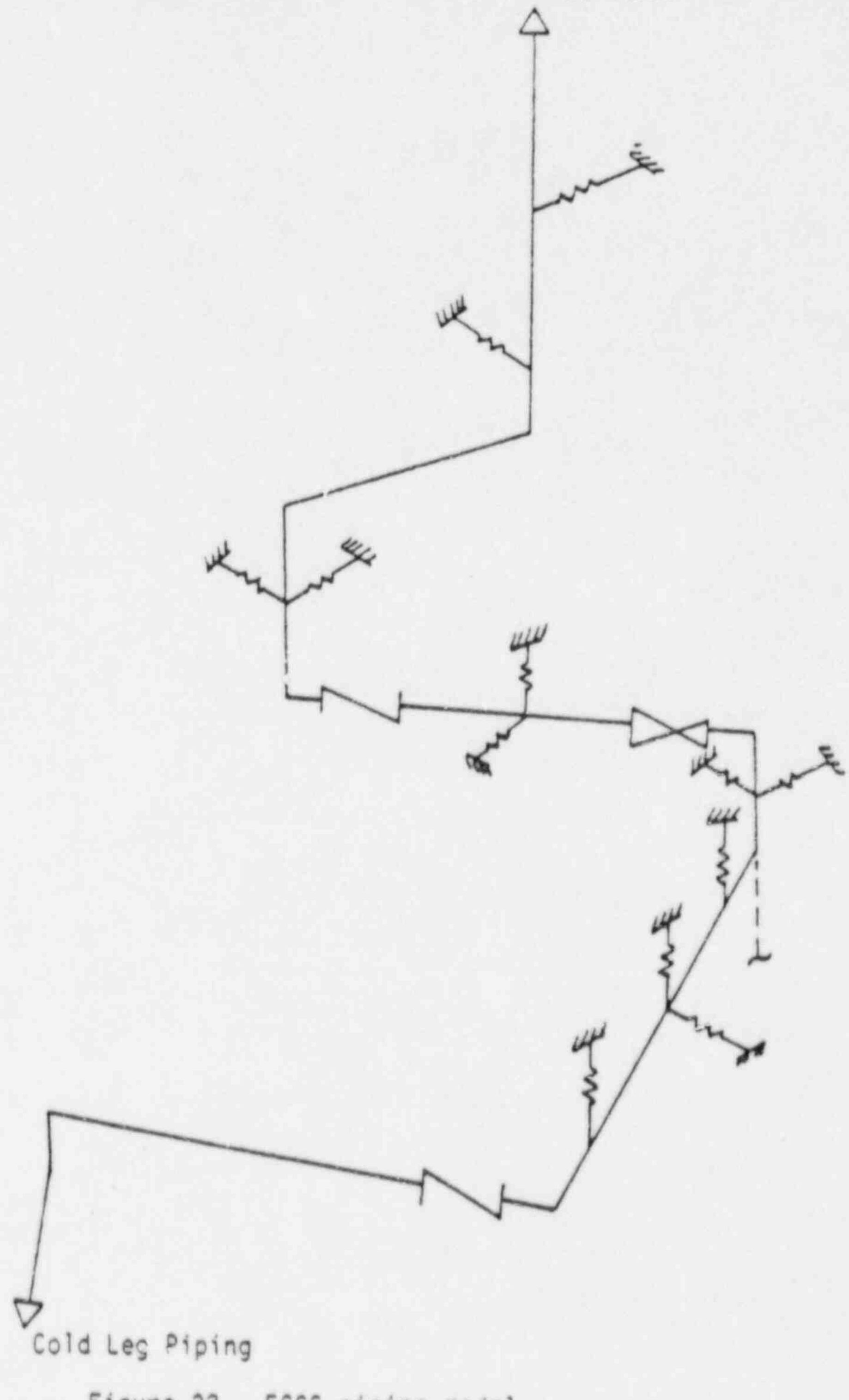

Figure 22. ECCS piping model.

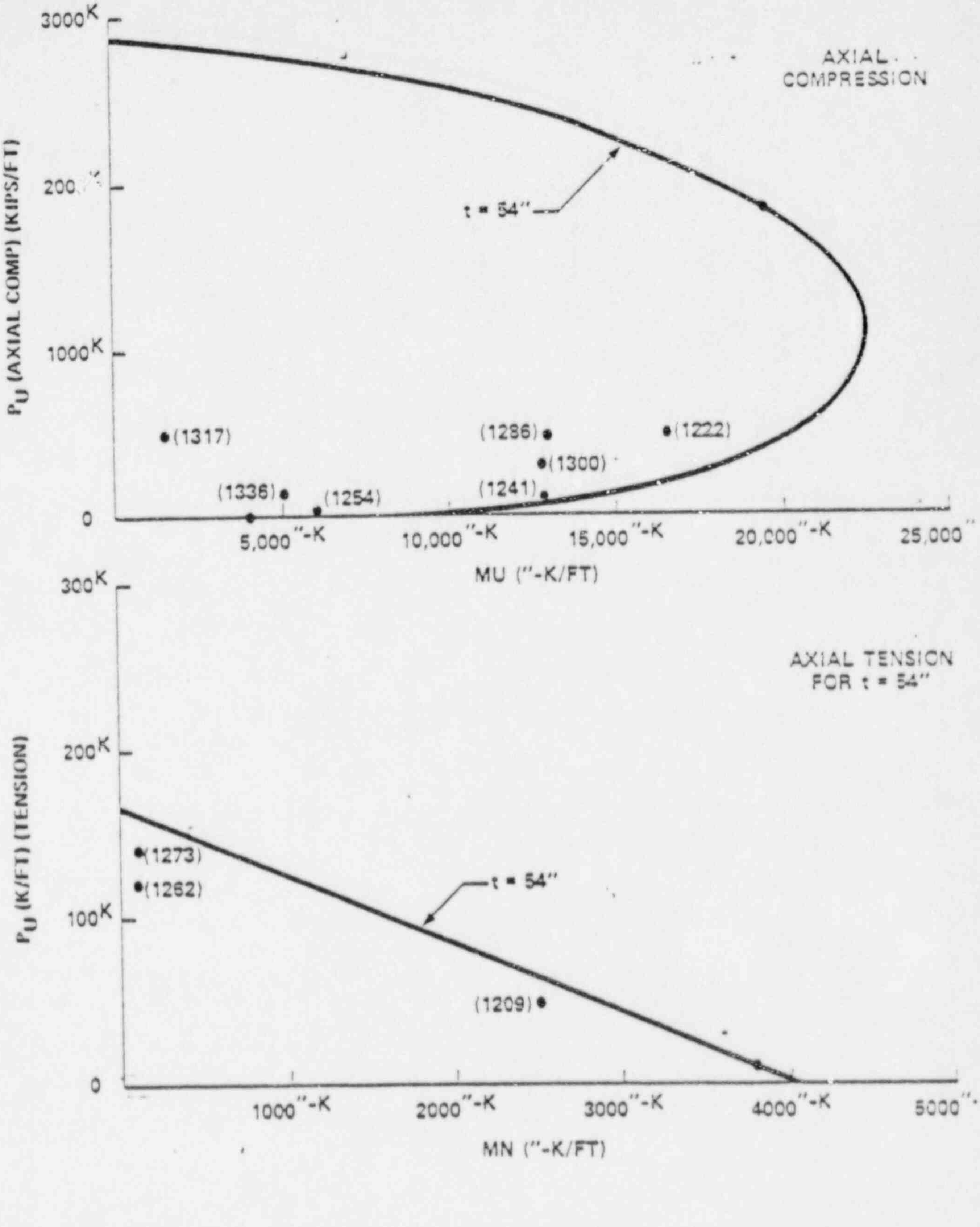

Figure 23. Primary shield wall interaction diagrams.

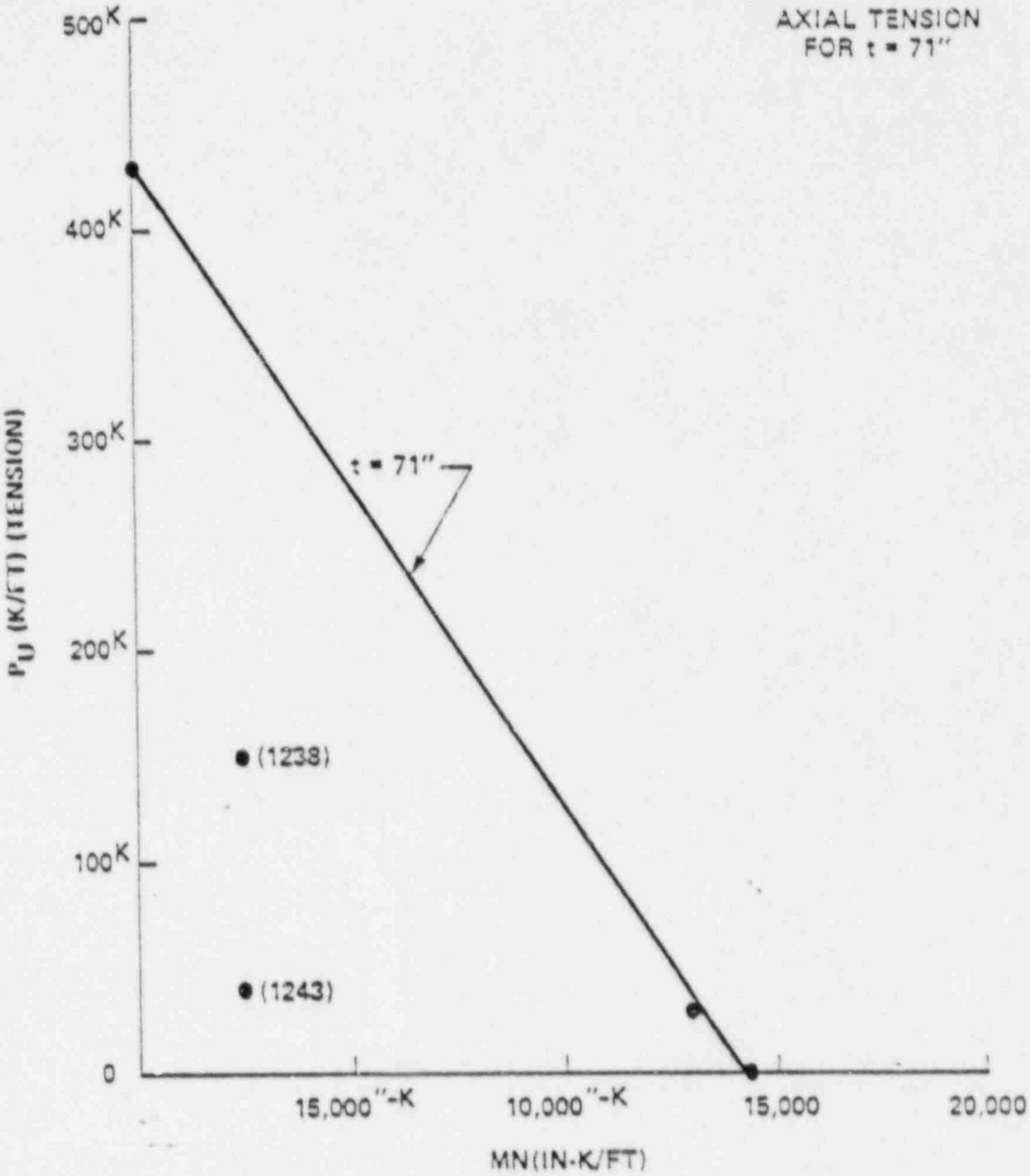

Figure 23. (Continued).

TABLE 1: MILLSTONE 2 STRUCTURAL RESPONSE AND STRESS SUMMARY

Component	Location	Calculated Value	Allowable Value	Percent Margin	Basis of Allowable
RPV support	Cold leg	6452 kips	6160 kips	- 5	ASME Code, Appendix F, 0.7 x Instability load
	Hot leg	7897 kips	7700 kips	- 3	
Steam generator supports	Pads	316 kips	9850 kips	+97	111% of design load
	Bolts	1055 kips	1856 kips	+43	Yield strength
	Lower stop	4531 kips	6810 kips	+33	111% of design load
	Lower keys	1582 kips	5610 kips	+72	111% of design load
	Upper keys	479 kips	1910 kips	+75	111% of design load
	Snubbers	1909 kips	3500 kips	+45	Test data
Primary coolant piping	RPV nozzles				
	Inlet	56040 in-kips	78965 in-kips	+29	ASME Code, Appendix F
	Outlet	144200 in-kips	279340 in-kips	+48	
	SG nozzles				
	Inlet	32120 in-kips	196715 in-kips	+84	
	Outlet	47590 in-kips	78965 in-kips	+40	
	RCP nozzles				
	Discharge	98790 in-kips	96810 in-kips	- 2	
	Suction	48860 in-kips	78965 in-kips	+38	
Reactor internals	Core support barrel				
	Lower flange	--	--	+ 2	ASME Code, Appendix F
	Lower support structure				
	Core support plate	--	--	+ 6	
	Upper guide structure				
	CEA shroud	--	--	+ 1	
	Grid beams	--	--	+ 2	
Fuel assemblies	End fittings				
	Castings	23.0 ksi	37.1 ksi	+38	ASME Code, Appendix F
	Posts	28.9 ksi	42.2 ksi	+32	
	Fuel rods	33.5 ksi	45.0 ksi	+26	
	Spacer grid				
	one-sided	5.495 kips	3.745 kips	-47	Test data
	through-grid	2.647 kips	2.345 kips	-13	Test data
CEDM	Nozzle	170 in-kips	203 in-kips	+16	ASME Code, Appendix F, 0.7 x Instability load
ECCS piping	Elbow	3300 in-kips	3850 in-kips	+14	ASME Code, Appendix F, 0.7 x Instability load
	Straight	5800 in-kips	7000 in-kips	+17	
Biological shield wall		For results see Figure 23			Interaction Diagrams

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CONCLUSION

In conclusion, there is reasonable evidence that the Millstone Unit 2 reactor systems would withstand the effects of an asymmetric LOCA event and that the reactor could be brought to a cold shutdown condition safely.

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