# APPENDIX 3.6A

## NONLINEAR PIPE WHIP ANALYSES - EMERGENCY CONDENSER PIPING INSIDE CONTAINMENT

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK
NUCLEAR GENERATING STATION

# NONLINEAR PIPE WHIP ANALYSES OF THE EMERGENCY CONDENSER PIPING INSIDE CONTAINMENT

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# 1.0 INTRODUCTION

This report, prepared by EDS Nuclear Inc. (EDS), for Jersey Central Power & Light Company, describes the analyses performed to determine the dynamic response of the inside containment portion of the Emergency Condenser piping system of the Oyster Creek Nuclear Generating Station following pipe rupture, and the selection of a rupture restraint system to prevent excessive pipe whip.

The purpose of these analyses was to determine the dynamic response and characteristics which influence the design of the rupture restraints for a "typical" inside containment piping system, and to evaluate the feasibility of providing rupture restraints for all high-energy piping systems inside the drywell, based on the results of the analyses and postulated break locations.

The Emergency Condenser piping system was selected for study, as it is considered to be "typical" of the inside containment piping in the following respects:

a. The piping is of moderate size, and operating conditions are representative of the high-energy systems in the plant.

b. The magnitude of blowdown forces associated with rupture is average for the systems.

c. Congestion and interference with adjacent systems are typical of the inside containment portion of the plant.

In addition, the Emergency Condenser piping is located close to potential load-carrying structures for attachment of rupture restraints. Hence, the design difficulties for the restraint system are less severe than for the majority of high-energy systems inside containment.

A mathematical model of the Emergency Condenser system from the reactor vessel connection to the postulated break location was constructed for use in the dynamic response analyses. The model consisted of lumped masses connected by massless pipe elements having nonlinear, inelastic structural properties and a nonlinear, inelastic spring representing the rupture restraint. Dynamic time history analyses were performed for shock loading resulting from an instantaneous circumferential break location. The analyses included the effects of pipe impace on the rupture restraint, generalized yield behavior of the piping and kinematic strain-hardening of the piping and restraint.

The blowdown force time histories were calculated at selected points in the piping system, and applied as time-varying concentrated loads in the nonlinear structural response analyses. Total fluid force time histories were calculated from the time-varying pressure and momentum conditions following rupture, and included the effects of pipe friction and two-phase compressible fluid flow.

The results of the structural response analyses indicated that a one-inch diameter U-bar of the type developed and tested by the General Electric Company, is optimum for the break case considered in the studies. Both the response of the piping system and rupture restraint are within acceptable limits, and the maximum energy-absorbing capabilities of the restraint are utilized. The maximum displacements at all points in the piping system are sufficiently small to ensure that no interference with adjacent systems will occur from pipe whip.

It should be noted that the rupture restraint system and design details developed from the results of the analyses are applicable to the particular piping system for a specific break case, and may not adequately restrain the piping for other potential break locations. In general, it is considered that similar rupture restraint systems will be necessary for other break cases in the Emergency Condenser piping and for the other high-energy piping systems in the plant.

It is concluded that, although the rupture restraint and support frame for the case considered in this study are structurally feasible, the structure which is required is massive and a number of potential problems exist in providing such restraints for all high-energy piping systems. These problems include space limitations, high radiation levels, and congestion which may result in severe installation penalties, including high costs, long plant down time, and "burn up" of plant maintenance personnel. In most cases, it will be virtually impossible to provide such restraints due to the severity of the problems discussed above and/or due to the magnitude of the loads imposed by lines larger than the one evaluated.

## 2.0 SYSTEM DESCRIPTION

The Emergency Condenser system prevents overheating of the reactor fuel in the event that reactor feedwater capability is lost and heat removal systems which require A.C. electrical power for operation are not available. In the event of reactor isolation and scram from power operation, the isolation condenser will remove decay heat without requiring an external power supply. The system has the capacity for the removal of decay heat without addition of water to the secondary side until pumping power can be restored for makeup.

The Emergency Condenser system operates by natural circulation without the need for driving power other than the D.C. system used to place the system in operation. The system consists of two condensers and associated piping and valves. Piping and valves are connected to the reactor vessel in such a fashion as to allow each to function independently. Each condenser consists of two tube bundles immersed in a large water storage tank.

In operation of the system, steam flows from the reactor, through the tubes of the heat exchangers; after condensing, it returns by gravity to the reactor. The condensers are located high in the reactor building to facilitate natural circulation. During operation, the water on the shell side of the condensers will boil and vent to the atmosphere while condensing steam inside the tube bundles.

The portion of the system located inside containment is subdivided into two sub-systems, each consisting of a 10-inch return line and a 10-inch supply line with associated valving. The analysis has evaluated the NE-5 supply line of loop A. This portion of the system consists of a single, 10-inch diameter, Schedule 80, stainless steel pipe from the reactor vessel connection to the drywell penetration. The geometry and location of the system with respect to surrounding systems and structures is shown in Figure 1.

# 3.0 METHODS OF ANALYSIS

The pipe whip analyses of the Emergency Condenser piping system proceeded in two stages: the first involved computing the fluid pressure and momentum transients throughout the piping system following rupture, and developing time histories of loading on the piping components; the second involved the construction of a mathematical model of the system and determining the structural response of the model to the time-varying loadings developed in the first stage. Following the structural response evaluation, design details were developed for a restraint structure subjected to the rupture restraint force time histories.

Nonlinear analysis techniques were used in the structural response evaluation, as the requirement that the response of the system remain in the elastic range may lead to unnecessarily conservative design requirements. The use of nonlinear, inelastic analysis procedures enables consideration of the large energy-absorbing capabilities of the system in the inelastic range, while still maintaining a conservative design basis. Thus, it was considered that the use of nonlinear dynamic analysis techniques for investigating structural adequacy was more appropriate for evaluating the feasibility of providing rupture restraints for the Oyster Creek plant.

The general analytical procedure, fluid transient analysis procedure and structural response analysis procedure are discussed in separate sections below.

## 3.1 General Analytical Procedure

The general procedure for determining an acceptable rupture restraint design may be summarized as follows:

a. Identify the potential break location(s) in the piping system and construct a mathematical model for the structural response analyses which adequately represents the dynamic and potential nonlinear stiffness characteristics of the system. The methods used in determining the potential break location(s) are discussed in Reference 2.01, and the mathematical model for these studies is discussed extensively in Section 4.4 below.

b. Determine pressure and momentum time histories following pipe rupture for the postulated break case(s) determined in a., above. From the pressure and momentum transients throughout the piping, calculate total force time histories at all direction changes in the system. In general, the force time histories vary both with location in the system, and with location of the postulated break. The development of the shock loading on the system is discussed in Section 3.2 below.

c. Select rupture restraint locations based on the magnitude and direction of the jet loads at the break location, and estimate the structural characteristics of the restraints. The strength properties of the restraints should be at least sufficient to prevent continued yielding of the restraints under steady-state blowdown forces.

d. Perform nonlinear, inelastic, time history analyses on the mathematical model of a. above, for the shock loads determined in b., using the restraint properties and locations determined in c. From the results of the response analyses, check the acceptance criteria for overall system response. The criteria used in these studies for acceptability of restraint locations, restraint properties and system response are discussed in Section 4.1 below.

e. Develop final restraint/support system design details if the acceptance criteria are met. If not, repeat Steps c. and d., above.

The procedure outlined above is iterative, as a number of criteria may influence the design and location of restraints. It should be noted that the structural properties of the restraint, support system and load-carrying structure may be included in the analyses, if it is considered that significant dynamic response coupling may exist between the piping and the support system, or if inelastic design of the load-carrying structure is warranted. However, the maximum dynamic forces in the system generally occur immediately after impact, and it is considered that these force magnitudes will not be sensitive to the properties of the load-carrying structure.

The above analytical procedure is summarized in Figure 2. Descriptions of the analyses for determining blowdown force time histories and system response are presented in Sections 4.2 and 4.5 below.

### 3.2 Evaluation of Shock Loading

The time histories of pressure and momentum transients throughout the piping following a circumferential break were evaluated using Program RELAP3, developed by Idaho Nuclear Corporation (Reference 3.01), for the calculation of transient flow conditions in primary reactor piping systems of arbitrary geometry. A mathematical model of the reactor vessel and the inside containment portion of the Emergency Condenser piping was constructed. The model consisted of a series of constant control volumes connected by flow paths monitoring the thermodynamic conditions of the steam throughout the system at uniform time intervals. Pipe friction effects were also included in the mathematical model.

Time history analyses for a postulated break case were performed by direct integration of the equations of momentum balance, mass balance and energy balance, using a two-phase equation of state available in the program. Total force time histories were computed from the time histories of pressure and momentum at the break location, and at each angle change of the piping system. These force time histories were subsequently used as the applied shock loads in the structural response analyses.

Discussions of the blowdown analyses and results are presented in Sections 4.2 and 4.3 below.

### 3.3 Evaluation of Structural Response

The structural response effects of shock loading on the Emergency Condenser piping were evaluated using EDS Program PWHIP. The program may be used to analyze three-dimensional piping systems of arbitrary geometry subjected to time-varying forces. The effects of inelastic pipe response, inelastic restraint behavior and initial clearances between the pipe and restraints were included in the analyses.

A mathematical model of the piping system was constructed for response computation. The model consisted of lumped masses connected by massless elements, with a sufficient number of mass points specified to enable an accurate evaluation of the dynamic response to be obtained. Nonlinear force-deformation characteristics were specified for the pipe and rupture restraint, as well as the initial clearance between the pipe wall and restraint. Reduced stiffness properties were included at all elbows of the system, in accordance with the procedures of the ANSI B31.1 Power Piping Code.

The structural properties of the complete system were assembled by the Direct Stiffness Method, and the time histories of response were computed by direct integration of the coupled equilibrium equations. The system stiffness properties were modified at each change in yield status of the structure, with corrective forces applied at the end of each time step to ensure that dynamic equilibrium was continually satisfied.

Shock loading of the system was represented by sets of time-varying concentrated forces applied at the break location and all direction changes of the piping system, as discussed in Section 3.2, above. Time histories of member forces throughout the system were computed and the yield status at all points of the piping was checked by the resultant-moment criterion discussed in Section 4.1.

## 4.0 DISCUSSION OF ANALYSES

In general, rupture-induced shock loads may be imposed on the piping system as a result of both jet forces at the break location and time-varying unbalanced forces on straight runs of piping associated with sudden pressure losses. Pressure variations, resulting in changing flow conditions, will propagate through the fluid until steady-state conditions are established. The transient flow conditions, and associated force time histories may be quite complex, and the use of simplified methods for evaluating blowdown forces can lead to unnecessarily conservative results. The use of a more detailed analysis technique was considered appropriate to the degree of refinement of the nonlinear structural response analyses.

The analyses and restraint design were performed for a "typical" pipe break case, and are not necessarily representative of "worst case" conditions in the plant. "Typical" conditions were chosen in order to obtain a more realistic assessment of the feasibility of providing rupture restraints for all high-energy piping systems in the plant. The Emergency Condenser piping was selected for study, and was considered to be representative of average inside-containment conditions for the following reasons:

a. The pipe is of moderate size and the magnitude of the blowdown forces is average for the systems.

b. The major portion of the pipe route is reasonably close to potential load-carrying structures, and does not present unusual support problems.

c. Interference and congestion in the area adjacent to the selected system are typical of the inside-containment portion of the plant.

In addition, the postulated break case selected for study is considered to represent a less restrictive design problem than for the majority of potential break locations inside containment, with respect to restraint orientation and access to load-carrying structures.

The assumptions and reference information used in the analyses, evaluation of blowdown forces, description of the mathematical model, and analyses performed are presented in separate sections below.

## 4.1 Assumptions and Reference Information

The piping geometry, pipe schedules and material types were obtained from References 1.01.03 through 1.01.12 and Reference 2.01. Both the reactor vessel penetration and drywell penetration were assumed to be full anchors. Constant support hangers were assumed to be inactive in the response analysis.

The assumptions and criteria used in evaluating the acceptability of restraints and the assumptions necessary for defining the nonlinear material stiffness characteristics are discussed separately below.

### 4.1.1 Analysis Acceptance Criteria

The sizes and locations of rupture restraints on the piping system may depend on both the response of the piping and of the restraint itself. In general, acceptable forces and strains in the restraints are not sufficient to ensure satisfactory response of the complete system. The following criteria were used to evaluate acceptability of response:

a. The maximum strains in the restraints and all points in the piping were not allowed to exceed 50 percent of ultimate strain for the appropriate materials.

b. Yielding of the pipe at rupture restraint locations was not permitted at any time during blowdown.

c. Continued yielding of the restraints and piping during the steady-state portion of blowdown was not permitted. (It should be noted that this does not preclude the development of permanent deformations in the system, but only requires that the system respond elastically under steady-state loads).

d. The displacements which occur during blowdown are not sufficient to cause interference with adjacent systems.

Assumption a., above, corresponds to accepted design procedures, permitting a sufficient factor of safety against failure of the restraints or piping. Yielding of the pipe at the restraint locations is not permitted (Assumption b.) in order to prevent excessive pipe rotation about the restraint, and to ensure that local yielding or buckling of the pipe wall is not sufficient to

cause a large change in the effective cross section geometry of the pipe. As steady-state blowdown forces are achieved, the entire system is required to return to the elastic range (Assumption c.), in order to prevent continuous long-term yielding of any components, which could result in secondary material failures. Assumption d., ensures that pipe rupture does not cause failure or loss of function of surrounding systems due to pipe whip impact.

### 4.1.2 Development of Material Properties

The Emergency Condenser piping consists of A312 TP316 stainless steel (Reference 2.01), and it was assumed, for purposes of analysis, that the rupture restraint is composed of TP304 stainless steel, in accordance with Reference 4.01. A description of the rupture restraint is presented in Section 4.4 below.

The nonlinear stress-strain properties for the pipe material were developed from data obtained from References 4.01 and 5.01, and are shown in Figure 4 . The strain-energy absorbing capacity of stainless steel at 0.1 percent, 25 percent, and ultimate strains were taken as 30 lbs./in.$^3$, 15,000 lbs./in.$^3$ and 30,000 lbs./in.$^3$, respectively (Reference 4.01). The elastic limit and ultimate strength of the material were assumed to be 30,000 psi and 70,000 psi, respectively (Reference 5.01). The above criteria were used to construct the idealized curve of Figure 4. This information provided the basis for defining the force-deformation properties of the restraint and the moment-curvature relationships for the pipe, for use in the nonlinear dynamic response analyses.

The ultimate strain for stainless steel was assumed to be 40-50 percent for both the piping and rupture restraint. This requires that the maximum strains in the system induced by shock loading be equal to or less than 20-25 percent, in accordance with Assumption a. of Section 4.1.1.

The material model for both piping and rupture restraints included kinematic strain-hardening behavior. At any point in the system, unloading from the inelastic region occurs along the elastic line for the appropriate material.

Yielding of the pipe was defined in terms of the two principal bending moments and torsional moment at each time step, as follows:

$$\sqrt{\left(\frac{M_1}{M_{1Y}}\right)^2 + \left(\frac{M_2}{M_{2Y}}\right)^2 + \left(\frac{M_T}{M_{TY}}\right)^2} \geq 1$$

in which $M_{1Y}$, $M_{2Y}$ and $M_{TY}$ are the two principal bending moments and torsional moment, respectively, for initial yielding in the outer fibers of a particular cross section. The above yield criterion permits consideration of nonlinear behavior under complex dynamic response of the system, as yielding may occur for general biaxial bending and torsion.

## 4.2 Discussion of Blowdown Analyses

The typical circumferential break case selected for study was obtained from Reference 2.01. The postulated break was assumed to occur at the end of elbow FW-5561 at Elevation 90 feet. For this break location, the support structure for the rupture restraint may be attached to the top of the biological shield wall.

The break was assumed to occur instantaneously over the entire pipe circumference. No flow restrictions on discharge at the break were considered, other than those occurring from the thermodynamic behavior of the fluid itself. The initial conditions of the steam prior to pipe rupture were assumed to be those of a saturated vapor at 1,100 psia.

Steady-state flow conditions following pipe rupture and initiation of blowdown were assumed to be controlled by "choking" conditions at the break location. The limiting mass flow from the break was defined by the two-phase flow model of Reference 6.01. The limiting steady-state flow conditions at all points in the system upstream of the break are governed by the conditions at the break and were automatically included in the calculation procedure.

The reactor vessel was modeled as a steam reservoir, with the fluid assumed to be essentially static. As neither the initial volume of steam in the reactor, or the rate of steam generation following rupture were accurately known, two cases were considered for analysis, in order to determine the sensitivity of blowdown force time histories to the assumed initial volume of steam:

The first case considered the reactor vessel to be a finite reservoir containing 5,000 cubic feet of saturated vapor at 1,100 psia. This was considered to be representative of a lower bound volume of steam. The second case considered the reactor vessel as an essentially infinite reservoir containing saturated vapor at 1,100 psia. The results of these analyses, and a discussion of the typical blowdown phenomena are presented in the next section.

## 4.3 Results of Blowdown Analyses

Analyses were performed to determine time histories of pressure and momentum throughout the piping following an instantaneous circumferential break. The pressure and momentum transients were then combined to produce total force time histories in the system. Total force time histories at two locations in the piping are presented in Figure 3, to illustrate typical blowdown behavior. Results are shown for the force time history at the elbow adjacent to the break and at the elbow adjacent to the reactor vessel connection.

The results indicate that steady-state pressures and flow rates are achieved rapidly near the break location following pipe rupture. The full mass flow rate is developed within approximately ten milliseconds. The subsequent blowdown forces remain nearly constant at approximately 65 kips, and are not significantly affected by the propagation of pressure transients through the system. It should be noted that the 65-kip steady-state blowdown force agrees closely with the theoretical result for choked flow of an ideal fluid, under the pressure and enthalpy conditions near the break. As seen from the figure, the total force first decreases, due to the sudden loss of pressure, and subsequently increases as the velocity, mass flow rate and momentum increase to the steady-state values.

For the location adjacent to the reactor vessel, the total force remains constant until the pressure loss resulting from the pipe rupture propagates through the piping and is reflected from the reactor reservoir as a pressure increase. As steady-state flow is established, the total force subsequently decreases to 65 kips at all points in the piping.

The results of the blowdown analyses for the finite and infinite reservoir cases indicated that the assumed initial steam volume does not affect the total force time histories during the portion of the blowdown which causes the maximum dynamic response of the piping. The results of the two cases were identical for the first few seconds of blowdown. This is substantially longer than the time required to achieve steady-state flow and maximum dynamic response of the piping.

## 4.4 Mathematical Model

A mathematical model of the Emergency Condenser system for use in the structural response analyses was constructed consisting of lumped masses connected by massless pipe elements. The number and locations of masses were selected to ensure that an accurate representation of the dynamic response would be obtained. The model of the system is shown in Figure 5, and the geometry, pipe schedules and material types were obtained from References 1.01.03 through 1.01.12 and Reference 2.01.

The structural properties of the system included the effects of bending, shear, torsional and axial deformations of the piping. At all elbows in the model, the nonlinear bending stiffnesses were reduced according to ANSI B31.1 Power Piping Code specifications. For purposes of analysis the connection of the piping to the reactor vessel was assumed to be a full anchor, and constant support hangers were assumed to be inactive.

The rupture restraint chosen for use in the response analyses was of the U-bar type, developed and tested by General Electric Company (Reference 4.01), shown schematically in Figure 6. This restraint type was selected because its structural properties are well-defined, and relatively extensive test data are available for confirming its effectiveness in preventing excessive pipe whip. The following U-bar geometry was chosen for the dynamic response analyses:

a. A bar diameter of one inch.

b. A total undeformed bar length of 52 inches. (Refer to Figure 6)

c. An initial clearance between the pipe wall and restraint of two inches.

It should be noted from b. and c. above, that the total effective clearance between pipe wall and restraint is 3.1 inches, as it was assumed in the analyses that the restraint does not exert significant forces on the pipe while deforming from the initial contact with the pipe to the fully-developed contact length shown in Figure 6.c. As full contact is developed, the force-extension characteristics of the U-bar with the above dimensions is shown in Figure 7.

The pipe insulation thickness was neglected in selecting the initial clearance, as it was assumed that the insulation material will offer negligible resistance to pipe movement. The insulation properties and "compressed" thickness can be considered in the analyses, but for most insulation types it was considered that these will have a negligible effect on the assumed clearance and dynamic response of the system. In general, it is desirable to maintain the smallest

restraint clearances which will permit free movement of the pipe under other types of loading, in order to minimize the impact forces caused by acceleration of the whipping pipe through the restraint gap.

## 4.5 Analyses Performed

Dynamic, nonlinear, time history analyses were performed on the mathematical model of Figure 5, using the blowdown force time histories discussed in Section 4.3 above. The final rupture restraint location is indicated in Figure 5 and the final restraint dimensions and properties used in the analyses are discussed in Section 4.4. The analyses were performed by direct integration of the dynamic equilibrium equations for the system. The time step for integration was chosen to be sufficiently small to accurately resolve response frequencies as high as 500 cps, in order to obtain both the gross response of the system in the low-frequency range and the high-frequency axial response of straight runs of piping associated with the time-varying unbalanced axial forces.

Several time history analyses were performed to determine the influence of material damping and the effects of restraint bar size on the dynamic response. The analysis cases may be summarized as follows:

Case 1: One-inch diameter U-bar, with "lower bound" material damping.

Case 2: One-inch diameter U-bar, with "upper bound" material damping.

Case 3: One and one-quarter-inch diameter U-bar, with "upper bound" material damping.

Selected results from each of these cases are discussed in Section 5.0, below.

Time history analyses were also performed for restraint locations at lower elevations on the vertical riser, in order to minimize the support structure necessary for attachment to the top of the biological shield wall. However, it was found that the allowable pipe strains and deflections discussed in Section 4.1.1 were exceeded, and that it was necessary to locate the restraint as close to the break location as possible. Hence, no results for these cases are presented.

Following the dynamic response analyses and evaluation of restraint size and location, design details were developed for a support structure adequate to withstand the maximum dynamic loads,and transmit the forces to the biological shield wall. It was required that the support structure remain relatively rigid under the dynamic loading imposed by the piping and rupture restraint, in order to minimize additional pipe movement resulting from displacement of the restraint structure. The development of the details of the restraint structure, together with the results of the nonlinear response analyses, are discussed in Section 5.0.

# 5.0 DISCUSSION OF RESULTS

The dynamic response analyses of the Emergency Condenser system were performed for a duration of loading sufficient to ensure that both the rapid initial blowdown force variations and the development of steady-state blowdown forces were included in the analyses. It was found that the maximum dynamic response of the system was achieved very rapidly following pipe rupture. Although steady-state structural response of the system was not achieved in the time duration of the analyses, the response at all points in the system had attenuated sufficiently to indicate convergence to steady-state levels.

Time histories of the rupture restraint strains and elongation, and of moments and displacements at selected points in the piping were obtained for the analysis cases discussed in Section 4.5. The structural response results and support structure design are discussed in Sections 5.1 and 5.2, respectively.

## 5.1 Structural Response of the System

Following pipe rupture, the free end of the pipe at the break is accelerated by the jet force of the discharging steam through the restraint gap, and impact occurs when the pipe displacement reaches the effective clearance between the pipe wall and restraint. The time required to close the restraint gap for the analysis cases considered is approximately nine (9) milliseconds.

It was found that the maximum moments and stresses occur in the piping near the reactor vessel connection. In this region of the system, yielding of the pipe occurs during the first 40 milliseconds of response, and is caused alternately by the two principal bending moments which result from both horizontal and vertical movement of the portion of the system near the break. It should be noted that the resultant moment associated with first yielding of the pipe is approximately 113 foot-kips. Time histories of the two principal bending moments at the reactor vessel connection are shown in Figure 8, for Case 1 of Section 4.5 above. Following initial yielding, the response near the reactor vessel returns to elastic levels, and all subsequent bending moment variations remain in the elastic range.

The maximum bending moment time history at the rupture restraint is shown in Figure 9 corresponding to Case 1. At this location yielding does not occur at any time during dynamic response, in accordance with Assumption b. of Section 4.1.1. As seen from the figure, the maximum moment occurs shortly after impact.

From a comparison of the moment time histories at the restraint and at the reactor vessel, it can be seen that the reactor vessel moments are not immediately affected by pipe impact, as the impact shock effects require several milliseconds to propagate through the piping. Resultant bending moment envelopes are presented in Figures 10 and 11, showing the maximum combined bending moments in the system at 15 and 35 milliseconds, respectively. Although the moments in some portions of the system exceed the yield moments for the pipe cross sections, the maximum strains in the piping remain well below the allowable value of 25 percent. However, some permanent deformation of the piping occurs in the region adjacent to the reactor vessel.

The rupture restraint force time histories for Cases 1 and 2 are shown in Figure 12. As discussed above, pipe impact occurs approximately nine milliseconds after the instantaneous break, and the elastic limit of the restraint is reached almost immediately. The peak load in the restraint of approximately 130 kips corresponds to an inelastic restraint elongation of approximately three and one-half inches. Subsequent unloading and loading cycles of the dynamic response are within the elastic range for the assumed bar size.

From a comparison of the time histories shown in Figure 12, it can be seen that the material damping properties assumed for the system do not significantly affect the maximum dynamic response. As would be expected, energy absorption in the system associated with material damping is much less significant than the energy absorption resulting from inelastic deformation. The subsequent unloading and loading cycles shown in the figure are, in general, governed by the dynamic properties of the restraint and support structure, and by the elastic material damping properties. However, the force time histories indicate that the dynamic response of the restraint following the initial inelastic deformation is not important for evaluating the adequacy of the restraint, provided the criteria discussed in Section 4. 1. 1 are satisfied.

The restraint extension time history for Case 1 is presented in Figure 13. As discussed above, the elastic extension of the restraint is achieved in a very short time interval, followed by a relatively large inelastic extension. The restraint experiences approximately three and one-half inches of permanent extension, as the subsequent unloading and loading cycles are elastic. For the U-bar dimensions assumed in the analyses, this corresponds to a maximum uniform strain of approximately 24 percent, which is within the optimum strain range for inelastic restraint design. The maximum restraint extension corresponds to a total pipe movement at the restraint of approximately six and one-half inches, as the effective clearance prior to rupture is approximately three inches.

Maximum displacements at selected points in the piping from the Case 1 analysis are presented in the table below.

MAXIMUM DISPLACEMENTS
CASE 1

| Mass Point (Figure 5) | Displacement (Inches) X | Y | Z |
|---|---|---|---|
| 2 | 0.04 | 0.08 | 0.06 |
| 5 | 0.23 | 0.92 | 0.48 |
| 8 | 0.47 | 1.10 | 0.58 |
| 11 | 1.25 | 1.19 | 0.58 |
| 14 | 2.03 | 1.39 | 0.59 |
| 17 | 2.95 | 1.37 | 0.59 |
| 20 | 4.64 | 1.19 | 0.59 |
| 22 | 5.63 | 1.24 | 0.59 |
| 24 | 6.04 | 0.77 | 0.59 |
| 26 | 6.25 | 1.31 | 0.74 |
| 29 | 7.82 | 2.38 | 1.64 |

Rupture restraint force time histories for the Case 1 and Case 3 analyses are presented in Figure 14, in order to illustrate the effect of bar diameter on the dynamic response. The one and one-quarter inch diameter bar of Case 3 is approximately 1.5 times stiffer than the one-inch diameter bar of Case 1, with a correspondingly higher elastic range. As shown by the force time histories, the stiffer restraint allows higher impact forces to be developed over a shorter time interval, as the stiff restraint permits higher elastic forces to occur, and absorbs less energy in the inelastic range. Thus, the total energy absorbed by the stiffer bar is less than that absorbed by the smaller bar, as most of the total energy absorbed in both cases is associated with inelastic deformation. A maximum strain in the restraint of 11 percent was calculated for Case 3.

From the Case 3 results, it was found that the pipe yields at the rupture restraint location following impact, permitting significant rotation of the pipe about the restraint. The stiffer restraint caused the pipe to absorb a greater proportion of the impact energy than in Case 1, and hence resulted in yielding of the pipe. As discussed in Section 4.1.1, yielding of the pipe at the

restraint is not permitted, and therefore the one and one-quarter inch U-bar is not considered to be appropriate to the system for the postulated break case.

It should be noted that, in general, the use of smaller bar diameters will result in lower restraint forces following impact, with correspondingly higher strains in the restraint. However, the feasibility of providing smaller restraints is governed by the maximum allowable strain for the material, and by the amount of pipe movement resulting from restraint elongation. A one-inch U-bar located as shown in Figure 5 is optimal for the piping system selected for study, and the postulated break location, as the maximum energy-absorbing capabilities of the restraint are utilized. Hence, it is considered that the Case 1 results represent the minimum dynamic forces which can occur during blowdown for the break location selected for study.

## 5.2 Design of Restraint Support Structure

The restraint force time history from the Case 1 analysis was used as the basis for developing design details of a structural system to transmit the maximum dynamic forces to the biological shield wall. In designing the structure, it was required that the structure remain essentially rigid with respect to the piping and rupture restraint, in order to limit any additional pipe movement resulting from deflection of the structure itself, as excessive pipe movement will cause interference with adjacent systems. Moreover, the response frequencies of the restraint structure must be relatively high compared to the frequency of the loading and unloading cycles in the rupture restraint, in order to avoid dynamic response amplification between the systems.

The restraint structure required for the above criteria is shown in Figures 15 and 16. For some systems or break cases the restraint structure may be designed for inelastic behavior, as in the case of the rupture restraints themselves. However, for most inside-containment piping systems in the Oyster Creek plant, the restrictions on pipe movement associated with congestion and interference require that the rupture restraint support structures be essentially rigid. Moreover, inelastic design of the restraint structure would be subject to essentially the same physical limitations on space and required connections to load-carrying structures. For these reasons it is considered that the restraint structure shown in Figures 15 and 16 is typical of those that would be required for the high-energy piping systems in the plant.

It should be noted that the restraint structure shown is applicable to a particular break case in the Emergency Condenser piping only. The rupture restraint and support frame for the case discussed in this study may not

provide adequate restraint of the system for other potential break locations, although similar design details may be appropriate for other potential breaks and other piping systems in the plant.

The rupture restraint and support frame developed in this study are structurally feasible. However, it should be noted that installation of the restraint system shown in Figures 15 and 16 would require permanent removal of the insulation on the reactor vessel where the restraint structure is located, as the clearance between the shield wall and vessel insulation is only seven inches. A number of potential problems exist in providing such restraints for all high-energy piping in an existing plant. These may be summarized as follows:

1. The restraint system shown is for a single break of a particular type. Similar systems will be necessary for other potential break locations in this piping system, as well as all other high-energy piping systems.

2. The construction of such a restraint system in an existing plant in most cases will require temporary removal of existing piping in the adjacent area, with subsequent problems associated with satisfactory replacement of components. In some cases, it may be necessary to reroute existing piping.

3. For most cases in which the restraint systems are attached to the shield wall, the construction of the restraints will require removal of the reactor vessel insulation. In many of these cases, the vessel insulation must be permanently removed at the restraint location.

4. The case chosen for study is regarded as "typical", and is not necessarily representative of "worst case" conditions. In most instances, it will not be possible to restrain the piping to prevent pipe whip, as there may be no load-carrying structures accessible to the restraint locations required for a particular break case. Alternatively, the required restraint system will be too large for the available space.

5. The space limitations in portions of the inside-containment area of the plant are too restrictive for the number of restraint systems required, with regard to both the space necessary for the restraints themselves and/or the space necessary for construction.

The conclusions and recommendations developed from these analyses regarding the feasibility of providing rupture restraints for the piping of the Oyster Creek plant are presented in Section 6.0, below.

# 6.0 CONCLUSIONS

The purpose of this study was to evaluate the dynamic response of a typical high-energy piping system of the Oyster Creek Nuclear Plant resulting from pipe rupture, and to evaluate the feasibility of providing rupture restraints for all high-energy piping inside containment.

The conclusions developed from this study may be summarized as follows:

1. For the system selected for study, the response of the system could be maintained within acceptable design limits by utilizing the energy absorbing capabilities of the piping and restraint. It was necessary to use nonlinear, inelastic analytical techniques for response evaluation, as the use of elastic analysis procedures will result in unacceptable design requirements. It should be noted that the use of such techniques would be necessary for the design of other restraint systems for the high-energy piping inside containment, and that the cost and level of effort required to obtain acceptable designs will be substantially higher than that associated with elastic analysis techniques.

2. By utilizing the procedures described above to restrict the magnitude of the loads developed, a restraint support structure could be designed. However, the structure required is massive, indicating that the use of such restraints as a means of protection is not practical for the following reasons:

   a. For "worst case" dynamic loads, it may not be possible to provide such restraint systems.

   b. In highly-congested portions of the drywell, the limitations on space in most areas are too restrictive for installation of similar restraint structures.

   c. For areas in which several restraint systems are required, the structural systems will interfere with each other.

   d. For certain break locations, there are no accessible load-carrying structures for attachment of the restraint systems. In the case of the larger piping systems, for example, the shield wall is the only viable load-carrying structure. The

shield wall is not accessible to the larger piping systems for some break cases.

e. The construction of such restraint systems may require temporary removal of existing piping, with subsequent problems associated with satisfactory replacement of components.

f. The radiation exposures to personnel resulting from the installation of restraints will be significant.

g. The high cost associated with the analysis and design, fabrication, installation, and the resulting plant down-time will be prohibitive.

h. The installation of restraint systems will interfere with the plant in-service inspection program to such an extent that certain required inspections may not be possible.

From the above conclusions, it is recommended that a surveillance program be instituted in lieu of the use of rupture restraints as a means of protecting against postulated pipe failures. It is recommended that rupture restraints be considered only for certain small lines, two inches and under, where load-carrying structures are readily accessible.

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# REFERENCES


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      12. (GE) 706E206, revision 1
      13. (Industrial Engineering Works), 722-P1, revision 1
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      20. (GE) 4069-4 (as built)
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   02. Correspondence

      01. Letter from R. J. Schemel (USAEC) to I. R. Finfrock (Jersey Central Power & Light Company), dated August 7, 1973, with Attachment A and Enclosure 1.

      02. Burns & Roe Calculations #157, 169, 490, 498 and 499.

2. EDS Nuclear Inc.

   01. Report to GPU Service Corporation entitled, "Oyster Creek Nuclear Plant, Pipe Whip Protection Inside Containment". Preliminary Summary Report, April, 1974.

3. Idaho Nuclear Corporation

   01. "RELAP3 - A Computer Program for Reactor Blowdown Analysis", USAEC Scientific and Technical Report, Contract AT(10-1)-1230 (TID-4500), June, 1970.

# APPENDIX

## COMPUTER PROGRAM DESCRIPTION

# PWHIP

EDS Program PWHIP is a general-purpose program for the dynamic, nonlinear, inelastic analysis of three-dimensional piping systems subjected to arbitrary time-varying forces. The program may be used to determine time histories of pipe displacements, member forces and strains, as well as restraint force and deformation time histories. The effects of inelastic pipe response, inelastic restraint behavior, and rupture restraint clearances are included in the analysis procedure.

The structure may be composed of elements of a variety of types, each having different behavior and yielding characteristics in order to permit maximum user flexibility. Restraints of different types may be specified and may include specified clearances between the pipe and restraint, as well as arbitrary orientation in space. Large displacement effects may be included to allow the direction of the restraining force to change as the piping system displaces.

The analysis is carried out by the Direct Stiffness Method, with step-by-step solution of the coupled equilibrium equations using a constant average acceleration assumption. The structure is modified each time a change in yield status of the structure occurs, and corrective forces are applied at the end of each time step to ensure that dynamic equilibrium is continually satisfied. The analysis also includes viscous damping effects. A static analysis for gravity, thermal and pressure loadings may precede the dynamic analysis. If a circumferential break (guillotine cut) is specified, any existing static forces and moments in the pipe at the cut are automatically applied as loads to the separated parts of the system.

The program has been verified for an extensive set of sample problems, including comparisons with theoretical solutions, hand calculations and experimental results, wherever possible. Response of systems within the elastic range has also been compared with benchmarked EDS piping programs for elastic analyses.

FIGURES

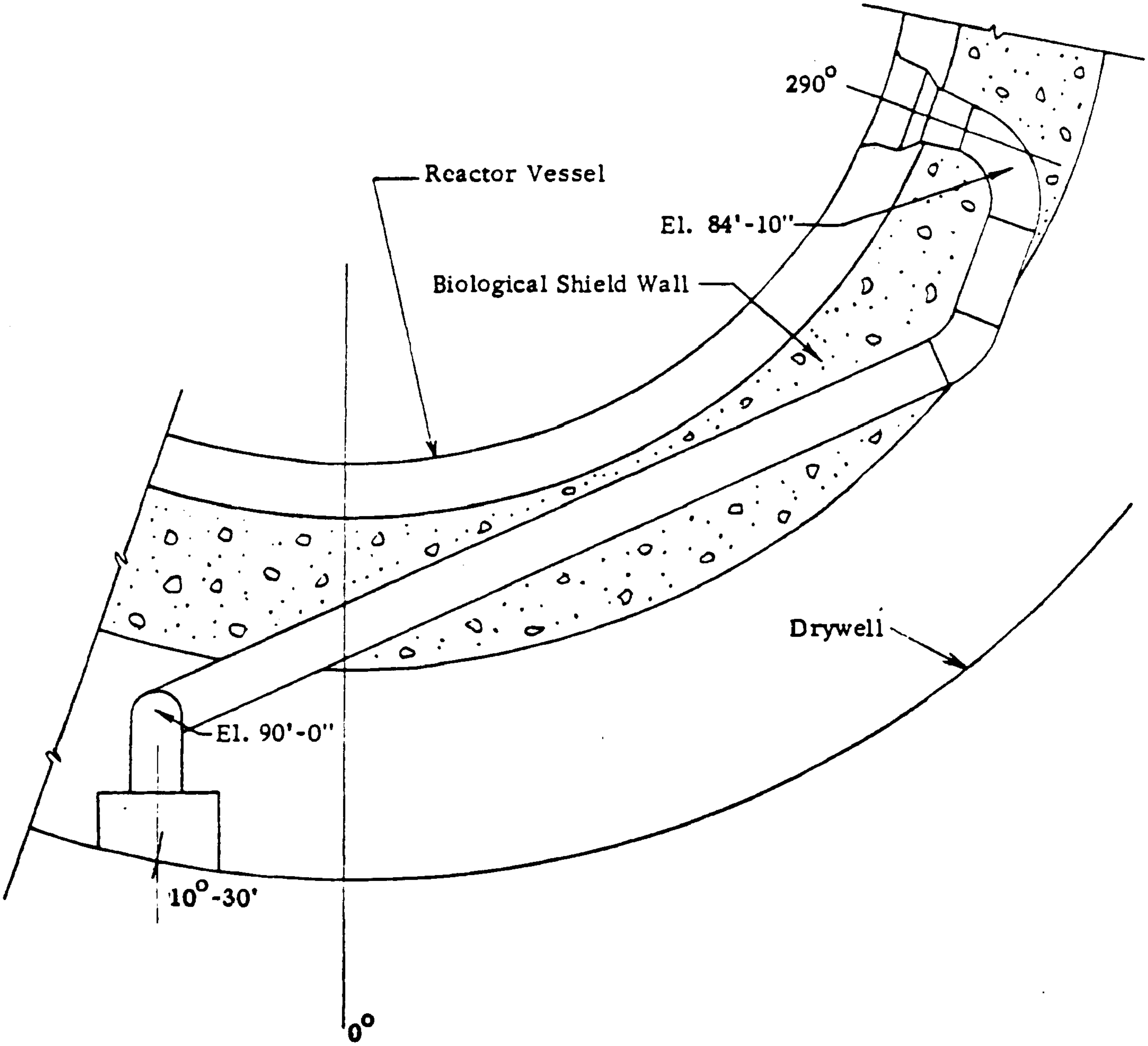


FIGURE 1: EMERGENCY CONDENSER PIPING INSIDE CONTAINMENT

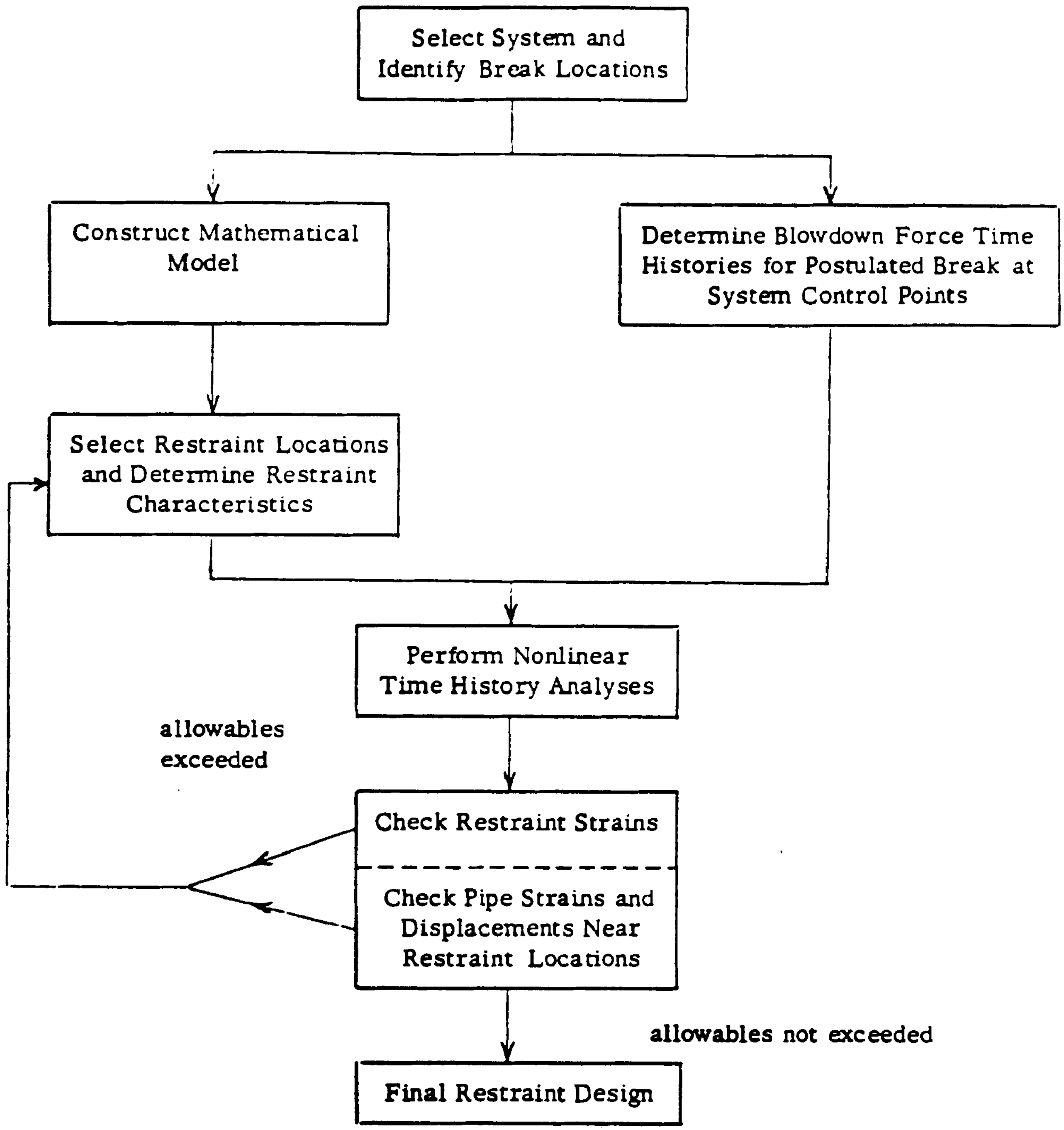


FIGURE 2: GENERAL ANALYTICAL PROCEDURE

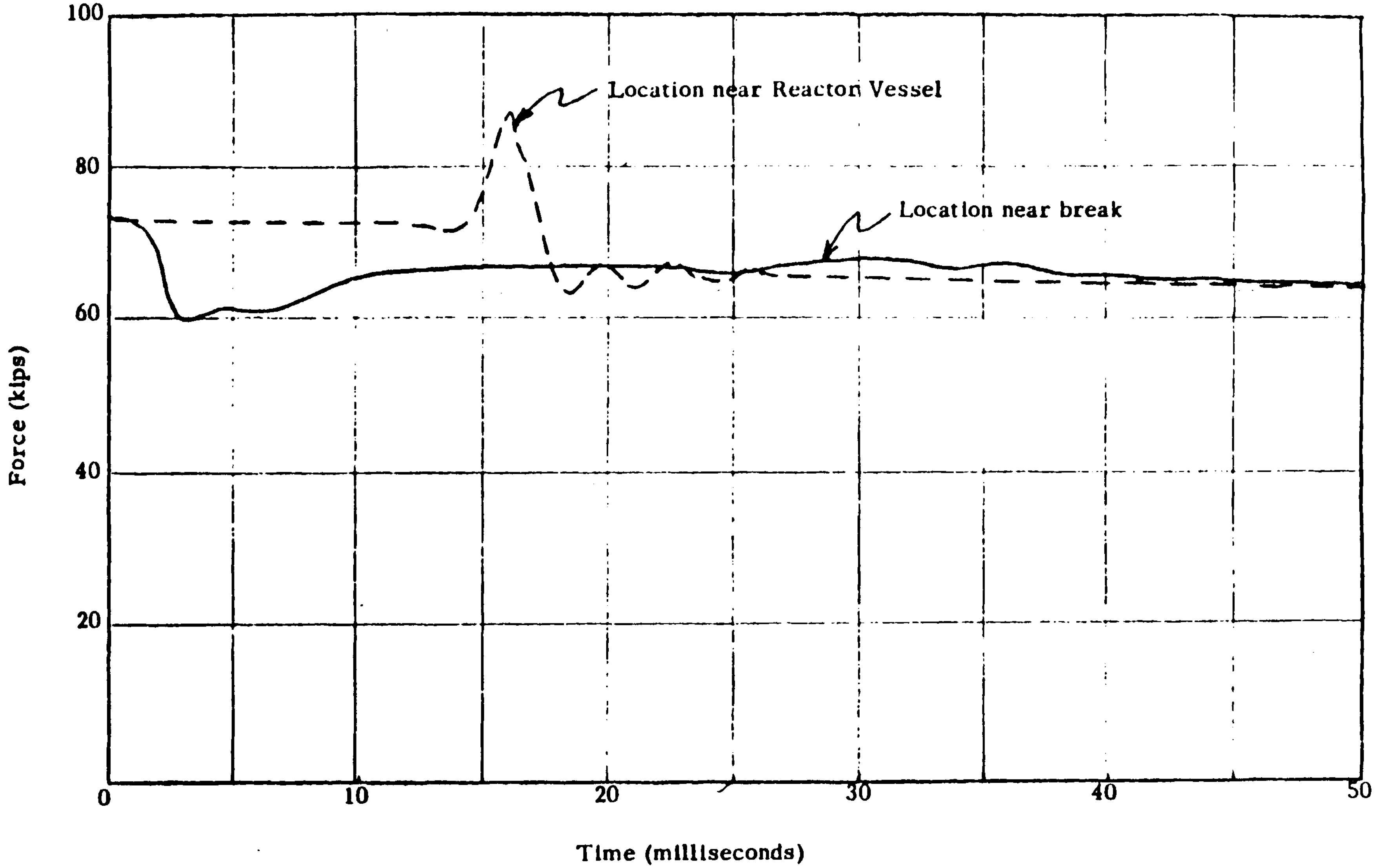


FIGURE 3: TYPICAL FLUID FORCE TIME HISTORIES FOLLOWING RUPTURE

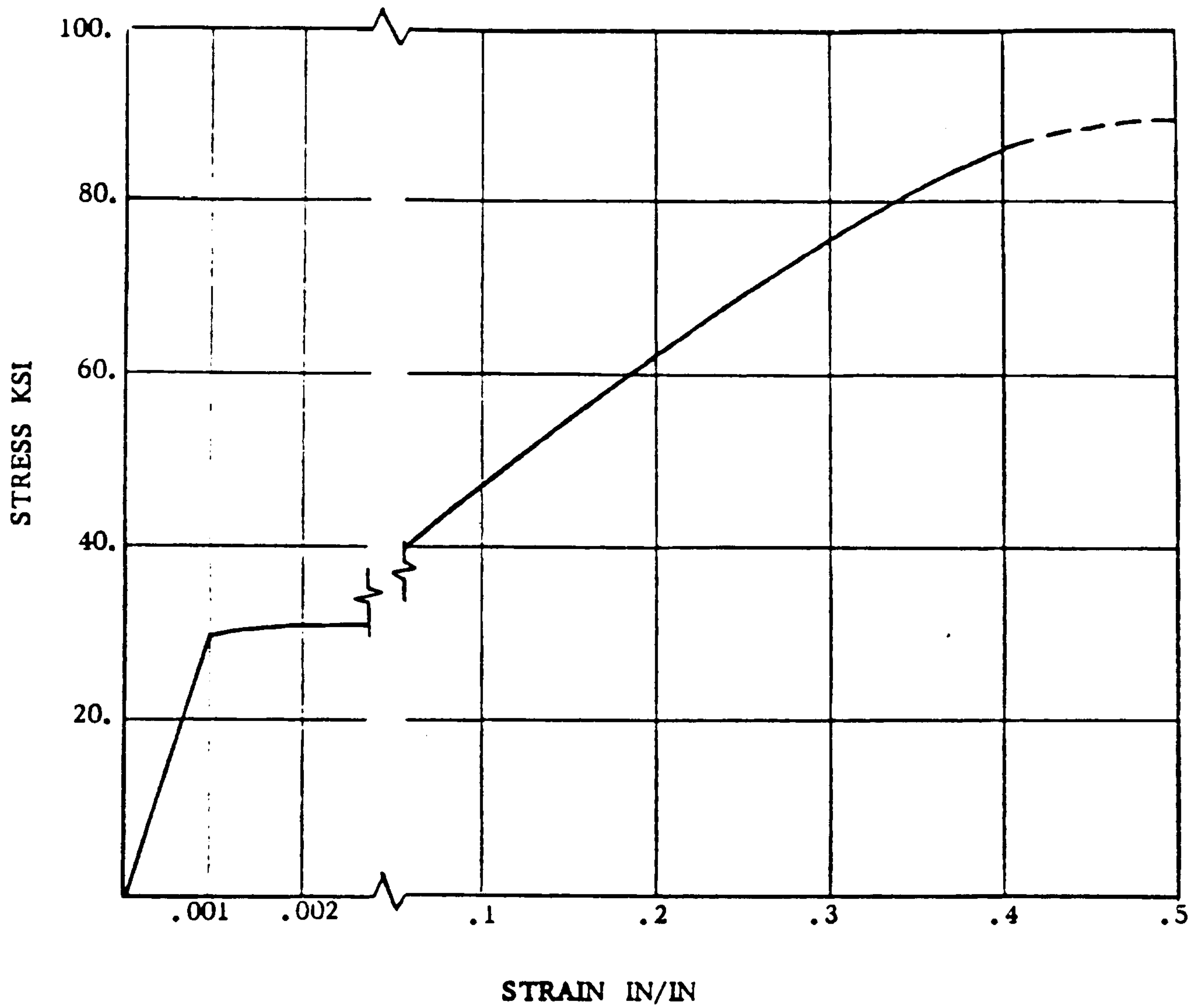


FIGURE 4: IDEALIZED STRESS-STRAIN PROPERTIES FOR STAINLESS STEEL

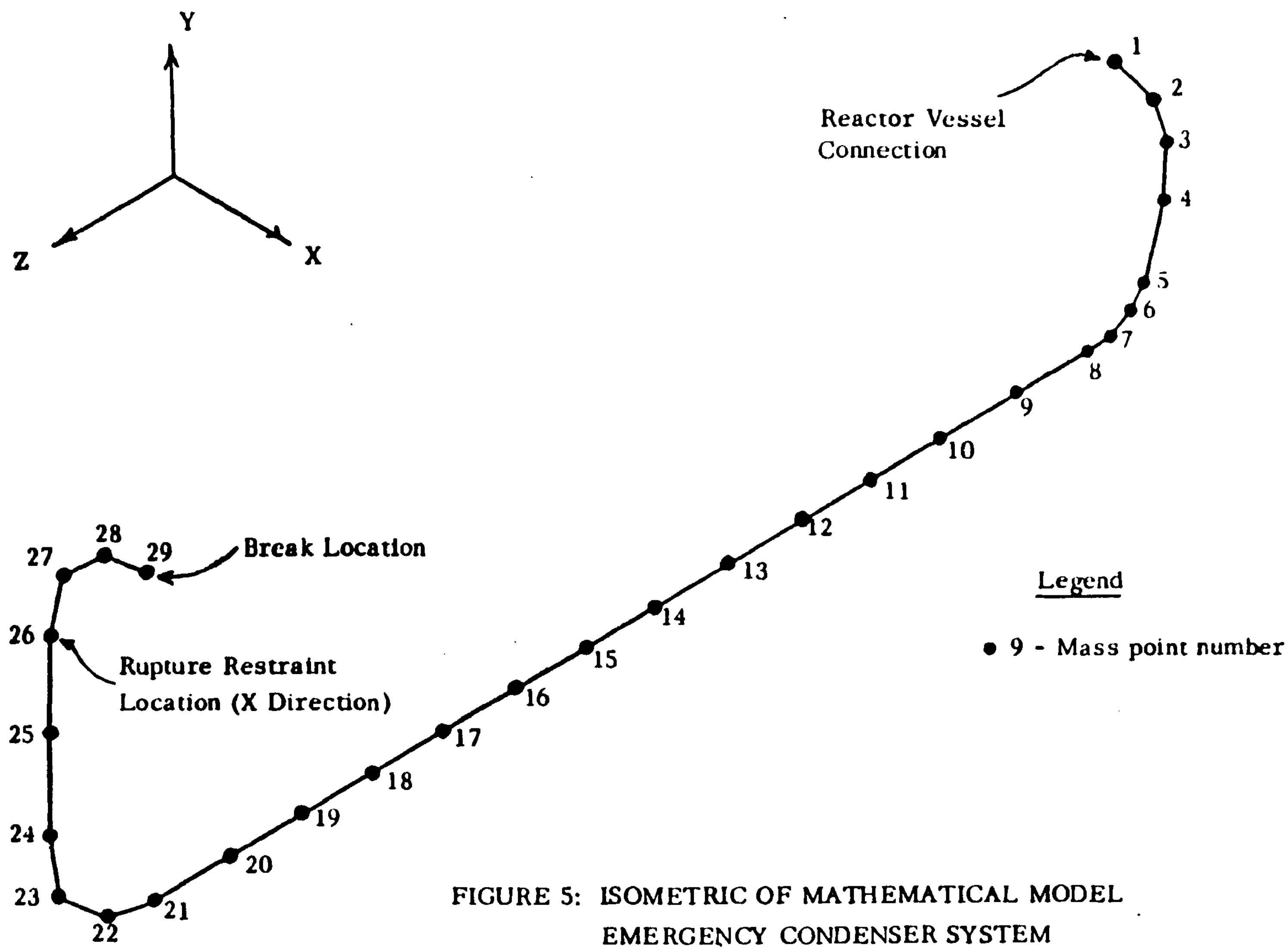


FIGURE 5: ISOMETRIC OF MATHEMATICAL MODEL
EMERGENCY CONDENSER SYSTEM
REACTOR TO NEO 1-A

| | |
|---|---|
| GPU SERVICE CORPORATION | OYSTER CREEK NUCLEAR PLANT<br>NONLINEAR PIPE WHIP ANALYSES |

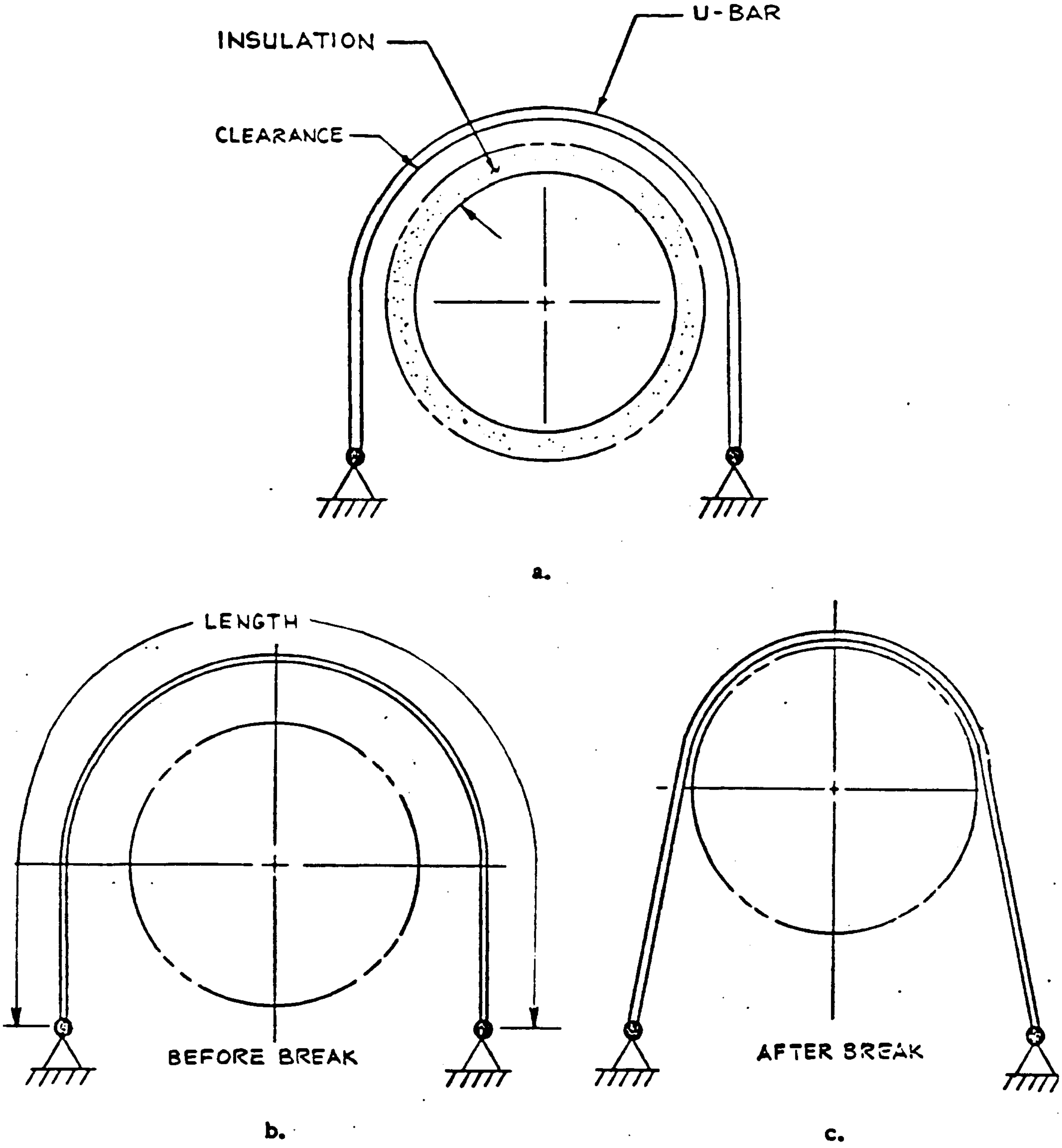


FIGURE 6: U-BAR RESTRAINT

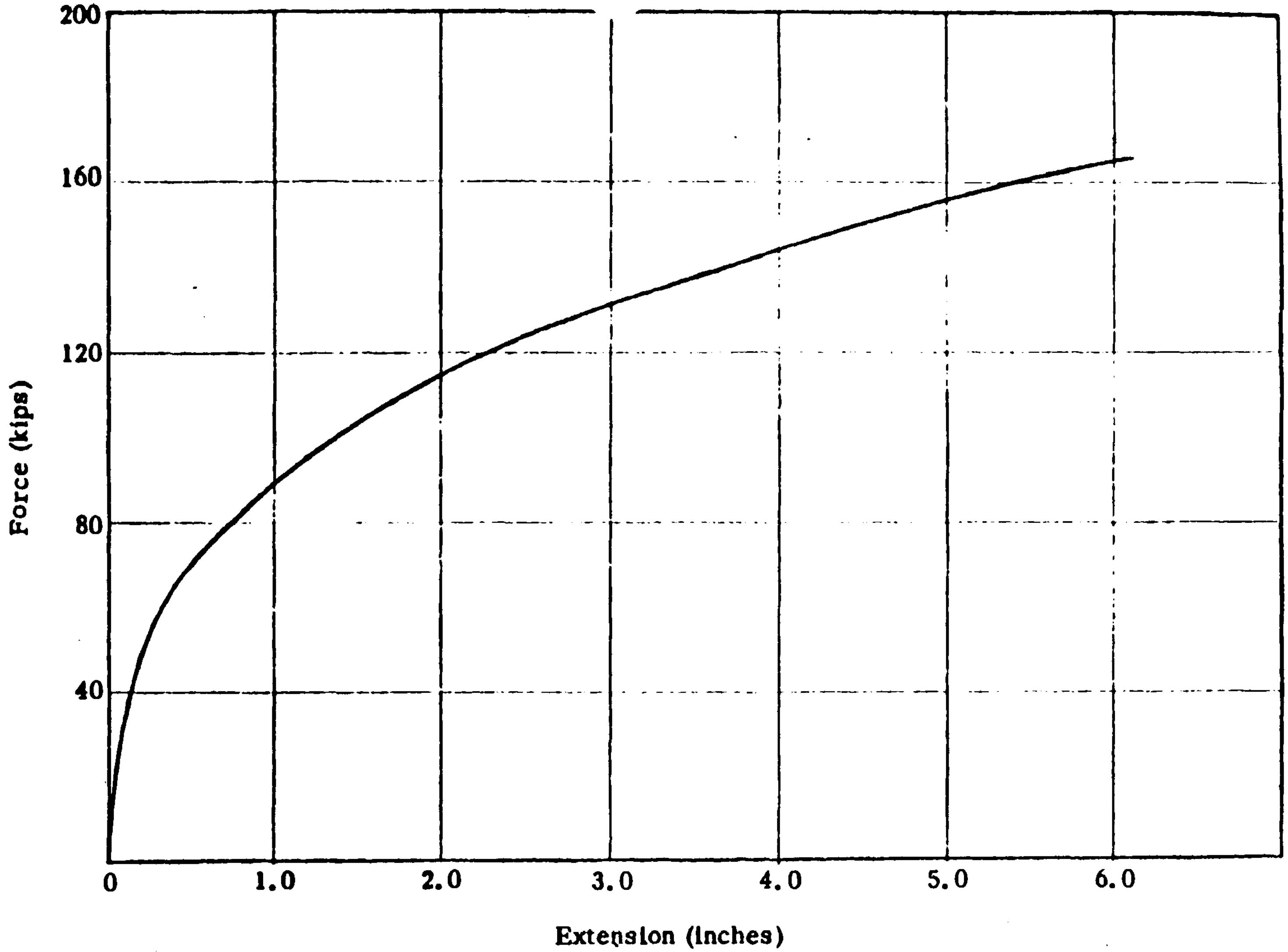


FIGURE 7: TYPICAL FORCE EXTENSION CHARACTERISTICS OF ONE-INCH DIAMETER U-BAR

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NONLINEAR PIPE WHIP ANALYSES

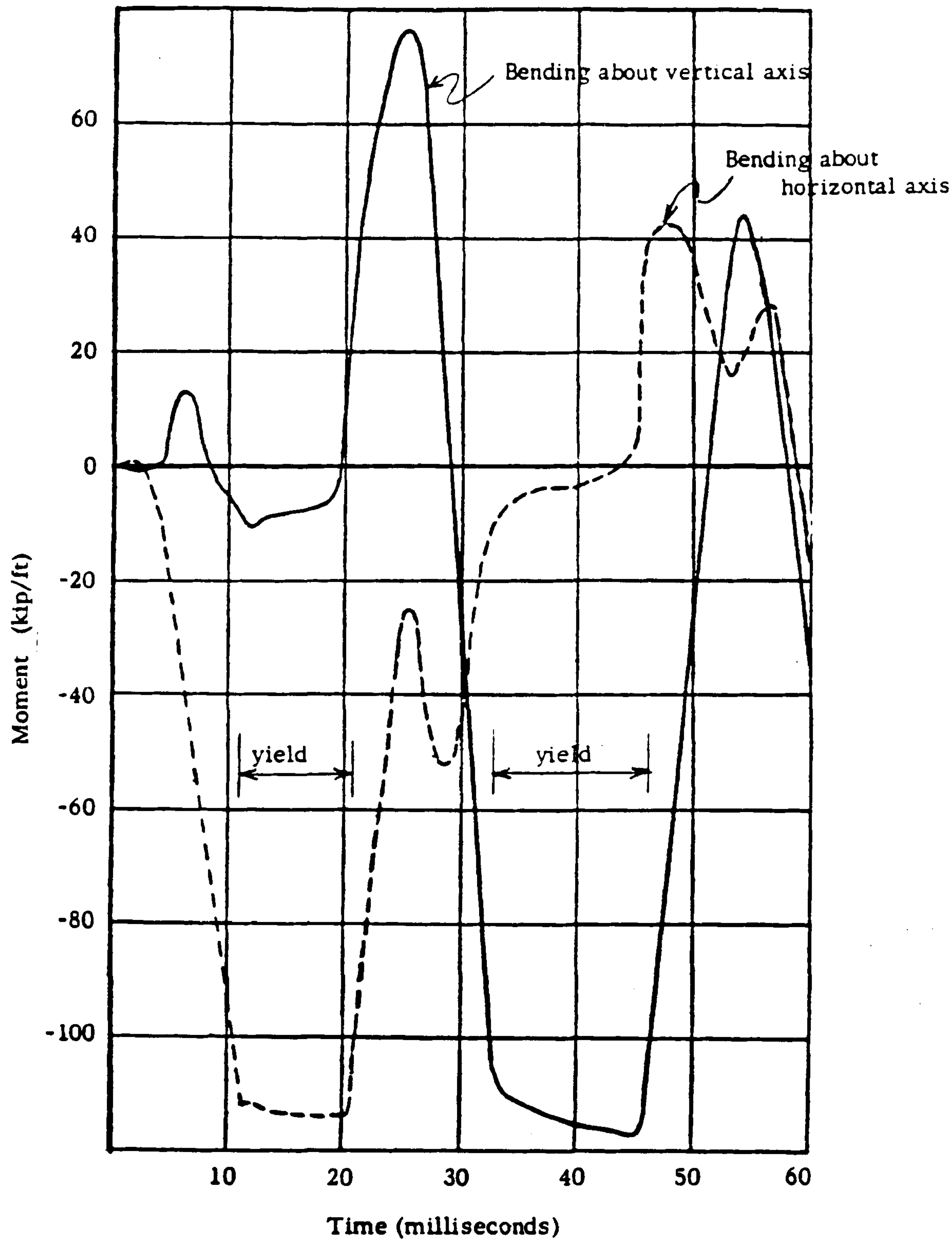



FIGURE 8: TIME HISTORIES OF BENDING MOMENT AT REACTOR VESSEL CONNECTION

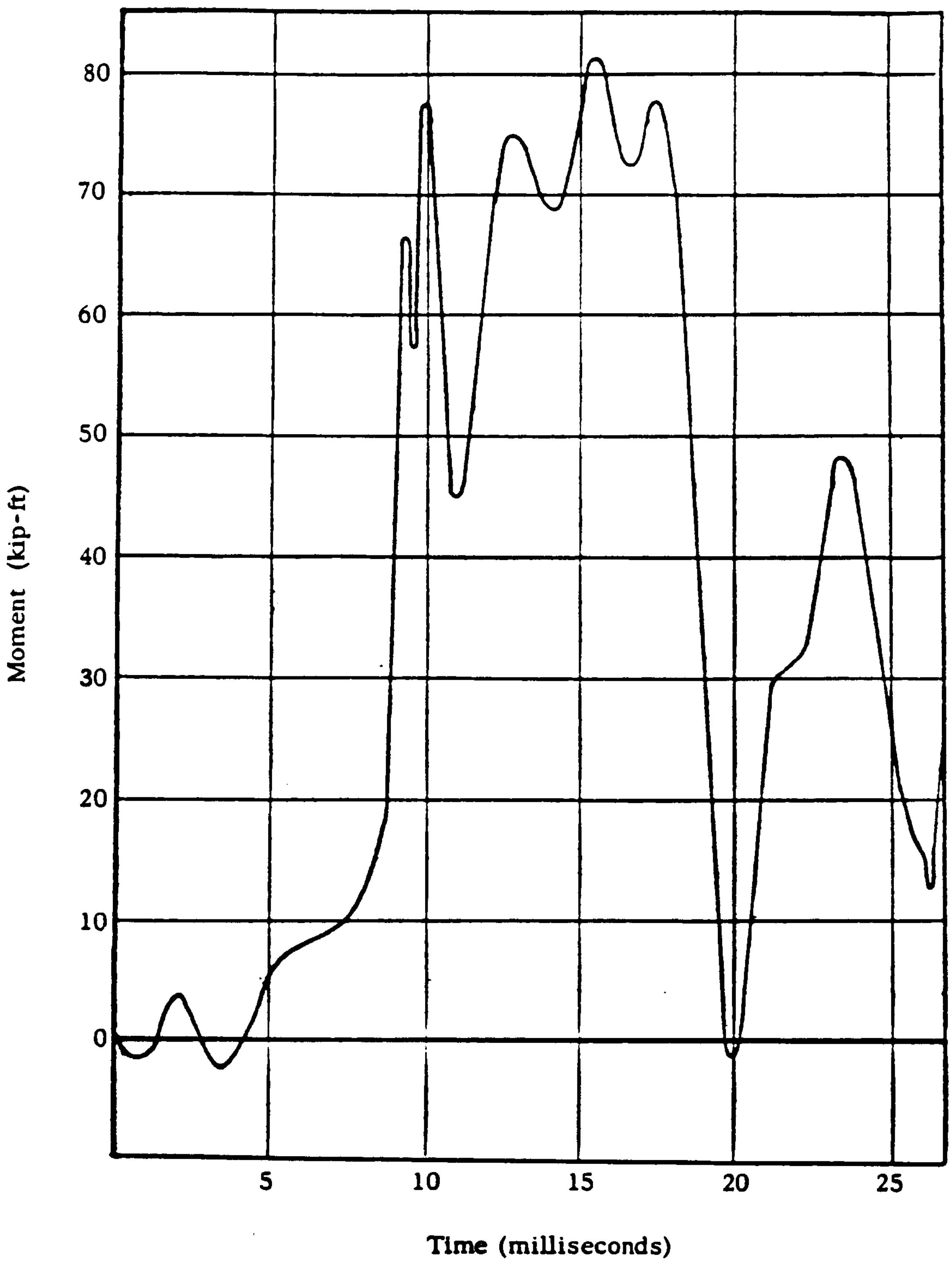



FIGURE 9: TIME HISTORY OF MAXIMUM BENDING MOMENT AT RUPTURE RESTRAINT LOCATION

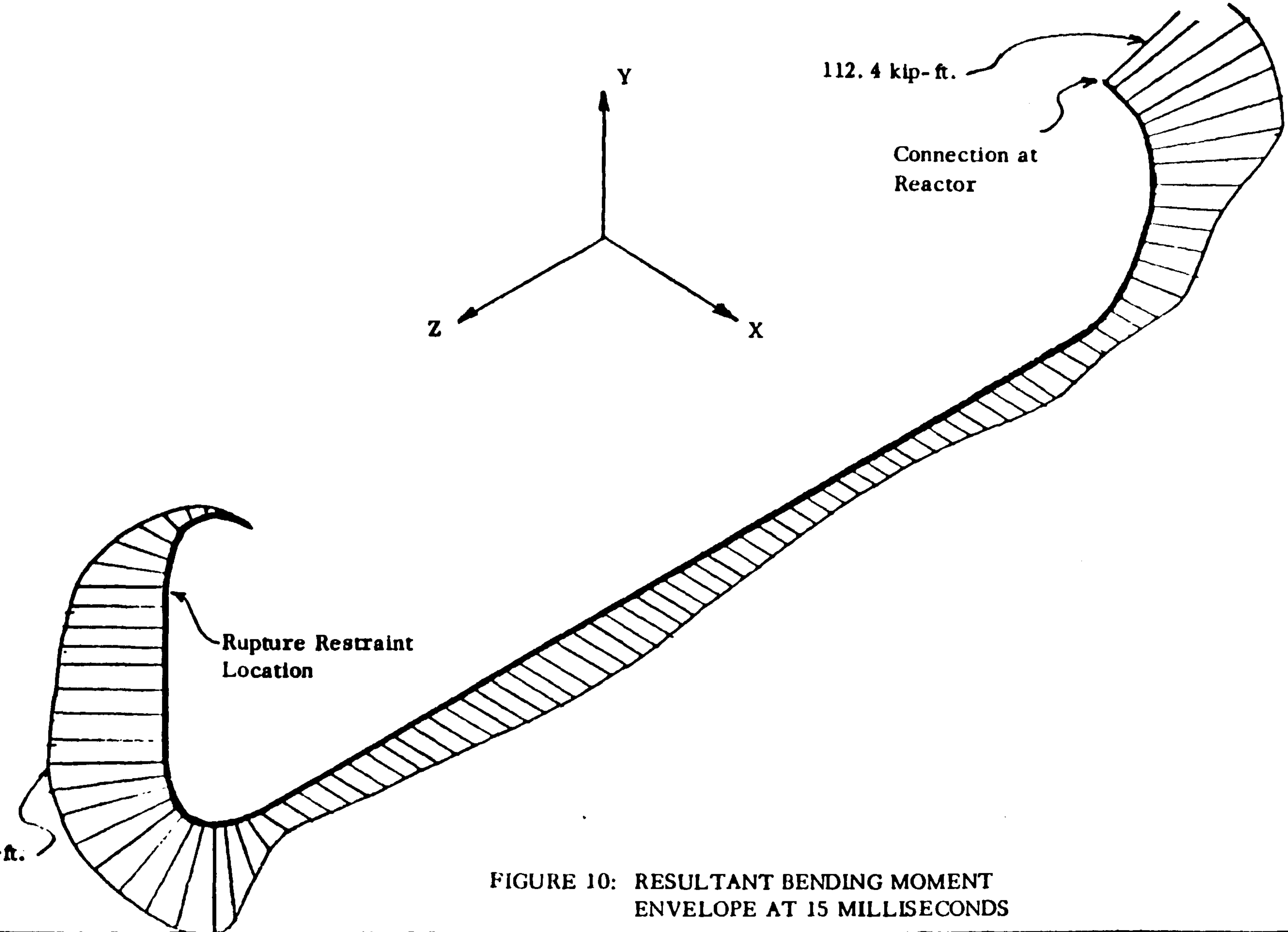



FIGURE 10: RESULTANT BENDING MOMENT ENVELOPE AT 15 MILLISECONDS

| GPU SERVICE CORPORATION | OYSTER CREEK NUCLEAR PLANT<br>NONLINEAR PIPE WHIP ANALYSES | eds nuclear |
|---|---|---|

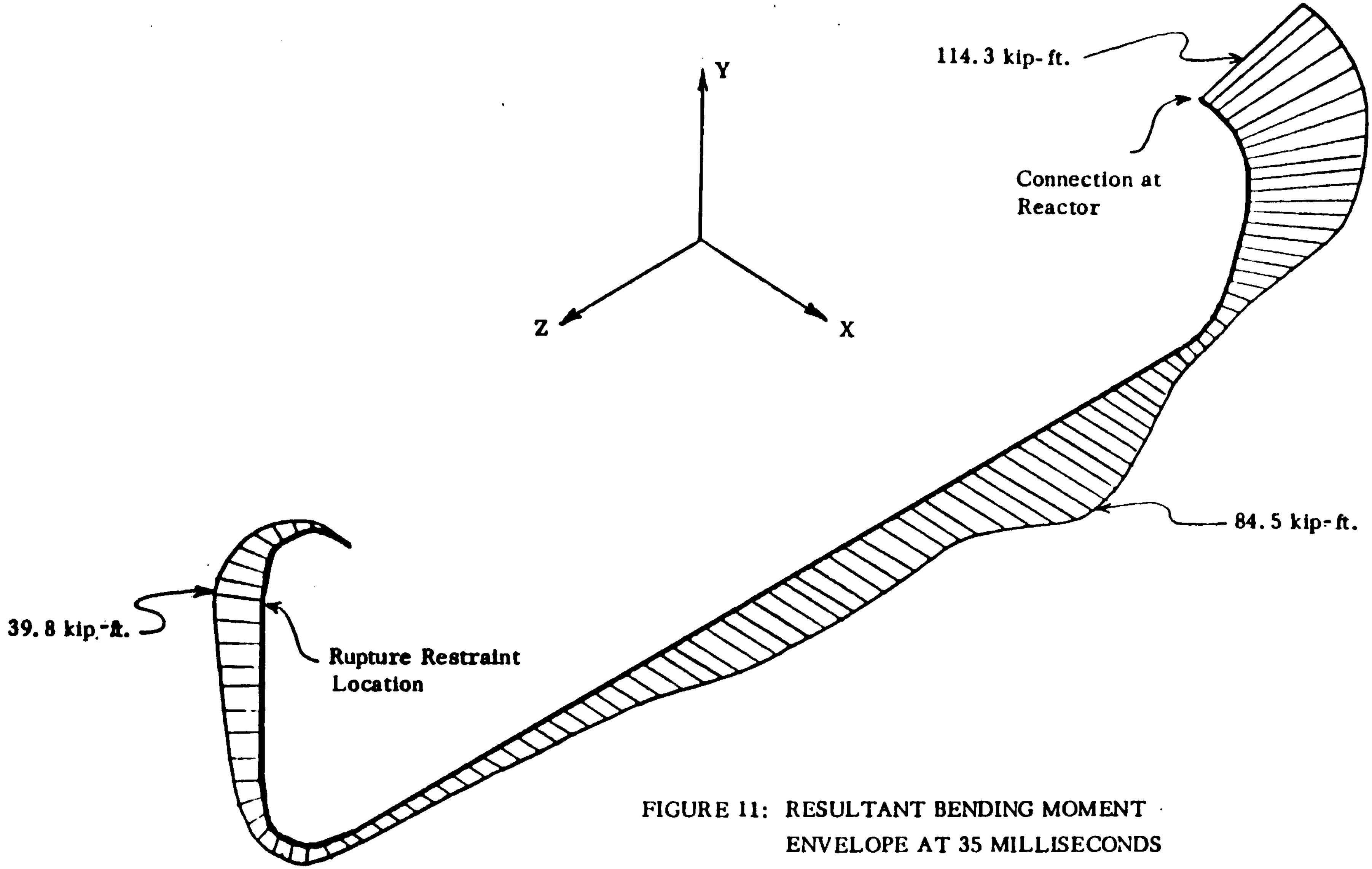



FIGURE 11: RESULTANT BENDING MOMENT ENVELOPE AT 35 MILLISECONDS

| GPU SERVICE CORPORATION | OYSTER CREEK NUCLEAR PLANT<br>NONLINEAR PIPE WHIP ANALYSES | eds nuclear |
|---|---|---|

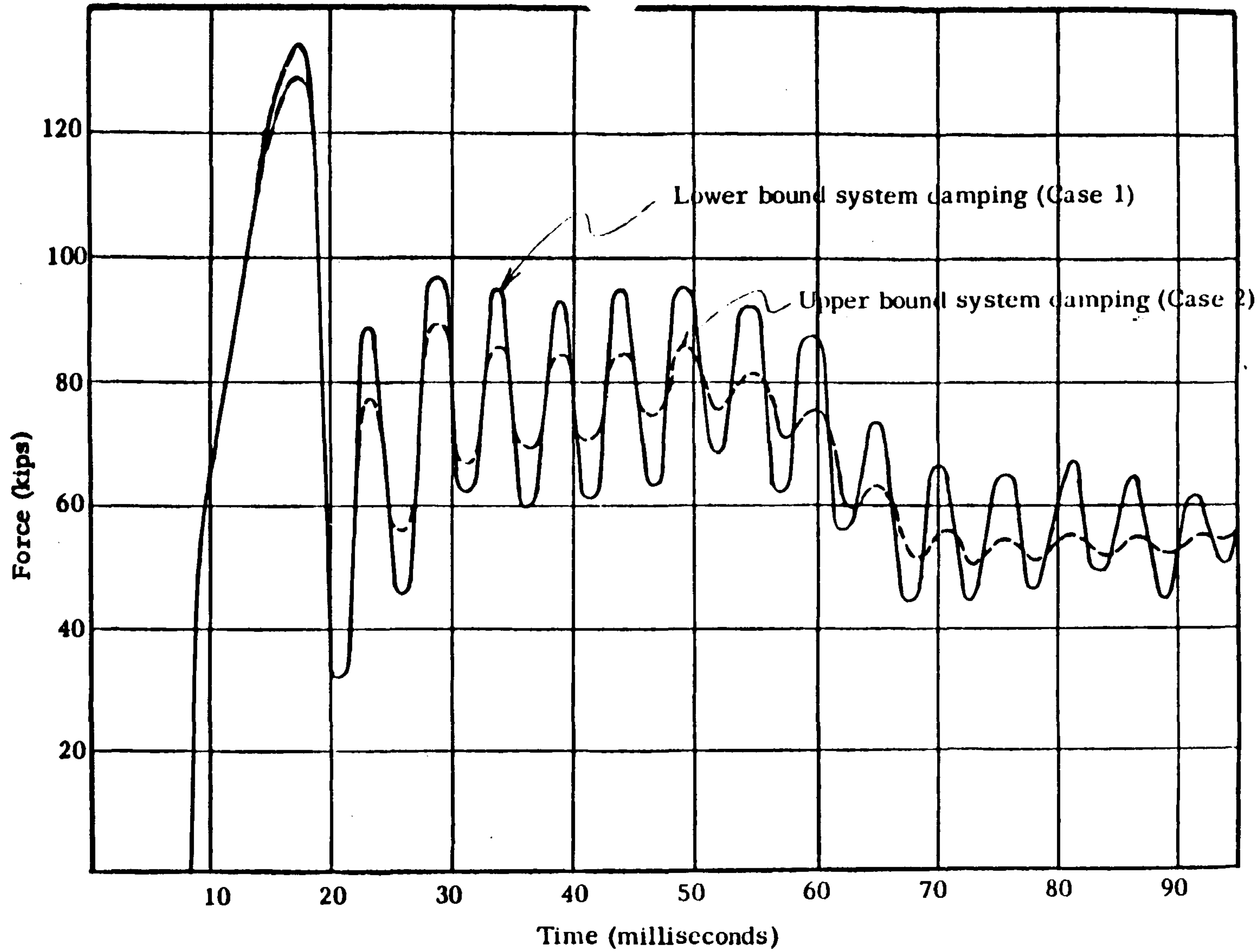



FIGURE 12: TIME HISTORY OF RUPTURE RESTRAINT FORCE
CASES 1 & 2

| | | |
|---|---|---|
| GPU SERVICE CORPORATION | OYSTER CREEK NUCLEAR PLANT<br>NONLINEAR PIPE WHIP ANALYSES | eds nuclear |

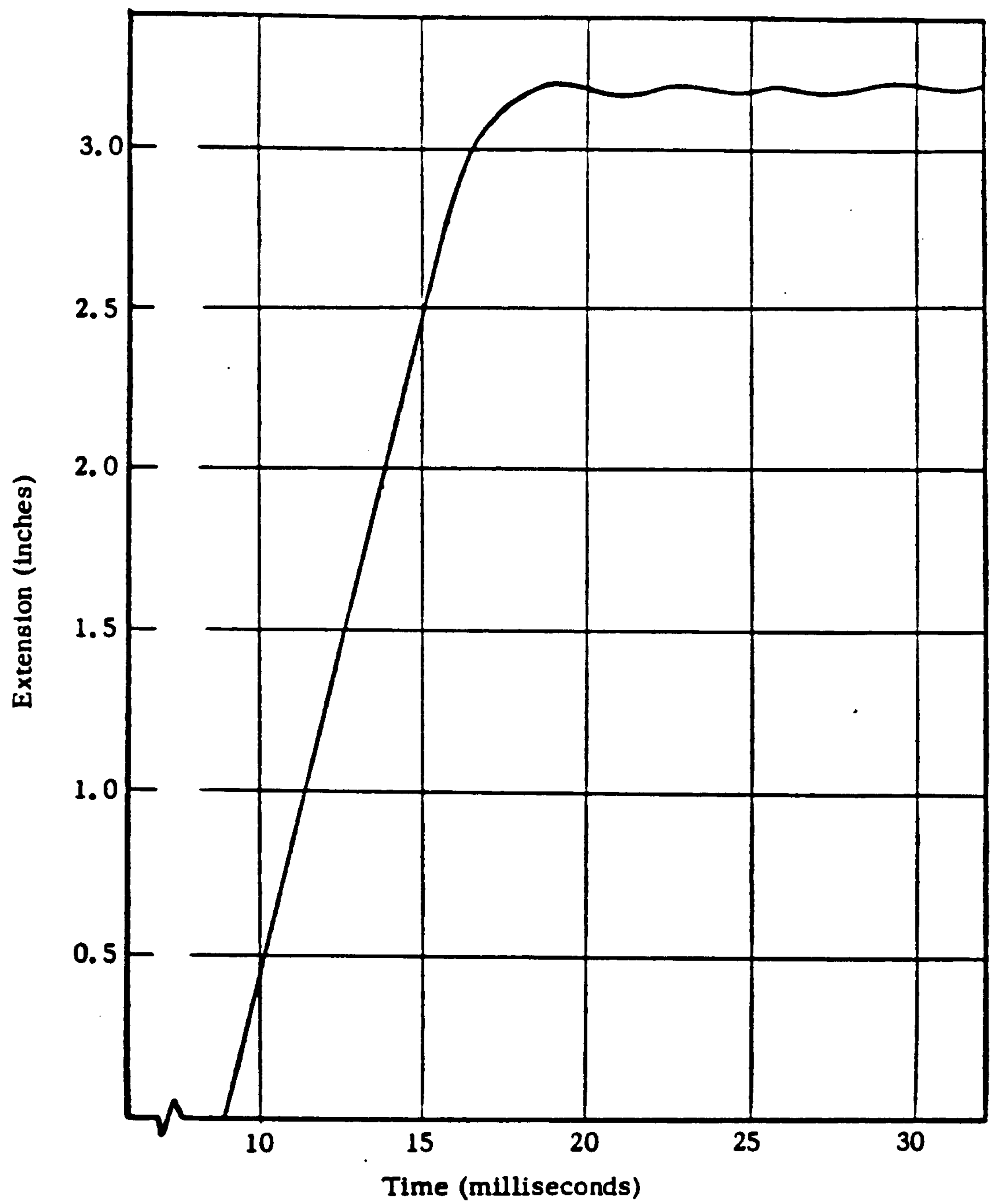



FIGURE 13: TIME HISTORY OF RUPTURE RESTRAINT EXTENSION

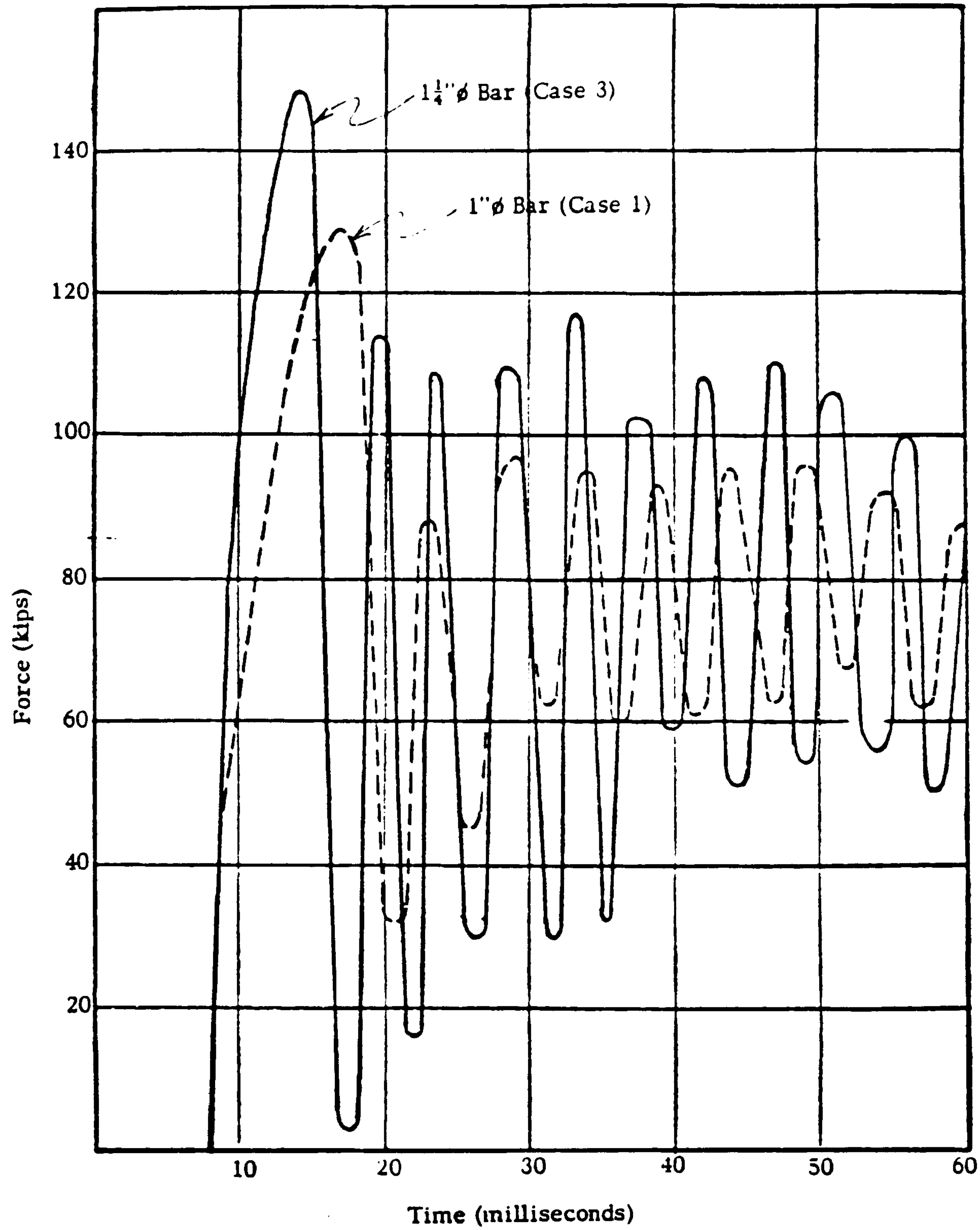



FIGURE 14: TIME HISTORY OF RUPTURE RESTRAINT FORCE
CASES 1 & 3

| | |
|---|---|
| GPU SERVICE CORPORATION | OYSTER CREEK NUCLEAR PLANT<br>NONLINEAR PIPE WHIP ANALYSES |

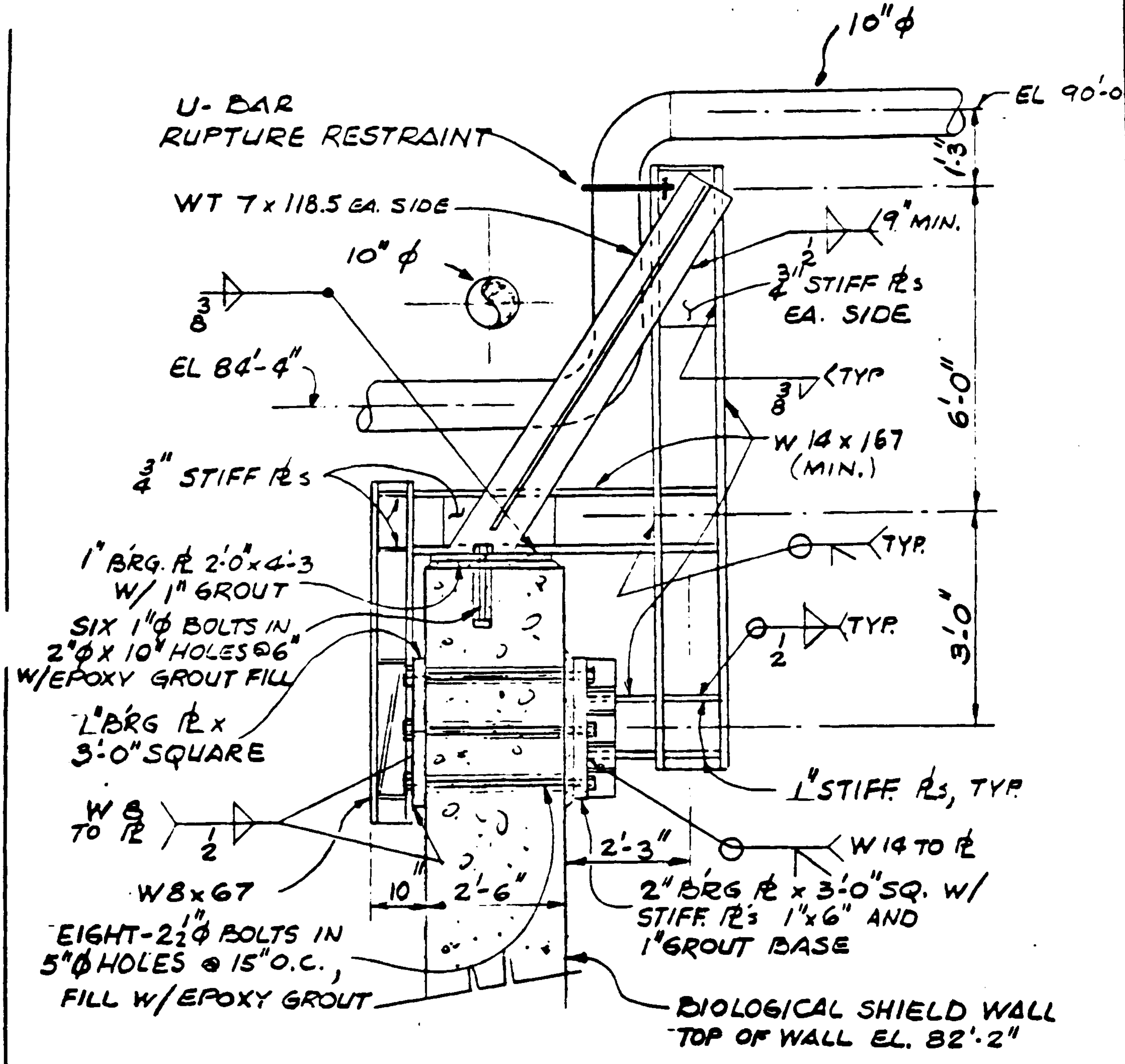



FIGURE 15: RUPTURE RESTRAINT FRAME (ELEVATION)

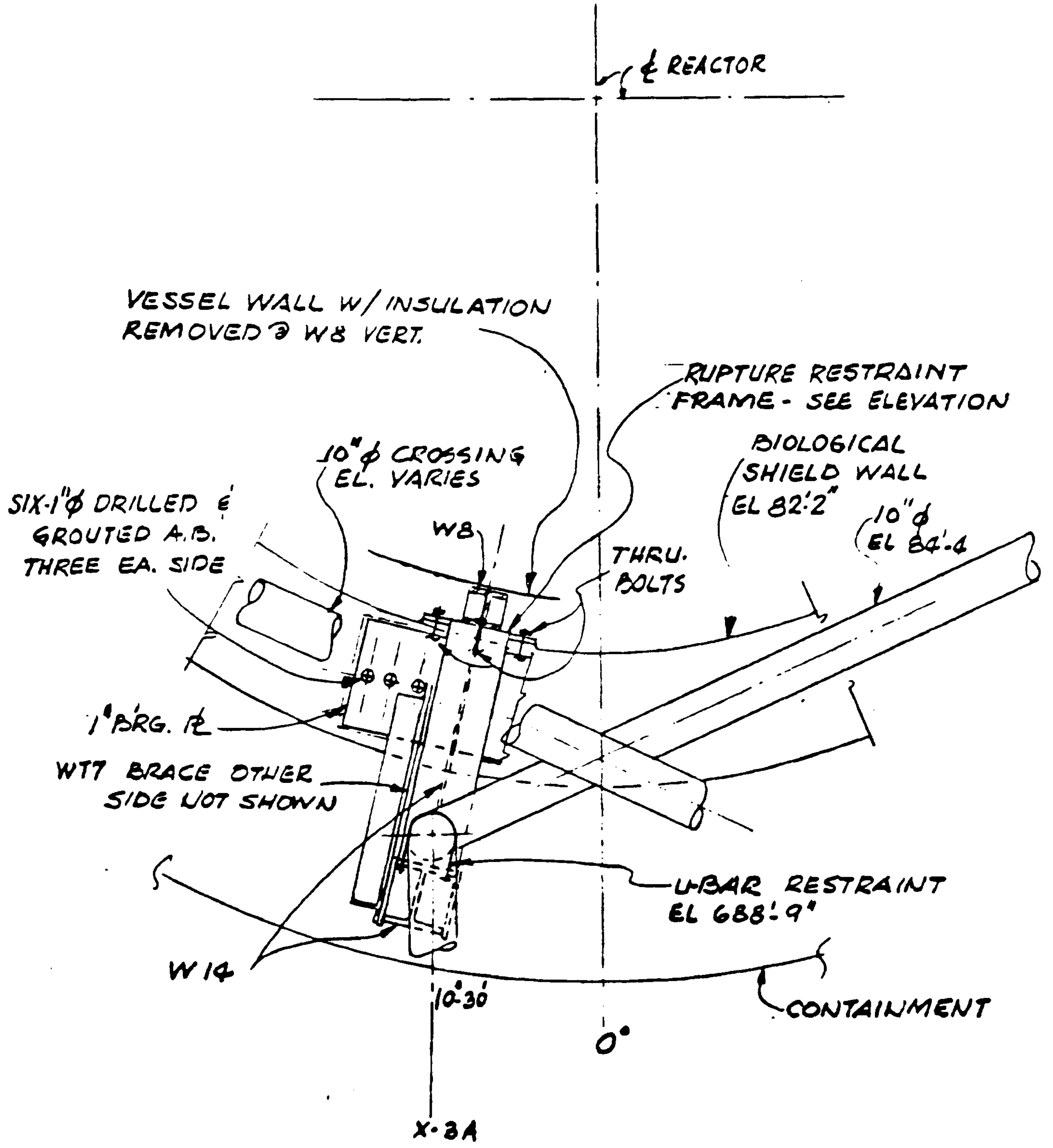



FIGURE 16: RUPTURE RESTRAINT FRAME (LOCATION PLAN)

| GPU SERVICE CORPORATION | OYSTER CREEK NUCLEAR PLANT<br>NONLINEAR PIPE WHIP ANALYSES | eds |
|---|---|---|

# APPENDIX 3.6B

## ANALYSIS OF PIPE BREAKS INSIDE CONTAINMENT

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK
NUCLEAR GENERATING STATION

# ANALYSIS OF PIPE BREAKS

# INSIDE CONTAINMENT

June, 1974

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

# ANALYSIS OF PIPE BREAKS INSIDE CONTAINMENT

## Table of Contents

## 1.0 INTRODUCTION AND SCOPE OF WORK

The following is a summary description of the pipe whip failure analysis inside containment performed in support of the Oyster Creek Nuclear Generating Station design.

The work for this project was performed as two parallel tasks. The first task involved the application of Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment", to the high energy lines inside containment and the evaluation of pipe failure interactions. The results of this first task are presented within this report. The second task involved the performance of a nonlinear elasto-plastic dynamic response analysis, involving a selected pipe failure in one of the high energy systems, and the design of an energy absorbing rupture restraint (i.e.: a rupture restraint that has gone into the plastic range) for the failed pipe system. The results of the second task are reported separately in response to AEC Question 2.a.

## 2.0 DETERMINATION OF PIPE BREAK LOCATIONS

In accordance with the requirements of USAEC Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment", the following criteria were used to determine potential pipe break locations inside containment on the Oyster Creek Nuclear Plant.

### 2.1 Systems Analyzed

Any system inside containment greater than 1-inch nominal pipe size, whose design pressure or temperature exceeded the following values was considered as a potential energy source requiring further pipe whip consideration.

| | |
|---|---|
| Design Temperature | 200°F |
| Design Pressure | 275 PSIG |

All other systems were considered as not having sufficient energy to require pipe whip consideration.

As a result of the above considerations, the following systems were determined as requiring pipe whip failure analysis:

Main Steam
Feedwater
Emergency Condenser
Liquid Poison
Core Spray
Shutdown Cooling
Reactor Vessel Head Cooling
Reactor Recirculation
Reactor Cleanup
Recirculation Bypass
Control Rod Drive Hydraulic Return
Reactor Vessel Vent Line
Reactor Vessel Drain Line

### 2.2 Break Location Criteria

For the systems listed above, pipe whip breaks were postulated to occur at the following locations in each piping run or branch run:

(A piping run interconnects components such as pressure vessels, pumps and anchors that act to restrain pipe movement. A branch run differs from a piping run only in that it originates at a piping intersection as a branch of the main pipe run.)

a. Terminal Ends

b. Any intermediate locations between terminal ends where the combined longitudinal stresses calculated for the loadings associated with the design seismic event plus operational plant conditions exceed 0.8 $(S_h + S_A)$.

c. Intermediate locations in addition to those determined in b. above, selected on a maximum stress level basis, in order to provide, as a minimum, two intermediate break points for each piping run or branch run.

2.3 Loading Combinations

In order to evaluate stress conditions within the piping systems against the stress criteria of 0.8 $(S_h + S_A)$, the following loading conditions were considered and combined:

Gravity Stress
Pressure Stress
Seismic Stress
Thermal Stress

2.4 Combination of Stresses

The four inputs to the combined stress at the various points were added together as follows:

| | | | |
|---|---|---|---|
| | Gravity Stress | - | 1500 psi or as calculated |
| plus | Pressure Stress | - | Per B31.1.0 |
| plus | Seismic Stress | - | Maximum Value from Seismic Stress Reports furnished by GE and B&R |
| plus | Thermal Stress | - | Maximum Value from Thermal Stress Reports furnished by GE and B&R |

The resultant combined stresses for each point on the analyses were tabulated and evaluated against the allowable stress (Refer to Appendix "A") for the pipe material to determine the location of high stress points. These points were then evaluated as described below as potential break point locations.

2.5 Break Locations

After combining stresses as described above, the results were compared to the allowable pipe whip stress [.8 $(S_h + S_A)$] for the material. Breaks were than postulated at all terminal ends (regardless of stress), at all points exceeding .8 $(S_h + S_A)$, and if necessary, at additional intermediate high stress points to provide a minimum of two intermediate break locations for each piping run or branch run.

The postulated break locations are indicated on the isometric drawings provided in Appendix "B" to this report.

## 3.0 INTERACTION ANALYSIS

The purpose of this section of the report is to describe the analysis of the effects of pipe whip and jet impingement resulting from postulated pipe breaks identified in Section 2.0 of this report in accordance with the requirements of USAEC Regulatory Guide 1.46.

### 3.1 Criteria and Assumptions

The following criteria and assumptions form the basis for the analysis and were uniformly applied throughout.

#### 3.1.1 Types of Postulated Breaks

Circumferential breaks occur at all postulated break points.

Longitudinal breaks occur at all postulated break points in lines greater than four-inch (4") nominal pipe size.

Circumferential and longitudinal breaks were considered to be nonsimultaneous occurrences and the effects of these breaks were, therefore, analyzed independently.

#### 3.1.2 Pipe Whip Occurrence

Pipe whip was assumed to occur as a result of a circumferential rupture in a system provided there was a sufficient reservoir of energy to force the formation of a plastic hinge about some point in the line. Table 3-1 of this report lists each system evaluated for the effects of pipe whip and its attendant energy reservoir.

The whipping force was applied along the axis of the ruptured pipe normal to the plane of the break. The resultant plastic hinges were assumed to occur at system anchors or at other intermediate locations as dictated by the complexity of the particular system configuration. (For branch lines, the terminal connections were considered as branch line anchors.) Plastic hinges formed as a result of the application of this force were assumed to form in either the bending or torsion mode depending upon the system configuration. The hinge location(s) and mode(s) of formation for each break point analyzed are described in Appendix "C", Reference C-1, Interaction Evaluation.

### 3.1.3 Jet Impingement Occurrence

A high energy steam jet was assumed to occur as a result of a longitudinal break at any point around the circumference of the break location in a direction normal to the axis of the ruptured pipe. The break was assumed to have an area of two times the cross-sectional area of the pipe resulting in the formation of a cone with angle of divergence of thirty degrees ($30^{o}$).

A high energy steam jet was assumed to occur as a result of a circumferential break. The jet path was assumed to be in the plane of movement (whip) emanating from the end of the whipping pipe.

### 3.1.4 Breaks at Penetration Assemblies

The penetration assemblies were assumed to withstand and transmit pipe rupture forces to support structures without plastic deformation. Additionally, the effects of jet impingement were not analyzed for breaks postulated to occur between the penetration assembly and the first isolation valve outside containment.

### 3.1.5 Selection of Targets

Structures, systems and components whose damage would result in either a loss of safety function or an increase in the severity of a pipe rupture accident were analyzed for the effects of interaction with whipping pipe and jet impingement

### 3.1.6 Interaction Consequences

The bases for evaluation the consequences of interactions between the high energy source systems and the selected targets were as follows:

A whipping pipe was considered to have sufficient energy to cause damage to:

a. Pipes of smaller nominal size and/or lighter wall thickness
b. The steel containment vessel
c. Electric conduit and cable trays
d. Electric motor operators

A steam jet was considered to have sufficient energy to cause damage to:

a. Electric cable trays
b. Electric motor operators

Reports prepared by MPR Associates, Inc. (Report No. MPR-285, May 7, 1971) and Burns and Roe, Inc. (Penetration Analysis for Jet Impingement Due to Pipe Rupture, April 24, 1968) demonstrate the ability of the steel containment vessel withstand the effects of jet impingement.

The analysis considered only the primary effects of interaction. It was assumed that a damaged target could not initiate a sequence of cascading interactions, i.e.: a damaged pipe was not evaluated as a potential source for pipe whip or jet impingement as a result of the primary interaction.

## 3.2 Interaction Matrix

The results of the analysis are shown on matrices in Appendix "C" of this report. The matrices were prepared on a system basis showing the potential interactions between each postulated break point (source) and each selected target.

Interactions were defined as follows:

| | | | |
|---|---|---|---|
| a. | Acceptable | - | Interaction causes no damage. |
| b. | Unacceptable | - | Interaction causes damage resulting in a loss of safety function or increase in severity of accident. |
| c. | No Interaction | - | Interaction physically not possible. |

A summary interaction matrix is presented in Table 3-2 of this report indicating the results of the analysis.

## 3.3 Results of Interaction Analysis

Table 3-3 of this report lists those postulated break points whose rupture resulted in unacceptable interactions.

TABLE 3-1

PIPING SYSTEMS AND ENERGY RESERVOIRS

| System | Energy Reservoir |
| --- | --- |
| Emergency Condenser | Reactor Vessel |
| Core Spray | Reactor Vessel |
| Reactor Cleanup | Reactor Vessel |
| Shutdown Cooling | Reactor Vessel |
| Liquid Poison | Reactor Vessel |
| Reactor Vessel Head Cooling | Reactor Vessel |
| Reactor Recirculation Loop | Reactor Vessel |
| Reactor Recirculation Bypass | Reactor Vessel |
| Control Rod Drive Hydraulic Return | Reactor Vessel |
| Main Steam | Reactor Vessel and Main Steam System Outside Containment |
| Feedwater | Reactor Vessel and Feedwater System Outside Containment |
| Reactor Vessel Vent | Reactor Vessel |
| Reactor Vessel Drain | Reactor Vessel |

## TABLE 3-2

## SUMMARY INTERACTION EVALUATION MATRIX

| SYSTEM | | TARGET | | |
|---|---|---|---|---|
| System | ISO | Containment | Piping | Electrical |
| Emergency Condenser NE01-B 10" NE-5 | B-1 | N | A | N |
| Emergency Condenser NE01-A 10" NE-5 | B-2 | N | A | N |
| Emergency Condenser NE01-B 10" NE-2 | B-3 | N | A | U |
| Emergency Condenser NE01-A 10" NE-2 | B-4 | N | A | U |
| Core Spray (South) 8" NZ-3 | B-5 | U | N | N |
| Core Spray (North) 8" NZ-3 | B-6 | N | A | N |
| Main Steam (South) 24" MS | B-7 | U | U | U |
| Main Steam (North) 24" MS | B-7 | U | U | U |
| Reactor Cleanup 6" ND-10 | B-8 | N | A | U |
| Reactor Cleanup 6" ND-1 | B-9 | N | A | N |
| Shutdown Cooling 14" NU-1, NU-4 | B-10 | N | A | U |
| Shutdown Cooling 14" NU-2, NU-3 | B-11 | N | N | U |
| Feedwater (South) 10" RF-2 | B-12 | N | U | N |
| Feedwater (South) 18" RF-2 | B-12 | U | A | U |
| Feedwater (North) 10" RF-2 | B-12 | N | U | U |
| Feedwater (North) 18" RF-2 | B-12 | U | N | U |
| Liquid Poison 1.5" NP-2 | B-13 | U | A | N |
| Reactor Vessel Head Cooling 2" RHC-2 | B-14 | N | N | N |
| Reactor Recirculation (Loops A, B, C, D, E) | B-15 | U | U | U |
| Reactor Recirculation (Bypass A, B, C, D, E) | B-16 | N | N | U |
| Control Rod Drive Hydraulic Return 3" NC-4, NC-2 | B-17 | U | N | U |

LEGEND
- U - Unacceptable Interaction (Damage Possible)
- A - Acceptable Interaction (Damage not Possible)
- N - No Interaction

TABLE 3-3

UNACCEPTABLE INTERACTIONS

| System | ISO Number | Break Point Number |
|---|---|---|
| Emergency Condenser NE01-B 10" NE-2 | B-3 | 1 |
| Emergency Condenser NE01-A 10" NE-2 | B-4 | 1 |
| Core Spray (South) 8" NZ-3 | B-5 | 2, 3 |
| Main Steam (South) 24" MS | B-7 | 1, 2, 3 |
| Main Steam (North) 24" MS | B-7 | 1, 2, 3 |
| Reactor Cleanup 6" ND-10 | B-8 | 1, 2, 3, 4, 5 |
| Shutdown Cooling 14" NU-1, NU-4 | B-10 | 1 |
| Shutdown Cooling 14" NU-2, NU-3 | B-11 | 1 |
| Feedwater (South) 10" RF-2 | B-12 | 1, 2, 3, 4, 8, 9 |
| Feedwater (South) 18" RF-2 | B-12 | 5, 6, 7, 12 |
| Feedwater (North) 10" RF-2 | B-12 | 1, 2, 3, 4, 8, 9 |
| Feedwater (North) 18" RF-2 | B-12 | 5, 6, 7, 12 |
| Liquid Poison 1.5" NP-2 | B-13 | 2 |
| Reactor Recirculation Loops A, B, C, D, E | B-15 | 1, 2, 3, 4 |
| Reactor Recirculation Bypass Loop D | B-16 | 2 |
| Control Rod Drive Hydraulic Return 3" NC-4, NC-2 | B-17 | 2, 3 |

## 4.0 PIPE WHIP PROTECTION PROGRAM

This section of the report describes the program recommended for protection against pipe failure inside containment and, further, evaluates each postulated break location for incorporation into that program.

## 4.1 Feasibility Evaluation of Protection Methods

The methods of failure protection evaluated were inservice inspection and surveillance, design and installation of rupture restraints, and installation of barriers.

### 4.1.1 Installation of Rupture Restraints

Based upon review of plant drawings and several site visits during periods of plant outage, it was determined, for reasons enumerated below, that installation of rupture restraints is unfeasible and does not offer a practicable solution to the problem of protection against the effects of pipe whip due to breaks at the postulated locations.

a. The drywell area around the break location is too congested to allow for installation of a restraint structure, thus necessitating temporary removal of existing structures, equipment and piping systems.

b. Installation of restraints would require rerouting of piping systems and relocation of existing equipment.

c. The restraint structure would interfere with normal maintenance or repair of existing equipment.

d. The restraint structure would hinder inservice inspection of piping system welds.

e. Adequate rupture load carrying structures are not available or accessible.

f. Installation of the restraints would result in high personnel radiation exposure.

### 4.1.2 Installation of Barriers

Installation of barriers for protection against pipe whip and jet impingement was considered unfeasible for reasons of congestion and lack of adequate support structures.

TABLE 4-1

PERCENT OF ALLOWABLE PIPE WHIP STRESS $.8\ (S_h + S_A)$

| System | Percent of Allowable | | | | | |
|---|---|---|---|---|---|---|
| | ≤30 | 30-39 | 40-49 | 50-59 | 60-69 | 70 ≥ |
| Emergency Condenser (Reactor to NE01-B) | 0 | 0 | 0 | 1 | 0 | 3 |
| Emergency Condenser (Reactor to NE01-A) | 0 | 0 | 0 | 1 | 0 | 3 |
| Emergency Condenser (NE01-B to Reactor) | 0 | 0 | 1 | 3 | 0 | 0 |
| Emergency Condenser (NE01-A to Reactor) | 0 | 0 | 1 | 3 | 0 | 0 |
| Core Spray (South) | 0 | 0 | 0 | 0 | 1 | 12 |
| Core Spray (North) | 0 | 0 | 0 | 0 | 2 | 9 |
| Main Steam (Both Lines) | 2 | 0 | 4 | 0 | 2 | 0 |
| Reactor Cleanup (Return to Recirc.) | 0 | 0 | 0 | 0 | 0 | 11 |
| Reactor Cleanup (Discharge from Recirc.) | 0 | 0 | 0 | 0 | 1 | 5 |
| Shutdown Cooling (Discharge from Recirc.) | 0 | 2 | 2 | 0 | 0 | 0 |
| Shutdown Cooling (Return to Recirc.) | 0 | 1 | 0 | 3 | 0 | 0 |
| Feedwater (Both Lines) | 8 | 16 | 0 | 0 | 0 | 0 |
| Liquid Poison | 0 | 0 | 0 | 0 | 5 | 0 |
| Reactor Vessel Head Cooling | 0 | 0 | 0 | 0 | 1 | 3 |

TABLE 4-2

BREAK POINTS REQUIRING ADDITIONAL SURVEILLANCE

| System | ISO Number | Break Point Number |
|---|---|---|
| Core Spray (South)<br>8" NZ-2 | B-5 | 2, 3 |
| Main Steam (South)<br>24" MS | B-7 | 2 |
| Main Steam (North)<br>24" MS | B-7 | 2 |
| Reactor Cleanup<br>6" ND-10 | B-8 | 1, 2, 3, 4, 5 |
| Liquid Poison<br>1.5" NP-2 | B-13 | 2 |
| Reactor Recirculation Bypass<br>Loop D | B-16 | 2 |
| Control Rod Drive Hydraulic<br>Return<br>3" NC-4, NC-2 | B-17 | 2, 3 |

## 5.0 CONCLUSION

The preceding report has served to:

a. Locate potential pipe rupture points, in accordance with USAEC Regulatory Guide 1.46.

b. Evaluate damage potential due to postulated breaks at these points.

c. Determine those postulated break points which require incorporation into a pipe whip protection program.

d. Evaluate and recommend method of protection against pipe break accident.

A total of 150 points were identified as potential break locations in the systems evaluated. Of these points, it was determined that 61 may cause unacceptable damage. Of those that could cause unacceptable damage, 13 points have stress levels greater than 50% of $.8\ (S_h + S_A)$.

# APPENDIX A

## CALCULATION OF ALLOWABLE STRESS

## APPENDIX "A"

## CALCULATION OF ALLOWABLE STRESS

### PROBLEM

Calculate allowable stress values for piping material in each system per Regulatory Guide 1.46.

### EQUATIONS

1. Allowable Stress = $0.8\ (S_h + S_A)$ - from Regulatory Guide 1.46

2. SA = $1.25\ S_c + 0.25\ S_h$ - from USAS B31.1.0, Para. 102.3.2.(c)

   $S_c$ = Material allowable stress at cold condition ($70^oF$)

   $S_h$ = Material allowable stress at hot condition ($575^oF$)

### Carbon Steel Systems - Allowable Stress Calculation

| Systems | Material |
|---|---|
| Main Steam | A-106 Grade C |
| Feedwater | A-106 Grade C |
| Shutdown Cooling | A-106 Grade C |

a. $S_c$ @ $70^oF$ = $S_h$ @ $575^oF$ = 17,500 psi - from USAS B31.1.0, Appendix A, Tables A-1 and A-2

b. **Substituting these values into Equation 2. above and solving for $S_A$ in terms of $S_h$:**

$$S_A = 1.25\ S_h + 0.25\ S_h$$

$$S_A = 1.5\ S_h$$

b. Substituting these values into Equation 2. above:

$$S_A = 1.25\ (18,750) + 0.25\ (17,125)$$

$$S_A = 27,718 \text{ psi}$$

c. Substituting this value into Equation 1. above:

$$\text{Allowable Stress} = 0.8\ (S_h + S_A) \text{ psi}$$

$$= 0.8\ (17,125 + 27,718) \text{ psi}$$

$$\text{Allowable Stress} = 35,784 \text{ psi}$$

# APPENDIX B

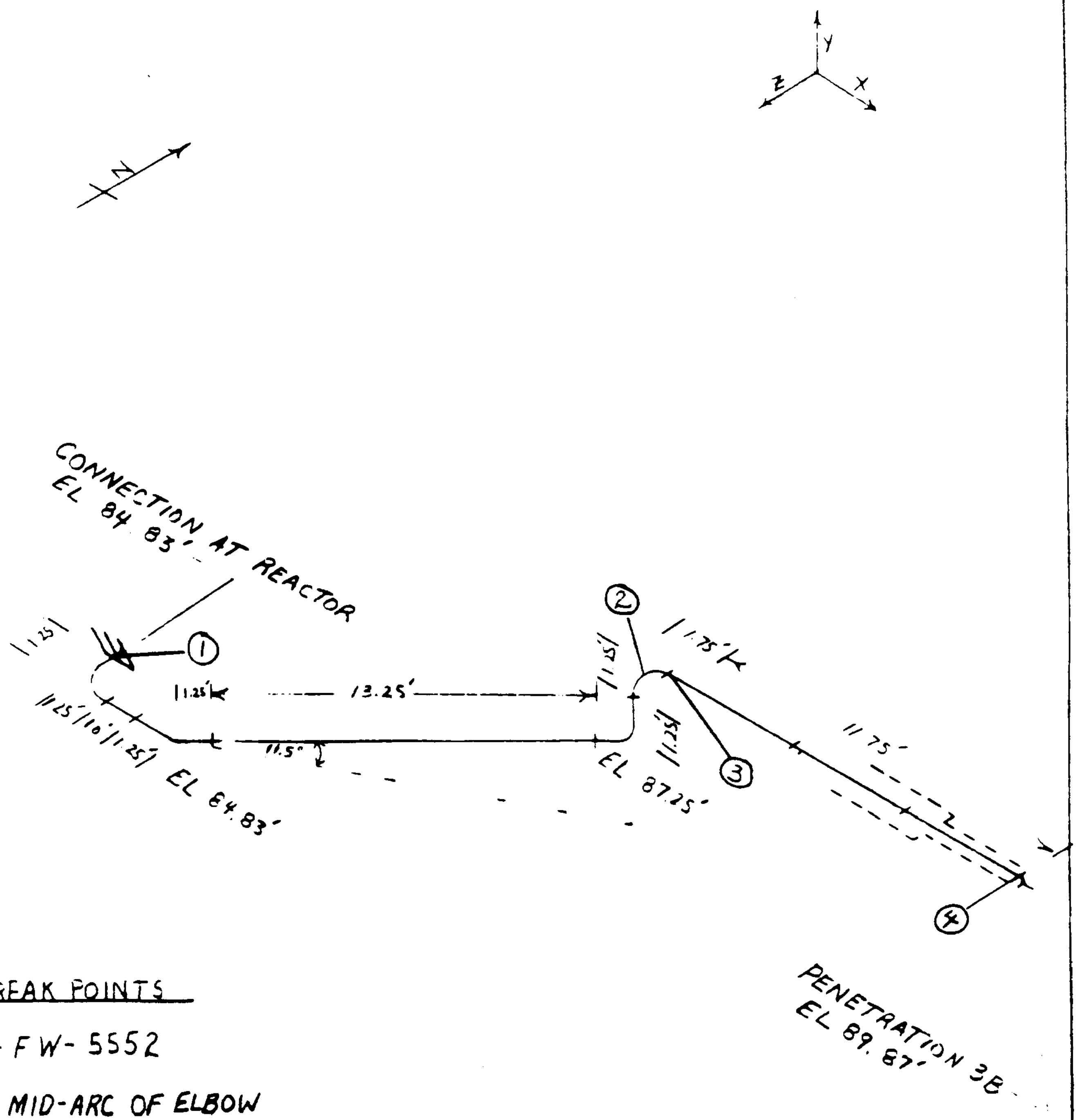
## EDS ISOMETRICS

# APPENDIX B

## ISOMETRIC DRAWINGS

Break point locations are referenced by the nearest field weld or shop weld number as applicable.

### List of Drawings




BREAK POINTS

1- FW- 5552

2- MID-ARC OF ELBOW

3- FW-5554

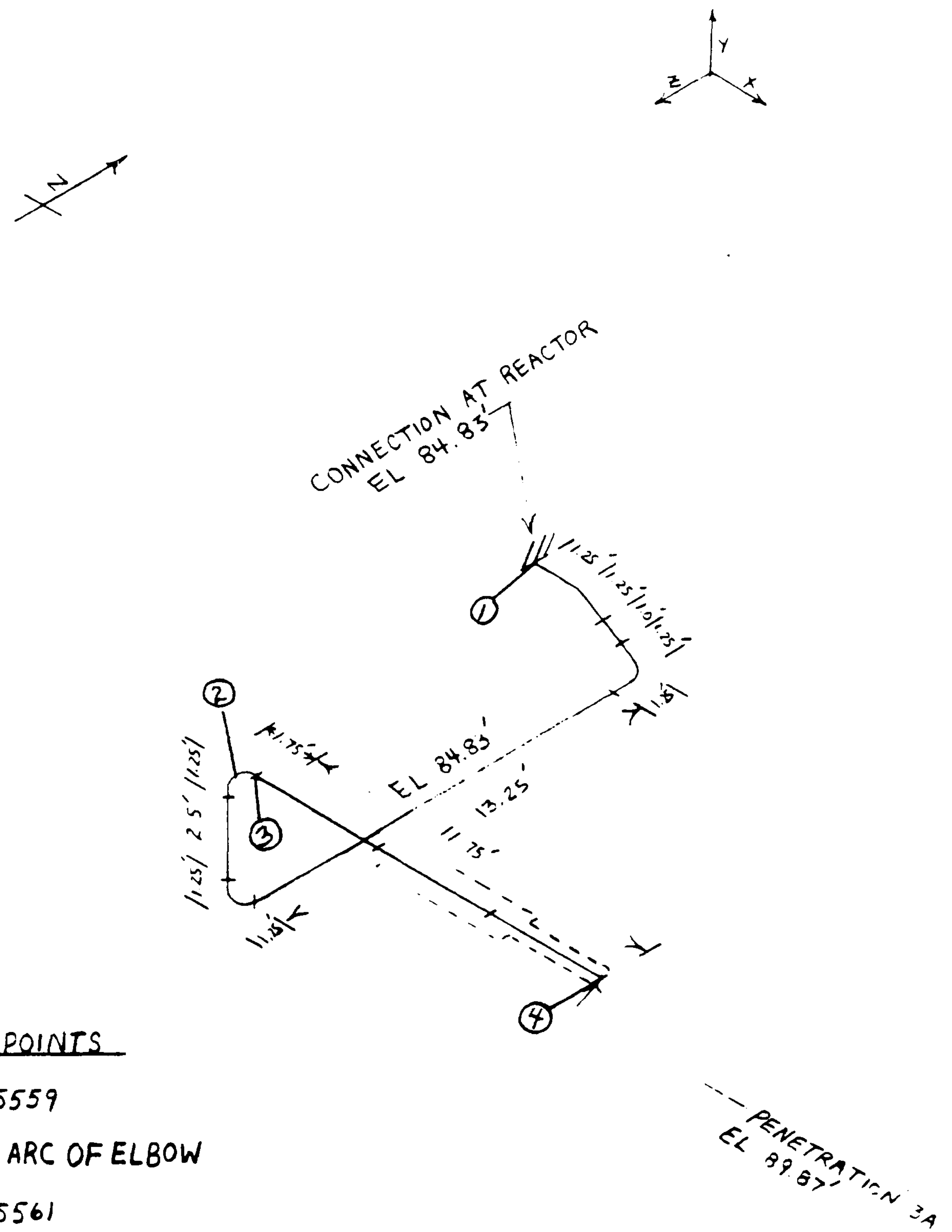
4- SW-AV-2

By: G.T.R
Date: 5/6/74

EMERGENCY CONDENSER SYSTEM (Reactor to NE01-B)

10" NE-5, SS-A312 TP316, Design - P-1250 psig, T-575°F

Figure B-1




BREAK POINTS

1-FW-5559

2-MID-ARC OF ELBOW

3-FW-5561

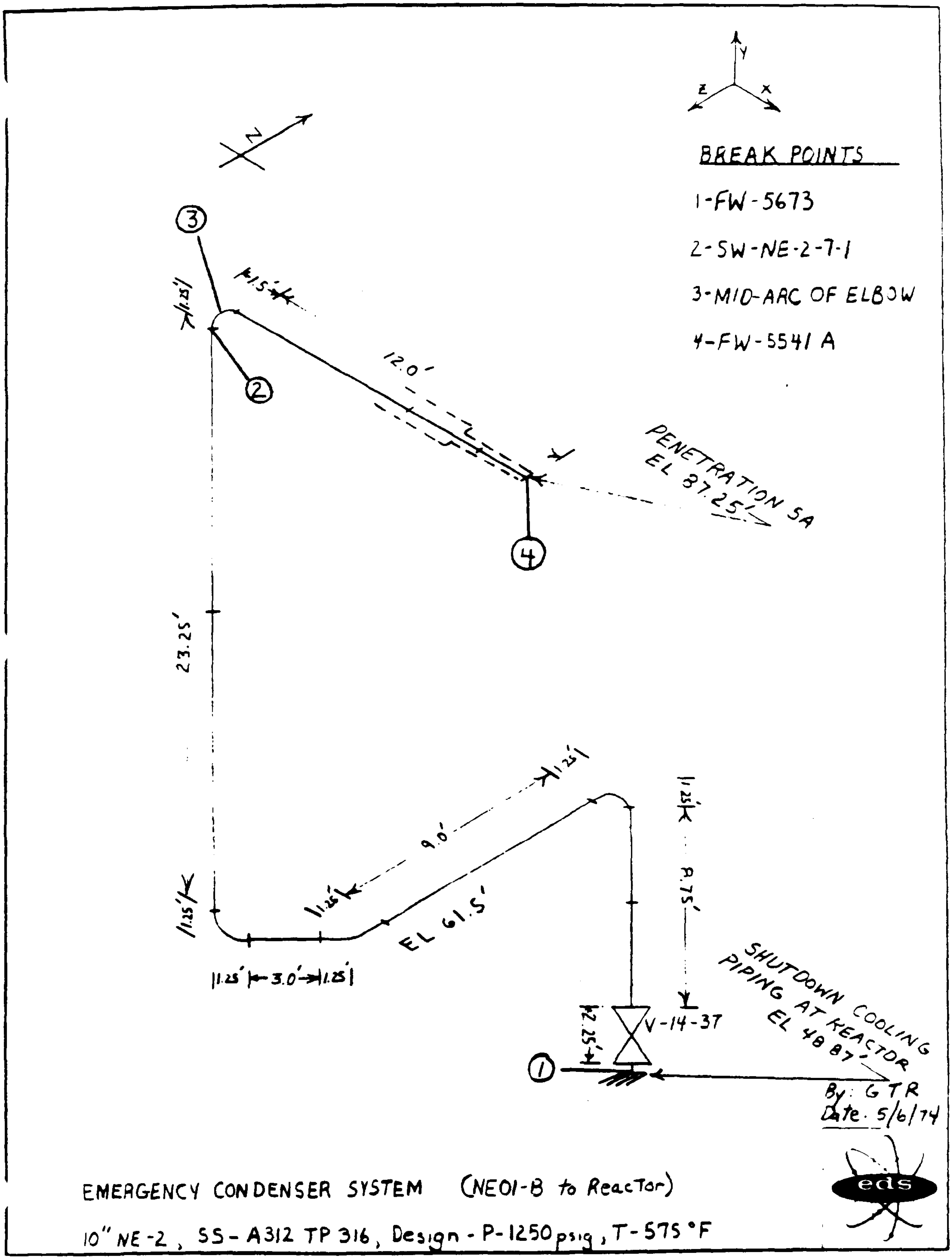
4-SW-AV-4

By: GTR
Date 5/6/74

EMERGENCY CONDENSER SYSTEM (Reactor to NE01-A)

10" NE-5, SS-A312 TP316, Design -P-1250psig, T-575°F

Figure B-2



Y
Z
X
N
BREAK POINTS
1-FW-5673
2-SW-NE-2-7-1
3-MID-ARC OF ELBOW
4-FW-5541 A
3
2
4
1
1.25'
1.5'
12.0'
PENETRATION 5A
EL 87.25'
23.25'
1.25'
9.0'
1.25'
1.25'
EL 61.5'
1.25'
3.0'
1.25'
1.25'
8.75'
2.25'
V-14-37
SHUTDOWN COOLING
PIPING AT REACTOR
EL 48 87'
By: GTR
Date. 5/6/74
EMERGENCY CONDENSER SYSTEM (NE01-B to Reactor)
10" NE-2, SS-A312 TP 316, Design-P-1250 psig, T-575 °F


Figure B-3

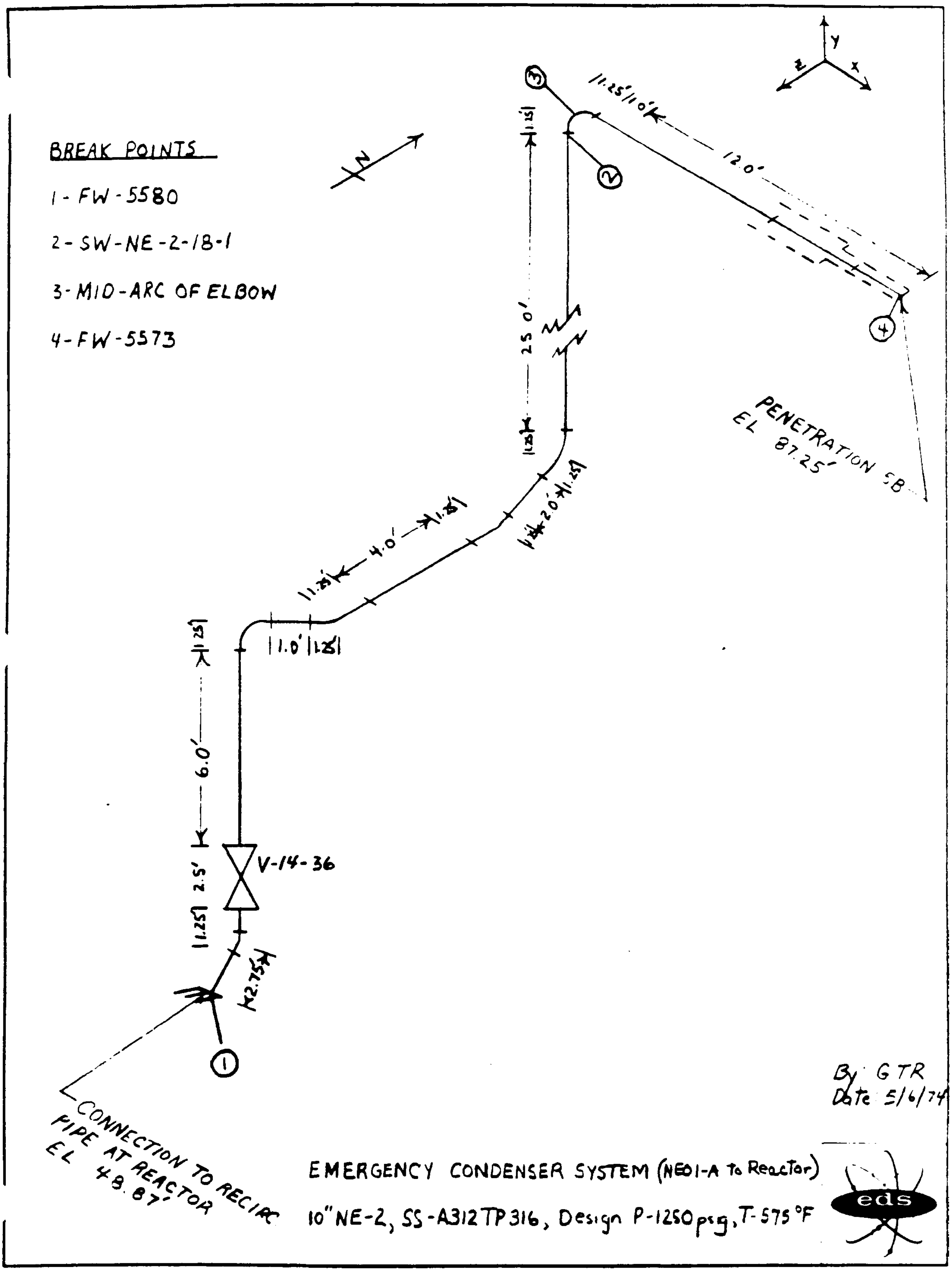
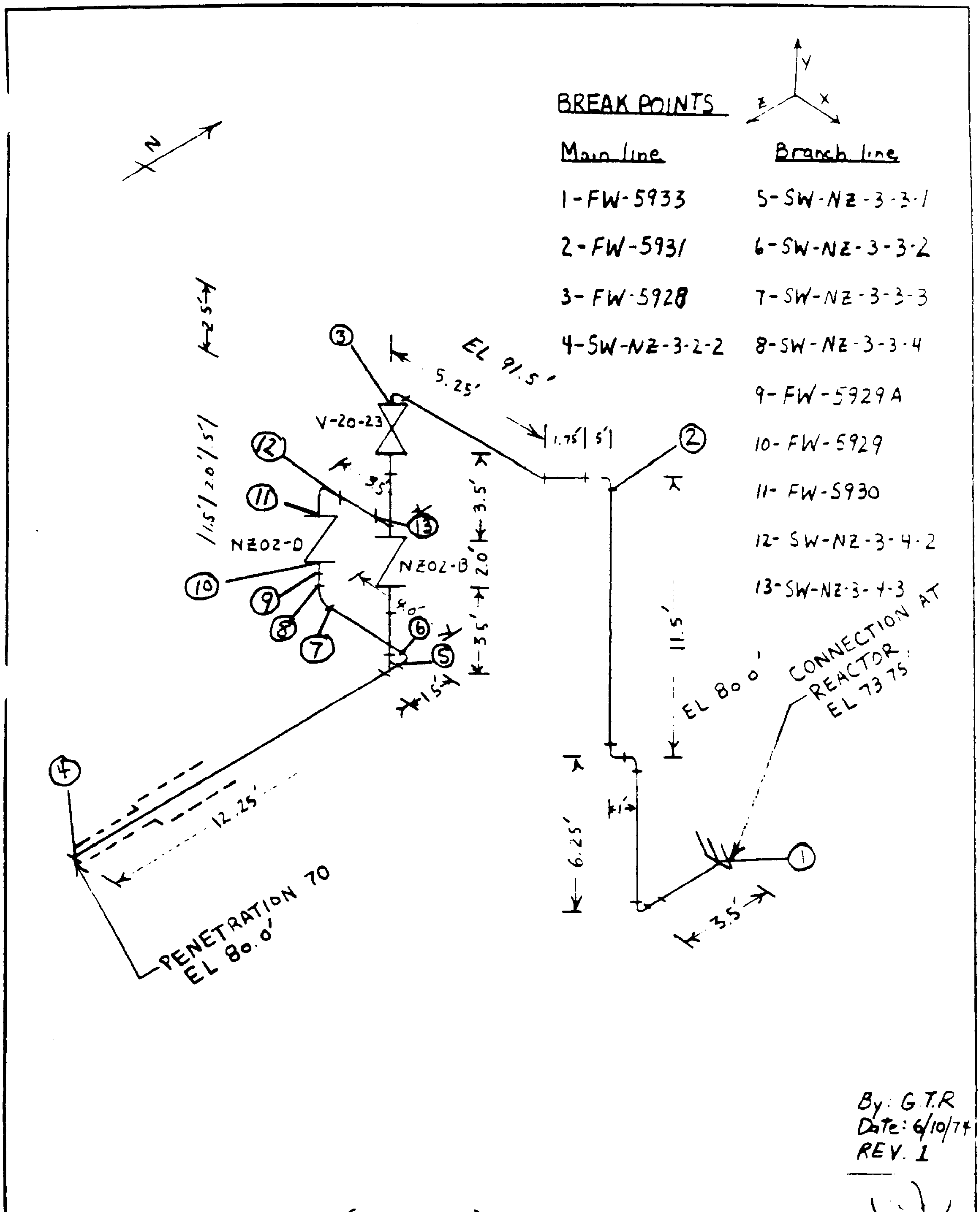

BREAK POINTS
1-FW-5580
2-SW-NE-2-18-1
3-MID-ARC OF ELBOW
4-FW-5573
Y
Z
X
N
11.25' 11.0'
12.0'
25.0'
PENETRATION 5B
EL 87.25'
2.0'
4.0'
11.0'
6.0'
2.5'
V-14-36
2.75
CONNECTION TO RECIRC
PIPE AT REACTOR
EL 48.87'
By: GTR
Date 5/6/74
EMERGENCY CONDENSER SYSTEM (NE01-A To Reactor)
10"NE-2, SS-A312TP316, Design P-1250psig, T-575°F
eds

Figure B-4




CORE SPRAY SYSTEM (SOUTH SIDE)

8" NZ-3, SS-A312 TP 316, Design P-1250 psig, T-575°F

Figure B-5

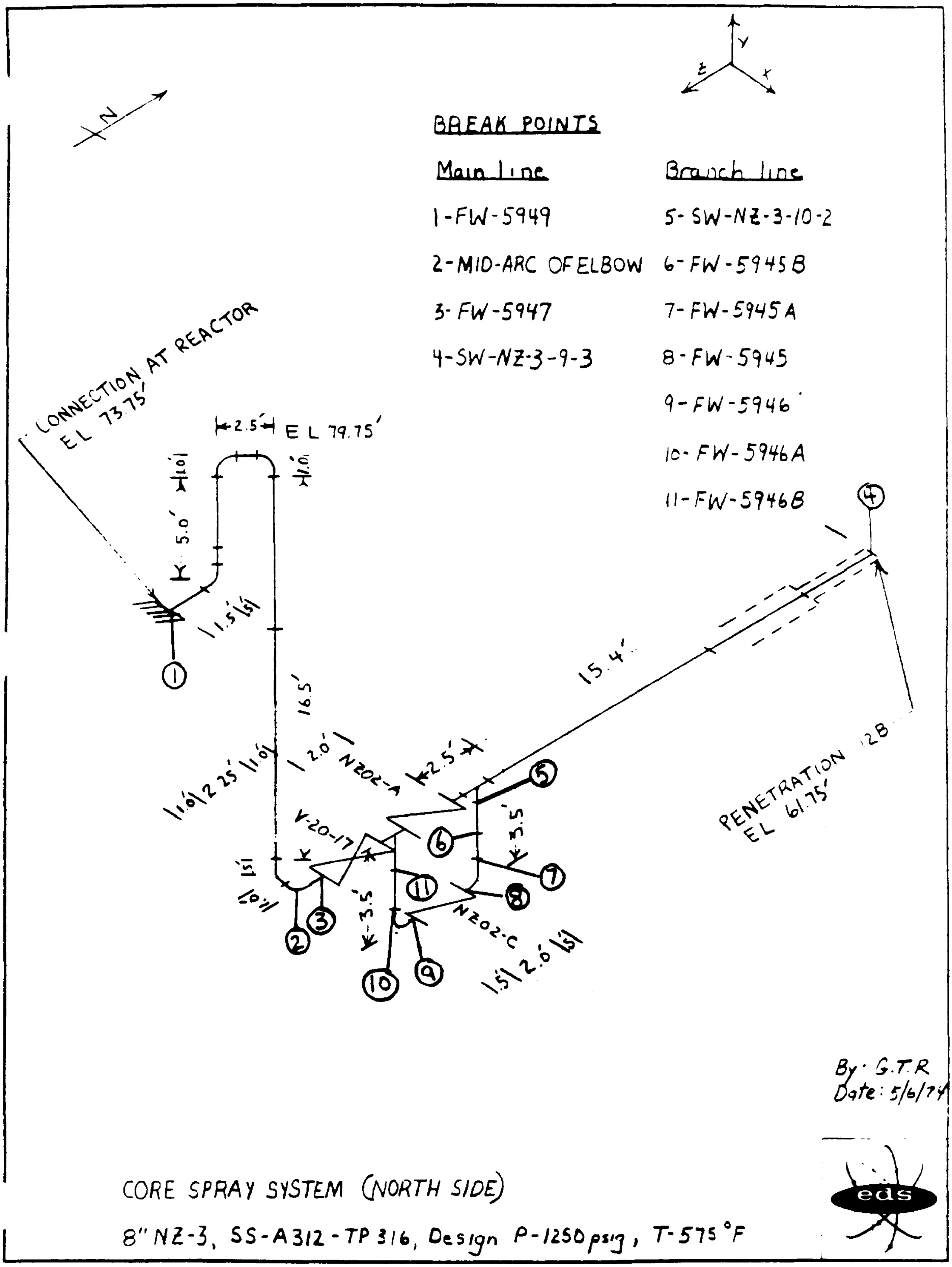
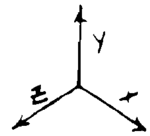

BREAK POINTS
Main line
1-FW-5949
2-MID-ARC OF ELBOW
3-FW-5947
4-SW-NZ-3-9-3
Branch line
5-SW-NZ-3-10-2
6-FW-5945B
7-FW-5945A
8-FW-5945
9-FW-5946
10-FW-5946A
11-FW-5946B
N
CONNECTION AT REACTOR
EL 73.75'
EL 79.75'
2.5'
1.0'
5.0'
1.5'
16.5'
2.0'
2.25'
NZ02-A
V-20-17
NZ02-C
3.5'
2.5'
15.4'
PENETRATION 12B
EL 61.75'
By: G.T.R
Date: 5/6/74
CORE SPRAY SYSTEM (NORTH SIDE)
8" NZ-3, SS-A312-TP316, Design P-1250 psig, T-575 °F

Figure B-6

BREAK POINTS

| South loop | North loop |
|---|---|
| 1-FW-1002 | FW-1001 |
| 2-SW-1162 | SW-1161 |
| 3-FW-1006 | FW-1005 |
| 4-SW-1376 | SW-1873 |



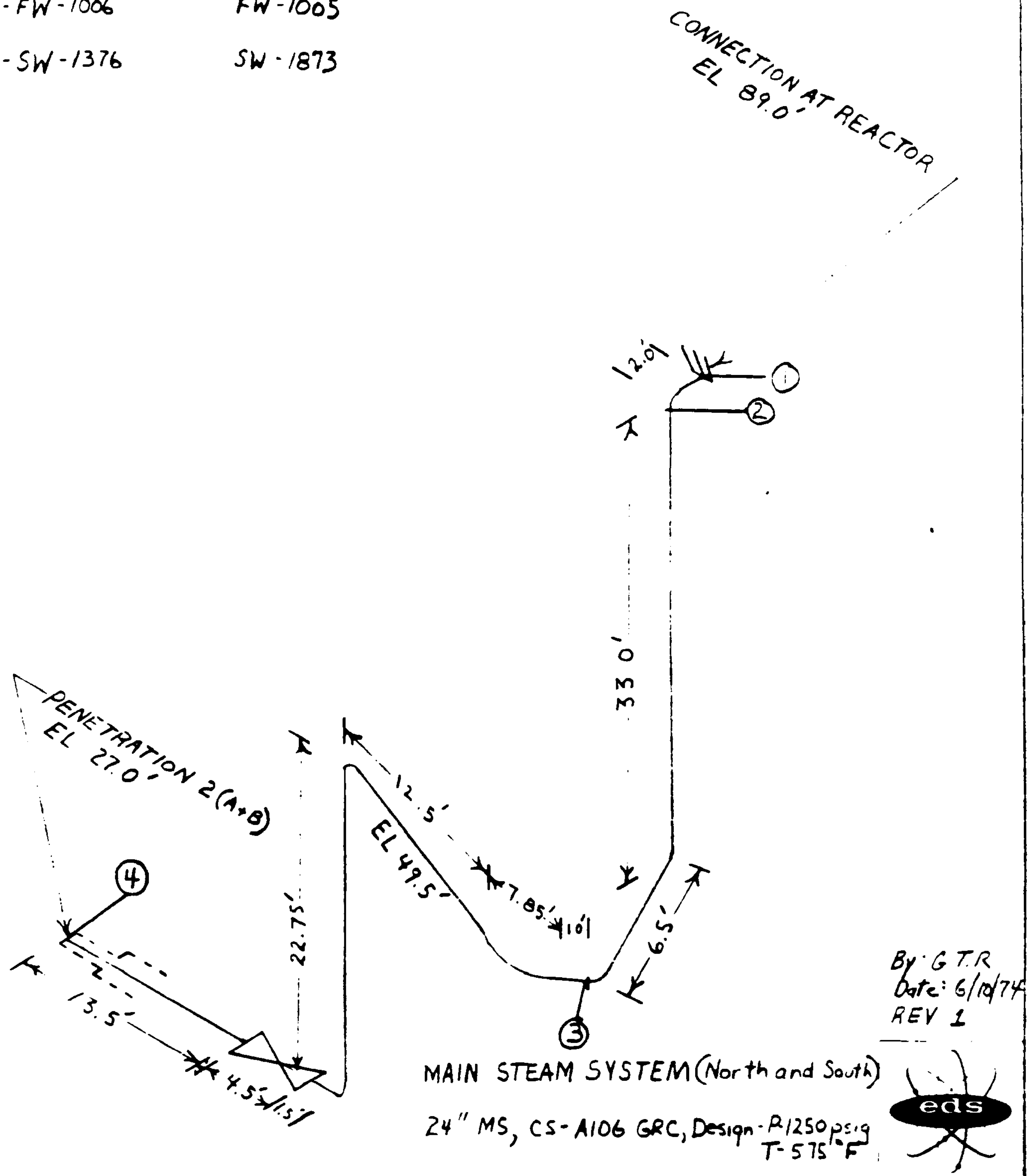




Figure B-7

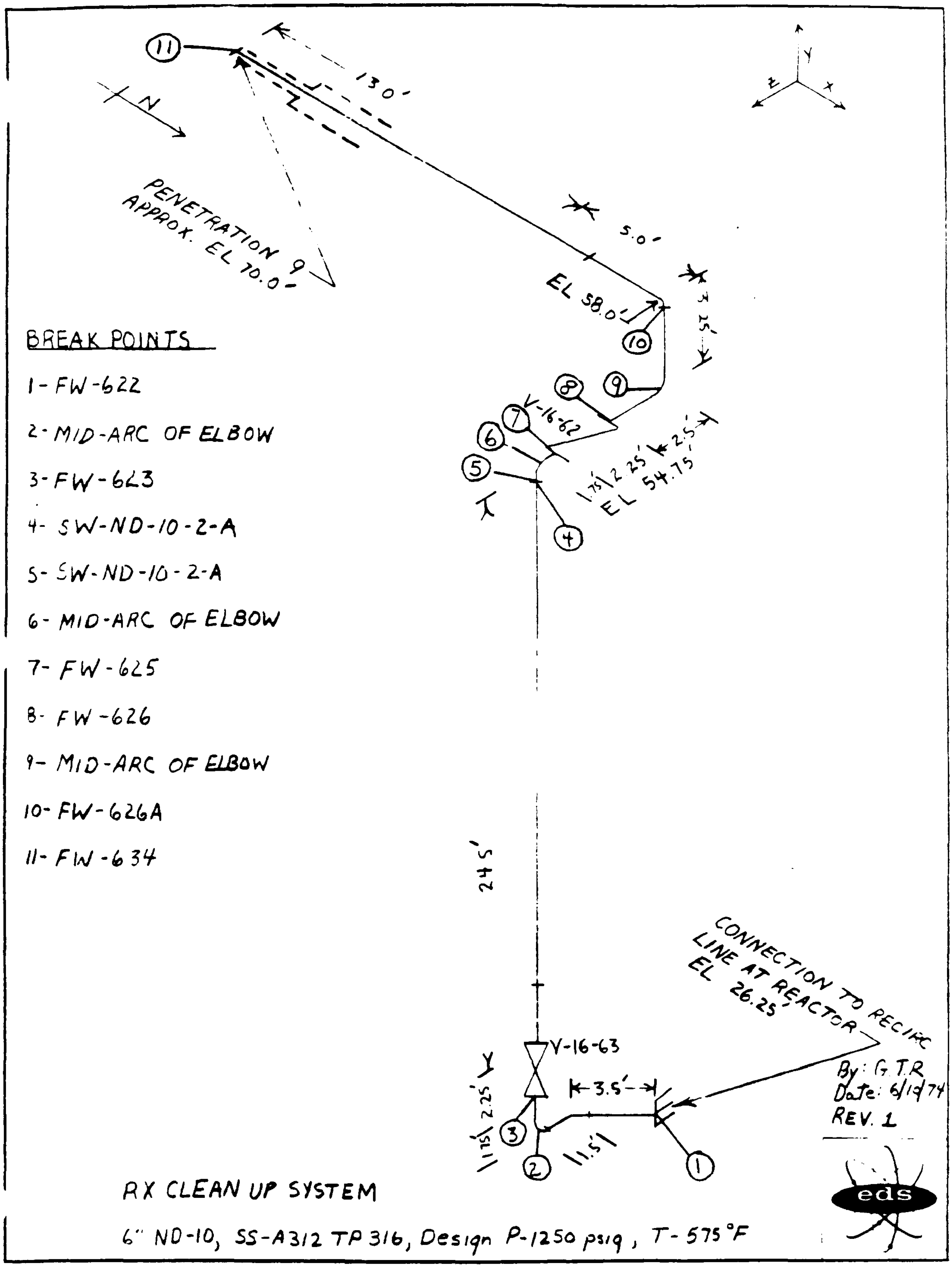
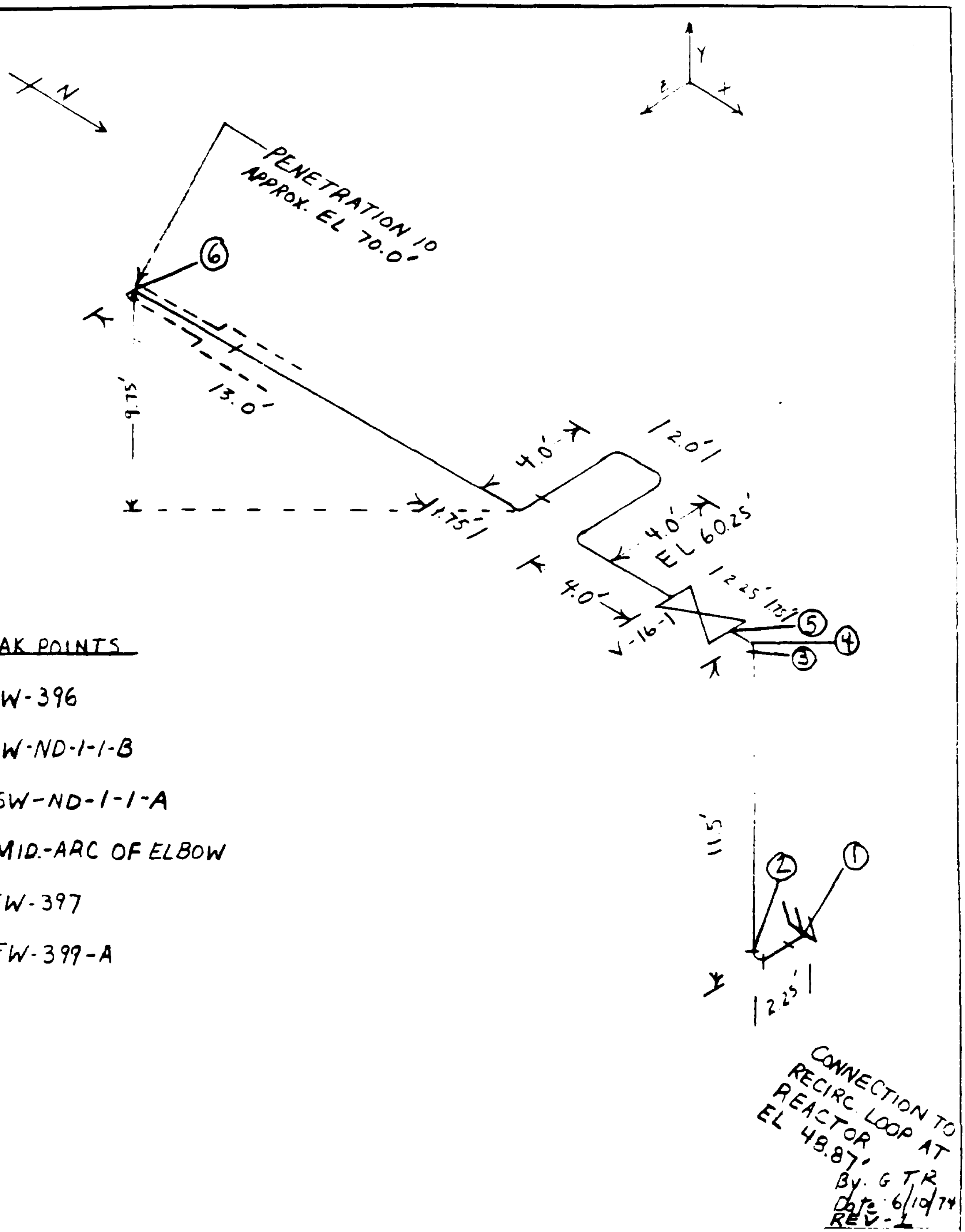

11
13.0'
N
PENETRATION 9
APPROX. EL 70.0'
5.0'
EL 58.0'
3.25'
10
9
8
7
V-16-62
6
5
4
2.5'
1.75' 2.25'
EL 54.75'
BREAK POINTS
1- FW-622
2- MID-ARC OF ELBOW
3- FW-623
4- SW-ND-10-2-A
5- SW-ND-10-2-A
6- MID-ARC OF ELBOW
7- FW-625
8- FW-626
9- MID-ARC OF ELBOW
10- FW-626A
11- FW-634
24.5'
CONNECTION TO RECIRC
LINE AT REACTOR
EL 26.25'
V-16-63
3.5'
2.25'
1.75'
1.5'
3
2
1
By: G.T.R.
Date: 6/17/74
REV. 1
eds
RX CLEAN UP SYSTEM
6" ND-10, SS-A312 TP 316, Design P-1250 psig, T-575°F

Figure B-8




BREAK POINTS

1-FW-396

2-SW-ND-1-1-B

3-SW-ND-1-1-A

4-MID-ARC OF ELBOW

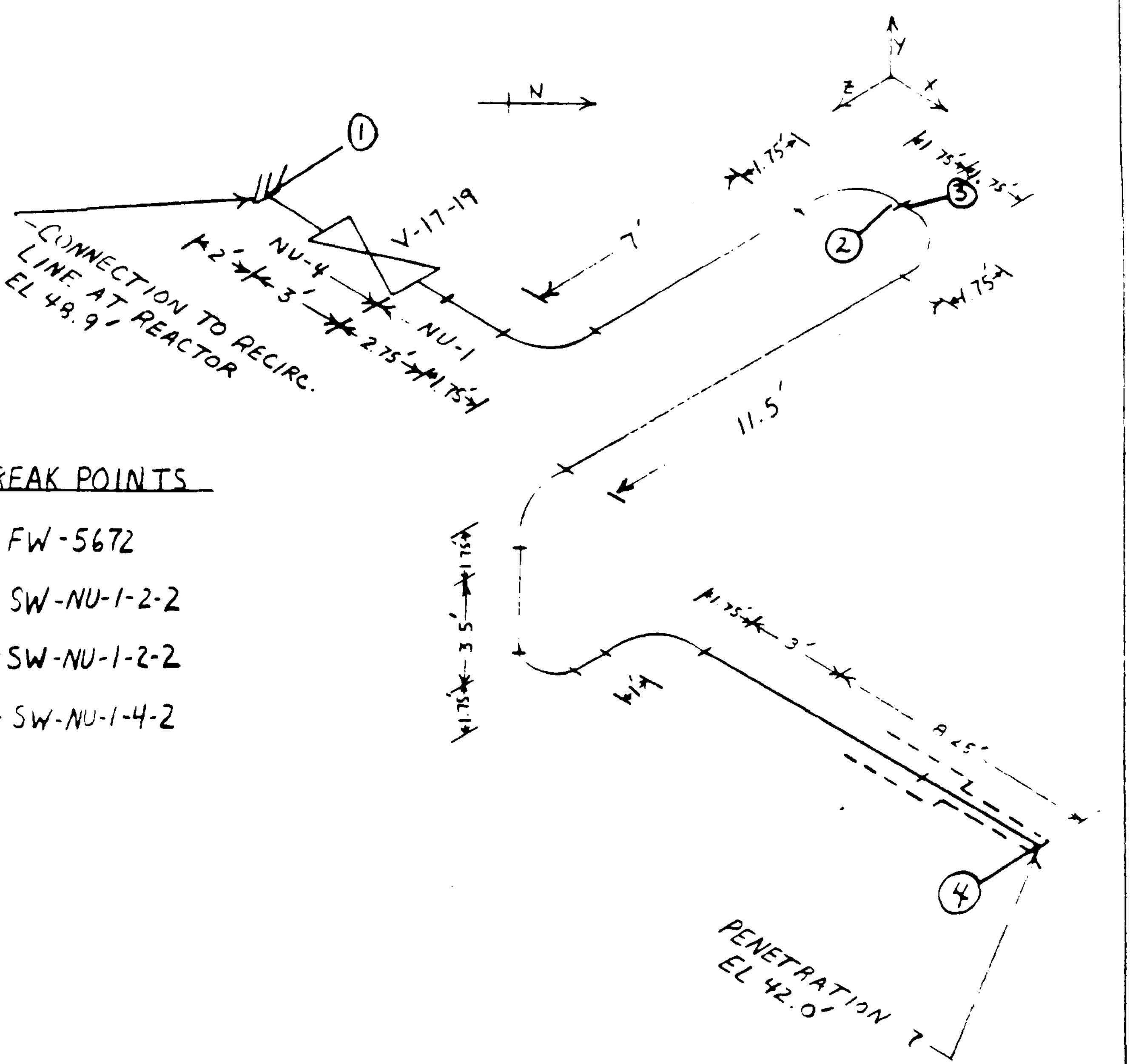
5-FW-397

6-FW-399-A

RX CLEANUP SYSTEM

6" ND-1, SS-A312 TP316, Design P-1250 psig, T-575 °F

Figure B-9




BREAK POINTS

1- FW -5672

2- SW -NU-1-2-2

3- SW -NU-1-2-2

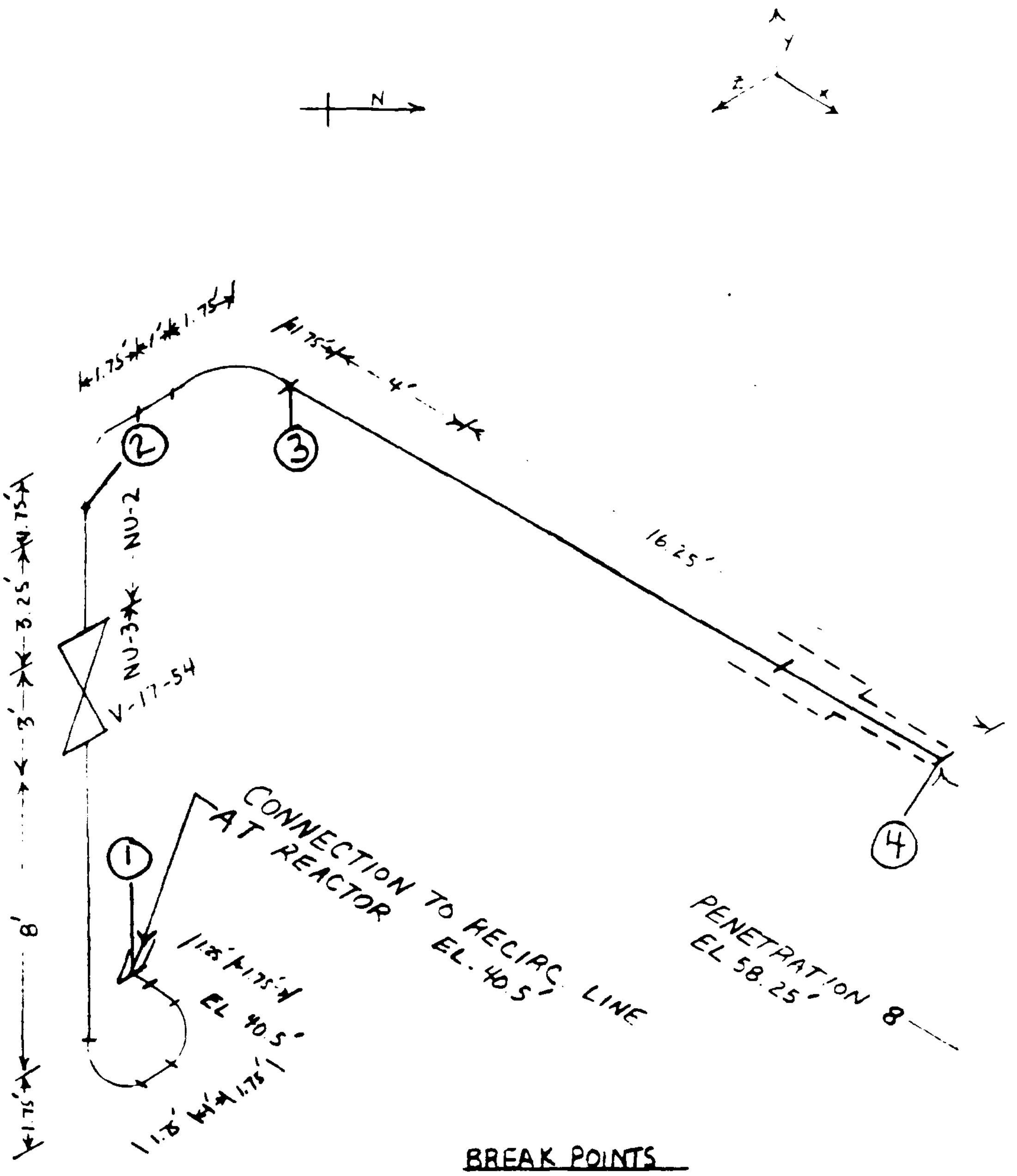
4- SW -NU-1-4-2

By: GTR
Date: 5/6/74

SHUTDOWN COOLING SYSTEM

14" NU-1, CS- A106 GRC, Design P- 1250 psig, T- 575 °F
14" NU-4, SS- A312 TP 316, Design P- 1250 psig, T- 575 °F

Figure B-10




BREAK POINTS

1- FW-5701

2- SW-NU-2-19-3

3- FW-5704

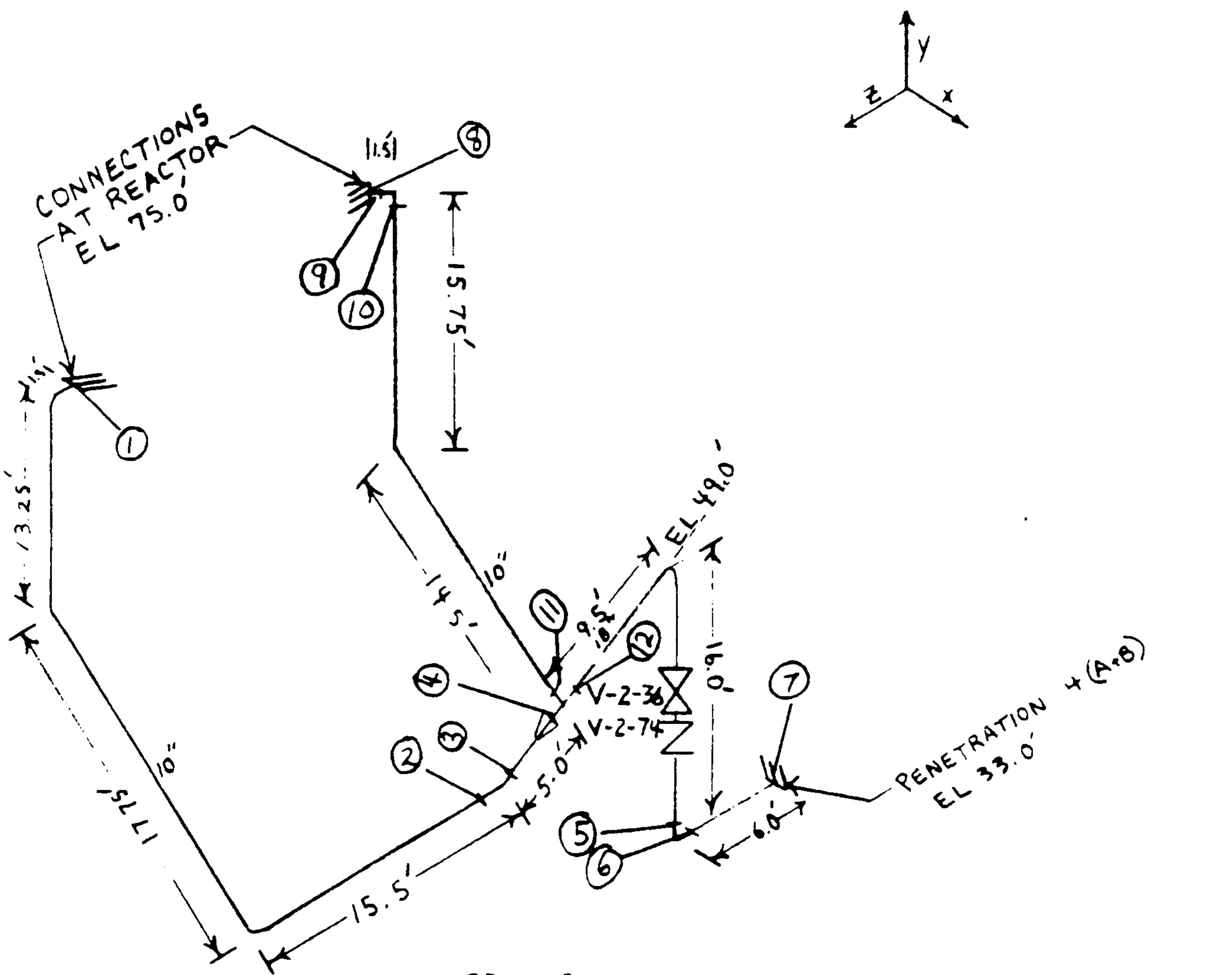
4- FW-5706

By: G T.R
Date: 5/6/74

SHUTDOWN COOLING SYSTEM

14" NU-2, CS-A106 GRC, Design P-1250 psig, T-575 °F
14" NU-3, SS-A312 TP316, Design P-1250 psig, T-575 °F

Figure B-11




## BREAK POINTS

| North loop Main line | North loop Branch line | South loop Main line | South loop Branch line |
|---|---|---|---|
| 1-FW-1278 | 8-FW-1269 | 1-FW-1287 | 8-FW-1283 |
| 2-SW-1007 | 9-SW-1013 | 2-SW-1022 | 9-SW-1028 |
| 3-FW-1270 | 10-SW-1012 | 3-FW-1284 | 10-SW-1027 |
| 4-SW-1476 | 11-FW-1267 | 4-SW-1477 | 11-FW-1281 |
| 5-SW-1172A | | 5-SW-1173A | |
| 6-FW-1263 | | 6-FW-1277 | |
| 7-SW-1364 | | 7-SW-1376 | |

By: G.T.R
Date: 6/10/74
Rev-1

FEEDWATER SYSTEM (North and South)

10" and 18" RF-2, CS-A106 GRC, Design P-1250 psig
T-575 °F

eds

Figure B-12

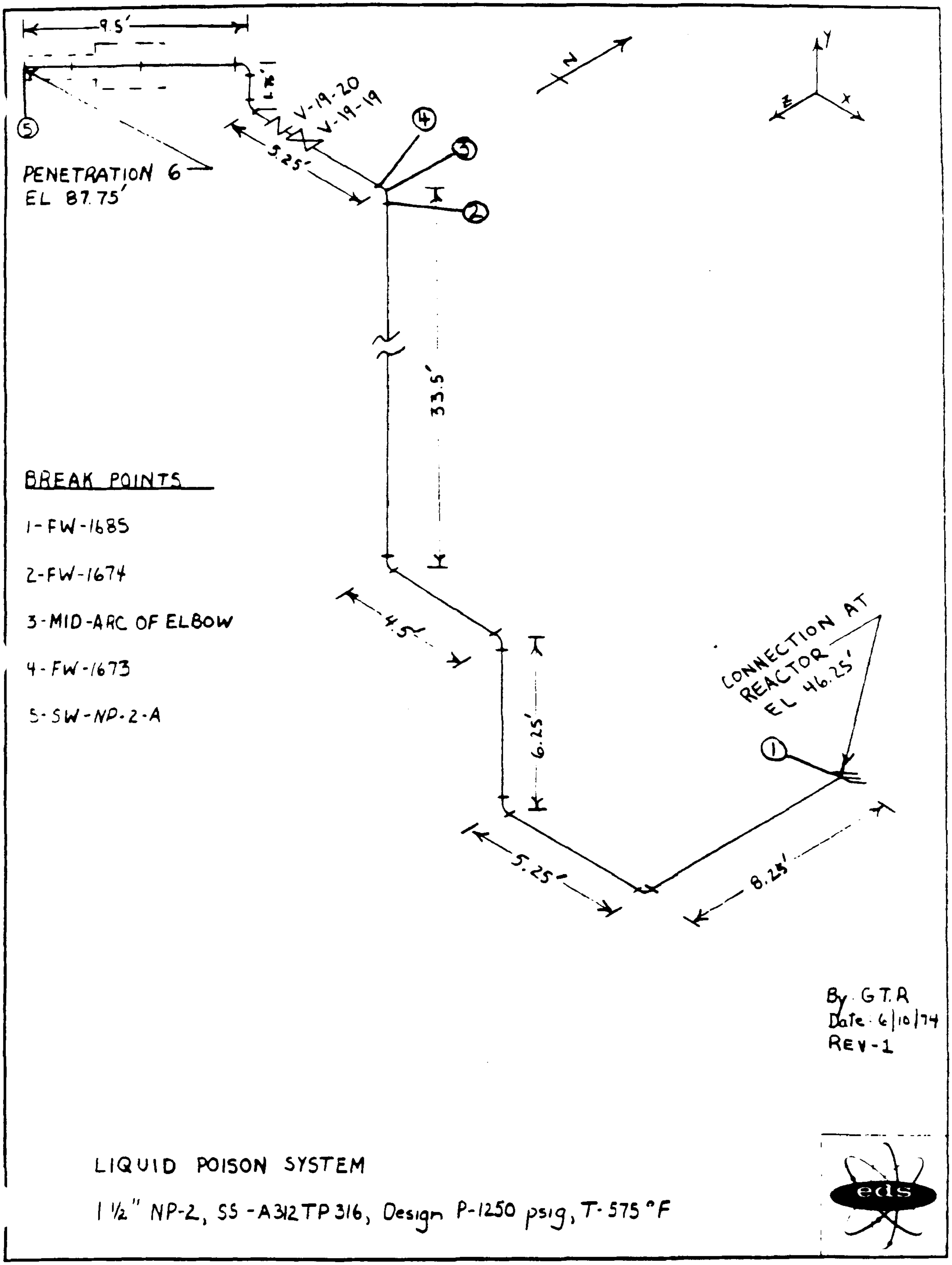
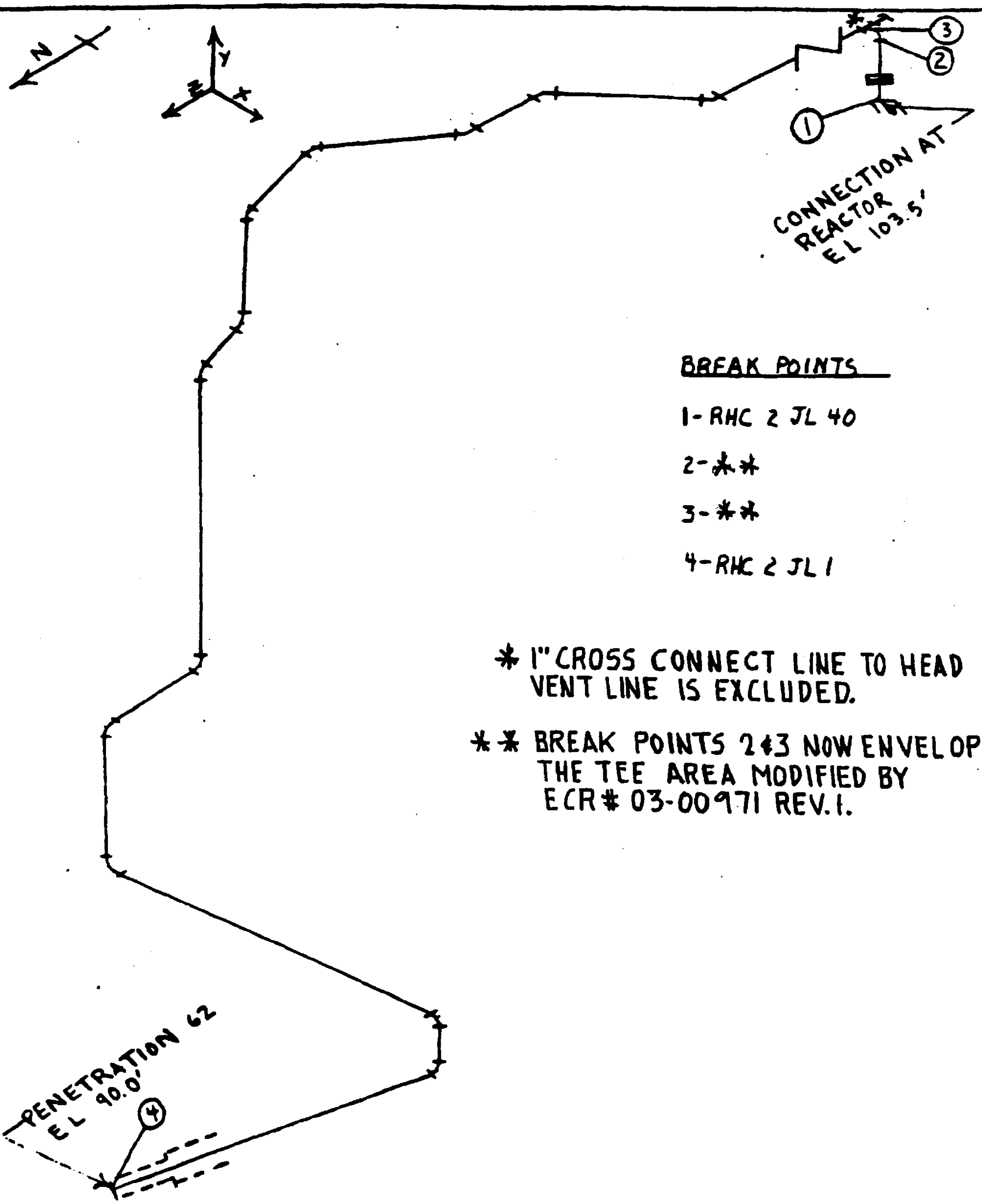

9.5'
1.5'
V-19-20
V-19-19
5.25'
PENETRATION 6
EL 87.75'
33.5'
BREAK POINTS
1-FW-1685
2-FW-1674
3-MID-ARC OF ELBOW
4-FW-1673
5-SW-NP-2-A
4.5'
6.25'
CONNECTION AT
REACTOR
EL 46.25'
5.25'
8.25'
By GTR
Date 6/10/74
REV-1
LIQUID POISON SYSTEM
1 ½" NP-2, SS-A312TP316, Design P-1250 psig, T-575 °F
eds

Figure B-13




REACTOR HEAD COOLING SYSTEM

2" RHC-2, SS-A312TP316, Design P-1250psig, T-575 °F

Figure B-14

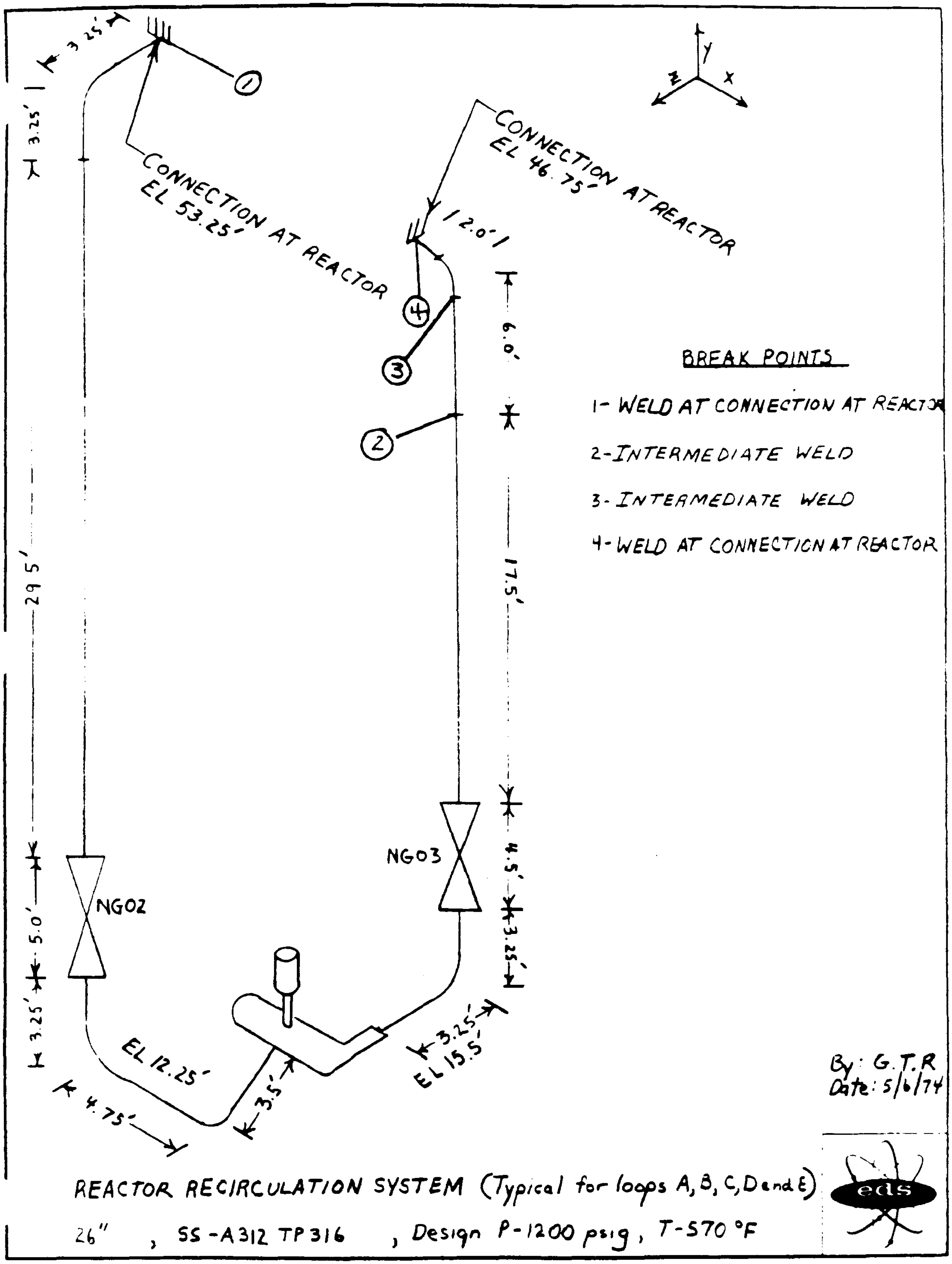
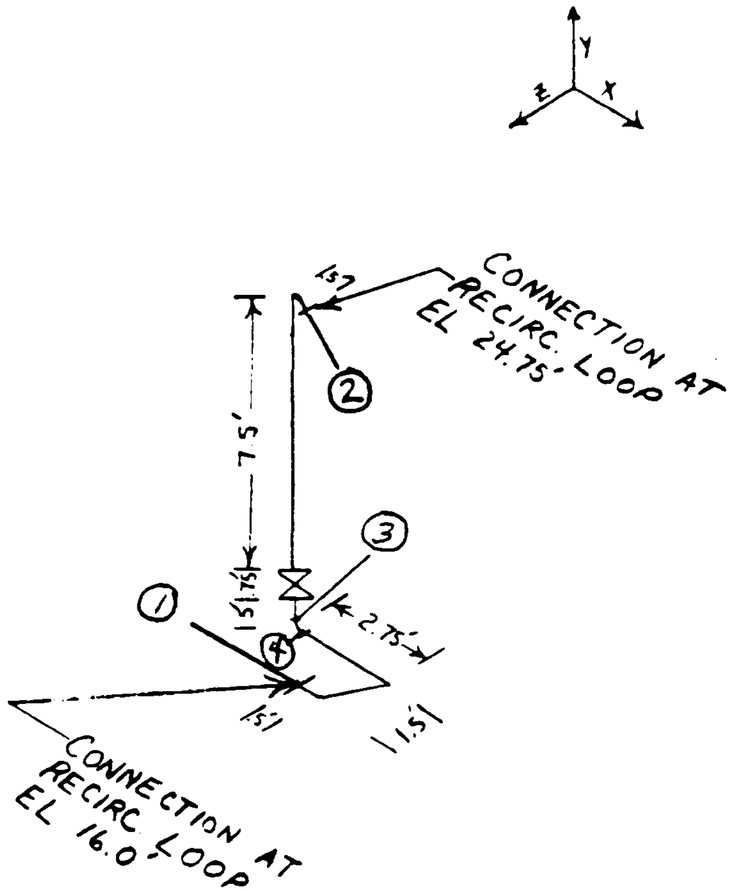

3.25'
3.25'
CONNECTION AT REACTOR
EL 53.25'
CONNECTION AT REACTOR
EL 46.75'
2.0'
6.0'
17.5'
29.5'
5.0'
3.25'
4.5'
3.25'
NG02
NG03
EL 12.25'
4.75'
3.5'
3.25'
EL 15.5'
y
x
z
BREAK POINTS
1- WELD AT CONNECTION AT REACTOR
2-INTERMEDIATE WELD
3- INTERMEDIATE WELD
4- WELD AT CONNECTION AT REACTOR
By: G.T.R
Date: 5/6/74
REACTOR RECIRCULATION SYSTEM (Typical for loops A, B, C, D and E)
26" , SS-A312 TP316 , Design P-1200 psig, T-570 °F
eds

Figure B-15




BREAK POINTS

1- WELD AT CONNECTION TO RECIRC LOOP (EL 16.0')

2- WELD AT CONNECTION TO RECIRC. LOOP (EL 24.75')

3- INTERMEDIATE WELD

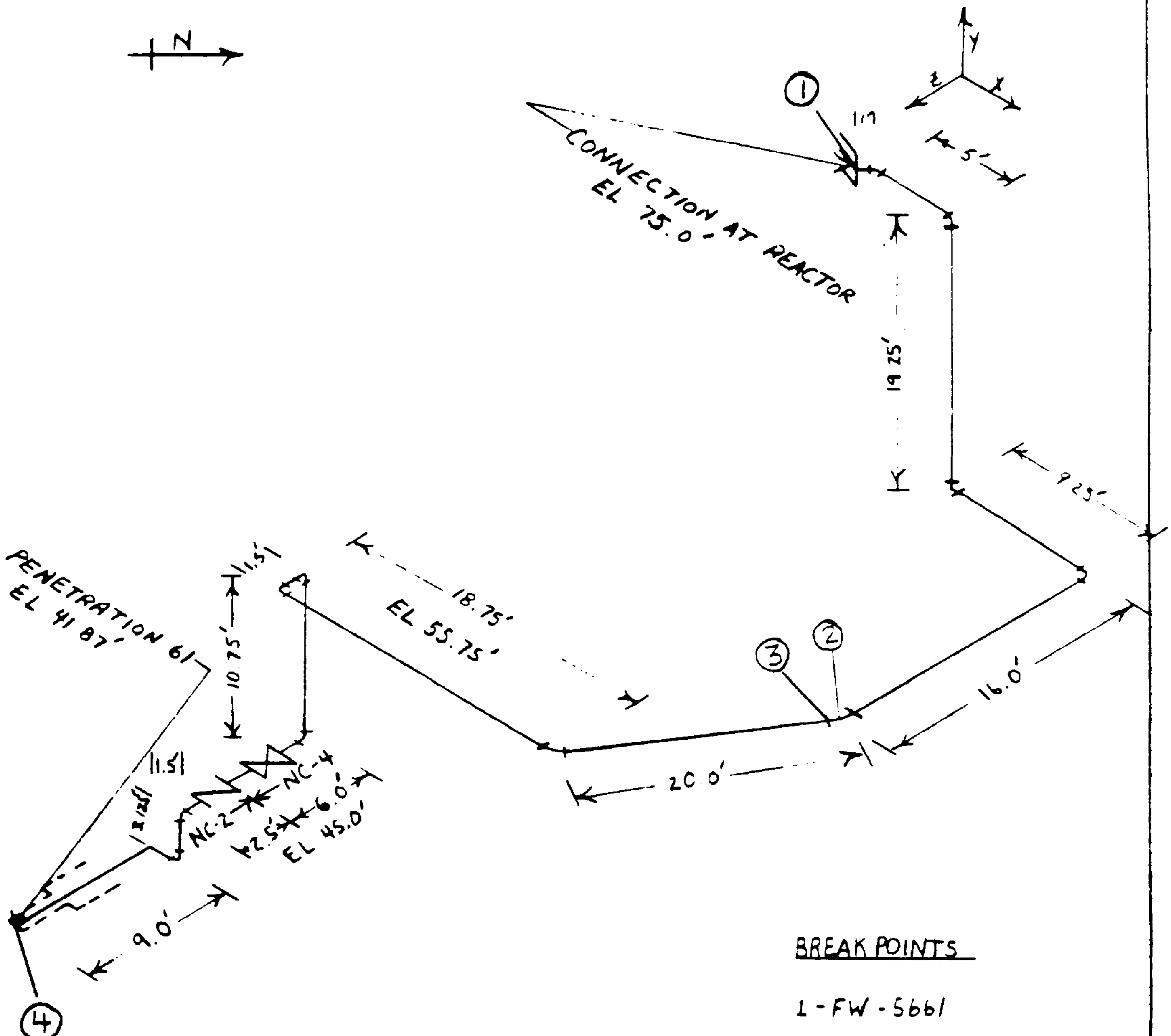
4- INTERMEDIATE WELD

By: G.T.R
Date: 5/6/74

REACTOR RECIRCULATION SYSTEM (Typical for bypass: A, B, C, D and E)

2" , SS-A312 TP 316 , Design P-1200 psig , T-150 °F

Figure B-16




BREAK POINTS

1 - FW - 5661

2 - MID-ARC OF ELBOW

3 - FW - 5664

4 - SW - NC-2-2-B

By: G.T.R.
Date: 5/6/74

CONTROL ROD DRIVE SYSTEM

3" NC-4, SS-A312 TP316, Design P-1250 psig, T-575 °F
3" NC-2, SS-A312 TP316, Design P-1250 psig, T-575 °F

Figure B-17

# APPENDIX C

## INTERACTION EVALUATION MATRICES

# APPENDIX C

## INTERACTION EVALUATION MATRICES

### List of Matrices

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| TARGET | | SOURCE System: EMERGENCY CONDENSER; Line: 10" NE-5 (to NE01-B); ISO: B-1; Bk. Pt. 1 | 2 | 3 | 4 | Line: 10" NE-5 (to NE01-A); ISO: B-2; Bk. Pt. 1 | 2 | 3 | 4 |
|---|---|---|---|---|---|---|---|---|---|
| Emergency Condenser | NE01-B 10" NE-5 | - | - | - | - | N | N | A | N |
| | NE01-A 10" NE-5 | N | A | N | N | - | - | - | - |
| | NE01-B 10" NE-2 | | N | | | N | N | N | N |
| | NE01-A 10" NE-2 | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | |
| | (North) 24" MS | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | |
| | 6" ND-1 | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | |
| | (Loop B) | | | | | | | | |
| | (Loop C) | | | | | | | | |
| | (Loop D) | | | | | | | | |
| | (Loop E) | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | |
| | (Bypass B) | | | | | | | | |
| | (Bypass C) | | | | | | | | |
| | (Bypass D) | | | | | | | | |
| | (Bypass E) | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | |
| | 12" NQ-2 | | | | | | | | |
| | 10" NQ-2 | | | | | | | | |
| | 8" NQ-2 | | | | | | | | |
| Reactor Vessel | | | | A | | | | A | |
| Biological Shield Wall | | | | N | | | A | N | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | N | | |
| | 14" (2 lines) | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | |
| Containment Vessel Shell | | | | | | | | | |
| Electrical | (Whip) | | | | | | | | |
| | (Impingement) | | | | | | | N | |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | System | EMERGENCY CONDENSER | | | | | | | |
| | Line | 10" NE-2 (from NE01-B) | | | | 10" NE-2 (from NE01-A) | | | |
| | ISO | B-3 | | | | B-4 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | |
| | NE01-B 10" NE-2 | - | - | - | - | | | | |
| | NE01-A 10" NE-2 | N | N | N | N | - | - | - | - |
| Core Spray | (South) 8" NZ-3 | | | | | N | N | N | N |
| | (North) 8" NZ-3 | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | |
| | (North) 24" MS | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | |
| | 6" ND-1 | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | A | | | | | | |
| | (South) 18" RF-2 | | N | | | | | | |
| | (North) 10" RF-2 | | | | | | A | | |
| | (North) 18" RF-2 | | | | | | N | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | |
| | (Loop B) | | | | | | | | |
| | (Loop C) | | | | | | | | |
| | (Loop D) | | | | | | | | |
| | (Loop E) | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | |
| | (Bypass B) | | | | | | | | |
| | (Bypass C) | | | | | | | | |
| | (Bypass D) | | | | | | | | |
| | (Bypass E) | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | |
| | 12" NQ-2 | | | | | | | | |
| | 10" NQ-2 | | | | | | | | |
| | 8" NQ-2 | | | | | | | | |
| Reactor Vessel | | | | | | | | | |
| Biological Shield Wall | | | | A | | | | A | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | N | | | | N | |
| | 14" (2 lines) | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | |
| Containment Vessel Shell | | | | | | | | | |
| Electrical | (Whip) | | | | | | | | |
| | (Impingement) | U | | | | U | | | |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | SOURCE | | | | | | | | | | | | |
| | System | CORE SPRAY | | | | | | | | | | | | |
| | Line | 8" NZ-3 (South) | | | | | | | | | | | | |
| | ISO | B-5 | | | | | | | | | | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | [illegible] | [illegible] | [illegible] | [illegible] |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | - | - | - | - | - | - | - | - | - | - | - | - | - |
| | (North) 8" NZ-3 | N | N | N | N | N | N | N | N | N | N | N | N | N |
| Main Steam | (South) 24" MS | | | | | | | | | | | | | |
| | (North) 24" MS | | | | | | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | | | | | | |
| | 6" ND-1 | | | | | | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | | | | | | |
| | (Loop B) | | | | | | | | | | | | | |
| | (Loop C) | | | | | | | | | | | | | |
| | (Loop D) | | | | | | | | | | | | | |
| | (Loop E) | | | | | | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | | | | | | |
| | (Bypass B) | | | | | | | | | | | | | |
| | (Bypass C) | | | | | | | | | | | | | |
| | (Bypass D) | | | | | | | | | | | | | |
| | (Bypass E) | | | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | | | |
| | 12" NQ-2 | | | | | | | | | | | | | |
| | 10" NQ-2 | | | | | | | | | | | | | |
| | 8" NQ-2 | | | | | | | | | | | | | |
| Reactor Vessel | | | | | | | | | | | | | | |
| Biological Shield Wall | | | | | | | | | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | | | | | | | | |
| | 14" (2 lines) | | | | | | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | | | | | | |
| Containment Vessel Shell | | | U | U | | N | N | N | N | N | N | N | N | N |
| Electrical | (Whip) | | N | N | | N | N | N | N | N | N | N | N | N |
| | (Impingement) | N | N | N | N | N | N | N | N | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | System | CORE SPRAY | | | | | | | | | | |
| | Line | 8" NZ-3 (North) | | | | | | | | | | |
| | ISO | B-6 | | | | | | | | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | | | | |
| | (North) 8" NZ-3 | - | - | - | - | - | - | - | - | - | - | - |
| Main Steam | (South) 24" MS | N | N | N | N | N | N | N | N | N | N | N |
| | (North) 24" MS | | | | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | | | | |
| | 6" ND-1 | | | | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | | | | |
| | (North) 10" RF-2 | | A | | | | | | | | | |
| | (North) 18" RF-2 | | N | | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | | | | |
| | (Loop B) | | | | | | | | | | | |
| | (Loop C) | | | | | | | | | | | |
| | (Loop D) | | | | | | | | | | | |
| | (Loop E) | | | | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | | | | |
| | (Bypass B) | | | | | | | | | | | |
| | (Bypass C) | | | | | | | | | | | |
| | (Bypass D) | | | | | | | | | | | |
| | (Bypass E) | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | |
| | 12" NQ-2 | | | | | | | | | | | |
| | 10" NQ-2 | | | | | | | | | | | |
| | 8" NQ-2 | | | | | | | | | | | |
| Reactor Vessel | | | | | | | | | | | | |
| Biological Shield Wall | | | A | A | | | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | | | | | | |
| | 14" (2 lines) | | | | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | | | | |
| Containment Vessel Shell | | | | | | | | | | | | |
| Electrical | (Whip) | | | | | | | | | | | |
| | (Impingement) | N | N | N | N | N | N | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | System | MAIN STEAM | | | | | | | |
| | Line | 24" MS (South) | | | | 24" MS (North) | | | |
| | ISO | B-7 | | | | B-7 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | |
| Main Steam | (South) 24" MS | - | - | - | - | | | | |
| | (North) 24" MS | N | N | N | N | - | - | - | - |
| Reactor Cleanup | 6" ND-10 | | | | | N | N | N | N |
| | 6" ND-1 | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | |
| Feedwater | (South) 10" RF-2 | U | | U | | | | | |
| | (South) 18" RF-2 | N | | N | | | | | |
| | (North) 10" RF-2 | | | | | U | | U | |
| | (North) 18" RF-2 | | | | | N | | N | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | |
| | (Loop B) | | | | | | | | |
| | (Loop C) | | | | | | | | |
| | (Loop D) | | | | | | | | |
| | (Loop E) | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | |
| | (Bypass B) | | | | | | | | |
| | (Bypass C) | | | | | | | | |
| | (Bypass D) | | | | | | | | |
| | (Bypass E) | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | U | U | |
| | (Supply) 1" NC-3 | | | | | | N | N | |
| | (Return) 3/4" NC-3 | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | |
| | 12" NQ-2 | | | | | | | | |
| | 10" NQ-2 | | | | | | | | |
| | 8" NQ-2 | | | | | | | | |
| Reactor Vessel | | | | | | | | | |
| Biological Shield Wall | | | A | A | | | A | A | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | N | N | | | N | N | |
| | 14" (2 lines) | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | |
| Containment Vessel Shell | | U | U | | | U | U | | |
| Electrical | (Whip) | N | N | | | N | N | | |
| | (Impingement) | N | N | U | N | N | N | U | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | SOURCE | | | | | | | | | | |
| | System | REACTOR CLEANUP | | | | | | | | | | |
| | Line | 6" ND-10 (To Recirculation Piping) | | | | | | | | | | |
| | ISO | B-8 | | | | | | | | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | | | | |
| | (North) 24" MS | | | | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | - | - | - | - | - | - | - | - | - | - | - |
| | 6" ND-1 | N | N | N | N | N | N | N | N | N | N | N |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | | | | |
| Reactor Vessel Head Cooling 2" RHC-2 | | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | | | | |
| | (Loop B) | | | | | | | | | | | |
| | (Loop C) | | | | | | | | | | | |
| | (Loop D) | | | | | | | | | | | |
| | (Loop E) | | | | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | | | | |
| | (Bypass B) | | | | | | | | | | | |
| | (Bypass C) | | | | | | | | | | | |
| | (Bypass D) | | | | | | | | | | | |
| | (Bypass E) | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | |
| | 12" NQ-2 | | | | | | | | | | | |
| | 10" NQ-2 | | | | | | | | | | | |
| | 8" NQ-2 | | | | A | A | A | A | | | | |
| Reactor Vessel | | | | | N | N | N | N | | | | |
| Biological Shield Wall | | | | | | | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | | | | | | |
| | 14" (2 lines) | | | | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | | | | |
| Containment Vessel Shell | | | | | U | U | | | | | | |
| Electrical | (Whip) | | | | N | N | | | | | | |
| | (Impingement) | U | U | U | N | N | N | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| TARGET | | SOURCE | | | | | |
|---|---|---|---|---|---|---|---|
| | System | REACTOR CLEANUP | | | | | |
| | Line | 6" ND-1 (From Recirculation Piping) | | | | | |
| | ISO | B-9 | | | | | |
| | Bk. Pt. | 1 | 2 | 3 | 4 | 5 | 6 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | |
| | NE01-B 10" NE-2 | | | | | | |
| | NE01-A 10" NE-2 | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | |
| | (North) 8" NZ-3 | | | | | | |
| Main Steam | (South) 24" MS | | | A | A | | |
| | (North) 24" MS | | | N | N | | |
| Reactor Cleanup | 6" ND-10 | | | | | | |
| | 6" ND-1 | - | - | - | - | - | - |
| Shutdown Cooling | 14" NU-1, NU-4 | N | N | N | N | N | N |
| | 14" NU-2, NU-3 | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | |
| | (South) 18" RF-2 | | | | | | |
| | (North) 10" RF-2 | | | | | | |
| | (North) 18" RF-2 | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | |
| | (Loop B) | | | | | | |
| | (Loop C) | | | | | | |
| | (Loop D) | | | | | | |
| | (Loop E) | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | |
| | (Bypass B) | | | | | | |
| | (Bypass C) | | | | | | |
| | (Bypass D) | | | | | | |
| | (Bypass E) | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | |
| | (Supply) 1" NC-3 | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | |
| | 12" NQ-2 | | | | | | |
| | 10" NQ-2 | | | | | | |
| | 8" NQ-2 | | | | | | |
| Reactor Vessel | | | | | | | |
| Biological Shield Wall | | | | | | A | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | N | |
| | 14" (2 lines) | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | |
| Containment Vessel Shell | | | | | | | |
| Electrical | (Whip) | | | | | | |
| | (Impingement) | N | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | System | SHUTDOWN COOLING | | | | | | | |
| | Line | 14" NU-1, NU-4 (From Recirculation Piping) | | | | 14" NU-2, NU-3 (To Recirculation Piping) | | | |
| | ISO | B-10 | | | | B-11 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | |
| | (North) 24" MS | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | |
| | 6" ND-1 | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | - | - | - | - | | | | |
| | 14" NU-2, NU-3 | N | A | A | N | - | - | - | - |
| Feedwater | (South) 10" RF-2 | | N | N | | N | N | N | N |
| | (South) 18" RF-2 | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | |
| | (Loop B) | | | | | | | | |
| | (Loop C) | | | | | | | | |
| | (Loop D) | | | | | | | | |
| | (Loop E) | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | |
| | (Bypass B) | | | | | | | | |
| | (Bypass C) | | | | | | | | |
| | (Bypass D) | | | | | | | | |
| | (Bypass E) | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | |
| | 12" NQ-2 | | | | | | | | |
| | 10" NQ-2 | | | | | | | | |
| | 8" NQ-2 | | | | | | | | |
| Reactor Vessel | | | | | | | | | |
| Biological Shield Wall | | | | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | | | |
| | 14" (2 lines) | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | |
| Containment Vessel Shell | | | | | | | | | |
| Electrical | (Whip) | | | | | | | | |
| | (Impingement) | U | N | N | N | U | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | System | FEEDWATER | | | | | | | | | | | |
| | Line | 10" RF-2 (South) | | | | 18" RF-2 (South) | | | | 10" RF-2 (South) | | | |
| | ISO | B-12 | | | | B-12 | | | | B-12 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 12 | 8 | 9 | [illegible] | [illegible] |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | | | | | |
| | (North) 8" NZ-3 | | | | | ↓ | | | | | | | ↓ |
| Main Steam | (South) 24" MS | | | | | A | | | | | | | A |
| | (North) 24" MS | ↓ | ↓ | ↓ | ↓ | N | | | | | | | N |
| Reactor Cleanup | 6" ND-10 | U | U | U | U | | | | | | | | |
| | 6" ND-1 | N | N | N | N | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | | |
| | 14" NU-2, NU-3 | ↓ | ↓ | ↓ | ↓ | | | | | ↓ | ↓ | ↓ | ↓ |
| Feedwater | (South) 10" RF-2 | - | - | - | - | ↓ | ↓ | ↓ | ↓ | - | - | - | - |
| | (South) 18" RF-2 | N | N | N | N | - | - | - | - | N | N | N | N |
| | (North) 10" RF-2 | | | | | N | N | N | N | | | | |
| | (North) 18" RF-2 | | | | | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | | | | | |
| Reactor Vessel Head Cooling 2" RHC-2 | | | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | | | | | |
| | (Loop B) | | | | | | ↓ | ↓ | | | | | |
| | (Loop C) | | | | | | A | A | | | | | |
| | (Loop D) | | | | | | N | N | | | | | |
| | (Loop E) | | | | | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | | | | | |
| | (Bypass B) | | | | | | | | | | | | |
| | (Bypass C) | | | | | | | | | | | | |
| | (Bypass D) | | | | | | | | | | | | |
| | (Bypass E) | | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | | |
| | 12" NQ-2 | | | | | | | | | | | | |
| | 10" NQ-2 | ↓ | | | | | | | | ↓ | ↓ | | |
| | 8" NQ-2 | U | | | | | | | | U | U | | |
| Reactor Vessel | | N | | | | | ↓ | ↓ | | N | N | ↓ | ↓ |
| Biological Shield Wall | | | | | | | A | A | | | | A | A |
| Main Steam Relief Valve Discharge | | | | | | | | | | | | | |
| | 8" (4 lines) | | | | | | N | N | | | | N | N |
| | 14" (2 lines) | | | | | | A | A | | | | | |
| Main Steam Safety Valve Discharge | | | | | | | | | | | | | |
| | 8" (16 lines) | | | | | ↓ | N | N | ↓ | | | | |
| Containment Vessel Shell | | | | | | U | U | | U | | | | |
| Electrical | (Whip) | ↓ | ↓ | ↓ | ↓ | N | N | ↓ | N | ↓ | ↓ | ↓ | ↓ |
| | (Impingement) | N | U | U | U | N | U | U | U | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| TARGET | | SOURCE | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| System | | FEEDWATER | | | | | | | | | | | |
| Line | | 10" RF-2 (North) | | | | 18" RF-2 (North) | | | | 10" RF-2 (North) | | | |
| ISO | | B-12 | | | | B-12 | | | | B-12 | | | |
| Bk. Pt. | | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 12 | 8 | 9 | [illegible] | [illegible] |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | | | | | |
| | (North) 24" MS | | | | | A | | | | | | | A |
| Reactor Cleanup | 6" ND-10 | | | | | N | | | | | | | N |
| | 6" ND-1 | | | | | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | | | | | |
| | (North) 10" RF-2 | - | - | - | - | | | | | - | - | - | - |
| | (North) 18" RF-2 | N | N | N | N | - | - | - | - | N | N | N | N |
| Liquid Poison - 1.5" NP-2 | | | | | | N | N | N | N | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | | | | | |
| | (Loop B) | | | | | | | | | | | | |
| | (Loop C) | | | | | | | | | | | | |
| | (Loop D) | | | | | | A | A | | | | | |
| | (Loop E) | | | | | | N | N | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | | | | | |
| | (Bypass B) | | | | | | | | | | | | |
| | (Bypass C) | | | | | | | | | | | | |
| | (Bypass D) | | | | | | | | | | | | |
| | (Bypass E) | | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | U | U | U | U | | | | | | | | |
| | (Supply) 1" NC-3 | N | N | N | N | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | | |
| | 12" NQ-2 | | | | | | | | | | | | |
| | 10" NQ-2 | | | | | | | | | | | | |
| | 8" NQ-2 | U | | | | | | | | U | U | | |
| Reactor Vessel | | N | | | | | | | | N | N | | |
| Biological Shield Wall | | | | | | | A | A | | | | A | A |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | N | N | | | | N | N |
| | 14" (2 lines) | | | | | | A | A | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | N | N | | | | | |
| Containment Vessel Shell | | | | | | U | U | | U | | | | |
| Electrical | (Whip) | | | | | N | N | | N | | | | |
| | (Impingement) | N | U | U | U | N | U | U | U | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | |
|---|---|---|---|---|---|---|
| | System | LIQUID POISON | | | | |
| | Line | 1.5" NP-2 | | | | |
| | ISO | B-13 | | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 5 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | |
| | NE01-B 10" NE-2 | | | | | |
| | NE01-A 10" NE-2 | | | | | |
| Core Spray | (South) 8" NZ-3 | | | A | A | |
| | (North) 8" NZ-3 | | | N | N | |
| Main Steam | (South) 24" MS | | | | | |
| | (North) 24" MS | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | |
| | 6" ND-1 | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | |
| | 14" NU-2, NU-3 | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | |
| | (South) 18" RF-2 | | | | | |
| | (North) 10" RF-2 | | | | | |
| | (North) 18" RF-2 | | | | | |
| Liquid Poison - 1.5" NP-2 | | - | - | - | - | - |
| Reactor Vessel Head Cooling | 2" RHC-2 | N | N | N | N | N |
| Reactor Recirc. | (Loop A) | | | | | |
| | (Loop B) | | | | | |
| | (Loop C) | | | | | |
| | (Loop D) | | | | | |
| | (Loop E) | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | |
| | (Bypass B) | | | | | |
| | (Bypass C) | | | | | |
| | (Bypass D) | | | | | |
| | (Bypass E) | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | |
| | (Supply) 1" NC-3 | | | | | |
| | (Return) 3/4" NC-3 | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | |
| | 12" NQ-2 | | | | | |
| | 10" NQ-2 | | | | | |
| | 8" NQ-2 | | | | | |
| Reactor Vessel | | | | | | |
| Biological Shield Wall | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | |
| | 14" (2 lines) | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | |
| Containment Vessel Shell | | | U | | | |
| Electrical | (Whip) | | N | | | |
| | (Impingement) | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | SOURCE | | | |
|---|---|---|---|---|
| System | REACTOR VESSEL HEAD COOLING | | | |
| Line | 2" RHC-2 | | | |
| ISO | B-14 | | | |
| Bk. Pt. | 1 | 2 | 3 | 4 |

| TARGET | | 1 | 2 | 3 | 4 |
|---|---|---|---|---|---|
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N |
| | NE01-A 10" NE-5 | | | | |
| | NE01-B 10" NE-2 | | | | |
| | NE01-A 10" NE-2 | | | | |
| Core Spray | (South) 8" NZ-3 | | | | |
| | (North) 8" NZ-3 | | | | |
| Main Steam | (South) 24" MS | | | | |
| | (North) 24" MS | | | | |
| Reactor Cleanup | 6" ND-10 | | | | |
| | 6" ND-1 | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | |
| | 14" NU-2, NU-3 | | | | |
| Feedwater | (South) 10" RF-2 | | | | |
| | (South) 18" RF-2 | | | | |
| | (North) 10" RF-2 | | | | |
| | (North) 18" RF-2 | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | - | - | - | - |
| Reactor Recirc. | (Loop A) | N | N | N | N |
| | (Loop B) | | | | |
| | (Loop C) | | | | |
| | (Loop D) | | | | |
| | (Loop E) | | | | |
| Reactor Recirc.. | (Bypass A) | | | | |
| | (Bypass B) | | | | |
| | (Bypass C) | | | | |
| | (Bypass D) | | | | |
| | (Bypass E) | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | |
| | (Supply) 1" NC-3 | | | | |
| | (Return) 3/4" NC-3 | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | |
| | 12" NQ-2 | | | | |
| | 10" NQ-2 | | | | |
| | 8" NQ-2 | | | | |
| Reactor Vessel | | | | | |
| Biological Shield Wall | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | |
| | 14" (2 lines) | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | |
| Containment Vessel Shell | | | | | |
| Electrical | (Whip) | | | | |
| | (Impingement) | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| TARGET | | SOURCE | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| System | | REACTOR RECIRCULATION LOOP | | | | | | | | | | | |
| Line | | Loop A | | | | Loop B | | | | Loop C | | | |
| ISO | | B-15 | | | | B-15 | | | | B-15 | | | |
| Bk. Pt. | | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | U | | | | | | | |
| | (North) 24" MS | | | | | N | | | | U | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | | N | | | |
| | 6" ND-1 | | | | | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | - | - | - | - | | A | A | | | | | |
| | (Loop B) | N | N | N | N | - | - | - | - | | A | A | |
| | (Loop C) | | | | | N | N | N | N | - | - | - | - |
| | (Loop D) | | | | | | | | | N | N | N | N |
| | (Loop E) | | A | A | | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | N | N | | | | | | | | | |
| | (Bypass B) | | | | | | | | | | | | |
| | (Bypass C) | | | | | | | | | | | | |
| | (Bypass D) | | | | | | | | | | | | |
| | (Bypass E) | | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | | |
| | 12" NQ-2 | U | | | | | | | | | | | |
| | 10" NQ-2 | N | | | | | | | | | | | |
| | 8" NQ-2 | | | | | | | | | | | | |
| Reactor Vessel | | | | | | | | | | | | | |
| Biological Shield Wall | | | A | A | | | A | A | | | A | A | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | N | N | | | N | N | | | N | N | |
| | 14" (2 lines) | | | | | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | | | | | |
| Containment Vessel Shell | | | | | U | | | | U | | | | |
| Electrical | (Whip) | U | | | U | | | | U | N | | | |
| | (Impingement) | A | U | U | U | A | U | U | U | A | U | U | U |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | System | REACTOR RECIRCULATION LOOP | | | | | | | |
| | Line | Loop D | | | | Loop E | | | |
| | ISO | B-15 | | | | B-15 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | |
| | (North) 24" MS | U | | | | | | | |
| Reactor Cleanup | 6" ND-10 | N | | | | | | | |
| | 6" ND-1 | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | |
| Reactor Vessel Head Cooling 2" RHC-2 | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | |
| | (Loop B) | | | | | | | | |
| | (Loop C) | | A | A | | | | | |
| | (Loop D) | - | - | - | - | | A | A | |
| | (Loop E) | N | N | N | N | - | - | - | - |
| Reactor Recirc. | (Bypass A) | | | | | N | N | N | N |
| | (Bypass B) | | | | | | | | |
| | (Bypass C) | | | | | | | | |
| | (Bypass D) | | | | | | | | |
| | (Bypass E) | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | |
| | 12" NQ-2 | | | | | | | | |
| | 10" NQ-2 | | | | | | | | |
| | 8" NQ-2 | | | | | | | | |
| Reactor Vessel | | | | | | | | | |
| Biological Shield Wall | | | A | A | | | A | A | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | N | N | | | N | N | |
| | 14" (2 lines) | | | | A | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | N | | | | |
| Containment Vessel Shell | | | | | U | U | | | U |
| Electrical | (Whip) | | | | U | U | | | U |
| | (Impingement) | A | U | U | U | A | U | U | U |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | System | REACTOR RECIRCULATION BYPASS | | | | | | | | | | | |
| | Line | Bypass A | | | | Bypass B | | | | Bypass C | | | |
| | ISO | B-16 | | | | B-16 | | | | B-16 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | | | | | |
| | (North) 24" MS | | | | | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | | | | | |
| | 6" ND-1 | | | | | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | | | | | |
| | (Loop B) | | | | | | | | | | | | |
| | (Loop C) | | | | | | | | | | | | |
| | (Loop D) | | | | | | | | | | | | |
| | (Loop E) | | | | | | | | | | | | |
| Reactor Recirc. | (Bypass A) | - | - | - | - | | | | | | | | |
| | (Bypass B) | N | N | N | N | - | - | - | - | | | | |
| | (Bypass C) | | | | | N | N | N | N | - | - | - | - |
| | (Bypass D) | | | | | | | | | N | N | N | N |
| | (Bypass E) | | | | | | | | | | | | |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | | | | | | | | |
| | (Supply) 1" NC-3 | | | | | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | | | | | |
| | 12" NQ-2 | | | | | | | | | | | | |
| | 10" NQ-2 | | | | | | | | | | | | |
| | 8" NQ-2 | | | | | | | | | | | | |
| Reactor Vessel | | | | | | | | | | | | | |
| Biological Shield Wall | | | | | | | | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | | | | | | | |
| | 14" (2 lines) | | | | | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | | | | | |
| Containment Vessel Shell | | | | | | | | | | | | | |
| Electrical | (Whip) | | | | | | | | | | | | |
| | (Impingement) | N | N | N | N | N | N | N | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | System | REACTOR RECIRCULATION BYPASS | | | | | | | |
| | Line | Bypass D | | | | Bypass E | | | |
| | ISO | B-16 | | | | B-16 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 |
| Emergency Condenser | NE01-B 10" NE-5 | N | N | N | N | N | N | N | N |
| | NE01-A 10" NE-5 | | | | | | | | |
| | NE01-B 10" NE-2 | | | | | | | | |
| | NE01-A 10" NE-2 | | | | | | | | |
| Core Spray | (South) 8" NZ-3 | | | | | | | | |
| | (North) 8" NZ-3 | | | | | | | | |
| Main Steam | (South) 24" MS | | | | | | | | |
| | (North) 24" MS | | | | | | | | |
| Reactor Cleanup | 6" ND-10 | | | | | | | | |
| | 6" ND-1 | | | | | | | | |
| Shutdown Cooling | 14" NU-1, NU-4 | | | | | | | | |
| | 14" NU-2, NU-3 | | | | | | | | |
| Feedwater | (South) 10" RF-2 | | | | | | | | |
| | (South) 18" RF-2 | | | | | | | | |
| | (North) 10" RF-2 | | | | | | | | |
| | (North) 18" RF-2 | | | | | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | | | | | |
| Reactor Recirc. | (Loop A) | | | | | | | | |
| | (Loop B) | | | | | | | | |
| | (Loop C) | | | | | | | | |
| | (Loop D) | | | | | | | | |
| | (Loop E) | | | | | | | | |
| Reactor Recirc. | (Bypass A) | | | | | | | | |
| | (Bypass B) | | | | | | | | |
| | (Bypass C) | | | | | | | | |
| | (Bypass D) | - | - | - | - | | | | |
| | (Bypass E) | N | N | N | N | - | - | - | - |
| Control Rod Drive | 3" NC-4, NC-2 | | | | | N | N | N | N |
| | (Supply) 1" NC-3 | | | | | | | | |
| | (Return) 3/4" NC-3 | | | | | | | | |
| Containment Spray | 14" CS-2, NQ-2 | | | | | | | | |
| | 12" NQ-2 | | | | | | | | |
| | 10" NQ-2 | | | | | | | | |
| | 8" NQ-2 | | | | | | | | |
| Reactor Vessel | | | | | | | | | |
| Biological Shield Wall | | | | | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines) | | | | | | | | |
| | 14" (2 lines) | | | | | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | | | | | |
| Containment Vessel Shell | | | | | | | | | |
| Electrical | (Whip) | | | | | | | | |
| | (Impingement) | N | U | N | N | N | N | N | N |

LEGEND:

U - Unacceptable Interaction (Damage Possible)

A - Acceptable Interaction (Damage not Possible)

N - No Interaction

| | | SOURCE | | | |
|---|---|---|---|---|---|
| | System | CONTROL ROD DRIVE HYDRAULIC RETURN | | | |
| | Line | 3" NC-4, NC-2 | | | |
| | ISO | B-17 | | | |
| TARGET | Bk. Pt. | 1 | 2 | 3 | - |
| Emergency Condenser | NE01-B 10" NE-5<br>NE01-A 10" NE-5<br>NE01-B 10" NE-2<br>NE01-A 10" NE-2 | N | N | N | N |
| Core Spray | (South) 8" NZ-3<br>(North) 8" NZ-3 | | | | |
| Main Steam | (South) 24" MS<br>(North) 24" MS | | | | |
| Reactor Cleanup | 6" ND-10<br>6" ND-1 | | | | |
| Shutdown Cooling | 14" NU-1, NU-4<br>14" NU-2, NU-3 | | | | |
| Feedwater | (South) 10" RF-2<br>(South) 18" RF-2<br>(North) 10" RF-2<br>(North) 18" RF-2 | | | | |
| Liquid Poison - 1.5" NP-2 | | | | | |
| Reactor Vessel Head Cooling | 2" RHC-2 | | | | |
| Reactor Recirc. | (Loop A)<br>(Loop B)<br>(Loop C)<br>(Loop D)<br>(Loop E) | | | | |
| Reactor Recirc. | (Bypass A)<br>(Bypass B)<br>(Bypass C)<br>(Bypass D)<br>(Bypass E) | | | | |
| Control Rod Drive | 3" NC-4, NC-2<br>(Supply) 1" NC-3<br>(Return) 3/4" NC-3 | -<br>N | -<br>N | -<br>N | -<br>N |
| Containment Spray | 14" CS-2, NQ-2<br>12" NQ-2<br>10" NQ-2<br>8" NQ-2 | | | | |
| Reactor Vessel | | | | | |
| Biological Shield Wall | | | | | |
| Main Steam Relief Valve Discharge | 8" (4 lines)<br>14" (2 lines) | | | | |
| Main Steam Safety Valve Discharge | 8" (16 lines) | | | | |
| Containment Vessel Shell | | | U | U | |
| Electrical | (Whip) | | N | N | |
| | (Impingement) | N | A | A | N |

## REFERENCE C-1

## INTERACTION EVALUATION

The following notes apply to Reference C-1, "Interaction Evaluation" contained on Pages C-19 through C-34:

(1) A circumferential break postulated at the terminal end of a line connected to the reactor vessel will not result in a pipe whip interaction with any of the listed targets due to the short length of the whipping pipe.

(2) A circumferential break postulated at the penetration assembly of a line connected to the reactor vessel will not result in a pipe whip interaction with any of the listed targets as the assembly will restrict the pipe from whipping.

(3) Sheet Numbers specified refer to Burns and Roe Drawing 2095 which consists of 10 sheets.

## REFERENCE C-1

## INTERACTION EVALUATION

PURPOSE: To identify the methods used to determine the interactions shown on the interaction evaluation matrices for each system.

EMERGENCY CONDENSER SYSTEM - 10" NE-5 (Reactor to NE01-B)

A circumferential break at point 1 will not result in an interaction with any of the listed targets [1].

A circumferential break at point 2 will most likely place the connection at the reactor in torsion causing the pipe to whip in such a manner as to result in an acceptable interaction with 10" NE-5 (reactor to NE01-A) at elevation 84.83', azimuth 0° of Sheet 6 [3]. Such a break could also cause the pipe to whip about a hinge formed at the first elbow downstream of the reactor which would have the same result.

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with the reactor vessel at elevation 85.5', azimuth 21° of Sheet 6 [3]. Such a break could also cause the pipe to whip about a hinge formed at the first elbow downstream of the reactor which would have the same result.

A circumferential break at point 4 will not result in an interaction with any of the listed targets [2].

A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

EMERGENCY CONDENSER SYSTEM - 10" NE-5 (Reactor to NE01-A)

A circumferential break at point 1 will not result in an interaction with any of the listed targets [1].

A circumferential break at point 2 will most likely place the connection at the reactor in torsion causing the pipe to whip in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 82.25', azimuth 350° of Sheet 5[3]. Such a break could also cause the pipe to whip about a hinge formed at the first elbow downstream of the reactor which would have the same result.

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with 10" NE-5 (reactor to NE01-B) at elevation 85', or with the reactor vessel at elevation 84', azimuth 335$^{o}$ of Sheet 6 (3). Such a break could cause the pipe to whip about a hinge formed at the first elbow downstream of the reactor which would have the same result.

A circumferential break at point 4 will not result in an interaction with any of the listed targets (2).

> A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

EMERGENCY CONDENSER SYSTEM - 10" NE-2 (NE01-B to Reactor)

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection to the Shutdown Cooling Piping in such a manner as to result in an acceptable interaction with 10" RF-2 (south) at elevation 75', azimuth 45$^{o}$ of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the first weld upstream of the connection which would have the same result.

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection to the Shutdown Cooling Piping in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 82', azimuth 25$^{o}$ of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the first weld upstream of the connection which would have the same result.

A circumferential break at point 4 will not result in an interaction with any of the listed targets (2).

> **A break at point 1 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 300$^{o}$-360$^{o}$ of Sheet 3 (3), or in an acceptable interaction with motor operators V-14-37, V-17-54 or V-17-19. A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.**

EMERGENCY CONDENSER SYSTEM - 10" NE-2 (NE01-A to Reactor)

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection to the Recirculation Piping in such a manner as to result in an acceptable interaction with 10" RF-2 (north) at elevation 75', azimuth 315° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the second elbow upstream of the connection which would have the same result.

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection to the Recirculation Piping in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 82', azimuth 335° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the second elbow upstream of the connection which would have the same result.

A circumferential break at point 4 will not result in an interaction with any of the listed targets (2).

> A break at point 1 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 0°-60° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

CORE SPRAY - 8" NZ-3 (South)

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 90', azimuth 70° of Sheet 6 (3). Such a break could also cause the pipe to whip about a hinge formed at the first elbow upstream of the reactor which would result in an unacceptable interaction with the containment vessel at elevation 89', azimuth 95° of Sheet 6 (3).

A circumferential break at point 3 will most likely place the connection at the reactor in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 90', azimuth 65° of Sheet 6 (3). Such a break could also cause the pipe to whip about a hinge formed at the first elbow upstream of the reactor which would have the same result.

A circumferential break at point 4 will not result in an interaction with any of the listed targets (2).

Circumferential breaks at points 5 thru 10 will not result in pipe whip as check valves NZ-02B and NZ-02D isolate these points from the high energy reservoir (reactor).

A circumferential break at any of points 11 thru 13 will most likely cause the pipe to whip about a hinge formed at the connection of the bypass line to the main line in such a manner as to result in no interactions with any of the listed targets.

> A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

CORE SPRAY SYSTEM - 8" NZ-3 (North)

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at point 2 will most likely place the connection at the reactor in torsion causing the pipe to whip in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 62', azimuth 240° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the connection at the reactor which would result in an acceptable interaction with 10" RF-2 (north) at elevation 75', azimuth 225° of Sheet 5 (3).

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 62, azimuth 250° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the second elbow upstream of the reactor which would have the same result.

A circumferential break at point 4 will not result in an interaction with any of the listed targets (2).

Circumferential breaks at points 5 thru 8 will not result in pipe whip as check valves NZ-02A and NZ-02C isolate these points from the energy reservoir (reactor).

A circumferential break at any of points 9 thru 11 will most likely cause the pipe to whip about a hinge formed at the connection of the bypass line to the main line in such a manner as to result in no interactions with any of the listed targets.

A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

MAIN STEAM SYSTEM - 24" MS (South)

A circumferential break at point 1 will most likely place the connection at penetration 2A in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 89', azimuth 90° of Sheet 6 (3), or with 10" RF-2 (south) at elevation 47', azimuth 80° of Sheet 4 (3).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at penetration 2A in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 89', azimuth 75° of Sheet 6 (3). Such a break could also cause the pipe to whip about a hinge formed at the third elbow downstream of the reactor which would result in an acceptable interaction with the Biological Shield Wall at elevation 82', azimuth 90° of Sheet 5 (3).

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 56', azimuth 90° of Sheet 5 (3). Such a break would also cause the pipe to whip about a hinge formed at the connection at penetration 2A which would result in an unacceptable interaction with 10" RF-2 (south) at elevation 49', azimuth 80° of Sheet 4 (3).

A circumferential break at point 4 will not result in interaction with any of the listed targets (2).

A break at point 3 will result in an unacceptable jet impingement interaction with the cable tray at 44', azimuth 150° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

MAIN STEAM SYSTEM - 24" MS (North)

A circumferential break at point 1 will most likely place the connection at penetration 2B in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 89', azimuth 270° of Sheet 6 (3), or with 10" RF-2 (north) at elevation 47', azimuth 260° of Sheet 4 (3).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at penetration 2B in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 89', azimuth 285° of Sheet 6 (3). Such a break could also cause the pipe to whip about a hinge formed

at the third elbow downstream of the reactor which would result in an unacceptable interaction with 3" NC-4 at elevation 75', azimuth 270° of Sheet 5 (3) or in an acceptable interaction with the Biological Shield Wall at elevation 82', azimuth 270° of Sheet 5 (3).

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 56', azimuth 270° or an unacceptable interaction with 3" NC-4 at elevation 75', azimuth 270° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the connection at penetration 2B which would result in an unacceptable interaction with 10" RF-2 (north) at elevation 49', azimuth 260° of Sheet 4 (3).

A circumferential break at point 4 will not result in interaction with any of the listed targets (2).

> A break at point 3 will result in an unacceptable jet impingement interaction with the cable tray at elevation 44', azimuth 210° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

REACTOR CLEANUP SYSTEM - 6" ND-10

A circumferential break at point 1 will not result in an interaction with any of the listed targets.

A circumferential break at either point 2 or 3 will most likely cause the pipe to whip about a hinge formed at the connection to the recirculation piping in such a manner as to result in no interactions.

A circumferential break at point 4 or 5 will most likely place the connection to the recirculation piping in torsion causing the pipe to whip in such a manner as to result in an acceptable interaction with 8" NQ-2 at elevation 37', azimuth 25° of Sheet 3 (3). Such a break could also cause the pipe to whip about a hinge formed at the connection to the recirculation piping which would result in an unacceptable interaction with the containment vessel at elevation 51', azimuth 65° of Sheet 4 (3).

A circumferential break at point 6 or 7 will most likely place the connection to the recirculation piping in torsion causing the pipe to whip in such a manner as to result in an acceptable interaction with 8" NQ-2 at elevation 37', azimuth 25° of Sheet 3 (3). Such a break could also cause the pipe to whip about a hinge formed at the connection to the recirculation piping which would have the same result.

A circumferential break at points 8 thru 11 will not result in pipe whip as check valve V-16-62 isolates these points from the energy reservoir (reactor).

> A break at point 1, 2, or 3 will result in an unacceptable jet impingement interaction with the cable tray at 40', azimuth $80^{o}$-$90^{o}$ of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

## REACTOR CLEANUP SYSTEM - 6" ND-1

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection to the recirculation piping in such a manner as to result in no interactions with any of the listed targets.

A circumferential break at either point 3 or 4 will most likely cause the pipe to whip about a hinge formed at the connection to recirculation piping in such a manner as to result in an acceptable interaction with 24" MS (south) at elevation 60', azimuth $90^{o}$ of Sheet 5 (3).

A circumferential break at point 5 will most likely place the connection to recirculation piping in torsion causing the pipe to whip in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 60', azimuth $95^{o}$ of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the first elbow downstream of the connection which would have the same result.

A circumferential break at point 6 will not result in an interaction with any of the listed targets (2).

> A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

## SHUTDOWN COOLING SYSTEM - 14" NU-1 and NU-4

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at either point 2 or 3 will most likely cause the pipe to whip about a hinge formed at the connection to the recirculation piping in such a manner as to result in an acceptable interaction with 14" NU-2 at elevation 49', azimuth $300^{o}$ of Sheet 4 (3).

A circumferential break at point 4 will not result in an interaction with any of the listed targets (2).

> A break at point 1 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 320° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

SHUTDOWN COOLING SYSTEM - 14" NU-2 and NU-3

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at points 2 thru 4 will not result in pipe whip as normally closed valve V-14-59 isolates these points from the high energy reservoir (reactor).

> A break at point 1 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 290°-300° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

FEEDWATER SYSTEM (SOUTH) - 10" RF-2 and 18" RF-2

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection between the tee and the reducer in such a manner as to result in an unacceptable interaction with 6" ND-10 at elevation 55', azimuth 75° of Sheet 4 (3), or with 8" NQ-2 at elevation 66', azimuth 45° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the third elbow upstream of the reactor which would have the same result.

A circumferential break at any of points 2 thru 4 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an unacceptable interaction with 6" ND-10 at elevation 55', azimuth 75° of Sheet 4 (3).

A circumferential break at point 5 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in no interactions with any of the listed targets. Such a break would also cause the pipe to whip about a hinge formed at the connection at penetration 4A which would result in an unacceptable interaction with the containment vessel at elevation 27', azimuth 170° of Sheet 2 (3), or in an acceptable interaction with 24" MS (south) at elevation 27', azimuth 170° of Sheet 2 (3).

A circumferential break at point 6 will most likely cause the pipe to whip about a hinge formed at the connection at penetration 4A in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 33', azimuth 165° of Sheet 2 (3). Such a break would also cause the pipe to whip about a hinge formed at the connection to the tee which would result in an acceptable interaction with loop C of the recirculation piping at elevation 35', azimuth 144° of Sheet 2 (3), or with the Biological Shield Wall at elevation 40', azimuth 145° of Sheet 3 (3), or in an acceptable interaction with 14" MS relief valve at elevation 47', azimuth 150° of Sheet 4 (3).

A circumferential break at point 7 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in an acceptable interaction with loop C of the recirculation piping at elevation 35', azimuth 144° of Sheet 2 (3), or with the Biological Shield Wall at elevation 40', azimuth 145° of Sheet 3 (3), or in an acceptable interaction with 14" MS relief valve at elevation 47', azimuth 150° of Sheet 4 (3).

A circumferential break at either point 8 or 9 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in an unacceptable interaction with 8" NQ-2 at elevation 66', azimuth 135° of Sheet 5 (3).

A circumferential break at point 10 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 74', azimuth 135° of Sheet 5 (3).

A circumferential break at point 11 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 60', azimuth 225° of Sheet 5 (3), or with 24" MS (south) at elevation 47', azimuth 135° of Sheet 4 (3).

A circumferential break at point 12 will most likely cause the pipe to whip about a hinge formed at the connection at penetration 4A in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 49', azimuth 175° of Sheet 4 (3).

A break at point 2, 3, 4, 6, 7 or 12 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 110°-115° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

FEEDWATER SYSTEM (NORTH) - 10" RF-2 and 18" RF-2

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection between the tee and the reducer in such a manner as to result in an unacceptable interaction with 3" NC-4 at elevation 56', azimuth 290° of Sheet 4 (3), or with 8" NQ-2 at elevation 66', azimuth 315° of Sheet 5 (3). Such a break could also cause the pipe to whip about a hinge formed at the third elbow upstream of the reactor which would have the same result.

A circumferential break at any of points 2 thru 4 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an unacceptable interaction with 3" NC-4 at elevation 56', azimuth 290° of Sheet 4 (3).

A circumferential break at point 5 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in no interactions with any of the listed targets. Such a break would also cause the pipe to whip about a hinge formed at the connection at penetration 4B which would result in an unacceptable interaction with the containment vessel at elevation 27', azimuth 190° of Sheet 2 (3), or in an acceptable interaction with 24" MS (north) at elevation 27', azimuth 190° of Sheet 2 (3).

A circumferential break at point 6 will most likely cause the pipe to whip about a hinge formed at the connection at penetration 4B in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 33', azimuth 195° of Sheet 2 (3). Such a break would also cause the pipe to whip about a hinge formed at the connection to the tee which would result in an acceptable interaction with loop D of the recirculation piping at elevation 35', azimuth 216° of Sheet 2 (3), or with the Biological Shield Wall at elevation 40', azimuth 215° of Sheet 3 (3) or in an acceptable interaction with 14" MS relief valve at elevation 47', azimuth 210° of Sheet 4 (3).

A circumferential break at point 7 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in an acceptable interaction with loop D of the recirculation piping at elevation 35', azimuth 216° of Sheet 2 (3), or with the Biological Shield Wall at elevation 40', azimuth 215° of Sheet 3 (3), or in an acceptable interaction with 14" MS relief valve at elevation 47', azimuth 210° of Sheet 4 (3).

A circumferential break at either point 8 or 9 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in an unacceptable interaction with 8" NQ-2 at elevation 66', azimuth 225° of Sheet 5 (3).

A circumferential break at point 10 will most likely cause the pipe to whip about a hinge formed at the connection to the tee in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 74', azimuth 225° of Sheet 5 (3).

A circumferential breat at point 11 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 60', azimuth 225° of Sheet 5 (3), or with 24" MS (north) at elevation 47', azimuth 225° of Sheet 4 (3).

A circumferential break at point 12 will most likely cause the pipe to whip about a hinge formed at the connection at penetration 4B in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 49', azimuth 185° of Sheet 4 (3).

> A break at point 2, 3, 4, 6, 7 or 12 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 210°-250° of Sheet 3 (3). A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

## LIQUID POISON SYSTEM - 1-1/2" NP-2

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at point 2 will most likely place connection at the reactor in torsion such as to result in an unacceptable interaction with the containment vessel at elevation 82', azimuth 155° of Sheet 6 (3).

A circumferential break at either point 3 or 4 will most likely place the connection at the reactor in torsion such as to result in an acceptable interaction with 8" NZ-3 (south) at elevation 79', azimuth 115° of Sheet 6 (3).

A circumferential break at point 5 will not result in an interaction with any of the listed targets (2).

> A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

## REACTOR VESSEL HEAD COOLING SYSTEM - 2" RHC-2

A circumferential break at point 1 will not result in an interaction with any of the listed targets (1).

A circumferential break at either point 2 or 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in no interactions with any of the listed targets.

A circumferential break at point 4 will not result in an interaction with any of the listed targets [(2)].

A break at any of the points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

## REACTOR RECIRCULATION SYSTEM - LOOP A

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection at the pump suction in a manner as to result in an unacceptable interaction with the cable tray at elevation 41', azimuth 42° of Sheet 3 [(3)], or with 12" NQ-2 at elevation 26', azimuth 42° of Sheet 4 [(3)].

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 40', azimuth 340°, or with loop E of the recirculation piping at elevation 40', azimuth 330° of Sheet 3 [(3)].

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 45', azimuth 340° or with loop E of the recirculation piping at elevation 45', azimuth 330° of Sheet 3 [(3)].

A circumferential break at point 4 will most likely place the connection at the pump discharge in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 45', azimuth 0° of Sheet 3 [(3)], or with the cable tray at elevation 41', azimuth 0° of Sheet 3 [(3)].

A break at point 1 will result in an acceptable jet impingement interaction with motor operator V-14-36, elevation 49', azimuth 25° of Sheet 4 [(2)]. A break at point 2, 3, or 4 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 0° of Sheet 3 [(3)].

## REACTOR RECIRCULATION SYSTEM - LOOP B

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection at the pump suction in such a manner as to result in an unacceptable interaction with 24" MS (south) at elevation 49', azimuth 110° of Sheet 4 [(3)].

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 40', azimuth 54° of Sheet 3 (3), or with loop A of the recirculation piping at elevation 40', azimuth 42° of Sheet 3 (3).

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 45', azimuth 54° of Sheet 3 (3), or with loop A of the recirculation piping at elevation 40', azimuth 42° of Sheet 3 (3).

A circumferential break at point 4 will most likely place the connection at the pump discharge in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 45', azimuth 72° of Sheet 3 (3), or with the cable tray at elevation 38', azimuth 72° of Sheet 3 (3).

> A break at point 1 will result in an acceptable jet impingement interaction with the motor operator for MS relief valve at elevation 49', azimuth 155° of Sheet 4 (3). A break at point 2, 3, or 4 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 72° of Sheet 3 (3).

REACTOR RECIRCULATION SYSTEM - LOOP C

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection at the pump suction in such a manner as to result in an unacceptable interaction with 24" MS (north) at elevation 49', azimuth 190° of Sheet 4 (3).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 40', azimuth 126° of Sheet 3 (3), or with loop B of the recirculation piping at elevation 40', azimuth 114° of Sheet 3 (3).

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 45', azimuth 126° of Sheet 3 (3), or with loop B of the recirculation piping at elevation 45', azimuth 114° of Sheet 3 (3).

A circumferential break at point 4 will most likely place the connection at the pump discharge in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 45', azimuth 144° of Sheet 3 (3), or with the cable tray at elevation 41', azimuth 144° of Sheet 3 (3), or an acceptable interaction with 14" MS relief valve at elevation 45', azimuth 138° of Sheet 3 (3).

> A break at point 1 will result in an acceptable jet impingement interaction with the motor operator MS relief valve at elevation 49', azimuth 240°. A break at point 2, 3, or 4 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 144° of Sheet 3 (3).

## REACTOR RECIRCULATION SYSTEM - LOOP D

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection at the pump suction in such a manner as to result in an unacceptable interaction with 24" MS (north) at elevation 49', azimuth 258° of Sheet 4 (3).

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 40', azimuth 198° of Sheet 3 (3), or with loop C of the recirculation piping at elevation 40', azimuth 186° of Sheet 3 (3).

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 45', azimuth 198° of Sheet 3 (3), or with loop C of the recirculation piping at elevation 45', azimuth 198° of Sheet 3 (3).

A circumferential break at point 4 will most likely place the connection at the pump discharge in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 45', azimuth 216° of Sheet 3 (3), or with the cable tray at elevation 41', azimuth 216° of Sheet 3 (3), or an acceptable interaction with 14" MS relief valve at elevation 45', azimuth 220° of Sheet 3 (3).

> A break at point 1 will result in an acceptable jet impingement interaction with motor operator for valve V-17-54 at elevation 53', azimuth 300° of Sheet 4 (3). A break at point 2, 3, or 4 will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 216° of Sheet 3 (3).

REACTOR RECIRCULATION SYSTEM - LOOP E

A circumferential break at point 1 will most likely cause the pipe to whip about a hinge formed at the connection at the pump suction in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 52', azimuth 370$^{o}$ of Sheet 4 [(3)], or with the cable tray at elevation 37', azimuth 288$^{o}$ of Sheet 3 [(3)].

A circumferential break at point 2 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 40', azimuth 370$^{o}$ of Sheet 3 [(3)], or with loop D of the recirculation piping at elevation 40', azimuth 258$^{o}$ of Sheet 3 [(3)].

A circumferential break at point 3 will most likely cause the pipe to whip about a hinge formed at the connection at the pump discharge in such a manner as to result in an acceptable interaction with the Biological Shield Wall at elevation 45', azimuth 270$^{o}$ of Sheet 3 [(3)], or with loop D of the recirculation piping at elevation 45', azimuth 258$^{o}$ of Sheet 3 [(3)].

A circumferential break at poing 4 will most likely place the connection at the pump discharge in torsion causing the pipe to whip in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 45', azimuth 288$^{o}$ of Sheet 3 [(3)], or with the cable tray at elevation 37', azimuth 288$^{o}$ of Sheet 3 [(3)].

A break at point 1 will result in an acceptable jet impingement interaction with motor operator for valve V-14-36 at elevation 49', azimuth 25$^{o}$ of Sheet 4 [(3)]. A break at point 2, 3, or 4 will result in an unacceptable jet impingement interaction with cable tray at elevation 40', azimuth 288$^{o}$ of Sheet 3 [(3)].

REACTOR RECIRCULATION SYSTEM - BYPASS (Typical for A, B, C, D, and E)

A circumferential break at either point 1 or 2 will most likely cause the pipe to whip about a hinge formed at the opposite connection at the recirculation loop piping in such a manner as to result in no interactions with any of the listed targets.

A circumferential break at either point 3 or 4 will most likely cause the pipe to whip about a hinge formed at the connection at the recirculation loop piping at elevation 24.75', in such a manner as to result in no interactions with any of the listed targets. Such a break would also cause the pipe to whip about a hinge formed at the connection to the recirculation loop piping at elevation 16' which would also result in no interactions with any of the listed targets.

A break at point 2 in loop D will result in an unacceptable jet impingement interaction with the cable tray at elevation 40', azimuth 216°. A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

CONTROL ROD DRIVE HYDRAULIC RETURN - 3" NC-4, NC-2

A circumferential break at point 1 will not result in an interaction with any of the listed targets [1].

A circumferential break at either point 2 or 3 will most likely cause the pipe to whip about a hinge formed at the connection at the reactor in such a manner as to result in an unacceptable interaction with the containment vessel at elevation 55', azimuth 315° of Sheet 4 [3].

A circumferential break at point 4 will not result in an interaction with any of the listed targets.

A break at point 2 or 3 will result in an acceptable jet impingement interaction with the motor operators for valves V-17-19 and V-14-37 both at elevation 49', azimuth 33° of Sheet 4 [3]. A break at any of the remaining points postulated in this line will not result in any jet impingement interactions with the cable tray or any of the motor operators.

REACTOR VESSEL DRAIN

Isometrics and stress analyses were not available for this system. However, inspection during site visits determined that breaks at any point in the system would not result in any unacceptable interactions.

REACTOR VESSEL VENT

Isometrics and stress analyses were not available for this system. However, inspection during site visits determined that breaks at any point in the system would not result in unacceptable interactions.

# APPENDIX 3.6C

## EVALUATION OF STRUCTURAL INTEGRITY - BIOLOGICAL SHIELD WALL UNDER PIPE WHIP LOADINGS

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK
NUCLEAR GENERATING STATION

# EVALUATION OF STRUCTURAL INTEGRITY OF THE BIOLOGICAL SHIELD WALL UNDER PIPE WHIP LOADINGS

June, 1974

# TABLE OF CONTENTS

## 1.0 INTRODUCTION

This report, prepared by EDS Nuclear Inc. for Jersey Central Power & Light Company, describes the analyses performed to determine the effects of pipe impact on the biological shield wall of the Oyster Creek Nuclear Generating Station, and to evaluate the structural adequacy of the shield wall under pipe impact, in combination with other types of concurrent loading.

The purpose of these studies was to evaluate the structural response of the biological shield wall following a postulated high-energy line break and subsequent unrestrained pipe whip. The structural integrity of the shield wall was evaluated with respect to both the gross structural response and local damage predictions, including perforation of the steel and depth of concrete penetration.

All analyses were performed for postulated "worst case" conditions of impact. The gross structural response was evaluated by performing dynamic time history analyses corresponding to the impact of a 24-inch diameter pipe at the top of the shield wall. Elastic analyses were performed, as the gross response of the shield wall was expected to remain in the elastic range. Local damage predictions were evaluated according to conservative penetration equations currently specified by the AEC.

The results of the analyses indicate that no gross structural damage will occur under "worst case" impact loadings, and that the shield wall is capable of withstanding the full spectrum of postulated breaks without incurring significant loss of load-carrying capability. Damage to the shield wall will be restricted to the local region of impact and will not significantly effect the overall structural capability.

## 2.0 DESCRIPTION OF STRUCTURE

The biological shield wall is a cylindrical structure composed of steel plate, steel column sections and concrete. The structure is approximately 45 feet high, has an inside diameter of 20 feet, 10 inches, and a total thickness of 29 inches. The shield wall functions as both a radiation shield for protection of plant personnel, and as a load-carrying structure for support of inside-containment piping.

The shield wall consists of the following structural components:

1. Twenty-five steel columns (27 WF 177 sections) at approximately uniform spacing circumferentially.

2. Steel plate (5/16-inch) comprising the outside surface, and 1/4-inch plate comprising the inside surface of the wall.

3. Poured concrete infill.

Both the inside and outside steel plate are provided with 1/2-inch studs at approximately 1'-6" spacing to ensure composite response of the steel and concrete. High density concrete is provided in the portion of the wall adjacent to the reactor core, with a specified unit weight of 210 $lbs/ft^3$, and standard weight concrete is provided for the remainder of the shield wall.

## 3.0 DISCUSSION OF ANALYSES

Analyses were performed to determine conservative estimates of both the overall response of the structure and the extent of local damage resulting from "worst case" pipe impact effects.

The gross structural response under impact loads was combined with conservative estimates of the response to other types of concurrent loadings, which were then compared to the structural capability of the shield wall. The capability of the structure for moments, shears and axial loads was evaluated in accordance with the following assumptions:

1. Composite behavior of the steel and concrete was assumed. (This is discussed further in Section 3.1 below.)

2. The concrete cannot sustain either tensile or shear stresses.

3. The compressive strength of the concrete was assumed to be 3,000 psi, with an associated allowable compressive stress equal to $0.75f'_c$.

4. The yield strength of the steel was assumed to be 36 ksi, with an associated allowable stress equal to $0.9\,f_y$.

The capability of the shield wall to sustain local damage was evaluated directly from empirical relationships derived from projectile impact experiments. The equations used to determine perforation or depth of penetration were those discussed by Amirikian (Reference 4.01) and Cottrell and Savolainen (Reference 5.01). These equations are currently accepted by the USAEC for use in local damage predictions associated with pipe impact.

### 3.1 Gross Structural Response Analysis

A mathematical model of the shield wall was constructed for the gross structural response analyses. The model consisted of lumped masses connected by massless elastic three-dimensional beam elements. A sufficient number of mass points was selected to accurately represent the relatively high-frequency wave transmission characteristics of the shield wall necessary for representing the response to impact loadings. Cross section properties of the elements included the composite behavior of the steel plate exterior, steel

column interior and concrete infill material, as the steel plates are provided with studs to ensure composite structural response. In addition, equivalent cross section properties were calculated and included at all elevations corresponding to the locations of hatches and penetrations.

An idealized impact force time history was constructed based on the blowdown force time histories developed by EDS for the pipe whip analyses of the Oyster Creek Emergency Condenser piping system. (Reference 2.02). The forcing function was constructed by extrapolating the previously-developed blowdown forces occurring at the break location to a 24-inch diameter pipe size. In addition, the initial portion of the resulting time history was further increased by a factor of 2.0, to account for the short-duration forces developed during impact. The force time history was postulated to act at the top of the cylindrical shield wall, as this impact location results in the largest dynamic response shears, moments and axial forces throughout the shield wall. It should be noted that this impact location would not be predicted from the postulated break locations (based on pipe stress criteria) for the 24-inch piping. Instead, the most conservative impact location was chosen for purposes of evaluating the maximum shield wall capability.

Dynamic elastic time history analyses were performed on the mathematical model discussed above, subjected to the postulated impact force time history. The analyses were performed using EDS program EDSGAP, originally developed by Wilson (Reference 6.01), and modified extensively by EDS. The program may be used to analyze three-dimensional structural systems of arbitrary geometry subjected to static or dynamic loading. The assumption of elastic behavior was considered to be appropriate for the analyses, as the gross stresses over the shield wall cross sections were expected to remain within the elastic range. A value of five percent structural damping was assumed in the analyses, corresponding to a combined material damping for steel and concrete.

Time histories of cross section moments and shears under the "worst case" impact loadings were obtained. The maximum shears and moments, in combination with those occurring from other types of concurrent loadings, were compared with the overall capability of the shield wall in accordance with the AEC criteria for factored load combinations. The load combinations considered in this study were those specified in Section C.1 of Reference 1.02.01, with the ultimate load capacities calculated as discussed in Section 3.0 above.

The results of the response analyses and evaluation of structural adequacy are discussed in Section 4.1 below.

### 3.2 Local Damage Analyses

Prediction of local damage was evaluated by calculating "threshold penetration" thicknesses of steel and concrete subjected to impact of both a segment of 10-inch diameter and 24-inch diameter pipe. Rigid-body impact was assumed for the calculations, as the assumption that no energy is absorbed by the impacting pipe results in conservative penetration predictions.

From an examination of the postulated break locations (Reference 2.01) for the case of a 10-inch diameter pipe, it was considered that a "worst case" impact would correspond to a missile consisting of an unfolded segment of 10-inch pipe striking the shield wall on edge. The kinetic energy of tne missile was assumed to be equivalent to the change in internal energy of the enclosed steam in undergoing a change of state from the operating conditions of the fluid to ambient conditions (Reference 3.01). This impact case is more severe than the case of impact by a whipping pipe of the same size, as the cross-sectional area of impact for the postulated case is smaller than the impact area of a whipping pipe, and hence higher stresses will be developed in the local region of the shield wall.

In the case of a 24-inch pipe break, it was found that no conditions exist for generation of a small missile, based on postulated break locations. It was therefore assumed that the most severe impact case consisted of a circumferential break and subsequent whip of the longest segment of pipe for which the shield wall is a possible target. The impact velocity and kinetic energy of impact were calculated from the mass of the pipe, maximum blowdown force and the maximum distance between the pipe segment and shield wall.

Spalling of the concrete will not occur for the Oyster Creek shield wall design, as steel plates are provided over both the entire inside and outside wall surfaces. Hence, concrete spalling was not included in the local damage evaluation.

The results of the local damage analyses are discussed in Section 4.2 below.

## 4.0 DISCUSSION OF RESULTS

The results of both the overall structural integrity analyses and the local damage analyses indicate that the shield wall is capable of withstanding the effects of a "worst case" pipe impact without incurring gross structural failure, perforation or significant loss of load-carrying capability. The results from the two phases of the study are discussed in separate sections below.

### 4.1 Gross Structural Response

The maximum dynamic moments and shears obtained from the structural response analyses discussed in Section 3.1 occur at the base of the shield wall. For the case of impact by a 24-inch diameter pipe, the maximum moment and shear are approximately 55,000 k-ft. and 1,350 kips, respectively.

Approximate seismic moments and shears were evaluated for combination with the above loads. Horizontal seismic loadings were based on a conservatively-estimated horizontal spectral acceleration of 0.5g at a frequency of 15 Hz, the first fundamental translational frequency of the shield wall. A factor of 1.5 was applied to the resulting base moment and shear to account for the contribution of higher modes of response. The moment and shear calculated at the base of the shield wall using the above procedure were 30,000 k-ft. and 950 kips, respectively.

It was found that the moments at the base of the shield wall control the capability of the structure for pipe impact and seismic loadings. The combination of pipe whip and SSE loadings results in a total base moment of slightly less than 30 percent of the structural capability. It was found that the capability of the shield wall for shears and axial forces was substantially larger than this margin.

It is concluded that the shield wall is capable of withstanding the postulated "worst case" pipe whip loadings without gross structural damage. The maximum loadings encountered are considerably less than the capability of the structure.

### 4.2 Local Damage

The results of the penetration calculations indicate that the case of a missile generated by a 10-inch diameter pipe break is more severe with regard to depth of penetration than the case of impact of a 24-inch pipe. This results

from the fact that the impact area is considerably smaller for the 10-inch pipe break.

The results of the calculations indicate that a concrete thickness of approximately 22 inches is sufficient to prevent perforation. The depth of penetration of the shield wall will be less than this amount, as the steel plate will absorb some of the energy of impact. The calculations for required steel plate thickness indicate that approximately one inch of steel is necessary to prevent perforation. Therefore, it is possible that the steel plate at the impact location will be penetrated, as the plate thickness on the outside surface of the shield wall is less than this amount. Similar calculations for the case of impact by a 24-inch pipe resulted in a considerably smaller estimate of concrete penetration depth.

Damage caused by pipe impact will be restricted to the local region of impact, as the design of the shield wall includes steel column members continuous through the height of the shield wall at approximately three-foot intervals over the circumference. (Reference 1.01.01). It is considered that these columns will restrict the development of cracking or crushing of the concrete to the region of impact enclosed by two adjacent columns. Such an extent of local damage will not significantly affect the gross structural capability of the shield wall, as the region of local damage is a small percentage of the shield wall cross section.

It is concluded that a whipping pipe or missile generated by pipe rupture will not perforate the shield wall, although perforation of the outer steel plate may possibly occur for the "worst case" impact. The depth of penetration will be less than 22 inches of concrete, and the region of damage to the concrete and outer steel plate will be restricted to approximately three feet of the shield wall circumference.

## 5.0 CONCLUSIONS

The conclusions developed from these studies may be summarized as follows:

1. The overall load-carrying capability of the structure is significantly greater than the combinations of loadings associated with the maximum structural responses resulting from the load types specified in Reference 1.02.01.

2. Perforation of the shield wall will not occur for the "worst case" pipe impact. Depth of penetration is predicted to be less than 22 inches of concrete, using the experimentally-derived relationships specified in Reference 1.02.01.

3. Damage will be restricted to the local region of impact, and will not significantly effect the overall structural capability of the shield wall.

Moreover, it is concluded that the shield wall is capable of withstanding the full spectrum of postulated breaks without incurring gross damage or significant loss of load-carrying capability, and that the design of the structure is such that impact by a whipping pipe is a condition which can be tolerated in the Oyster Creek Nuclear Plant.

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# REFERENCES


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# APPENDIX

## EDS COMPUTER PROGRAM DESCRIPTION

# EDSGAP

EDS program EDSGAP is a general-purpose finite element program for linear elastic analyses of arbitrary structural systems. The program contains the following element types:

1. General beam
2. Truss
3. Two-dimensional plane stress/plane strain
4. Three-dimensional solid
5. Axisymmetric solid
6. Plate and shell
7. Translational/rotational spring

These element types may be used both singly and in compatible combinations. The program includes static and dynamic options, as discussed below. Out-of-core storage may be utilized for solution of the equations of equilibrium, storage of problem data and storage of solution results. The program has virtually no restrictions on size of the structural system to be analyzed. EDSGAP is based on the program SAP developed by E. L. Wilson of the University of California at Berkeley. However, many improvements have been incorporated into EDSGAP to increase its capabilities and efficiency.

Static analyses are performed using the Direct Stiffness Method, in which element stiffness matrices are formed according to virtual work principles and assembled to form a global stiffness matrix for the system, relating external forces and moments to joint displacements and rotations. Applied static loads may be specified as combinations of concentrated forces, thermal expansion loads, pressure forces, and inertia (body) forces. The equations of equilibrium of the system are solved for joint displacements and rotations by Gaussian reduction techniques.

Dynamic options within the program include calculation of undamped natural frequencies and normal modes of vibration using either the Determinant Search or Subspace Interaction techniques, and computation of time history response by either the Mode Superposition technique or direct integration of the equations of dynamic equilibrium. Dynamic loadings may be specified as

combinations of arbitrary applied force and moment time histories and three independent orthogonal component time histories of acceleration.

EDSGAP has been used for soil-structure interaction analyses on several nuclear power facilities, including Atlantic Generating Station, Newbold Island and Douglas Point, for pressure transient piping response analyses on Rancho Seco, Oconee, Calvert Cliffs, Donald C. Cook and Salem Generating Stations, and for conceptual design review studies on the GE MARK III Reactor Building.

The program has been verified for the various element types by an extensive set of sample problems, including comparisons with hand calculations or theoretical solutions, wherever possible, and has been benchmarked against EDS programs PISOL1A and PISOL3A for static and dynamic analyses of complex piping systems.

# APPENDIX 3.7A

## SEISMIC ACCELERATION FLOOR RESPONSE SPECTRA FOR THE REACTOR BUILDING

URS JAB 8151

EDCC
FILE W.O. 5510-66013, BA 402030
ENGRG. MECHANICS SECTION

3.7

URS

# SEISMIC ACCELERATION FLOOR RESPONSE SPECTRA FOR THE REACTOR BUILDING AT OYSTER CREEK NUCLEAR POWER PLANT

December 1981

prepared for
GPU Nuclear
Parsippany, New Jersey

prepared by
URS/John A. Blume & Associates, Engineers
130 Jessie Street (at New Montgomery)
San Francisco, California 94105

URS/JAB 8151

# URS

# SEISMIC ACCELERATION FLOOR RESPONSE SPECTRA FOR THE REACTOR BUILDING AT OYSTER CREEK NUCLEAR POWER PLANT

December 1981

prepared for
GPU Nuclear
Parsippany, New Jersey

prepared by
URS/John A. Blume & Associates, Engineers
130 Jessie Street (at New Montgomery)
San Francisco, California 94105

FORM 4.1-1

DOCUMENT APPROVAL SHEET*

..MENT TITLE: Seismic Acceleration Floor Response Spectra for the Reactor Building at Oyster Creek Nuclear Power Plant

...MENT TYPE: Criteria ☐ Interface ☐ Report ☒ Specification ☐ Other ☐

..JECT NAME: Oyster Creek Floor Spectra

..B NO.: 8151

..IENT: GPU Service Co.

* * * * *

..his document has been prepared in accordance with the URS/Blume Quality Assurance ..anual and project requirements. Initial issue (Rev. 0):

..epared by: Ahmad F. Kaleri Date: 2/24/82

..eviewed by: L. E. Malik Date: 3/2/82

..roved by: Dilip P. Thaveri Date: 3/5/82

..EVISION RECORD:

| Revision No. | Prepared By | Reviewed By | Approved By/ Date | Description of Revision |
|---|---|---|---|---|
| | | | | |

*Revised in Revision 8

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### APPENDIX



### TABLE



### FIGURES

CONTENTS (Continued)

1. INTRODUCTION

...port describes the work performed by URS/John A. Blume & Associates, ...ers (URS/Blume) for GPU Nuclear (GPU), under Purchase Order No. 70049. ...k described comprises the generation of seismic acceleration floor ... spectra for the reactor building of the Oyster Creek Nuclear Power located near Forked River, New Jersey.

...ackground

... Creek is a 620-MW boiling water reactor (BWR) plant that went into ...on in 1969. It is one of 11 older United States nuclear plants now ...eviewed under the Systematic Evaluation Program (SEP) of the U.S. ... Regulatory Commission (NRC). Seismic design has emerged as one of ...st significant issues addressed in the SEP, primarily because of the ...evelopments in the state of the art since the late 1950s to middle ... when many of the SEP plants were designed. Many of these older plants ... were not seismically qualified or were qualified using technology that ...ot be entirely acceptable today.

...ng the last three years, URS/Blume has conducted a series of studies on Oyster Creek reactor building. Soil-structure interaction (SSI) effects ... the Oyster Creek reactor building were estimated in a parametric study[1] ...or Jersey Central Power & Light Company (JCP&L). The study investigated the influence of several parameters on the response of the structures. These parameters are the free-field input-response spectrum, the flexibility of the foundation and radiation damping into the soil medium, and the shear stiffness of the soil. A second study[2] for JCP&L identified the areas of seismic analysis where significant changes have taken place in seismic analysis procedures and regulatory requirements since the time of the initial seismic qualification of the Oyster Creek plant and, furthermore, whenever possible, the study identified the impact of such changes on the original estimates of the seismic inputs to structure, SSI, dynamic modeling, and analytical procedures.

It is our understanding that much of the Oyster Creek Category I electrical equipment, and possibly some mechanical equipment and components, must be

evaluated for seismic motion. Acceleration floor response spectra are used for seismic qualification of Category I equipment. Such spectra are typically used in conjunction with analysis or testing, or with a combination of the two, to verify seismic adequacy of piping or equipment.

### 1.2 Scope of Work

The work reported herein consists of generating horizontal and vertical acceleration floor response spectra for selected locations of the reactor building. The seismic motion is specified to be the NRC site-specific spectra developed for the Oyster Creek site.[3] For generation of the horizontal east-west floor response spectra, the original fixed-based lumped-mass model of the reactor building, developed by URS/Blume in 1965,[4] is used. This model is, however, modified to incorporate SSI effects.

For the vertical floor response spectra, the effects of the out-of-plane behavior of the floor slabs are considered, as well as effects of amplification of motion through the supporting columns and walls.

### 1.3 Organization of the Report

Chapter 2 of this report describes the analysis procedures, while the results of the analysis are given in Chapter 3. Chapter 4 presents the summary and conclusions of this study. The criteria document used in this work is given in Appendix A.

## 2. ANALYSIS PROCEDURES

### 2.1 Description of the Reactor Building

The reactor building is a partially reinforced concrete and partially steel structure that houses the reactor and its auxiliary systems. The base is approximately 140 ft x 140 ft and is embedded 50 ft below grade. The reactor vessel and the recirculation system are contained inside the drywell of a pressure-suppression containment system. The drywell has massive reinforced concrete walls 4 ft to 6 ft in thickness. The primary containment system consists of the drywell, vent pipes, and a pool of water contained in the suppression chamber. The reactor building encloses the primary containment system, thereby providing a second containment. In addition, all refueling equipment is inside the building, including the spent fuel storage pool and the new fuel storage valut. The outside walls of the reactor building is constructued of reinforced concrete up to elevation 119 ft 3 in. Above this elevation is a steel braced frame superstructure which carries an overhead crane.

### 2.2 Seismic Input

The seismic input for the Oyster Creek reactor building is defined by the site-specific free-field acceleration response spectrum (design response spectrum) developed by the NRC.[3] However, seismic input is also needed in the form of an acceleration time history for the development of floor acceleration response spectra. The input acceleration time history should be such that its response spectrum closely matches the design response spectrum.

The prodecure to develop an acceleration time history for the Oyster Creek site was to start with a record of the S69E component of the Wheeler Ridge earthquake as recorded at Taft, California. This particular earthquake record was chosen because of seismological similarities between the Wheeler Ridge earthquake and the postulated earthquake for Oyster Creek sites.

This particular record of the Wheeler Ridge earthquake was then modified iteratively, using a proprietory URS/Blume computer program (SMSPC), until the response spectrum of the modified acceleration time history matched closely with the design response spectrum. In the modification process, only the spectral

amplitudes of the original time history were modified while the phase angles remained the same as those in the original time history.

Figure 1 shows the time history used to develop the floor acceleration response spectra for the Oyster Creek reactor building. Figure 2 shows the comparison of the response spectrum of this time history with the design response spectrum at a modal damping ratio of 10% of critical.

### Horizontal Dynamic Model

The scope of work specified that the horizontal east-west model developed by URS/Blume in 1965[4] be used for horizontal floor spectra generation. During parametric analysis of the Oyster Creek reactor building,[1] we had to reassemble this model from the 1965 report[4] to be suitable for use in present-day computer codes which are different from those used in 1965. This required some validation studies to justify that the model does, in fact, have the same dynamic characteristics as those used in the 1965 study. This model has been used in the present study.

The most important factor that the scope of work requires to be considered in the present study is the SSI effects on the responses and floor spectra of the reactor building of the Oyster Creek plant. Hence, the original fixed-base lumped-mass model is modified (see Figure 3) through the use of lumped parameters to simulate the SSI effects. Section 2.4 presents in detail the methodology adopted to estimate SSI effects in this study.

### 2.4 Soil-Structure Interaction Effects

SSI is a very recent development in the field of seismic analysis of structures. Consequently, all the available methodologies are based on simplifying assumptions which limit their applicability and require great care in their use as engineering judgment and a comprehensive understanding of a wide variety of related subjects (e.g., soil dynamics, structural dynamics, wave theory, structural mechanics) is necessary. The limited development of this field also makes it often difficult to conclusively answer all questions raised on the approach used in analyzing a structure such as the Oyster Creek reactor building where SSI effects are expected to have a significant influence on the structural response and especially on the floor response spectra.

Due to these problems, analysts have often adopted the approach of a parametric study to bound the results rather than to use a single solution. The LLL study[5] is a very good example of this bounding parametric approach.

The two most commonly used SSI methods at present are the finite element and the infinite half space (lumped parameter) approaches. Both methods have their advantages and disadvantages. However, since the lumped parameter approach is used in the current study, this will be the only procedure discussed here.

In general, the lumped parameter approach to SSI is based on developing a series of "soil springs" which represent the flexibility of the foundation material under the structure and a set of viscous dampers which represent the "radiation damping" (and material) effects of SSI. The equations are based on assuming the soil to be an infinite half space and a rigid foundation with a given mass "glued" to the half space. Recently, Novak et al.[6] and Kausel et al.[7] have suggested modification to the lumped parameters to account for the effects of embedment such as is the case with the Oyster Creek reactor building foundation.

There are several issues associated with the lumped parameter approach which are discussed separately below. These are:

a. Effect of variation in soil properties
b. Effect of embedment
c. Radiation damping in the soil

It should be noted that considerable insight into all these problems has been accumulated by URS/Blume from its original seismic analyses[4] of Oyster Creek in 1965, the parametric study[1] conducted by URS/Blume for Oyster Creek in 1979, the review[2] of the Oyster Creek seismic analysis, and the NRC study reported in Reference 5. This insight allowed us to limit some of the parametric studies that are often used for plants which do not have as much supporting material.

a. Effects of Variation of Soil Properties. The single most important parameter in the lumped parameter approach to SSI is the shear modulus, $G$, of the

soil. Common practice found acceptable to the NRC in accordance with the Standard Review Plan[8] is to use average values of $G$ as well as values ±50% of the average and then envelop the results of the three analyses. This was the approach used by URS/Blume for the structural responses in the parametric study[1] for the Oyster Creek reactor building.

...ng its review of the NRC study,[5] the Senior Seismic Review Team (SSRT) of ... developed a specific set of guidelines for use in the SSI review of SEP ...nts. These guidelines (Appendix C, Reference 5) state that:

> *To account for uncertainty in soil properties, the soil stiffnesses (horizontal, vertical, rocking, and torsional) employed in analysis shall include a range of soil shear moduli bounded by (a) 50 percent of the modulus corresponding to the best estimate of the large strain condition, and (b) 90 percent of the modulus corresponding to the best estimate of the low strain condition. For purposes of structural analysis three soil modulus conditions generally will suffice corresponding to (a) and (b) above, and (c), a best estimated shear modulus.*

...e guidelines are difficult to interpret as they do not specify what is considered to be low strain, high strain, and average values. Nevertheless, based on the data from the nearby Forked River plant, and the opinion of the URS/Blume geotechnical department which is very familiar with the soil conditions at Oyster Creek from their work on developing the site specific spectra,[9] a value of 6,000 ksf for the shear modulus seems to represent a good estimate of 90% of the soil modulus at low strain conditions (condition b of SSRT guidelines). This is the same value that the NRC study[5] on Oyster Creek used to satisfy this condition.

Available data for variations of shear modulus with the strain level in sand[10] leads to an estimate of $G$ value at high strain of about 3,000 ksf and at an average strain during SSE motion of about 4,000 ksf.

Hence, to follow the SSRT guidelines would lead to considering soil shear moduli of 1,500 ksf, 6,000 ksf, and 4,000 ksf to satisfy their conditions a, b, and c, respectively. Among these three values, a $G$ of 1,500 ksf is totally unrealistic for Oyster Creek since it translates to a soil with a shear wave

velocity of about 600 fps which is consistent with formations such as San Francisco Bay mud. Furthermore, the NRC study[5] on Oyster Creek also disregarded the results obtained from using $G = 1{,}500$ ksf (as stated in Appendix A of the referenced study), as they found the spectra to be essentially bounded by the responses from using $G$ values of 4,000 ksf and 6,000 ksf. Hence, only values of $G$ equal to 6,000 and 4,000 ksf are considered in this study.

Furthermore, in the case of calculating building responses (story shears and story overturning moments), a careful study of the results of the Oyster Creek parametric study[1] conducted by URS/Blume, where the average ($G = 6{,}000$ ksf) and the ±50% ($G = 4{,}000$ ksf and 9,000 ksf) values were used, shows that the variations in structural responses were fairly small. Given all the assumptions and approximations inherent in the overall analysis, these variations do not warrant multiple analyses. Building responses shown in Figure 21 of the NRC study[5] confirm this same trend. Hence, the reactor building responses are calculated for a value of $G = 6{,}000$ ksf only.

For the case of generating floor response spectra, two values of $G = 6{,}000$ ksf and $G = 4{,}000$ ksf are used and the results enveloped. The NRC study[5] and the URS/Blume parametric study[1] show a frequency shift in the peak of the floor spectra due to changes in soil properties and using the envelop of the two values of $G$ is sufficient to adequately account for the effects of this frequency shift on the floor spectra.

b. Embedment Effect. The SSRT has recommended[5] that only 50% of the embedment effects be considered to compensate for possible soil separation from the structure and other soil variations in the backfill. The SSRT recommendations of using 50% of the embedment is used in this study. The effects of embedment are calculated according to procedures developed by Kausel et al.[7]

c. Radiation Damping in the Soil. The most difficult problem to be resolved in using the lumped parameters for SSI effects is the radiation damping. The viscous dampers (dashpots) which are calculated from the infinite half-space solutions are frequency-dependent functions. The structural damping, on the other hand, is commonly given by percentage of critical modal damping which is best suited for a modal superposition time history analysis.

The most common approach to date has been to develop some form of equivalent composite modal damping that combines the structural and radiation dampings in one equivalent percent of critical damping value for each structural mode. This has been the basis that URS/Blume used in its parametric study[1] for Oyster Creek and NRC used in their evaluation.[5] This approach loses the frequency dependence of radiation damping and the equivalence of composite modal damping to the frequency dependent radiation damping is extremely difficult to demonstrate.

The SSRT recognized this problem during the review of the NRC study[5] and has suggested the following guidelines (Appendix C, Reference 5):

> *The geometric damping (radiation energy dissipation) is recognized to be frequency-dependent. However, in order to reduce the calculational effort (at least initially), and to be sure that excessive damping is not employed, it is recommended that values of damping be estimated theoretically (on frequency-independent basis) as follows:*
>
> *i) Horizontal to be taken as 75 percent of the theoretical values.**
>
> *ii) Vertical to be taken as 75 percent of the theoretical value.**
>
> *iii) Rotation (rocking and torsional) to be taken at 100 percent of the theoretical value.**
>
> *In the case of layered systems, the approach employed in establishing these values needs to be justified.*
>
> *4. The following analysis approaches are considered to be acceptable.*
>
> *i) When all composite modal damping ratios** are less than 20 percent, modal superposition approaches can be used without any validation check.*
>
> *ii) If in investigating the use of modal superposition approaches it is ascertained that a composite modal damping ratio** exceeds 20 percent, one must perform a validation analysis. To perform this validation, it is generally acceptable to use a time-history analysis in which the energy dissipation associated with the structure is included with the structural elements, and that associated with the soil is included with the soil elements. In structures response spectra obtained from a modal damping throughout*

*the frequency range of interest must be similar to or more conservative than those obtained from the validation analyses.*

---

**As calculated by generally accepted methods, as for example given in the book* *__Vibration of Soils and Foundations__*, *by F. E. Richart, Jr., J. R. Hall, Jr., and R. D. Woods, Prentice-Hall, Inc., 1970.*

***As defined by generally accepted methods.*

T· :RC study[5] for Oyster Creek did not follow these guidelines as they were d ..oped after most of the work was completed. To assess the effect of some c nese guidelines, a parametric study was conducted in which the floor spec- from a direct integration procedure were compared with those of a modal .rposition procedure. In the direct integration procedure, a damping ma- (as opposed to composite modal damping in the modal superposition method) developed which included the soil radiation damping (dashpot) coefficients .perly without using any manipulations to convert them to equivalent compo- .e modal damping. However, since the analysis was done in the time domain : frequency dependence of the coefficients could not be handled and one .ngle value for each coefficient was used.

:ne floor spectra developed by the equivalent composite modal damping approach .ere below those computed by the more exact direct integration dashpot approach. Hence, the former had to be scaled up to envelop the dashpot results (see Figures A-8 through A-12 of Reference 5). Thus, the approach to use the approximate composite modal damping not only required more analytical work in the form of a validation study by the direct integration analysis mentioned above, but also introduced conservatisms into the final floor spectra as they were scaled by a uniform factor to envelop the validation study spectra.

In response to similar problems, URS/Blume has developed an advanced proprietary program FREDA that can be used to calculate the structural responses and floor spectra in the frequency domain. In this procedure the frequency dependence of the radiation damping is maintained without resorting to the manipulations suggested in the SSRT guidelines which are of weak technical bases and contain undue conservatisms. The SSRT has stated in the presentation of their guidelines that more rigorous approaches, such as that adopted in this study, would be acceptable in lieu of these guidelines.

The benefits of the approach used in this study are that:

- 25% reduction in the radiation damping for the horizontal and the vertical directions required by the SSRT guidelines is not necessary.
- A limit of 20% maximum composite modal damping is not necessary.
- The step of calculating a composite modal damping representing the combined structural and soil radiation damping is not necessary.
- Undue conservatism in the final results are reduced as a consequence.

### Summary of Horizontal Dynamic Model Modifications to Include SSI Effects

summary, the following modifications to the URS/Blume 1965 horizontal dy-.aic model are made for the present study:

1. The fixed-base condition has been replaced by springs and dashpots representing the soil impedances (Figure 3). The soil impedances are frequency-dependent functions which are computed by the formulae presented by Kausel et al.[7] Half of the reactor embedment effects are used to calculate the soil impedances in conformance with SSRT guidelines.
2. Structural responses are generated from the model which uses $G = 6{,}000$ ksf for soil modulus in calculating soil impedance values.
3. Two models that use soil modulus values of 6,000 ksf and 4,000 ksf to compute the soil impedance values have been used to generate the floor response spectra.

## 2.6 Vertical Dynamic Models

The model developed for the Oyster Creek plant by URS/Blume in 1965[4] can be modified to estimate vertical responses. However, the lumped mass cantilever model neglects the out-of-plane flexibility of the floor slabs which may significantly influence the vertical floor spectra.

In general, the lumped mass cantilever model is adequate for structures where floor slabs can be shown to be rigid (fundamental frequency higher than 33 Hz) in their out-of-plane dynamic behavior. This is not the case in most of the

Oyster Creek reactor building where the floors are supported vertically by massive drywell walls, outside walls of varying thicknesses, interior walls that do not continue to the floor mat, and columns which at times continue for several floors. Only the floor at elevation 0 ft 0 in. is found to be rigid in its out-of-plane behavior along with the foundation mat.

This represents a very complex vertical dynamic system and the vertical amplifications above grade level vary considerably throughout a floor at any given elevation. Furthermore, input to a floor at a given elevation will vary as the walls and columns do not uniformly amplify the ground motion input. These local variations in the input motions to the floor slab cannot be accounted for in the lumped mass cantilever model.

To account for all the complex aspects of the vertical response in an efficient form, three dynamic vertical models were used. A lumped mass model was used to generate floor spectra at elevations -19 ft 0 in., 0 ft 0 in., and 156 ft 9 in.; a detailed finite-element model to generate floor spectra at elevations 23 ft 6 in., 51 ft 3 in., 75 ft 3 in., 95 ft 3 in., and 119 ft 3 in.; and a beam model to generate spectra at the overhead bridge crane supports at elevation 138 ft 0 in. Following is a detailed description of each of these three models.

3.7.1.10

a. Lumped-Mass Vertical Model. The lumped-mass model developed by URS/Blume[4] in 1965 is modified to obtain a vertical model suitable for calculating floor spectra at elevations -19 ft 0 in., 0 ft 0 in., and 158 ft 9 in. The modification involves changing the fixed base condition to include a frequency-dependent vertical spring and dashpot to represent SSI effects (see Figure 4). The lumped-mass model is considered adequate for calculating floor spectra at the above elevations since the base mat (elevation -19 ft 0 in.) and first floor (elevation 0 ft 0 in.) of the reactor building are both rigid in their out-of-plane responses (i.e., have a fundamental frequency higher than 33 Hz). The spectra at elevation 158 ft 0 in. is at the top of columns and does not represent the magnifications of motions elsewhere in the roof.

b. Detailed Vertical Model. A detailed finite-element model of the reactor building between elevations 23 ft 6 in. and 119 ft 3 in. is developed to

generate vertical floor spectra at various locations in floors between these two elevations. The floor slabs and walls (except for the drywell) are modeled with quadrilateral plate elements with five degrees of freedom at each node. Floor beams are modeled using prismatic beam elements with adequate accounting for the difference between the neutral axis of the beam and slab in the floor. The drywell is modeled by a series of vertical prismatic beam-column elements connected to the floor plate elements. Figure 5 shows a schematic representation of the plate element mesh of the floor at elevation 95 ft 3 in. and a partial view of the finite-element mesh of the exterior building walls and beam-column elements representing the dry well. The input to this model is the vertical acceleration time history at elevation 23 ft 6 in. calculated from the lumped mass model described above. The locations in each floor where floor spectra are calculated were identified by GPU.

c. Detailed Vertical Crane Model. A detailed model of the overhead bridge crane support system is developed in order to generate the vertical floor spectra at this level. The mathematical model is presented in Figure 6. The steel columns are modeled as truss elements while the girders supporting the crane rail are modeled as beam elements. The crane is assumed to be in the center span between column lines R3 and R4 for worst case consideration. The weight of crane was specified by GPU to be 125 tons for the hoist, 170 kips for the bridge, and 88 kips for the trolley. These masses are incorporated in the mathematical model. The input to this model is the vertical acceleration time history at elevation 119 ft 3 in. computed from the lumped-mass model described above.

3. DISCUSSION OF RESULTS

:uctural Response for Horizontal Seismic Input

:ctural responses in the east-west direction, comprise of the maximum . floor accelerations, maximum relative floor displacements, maximum :ears, and maximum story overturning moments, are calculated for the .se in the horizontal direction. The solutions are for the modified .al dynamic model described in Chapter 2. For this analysis, a single 6,000 ksf for soil shear modulus is used and half of the reactor em- for effects of embedment on soil impedance values. The results are : from frequency domain analysis using the URS/Blume proprietary com- ogram FREDA. Table 1 presents the computed structural responses at it levels of the building.

..or Response Spectra

: horizontal east-west direction, floor acceleration spectra are gen- at nine floors, elevations 156'-9", 138'-0", 119'-3", 95'-3", 75'-3", , 23'-6", 0'-0", and -19'-0", using the lumped-mass stick model. The a curves are generated for 2, 3, 4, and 7% of critical damping. Two of floor response spectra are developed for soil shear modulus values 000 ksf and 4,000 ksf. The two sets of spectra are enveloped to produce inal raw spectra. Peaks in the raw spectra at structural frequencies are broadened and the rest of the spectra smoothed in accordance with project criteria. The spectra computations are done through the URS/Blume proprietary computer program FLSPEC which calculates, broadens the peaks, smooths, plots, and digitizes the floor response spectra in a single run for the various damping values requested. Figures 7 to 15 show the final smooth floor spectra curves for the horizontal east-west direction.

15% See Att A

The floor acceleration response spectra in the vertical direction have been computed from the three mathematical models described in Chapter 2. Floor spectra are generated at fourteen locations in the structure as requested by GPU. The procedures of enveloping the spectra for the two values of soil-shear modulus, the peak broadening, and spectra smoothing are all done using the same procedures described above for the horizontal spectra. Figures 16 to 29 show the final smooth floor spectra for the vertical direction.

## 4. SUMMARY AND CONCLUSIONS

Horizontal and vertical broadened and smoothed acceleration response spectra are generated for the Oyster Creek reactor building. These spectra will subsequently be used in qualifying various items of Category I electrical and mechanical equipment for seismic motion.

A written document delineating the criteria to be used in the modeling, analyses, and generation of floor response spectra of the reactor building is prepared at the inception of the work. The rest of the work is started after review and approval of this document by GPU.

A synthetic acceleration time history is generated whose spectrum closely matches the NRC site-specific spectrum for the Oyster Creek plant. This time history is later used for the floor spectra generation.

The east-west horizontal lumped-mass stick model developed by URS/Blume in 1965, modified to consider SSI effects, is used for structural response computation as well as horizontal floor acceleration spectra generation. Soil impedances - stiffnesses (springs) and radiation damping (dashpots) - are considered at the base of the structure. Solutions are computed in the frequency domain to properly consider the frequency-dependence of the soil impedances. A single soil-shear modulus value of 6,000 ksf is considered for calculation of structural responses such as displacements, accelerations, shears, and overturning moments. Two bounding values of soil shear moduli of 6,000 ksf and 4,000 ksf are used for generation of floor response spectra which are then enveloped to produce the final floor spectra.

Three models are used to generate the floor response spectra in the vertical direction: lumped-mass stick model to generate the floor spectra at elevations -19'-0", 0'-0", and 156'-9"; a detailed finite-element model for floor response spectra at elevations 23'-6", 51'-3", 75'-3", 95'-3", and 119'-3"; and a detailed crane model to generate the floor spectra at the crane supports.

## 5. REFERENCES


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4. *Jersey Central Nuclear Reactor Project - Earthquake Analysis: Reactor Building*, report prepared for General Electric Company by John A. Blume & Associates, Engineers, San Francisco, California, June 1965.

5. *Seismic Review of the Oyster Creek Nuclear Power Plant as Part of the Systematic Evaluation Program, NUREG/CR-1981, UCRL-53018, RD, RN*, prepared for U.S. Nuclear Regulatory Commission by Lawrence Livermore Laboratory, Livermore, California, November 1980.

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8. U.S. Nuclear Regulatory Commission, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-75/087*, LWR Edition, March 1979.

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10. Seed, H. B., and I. M. Idriss, *Soil Moduli and Damping Factors for Dynamic Response Analysis*, Earthquake Engineering Research Center, Report No. EERC 70-10, University of California, Berkeley, California, December 1970.

TABLE 1

SEISMIC RESPONSES OF THE HORIZONTAL REACTOR BUILDING MODEL (SOIL MODULUS = 6,000 ksf)*

| Elevation | Maximum Relative Displacement (in.) | Maximum Absolute Acceleration (g) | Maximum Story Shears (kips) | Maximum Overturning Moment (k-ft) |
|---|---|---|---|---|
| 156'-9" | 0.6653 | 0.988 | -- | -- |
| 138'-0" | 0.2696 | 0.504 | 671 | 12,581 |
| 119'-0" | 0.0742 | 0.300 | 938 | 30,045 |
| 95'-3" | 0.0633 | 0.264 | 2,806 | 96,340 |
| 75'-3" | 0.0524 | 0.235 | 6,793 | 224,450 |
| 51'-3" | 0.0368 | 0.203 | 10,324 | 465,160 |
| 23'-6" | 0.0171 | 0.186 | 13,914 | 838,840 |
| 0'-0" | 0.0089 | 0.175 | 16,208 | 1,206,600 |
| -19'-0" | 0.0000 | 0.164 | 17,911 | 1,542,400 |

*These results are for the east-west horizontal model for seismic input in the east-west direction.

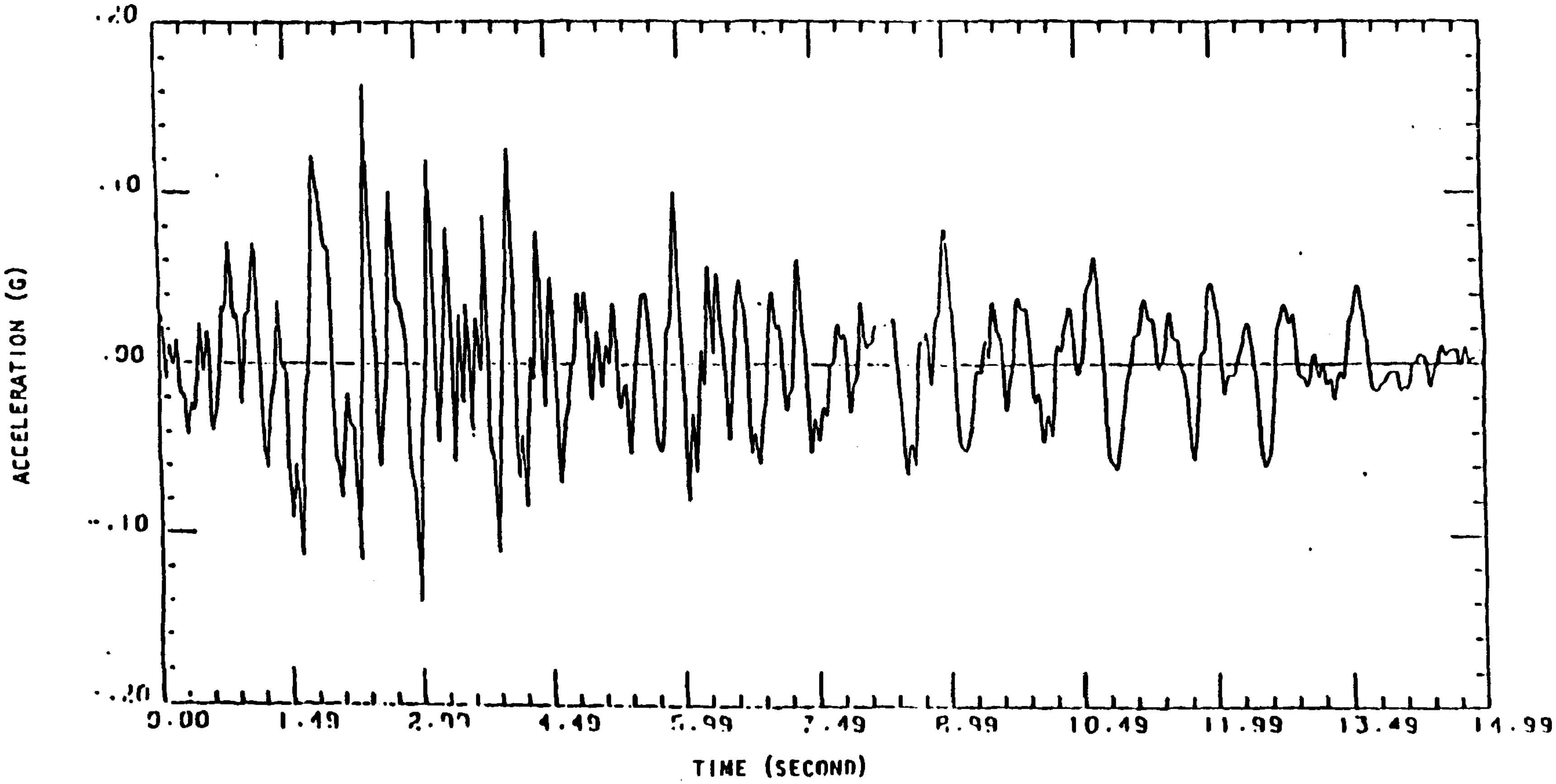



FIGURE 1 TIME-HISTORY COMPATIBLE WITH OYSTER CREEK SITE-SPECIFIC DESIGN SPECTRUM (PGA = 0.165g)

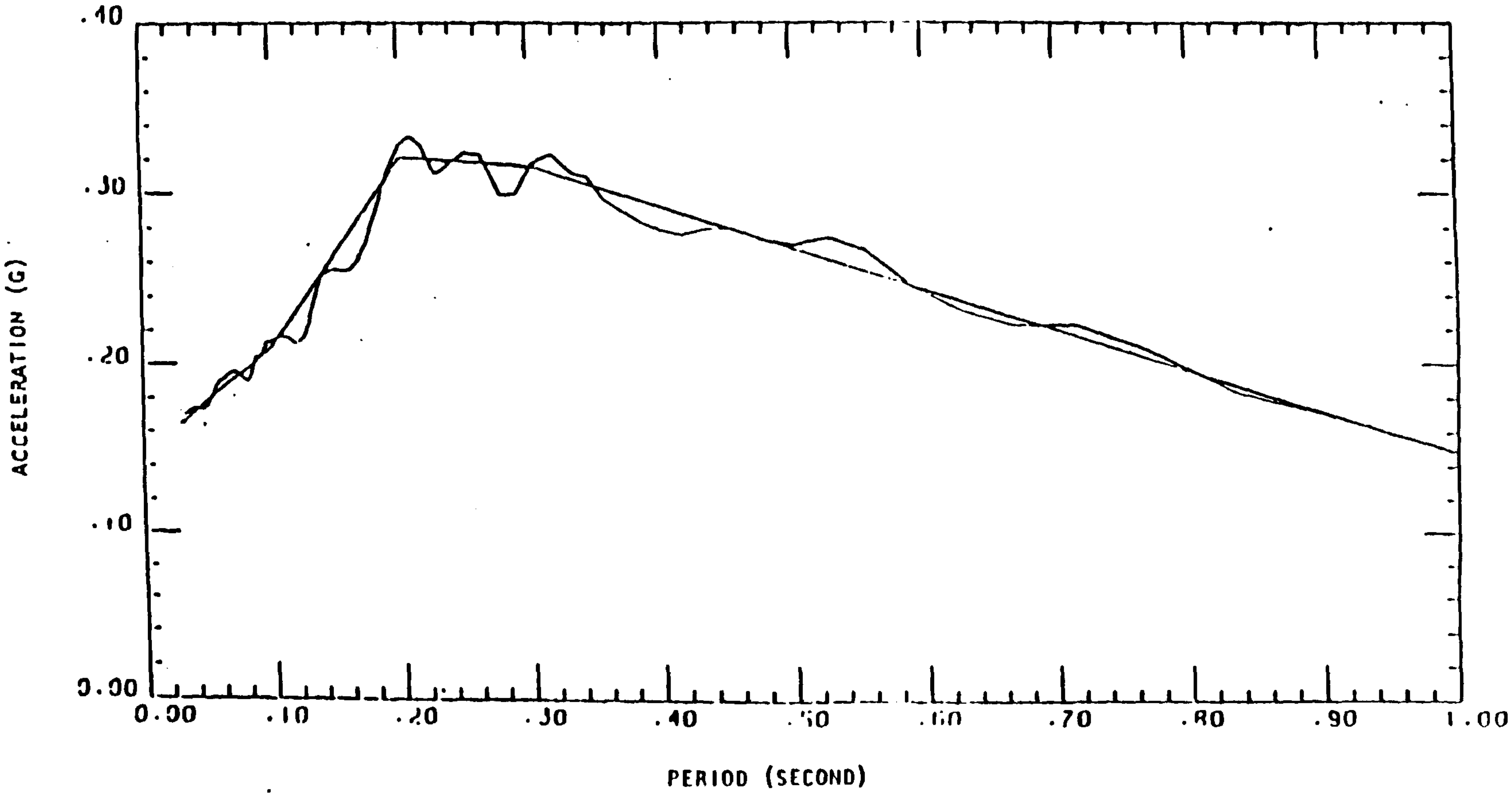



FIGURE 2 COMPARISON OF 10% DAMPED SITE-SPECIFIC OYSTER CREEK DESIGN RESPONSE SPECTRUM WITH RESPONSE SPECTRUM OF THE TIME-HISTORY

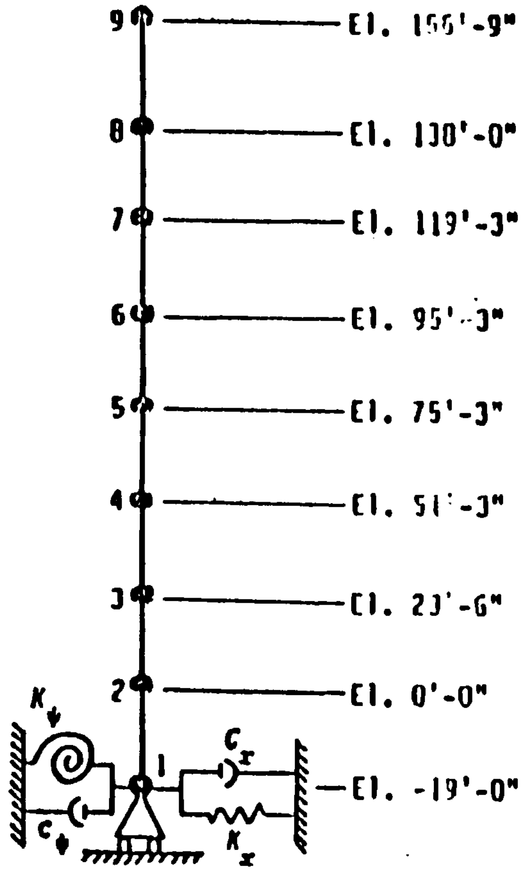



FIGURE 3 OYSTER CREEK REACTOR BUILDING HORIZONTAL EAST-WEST MODEL

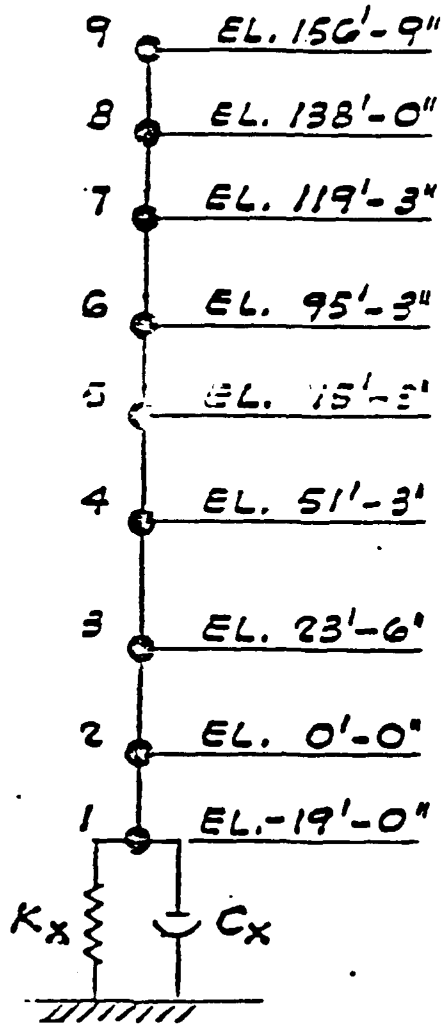



FIGURE 4 OYSTER CREEK REACTOR BUILDING
VERTICAL LUMPED-MASS MODEL

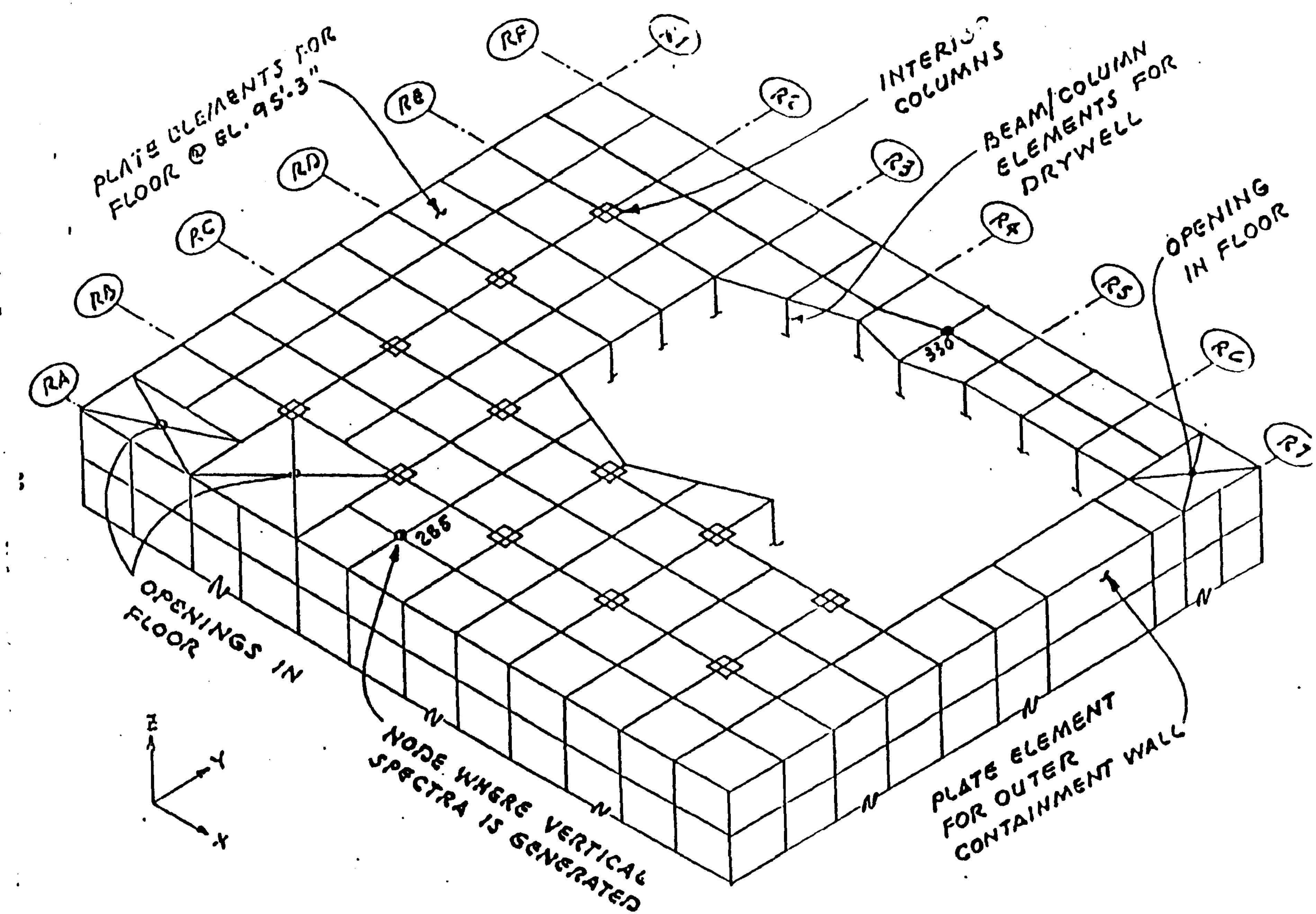



FIGURE 5 SCHEMATIC REPRESENTATION OF THE

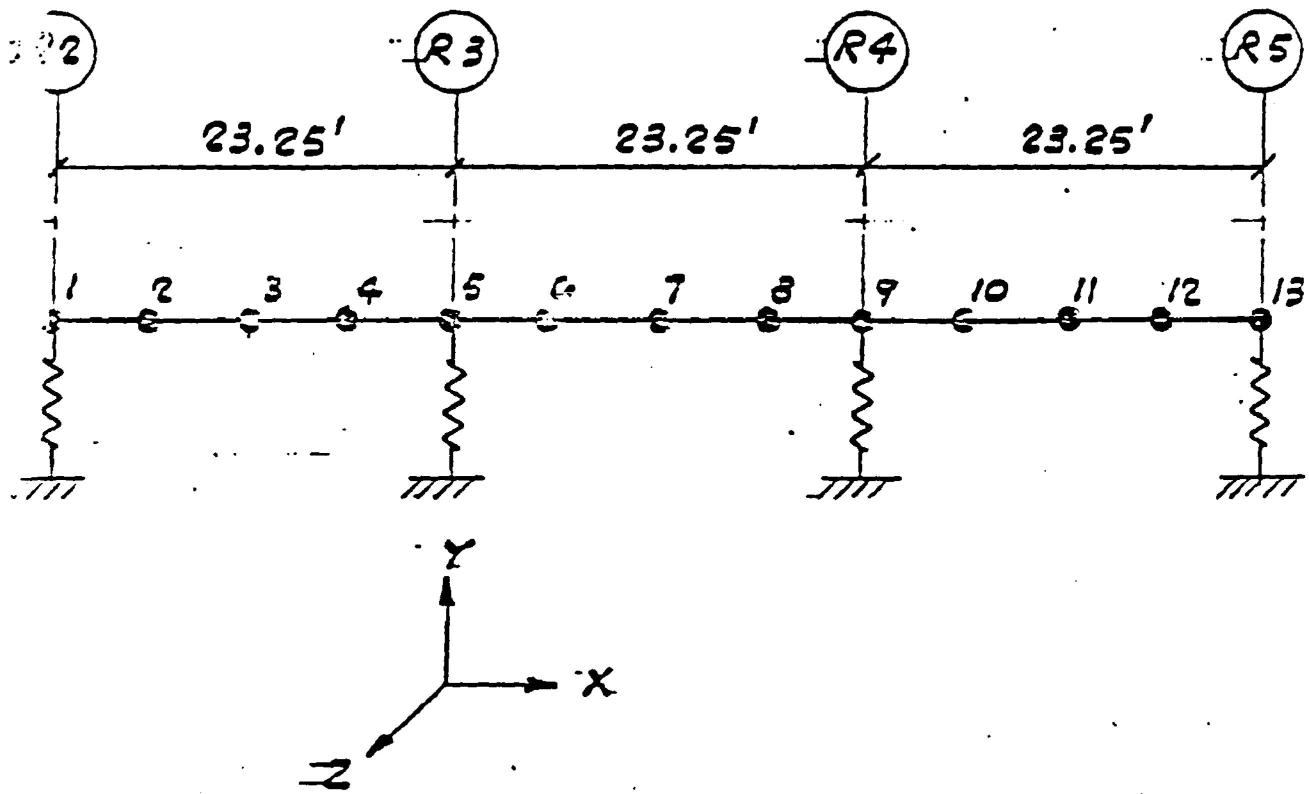



FIGURE 6 OVERHEAD CRANE VERTICAL MODEL

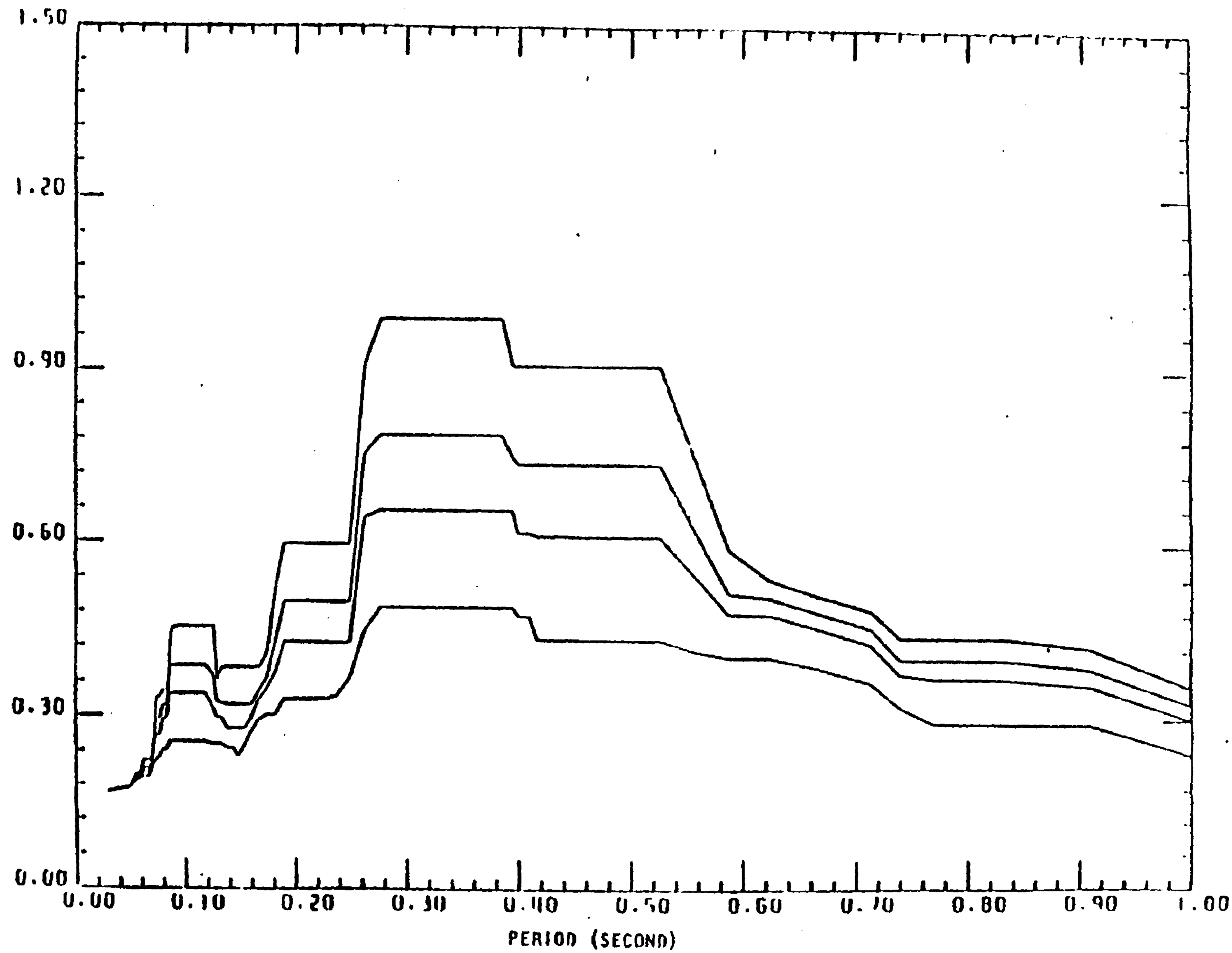
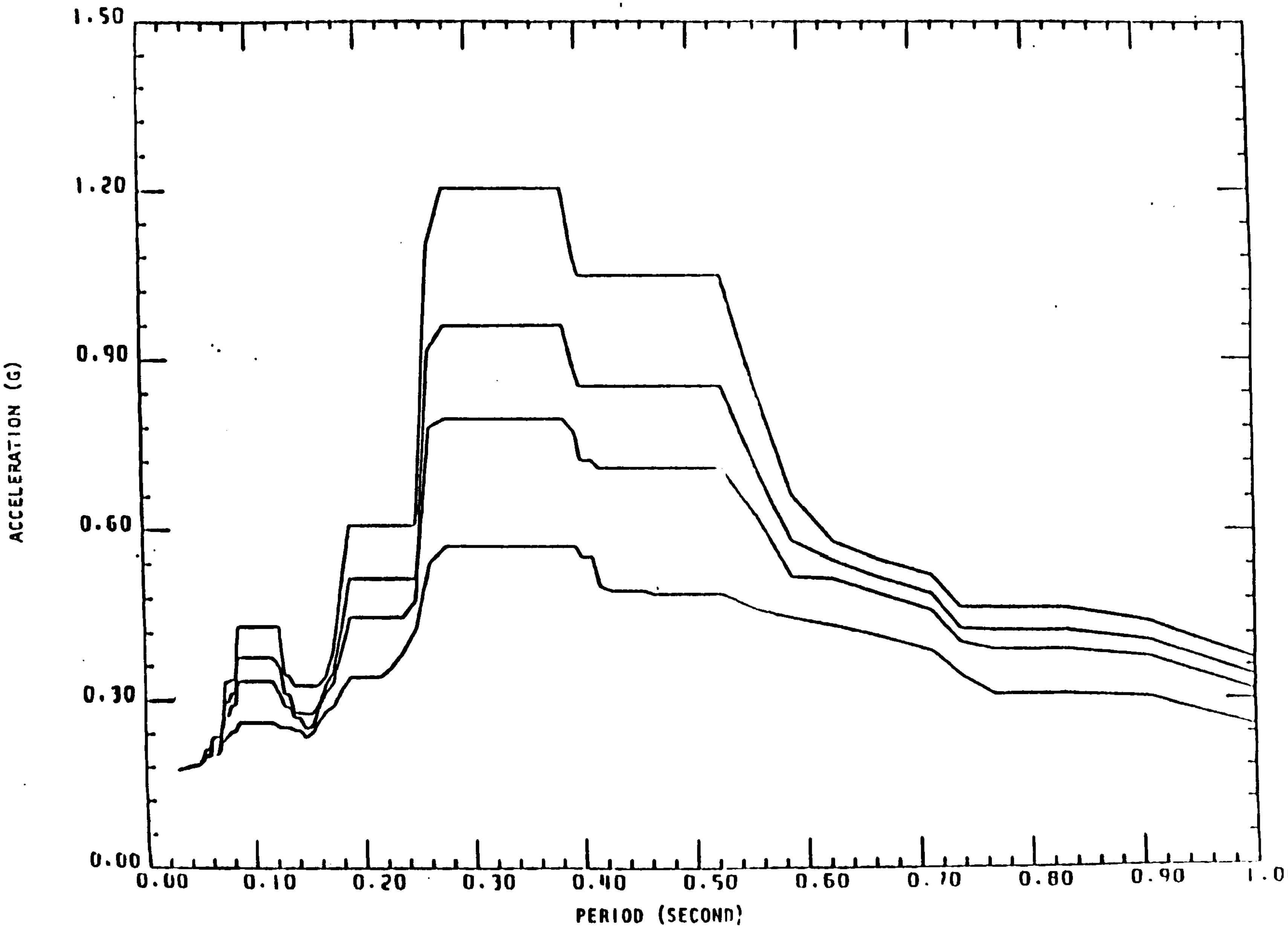



FIGURE 7 HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 1 El. -19 FT (2%, 3%, 4% and 7% Damping)


ACCELERATION (G)
1.50
1.20
0.90
0.60
0.30
0.00
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
PERIOD (SECOND)

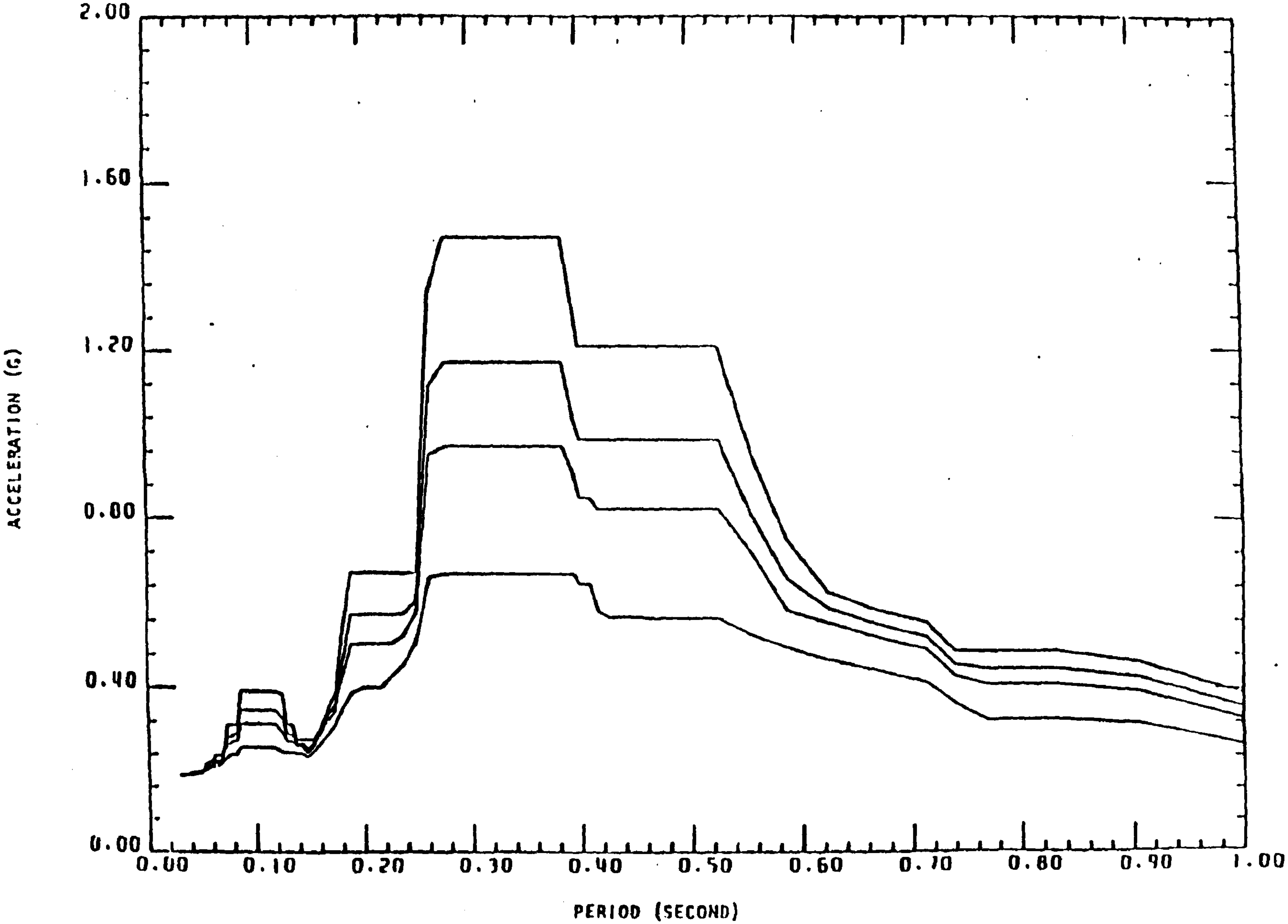
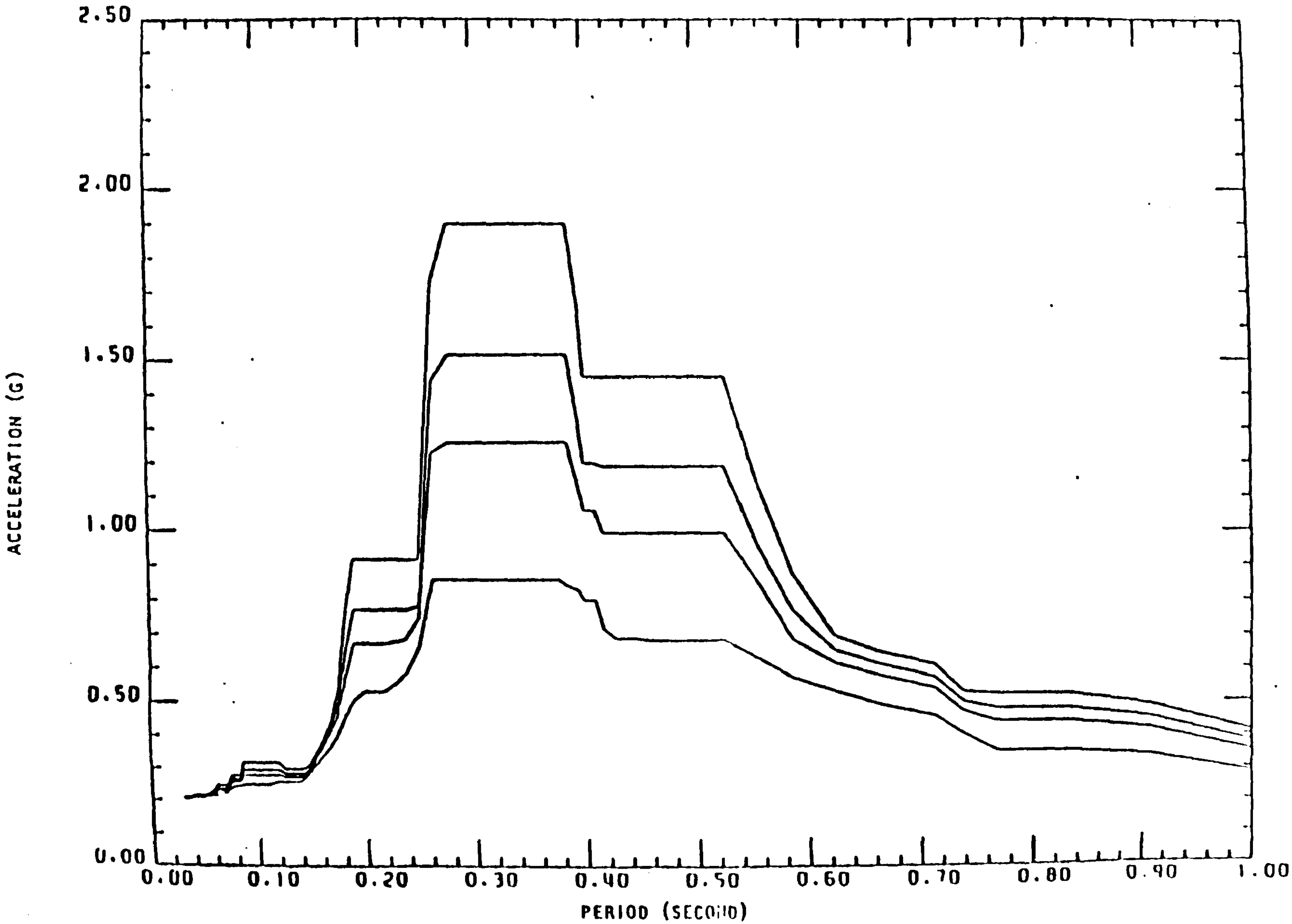



FIGURE 9 HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 3 EL. 23 FT 6 IN (2%, 3%, 4% and 7% Damping)


2.50
2.00
1.50
1.00
0.50
0.00
ACCELERATION (G)
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
PERIOD (SECOND)

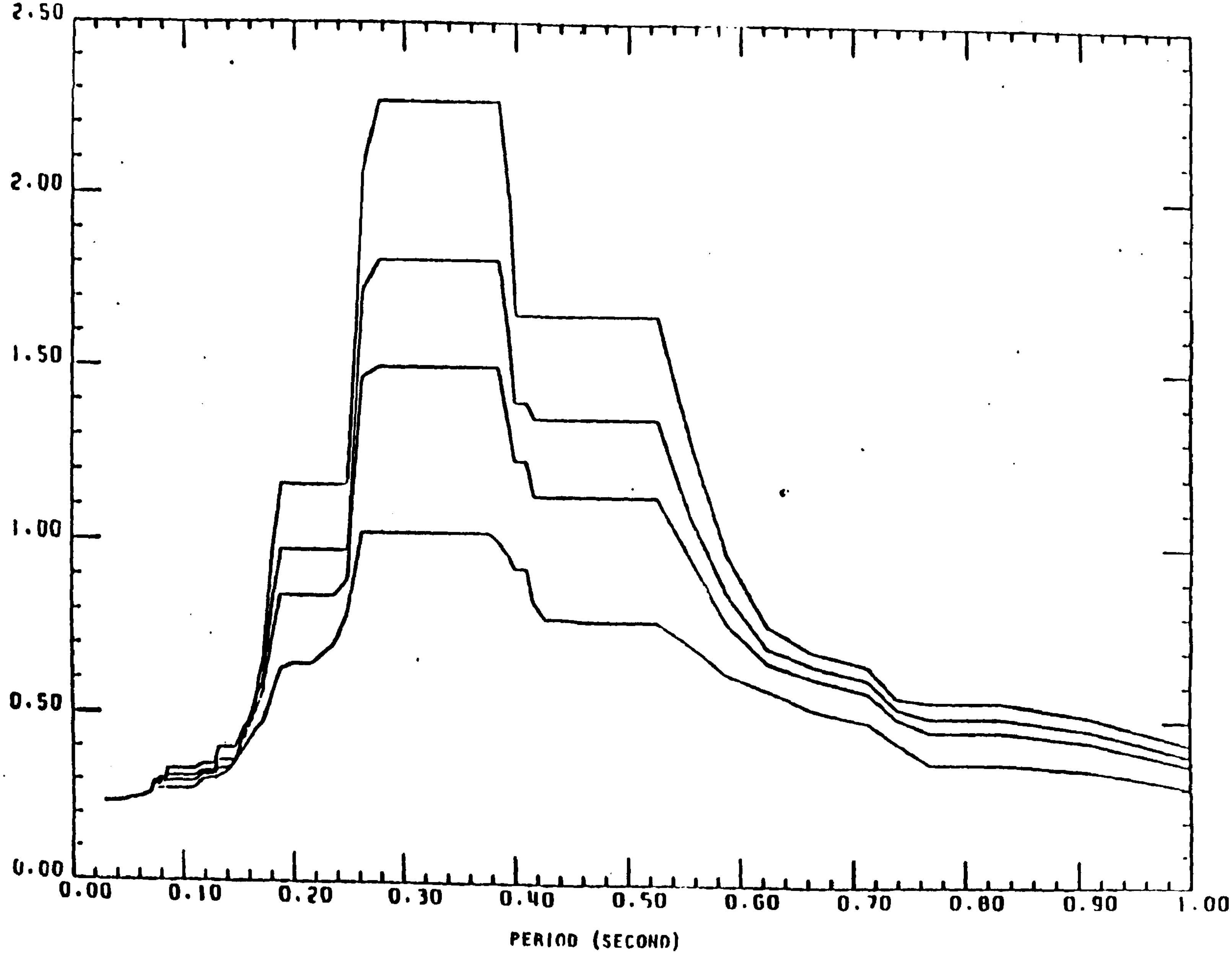



FIGURE 11. HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 5 EL. 75 FT 3 IN (2%, 3%, 4% and 7% Damping)

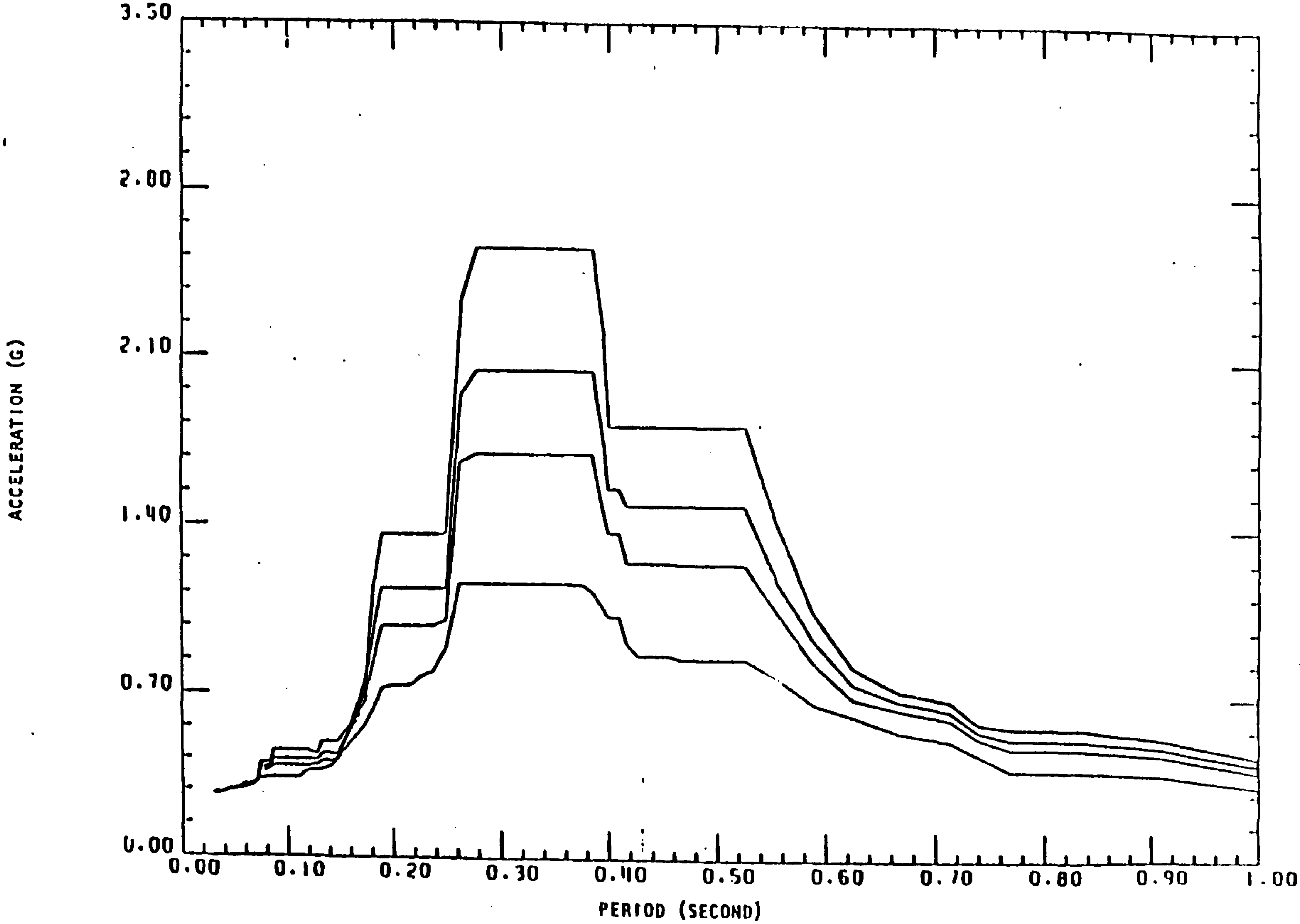



FIGURE 12 HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 6 EL. 95 FT 3 IN (2%, 3%, 4% and 7% Damping)

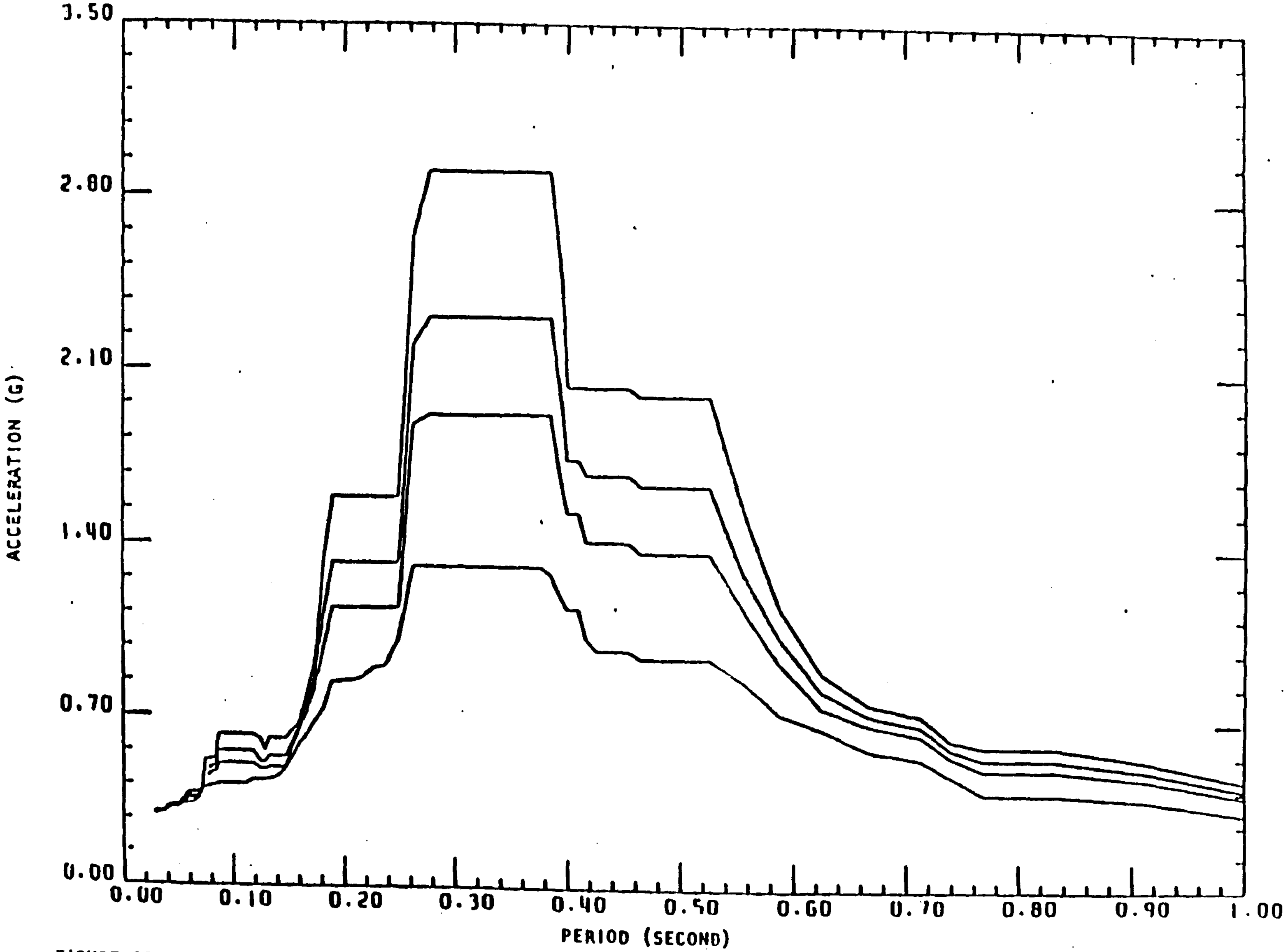



FIGURE 13 HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 7 EL. 119 FT 3 IN (2%, 3%, 4% and 7% Damping)

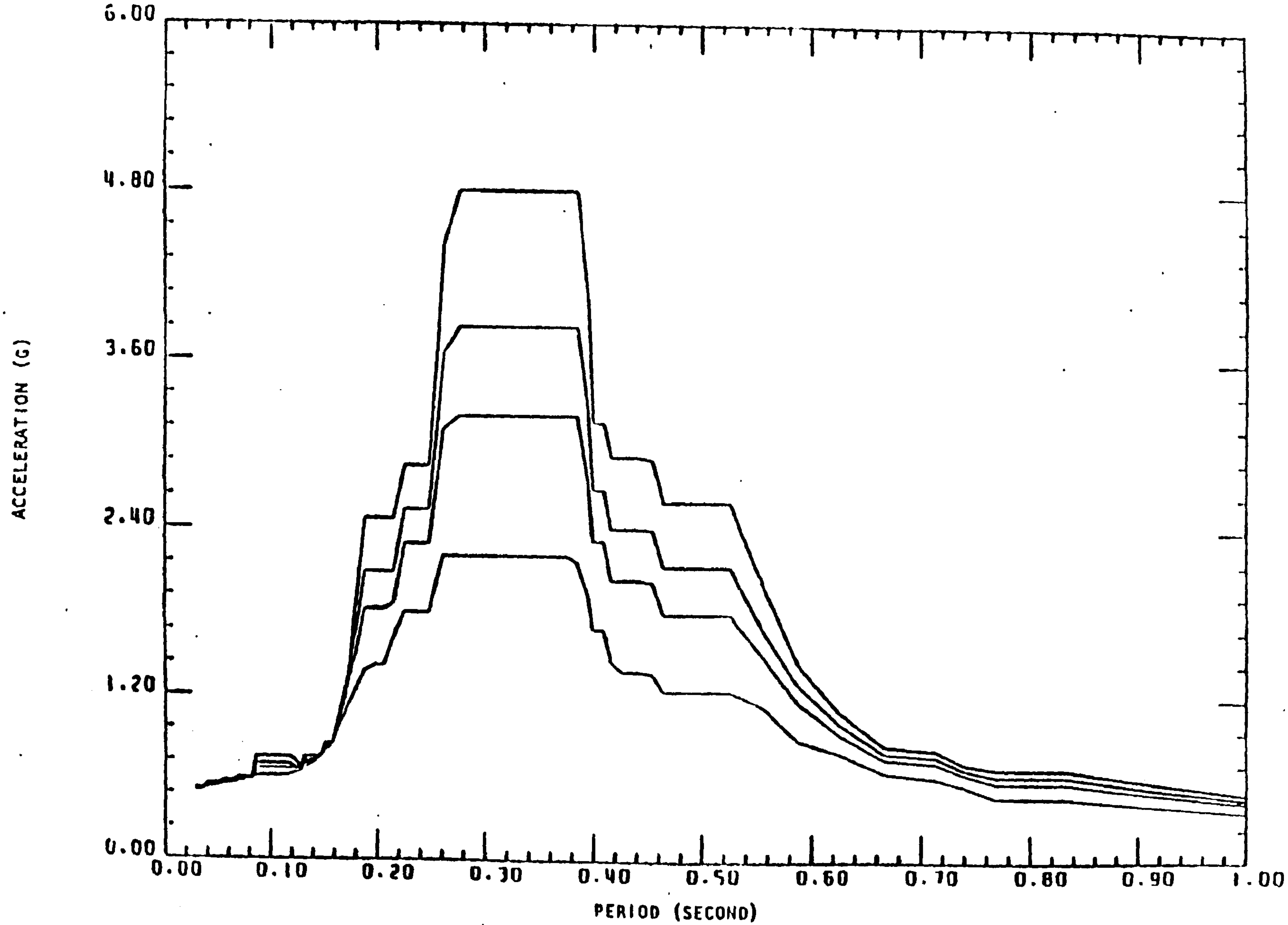



FIGURE 1[illegible] HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 8 EL. 138 FT 0 IN (2%, 3%, 4% and 7% Damping)

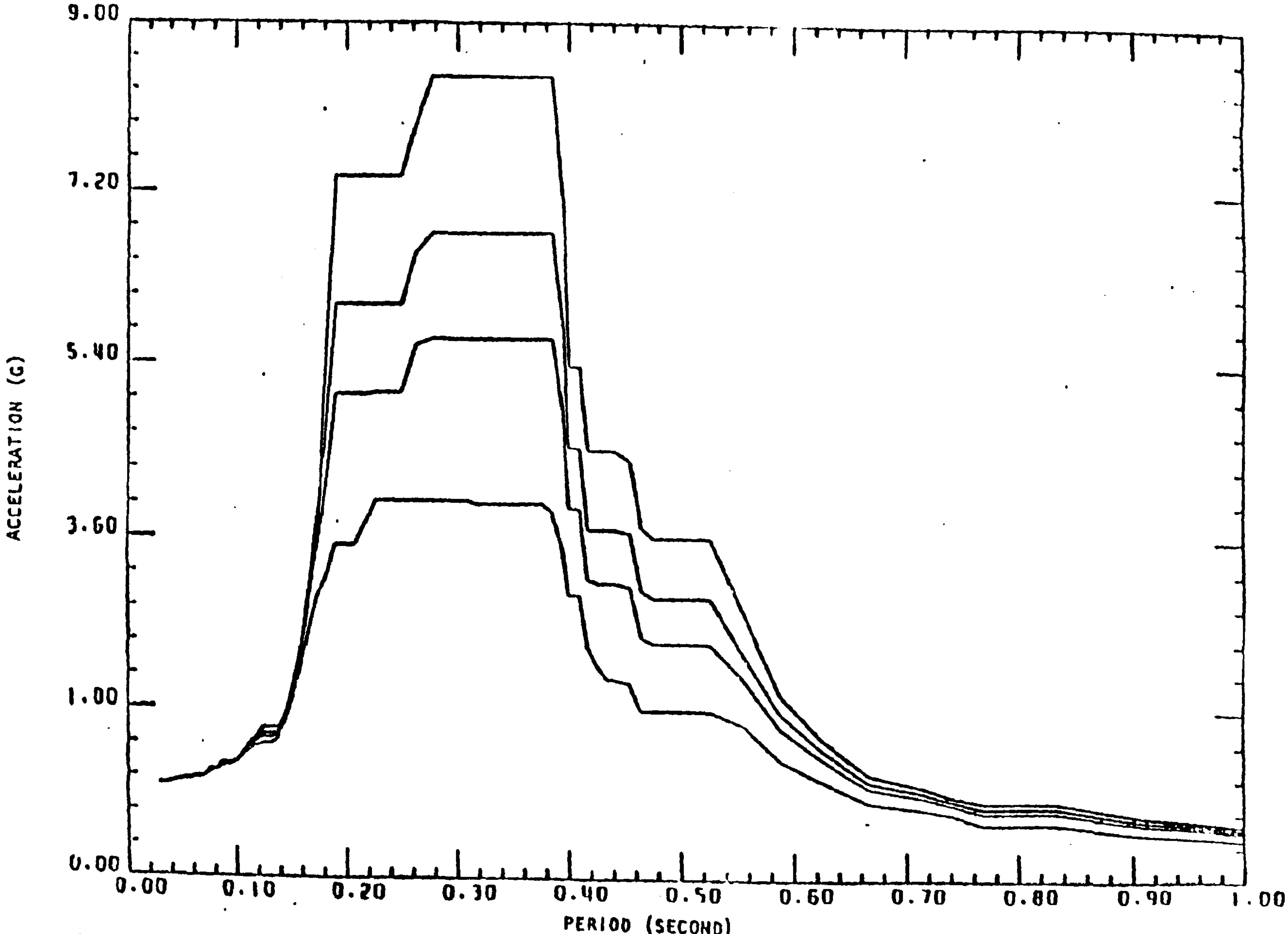
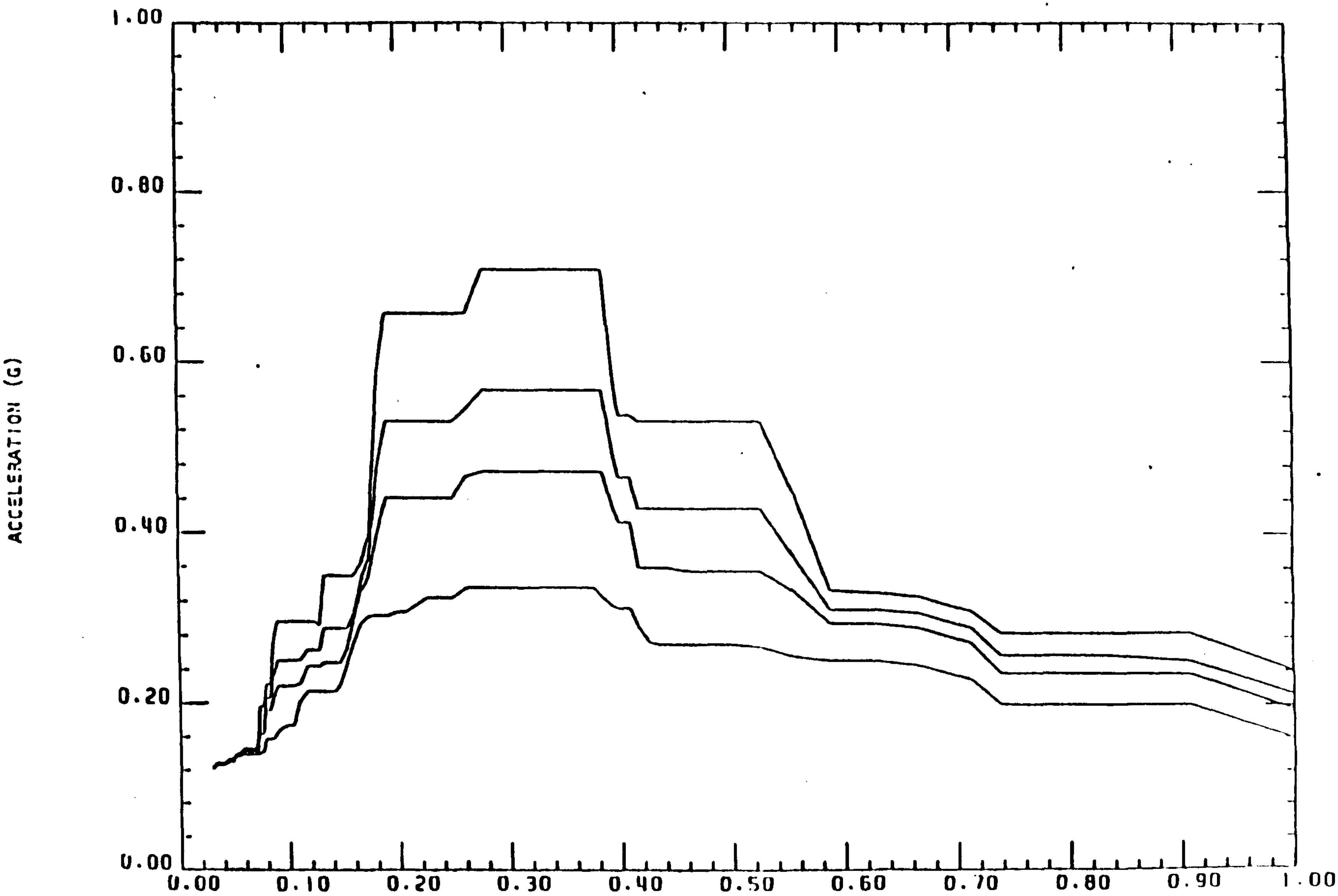



FIGURE 15 HORIZONTAL EAST-WEST FLOOR RESPONSE SPECTRA FOR NODE 9 EL. 156 FT 9 IN (2%, 3%, 4% and 7% Damping)


ACCELERATION (G)
1.00
0.80
0.60
0.40
0.20
0.00
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00

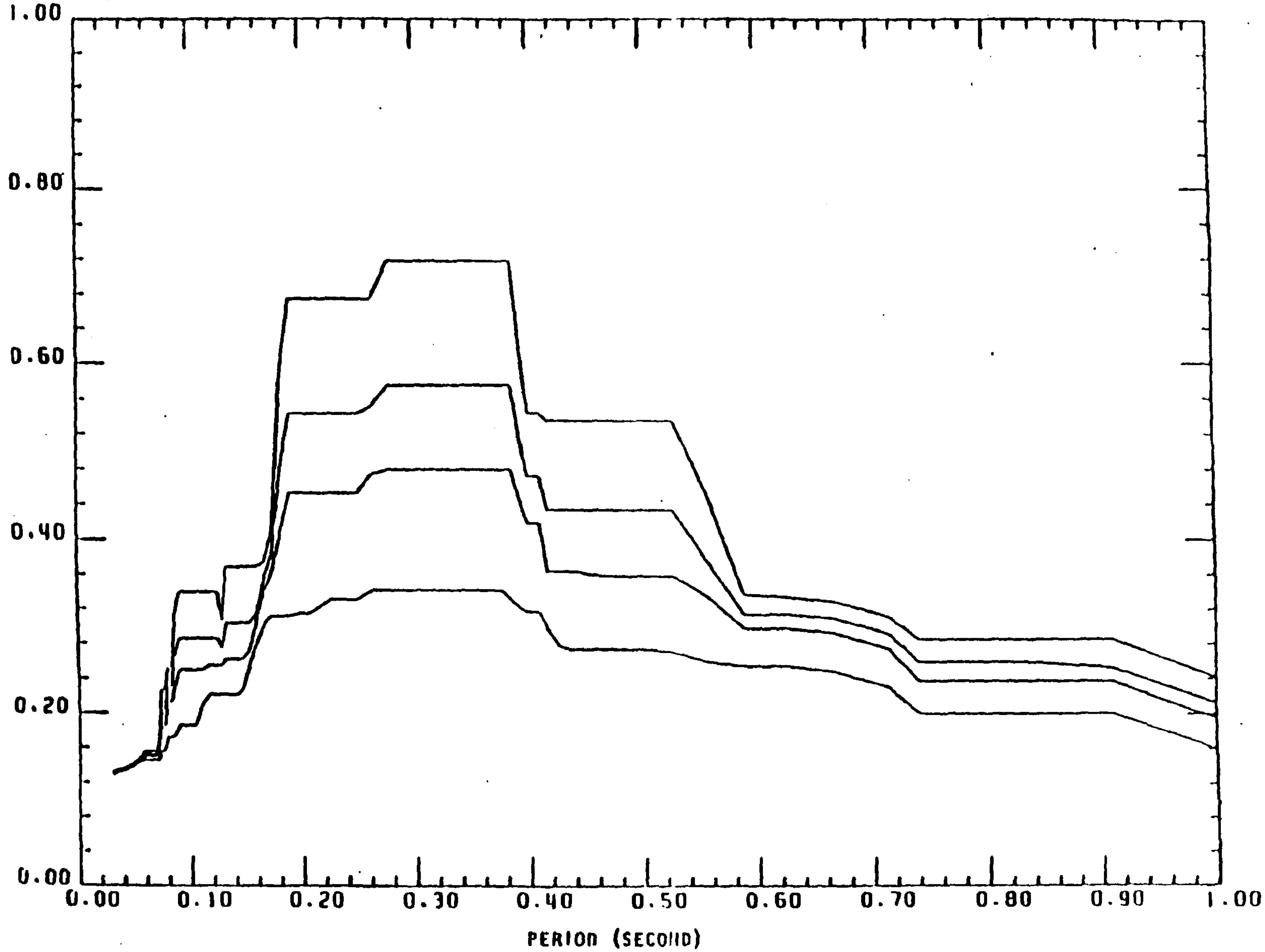
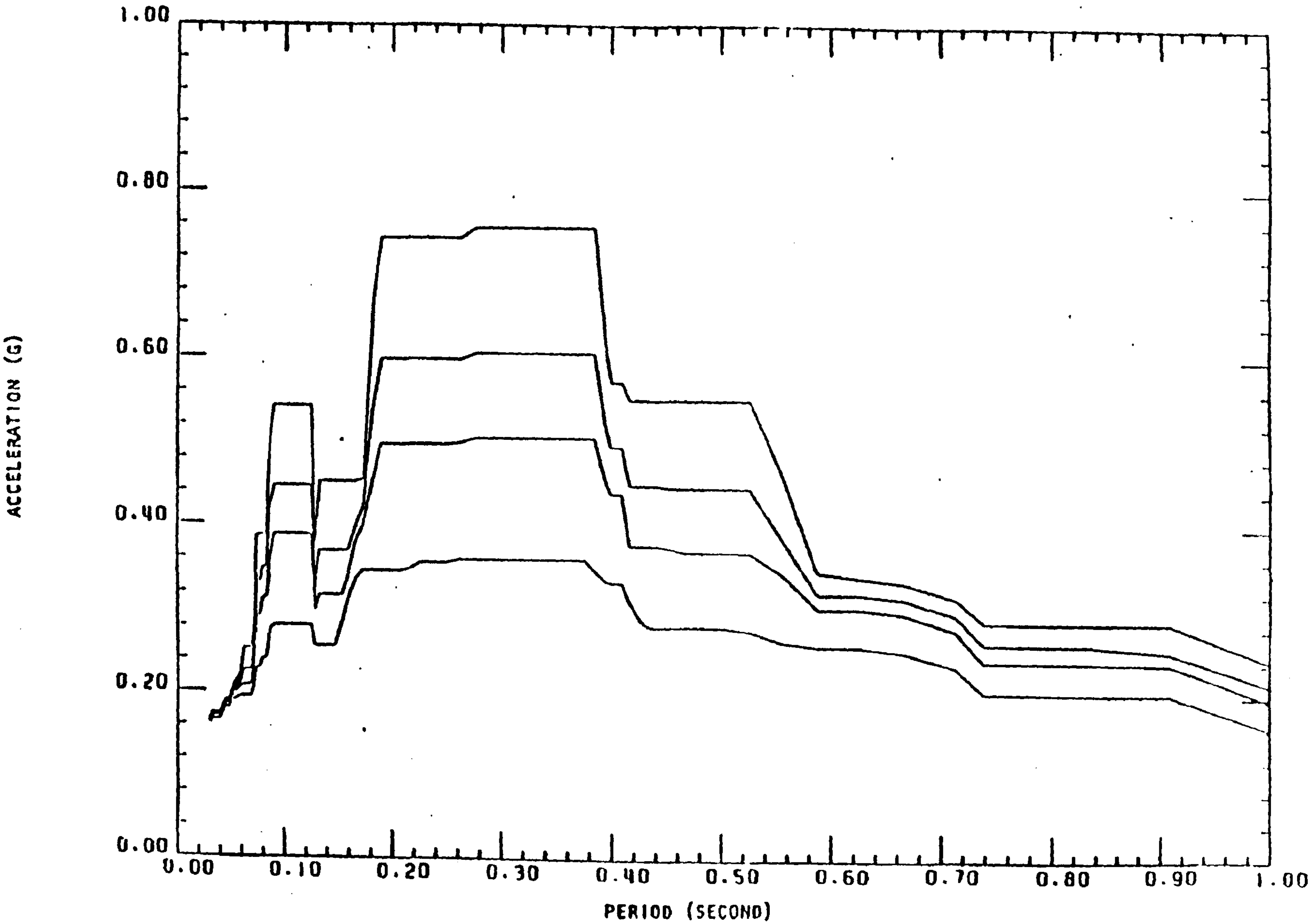
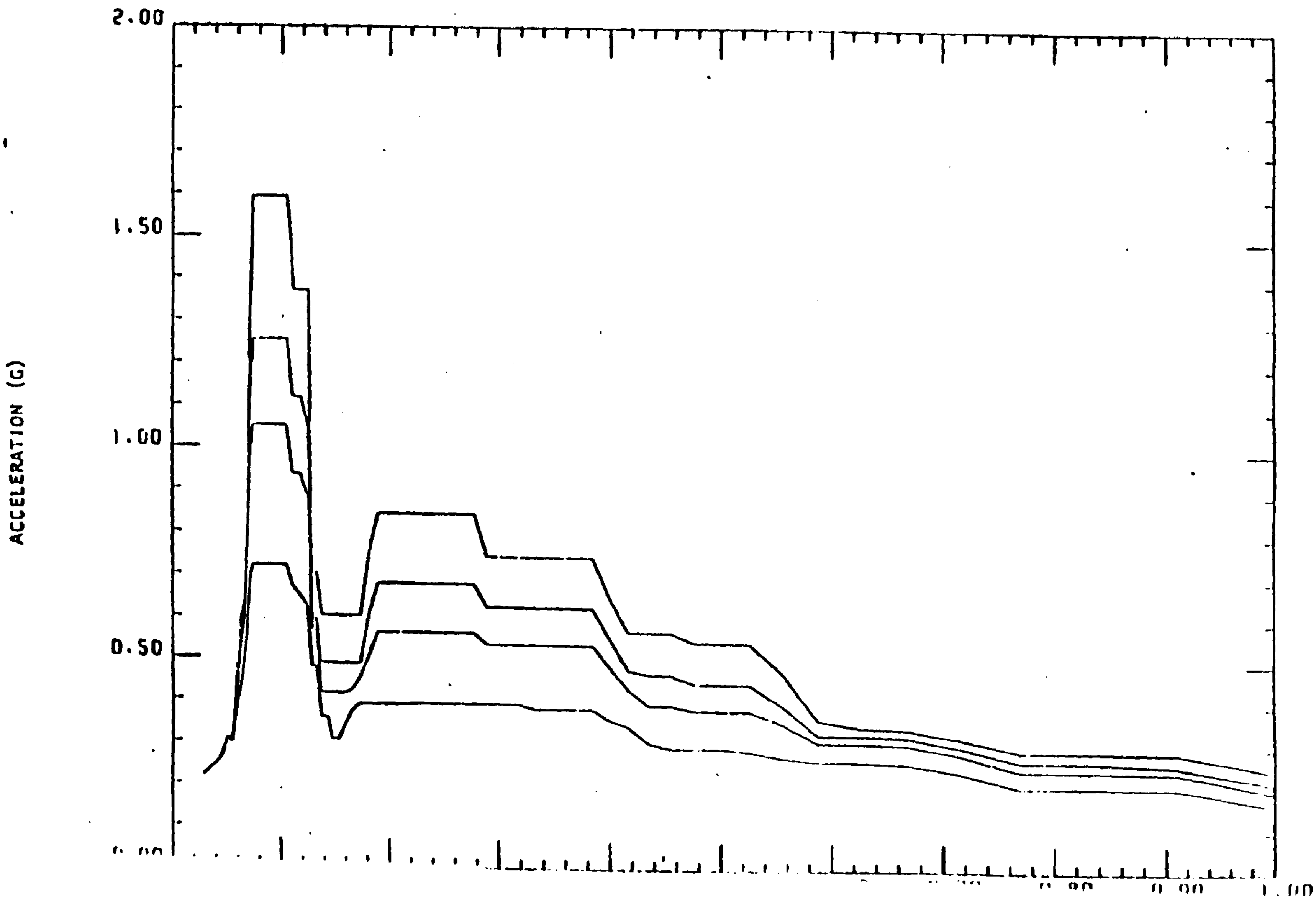



FIGURE 17 VERTICAL FLOOR RESPONSE SPECTRA FOR NODE 2 EL. 0 FT (2%, 3%, 4% and 7% Damping)




... [illegible] 3%, 4% and 7% Damping)


2.00
1.50
1.00
0.50
ACCELERATION (G)
1.00

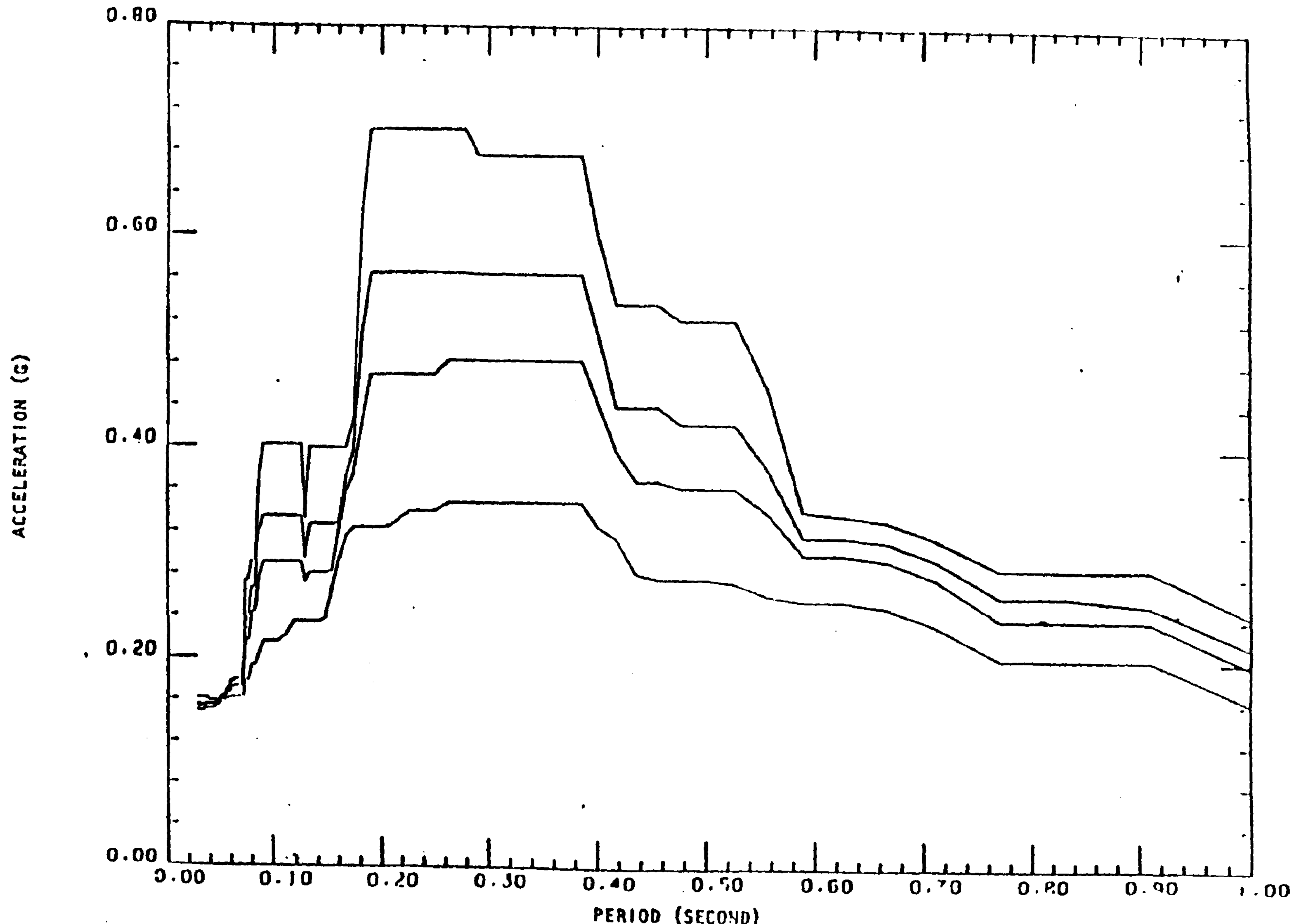
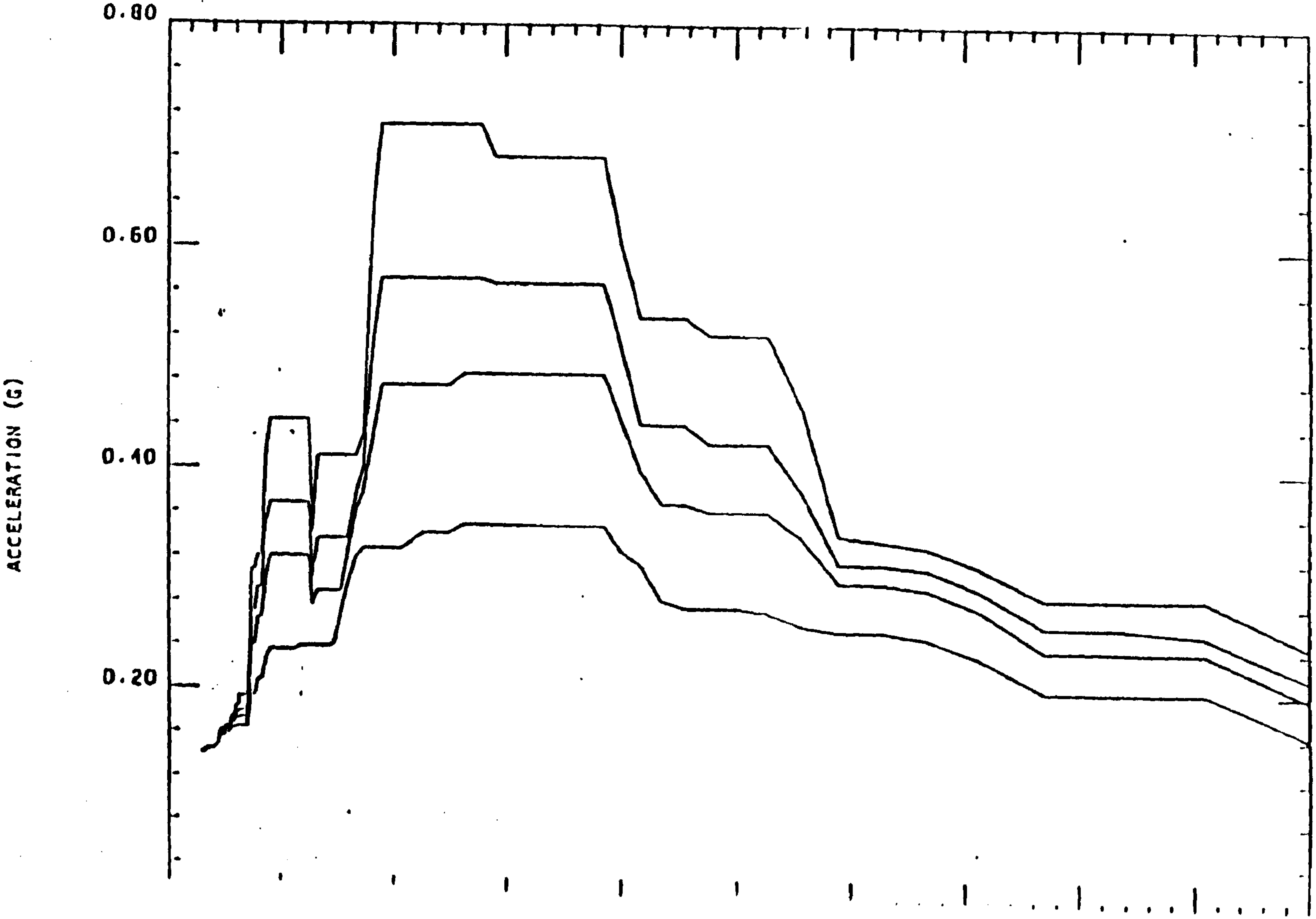



FIGURE 20 VERTICAL FLOOR RESPONSE SPECTRA FOR THE PANEL BETWEEN COLUMN LINES RA-RB AND R6-R7 (2%, 3%, 4% and 7% Damping), El. 23 FT 6 IN


0.80
0.60
0.40
0.20
ACCELERATION (G)

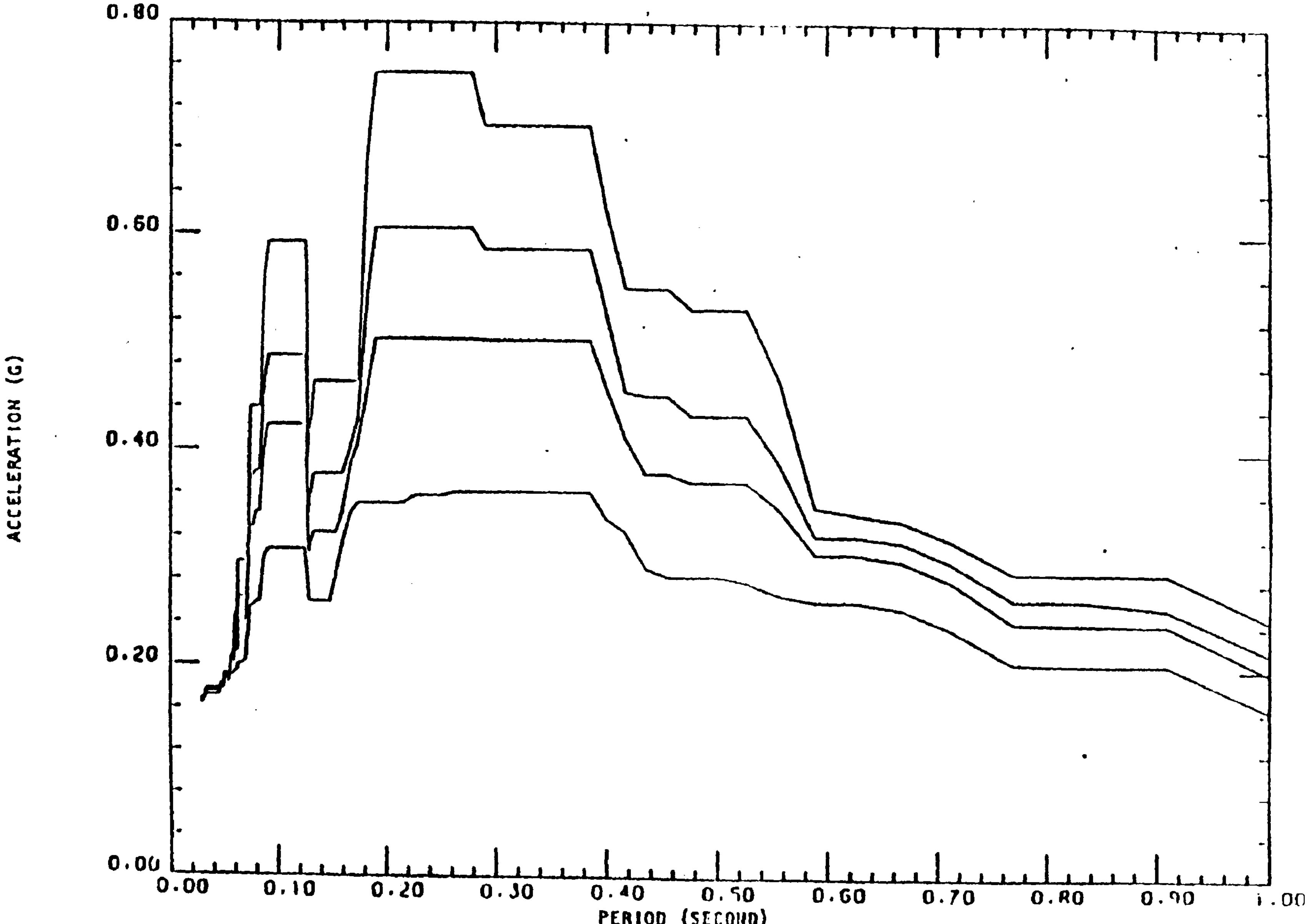



FIGURE 22 VERTICAL FLOOR RESPONSE SPECTRA FOR THE PANEL BETWEEN COLUMN LINES RD-RE
[illegible] 7% Damping) EL. 51 FT 3 IN

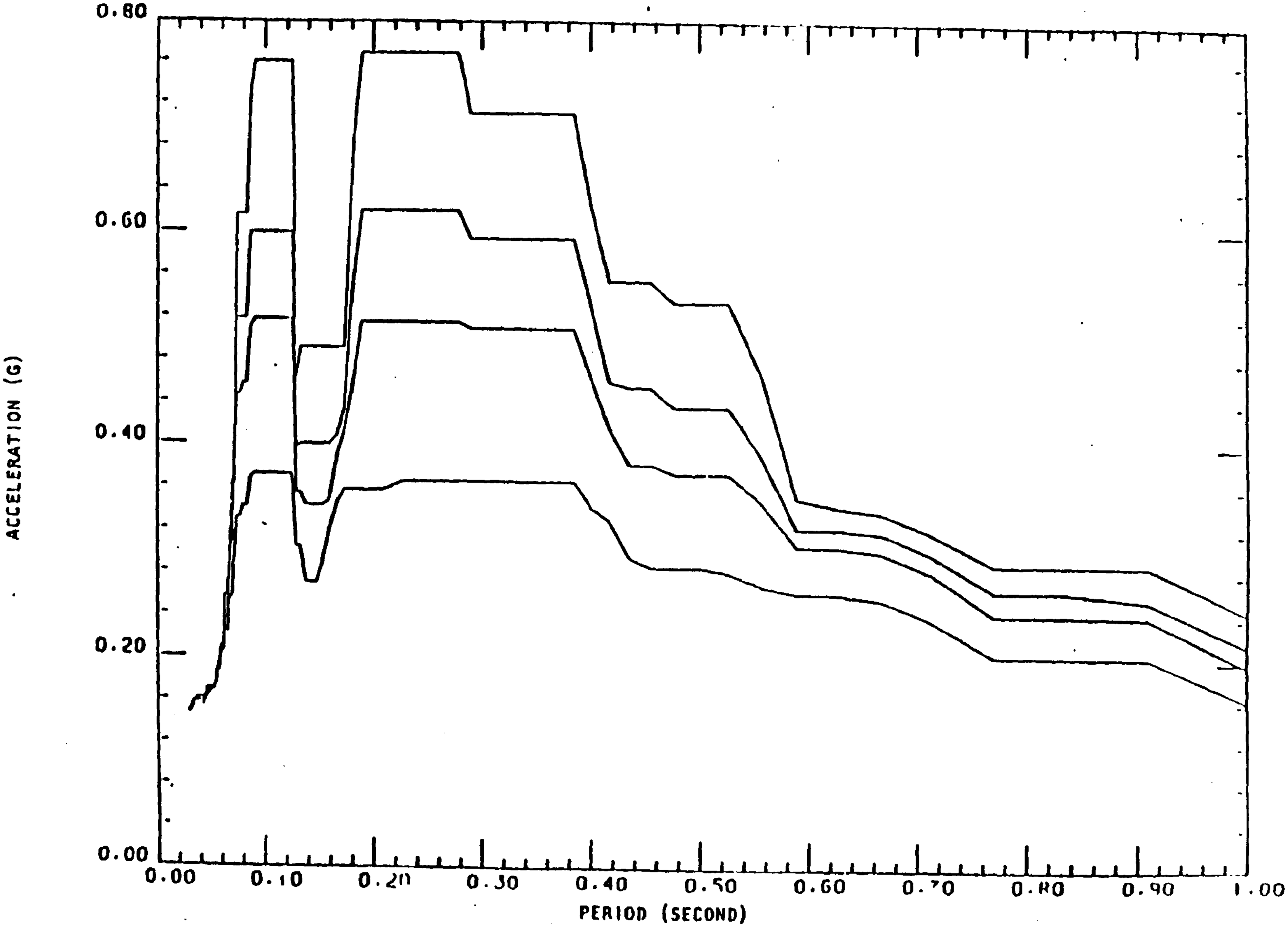
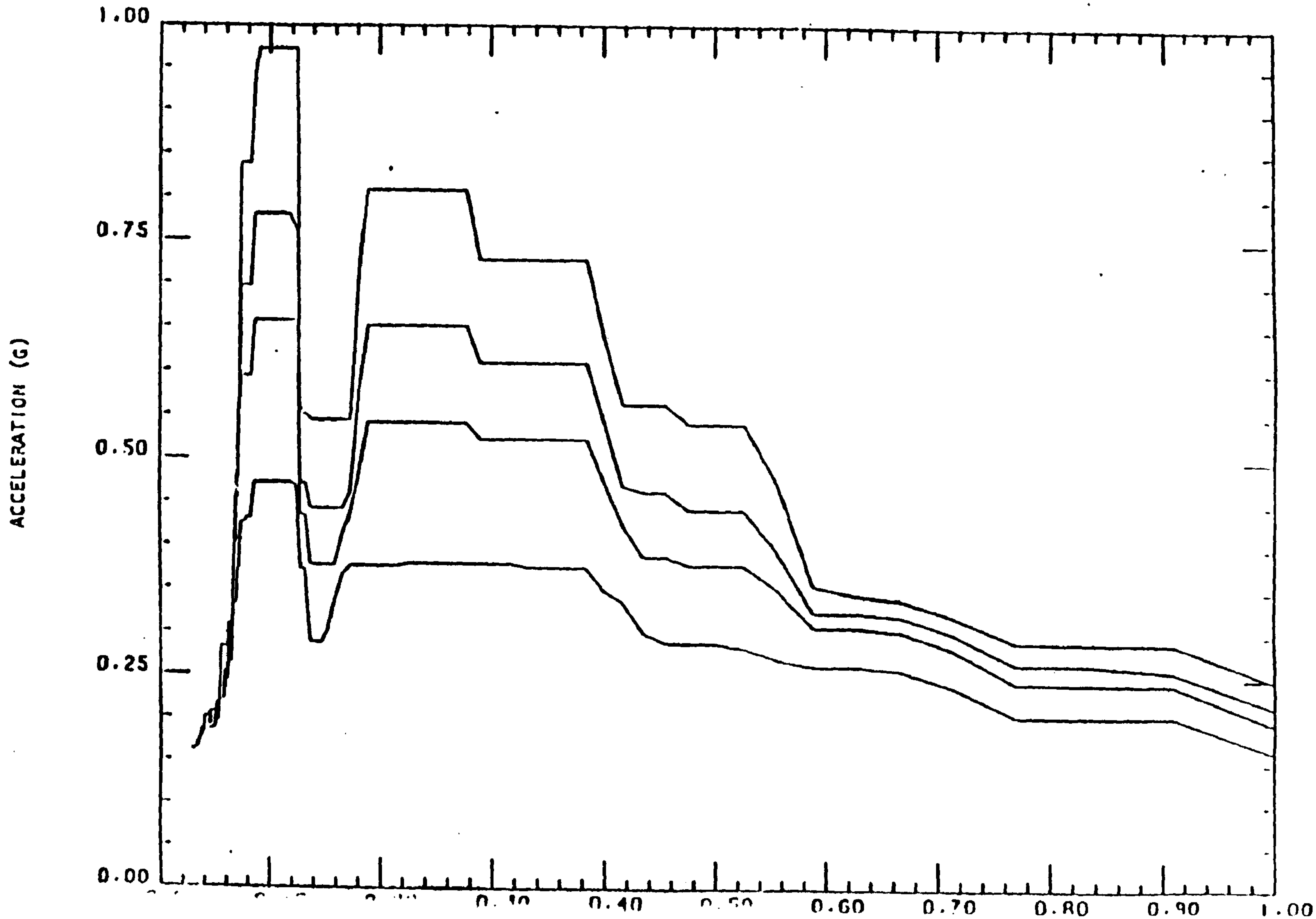
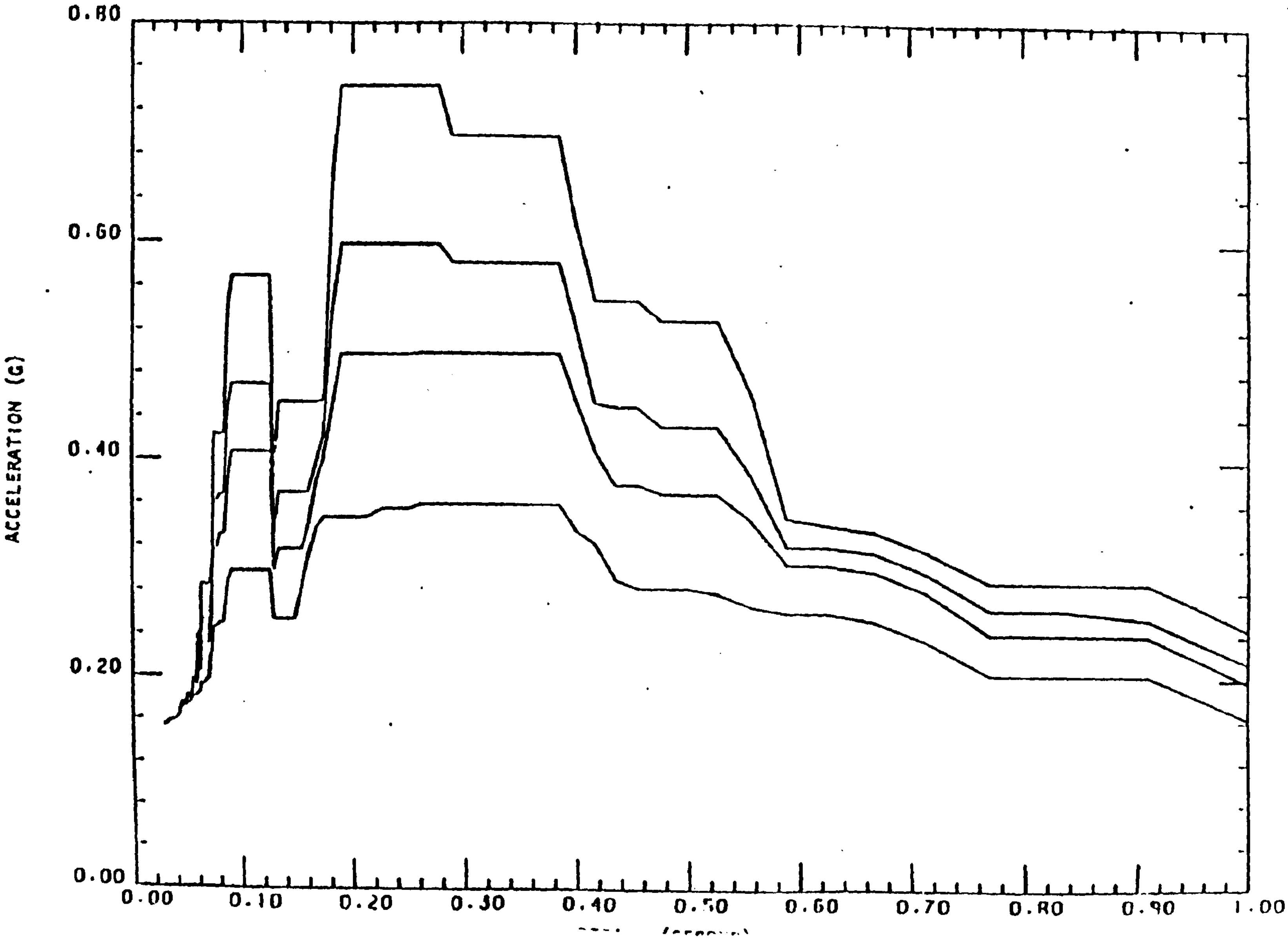
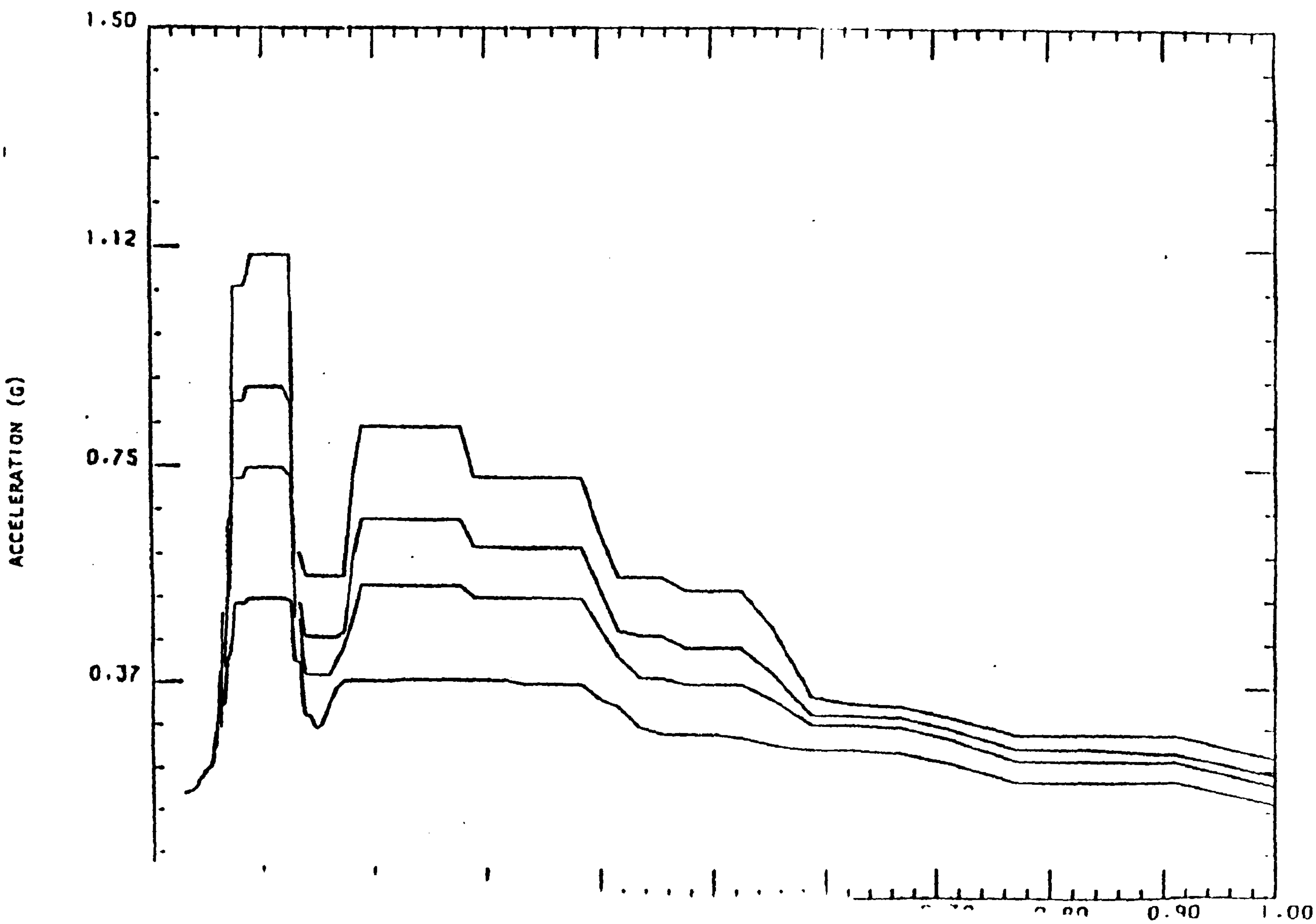



FIGURE 23. VERTICAL FLOOR RESPONSE SPECTRA FOR THE PANEL BETWEEN COLUMN LINES RA-RB


ACCELERATION (G)
1.00
0.75
0.50
0.25
0.00
0.30
0.40
0.60
0.70
0.80
0.90
1.00


ACCELERATION (G)
0.00
0.20
0.40
0.60
0.80
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00


ACCELERATION (G)
1.50
1.12
0.75
0.37
0.90
1.00

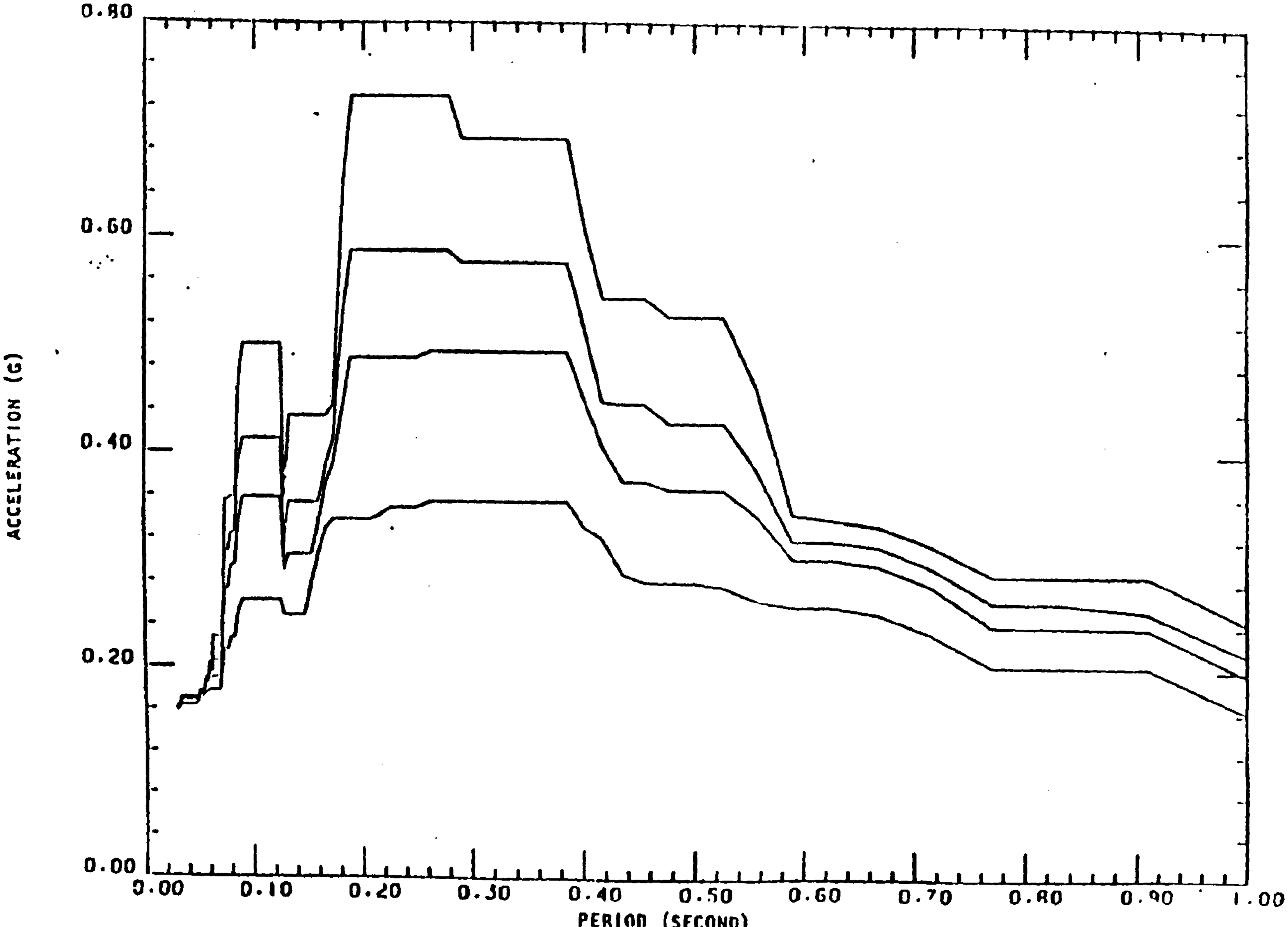
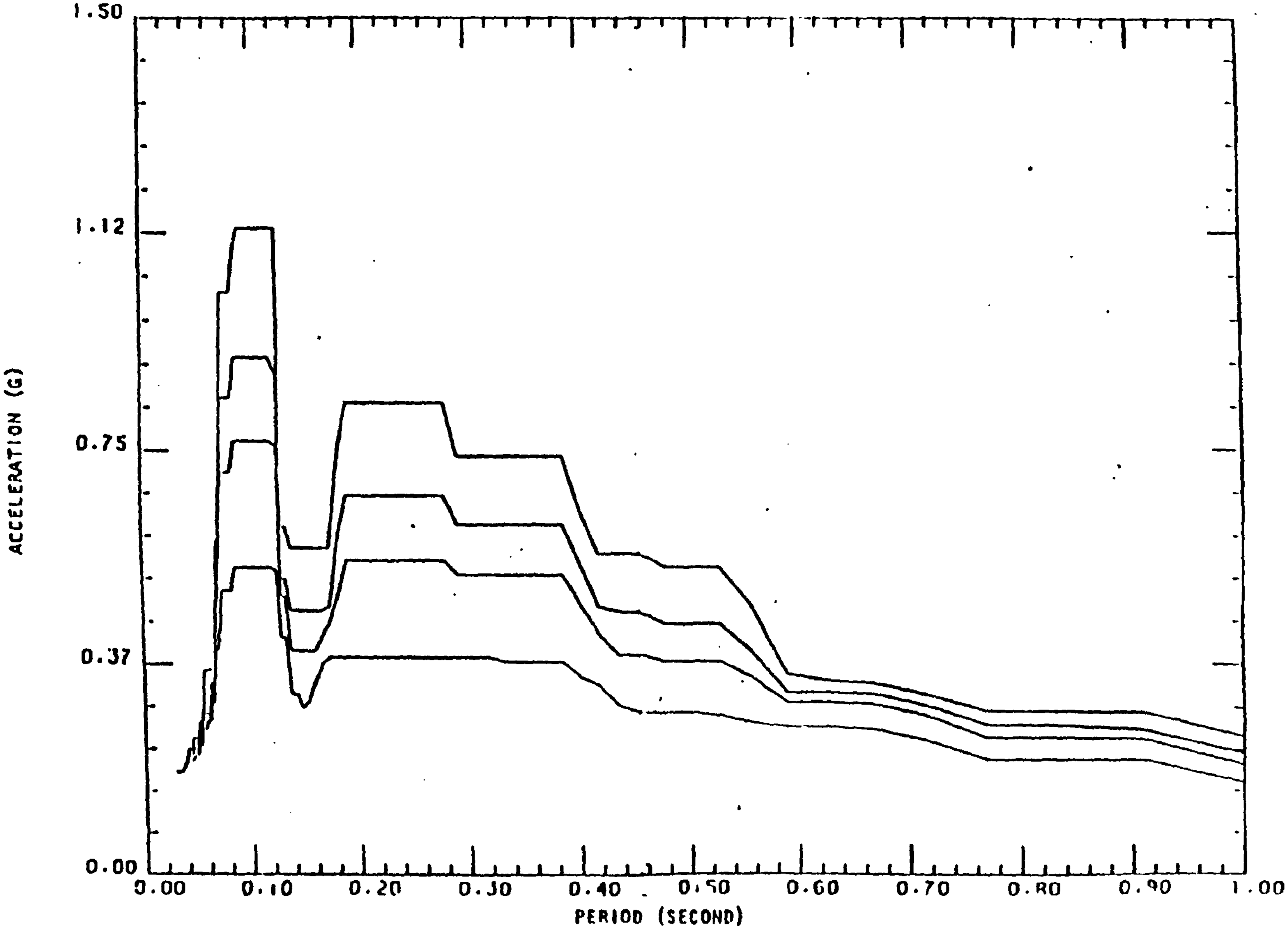
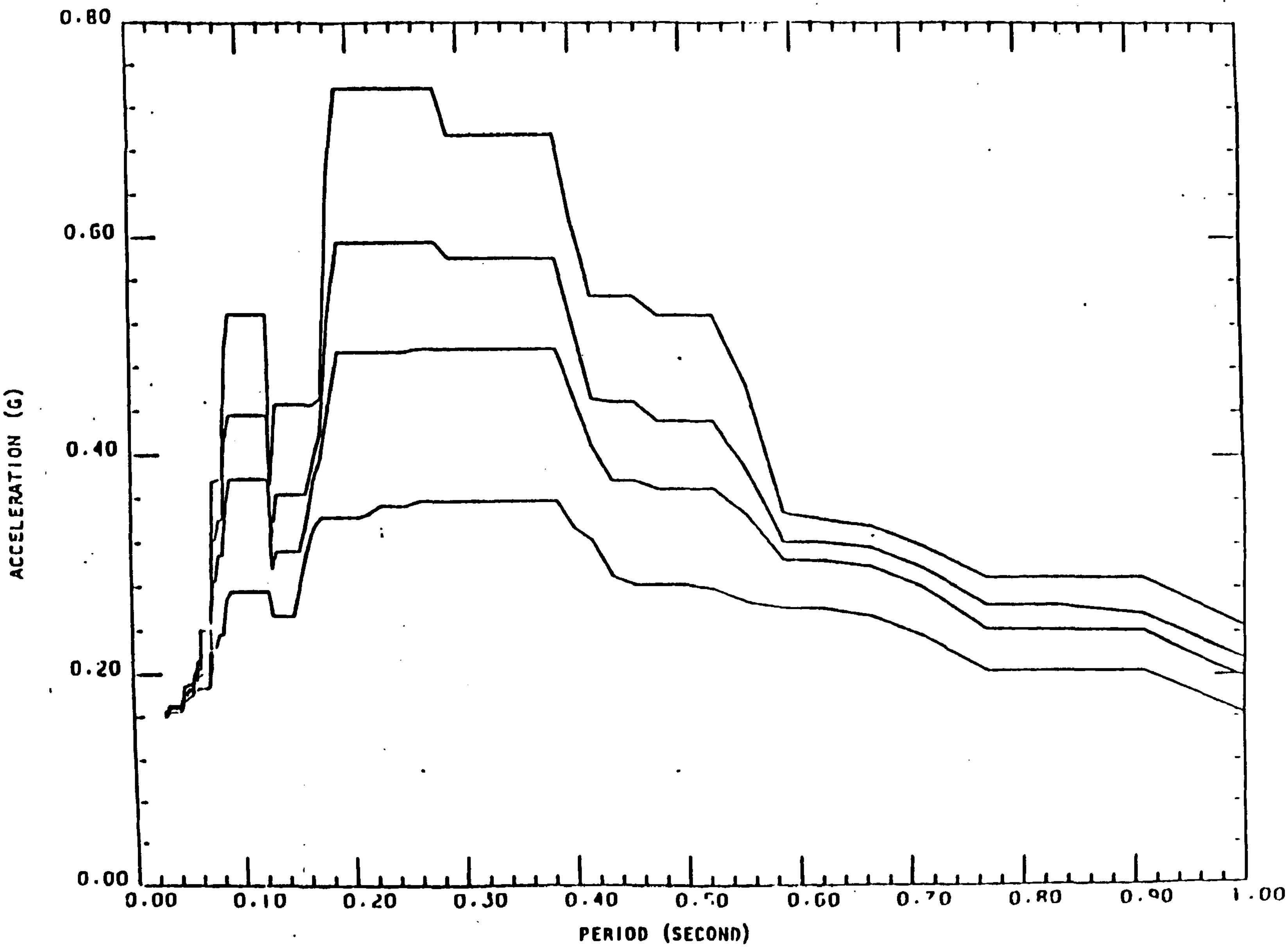



FIGURE 27 VERTICAL FLOOR RESPONSE SPECTRA FOR THE PANEL BETWEEN COLUMN LINES RE-RF AND Rb-RE (2%, 3%, 5% and 7% Damping), EL. 95 FT 3 IN




RESPONSE SPECTRA FOR THE PANEL BETWEEN COLUMN LINES RB-RC


0.80
0.60
0.40
0.20
0.00
ACCELERATION (G)
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
PERIOD (SECOND)

... FOR THE REACTOR BUILDING

...CLEAR POWER PLANT

Criteria

...ia Document

...ared for

... Nuclear

...y, New Jersey

...ust 1981

.. Blume & Associates, Engineers

...ie Street (at New Montgomery)

...rancisco, California 94105

## CONTENTS





bd

1. Introduction

Oyster Creek is a 620-MW boiling water reactor (BWR) plant that went into operation in 1969. It is one of 11 older United States nuclear plants now being reviewed under the Systematic Evaluation Program (SEP) of the U.S. Nuclear Regulatory Commission (NRC).

Oyster Creek Category 1 electrical equipment, and possibly some mechanical equipment and components, are subject to seismic evaluation. Consequently, acceleration floor response spectra are required for proper seismic qualification of category 1 equipment.

This criteria document describes the methods and techniques to be used in the development of floor response spectra for the reactor building at Oyster Creek.

2. Basic Approach

This criteria will be based primarily on the following regulatory standards and documents:

*USNRC Standard Review Plan - Section 3.7.2* (ref. 1)

*USNRC Regulatory Guides* (ref. 2)

*Seismic Review of the Oyster Creek Nuclear Power Plant as Part of the Systematic Evaluation Program, NUREG/CR-1981.* (ref. 3)

*Initial Review and Recommendations for Site-Specific Spectra at SEP Sites.* (ref. 4)

U.S. Nuclear Regulatory Commission, *Development of Criteria for Seismic Review of Selected Nuclear Power Plants, NUREG/CR-0098*, May 1978. (ref. 5)

3. Seismic Input

The free field horizontal seismic input has been described in terms of the response spectrum as shown in Figure 1 (ref. 9) which has been recommended by the NRC for use at the Oyster Creek site. The vertical free field spectrum will be considered to be 2/3 the horizontal throughout the frequency range of interest. The seismic input is also required in terms of time-histories for the computation of floor response spectra as well as for the response time-history analyses.

The free field acceleration time-histories will be developed such that their response spectra will match the input free field spectra described above.

## 4. Material Properties

The following material properties will be used in this project:

Concrete modulus, $E_c = W_c^{1.5}\ 33\sqrt{f_c'}$ (ref. 10) R1

Steel modulus, $E_s$ = 29,000 ksi

Concrete Poisson's Ratio, $\nu_c$ = 0.17

Steel Poisson's Ratio, $\nu_s$ = 0.25

Structural modal damping, $\xi$ = 10% of critical

Soil shear modulus, $G_s$ = 6,000 ksf will be used to calculate structural responses

Soil shear modulus, $G_s$ = 6,000 ksf and 4,000 ksf will be used in calculating floor response spectra. The final floor response spectra will be an envelope of the spectra from the two values of $G_s$ used.

Soil poisson's ratio, $\nu_{soil}$ = 0.475

Soil weight density = 126 lb/cft.

## 5. Analytical Methods

### 5.1 Seismic-Dynamic Model

The fixed-base lumped mass reactor building developed by URS/Blume (ref. 8) will be used. The SSI effects will be simulated using the lumped parameter approach. This approach to SSI is based on developing a set of 'soil springs' which represent the flexibility of the foundation and a set of viscous dampers which represent the 'radiation damping'. The expressions for these frequency-dependent springs and dampers will be taken from references 6 and 7.

Per SSRT recommendations (ref. 3), only 50% of the embedment effects will be considered for SSI.

The response for the vertical component of ground motion will be calculated on the bases of a vertical dynamic model which will use finite elements and/or equivalent beam elements to model the out of plane response of the floors. Supporting elements which are not rigid in the vertical direction will be included in the model. The R1 rigid supporting elements will be modelled by appropriate boundary elements at their point of connection to the floor slabs. Columns, and walls which do not continue to the foundation will be appropriately included in the model.

### 5.2 Seismic Analysis

Seismic analyses of the dynamic models will be undertaken for horizontal and vertical motions. In order to properly consider the frequency dependence of soil impedance (springs and dampers) coefficients, the analysis will be performed in the frequency domain. The maximum story shears, maximum story overturning moments, maximum absolute floor accelerations and maximum relative floor displacements will be obtained from these analyses. The response acceleration time-histories at different mass points will also be generated to develop the floor spectra.

## 6. Floor Response Spectra

Horizontal and vertical floor response spectra will be generated for 2, 3, 4 and 7% of critical damping for all applicable locations. Peaks in the floor response spectra at structural frequencies less then 33 hertz will be broadened by 15% on either side of the peak per Reg. Guide 1.122 (ref. 2) and the rest of the spectra will be smoothed.

## 7. References


1. U.S. Nuclear Regulatory Commission, [illegible] *Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-75/087*, LWR Edition, March 1979.

2. U.S. Nuclear Regulatory Commission, *Regulatory Guide 1.122, Revision 1*, February 1978.

3. *Seismic Review of the Oyster Creek Nuclear Power Plant as Part of the Systematic Evaluation Program, NUREG/CR-[illegible], UCRL-53018, RD, RM*, prepared for U.S. Nuclear Regulatory Commission by Lawrence Livermore Laboratory, Livermore, California, [illegible] 1980.

4. *Initial Review and Recommendations for Site-Specific Spectra at SEP Sites*, Memorandum for D. Crutchfield, [illegible] Chief, Systematic Evaluation Program Branch, from R[illegible] Jackson, Chief, Geosciences Branch, DE, U.S. Nuclear Regulatory Commission, Washington, D.C., June 23, 1980.

5. U.S. Nuclear Regulatory Commission *Development of Criteria for Seismic Review of Selected Nuclear Power Plants, NUREG/CR-0098*, May 1978.

6. Kausel, E., R.V. Whitman, J.P. Mo[illegible] E. Elsabee, *The Spring Method for Embedded Foundations*, published [illegible] Nuclear Engineering Design 48, pp. 337-392, 1978.

7. Kausel, E., R. Ushijima, *Vertical and Torsional Stiffness of Cylindrical Footings*, Publication No. R79-6, Department of Civil Engineering, MIT, February 1979.

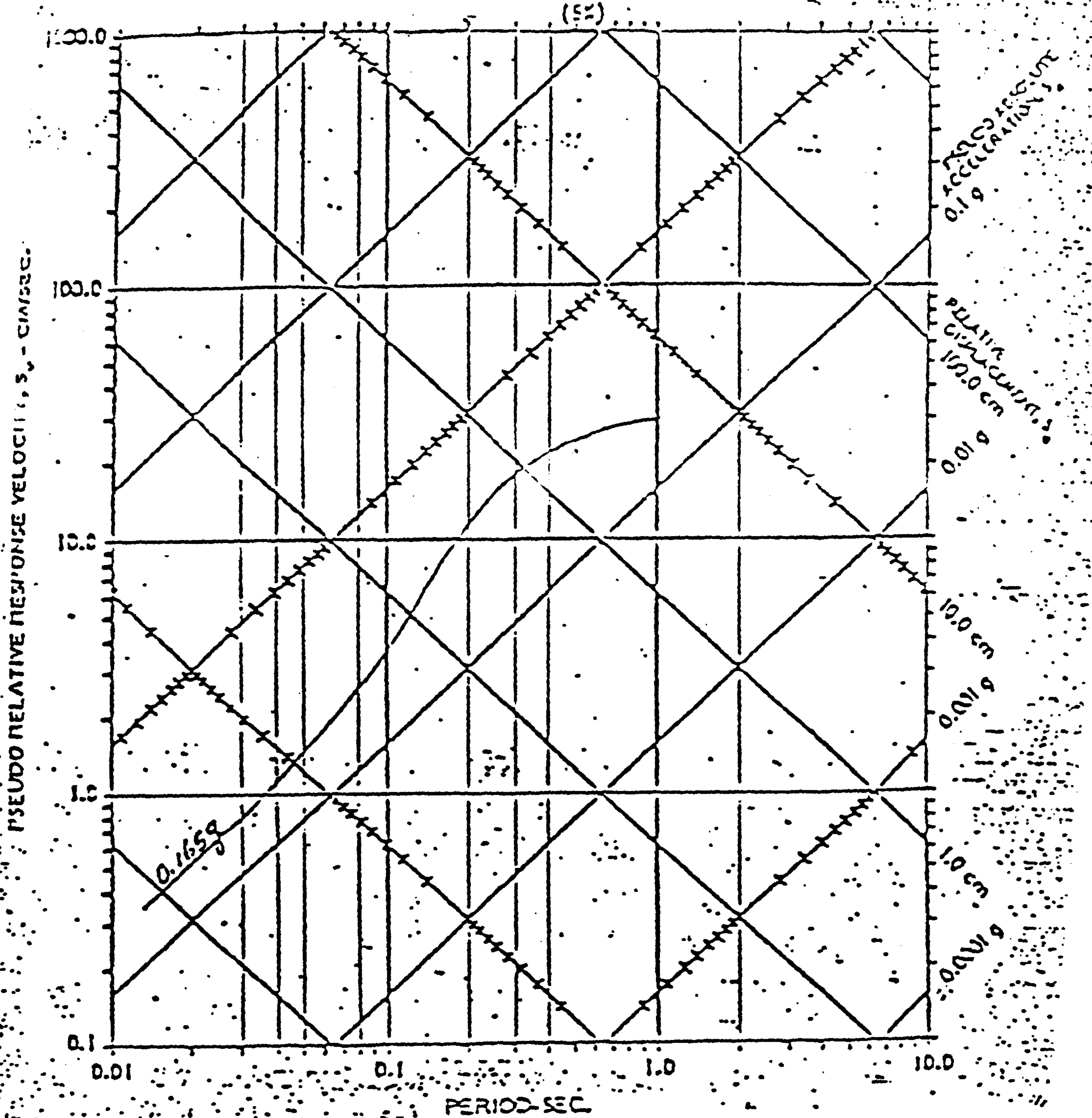
8. *Jersey Central Nuclear Reactor Project - Earthquake Analysis: Reactor Building*, report prepared for General Electric Company by John A. Blume & Associates, Engineers, San Francisco, California, June 1965.

9. GPU Service Specification 1302-43-[illegible], Revision 1, dated May 4, 1981.

10. American Concrete Institute, ACI Standard 318-77, December 1977.

FIG. 1

Site Specific Spectrum




Oyster Creek Site
(5% Damping)

## APPENDIX B

List of Computer Codes

## List of Computer Codes

The computer programs used in this project by URS/Blume are:

1. SAP4 Version N&E 4.2
2. SAP4 Version N&E 4.2B - Project Specific
3. JAB/FLSPEC Version 1.0
4. SMSPC3 Version 1.0
5. FREDA Version 0.0 - Project Specific

The above programs have been fully verified in accordance with requirements of the URS/Blume Quality Assurance Program.

In addition the following four subroutines have been used:

1. INTERP
2. MODE
3. ENVEL
4. JMT

These have been verified in accordance with URS/Blume Project Specific Computer Program QA requirements.

# APPENDIX C

Letter Regarding Referenced Documents

URS

AN INTERNATIONAL PROFESSIONAL SERVICES ORGANIZATION

URS/JOHN A. BLUME & ASSOCIATES, ENGINEERS
130 JESSIE STREET (AT NEW MONTGOMERY)
SAN FRANCISCO, CALIFORNIA 94105
TEL: (415) 397-2525
CABLE. BLUMENGRS

NEW YORK
SAN FRANCISCO
WASHINGTON D.C.
DALLAS
SEATTLE
DENVER
KANSAS CITY
HONOLULU
NEW ORLEANS
SAN MATEO
HONG KONG
BOLIVIA

February 1, 1982

Leon Garibian
General Public Utility Service Corp.
100 Interpace Parkway
Parsippany, New Jersey 07054

Subject: Oyster Creek Reactor Building Floor Response Spectra

References: 1. URS/Blume draft report entitled: *Seismic Acceleration Floor Response Spectra for the Reactor Building at Oyster Creek Nuclear Power Plant*

2. GPU Specification entitled: *Oyster Creek Nuclear Station, Reactor Building Structure, Generation of Seismic Floor Response Spectra (Specification No. 1302-43-001)*

Dear Leon:

In response to your telephone inquiry, this letter is to confirm that our work to generate the floor response spectra and other building responses presented in the referenced URS/Blume report has complied with all the relevent references contained in the GPU referenced specifications.

Should you have any further questions in this regard, please do not hesitate to contact Dr. Lincoln Malik or myself.

Very truly yours,

Ahmad F. Kabir

Ahmad F. Kabir
Deputy Manager
Structures Department
Nuclear & Energy Division

cav



km

# APPENDIX 3.7B

## SITE-SPECIFIC RESPONSE SPECTRA

URS

# SITE-SPECIFIC RESPONSE SPECTRA FOR THE OYSTER CREEK NUCLEAR POWER PLANT

DECEMBER 1979

prepared for

Jersey Central Power and Light Company
Morristown, New Jersey

prepared by

URS/John A. Blume & Associates, Engineers
130 Jessie Street (at New Montgomery)
San Francisco, California 94105

# SITE-SPECIFIC RESPONSE SPECTRA
# FOR THE OYSTER CREEK NUCLEAR POWER PLANT

prepared for

Jersey Central Power and Light Company
Morristown, New Jersey

December 1979

URS/John A. Blume & Associates, Engineers
130 Jessie Street (at New Montgomery)
San Francisco, California 94105

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TABLES



FIGURES

# INTRODUCTION

This report describes work performed by URS/John A. Blume & Associates, Engineers (URS/Blume), in developing a site-specific response spectrum for the Oyster Creek Nuclear Power Plant. The report is prepared for the Jersey Central Power & Light Company of Morristown, New Jersey. The spectrum is intended for use in engineering design review and analysis of existing facilities at the Oyster Creek plant.

At this time, the use of site-specific response spectra is encouraged by the U.S. Nuclear Regulatory Commission (NRC) staff. This is because of improved analytical methods and the increased number of strong motion earthquake records available for use in analysis.

The methodology used in the past to establish seismic criteria for most nuclear power plants in the eastern United States utilized the generic response spectra presented in NRC Regulatory Guide 1.60. This spectrum is anchored at a zero-period peak ground acceleration based upon the Modified Mercalli Intensity (MMI) of the Safe Shutdown Earthquake (SSE). The appropriate level of peak ground acceleration for an earthquake of a given intensity is determined from published studies of strong motion earthquake data that relate intensity to peak ground acceleration.

The problems associated with this method of developing response spectra, which have been widely discussed, are generally recognized to be significant. Some of the major problems cited are the highly subjective nature of earthquake intensity ratings, the large scattering of data in the intensity-acceleration relationship, and the conservatism of the NRC Regulatory Guide 1.60 spectra, which are based on ground motion records of earthquakes of varying magnitudes and epicentral distances in different geologic environments.

These problems are alleviated to a large extent by the technique for developing a site-specific response spectrum that is used in this study. This technique is based on selection of a suite of strong motion earthquake records that characterize both the SSE (i.e., magnitude and epicentral distance) and the geologic conditions of the power plant site.

This study begins with determination of the magnitude appropriate to the SSE maximum MMI of VII developed in accordance with regulatory procedures prior to design of the Oyster Creek Nuclear Power Plant. Since this is the same intensity that would be arrived at following the current regulations of 10 CFR 100, Appendix A, the study does not review this development.

The following sections of this report explain the methodology utilized in developing the site-specific response spectra for the Oyster Creek plant; they describe the calculations, provide the input parameters utilized, including the suite of strong motion earthquake records, and present results of the calculations and the site-specific response spectra.

## SUMMARY

In the mean value site-specific response spectrum developed for the Oyster Creek Nuclear Power Plant site, shown on Figure 1, the zero-period peak ground acceleration is 0.07g.

The basic input parameters used to develop this spectrum are as follows. The SSE is an event of Modified Mercalli Intensity VII immediately adjacent to the site. The corresponding magnitude ($m_{bLg}$) is 5.3. This site is underlain by moderately deep alluvium. The characteristic epicentral distance is 5 km. The earthquake mechanism is considered to be characterized by normal faulting.

METHODOLOGY

A computer program based on a new way of predicting the maximum response of single-degree-of-freedom systems to which the response spectra shape can be anchored was implemented to predict ground motion spectra at specific sites in the eastern United States.

The methodology employed is an application of the random vibration theory (Vanmarcke, 1976) with the data needed for the solution based on empirical work (Street and Turcotte, 1977) and engineering judgment. The anchor point of the response spectrum at period $T$ = 1 sec and the maximum expected ground acceleration for rock is obtained for a specified spectral moment ($\tau_{DS}$), a given epicentral distance, a strong motion duration, and a description of the spectrum within the period range of interest. The desired spectral shape can also be scaled with the corresponding transfer function so that the soil characteristics of the local site are included.

## Response Spectra

The maximum response is of practical value in the analysis and design of elastic structures. For a simple one-degree-of-freedom mathematical model of a structure with a natural period $T$ and a damping ratio $\beta$, a plot of the peak response to a given excitation as a function of the period is known as the response spectrum.

It is useful to present the displacement, velocity, and acceleration spectra in one plot, as in the tripartite response spectra. Here the ideal characteristics of a double-degree-of-freedom oscillator are exploited so that relative displacement ($S_D$) and pseudo-velocity ($S_v = W_n S_D$) are both presented.

## Random Vibration Method

Because of our limited knowledge of the physical process that produces seismic motion and the almost total absence of strong motion data for the eastern United States, a stochastic description of the earthquake process becomes inevitable. Such a description is very useful in predicting the occurrence of events because of the availability of the analytical tools developed in the theory of random processes.

When the theory of random process is applied to a time history ($x(t)$), the whole ensemble of possible time histories which might have occurred as a result of the phenomena under consideration, not just one time history, are described. An individual time history belonging to the ensemble is called a realization.

A major simplification is normally used, in that the random process is assumed to be invariant with time in its statistical description. In this case, the stochastic or random process is called a stationary stochastic process, and we can write the autocorrelation function ($E[X(t_1)X(t_2)]$), which is the expected value of the product of the value of $x(t)$ at two different times $t_1$ and $t_2$, as a function of the difference $T = t_2 - t_1$ : $E[X(t_1)X(t_2) = R_x(T)]$, where $R_x(T)$ is the autocorrelation function of the stationary stochastic process $X(t)$.

The spectral function, $S_x(\omega)$, can be defined as the Fourier pair of $R_x(t)$, so that

$$R_x(t) = \int_{-\infty}^{\infty} S_x(\omega)e^{i\omega t}d\omega \tag{1}$$

and

$$S_x(\omega) = \frac{1}{2\pi}\int_{-\infty}^{\infty} R_x(t)e^{-i\omega t}dt \tag{2}$$

Both the autocorrelation and the spectral functions are a complete representation of the stationary stochastic process $x(t)$

Empirically derived relationships that relate the observed vertical displacement spectrum $\Omega_r(\omega)$ with the source spectral function $S^*(\omega)$ at a fixed distance are available. Among them, the one found in Street and Turcotte (1977) is applicable for the eastern United States. It is:

$$S^*(\omega) = \begin{cases} 4\pi\rho\beta^3 r_o(r/r_o)\ \Omega_r(\omega), \text{ when } r \le r \\ 4\pi\rho\beta^3 r_o(r/r_o)^{1/2}\ \Omega_r(\omega), \text{ when } r > r \end{cases} \tag{3}$$

where $\rho$ = soil density, taken as 2.5 gm/cm$^{-3}$; $\beta$ = damping, taken as 3.5 km/sec$^{-1}$; $r_o$ = a fixed distance parameter = 100 km; $S^*(\omega)$ = source spectrum at $r_o$ distance; and $\Omega_r(\omega)$ = observed vertical displacement spectrum.

from the same reference it can be observed that the frequencies around $T$ = 1 sec, ($\omega = 2\pi$), are well behaved. Also, the following relation between observed $m_{bLg}$ and $S^*(T = 1\ \text{sec})$ has been derived:

$$S^*(T = 1\ \text{sec}) = 10^{(17.5 + m_{bLg})} \tag{4}$$

It is easy to obtain from Equation 3 the site specific vertical displacement spectral level $\Omega_r(T = 1\ \text{sec})$ for any specified distance $r$. This site vertical acceleration amplitude spectrum $A_r(T = 1\ \text{sec})$ is readily calculated by:

$$A_r(T = 1\ \text{sec}) = \Omega_r(T = 1\ \text{sec}) \cdot (2\pi)^2 \tag{5}$$

The development up to this point has been for vertical motion (used in obtaining $m_{bLg}$). This can be transformed to horizontal acceleration amplitude by multiplying by the ratio of horizontal to vertical acceleration. In this case a mean value of 2.4 was obtained for this ratio from a study of 70 strong motion records of eastern United States earthquakes.

The power spectral density function (one-sided) of the ground motion $G_r(\omega)$ can be estimated by smoothing the Fourier acceleration spectrum,

$$G_r(T = 1\ \text{sec}) = \frac{1}{\pi s}|A_r(T = 1\ \text{sec})|^2 \tag{6}$$

where $s$ denotes strong-motion duration. As the assumption of stationarity implies a uniform energy distribution in time, the duration $s$ must be adjusted so that an equivalent duration, called $s_o$, and the averaged $G_r(\omega)$ give the total energy content of the earthquake. Here the definition given in Vanmarcke and Lai (1977) will be used to that effect.

We need the predicted response spectrum ordinate at period $T$ = 1 sec to anchor the selected shape of the normalized response spectra. A nonstationary random vibration analysis following that of Vanmarcke and Lai (1977) is done to accomplish this task: Given $G_r(\omega)$ of a single-degree-of-freedom system and the strong motion (equivalent) duration $s_o$, the pseudo-velocity response spectra $S_V(T,B)$ is predicted with the general form

$$S_V(T,B) = a_p \sigma_v(s_o) \tag{7}$$

in which $\sigma_v(s_o)$ = time-dependent standard deviation of the pseudo-velocity response, evaluated at $t = s_o$; $\alpha_p$ is a peak factor function of the probability of nonexceedance, $p$.

The variance $\sigma_v^2(t)$ is obtained by integrating the spectral density function over all frequencies,

$$\sigma_v^2(t) = \int_{-\infty}^{\infty} G(\omega,t)d\omega \tag{8}$$

An approximate solution for moderate natural frequencies, including $T$ = 1 sec ($\omega = 2\pi$) and for relatively large damping values (such as $\beta$ = 0.05) is, for $T$ = 1 sec:

$$\sigma_v^2(s_o) = \left[\frac{G_r(T = 1 \text{ sec})}{8\beta}\right]^{1/2} \tag{9}$$

The peak factor is the product of an analysis of the maximum. Exact solutions for $\alpha_p$ do not exist, but an approximate solution that has been extensively checked is, for $T$ = 1 sec:

$$\alpha_p = [2\, ln\, (-2S_o/ln\, p)]^{1/2} \tag{10}$$

From Equations 7, 9, and 10, we have

$$S_v(T = 1 \text{ sec}, \beta) = \left[\frac{G_r(T = 1 \text{ sec})\, ln\, (-2s_o/ln\, p)}{4\beta}\right]^{1/2} \tag{11}$$

Another value that can be approximated from these data is the peak ground acceleration, $a_g$. This is done empirically; from a set of strong motion data the ratio

$$k = \frac{a_g}{S_v(T = 1 \text{ sec}, 0.1)} \tag{12}$$

was estimated. For $a_g$ in cm/sec$^2$ and $s_v$ in cm/sec the mean value was $k$ = 10.46.

Input Parameter for Determining Peak Ground Acceleration

In order to determine the peak ground acceleration, certain parameters are needed as input. The primary input is the maximum magnitude expected at the

site. The magnitude scale used is $m_{bLg}$ developed by Nuttli (1973) and is most appropriate for the East Coast. Since the maximum historic event associated with the Oyster Creek site is known in terms of epicentral intensity (MMI VII), a relation is needed between epicentral intensity and $m_{bLg}$. Using relations developed by Nuttli and Zollwig (1974) and Street and Turcotte (1977) relating intensity to $m_b$ (body wave magnitude) and $m_b$ to $m_{bLg}$ (magnitude determined from the higher mode $L_g$ wave), the following relation between $m_{bLg}$ and intensity was derived:

$$m_{bLg} = .485\, I_o + 1.92 \tag{13}$$

Using the maximum MMI determined for the Oyster Creek site of VII one obtains an $m_{bLg}$ value of 5.3.

The next parameter needed is the distance from the source to the site. A distance of 5 km was chosen to approximate the condition in which the earthquake occurs under the site. This distance would represent an average focal depth for the site and eliminates the problem of having the acceleration achieve unrealistically high values as the source-site distance approaches zero.

The duration of strong shaking was estimated to be approximately three to five seconds (Bolt, 1973). These values are also consistent with estimates of duration measured for the strong motion records used to develop the site response spectra. The last parameter to input is the damping value, which for characterization purposes was put at 5%.

Using the parameters of $m_{bLg}$ = 5.3, distance = 5 km, duration = 3.0 sec, and damping = 0.05, a calculated ground acceleration of 0.072g was obtained for the site. This value of acceleration was then used to anchor the site specific design spectrum derived for Oyster Creek. In order to see what a slight variation of the parameters would do to the ground acceleration, a preliminary sensitivity study was performed, the results of which are presented in Table 1.

TABLE 1

PEAK GROUND ACCELERATION (g) FOR VARIOUS $m_{bLg}$, DISTANCE, AND DURATION OF STRONG MOTION

| Distance (km) | | | 5 | | 10 | | 15 | |
|---|---|---|---|---|---|---|---|---|
| Duration (sec) | | | 3.0 | 5.0 | 3.0 | 5.0 | 3.0 | 5.0 |
| $m_{bLg}$ | 5.1 | mean | .045g | .039g | .023g | .020g | .015g | .013g |
| | | mean + 1σ | .057g | .048g | .029g | .024g | .019g | .016g |
| | 5.3 | mean | .072g | .062g | .036g | .031g | .024g | .021g |
| | | mean + 1σ | .091g | .075g | .045g | .038g | .030g | .025g |
| | 5.5 | mean | .114g | .098g | .057g | .049g | .039g | .033g |
| | | mean + 1σ | .144g | .119g | .072g | .060g | .048g | .040g |

Site Specific Design Spectra

A total of 34 strong motion records were used to develop site specific response spectra. Twenty records were from the western United States and fourteen from the 1976 Friuli, Italy, earthquake. The accelerograms were chosen to satisfy certain criteria. The records were all recorded at soil sites and at distances of less than 54 km. The earthquakes ranged in magnitude from 4.7 to 6.1. The peak accelerations varied from .027g to .308g, with a mean peak acceleration of .115g and standard deviation of ±.073 (see Table 2). The spectra were all normalized to the zero-period acceleration and then statistically combined (assuming lognormal distribution) to obtain a mean response spectrum. The spectrum was then anchored to the peak ground acceleration value determined by the above method. It is shown in Figure 1. The spectra for 0, 2, 5, and 10% damping are shown together in Figure 2.

# TABLE 2

## UHS DATA SET - OYSTER CREEK

| I.D. No | Earthquake Location | Date/Time | Accelerometer Station | $M_L$ | Epicentral distance (km) | Component & pga (g) |
|---|---|---|---|---|---|---|
| A010 | San Jose, Ca. | 9-4-55 | Bank of America, San Jose | 5.8 | 10 | N31W: .102<br>N59E: .108 |
| A013 | San Francisco, Ca. | 3-22-57 | Southern Pacific Bldg., San Francisco | 5.3 | 17 | N45E: .047<br>N45W: .046 |
| A018 | Central California | 4-8-61 | Hollister City Hall | 5.6 | 21 | S01W: .065<br>N89W: .178 |
| B023 | Long Beach, Ca. | 10-2-33 | Hollywood Storage Bldg., basement, Los Angeles | 5.4 | 38 | N90W: .027<br>N .033 |
| B031 | Wheeler Ridge, Ca. | 1-12-54 | Taft Lincoln School Tunnel | 5.9 | 54 | N21E: .065<br>S69E: .068 |
| U305 | Central California | 4-25-54 | Hollister Public Library | 5.3 | 36 | S01W: .053<br>N89W: .050 |
| V316 | Torrance-Gardena, Ca. | 11-14-41 | Long Beach Public Utilities Bldg., Los Angeles | 5.4 | 6 | N: .040<br>E: .055 |
| V329 | Southern California | 3-18-57 | Pt. Hueneme Research Lab. | 4.7 | 5 | S: .167<br>W: .089 |
| V335 | Lytle Creek, Ca. | 9-12-70 | Hall of Records, San Bernardino | 5.4 | 30 | N: .116<br>E: .059 |
| W339 | - | - | Southern California Edison, Colton | 5.4 | 34 | S: .062<br>E: .059 |
| FCC55 | Friuli, Italy | 5-11-76/ 224402 | Forgaria-Cornino | 5.3 | 10 | NS: .190<br>EW: .308 |
| FC131 | - | 9-11-76/ 163132 | Forgaria-Cornino | 5.5 | 16 | NS: .095<br>EW: .115 |
| TAR 333 | - | 9-11-76/ 163132 | Tarcento | 5.5 | 8 | NS: .204<br>EW: .105 |
| FC138 | - | 9-11-76/ 163500 | Forgaria-Cornino | 5.9 | 15 | NS: .333<br>EW: .235 |
| B143 | - | 9-11-76/ 163500 | Buia | 5.9 | 14 | NS: .233<br>EW: .108 |
| FC152 | - | 9-15-76/ 031519 | Forgaria-Cornino | 6.1 | 10 | NS: .263<br>EW: .218 |
| B156 | - | 9-15-76/ 031519 | Buia | 6.1 | 6 | NS: .110<br>EW: .096 |

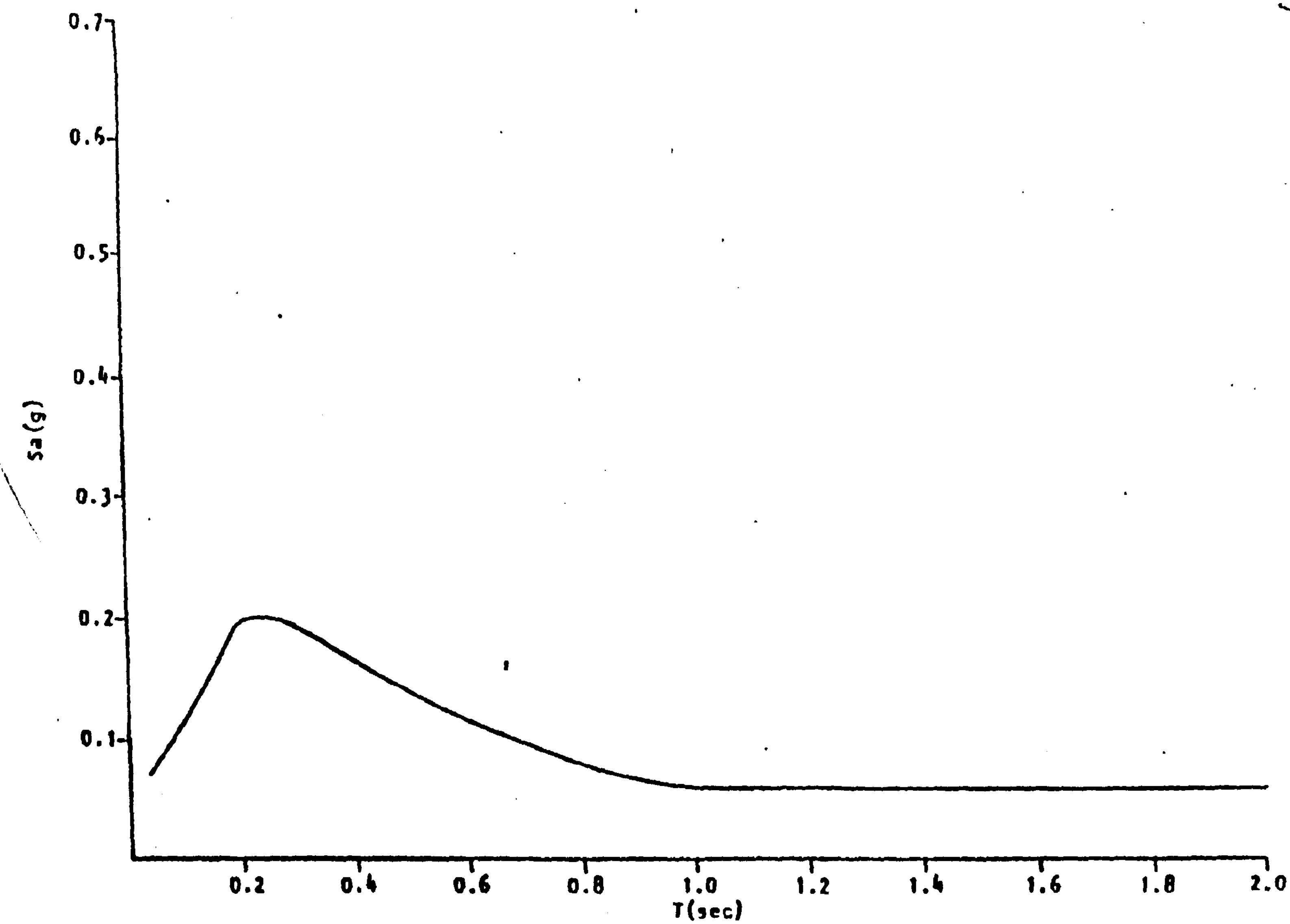


FIGURE 1 OYSTER CREEK MEAN HORIZONTAL SPECTRA FOR 5% DAMPING

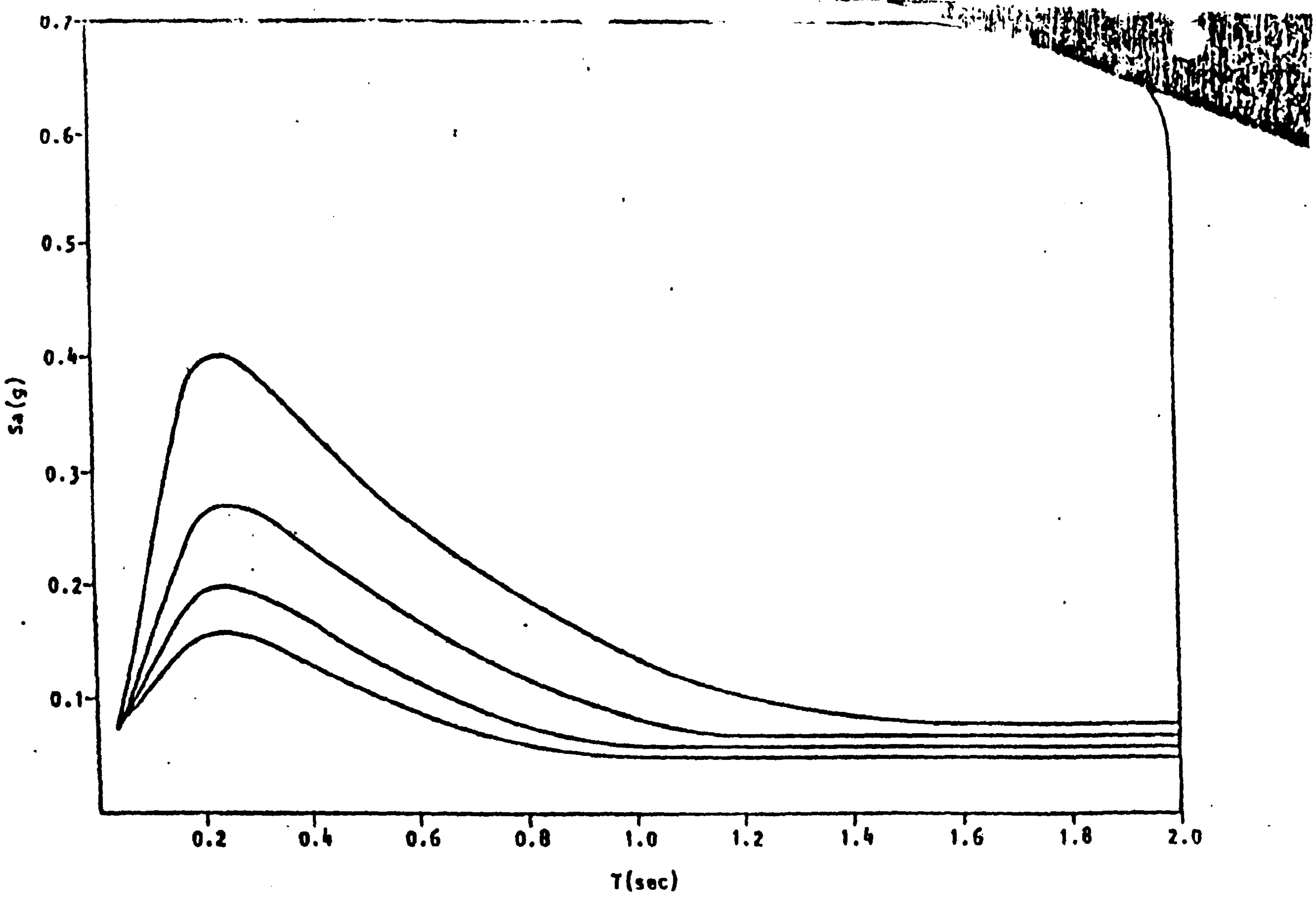


FIGURE 2 OYSTER CREEK MEAN HORIZONTAL SPECTRA FOR 0, 2, 5, 10% DAMPING

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APPENDIX 3.7C

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Note

Dynamic Design of Suppression Chamber
Suction Header is governed by the Mark-I
Containment Long Term Program described
in Section 3.8.2 of the UFSAR.

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