TABLE 3.5-1

CONTROL ROD DRIVE MECHANISM HOUSING PLUG

MISSILE CHARACTERISTICS

Plug Weight: 11 pounds

Plug O.D.: 2.75 inches

Travel, ft	Velocity, ft/sec	Kinetic Energy, lb-ft
1	240	9,750
2	335	19,000
3	370	23,300
4	415	29,200
5	440	33,000

TABLE 3.5-2

CONTROL ROD DRIVE SHAFT - MISSILE CHARACTERISTICS

Diameter = 1.75 inches

Length = 300 inches

Weight = 120 pounds

Drive Shaft Travel Outside Housing*, ft	Drive Shaft Velocity, ft/sec	Drive Shaft Kinetic Energy, lb-ft
1	151	42,900
2	162	49,000
3	171	55,000
4	179	60,200
5	189	66,500

* Distance from top of rod travel housing to bottom of missile shield.

TABLE 3.5-3

CONTROL ROD DRIVE SHAFT AND MECHANISM - MISSILE CHARACTERISTICS

Missile Weight: 1500 pounds

Impact O.D.: 3.75 inches

Travel, ft	Velocity, ft/sec	Kinetic Energy, lb-ft
1	240	9,750
2	335	19,000
3	370	23,300
4	415	29,200
5	440	33,000

TABLE 3.5-4

VALVE - MISSILE CHARACTERISTICS

Missile Description	Weight, lb	Flow Discharge Area, in.^{2}	Thrust Area, in.^{2}	Impact Area, in.^{2}	Wt. to Imp. Area Ratio	Velocity FPS
Safety Relief Valve Bonnet, (3 in. x 6 in. x 6 in.)	350	2.86	80	24	14.6	110
3 in. Motor Operated Isolation Valve Bonnet (plus motor and stem) (3 in.)	400	5.5	113	28	14.1	135
2 in. Air Operated Relief Valve Bonnet (plus stem)	75	1.8	20	20	3.75	115
3 in. Air Operated Spray Valve Bonnet (plus stem)	120	5.5	50	50	2.4	190
4 in. Air Operated Spray Valve Bonnet	200	9.3	50	50	4.0	190

TABLE 3.5-5

PIPING TEMPERATURE ELEMENT ASSEMBLY - MISSILE CHARACTERISTICS

1. For a tear around the weld between the boss and the pipe:

Characteristics	"without well"	"with well"
Flow Discharge Area, in.^{2}	0.11	0.60
Thrust Area, in.^{2}	7.1	9.6
Missile Weight, lb	11.0	15.2
Area of Impact, in.^{2}	3.14	3.14
$\left(\frac{\text{Missile Weight}}{\text{Impact Area}}\right)$, psi	3.5	4.84
Velocity, fps	20.0	120.0

2. For a tear at the junction between the temperature element assembly and the boss for the "without well" element and at the junction between the boss and the well for the "with well" element:

Characteristics	"without well"	"with well"
Flow Discharge Area, in.^{2}	0.11	0.60
Thrust Area, in.^{2}	3.14	3.14
Missile Weight, lb	4.0	6.1
Area of Impact, in.^{2}	3.14	3.14
$\left(\frac{\text{Missile Weight}}{\text{Impact Area}}\right)$, psi	1.27	1.94
Velocity, fps	75.0	120.0

TABLE 3.5-6

CHARACTERISTICS OF OTHER MISSILES
POSTULATED WITHIN REACTOR CONTAINMENT

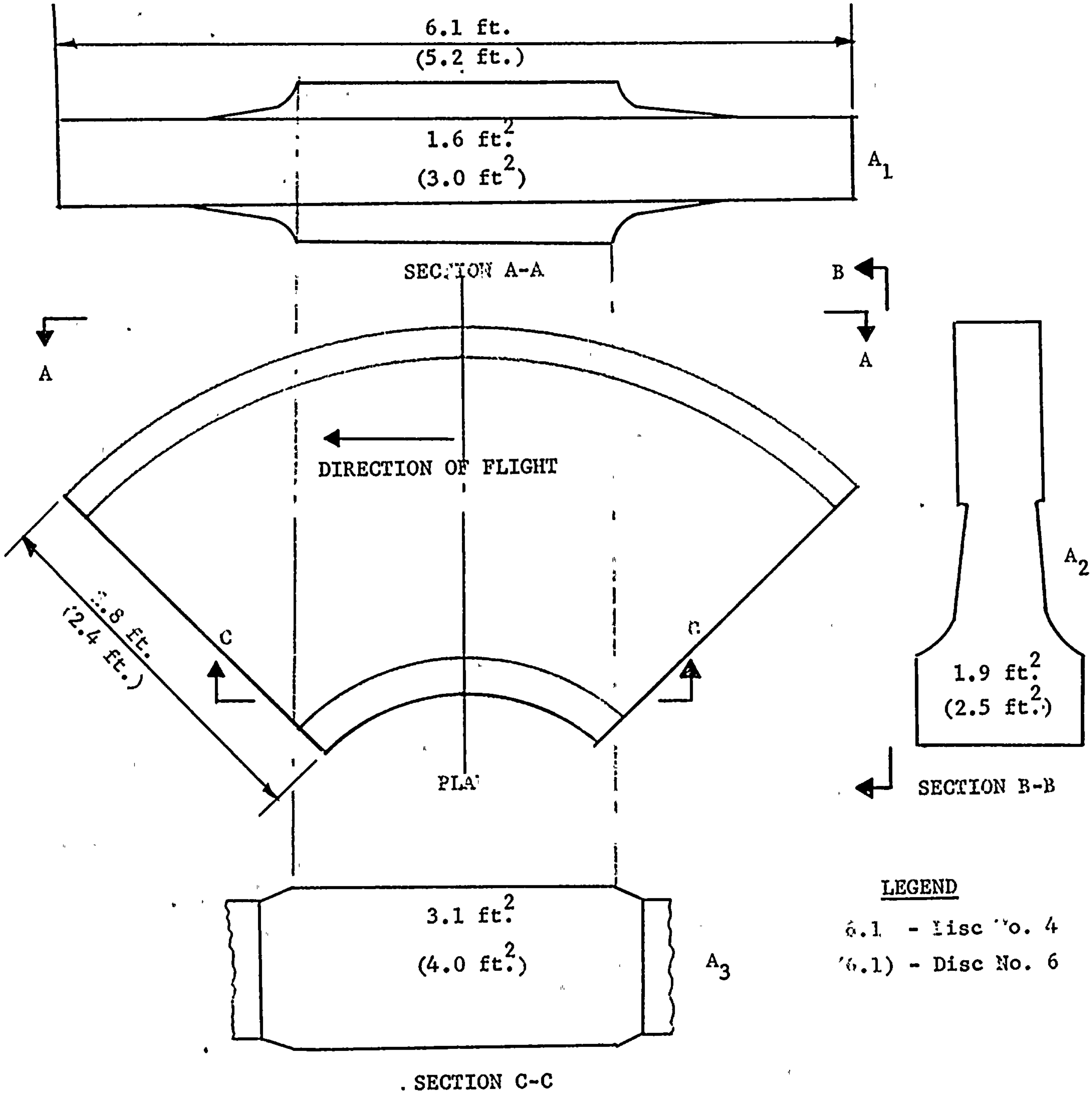
	Reactor Coolant Pump Temperature Element	Instrument Wall of Pressurizer	Pressurizer Heaters
Weight, lb	0.25	5.5	15.0
Discharge Area, $in.^2$	0.50	0.442	0.80
Thrust Area, $in.^2$	0.50	1.35	2.4
Impact Area, $in.^2$	0.50	1.35	2.4
$\left(\frac{\text{Missile Weight}}{\text{Impact Area}}\right)$, psi	0.5	4.1	6.25
Velocity, fps	260.0	100.0	55.0

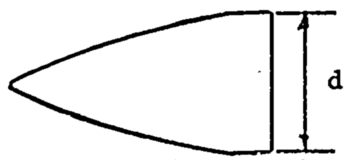
TABLE 3.5-7

BUILDING DESIGN DATA USED IN TURBINE MISSILE IMPACT ANALYSIS

BARRIER	THICKNESS	MATERIAL	LARGEST IMPACT ANGLE (MEASURED FROM SURFACE)
Fuel Handling Building Roof (Composite Roof)	0.5 in. 2.0 in. 0.125 in.	Celotex Board Zonolite (light weight concrete) Metal Decking	90°
Auxiliary Bldg. Roof (various locations)	40 in. 33 in. 18 in.	Concrete (Area H, El. 163'-4") Concrete (Area K, El. 140'-0") Concrete (Vent.Room Area K, El. 165'-10")	90°
Control Room North and South wall	36 in.	Concrete	25°
Auxiliary Bldg. North and South wall	36 in.	Concrete	22°
Turbine Bldg. deck @ El. 140'-0"	12 in.	Concrete	90°
Containment Side El. 140'-0"	44 in.	Concrete and 0.25 in. steel liner (Heavily reinforced with woven #18 reinforcing Bar)	90°
Top of Containment	30 in.	Concrete and 0.25 in. steel liner (Heavily reinforced with woven #18 reinforcing Bar)	90°

4



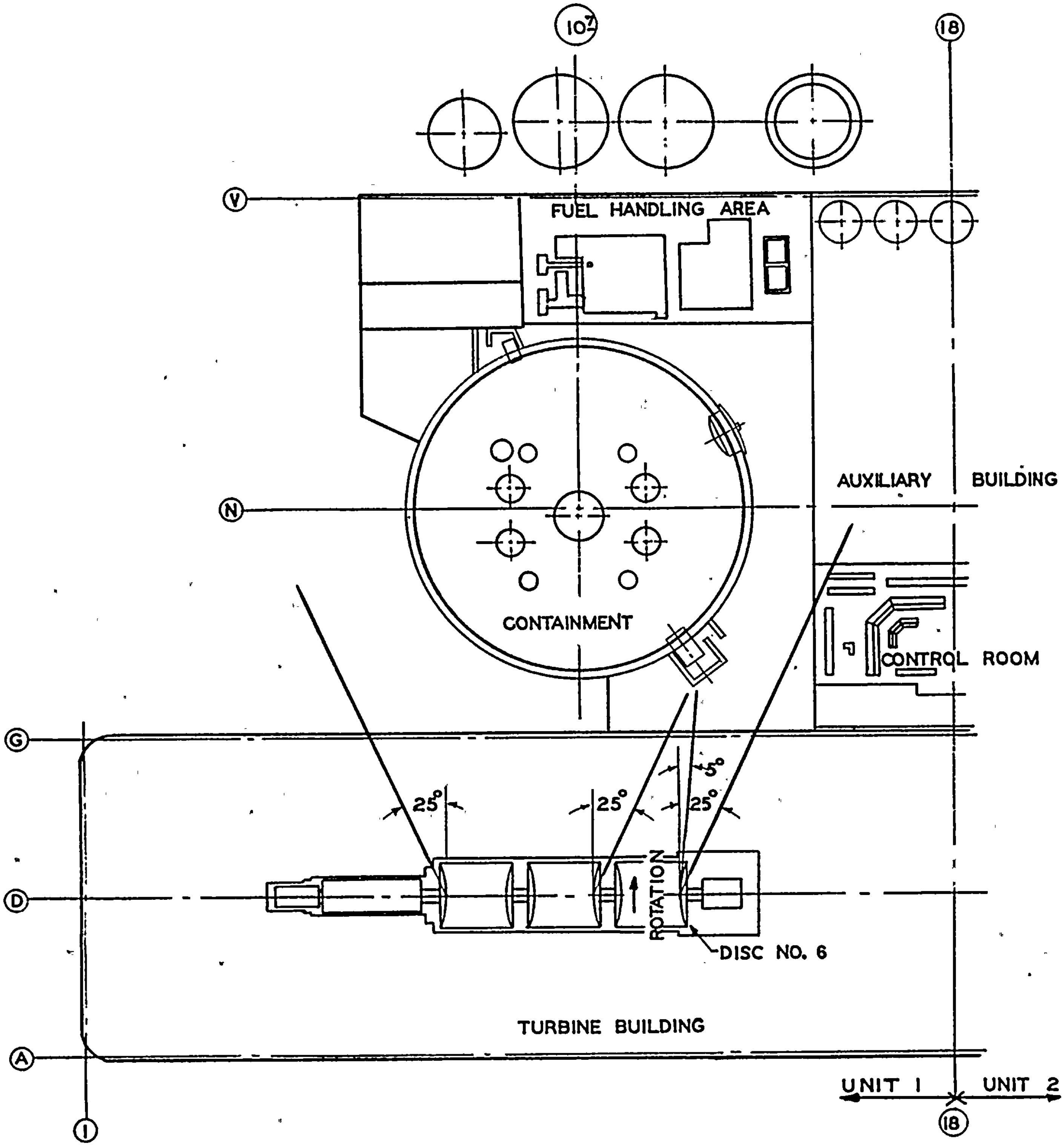


TYPICAL BALLISTIC MISSILE

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.5-0C
PHYSICAL DIMENSIONS OF
TURBINE MISSILES

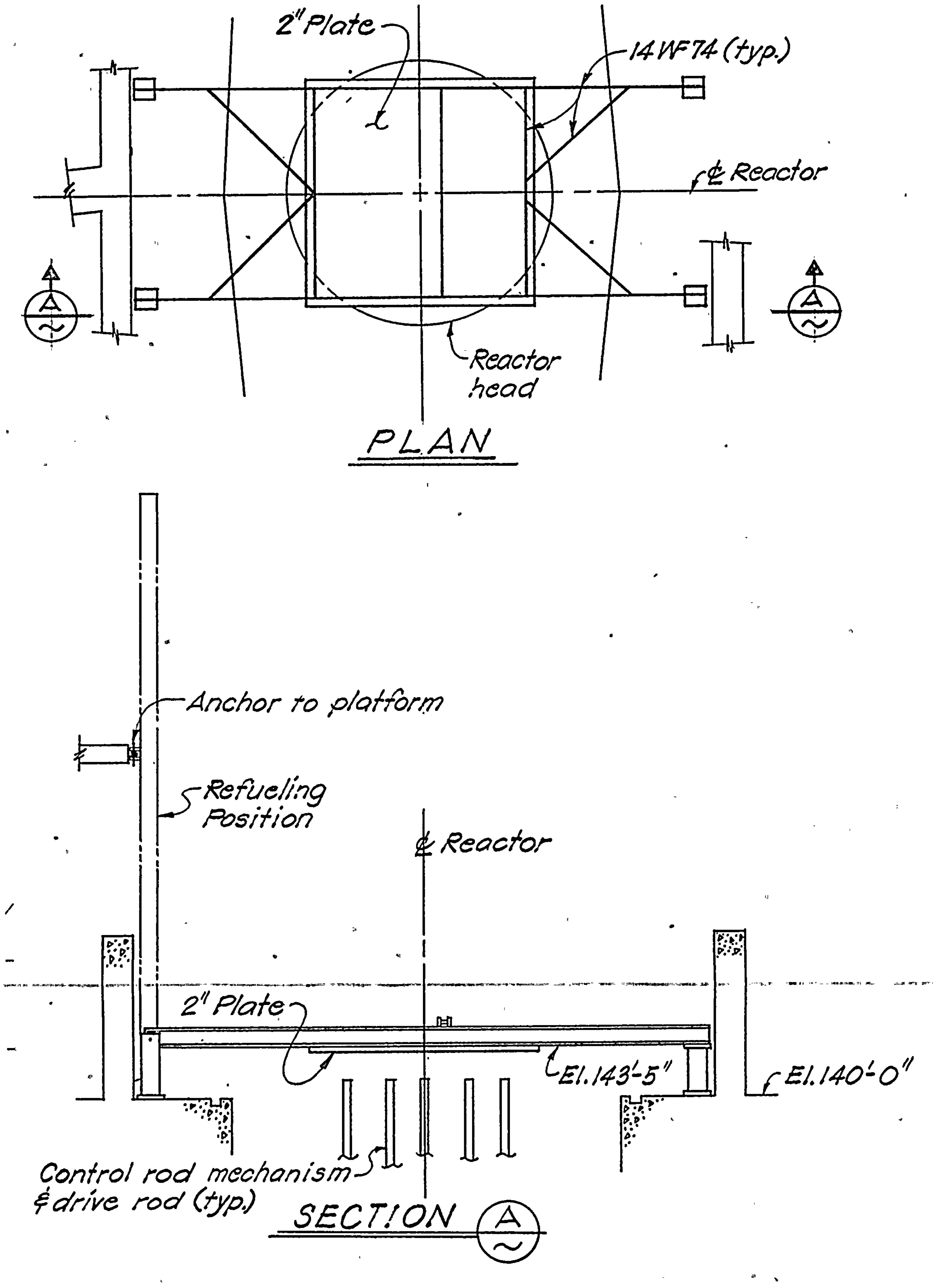
Amendment 9 May 1974



NOTE:
UNIT 2 IS SIMILAR EXCEPT OPPOSITE HAND

UNITS 1 AND 2
DIABLO CANYON SITE

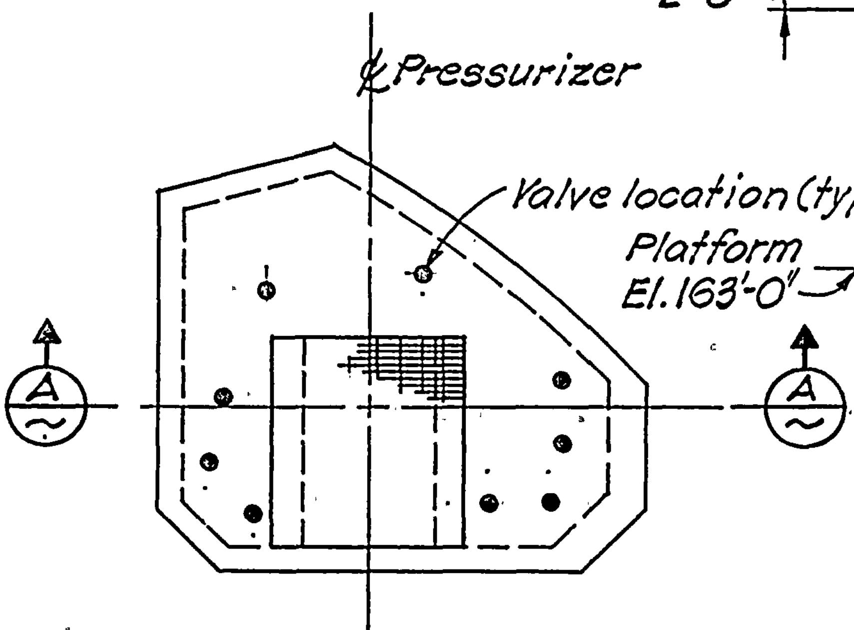
FIGURE 3.5-1
EJECTION PATH OF DISC NUMBER 6
FROM LOW PRESSURE TURBINE



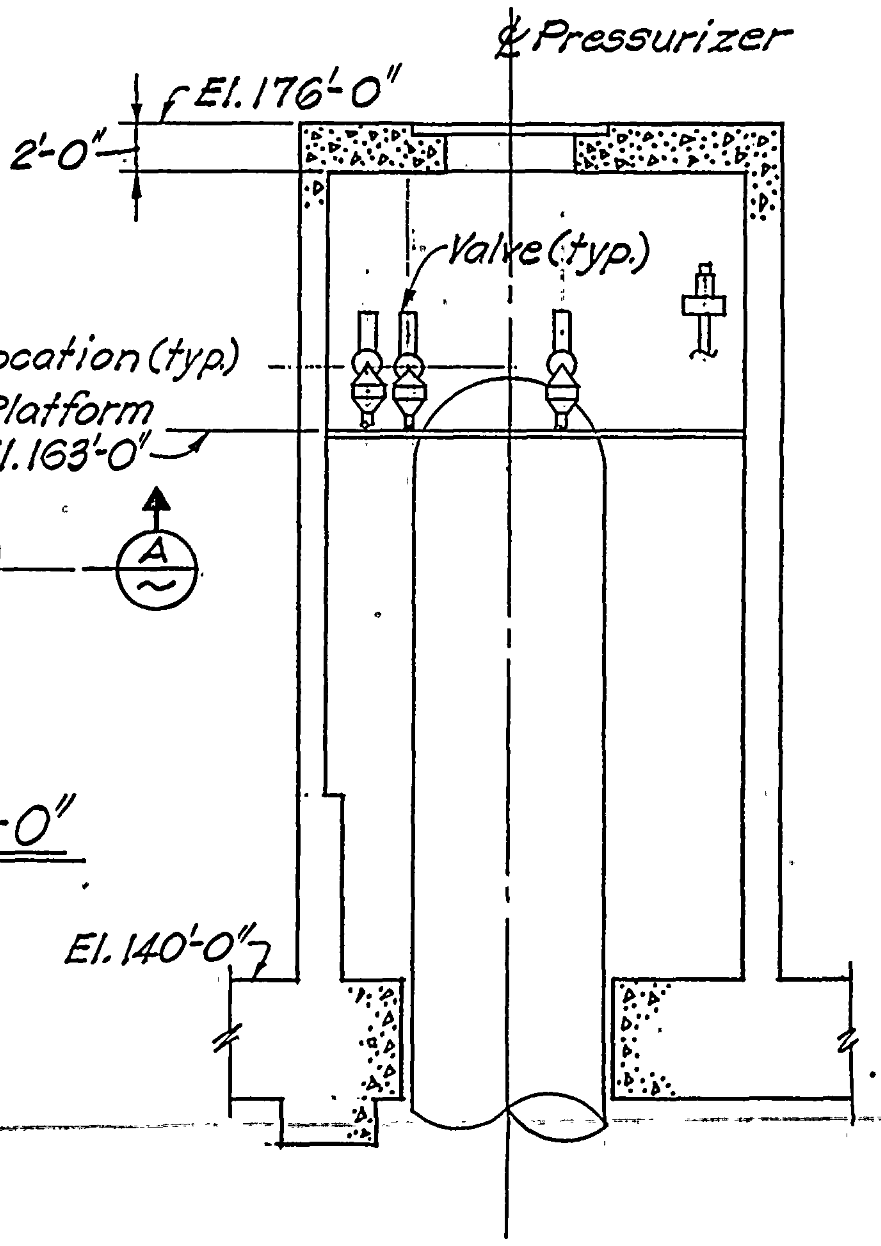
UNITS 1 AND 2 DIABLO CANYON SITE

FIGURE 3.5-0A

CONTAINMENT STRUCTURE REACTOR MISSILE SHIELD



PLAN AT EL. 176'-0"

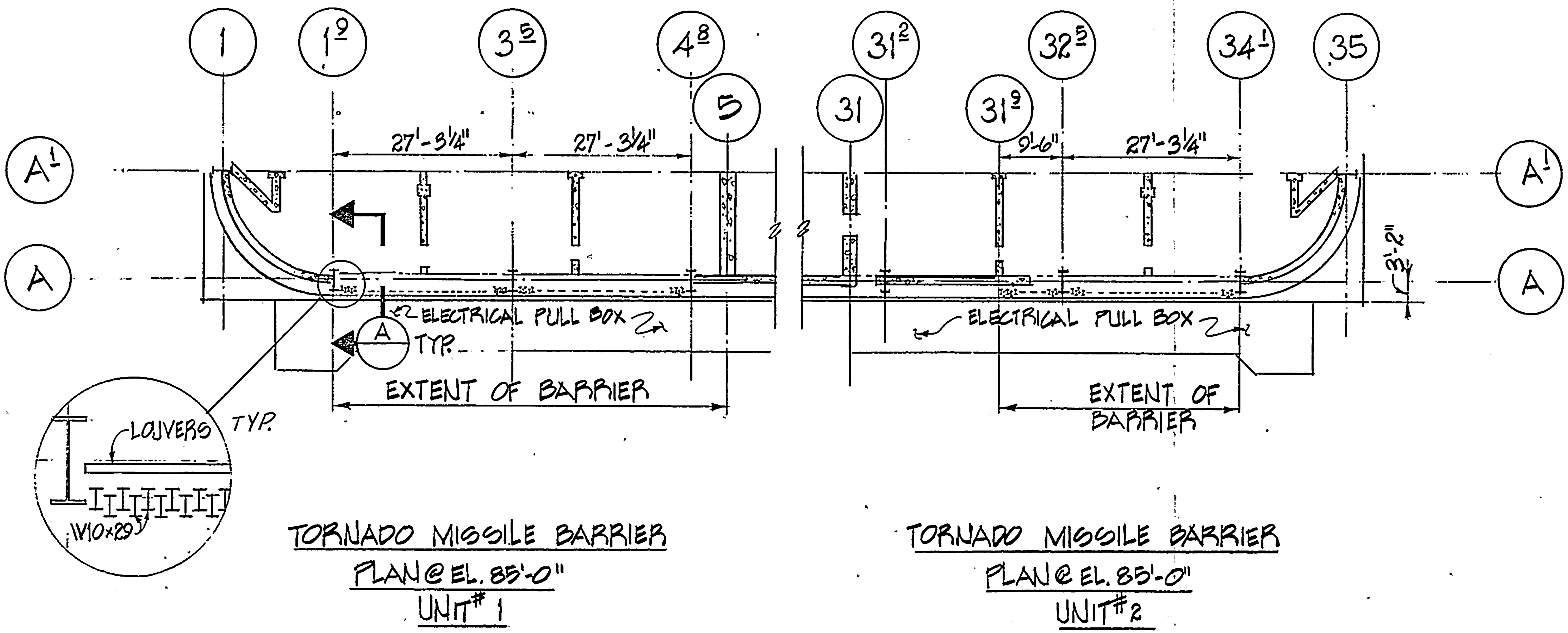


SECTION A

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.5-0B

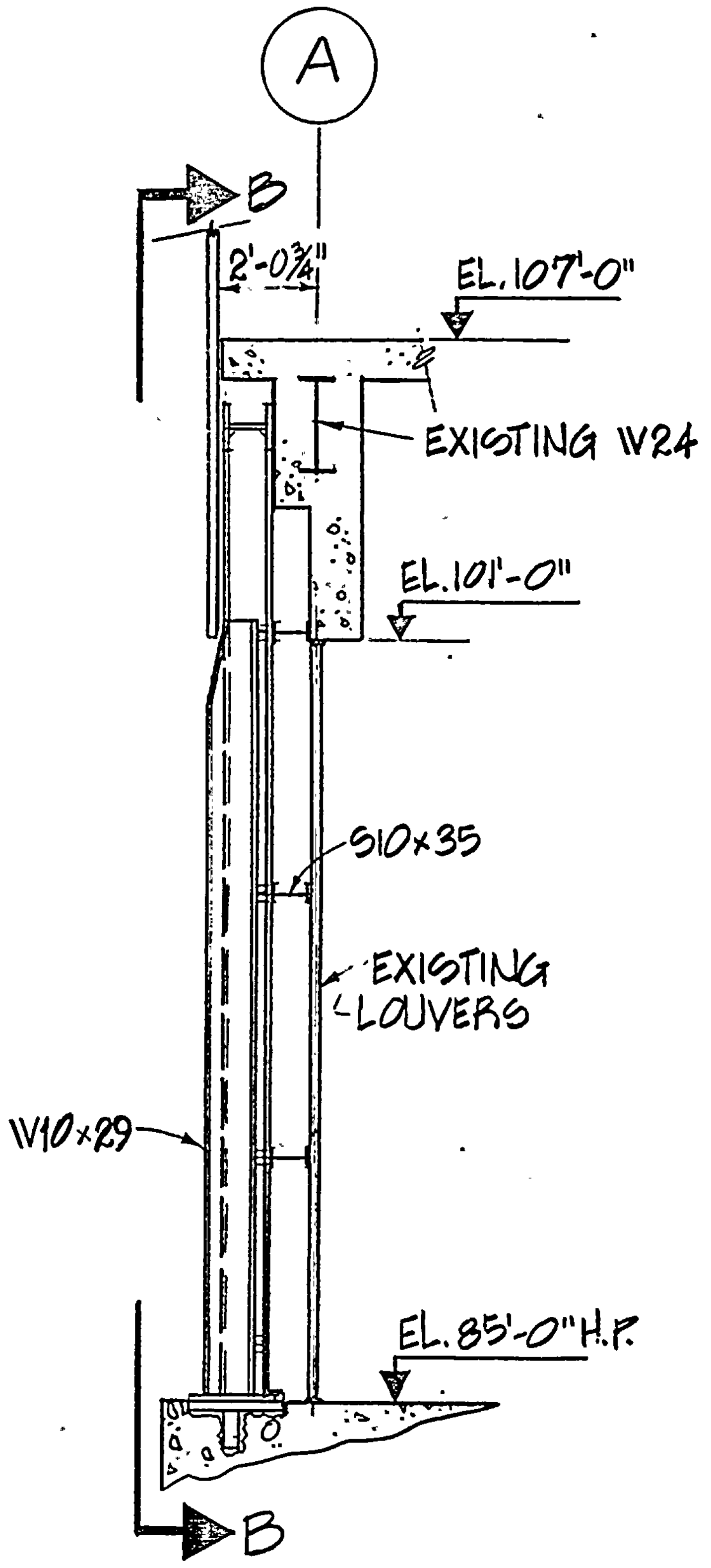
CONTAINMENT STRUCTURE
PRESSURIZER MISSILE SHIELD



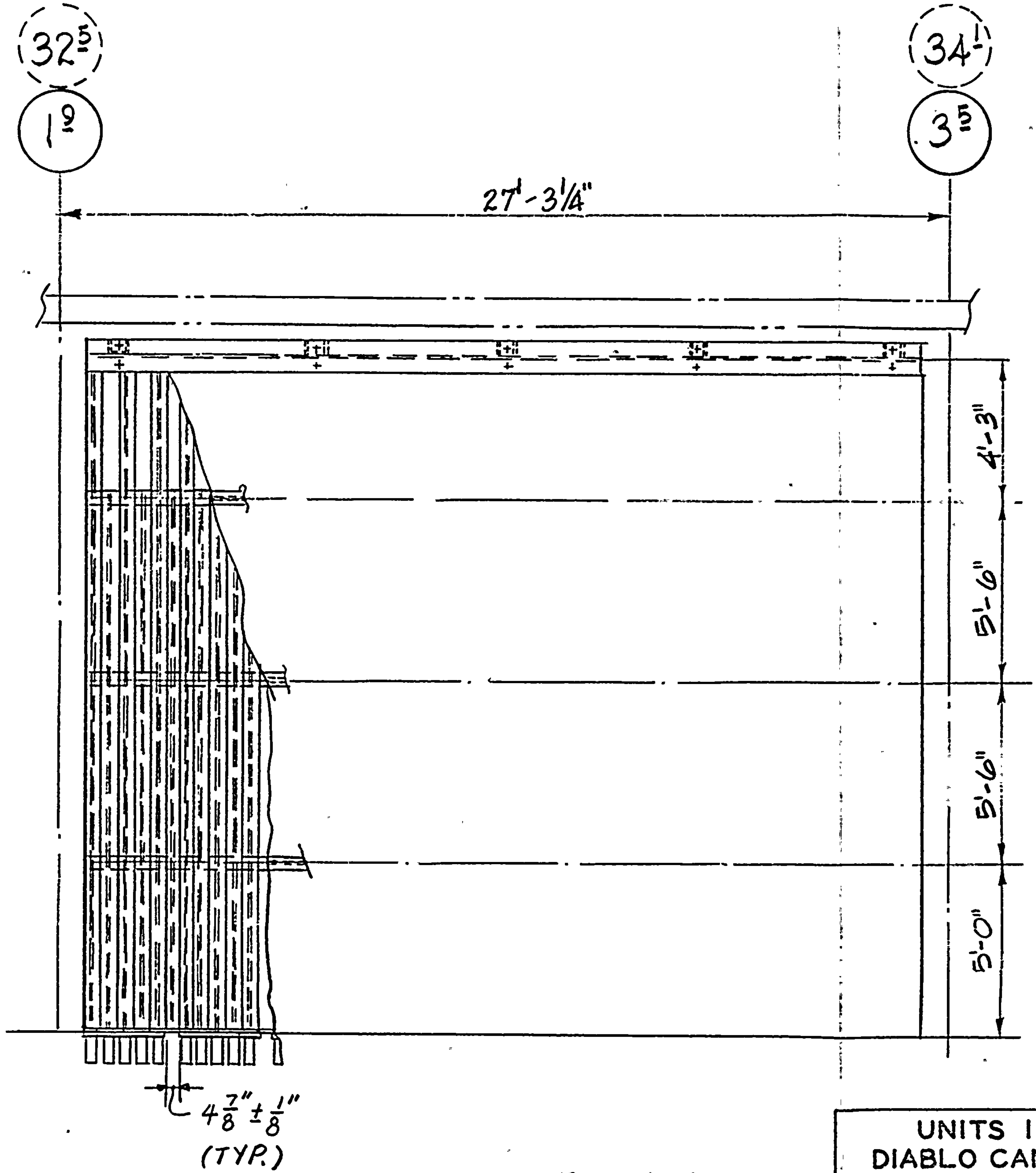
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.5-2 (SHEET 1 OF 2)
TURBINE BUILDING
TORNADO MISSILE BARRIER
FOR DIESEL GENERATOR LOUVERS

Amendment 33 August 1975



SECTION A-A



SECTION B-B

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.5-2 (SHEET 2 OF 2)
TURBINE BUILDING
TORNADO MISSILE BARRIER
FOR DIESEL GENERATOR LOUVERS

3.6 CRITERIA FOR PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH A POSTULATED RUPTURE OF PIPING

Special measures have been taken in the design and construction of the plant to protect the public against the consequences of dynamic effects associated with postulated piping ruptures both inside and outside the containment. The containment and all essential equipment inside and outside the containment, particularly components of the reactor coolant pressure boundary and other safety related components, are protected against the effects of blowdown jet, reactive forces, and pipe whip resulting from the postulated rupture of piping. This section presents the criteria, which have been established for the Diablo Canyon plant, for protection against these dynamic effects.

The design loading combinations, the design condition categories (normal, upset, emergency and faulted), and design stress limits applied to components, supports and pipe whip restraints of essential Design Class I components and piping of fluid systems within the Containment are described in Sections 3.9 and 5.2. Section 5.2 adresses Class A components while Section 3.9 addresses the remaining components. A discussion of potential missiles is presented in Section 3.5. 3

3.6.1 GENERAL CRITERIA

General Criteria - Piping Inside Containment

The following general criteria and discussion apply to the primary reactor coolant loop piping described in Subsection 3.6.2 and to the other piping in the Containment described in Subsection 3.6.3.

Engineered safety features are provided for core cooling and boration, pressure reduction, and activity confinement in the event of a loss of reactor coolant or steam or feedwater line break accident to assure that the public is protected in accordance with 10 CFR 100 guidelines. These safety systems have been designed to provide protection for a Reactor Coolant System pipe rupture of a size up to and including a double ended severance of a Reactor Coolant System main loop.

In order to assure the continued integrity of the vital components and the engineered safety systems, consideration is given to the consequential effects of the pipe break itself in order to meet the following criteria:

1. The minimum performance capabilities of the engineered safety systems must not be reduced below that required to protect against the postulated break;
2. The containment leaktightness must not be decreased below the design value, if the break leads to a loss of reactor coolant;* and
3. A Reactor Coolant System pipe break must not cause a steam-feedwater system pipe break and vice versa.

The fluid discharge from ruptured piping would produce reaction and thrust forces in the piping systems. The effects of these forces have been considered in assuring that the general criteria and performance of engineered
3 safety systems are satisfied.

The piping connections to the primary reactor coolant loops fall into the general categories illustrated in Figure 3.6-1. These categories are defined by the direction of flow to or from the primary reactor coolant loops and by the associated valve configuration. A rupture of these lines conceivably could cause uncontrolled loss of reactor coolant depending on the precise location of the break and the line configuration.

In establishing the dynamic effects criteria, uncontrolled loss of reactor coolant is assumed to occur for a pipe break out to the restraint of the second, normally open, automatic isolation valve (Case II, Figure 3.6-1) on outgoing lines and out to and including the second check valve on incoming lines normally with flow (Case III, Figure 3.6-1). It is assumed that motion of the unsupported line containing the isolation valves could cause functional failure of the operators of both valves. A pipe break beyond the restraint or

*The containment is defined here as the containment structure liner and penetrations, and the steam generator shell, the steam generator steam side instrumentation connections, the steam feedwater, blowdown and steam generator drain pipes within the containment structure second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line closes. Accordingly, both of the automatic isolation valves must be suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, frequent testing of the capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV, Figure 3.6-1), uncontrolled loss of reactor coolant is assumed to occur for pipe breaks on the reactor side of the valve.

It is assumed that a break of the piping associated with Engineered Safety Features does not occur during the injection phase following a loss of coolant. During the recirculation phase, a leak or equivalent break resulting in a maximum flow of 50 gpm is assumed to be credible. This value is based on the flow that would result from the complete failure of a residual heat removal pump seal. Therefore, special provisions for protection against the dynamic effects resulting from pipe rupture are not made for the systems in operation during the injection and recirculation phase following a LOCA.

Breaks are postulated at points of high fatigue or points of high primary plus secondary stress for normal and upset operating conditions.

The piping of the reactor coolant loops was designed to ANSI B31.1. Design was completed prior to both the issuance of Regulatory Guide 1.46 and the nuclear piping codes B31.7 and ASME III to which the break criteria of the recent Westinghouse report WCAP-8082[(4)] specifically apply. Consequently, these documents were not available when the discrete break locations for the reactor coolant loop (RCL) were determined. However, a comparison of the postulated break locations for the RCL and those of WCAP-8082 shows that the break locations are similar. 8

The breaks in the hot and cold legs were placed in the straight run outside of the primary shield wall. These circumferential breaks were chosen so as to allow full double-ended pipe separation and full discharge flow rather than limited area breaks and limited flow which would be obtained from a break inside the shield wall or at the reactor vessel nozzles. These RCL break locations were chosen for the analysis since they would result in conservative dynamic forcing functions and conservative resultant system stresses. The criteria used to determine the break locations on the RCL provide equivalent conservatism and result in equivalent protection to the criteria of Regulatory Guide 1.46. 8

General Criteria - Piping Outside Containment

The criteria which apply to the evaluation of the dynamic effects associated with postulated pipe rupture outside the containment are in accordance with those given in Mr. Giambusso's letter of December 18, 1972 on this subject.

3.6.2 SPECIFIC CRITERIA - PRIMARY REACTOR COOLANT PIPING

Locations Where Design Basis Piping Breaks are Postulated to Occur

To insure integrity and design adequacy of the primary reactor coolant loop piping and equipment supports system in the event of a highly improbable pipe rupture accident, a number of pipe rupture break locations are postulated. The primary reactor coolant loop has been analyzed for the design pipe breaks listed in this Subsection and shown on Figure 3.6-4. These discrete break locations and types were determined by an engineering approach which employs, as its basis, stress and fatigue analyses, system considerations, operational characteristics, and loading conditions. The dynamic analyses of the primary reactor coolant loop piping and equipment supports system for each of these break locations assure that public health and safety will be adequately protected.

Location and Type of Postulated Primary Coolant Loop Failure

1. Straight portion of hot leg piping - guillotine
2. Straight portion of cold leg piping - guillotine
3. Steam generator inlet nozzle - guillotine
4. Steam generator outlet nozzle - guillotine
5. Reactor coolant pump inlet nozzle - guillotine
6. 50° elbow - split
7. Flow entrance to the 90° elbow - guillotine
8. RHR primary loop connection - guillotine
9. Safety injection/primary coolant loop connection - guillotine
10. Pressurizer surge/primary coolant loop connection - guillotine
11. Loop closure weld in crossover leg - guillotine

The break area for both guillotine and longitudinal breaks can be assumed to be less than the cross sectional area of the pipe when analytically or experimentally substantiated. In the absence of this data, the break area is the cross sectional area of the pipe. The break length for the longitudinal breaks is considered to be equal to two pipe diameters. For the breaks listed in this subsection and shown on Figure 3.6-4, the break area is conservatively assumed to be the cross sectional area of the pipe.

Thrust-Time Relationship

The blowdown forces caused by a rupture of a primary reactor coolant loop pipe include the specific jet thrust at the break location and the internal hydraulic forces resulting from the acceleration of the fluid within the broken and unbroken loops.

Hydraulic forcing functions are claculated for the ruptured and intact reactor coolant loops as a result of a postulated LOCA. These forces result from the transient flow and pressure histories in the Reactor Coolant System. The calculation is performed in two steps. The first step is to calculate the transient pressure, mass flow rates and thermodynamic properties as a function of time. In the second step the results obtained from the hydraulic analysis, along with input of areas and direction coordinates,are used to calculate the time history of forces at appropriate locations in the reactor coolant loops. 3

The analysis is performed on integrated analytical models including the steam generator and reactor coolant pump, the associated supports and restraints,and the attached piping. An elastic-dynamic three-dimensional model of the reactor coolant loop is constructed. The boundary of the analytical model is, in general, the interface between the foundation concrete and the support structure. The deformation of the reinforced concrete foundation supports is considered where applicable to the reactor coolant loop model.

The steps in the analytical method are:

1. The initial deflected position of the reactor coolant loop model is defined by applying the initial steady state condition of the unbroken reactor coolant loop model.

2. Natural frequencies and normal modes of the broken loop are determined.

3. The initial deflection, natural frequencies, normal modes, and time-history forcing functions are used to determine the time-history forcing dynamic deflection response of the lumped mass representation of the reactor coolant loop.

4. The forces imposed upon the equipment supports and restraints by the loop are obtained by multiplying the support stiffness matrix and the time history of displacement vector at the support point.

5. The time-history dynamic deflections at mass points are treated as an imposed deflection condition on the ruptured loop reactor coolant loop model and internal forces, deflections, and stresses at each end of the members on the reactor coolant loop piping systems are computed.

3

The results are used to verify the adequacy of the piping, equipment supports and restraints.

The hydraulic model represents the behavior of the coolant fluid within the entire Reactor Coolant System. Key parameters calculated by the hydraulic model are pressure, mass flow rate and density. These parameters are supplied to the thrust calculation, together with appropriate plant layout information to determine the time-dependent loads exerted by the fluid on the loops. In evaluating the hydraulic forcing functions during a postulated LOCA, the pressure and momentum flux terms are dominant. The inertia and gravitational terms are taken into account in evaluation of the local fluid conditions in the hydraulic model.

The blowdown hydraulic analysis provides the basic information concerning the dynamic behavior of the reactor core environment for the loop forces, reactor kinetics and core cooling analysis. This analysis requires the ability to predict the flow, quality and pressure of the fluid throughout the reactor system. The SATAN-V code was developed with a capability to provide this information.

The SATAN-V computer code performs a comprehensive space-time dependent analysis of a loss-of-coolant accident and is designed to treat all phases of the blowdown. The stages are: a) a subcooled stage where the rapidly changing pressure gradients in the subcooled fluid exert an influence upon the Reactor Coolant System internals and support structures, b) a two-phase depressurization stage and c) a saturated stage.

The code employs a one-dimensional analysis in which the entire Reactor Coolant System is divided into control volumen. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coastdown and cavitation, core and steam generator heat transfer including the W-3 DNB correlation, in addition to the reactor kinetics,are incorporated in the code. The SATAN-V code is described in Subsection 15.4.1.

The blowdown hydraulic loads on primary loop components are computed from the following equation: 3

$$F = 144A \left[(P-14.7) + \left(\frac{[\dot{m}]^2}{144\rho g A_m}\right)\right]$$

which includes both the static and dynamic effects. The symbols and units are:

F = Force, lb_f
A = Aperture area, ft^2
P = System pressure, psia
m = Mass flow rate, lb/sec
ρ = Density, lb_m/ft^3
g = Gravitational onstant = 32.174 ft/sec^2
A_m = Mass flow area, ft^2

In the model to compute forcing functions, the main reactor coolant loop system is represented by a similar model as employed in the blowdown analysis. The entire loop layout is described in a global coordinate system. Each node is fully described by: a) blowdown hydraulic information and b) the orientation of the steamlines of the force nodes in the system, which includes flow areas and projection coefficients along the three axes of the global coordinate system. Each node is modeled as a separate control volume with one or two flow apertures associated with it. Two apertures are used to simulate a change in flow direction and area. Each force is divided into its x, y and x components using the projection coefficients. The force components are then summed over the total number of apertures in any one node to give a total x force, total y force and total z force. These thrust forces serve as input to the piping/restraint dynamic analysis.

The dynamic analysis of reactor coolant loops employs displacement method, lumped parameter, stiffness matrix formulation and assumes that all components behave in linear elastic manner.

Thrust calculations are made at each of these nodes using the STHRUST computer code.

3 Load Combinations and Allowable Stresses

A dynamic analysis is performed on the reactor coolant loop/support system for the pipe break cases discussed above to determine component and component support loadings. As described in Section 5.2, these loadings are combined with seismic and other loadings and the associated stresses are held within faulted condition limits.

Equipments Support Criteria

It is essential that the equipment support structures (reactor pressure vessel, steam generator and reactor coolant pump) be protected from the impact of large whipping pipes or be designed to resist such impact. This protection is accomplished by separation of equipment and piping, or by providing pipe restraints to prevent the formation of a plastic hinge mechanism. If any branch pipes are supported from equipment support structures, the reaction force resulting from a rupture of these lines is considered in designing the equipment supports. Small pipes are assumed to cause no significant damage to equipment supports.

In order to achieve an adequate primary reactor coolant loop design, the equipment support structures must be designed to prevent both uplift and overturning of the equipment under the action of normal and abnormal forces.

Reactor Vessel

The reactor vessel support structure is designed to resist thrusts that are considered to originate from the following two sources: a) the reactions of the blowdown forces in the primary reactor coolant loop piping which are eventually transmitted to the reactor nozzles, and b) the forces within the reactor pressure vessel shell acting on the reactor internals and shell wall.

The superposition of these effects, in time-history form, permits accurate determination of the loads transmitted to the reactor vessel support structure. The design and details of the reactor vessel support structure are further discussed in Subsection 5.5.14.

Steam Generator

The steam generators are supported in a manner that allows for thermal expansion of the equipment from cold to operating condition. In addition, the rupture of the steam side of a steam generator, as well as rupture of steam, feedwater, blowdown and instrument piping as a result of thrust forces created by the rupture of a primary reactor coolant loop pipe, is prevented. The steam generators are also supported in a manner that prevents rupture of a primary reactor coolant loop pipe as a result of thrust forces created by the rupture of a steam or feedwater line.

Guides and restraints are employed, where required, to limit the motion of the steam generators under the reaction forces that result from a primary reactor coolant loop pipe break, to a distance that is compatible with the flexibility of the steam and feedwater piping. Also, the motion of the steam generators, under the reaction forces due to a steam or feedwater pipe break, is limited to a distance that is compatible with the flexibility of the primary reactor coolant piping. The design and details of the steam generator support structures are further discussed in Subsection 5.5.14.

Reactor Coolant Pumps

Each reactor coolant pump is supported in a manner that would limit its displacement short of the primary shield, secondary shield, the steam generator,e steam generator supports, equipment and piping in adjacent loops, and the hot leg of the affected loop as a result of a rupture occurring in either the pump suction or discharge piping. The design and details of the reactor coolant pump support structures are further discussed in Subsection 5.5.14.

Jet Dynamic Force

A jet dynamic force will result from any of the pipe breaks postulated above. The force, caused by the momentum change of fluid flowing through the break, is a function of the upstream fluid conditions, fluid enthalpy, source pressure, pipe flow restrictions, friction and dimensions. Structural barriers and physical separation by plant layout have been used in the design to limit the effects of impingement. Where necessary, the jet forces resulting from the pipe break have been computed using the following method:

Jet dynamic forces on structures are calculated as:

$$F_j = C_j\ (1.26\ PA) \qquad \text{(Reference 2)}$$

Where: F_j = jet dynamic force acting on a structure

C_j = factor to account for the dynamic nature of the load. In determining the value of C_j, inelastic behavior is assumed.

P = system operating pressure

A = cross-sectional area of pipe

30 The above loads were considered in the structural design.

3.6.3 SPECIFIC CRITERIA - OTHER PIPING INSIDE CONTAINMENT

The containment and all essential equipment within the containment, particularly components of the reactor coolant pressure boundary and other safety related components, have been protected against the effects of pipe whip resulting from postulated rupture of piping. The criteria for locating and minimizing these effects are described in this Section.

<u>Systems in Which Design Basis Piping Breaks are Postulated to Occur - Piping Inside Containment</u>

The following systems (in addition to the primary coolant loops discussed in the previous section), have been evaluated with regard to the dynamic effects associated with a ruptured pipe:

1. Reactor Coolant System
    a. Pressurizer Surge Line
    b. Pressurizer Spray Line
    c. Pressurizer Relief and Safety Valve Lines
    d. Remote Temperature Detector Bypass Lines
    e. Drains

2. Chemical and Volume Control System
    a. Charging Line and Auxiliary Spray Line
    b. Reactor Coolant Pumps Seal Water Injection 3
    c. Letdown Line
    d. Excess Letdown Line
    e. Reactor Coolant Pumps - Seal Vent and Leakoff

3. Safety Injection System
    a. Accumulator Injection Lines
    b. Safety Injection Lines

4. Residual Heat Removal System
    a. Residual Heat Removal Supply
    b. Residual Heat Removal Return (Accumulator Injection)

5. Turbine Steam Supply System
    a. Main Steam Lines
    b. Feedwater Lines
    c. Steam Generator Blowdown and Drain Lines

<u>Criteria for Protection of Engineered Safeguard Features - Piping Inside Containment</u>

In addition to the three criteria on the consequential effects of the pipe break itself, as given in Section 3.6.1, propagation of damage must be limited in type and/or degree as follows:

4. A pipe break which is not a loss of reactor coolant must not cause a loss of reactor coolant or steam or feedwater line break.

5. Branch lines connected to the Reactor Coolant System are defined as "large" if they have an inside diameter greater than 4 inches (up to the largest connecting line, which is the pressurizer surge line.) Rupture of these lines results in a rapid blowdown from the Reactor Coolant System, and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

   In addition to the above (1 through 4), large piping must be restrained so that:

   a. Propagation of the break to the unaffected loops is
   3 prevented to assure the delivery capacity of the accumulators and low head pumps.

   b. Propagation of the break in the affected loop is permitted to occur but must not exceed 20 percent of the area of the line which initially ruptured. This criterion has been voluntarily applied so as not to substantially increase the severity of the loss of coolant.

6. Branch lines connected to the Reactor Coolant System are defined as "small" if they have an inside diameter equal to or less than 4 inches.

   Westinghouse Nuclear Energy System tests have demonstrated that lines hitting equal or larger size lines of the same schedule will not cause failure of the line being hit, e.g., failure of a 1-inch line will not cause subsequent failure of a 1-inch or larger size line. The reverse, however, is assumed to be probable; i.e., a 4-inch line, should it fail and whip as a result of the fluid discharged through the break, is assumed to cause failure of smaller lines such as neighboring 3-inch or 2-inch lines. In this case, the total break area must be limited to less than 12.5 square inches.

In the unlikely event that one of the small pressurized lines should fail and result in a loss of coolant accident, the piping is restrained or arranged to meet the following requirements in addition to (1 through 4) above.

a. Break propagation must be limited to the affected leg; i.e. propagation to the other leg of the affected loop and to the other loops must be prevented.

b. Propagation of the break in the affected leg is permitted but must be limited to a total break area of 12.5 square inches (4-inch inside diameter). The exception to this case is when the initiating small break is the high head safety injection line. Further propagation must not be permitted for this case.

c. Damage to the high-head safety injection lines connected to the other leg of the affected loop or to the other loops must be prevented.

d. Propagation of the break to high-head safety injection lines connected to the affected leg must be prevented if the line break results in a loss of core cooling capability due to a spilling injection line. 3

As stated above, a small pipe break in one of the legs of a given loop must not cause a break in the opposite leg of the same loop or in other loops. A rupture of the resistance temperature detector (RTD) bypass loop would constitute an exception to this criterion. If a break occurs at any location in the RTD bypass, however, the resulting blowdown, because of the relatively high flow resistance in the line, is less severe than from a rupture directly adjacent to the reactor coolant pipe. Safety injection capacity for this case is provided by the high-head connections.

7. Restraint on the lines are necessary in order to prevent impact on and subsequent damage to the neighboring equipment or piping. Restraint type and spacing must be chosen such that a plastic hinge (unrestrained rotation) on the pipe at the two support points closest to the break will not be formed.

Pipes are allowed to form plastic hinges in areas or arrangements where:

a. Whipping free sections cannot reach equipment or other pipes for which protection is required, or

b. protective barriers prevent the whipping pipe from impacting on components or pipes requiring protection or

c. the internal energy of the pipe is insufficient to impair the function of any equipment or structure, i.e., the design temperature is less than 200°F and the design pressure is below 275 psig.

Use of Piping Routing to Minimize Restraints - Piping Inside Containment

The piping is routed so that whipping of two free sections cannot reach equipment or other pipes for which protection is required. Barriers are utilized, where available, to prevent the whipping pipe from impacting on equipment or piping requiring protection. For example, the crane wall, operating floor,
3 and refueling cavity walls serve as barriers between the reactor coolant loops and the containment liner. Except for Emergency Core Cooling System lines attached to main reactor coolant loops, the engineered safety features are located outside of the crane wall. The Emergency Core Cooling System lines that penetrate the crane wall are routed outside of the crane wall so as to penetrate it in the vicinity of the loop to which they are attached. The results of analyses demonstrate that pipe whip resulting from a postulated break in an Emergency Core Cooling System line inside the crane wall will not cause damage in excess of that allowed by the established criteria.

Pipe Restraint Design and Location - Piping Inside Containment

Where the requirements as outlined above cannot be satisfied by judicious routing of the piping, pipe whip restraints are designed and located as outlined below:

1. Location of Pipe Whip Restraints

   Restraints are located at each zone of the piping over 1-inch size where formation of a plastic hinge could endanger a structure, system or component vital for safety. Design was completed prior to issuance of AEC Regulatory Guide 1.46 in May 1973.[1] The piping design on all of these systems is to ANSI B31.1-1967.

Comparison of the original criteria for determination of postulated pipe break locations with those in AEC Regulatory Guide 1.46 shows that they resulted in substantially more restraints being installed than would have been required by AEC Regulatory Guide 1.46 for ASME Code Class 1, 2, and 3 piping (PG&E Code Class A, B and C) within the containment (see Figure 3.6-2 and Tables 3.6-1 and 3.6-2). Breaks were postulated at all terminal ends and all locations, regardless of piping stress level, where plastic hinges could form and endanger safety systems or components. Regulatory Guide 1.46 requires a minimum of two postulated break locations per piping run or branch run, i.e., one for each terminal end, provided that combined stresses are below certain limits. Piping combined stresses due to dead load, seismic load pressure and thermal effects have been compared with those limits specified in Regulatory Guide 1.46. The combined stresses do not exceed these limits on any piping. Therefore, only terminal end break locations would have been postulated if Regulatory Guide 1.46 had been applied; and more restraints are provided than required by Regulatory Guide 1.46. 8

2. Summary of Pipe Whip Effects

Lines classified as high energy have been evaluated against Regulatory Guide 1.46 as minimum design criteria. These lines are summarized in Tables 3.6-1 and 3.6-2 which are check lists of pipe whip effects from postulated pipe ruptures inside the containment.

Calculation of Blowdown Forces and Plastic Hinge Moments - Piping Inside Containment

Blowdown Forces and jet impingement forces due to the postulated piping breaks on lines in the containment (other than the reactor coolant loops) were calculated from the formula

$F_B = 1.26\ P_oA$ (See Reference 2).

where: F_B = steady state blowdown force (lb)

1.26 = dimensionless factor F_B/P_oA for saturated steam and flashing liquid lines (See Reference 2)

P_o = line Pressure at time = 0

A = cross-sectional flow area of pipe with a full-area break.

Moments required to form a plastic hinge were calculated from the formula

$$M_p = \frac{KS_yI}{R_o}$$

where: M_p = plastic moment

$K = M_p/M_y$ where M_y is the moment to produce yielding on extreme fiber (See Reference 3)

$$M_p = 2.5\ S_y \frac{I}{R_o}$$

$$S_y = M\frac{R_o}{I}$$

$$M_y = S_y\frac{I}{R_o}$$

?

$$M_p = 2.5\ M_y$$

S_y = yield stress of material at temperature

I = moment of inertia of piping = $\frac{\pi}{4} (R_o^4 - R_i^4)$

R_o = outside radius of pipe

R_i = inside radius of pipe

K = 2.5 for materials in the piping systems, based on Westinghouse test results as outlined in WCAP 7287 (3)

Design of Restraint Structures - Piping Inside Containment

In determining their design load the pipe rupture restraints are considered independent of dead and live load supports and of seismic restraints. In equation form: $Y = Y_r$

where: Y = section strength required to resist design loads,

Y_r = equivalent static load on a pipe rupture restraint generated by the reaction on the broken high energy pipe during a postulated break, the load, Y_r, includes a minimum dynamic factor of 3 to account for the dynamic effects associated with a postulated pipe rupture $(1.26\ P_oA \times 3)$.

The allowable stress for the pipe rupture restraint is the yield stress of the material. Due to the high rate of strain that the restraint would experience after pipe rupture, the static yield strength of the material is increased by 5%. A typical restraint is shown in Figure 3.6-3.

3.6.4 SPECIFIC CRITERIA - PIPING OUTSIDE CONTAINMENT

Systems in Which Design Basis Piping Breaks Occur - Piping Outside the Containment

The following criteria and definitions apply to the selection of piping systems outside containment for evaluation of the dynamic effects associated with postulated pipe rupture.

1. All systems having a service temperature greater than 200°F or a design pressure greater than 275 psig are considered.

Open crack breaks are postulated to occur in the most adverse locations in piping having fluid temperature or pressure greater than the above.

Design basis breaks, in addition to crack breaks, are postulated in that portion of those high energy systems where both temperature and pressure exceed these levels. The criteria for determining the location of design basis breaks are defined later this Subsection.

2. Piping either encased in concrete or protected from structures, systems, and components important to safety by barriers is not considered.

3. Piping physically located such that unrestrained motion (pipe whip) in any direction about a plastic hinge formed after a pipe rupture could not impact any structure, system, or component important to safety is not considered.

The systems which contain high energy lines located outside of containment
in which both open crack and design basis piping breaks are postulated to 3
occur are:

- Condensate System
- Feedwater System
- Turbine Steam Supply System (Main Steam System)
- Extraction Steam and Heater Drip System
- Chemical and Volume Control System
- Safety Injection System
- Residual Heat Removal System
- Turbine and Generator Associated Systems

Open crack breaks are also postulated in the Auxiliary Steam System, as it contains piping at a temperature which exceeds 200°F.

Design Basis Piping Break Criteria - Piping Outside the Containment

The selection of design basis breaks is generally based on results of the piping stress analysis. These analyses consider effects of pressure, deadweight, thermal expansion during normal operating, upset and test conditions and the Design Earthquake (DE). Where such stress analyses are not available, the design basis breaks are postulated to occur at locations which would result in the most severe consequences.

Design basis breaks in straight or curved pipe 4 inches in diameter or greater are assumed to be longitudinal or circumferential, with the break area equal to the flow area of the pipe. Longitudinal breaks may have any orientation around the circumference of the pipe. Design basis breaks in pipe 1 to 4 inches in diameter are assumed to be circumferential only, with break area equal to flow area of the pipe. Design basis breaks at branch points are assumed to be circumferential in branch lines and longitudinal in run lines, with break area equal to flow area of the branch. The criteria for selection of design basis break locations in each piping run are as follows:

1. Postulate breaks at all terminal points (anchors or rigid equipment).

3 2. Postulate breaks at all branch points (terminal point of the branch run).

3. Postulate intermediate breaks between terminal points wherever thermal expansion stresses exceed 80% of S_A, where S_A is as defined in ANSI B31.1 - 1967.

4. Postulate intermediate breaks between terminal points wherever primary stress (pressure, weight, DE) plus thermal expansion stress exceeds 80% of $(S_h + S_A)$, where S_h and S_A are as defined in ANSI B31.1 - 1967. The summation of stresses (pressure + weight + DE + thermal) is referred to as "combined stress".

5. As a minimum, two intermediate breaks are selected at locations of highest stress.

6. For piping runs where DE stress analyses are not available, breaks are postulated at locations where pipe break would yield most severe consequences.

Crack breaks are assumed to have a flow area equal to one-half the pipe diameter times one-half the pipe thickness and are postulated in the most adverse orientations and locations throughout the piping.

Design Loading Combinations - Piping Outside Containment

Piping response analyses are performed on high-energy piping systems at those postulated break locations for which unrestrained pipe motion about a plastic hinge could impact or endanger vital systems. Factors and criteria which are considered are:

1. The dynamic nature of the loading.

2. Pipe impact effects due to gaps in piping restraints.

3. Nonlinear (elastic-plastic) pipe and restraint material properties and the effect of rapid strain rate on material properties.

4. For circumferential breaks in a pipe, whip occurs upon attainment of 50% of uniform ultimate strain at a plastic hinge due to loading from the blowdown reactive forces. The pipe whip is characterized by unrestrained motion of the pipe about the hinge in the direction governed by the vector thrust of the break force. 3

5. For longitudinal breaks, failure occurs upon attainment of a hinge mechanism with 50% of uniform ultimate strain on each hinge.

6. Lower-bound piping material properties are used for prediction of pipe whips.

7. Both lower and upper-bound piping material properties are used for prediction of loads on anchors and restraints.

8. Piping loads on the rupture restraints are limited to the equivalent of 50% of uniform ultimate strain in the restraint materials. The nonlinear material properties of the restraint are considered.

9. The following set of load combinations and allowable limits are used in evaluating Design Class I concrete structures for the effects of high energy pipe breaks:

$\underline{\text{Criterion 1.}}$ $U > D + L + T_a + R_a + 1.5\ P_a$

$\underline{\text{Criterion 2.}}$ $U > D + L + T_a + R_a + 1.25\ P_a + Y_r + Y_j + Y_m + 1.25\ Feqo$

$\underline{\text{Criterion 3.}}$ $U > D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + Feqs$

Where:

U = Section design capability

D = Dead load

L = Live load

T_a = Thermal loads due to postulated break

R_a = Pipe reactions under thermal conditions due to postulated break

P_a = Pressure equivalent static loading within or across a compartment and/or building, including an appropriate dynamic load factor.

Y_j = Jet impingement loads, including an appropriate dynamic load factor.

Y_m = Missile impact loads, including an appropriate dynamic load factor.

3 Feqo = Loads generated by Design Earthquake.

Feqs = Loads generated by Double Design Earthquake.

10. The following set of load combinations and allowable limits are used in evaluating those Design Class I steel structures outside the containment whose function is to provide protection against the effects of high energy pipe breaks.

Structures Evaluated Using the Elastic Working Stress Method

$\underline{\text{Criterion 1.}}$ $1.6S = D + L + T_a + R_a + P_a$

$\underline{\text{Criterion 2.}}$ $1.6S = D + L + T_a + R_a + P_a + Y_j + Y_r + Y_m + Feqo$

$\underline{\text{Criterion 3.}}$ $1.6S = D + L + T_a + R_a + P_a + Y_j + Y_r + Y_m + Feqs$

Structures Evaluated Using Plastic Design Methods

$\underline{\text{Criterion 1.}}$ $0.9Y = D + L + T_a + R_a + 1.5\ P_a$

$\underline{\text{Criterion 2.}}$ $0.9Y = D + L + T_a + R_a + 1.25\ P_a + Y_j + Y_r + Y_m + 1.25\ Feqo$

$\underline{\text{Criterion 3.}}$ $0.9Y = D + L + T_a + R_a + 1.0\ P_a + Y_j + Y_r + Y_m + 1.0\ Feqs$

Where:

S = Required section strength based on elastic design methods.

Y = Required section strength based on plastic design methods.

Dynamic Analysis - Piping Outside Containment

Analysis to determine the effects of a rupture of the main steam and feedwater piping, between the containment and the Turbine Building, is nearly complete and is reported in detail in Appendix 3.6A. This section of piping, because of its high energy content and proximity to safety related equipment and structures, is considered to possess the most potential for damage. Pipe break effects analyzed include pipe whip, jet impingement, pressurization of compartments, water flooding and the environmental effects of pressure, temperature and humidity.

Analysis to determine the effects of a rupture of the remainder of the main steam and feedwater piping, along with the other high energy piping systems defined earlier in this section, is currently in progress.

Locations chosen for design basis breaks were selected in accordance with the break criteria presented earlier in this section. These locations for the main steam and feedwater piping, between the containment and the turbine building, are shown in Appendix 3.6A, Figures 7-3 and 7-9. At all branch locations, the break is circumferential in the branch line and longitudinal in the run line, with break area equal to flow area of the branch. At all other design basis break locations both circumferential and longitudinal breaks are postulated.

Fluid blowdown thrust-time histories resulting from a pipe rupture are determined using PRTHRUST, a computer code derived from RELAP3, the Atomic Energy Commission's presently accepted loss of coolant accident computer code. The assumptions used for these analysis together with representative mathmatical models and typical results are presented in Appendix 3.6A.

At design basis break locations fluid forces determined from the blowdown analysis are utilized for the jet impingement evaluations. The method of determining impacted targets and resulting loadings is discussed in detail in Appendix 3.6A.

Pipe whip analyses of the main steam and feedwater piping, between the containment and the turbine building, resulting from ruptures at the identified locations were determined using computer program PIPERUP. PIPERUP determines the non-linear, elastic-plastic response of three-dimensional piping restraint systems to the fluid blowdown force-time histories defined above. Gaps between the piping and rupture restraints, as well as non-linear properties of the restraints are included in the analysis. A description of the analytical methods used in the analysis, mathematical models of the piping systems, and representative results are also presented in Appendix 3.6A. Preliminary results of this analysis indicate that the locations of restraints shown in Appendix 3.6A, Figures B-1 through B-6, will be adequate to prevent all whips which could result from breaks at the identified locations.

Protective Measures - Piping Outside Containment

All piping anchors, rupture restraints, restraint attachments, and safety related building structures are analyzed in accordance with the design load combinations described earlier in this section. Modifications and additions to existing structure and rupture restraints proposed to provide the necessary protection are identified in Appendix 3.6A.

Analysis to determine effects of jet impingement loadings or safety related equipment and conduits is currently in progress. Where found necessary, modifications such as impingement barriers and equipment relocations will be accomplished.

Affected equipment and conduits are also being investigated to ensure their operability in the post break temperature, pressure, and humidity environment.

3.6.5 REFERENCES

(1) "Protection Against Pipe Whip Inside Containment," AEC Regulatory Guide 1.46, May 1973.

(2) Moody, F.J., ASME Paper ~~69~~ 6 9-HT-31, 1969.

(3) Westinghouse Nuclear Energy Systems, WCAP 7287, January 1969.

8 (4) Westinghouse Nuclear Energy Systems, WCAP 8082, June 1973.

Table 3.6-1 Check List of Dynamic Effects From Postulated Rupture of Pipe Connected to the Reactor Coolant System

Large Break (>4" I.D.) Resulting in a Loss of Coolant	Location Description	Loop	Size	No. of Rupture Restraints	Is the Containment Liner Protected?	Is Break Propagation Prevented and Low Head Safety Injection Maintained to the Unaffected Loops?	Is Break Propagation in the Affected Loop Limited to 20%?	Is Damage to Steam System Prevented? Steam Line	Feed Line	Aux. Feed	Blow-Down	Level Taps	Sample Line	Is the Integrity of Equipment Supports Maintained? S.G.	Pump	Is Line Restrained to Meet Criteria with Regulatory Guide 1.46 as a Minimum?
Pressurizer Surge (16)	Hot Leg to Pressurizer	2	14	11	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Accumulator Injection (253)	Check Valve 8948A to Cold Leg	1	10	1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Accumulator Injection (254)	Check Valve 8948B to Cold Leg	2	10	1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Accumulator Injection (255)	Check Valve 8948C to Cold Leg	3	10	1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Accumulator Injection (256)	Check Valve 8948D to Cold Leg	4	10	1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Residual Heat Removal Supply (109)	Hot Leg to Valve 8702	4	14	5	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Low Head Safety Injection (235)	Loop to Check Valve	1	6	2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Low Head Safety Injection (236)	Loop to Check Valve	2	6	3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Low Head Safety Injection (237)	Loop to Check Valve	3	6	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Low Head Safety Injection (238)	Loop to Check Valve	4	6	3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pressurizer Relief Line (230)	Pressurizer to 1171 Take Off	-	6	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Pressurizer Safety Lines (727, 728, 729)	Pressurizer to Safety Valves	-	6	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1

Table 3.6-1 Check List of Dynamic Effects From Postulated Rupture of Pipe Connected to the Reactor Coolant System

Large Areas (>4" I.D.) not Resulting in a Loss of Coolant	Location Description	Loop	Size	No. of Rupture Restraints	Is a Loss of Coolant Prevented From Occurring?	Is Damage to Steam System Prevented? Steam Line	Feed Line	Aux. Feed	Blow-Down	Level Taps	Sample Line	Is the Integrity of Equipment Supports Maintained? S.G.	Pump	Is Boration Capability Maintained?	Is Cold Shutdown Capability Maintained? RHR Suction	RHR Return	Is Line Restrained to Meet Criteria with Regulatory Guide 1.46 as a Minimum?
Accumulator Injection (253, 1294)	Accumulator to Check Valve 8948A	1	10	6	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Accumulator Injection (254, 1295)	Accumulator to Check Valve 8948B	2	10	7	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Accumulator Injection (255, 1296)	Accumulator to Check Valve 8948C	3	10	7	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Loop 4 only	Yes
Accumulator Injection (236, 1297)	Accumulator to Check Valve 8948D	4	10	5	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Loop 3 only	Yes
Residual Heat Removal Supply (109, 927)	Valve 8702 to Cont. Pen. 27	4	14	5, 4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes
Residual Heat Removal Return (2576, 120)	Accum. Disch. to Cont. Pen. 26	3	8, 14	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Note 4
Residual Heat Removal Return (2575, 120)	Accum. Disch. to Cont. Pen. 26	4	8, 12	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Note 4
Low Head Safety Injection (235)	Loop Isol. Valve to Cont. Pen. 24	-	6	-	Yes												Note 2
Low Head Safety Injection (236)	Loop Isol. Valve to Cont. Pen. 24	-	6	-	Yes												Note 2
Low Head Safety Injection (237)	Loop Isol. Valve to Cont. Pen. 25	-	6	-	Yes												Note 2
Low Head Safety Injection (238)	Loop Isol. Valve to Cont. Pen. 25	-	6	-	Yes												Note 2
Pressurizer Relief Line (17)	From 4X6 to Header (23)	-	6	-	No	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 3
Pressurizer Safety Lines (19, 20, 21)	Safety Valves to Header (23)	-	6	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 3

Table 3.6-1 Check List of Dynamic Effects From Postulated Rupture of Pipe Connected to the Reactor Coolant System

Small Areas (≤4" I.D.) Resulting in a Loss of Coolant	Location Description	Loop	Size	No. of Rupture Restraints	Is the Containment Liner Protected?	Is Break Propagation in the Affected Leg Limited to 4" I.D. (12.5 in^2)?	Is Break Propagation in the Other Leg of the Affected Loop Prevented?	Is Break Propagation in the Unaffected Loops Prevented?	Is Damage to Steam System Prevented? Steam Line	Feed Line	Aux. Feed	Blow-Down	Level Taps	Sample Line	Is High Head Safety Injection Line Integrity Maintained to Avoid Spilling Line? If Not How Is High Head Safety Injection Maintained?	Is Line Restrained to Meet Criteria with Regulatory Guide 1.46 as a Minimum?
Pressurizer Spray (12, 15)	Cold Leg to Pressurizer	1	4	3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pressurizer Spray (14)	Cold Leg to Pressurizer	2	4	2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
RTD Bypass	Hot-Cold Leg to Pump Suction			-												Note 5
RTD Bypass	Hot-Cold Leg to Pump Suction			-												Note 5
RTD Bypass	Hot-Cold Leg to Pump Suction			-												Note 5
RTD Bypass	Hot-Cold Leg to Pump Suction			-												Note 5
Charging (50)	Check Valve 8379A to RC Piping	3	3	1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Loop 4 only	Yes
Charging (246)	Check Valve 8379B to RC Piping	4	3	1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Loop 3 only	Yes
Auxiliary Spray (51)	Check Valve 8377 to Line (15)	-	2	-	Yes	Yes	-	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
RC Pump Seal Water Injection (54)	Check Valve 8368 to RC Pump	1	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
RC Pump Seal Water Injection (55)	Check Valve 8368 to RC Pump	2	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
RC Pump Seal Water Injection (56)	Check Valve 8368 to RC Pump	3	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
RC Pump Seal Water Injection (57)	Check Valve 8368 to RC Pump	4	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
Letdown (24)	RC Piping to Support Downstream of Valve LCV 459	2	3	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
Excess Letdown (61)	RC Piping to Support Downstream of Valve 8167	2	1	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
RC Seal Vent (1495)	Pump to Valve			-												Note 3
RC Seal Vent (1496)	Pump to Valve			-												Note 3
RC Seal Vent (1497)	Pump to Valve			-												Note 3
RC Seal Vent (1498)	Pump to Valve			-												Note 3
RC Leakoff (58)				-												Note 3
RC Leakoff (59)				-												Note 3
RC Leakoff (60)				-												Note 3
RC Leakoff (61)				-												Note 3
Drain 958	RC Piping to Downsteam of Valve	1	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
Drain 959	RC Piping to Downstream of Valve	2	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
Drain 960	RC Piping to Downstream of Valve	3	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
Drain 961	RC Piping to Downstream of Valve	4	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
Pressurizer Relief Lines (1171, 1172, 1195)	1171 Take Off to Valve (N.C.)	3	2	2 each at valves	Yes	-	-	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Table 3.6-1 Check List of Dynamic Effects from Postulated Rupture of Pipe Connected to the Reactor Coolant System

Small Break (≤4" I.D.) not Resulting in a Loss of Coolant	Location Description	Loop	Size	No. of Rupture Restraints	Is a Loss of Coolant Prevented From Occurring?	Is Damage to Steam System Prevented?						Is Boration Capability Maintained?	Is Cold Shutdown Capability Maintained		Is Line Restrained to Meet Criteria with Regulatory Guide 1.46 as a Minimum?
						Steam Line	Feed Line	Aux. Feed	Blow-Down	Level Taps	Sample Line		RHR Suction	RHR Return	
Charging High Head SIS (50, 49)	Upstream of Check Valve 8379A	3	3	1	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	Note 4
Charging High Head SIS (246)	Upstream of Check Valve 8379B	4	3	1	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	Note 4
Auxiliary Spray (51)	Charging Line to Isolation Valve	-	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 4
RC Pump Seal Water Injection (54)	Upstream of Check Valve	1	-	-	-	-	-	-	-	-	-	-	-	-	Note 3
RC Pump Seal Water Injection (55)	Upstream of Check Valve	2	-	-	-	-	-	-	-	-	-	-	-	-	Note 3
RC Pump Seal Water Injection (56)	Upstream of Check Valve	3	-	-	-	-	-	-	-	-	-	-	-	-	Note 3
RC Pump Seal Water Injection (57)	Upstream of Check Valve	4	-	-	-	-	-	-	-	-	-	-	-	-	Note 3
Letdown (24)	Beyond Restraint Down-stream of Isolation Valve	2	3	2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Excess Letdown (63)	Beyond Restraint Down-stream of Isolation Valve	2	1	2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 6
Pressurizer Relief Lines (1171, 1172, 1195)	Downstream of Valves (N.C.)	-	3	1 ea.	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

( ) = PG&E Line Number

Note 1 - Affected area limited by enclosure.

Note 2 - Operates only during the injection and recirculation phase following a LOCA. Rupture not postulated.

Note 3 - Due to pressure and flow conditions in these pipes during operation of these lines, whipping is not assumed to occur.

Note 4 - Whipping allowed with no services affected.

Note 5 - Blowdown resulting from pipe rupture is less severe than from a rupture directly adjacent to reactor coolant pipe and can be handled by high-head safety injection.

Note 6 - Whipping allowable with no services affected with restraints at valves to prevent LOCA.

TABLE 3.6-2 CHECKLIST OF DYNAMIC EFFECTS OF OTHER POSTULATED PIPE RUPTURES INSIDE THE CONTAINMENT

Steam System Break not Resulting in a Loss of Coolant	Location Description	Loop	Size	No. of Rupture Restraint	Is Loss of Coolant Prevented from Occurring?	Is Break Propagation Prevented to Steam Piping in the Unaffected Loops?	Is Safety Injection Maintained to all R.C. Loops?	Is the Containment Liner Protected?	Is Integrity of the Steam Generator Supports Maintained?	Is Boration Capability Maintained?	Is Cold Shutdown Capability Maintained?	Is Line Restrained to Meet Criteria of Regulatory Guide 1.46 as a Minimum?
Main Steam Line (225)	From Steam Generator Nozzle to Containment Penetration	4	28	5	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Main Steam Line (226)	From Generator Nozzle to Containment Penetration	3	28	5	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Main Steam Line (227)	From Steam Generator Nozzle to Containment Penetration	2	28	5	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Main Steam Line (228)	From Steam Generator Nozzle to Containment Penetration	1	28	5	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Feed Water Line (554)	From Steam Generator Nozzle to Containment Penetration	1	16	4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Feed Water Line (555)	From Steam Generator Nozzle to Containment Penetration	2	16	4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Feed Water Line (556)	From Steam Generator Nozzle to Containment Penetration	4	16	4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Feed Water Line (557)	From Steam Generator Nozzle to Containment Penetration	3	16	4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Steam Generator Blow Down Line (1059)	From Steam Generator Nozzle to Containment Penetration	1	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Blow Down Line (1060)	From Steam Generator Nozzle to Containment Penetration	2	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Blow Down Line (1061)	From Steam Generator Nozzle to Containment Feneration	3	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Blow Down Line (1062)	From Steam Generator Nozzle to Containment Eneration	4	2	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Drain Line (1055)	From Steam Generator Nozzle to Blowdown Line	1	1	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Drain Line (1056)	From Steam Generator Nozzle to Blowdown Line	2	1	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Drain Line (1057)	From Steam Generator Nozzle to Blowdown Line	3	1	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1
Steam Generator Drain Line (1058)	From Steam Generator Nozzle to Blowdown Line	4	1	-	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Note 1

( ) IG&E Line Number

NOTE 1: In case of pipe rupture, pipe is allowed to whip with no services affected.

Nuclear Services Corporation

APPENDIX 3.6A

INTERIM REPORT

EVALUATION FOR EFFECTS OF

POSTULATED PIPE BREAK OUTSIDE CONTAINMENT

FOR

DIABLO CANYON UNIT 1

APPENDIX 3.6

EVALUATION FOR EFFECTS OF POSTULATED PIPE BREAK OUTSIDE CONTAINMENT

MAY 10 1974

Revision 1

April 12, 1974

PGE-01-02

INTERIM REPORT
EVALUATION FOR EFFECTS OF
POSTULATED PIPE BREAK OUTSIDE CONTAINMENT
FOR
DIABLO CANYON UNIT 1

Prepared for

Pacific Gas and Electric Company

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

Prepared by: D. P. Munson
D. P. Munson

V. Pichai
V. Pichai

C. E. Willey

C. R. Pignoli

A. J. Elliott
A. J. Elliott

H. Durlofsky

Approved by: G. A. Randall
G. A. Randall

Issued by:
R. E. Keever

Date: 5-6-74

MAY 10 1974

TABLE OF CONTENTS

		Page
1.0	SUMMARY	3.6A-7
2.0	INTRODUCTION	3.6A-8
3.0	PIPE RUPTURE CRITERIA	3.6A-11
4.0	PLANT DESCRIPTION	3.6A-14
5.0	IDENTIFICATION OF HIGH ENERGY SYSTEMS	3.6A-17
	5.1 Systems Containing High Energy Fluid	3.6A-17
	5.2 Systems in Which Design Basis Breaks are Postulated	3.6A-18
	5.3 Systems in Which Open Cracks are Postulated	3.6A-19
6.0	IDENTIFICATION OF ESSENTIAL STRUCTURES AND EQUIPMENT	3.6A-20
	6.1 Normal Shutdown	3.6A-20
	6.2 Emergency Shutdown	3.6A-21
	6.2.1 Emergency Shutdown with a Main Steam Line Rupture	3.6A-22
	6.2.2 Emergency Shutdown with a Feedwater Rupture	3.6A-23
	6.3 Essential Equipment Required for a Safe Shutdown	3.6A-24
	6.3.1 Main Steam Line Rupture	3.6A-24
	6.3.2 Feedwater Line Rupture	3.6A-33
	6.4 Equipment Belonging to the Reactor Protection System	3.6A-33
	6.5 Other Vital Equipment	3.6A-34
7.0	PIPE RUPTURE ANALYSIS	3.6A-36
	7.1 Main Steam Piping	3.6A-37
	7.1.1 Determination of Postulated Break Locations	3.6A-37
	7.1.2 Vulnerability of Essential Equipment to Damage	3.6A-39
	7.1.3 Pipe Whip Analysis	3.6A-45
	7.1.4 Jet Effects	3.6A-47
	7.1.5 Compartment Pressurization	3.6A-51
	7.1.6 Flooding Analysis	3.6A-54
	7.1.7 Environmental Effects	3.6A-54
	7.1.8 Building and Structural Evaluation	3.6A-55
	7.2 Feedwater Piping	3.6A-57
	7.2.1 Determination of Postulated Break Locations	3.6A-57
	7.2.2 Vulnerability of Essential Equipment to Damage	3.6A-58
	7.2.3 Pipe Whip Analysis	3.6A-59
	7.2.4 Jet Effects	3.6A-60

MAY 10 1974

Page

	7.2.5	Compartment Pressurization	3.6A-60
	7.2.6	Flooding Analysis	3.6A-61
	7.2.7	Environmental Effects	3.6A-63
	7.2.8	Building and Structural Evaluation	3.6A-63
8.0	REFERENCES		3.6A-84
	APPENDIX A	Thermal-Hydraulic Analysis	3.6A-89
	APPENDIX B	Pipe Whip Analysis	3.6A-149
	APPENDIX C	Jet Effects Analysis	3.6A-187
	APPENDIX D	Structural Analysis	3.6A-221

1.0 SUMMARY

This interim report, prepared for the Pacific Gas and Electric Company, presents the results to date of the investigation being conducted to determine the consequences of a postulated rupture in high-energy fluid piping outside the containment at Diablo Canyon Unit 1. The investigation is being conducted in accordance with, and in response to, the Atomic Energy Commission letter to Mr. Frederick T. Searls, Vice President and General Counsel for the Pacific Gas and Electric Company, dated December 18, 1972. Included within this report are the definition of criteria and methods being employed in the analyses, the identification of the high-energy fluid piping outside containment and the structures and equipment essential for safe shutdown following a postulated rupture of this piping. Also given is a summary of analyses results for a break in the main steam piping between the containment and the turbine stop valves and a break in the feedwater piping between the containment and the feedwater pumps, including proposed design modifications. Of all the high-energy piping outside containment, this section of piping, because of its size and location, is considered to possess the most potential for damage. Pipe break effects analyzed include pipe whip, jet impingement, pressurization of compartments, water flooding and the environmental effects of pressure, temperature and humidity.

2.0 INTRODUCTION

In the letter (Reference 1) to the Pacific Gas and Electric Company concerning Diablo Canyon Reactor Units 1 and 2, the Atomic Energy Commission has requested that they be provided with analyses and other relevant information needed to determine the consequences of a postulated rupture, outside containment, of a pipe containing a high-energy fluid. In response to this request, Nuclear Services Corporation, in assistance to the Pacific Gas and Electric Company, is conducting an investigation of all high energy piping systems outside the containment to obtain the necessary information. The investigation is being conducted in the following steps:

A. The piping outside containment which contain high-energy fluid are first identified.

B. Those structures, systems, and components necessary to assure that a safe shutdown condition of the reactors can be accomplished and maintained following a postulated pipe rupture in any of this high-energy fluid piping are also identified.

C. Those piping runs of high-energy fluid systems which are not isolated from, or physically separated enough to preclude potential damage to, the structures, systems, or components established as important to safety for a rupture in that run are then identified for further analyses.

D. For each of these identified piping runs, locations are established for postulated pipe breaks.

E. For those piping runs in which design basis breaks are postulated, the appropriate analyses are performed to define the potential damage from effects of the break. Effects being considered include pipe whip, jet impingement, pressurization of compartments, water flooding and the environmental effects of pressure, temperature and humidity.

F. For those piping runs in which open cracks are postulated, analyses are performed to define the potential damage from the environmental effects.

G. Analyses of those affected structures, systems and components established as important to safety are then performed to evaluate the extent of this potential damage, and to determine if the damage can be accommodated without impairment of required function.

H. Where the results of the analyses show that the required functions of the necessary structures, systems and components cannot be assured, design modifications are proposed to provide protection.

This report defines the criteria being applied to the analyses, the methods employed, and a summary of the results of analyses completed to date. The piping systems outside containment containing high-energy fluid have been identified, and are discussed in Section 5. The systems found to contain the highest energy are the main steam and feedwater systems. Identification of the structures and equipment necessary to mitigate the consequences of a rupture in these systems is given in Section 6. The portions of the Unit 1 main steam and feedwater piping judged to present the most potential for damage to necessary structures and equipment are between the containment and the anchor in the turbine building at column line G. A summary of analytical results for a break in these lines is presented in Section 7. A description of the respective analyses performed, the criteria employed and typical results obtained, including proposed design modifications, are given in the Appendices.

Analyses on the remaining portions of the main steam and feedwater systems are currently in progress with typical results to date also included in this report. Final results of these analyses along with results of the other high-energy fluid systems of lower consequence will be presented in later reports.

3.0 PIPE RUPTURE CRITERIA

The chief criterion and basic objective of this investigation is to assure that the Diablo Canyon nuclear plant is designed so that the reactors can be shutdown and maintained in a safe shutdown condition in the event of a postulated rupture, outside containment, of a pipe containing a high-energy fluid, including the double-ended rupture of the largest pipe in the main steam and feedwater systems. The guidelines and criteria to be employed in the achievement of this objective are specified in the initial letter (Reference 1) from the AEC of December 18, 1972, to Mr. Frederick T. Searls, which included the attachment listing the general information required for consideration; the follow-up letter (Reference 2) from the AEC of January 29, 1973, which included an errata sheet for this attachment; "Structural Design Criteria for Evaluating the Effects of High-Energy Pipe Breaks on Category I Structures Outside Containment"; Document (B), Structural Engineering Branch of the Directorate of Licensing, which was enclosure 3 to the letter (Reference 3) from the AEC of August 13, 1973; and the project FSAR (Reference 4). The specifics of these guidelines and criteria as they are being employed in this investigation of Diablo Canyon are set forth in detail in Reference 5 and in other sections of this report.

Basically, the criteria may be separated into two types: those which determine the system, types of breaks, and locations of breaks to be analyzed; and those which state requirements for protection against the consequences of the selected breaks. The criteria for determining the systems to be considered, and the locations of postulated breaks in those systems, are given in detail in Section 5 and 7, respectively. Some of the more important criteria applied in the determination of the acceptability of the consequences of a break at these locations are:

- The most consequential single failure of an active component is assumed to occur coincidentally with the damage resulting from the effects of the pipe break.

- Concurrent loss of off site power is assumed.

- Required redundancy must be preserved in those protection systems, electrical systems, engineered safety feature equipment, and their interconnecting cables and cable penetrations necessary to mitigate the consequences of the pipe break and place the reactor in a safe shutdown condition.

- For those environmentally induced failures caused by an open crack, or a design basis pipe break which would not of itself result in protective action but does disable protection functions, a loss of redundancy is acceptable but the minimum required function of the system must be preserved.

- The failure of any structure caused by the pipe break must not damage any structure containing safety and shutdown equipment to the extent that the capability to mitigate the consequences of the pipe break and bring the reactor to a safe shutdown condition is impaired.

- The integrity of the containment must be preserved.

- The integrity of the control room must be preserved.

- The ability to bring the reactor to a safe shutdown condition from the control room or from outside the control room must be preserved.

4.0 PLANT DESCRIPTION

Diablo Canyon Unit 1 is a 4-loop, pressurized water reactor plant located on a coastal site near San Luis Obispo, California. A nearly identical Unit 2 is being built on the same site contiguous to Unit 1. Consequently, some facilities are shared between the two plants. A general area layout of Unit 1, with the match line between units, is shown in Figure 4-1. As indicated on this figure, the plant is divided into areas, with each area given an alphabetical designation. Areas A, B, C, and D comprise the Unit 1 turbine building, with area E being shared between the two units. Areas H and K make up the shared auxiliary building. The turbine building houses the turbine generator, main condenser, pumps, feedwater heaters, other related main power loop equipment, and some station auxiliary equipment. The auxiliary building houses the control room, ventilation equipment, battery room, safeguards instrument room, and other related safety and auxiliary equipment. The general layout of equipment in these buildings is shown in Figures 1.2-4 through 1.2-27 of the FSAR (Reference 4).

A number of high-energy fluid piping systems outside the containment have been identified for Unit 1 and are discussed in Section 5 of this report. Of these systems, the main steam and feedwater lines, because of their large sizes and high pressures, will present the greatest potential for damage from pipe rupture.

Two main steam and two feedwater lines exit the north side of the containment and run along the containment exterior to area GW. These lines are carried in a structural steel pipeway which acts as a rupture restraint and pipe support. The other two sets of steam and feedwater lines leave the south side of the containment within areas GE and GW and turn west to enter the turbine building. These lines are restrained against pipe whip by several rupture restraints within areas GE and GW. From the anchor at column line G, the steam and feedwater lines run into areas B, C, and D of the turbine building. Other high-energy fluid piping lines run through all of these areas as well as portions of areas H and K in the auxiliary building.

MAY 10 1974

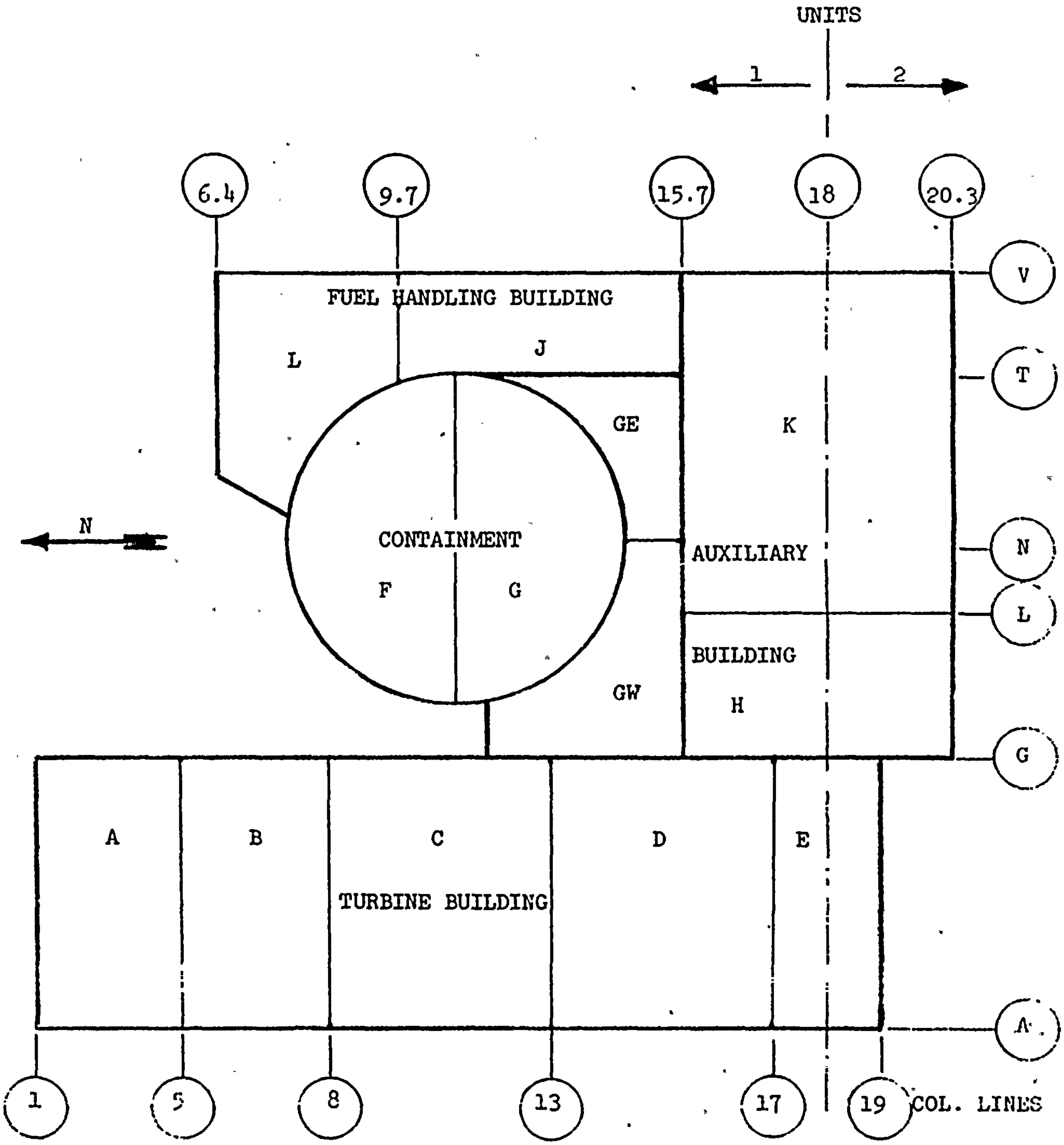


FIGURE 4-1. AREA DESIGNATION PLAN, DIABLO CANYON - UNIT 1

MAY 10 1974

5.0 IDENTIFICATION OF HIGH ENERGY SYSTEMS

5.1 Systems Containing High Energy Fluid

High energy piping systems are defined in Reference 1 as those which have a service temperature above 200°F or a pressure above 275 psig. The plant operational conditions under which this definition applies include reactor start-up, operation from hot standby to rated power and reactor cooldown to the cold shutdown condition.

The following systems at Diablo Canyon Unit 1 contain lines outside the containment which fall under this high energy piping definition:

Condensate System

Feedwater System, including

- Main Feedwater Piping
- Auxiliary Feedwater Piping

Turbine Steam Supply, including

- Main Steam Piping
- Steam Generator Blowdown Piping
- Steam to Auxiliary Feedwater Pump Piping

Extraction Steam and Heater Drip System

Auxiliary Steam System (Only temperature exceeds 200°F/275 psig criterion)

Chemical and Volume Control System

Safety Injection System

Residual Heat Removal System

Nuclear Steam Supply Sampling System

Component Cooling Water System

Fire Protection System (Only pressure exceeds 200°F/275 psig criterion)

Turbine and Generator Associated Systems

Nitrogen and Hydrogen Systems (Only pressure exceeds 200°F/275 psig criterion)

In accordance with the criteria of Reference 1, evaluation of breaks in the nuclear steam supply sampling system and the nitrogen and hydrogen system is not required as the portions of those lines which contain high energy fluids have a nominal pipe diameter less than or equal to one inch. In addition, the only portion of the fire protection system which contains a high energy fluid is of small size (diameter of 1-1/2 inches) and is physically remote from any safety related equipment (located in the turbine building near the intersection of column lines 16 and A). No further analyses are therefore required for these systems.

5.2 Systems in Which Design Basis Breaks are Postulated

Design basis breaks are postulated in that portion of those high energy systems where both the temperature and pressure exceed the 200°F and 275 psig limits. These systems, as defined in the plant piping and instrument diagrams, are:

MAY 10 1974

Condensate System

Feedwater System, including

- Main Feedwater Piping
- Auxiliary Feedwater Piping

Turbine Steam Supply System, including

- Main Steam Piping
- Steam Generator Blowdown Piping
- Steam to Auxiliary Feedwater Pump Piping

Extraction Steam and Heater Drip System

Chemical and Volume Control System

Safety Injection System

Residual Heat Removal System

Component Cooling Water System

Turbine and Generator Associated Systems

5.3 Systems in Which Open Cracks are Postulated

Open cracks are postulated in those systems where either the temperature or pressure exceeds the 200°F and 275 psig limits. Therefore, open cracks, as well as design basis breaks, are postulated in all of the systems identified in Section 5.2. In addition, open cracks are also postulated in the auxiliary steam system, as it contains piping where the fluid temperature exceeds the 200°F limit.

6.0 IDENTIFICATION OF ESSENTIAL STRUCTURES AND EQUIPMENT

Under conditions of a postulated main steam or feedwater pipe rupture, in order to effect and maintain the plant in a safe shutdown condition, it is necessary that the integrity of the containment and auxiliary building be preserved. The auxiliary building contains most of the engineered safeguards pumps and associated equipment,control room, batteries and vital AC and hot shutdown remote panel. Area A, which is at the northern end of the turbine building, houses the diesel generator units, 4.16 KV and 480 V buses and their switchgear and the cable spreading room. Area A is a Class 1 structure.

6.1 Normal Shutdown

The functions necessary for safe shutdown are available from instrumentation channels that are associated with the major systems in both the primary and secondary sides of the Nuclear Steam Supply System. There are no individually identifiable safe shutdown systems per se. However, prescribed procedures for securing and maintaining the plant in a safe condition can be instituted by appropriate alignment of selected fluid and control systems. The system functions required to be aligned for maintaining normal, controlled safe shutdown of the reactor are the minimum number that will:

1. Prevent the reactor from achieving criticality.

2. Provide an adequate heat sink such that design and safety limits are not exceeded.

MAY 10 1974

The designation of systems that can be used for safe shutdown depends on identifying those systems which provide the following capabilities for maintaining a safe shutdown condition:

1. Reactor coolant boration

2. Adequate supply for auxiliary feedwater

3. Residual heat removal

All systems required for a safe shutdown associated with these functions have been designed in accordance with the single failure criteria described in Reference 4, page 7.3-4.

6.2 Emergency Shutdown

The following subsections identify systems and vital equipment which are required to function for safe shutdown of the reactor consequent to rupture of main steam or feedwater lines outside the containment.

An emergency shutdown may be caused by a main steam or feedwater pipe rupture. In this context, it is assumed that an emergency mode of shutdown is necessary even though normal mode of shutdown may be sufficient. The main steam and feedwater pipe ruptures are analyzed independently and not postulated to occur concurrently. The events requiring initiation of Engineered Safety

Features are assumed to occur simultaneously with loss of offsite power. Therefore, the function of onsite power sources (diesel generators, batteries and vital instrument ac) must be preserved.

6.2.1 Emergency Shutdown with a Main Steam Line Rupture

The equipment necessary for a safe shutdown following a main steam line rupture is the same regardless of the pipe break location. The effect of one steam line break is the loss of one steam generator for reactor decay heat removal after reactor trip.

For a large steam line break the following must be available to accomplish safety functions:

A. Safety injection to pump borated water into the core, thereby limiting the core power transient following the break.

B. Isolation of main feedwater to the steam generators to limit the reactor coolant system cooldown.

C. Closure of main steam isolation valves to limit the reactor coolant system cooldown.

D. Auxiliary feedwater is required to dissipate decay heat. In the event of a concurrent loss of offsite power, at least one of the three auxiliary feed pumps would be required. This pump must be available and deliver full flow within one minute.

In order to cool the plant down to the residual heat removal system pressure, auxiliary feedwater must be available and the steam generator power-operated relief valves must be operable.

6.2.2 Emergency Shutdown with Feedwater Line Rupture

For a rupture of the main feedwater line between the containment and the feedwater check valve, which results in the blowdown of one steam generator, the following equipment must be available to accomplish safety functions:

A. Safety injection to pump borated water into the core, thereby limiting the core power transient following the break.

B. Closure of feedwater isolation valves.

C. Auxiliary feedwater to the intact steam generators within 10 minutes. For a large feedwater line break, there will be considerable water discharge from the affected steam generator. The water loss out of the steam generator lessens the plant's ability to dissipate decay heat. Furthermore, assuming the failure of one of the auxiliary feedwater pumps, most auxiliary feedwater will be discharged through the break until operator action is taken to re-align the auxiliary feedwater system. One-half of the auxiliary feedwater system capacity (provided by the two motor-driven or the one turbine-driven pump) must deliver full flow to the unaffected steam generators within 10 minutes.

In order to cool the plant down to the residual heat removal system pressure, auxiliary feedwater from at least one auxiliary feedwater pump must be available and the steam generator power operated relief valves must be operable.

For a large break between the feedwater pump and the main feedwater check valve, the feedwater line check valve will prevent water or steam release from any of the steam generators through the break. A large break at this point is, thus, essentially a loss of normal feedwater. In this case, the equipment which must be available to accomplish the safety function consists of the auxiliary feedwater system. At least one of the three auxiliary feedwater pumps must be available. Safety injection is not required for this case.

6.3 Essential Equipment Required for a Safe Shutdown

Based on the functional requirements of safe shutdown systems required for a main steam or feedwater pipe rupture as given in Section 6.2, the following essential equipment will be required for each function.

6.3.1 Main Steam Line Rupture

A. Equipment Required to Initiate Safety Injection

Following a steam line rupture, the Emergency Core Cooling System is automatically actuated to deliver borated water from the boron injection tank to the Reactor Coolant System. The response of the Emergency Core Cooling System following a steam line break is identical to its response during the injection mode of operation following a loss of coolant accident.

MAY 10 1974

The safety injection signal initiates the actions described for the injection mode of the loss of coolant accident, even though not all of these actions are required following a steam line rupture; e.g., the residual heat removal pumps are not required since the Reactor Coolant System pressure will remain above the pump shutoff head.

The delivery of the concentrated boric acid from the boron injection tank provides negative reactivity to counteract the increase in reactivity caused by the system cooldown. After the concentrated boric acid is delivered from the boron injection tank, the charging pumps continue to supply borated water from the refueling water storage tank, until enough water has been added to the Reactor Coolant System to make up for the shrinkage due to cooldown. The safety injection pumps also deliver borated water from the refueling water storage tank for the interval when the Reactor Coolant System pressure is less than the shutoff head of the safety injection pumps. After the pressurizer water level has been restored, the injection is manually terminated.

Equipment outside the containment required to initiate a safety injection, or be initiated by a safety injection signal, and its location is:

(1) Safety injection is initiated by high differential pressure between any two steam lines (besides other signals originating inside the containment). Steam line pressure is sensed by three pressure transmitters for each line which are located in mechanical panels in area GW at 115'

elevation and in area F outside the containment at about 110' elevation. The instruments and the instrument leads are vital equipment. The steam line pressure transmitters are:

Area F (north of containment)	Area GW
PT 514	PT 534
PT 515	PT 535
PT 516	PT 536
PT 524	PT 544
PT 525	PT 545
PT 526	PT 546
and their leads	and their leads

(2) Other safeguards and reactor protection signals originate inside the containment. Their cables penetrate through the containment in area GW between elevations 115' and 140', and are routed to the auxiliary building, area H.

(3) Safety injection signal automatically initiates the following actions:

(a) Starts the diesel generators numbers 1-1, 1-2 and 1-3. The diesel generators and their auxiliaries are located at the northern end of the turbine building in a class 1 structure (area A) at the 85' elevation. Area A also houses the vital

MAY 10 1974

cable spreading room at elevation 104' and the vital 4.16 KV busses F, G and H and their switchgear at elevation 119'.

(b) Starts the charging pumps number 1-1 and 1-2, the safety injection pumps number 1-1 and 1-2, and the residual heat removal pumps number 1-1 and 1-2. The charging pumps are located at elevation 73', the safety injection pumps at elevation 85' and the residual heat removal pumps at elevation 60' of the auxiliary building. The residual heat exchangers number 1-1 and 1-2 are located at elevation 73' of the auxiliary building. The pump motors are supplied from 4160V vital buses F, G or H. The power, control and indication circuits of these pumps are vital.

(c) Aligns the charging pumps and safety injection pumps for injection by closing or opening the valves associated with the charging pump, boron injection tank, volume control tank, and the refueling water storage tank. These valves are located in the auxiliary building and in areas GE/GW (above the 91' elevation and below the slab at 115' elevation) and at the refueling water tank. Therefore, these are at different locations and elevations from those of the main steam and feedwater piping.

(d) The charging pumps during the safety injection initiation are associated with volume control tank number 1-1 and boron injection tank number 1-1. These tanks are located at elevation 91' and 73' of the auxiliary building respectively.

(e) The component cooling water pumps 1-1, 1-2 and 1-3 and the auxiliary salt water pumps 1-1 and 1-2 are also initiated by a safety injection signal. The component cooling water pumps are located at elevation 73' of the auxiliary building. The auxiliary salt water pumps are located at the intake structure. The pump motors are supplied from 4160V vital buses F, G or H. The valves and instruments of the component cooling water heat exchangers number 1-1 and 1-2 and their associated electrical circuits are located at elevation 85' of the turbine building, area D. The component cooling water surge tank is located on top of the auxiliary building and the surge tank connection header lines run near the main steam and feedwater lines in areas GE, GW. The component cooling water system and the auxiliary salt water system are vital equipment required for safe shutdown of the plant.

B. <u>Equipment Required to Initiate Feedwater Isolation</u>

(a) Safety injection signal causes feedwater isolation and trips both main feedwater pumps. Each feedwater line has a motor-operated valve in series with a check valve and a flow control/isolation valve. These are located in area GW and area F outside the containment. The associated electrical circuits required to initiate these valves are located in their vicinity and are vital.

MAY 10 1974

The feedwater isolation valves are:

F.W. Line 1-1 (North of Containment)	F.W. Line 1-2 (North of Containment)	F.W. Line 1-3 (Area GW)	F.W. Line 1-4 (Area GW)
FCV-510 FCV-438	FCV-520 FCV-439	FCV-530 FCV-440	FCV-540 FCV-441

C. Equipment Required to Initiate Main Steam Isolation

1. High steam flow in two out of four lines in coincidence with either low-low primary coolant average temperature in two out of four lines or low steam pressure in two out of four lines; or

2. High-high containment pressure

The main steam line pressure instruments are located outside the containment and were covered in paragraph A, Equipment Required to Initiate Safety Injection. The other signals originate inside the containment and their circuits are routed through the electrical penetrations in area GW to the auxiliary building, area H.

The main steam line isolation valves are:

Main Steam Line 1-1 (North of Containment)	Main Steam Line 1-2 (North of Containment)	Main Steam Line 1-3 Area GW	Main Steam Line 1-4 Area GW
FCV-41	FCV-42	FCV-43	FCV-44

The associated control and indication circuits for these valves are vital.

MAY 10 1974

D. Auxiliary Feedwater Pump Automatic Initiation

The two motor-driven auxiliary feedwater pumps are fed from 4160V vital buses F and H. They are automatically initiated by any one of the following signals:

1. Steam generator low-low level (1 out of 4) signal or both main feedwater pumps tripped signal; not coincident with a blackout signal,

2. Safeguards sequence signal, or

3. Blackout signal.

The turbine-driven auxiliary feedwater pump is automatically initiated by either one of the following signals:

1. Steam generator low-low level (1 out of 4)

2. Vital bus undervoltage

In addition, the three pumps can be manually started for the purpose of plant startup or shutdown.

The three auxiliary feedwater pumps 1-1, 1-2 and 1-3 are vital equipment. They are located in area J, at elevation 100' of the auxiliary building. The motor-driven pumps 1-2 and 1-3 are supplied from 4160V vital buses H and F respectively. These switchgears and their control and indication circuits are vital. The turbine driver of pump 1-1 is supplied by steam control valves FCV-37 and FCV-38 from steam leads 2 and 3. These valves, their control and indication are vital equipment.

The following auxiliary feedwater control valves and their control circuitry are vital equipment.

F.W. Line 1-1 (North of Containment)	F.W. Line 1-2 (North of Containment)	F.W. Line 1-3 Area GW	F.W. Line 1-4 Area GW
LCV-106	LCV-107	LCV-108	LCV-109
LCV-110	LCV-111	LCV-113	LCV-115

The auxiliary feedwater flow elements FE-213, FE-214, FE-215 and FE-216 located in the area north of containment and in areas GE, GW are vital instruments required to modulate auxiliary feedwater flow.

The auxiliary feedwater pump discharge pressure instruments PT 432, PT 433 and PT 434 and flow instrument FI 11 and their circuits are vital equipment. The sensors are located in panel 31 in area J of the auxiliary building at elevation 100'. The pump discharge pressures are indicated in the main control room and hot shutdown panels. The pump discharge pressure signals are also used to control the auxiliary feedwater control valves LCV 110, LCV 111, LCV 113 and LCV 115. The auxiliary feedwater flow is indicated on panel 31.

MAY 10 1974

E. Vital Equipment in the Turbine Building

For a steam line or feedwater line rupture in the turbine building, the following vital equipment and electrical circuits must be evaluated for any adverse effects.

1. There are two main turbine first stage pressure signals which originate in the turbine building. The two pressure transmitters sensing these signals, PT 506 and PT 507 are located in mechanical panels at elevation 124' of the turbine building. Their associated electrical circuits are located below the floor at the 140' elevation. This signal is used to program the high steam line flow for actuation of safety injection and is also used for various reactor trip actuation signals.

2. Other vital circuits are also located in the turbine building and are routed below the floors at elevations 140', 119' and 105'. They perform the following functions:

   - Automatic initiation of safeguard equipment belonging to one train.

   - Control of the component cooling water valves and the auxiliary salt water system valves associated with the component cooling water heat exchangers.

MAY 10 1974

3. The component cooling water heat exchangers which are the ultimate heat sink for plant shutdown are located in area D near the east wall at elevation 85' of the turbine building.

6.3.2 Feedwater or Condensate Line Rupture

The systems and equipment required to function for this event are given in Section 6.2.2. The sequence of events and the equipment necessary to function following the accident (safety injection, feedwater isolation, and automatic initiation of the auxiliary feedwater pumps) have been described in Section 6.3.1, paragraphs A, B and D, respectively.

6.4 Equipment Belonging to Reactor Protection System

Equipment not belonging to the reactor safeguards system, but part of the reactor protection system is also considered vital. Equipment of this category in the vicinity of the main steam and feedwater lines includes:

- Feedwater flow instrumentation consisting of one set of flow transmitters on each line. These are located in mechanical panels in area GW and in panels in the area north of containment.

- Containment electrical penetraitons for vital circuits.

6.5 Other Vital Equipment

In addition to the equipment described in Section 6.3, the following equipment not in the vicinity of main steam or feedwater piping are vital for the functions required by Section 6.3:

A. Power Sources

In addition to the diesel generators, their auxiliaries and vital 4.16 KV switchgear already described, the following on-site power sources and their power, switchgear and control circuits are required:

(1) 480V AC load center and switchgear 1F, 1G, 1H, and 1-2 located at elevation 100' in area H of the auxiliary building.

(2) DC batteries 11, 12 and 13, their battery chargers, inverters, and panels, located at elevation 115', in area H of the auxiliary building.

B. Control Stations

(1) Control room and control room air conditioning at elevation 140' of the auxiliary building, area H.

(2) Hot shutdown remote panel at elevation 100' of auxiliary building, area H.

MAY 10 1974

(3) All required electrical power, control and instrumentation cables.

C. Ventilation

The ventilation systems for the engineered safeguard pumps and DC battery rooms which are located in the auxiliary building, area H.

D. Safeguards Systems Piping

The safeguards systems piping, valves and the valve control circuits located in the auxiliary building and in areas GE, GW below the slab at 115' elevation.

E. Auxiliary Lubrication Oil Pumps

The component cooling water pump auxiliary lubicating oil pumps 1-1, 1-2 and 1-3 and the charging pump auxiliary lubricating oil pumps 1-1 and 1-2 are vital for operation of the main pumps.

MAY 10 1974

7.0 PIPE RUPTURE ANALYSIS

Analysis to determine the consequences of a rupture of high energy piping on the ability of the plant to safely shut down from a normal operating or hot standby condition, and remain in the safe shutdown position, was conducted in the following manner. First, the high energy systems in the plant were identified according to the criteria presented in Section 5.0. Secondly, discrete locations at which design basis breaks were postulated to occur were selected in accordance with the criteria of References 1 and 2. At each break location, an evaluation was then conducted to determine if an unrestrained motion of the piping about its nearest hinge point (i.e. pipe whip) would result in an unacceptable degree of damage to equipment (including control and instrumentation conduits) required for a safe shutdown. At those break locations where such unacceptable damage could result from an unrestrained pipe whip, a dynamic nonlinear analysis of the piping and piping restraint system was conducted to determine if the pipe whip could actually occur. In some cases, modifications to existing piping restraints and addition of new restraints have been proposed to prevent such whip.

Jet impingements which result from fluid discharging through the pipe opening were also evaluated with regard to overall plant safety. Such jets can result in loadings on mechanical equipment, electrical conduits, other piping, and building structure. They also may cause pressurization of building compartments, floor flooding, and high temperature/high humidity environments in which safety equipment would be required to function.

Finally, building primary structures (i.e. shear walls, floors, containment, etc.) were evaluated to ensure structural integrity and conformance with the structural loading combinations specified by Reference 3.

7.1 Main Steam Piping

7.1.1 Determination of Postulated Break Locations

7.1.1.1 Design Basis Breaks

In accordance with Reference 1, as the main steam piping is greater than four inches in diameter, design basis breaks are postulated in both the longitudinal and circumferential directions. For each type, the break area is equal to the flow area of the pipe. Longitudinal breaks can occur at any orientation around the pipe circumference, with a break length of two times the pipe diameter. Circumferential breaks separate the piping axially, with each portion of the severed piping moving independently. Design basis breaks at branch points are assumed to be circumferential in branch lines and longitudinal in run lines, with the break area equal to the flow area of the branch.

The criteria used for selection of design basis break locations are as follows:

A. Postulate breaks at all terminal points (anchors or rigid equipment).

B. Postulate breaks at all branch points (terminal point of the branch line).

C. Postulate intermediate breaks between terminal points whenever primary stress (pressure, weight, DE) plus secondary stress (thermal) exceeds 80% of (S_h + S_A) or when secondary stress alone exceeds 80% of S_A, where S_h and S_A are defined in ANSI B.31.1 1967. The summation of stresses, Pressure + Weight + DE + Thermal, is referred to as "combined stress" in this report.

D. As a minimum, two intermediate breaks are selected at locations of highest stress.

E. For those piping runs where the design earthquake (DE) stress analyses are not available, breaks are postulated to occur at each location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges and welded attachments.

To select intermediate pipe break locations for the portion of the main steam piping between the containment and the G row anchor, piping stresses are thus plotted versus locations along the pipe. The stress values at all locations are then compared to the Atomic Energy Commission stress limits discussed in Paragraph C, above. The resulting plots are shown in Figures 7-1 and 7-2. Stress information was taken from Reference 6. As can be seen from the stress plots, there were no points which exceeded the Atomic Energy Commission stress criteria discussed above. Therefore, intermediate break locations were determined by choosing the more highly stressed points.

The break locations thus selected for the main steam piping, from containment to G row anchor, are shown on the isometric of Figure 7-3. At each postulated break location, two node numbers are specified. The first number corresponds to that used by Pacific Gas and Electric Company in their thermal stress analysis, while the second number corresponds to that used by Nuclear Services Corporation for the pipe rupture analyses. These main steam break locations are also listed in Tables 7-1 through 7-4, together with the criteria used for their selection. Hereafter, all break locations will be referred to only by the node numbers used in the pipe rupture analyses.

Design earthquake stress analyses of the main steam piping are not available for those sections within the turbine building. Therefore, breaks are postulated to occur at each location of potential high stress or fatigue (as per paragraph E).

7.1.1.2 Crack Breaks

In addition to the design basis breaks defined above, crack breaks with a flow area equal to one-half the pipe diameter times one-half the pipe thickness were postulated at all potentially adverse locations for all piping.

7.1.2 Vulnerability of Essential Equipment to Damage

7.1.2.1 Equipment Near Main Steam Lines 1 and 2, Containment to G Row Anchor

Main steam lines 1 and 2 exit the north side of the containment at the 129' elevation (area F). They proceed to drop in elevation to 112'6" and then

MAY 10 1974

follow the side of the containment, first in a southwest, and then in a southerly direction back towards the turbine building. They penetrate the turbine building through column 15 wall, turn 90° and head west, and are anchored at G row. Equipment and structures necessary for a safe shutdown of the plant which could be adversely affected by a rupture of these lines have been identified as follows:

- Primary containment
- J wall
- Wall at 15^Z
- Feedwater isolation valves - lines 1 and 2
- Main steam relief and safety valves
- Main steam check/isolation valve assembly (on the unruptured line)
- Auxiliary feedwater lines
- Main steam lead 2 supply valve to auxiliary feedwater pump turbine
- Electrical conduits to main steam and feedwater isolation valves and to feedwater control valves
- Feedwater flow sensors
- Main steam pressure sensors
- Auxiliary feedwater control valves and their electrical circuits
- In addition to the above equipment, the main steam and feedwater piping, vital conduits, and auxiliary feedwater piping are all supported by a common restraint structure called the pipeway. Although a localized failure of the pipeway could be tolerated, gross damage which would result in its collapse could not.

The conservatisms which must be applied with regard to pipe trajectories and structural impact loads during an unrestrained pipe whip have led to the conclusion that such whips of main steam lines 1 and 2 from containment to G row anchor cannot be justified. Therefore, the piping restraint system must be adequately designed to prevent whips which would result from breaks at locations shown in Figure 7-3.

7.1.2.2 Equipment Near Main Steam Lines 3 and 4, Containment to G Row Anchor

Main steam lines 3 and 4 exit the containment in area GW at the 129' elevation. They traverse in a southerly direction towards the wall at 15^Z, turn 90° and run west, paralleling the 15^Z wall. Near column row K, they turn 90° and run north, make another 90° turn and penetrate through J wall into the pipeway area. Following another series of turns and a drop in elevation to 112'6", they run parallel to main steam lines 1 and 2 and are also anchored at G row. Equipment and structures necessary for a safe plant shutdown which could be adversely affected by a rupture of these lines are as follows:

- Primary containment
- J wall
- Interior columns
- Wall at 15^Z
- Floors at elevation 115' and 140'
- Feedwater isolation valves - lines 3 and 4

MAY 10 1974

- Feedwater control valves - lines 3 and 4
- Auxiliary feedwater - lines 3 and 4
- Main steam lead 3 supply valve to auxiliary feedwater pump turbine
- Conduits to main steam and feedwater isolation valves and to feedwater control valves
- Main steam pressure sensors
- Feedwater flow sensors
- Electrical conduits and penetrations into the containment
- Main steam isolation/check valve assembly (on the unruptured line)
- Auxiliary feedwater control valves and their electrical circuits.

In addition, if the wall at 15[7] was destroyed, the following equipment could be impaired.

- Cable spreading room
- Hot shutdown remote panel
- 480 Volt load centers 1F, 1G, 1H
- Lighting distribution panels 13, 14, and 15
- 480 Volt load center No. 1-2
- Control room
- Safeguards instrument room #1, Trains A and B
- Batteries, associated equipment and vital AC.

Schematics of this equipment in relationship to main steam leads 3 and 4 are shown in Figures 7-4 through 7-7.

Again, the conservatisms which must be applied if one allows the main steam piping to whip in area GW would show unacceptable damage to the above safety related equipment. Thus, the restraint system on main steam lines 3 and 4, from containment to G row anchor, must also be capable of preventing whips resulting from the postulated breaks shown in Figure 7-3.

7.1.2.3 Equipment Near Main Steam Lines, Turbine Building Area

Past the G row anchor, main steam lines 1, 2, 3 and 4 traverse across the turbine building in a west direction at the 112'6" elevation. Between rows C and E they turn and run vertically, turn again at elevation 144'6" and are anchored at the turbine stop valves. Equipment and structures necessary for a safe plant shutdown which could be adversely affected by a rupture of these lines are as follows:

- H wall
- Wall at column 17
- Turbine building plate columns along column lines A and G
- Floor at elevation 140'
- Component cooling water heat exchanger
- Essential identified electrical conduits (see section 6.3.1)

The relationship of the above structures and equipment is shown in the schematics of Figures 7-8 through 7-10.

MAY 10 1974

In addition to the above mentioned essential equipment and structures the diesel generators and cable spreading room are housed at the north end of the turbine building in a seismic category 1 structure, area A (see Figure 7-9). Thus any damage resulting from a pipe rupture in the turbine building should not propagate into area A and prevent these components from performing their minimum required function.

It is proposed to protect the above safety related structures and equipment from the effects of pipe whip and jet impingement by the following method:

A. Perform an evaluation of the turbine building to determine the amount of structural damage resulting from an unrestrained pipe whip that the building could sustain and still retain its capability to survive the double design earthquake and support the design live and dead building loads. Those structures thus found not to be necessary to demonstrate overall building structural integrity need not be protected from the effects of pipe whip.

B. At each postulated break location, determine if the trajectory and kinetic energy associated with an unrestrained pipe whip will result in a larger amount of structural damage than the building can withstand and still retain its overall integrity.

C. For those unrestrained pipe whips which thus show an unacceptable degree of damage to building structure or which would impact safety related equipment (i.e., the component cooling water heat exchanger) it is proposed to prevent this damage by the construction of rupture restraints. These restraints will either prevent the whip entirely, or control the whip such that it does not damage either safety related equipment or structures. In addition, it is proposed to construct impingement barriers where necessary to prevent an unacceptable degree of damage to the above mentioned structures and equipment.

Pipe rupture analyses in the turbine building are currently in progress based upon this method.

7.1.3 Pipe Whip Analysis

At each postulated break location on the main steam piping, an evaluation was made to determine which system operating condition would result in the most severe hydrodynamic blowdown force-time history. In each case, it was determined that the most severe condition was hot standby. At representative break locations along the piping, the resulting blowdown force-time histories were determined using computer program PRTHRUST (Reference 7). This program, along with the analytical methods used are described in Appendix A, Section A.2. Advantage was taken of system symmetry where applicable. Typical results of this analysis for the main steam piping are shown in Appendix A, Figures A-3, A-4, and A-5.

Pipe whip analyses of the main steam piping system resulting from ruptures at the identified break locations were conducted with the use of computer program PIPERUP (Reference 8). PIPERUP determines the dynamic nonlinear response of a piping and restraint system to the time dependent blowdown forces described above. A description of the methods of analysis incorporated in PIPERUP is provided in Appendix B, Section B.2. The mathematical models constructed of the main steam piping between the containment and G row anchor, for use in the pipe whip analyses, are shown in Appendix B, Figures B-1 through B-4 for lines 1 through 4, respectively. The mathematical model of the main steam piping in the turbine building is shown in Figure B-7. The results of the pipe whip analyses include reaction loads on the piping restraints; deflections, shears, and moments at each pipe node point; plus description of yielding and ultimate hinge formation in the piping if it occurs.

Advantage is taken of symmetry when possible. That is, if a break at one piping location is typical of another break at another location on either the same line, or a different line, only one analysis need be performed. Of course, this is only possible if the blowdown forces, piping geometry and characteristics, as well as restraint characteristics are all nearly identical.

Some typical results of this analysis for breaks on main steam lines 1 through 4, from containment to G row anchor, are shown in Appendix B, Section B.3. Included in the typical results tables of Appendix B are maximum values of piping displacements, moments, strains, and support reaction loads. The results of these analyses have determined that the location of the restraint system shown in Figures B-1 through B-4 is adequate to prevent all whips which could result from breaks at the identified locations.

Pipe whip analysis of the main steam piping in the turbine building is currently in progress using the method presented in Section 7.1.2.3.

7.1.4 Jet Effects

7.1.4.1 Main Steam Piping from Containment to G Row Anchor

The jet impingement evaluation of main steam lines 1 through 4, from containment to G row anchor has determined jet impingement loadings on safety related mechanical equipment, electrical and instrumentation conduits, other piping, and basic building structure (walls, columns, etc.). Detailed descriptions of the criteria and methods of analysis used in this evaluation are provided in Appendix C.

At several of the main steam design basis break locations, jet impingement evaluations showed unacceptable degrees of damage to safety related equipment. These locations were at:

A. Main steam line 1: breaks located on elbows from 1185 to 1180 and 1120 to 1115 (see Figure 7-3). Large potential damage to:

- Main steam and feedwater isolation valves
- Auxiliary feedwater lines
- Vital electrical and instrumentation conduits
- Feedwater control valves
- Main steam and feedwater flow sensors

B. Main steam line 2: Breaks located on elbows from 2190 to 2185 and 2080 to 2075 (see Figure 7-3). Large potential damage to:

MAY 10 1974

- Main steam and feedwater isolation valves
- Auxiliary feedwater lines and control valves
- Vital electrical and instrumentation conduits
- Feedwater control valves
- Main steam and feedwater flow sensors

C. Main steam line 3: Breaks located on elbow from 3180 to 3170. Large potential damage to:

- Feedwater control valves
- Vital electrical and instrumentation conduits
- Electrical penetrations and pullboxes on containment wall
- Main steam and feedwater isolation valves
- Auxiliary feedwater lines and control valves
- Main steam and feedwater flow sensors

D. Main steam line 4: Breaks located on elbow from 4070 to 4067: Large potential damage to:

- Feedwater control valves
- Vital electrical and instrumentation conduits
- Electrical penetrations and pullboxes on containment wall
- Main steam and feedwater isolation valves
- Auxiliary feedwater lines and control valves
- Main steam and feedwater flow sensors

MAY 10 1974

Rerouting of the above mentioned safety equipment to protect it by means of separation would be nearly impossible without major redesigns of the plant. Even then, the impingement effects associated with design basis breaks would only be lessened and not eliminated. As an alternate, the erection of impingement barriers to protect equipment routed along existing lines would be very difficult for two reasons: (1) the extremely large forces the impingement barriers must be designed to withstand (i.e. approximately 800 kips), and (2) the great areas over which the impingements can act.

The above difficulties associated with protecting safety related equipment from jet impingements demonstrate the necessity to limit the jet thrusts resulting from design breaks at several of the postulated break locations. It is proposed that this flow limitation be accomplished by means of sleeves which extend over the break location. These sleeves are shown and described in detail in Appendix D, Section D.3. The locations where these sleeves are to be placed are shown in Appendix B, Figures B-1 through B-4 for lines 1 through 4, respectively. The piping restraint systems shown in Figures B-1 through B-4 are designed to maintain the process pipe within the sleeve for either a longitudinal or circumferential pipe break within the sleeve. In this manner, the only jet impingement that can occur results from the fluid ejected in the small annular space between the process pipe and the sleeve. The resulting potential target areas and jet impingement loadings are, of course, much smaller.

Some typical results of the jet impingement analysis (targets and impingement loads) are given in Appendix C, Section C.4.1. Analyses to determine the overall system effects of these loadings are still in progress.

7.1.4.2 Main Steam Piping in Turbine Building

As described in Section 7.1.1, design basis breaks of the main steam piping in the turbine building are postulated to occur at each location of potential high stress or fatigue. Jet impingement evaluations at representative locations showed large potential damage to:

- 4 KV bus auto transfer circuit (train H), which automatically initiates safeguards equipment of one train
- Diesel generator 11 auto trip circuit
- Diesel generator auto start back-up circuit
- Component cooling water pump standby start circuit
- Component cooling water supply header valve control circuits
- First stage turbine pressure circuits
- Auxiliary feedwater pump auto start circuits on main feedwater pump trip
- Component cooling water heat exchangers and their associated valves

Construction of impingement barriers (and rupture restraints) to protect this equipment from design basis breaks of the main steam piping and breaks of other high energy lines in the vicinity would be impractical due to the large areas that will need protection. Therefore, it has been proposed to reroute

MAY 10 1974

most of the above vital circuits outside of the turbine building, protecting them by means of separation. However, as the first stage turbine pressure signal originates at the turbine stop valves, this circuit cannot be similarly separated. Instead, it has been proposed to reroute this circuit behind one of the turbine pedestals (see Figure 7-8) down to the 85' base floor slab, at which point it will be imbedded in the floor until its penetration into the auxiliary building. The turbine pedestal and the base floor slab will function as a combination impingement barrier and protective restraint.

As described in Section 7.1.2.3 it is proposed to protect the component cooling water heat exchanger, along with its associated electrical conduits, valves and piping from the effects of pipe whip by the use of selected rupture restraints and from the effects of jet impingements by the construction of an impingement barrier.

7.1.5 Compartment Pressurization

A rupture of main steam lines 1 and 2, between the containment and their entrance into the turbine building, would result in the steam being released directly into the atmosphere. Hence, no building compartments would be pressurized.

A rupture of main steam lines 3 and 4, between the containment and their penetration through J wall into the pipeway area, would occur in area GW at the 115' elevation. The design basis breaks at nodes 3180 and 3170 on line 3, and at nodes 4070 and 4067 on line 4 are sleeved (see Figures B-3 and B-4 for lines 3 and 4, respectively); hence steam can only flow in the

MAY 10 1974

small annular space between the sleeve and the process pipe. The resulting building pressurizations would be small. For each line, the two remaining design basis break locations within area GW are at the containment anchor (nodes 3295 and 4135 for lines 3 and 4, respectively) and at the relief valve header tee (nodes 3273 and 4115 for lines 3 and 4), see Figure 7-3. The circumferential breaks at the containment anchors are the more severe case as fluid can discharge from both sides of the severed piping simultaneously. Thus, it was the case analyzed.

Following a rupture of main steam line 3 or 4 at the containment anchor, steam will discharge from both ends of the severed piping into area GW. This will cause safety injection and main steam isolation signals, thereby closing the main steam isolation valves, feedwater motor operated isolation valves, and feedwater control valves. Steam flowing from the steam generator side of the break will continue until the feedwater lines are isolated and the residual water in the steam generator has flashed. Reverse flow steam coming from the turbine side of the break would normally close the check valve, thus cutting off the flow. However, postulating the worst case active component failure, the check valve is assumed not to close. Thus, steam from the turbine side of the break is assumed to continue until the main steam isolation valves on the three unbroken lines close, and the remaining residual steam in the lines flow out of the break.

Immediately following the break, steam discharged into area GW will result in a rapid pressurization of this compartment. Steam will also flow through

IAY 10 1974

the doorway into area GE, thus pressurizing it as well. It is proposed to prevent steam from entering the auxiliary building (area K) and the fuel handling building (area J) by means of closed doors. After a short time, venting from these compartments (most notably through the 98 ft^2 opening in J wall into the atmosphere) will exceed the steam discharged from the break, and the compartment pressures will fall.

Enthalpy and flow rates of the steam into each of the compartments and resulting pressures, are determined by computer program PRTHRUST (Reference 7). The resulting compartment pressurizations are verified by using a modified version of the computer program CONTEMPT (Reference 9). Detailed descriptions of the methods of analysis and values of data used are presented in Appendix A, Section A.3. The results of this analysis are also presented in Appendix A. The overall effects of these compartment pressures are evaluated in Appendix D, Section D.4, and were found not to damage any safety related structure to an unacceptable degree.

A rupture of the main steam piping within the turbine building would not result in any significant level of pressurization due to the extremely large volume of this compartment. At worst, any small overpressures that could be present would leak to the atmosphere through the wall siding. For this reason, no compartment pressurization analysis of the turbine building was deemed necessary.

MAY 10 1974

7.1.6 Flooding Analysis

Building flooding resulting from a rupture of the main steam piping can only occur from steam condensation within a building compartment. As the feedwater piping parallels the main steam, the resulting water levels would be bounded by breaks of the feedwater piping.

7.1.7 Environmental Effects

The purpose of the building environmental analysis is to determine the temperature and humidity response in compartments in which safety related equipment is required to operate following a process line rupture.

A rupture of main steam lines 1 or 2, between the containment and their entrance into the turbine building, would release the steam directly to the atmosphere. Therefore, operating conditions for safety related equipment would not be affected.

A rupture of main steam lines 3 or 4, between the containment and their penetration through J wall into the pipeway area, would result in higher than ambient temperature and humidity in areas GW and GE. For the same reason as discussed in Section 7.1.5 above, the most severe (i.e. highest temperature) of the design breaks in the GW area would occur at the containment anchors (nodes 3295 and 4135 for lines 3 and 4, respectively). The criteria and method of analysis used to determine these environments as a function of time are given in Appendix A, Section A.4. The results of

MAY 10 1974

these analyses are summarized below, under various postulated active component failures. Average temperature is defined as a quasi steady state value which is assumed to exist until the room returns to a 100°F temperature.

Assumed Active Component Failure	Average Temperature (°F)	Exposure Time (sec)
Main Steam Check Valve Failure	215	50
Feedwater Control Valve Failure	200	120
Postulated Break Underneath Sleeve with Feedwater Control Valve Failure	180	320

The maximum temperature determined within areas GE and GW was found to be 300°F, but this exists for less than 2 seconds. The relative humidity reaches 100% within 10 seconds, and will remain at this value over a long period of time.

The analysis to determine the capability of safeguard equipment to survive in this environment is currently in progress.

7.1.8 Building and Structural Evaluation

Analysis of the piping restraint system considers the combined effects of normal operating loads (i.e. deadload, etc.) in conjunction with seismic and accident loads (i.e. pipe impact) on each piping restraint. The loading combinations used are those specified by Reference 3. Evaluations are made on the restraint anchorage system, as well as on the restraint itself. Effects of buckling and instabilities are included. A more detailed description of this analysis is provided in Appendix D, sections D.2, D.3, and D.4.

The results of this analysis in area GW and the pipeway area are complete. They show that, based upon the proposed restraint modifications and additions given in section D.2 and D.4, the restraint system (including anchorages) will be adequately designed to withstand piping restraint loads resulting from breaks at the identified locations.

Analysis of building primary structure (i.e. walls, columns, containment, etc.) considers the combined effects of normal operation (i.e. live loads, deadloads, seismic) in conjunction with accident loads (i.e. jet impingements, restraint loads, compartment pressurization). The loading combinations used are those specified in Reference 3. A more detailed description of this criteria and method of analysis is provided in Appendix D, section D.4.

The results of the building structural analysis of areas GW, GE and the containment show that the building primary structure is adequately designed to withstand the structural loadings following a postulated pipe rupture between the containment and the G line anchor.

Analysis of the turbine building structure for a break of the main steam piping is currently in progress.

7.2 Feedwater Piping

7.2.1 Determination of Postulated Break Locations

7.2.1.1 Design Basis Breaks

Like the main steam piping, as the feedwater piping is greater than four inches in diameter, design basis breaks are assumed in both the longitudinal and circumferential directions. The criteria used for selection of design basis break locations is that given in section 7.1.1.1. Again, for the portion of feedwater piping between the containment and G row anchor, plots of piping stresses versus distance were used for the selection of intermediate break locations. The resulting plots of thermal and combined stresses for feedwater lines 1 through 4, from containment to G row anchor, are shown in Figure 7-11. As can be seen from these plots, there are no points between the terminal anchors which exceed the AEC stress criteria discussed in section 7.1.1.1. Therefore, intermediate break locations were determined by choosing the more highly stressed points.

The break locations thus selected for the feedwater piping, from containment to G row anchor, are shown on the isometric of Figure 7-12. These break locations are also listed in Tables 7-5 and 7-6, together with the criteria used for their selection.

MAY 10 1974

Similar to the main steam piping, breaks of the feedwater piping within the turbine building are postulated to occur at each location of potential high stress or fatigue.

7.2.1.2 Crack Breaks

In addition to the design basis breaks defined above, crack breaks with a flow area equal to one-half the pipe diameter times one-half the pipe thickness were postulated at all potentially adverse locations for all piping.

7.2.2 Vulnerability of Essential Equipment to Damage

7.2.2.1 Equipment Near Feedwater Lines 1 and 2

Feedwater lines 1 and 2 closely parallel main steam lines 1 and 2, all the way from containment into the turbine building. Therefore the safety related equipment and structures listed in Section 7.1.2.1 which could be adversely affected by a break of main steam lines 1 or 2, could also be adversely affected by a break of feedwater lines 1 or 2. Thus, the feedwater piping restraint system must be adequately designed to prevent whips which would result from breaks at the locations shown in Figure 7-12.

7.2.2.2 Equipment Near Feedwater Lines 3 and 4

Feedwater lines 3 and 4 also closely parallel main steam lines 3 and 4 in the GW area. Thus, the safety related structures and equipment identified in Section 7.1.2.2 could also be adversely affected by an unrestrained whip of feedwater piping in area GW. For this reason, feedwater lines 3 and 4 must be adequately restrained to prevent all whips resulting from the break locations shown in Figure 7-12.

7.2.2.3 Equipment Near Feedwater Piping, Turbine Building Area

The equipment and building structure identified in Section 7.1.2.3 as being located near the main steam piping, is also vulnerable to damage from pipe whip effects associated with a feedwater break within the turbine building. Thus the means of protection discussed in Section 7.1.2.3 also applies to the feedwater piping.

7.2.3 Pipe Whip Analysis

On the feedwater piping an evaluation was also made to determine which system operating condition would result in the most severe blowdown loads. Again, the hot standby condition was found to be the most severe case. Typical results of this analysis are also presented in Appendix A, Figures A-6, A-7, and A-8.

Pipe whip analyses of the feedwater piping resulting from ruptures at locations shown in Figure 7-12 were also conducted with the use of computer program PIPERUP. The mathematical models constructed of the feedwater piping between the containment and G row anchor for use in the pipe whip analyses are shown in Figures B-5 and B-6. These analyses have been completed, and show that the locations of restraints given on Figures B-5 and B-6 are adequate to prevent whips which could result from breaks at the identified locations. Typical results of these analyses are also given in Appendix B, Section B.3.

The mathematical model constructed of the feedwater piping in the turbine building is shown in Figure B-8. This analysis is currently in progress.

7.2.4 Jet Effects

Similar to the main steam system, jet impingement evaluation of the feedwater piping between containment and the G row anchor at several of the feedwater design basis break locations showed unacceptable degrees of damage to safety related equipment. Thus, it is also proposed to limit the jet thrust at several break locations by means of sleeves. The locations where these sleeves are to be placed are shown in Appendix B, Figures B-5 and B-6. For a design basis break within a sleeve, the restraint systems shown in Figures B-5 and B-6 are designed to maintain the process pipe within the sleeve.

Typical results of the feedwater jet impingement analysis (targets and impingement loads) are also given in Appendix C, Section C.4. Analyses to determine the overall system effects of these loadings are still in progress.

7.2.5 Compartment Pressurization

A rupture of feedwater lines 1 and 2, between the containment and their entrance into the turbine building, would result in the water (and flashed steam) being discharged directly into the atmosphere. Hence, no building compartments would be pressurized.

Within area GW, feedwater lines 3 and 4 closely parallel main steam lines 3 and 4. A rupture of the feedwater piping in GW would result in smaller compartment pressurization (than a rupture of the main steam piping) because of the smaller feedwater pipe size and lower fluid energy.

Similar to the main steam, a rupture of the feedwater piping within the turbine building would not result in any significant level of pressurization due to the large volume of this compartment.

7.2.6 Flooding Analysis

A rupture of feedwater lines 1 and 2, between the containment and their entrance into the turbine building would result in the water spilling directly onto the ground, just outside of containment area F. From this location, a portion of the water would flash, while the remainder would be absorbed into the soil and run off towards the ocean.

A rupture of feedwater lines 3 and 4 between the containment and their penetration through J wall into the turbine building would occur in area GW at the 115' elevation. Within area GW sleeves have been proposed for design basis breaks at nodes 1303, 1305, 1329, and 1331 on line 1 and at nodes 1403, 1405, 1429, and 1431 on line 2 (see Appendix B, Figure B-6). The majority of the water ejected in the small annular space between the process pipe and the sleeve would flash, with the resulting water levels on the floor being minimal. At branch break locations 1307 and 1407 (3"

MAY 10 1974

auxiliary feedwater inlet) and at 1347 and 1441 (6" bypass tees) water levels resulting from discharges through a broken branch would again be small due to the small flow area. Thus, the postulated break locations with the largest flooding potential for area GW (and connected areas) are at the containment anchor, nodes 1300 and 1400 for lines 3 and 4, respectively.

The criteria and method of analysis used to determine the resulting water levels and compartments affected are given in Appendix A, section A.5. The results of this analysis are shown in Figure A-16. As this plot shows, the peak water level in area GW at the 115' elevation is approximately 5 inches. Water also drains into area GE, but its peak level is less than 1 inch. Final drainage of the water is provided by a 13 foot wide opening in J wall, which will discharge the water directly onto the ground.

A rupture of the feedwater piping in the turbine building would result in essentially the same flooding conditions regardless of the break location. That volume of water from the steam generator side of the break between the check valves and the break opening will be acted upon by line pressure and discharged. Similarly, the water between the break and the feedwater pumps will also be discharged. The condensate pumps are conservatively assumed to run down until they drain the heater #2 drain tank and the hotwells. At this time they will be tripped due to a low pump suction head. Water thus discharged through the break will drain down to the 85' elevation

MAY 10 1974

floor, where it will drain into the turbine building sump volumes. This analysis showed that the total volume of water thus discharged is smaller than the turbine building sump volume, hence a flooding potential in this area does not exist.

7.2.7 Environmental Effects

A rupture of feedwater lines 1 and 2, between the containment and their entrance into the turbine building would release the fluid directly to atmosphere. Hence, no safety related equipment would be exposed to a resulting adverse environment.

A rupture of feedwater lines 3 and 4 in area GW or a rupture of the feedwater piping in the turbine building would be less severe, environmentally, than a rupture of the main steam piping due to the lower fluid discharge rate and lower fluid energy. Therefore, no additional environmental analysis is required.

7.2.8 Building and Structural Evaluation

The criteria and method of analysis used for evaluating the piping restraints and building primary structure resulting from loads imposed by ruptures of the feedwater piping is identical to that used for evaluation of main steam ruptures. It is summarized in Appendix D.

The results of this evaluation in area GW and the pipeway area have been completed and are also summarized in Appendix D. Briefly, this analysis has found that, based upon the proposed restraint modifications and additional restraints specified in Appendix D, the restraint system and building primary structure will satisfy the loading criteria specified by Reference 3.

PROJECT DIABLO CANYON UNIT 1

SUBJECT MAIN STEAM PIPING

TABLE 7-1

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM LINE 1 - CONTAINMENT TO G ROW ANCHOR

Location (NSC Node No.)	Location Type	Reason for Selection
1215	Anchor at Containment	Terminal point
1212	Relief valve outlet tee	Branch point
1210	Relief valve outlet tee	Branch point
1200	Relief valve outlet tee	Branch point
1195	Relief valve outlet tee	Branch point
1190	Relief valve outlet tee	Branch point
1185	Elbow	High thermal & combined stresses
1180	Elbow	High thermal & combined stresses
1120	Elbow	High thermal & combined stresses
1115	Elbow	High thermal & combined stresses
1060	Elbow	High combined stress
1055	Elbow	High combined stress
1052	PCV header tee	Branch point
1050	G Row Anchor	Terminal point
1315	Tee on PCV header	Branch point

PROJECT. DIABLO CANYON UNIT 1

SUBJECT MAIN STEAM PIPING

TABLE 7-2

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM LINE 2 - CONTAINMENT TO G ROW ANCHOR

Location (NSC Node No.)	Location Type	Reason for Selection
2225	Anchor at Containment	Terminal point
2220	Relief valve outlet tee	Branch point
2215	Relief valve outlet tee	Branch point
2205	Relief valve outlet tee	Branch point
2200	Relief valve outlet tee	Branch point
2195	Relief valve outlet tee	Branch point
2190	Elbow	High thermal stress
2185	Elbow	High thermal stress
2182	Steam to Aux FW pump outlet tee	Branch point
2080	Elbow	High combined & thermal stresses
2075	Elbow	High combined & thermal stresses
2015	Elbow	High combined stress
2010	Elbow	High combined stress
2005	PCV header tee	Branch point
2001	G row anchor	Terminal point
2245	Tee on PCV header	Branch point

MAY 10 1974

TABLE 7-3

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM LINE 3 - CONTAINMENT TO G ROW ANCHOR

Location (NSC Node No.)	Location Type	Reason for Selection
3295	Anchor at Containment	Terminal point
3282	Steam to Aux FW pump outlet tee	Branch point
3273	RV header tee	Branch point
3510	Elbow	High thermal stress
3540	Elbow	High thermal stress
3550	Relief valve outlet tee	Brnach point
3560	Relief valve outlet tee	Branch point
3570	Relief valve outlet tee	Branch point
3580	Relief valve outlet tee	Branch point
3590	Relief valve outlet tee	Branch point
3600	Relief valve outlet tee	Branch point
3180	Elbow	High thermal & combined stresses
3170	Elbow	High thermal & combined stresses
3140	Elbow	High thermal & combined stresses
3130	Elbow	High thermal & combined stresses
3080	Elbow	High thermal stress
3070	Elbow	High thermal stress
3060	PVC header tee	Branch point
3050	G Row Anchor	Terminal point
3430	Tee on PCV header	Branch point

PROJECT DIABLO CANYON UNIT 1

SUBJECT MAIN STEAM PIPING

TABLE 7-4

SUMMARY OF DESIGN BASIS BREAKS

MAIN STEAM LINE 4 - CONTAINMENT TO G ROW ANCHOR

Location (NSC Node No.)	Location Type	Reason for Selection
4135	Anchor at Containment	Terminal point
4115	RV header tee	Branch point
4140	Elbow	High thermal stress
4145	Elbow	High thermal stress
4165	Relief valve outlet tee	Branch point
4170	Relief valve outlet tee	Branch point
4175	Relief valve outlet tee	Branch point
4180	Relief valve outlet tee	Branch point
4185	Relief valve outlet tee	Branch point
4195	Relief valve outlet tee	Branch point
4070	Elbow	High combined & thermal stresses
4067	Elbow	High combined & thermal stresses
4055	Elbow	High combined & thermal stresses
4050	Elbow	High combined & thermal stresses
4025	Elbow	High thermal stress
4020	Elbow	High thermal stress
4005	PCV header tee	Branch point
4001	G Row Anchor	Terminal point
4015	Tee on PCV header	Branch point

MAY 10 1974

TABLE 7-5

SUMMARY OF DESIGN BASIS BREAKS

FEEDWATER LINES 1 AND 2, CONTAINMENT TO G ROW ANCHOR

Location (NSC Node No.)	Location Type	Reason for Selection
1100	Anchor at Containment	Terminal point
1103	3" Aux Feedwater tee	Branch point
1111	Elbow	High thermal & combined stresses
1113	Elbow	High thermal & combined stresses
1143	6" Bypass tee	Branch point
1151	Elbow	High thermal & combined stresses
1153	Elbow	High thermal & combined stresses
1163	Junction of FW1 with FW2	Branch point
1167	6" Bypass tee	Branch point
1375	Junction of FW3 with FW1	Branch point
1000	G Row Anchor	Terminal point
1200	Anchor at Containment	Terminal point
1203	3" Aux Feedwater tee	Branch point
1211	Elbow	High thermal stress
1213	Elbow	High thermal stress
1237	6" Bypass Tee	Branch point
1239	Elbow	High thermal & combined stresses

TABLE 7-5 (cont'd)

SUMMARY OF DESIGN BASIS BREAKS

FEEDWATER LINES 1 AND 2, CONTAINMENT TO G ROW ANCHOR

Location (NSC Node No.)	Location Type	Reason for Selection
1241	Elbow	High thermal & combined stresses
1255	6" Bypass tee	Branch point
1261	Elbow	High combined stress
1263	Elbow	High combined stress

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1

SUBJECT FEEDWATER PIPING

TABLE 7-6

SUMMARY OF DESIGN BASIS BREAKS

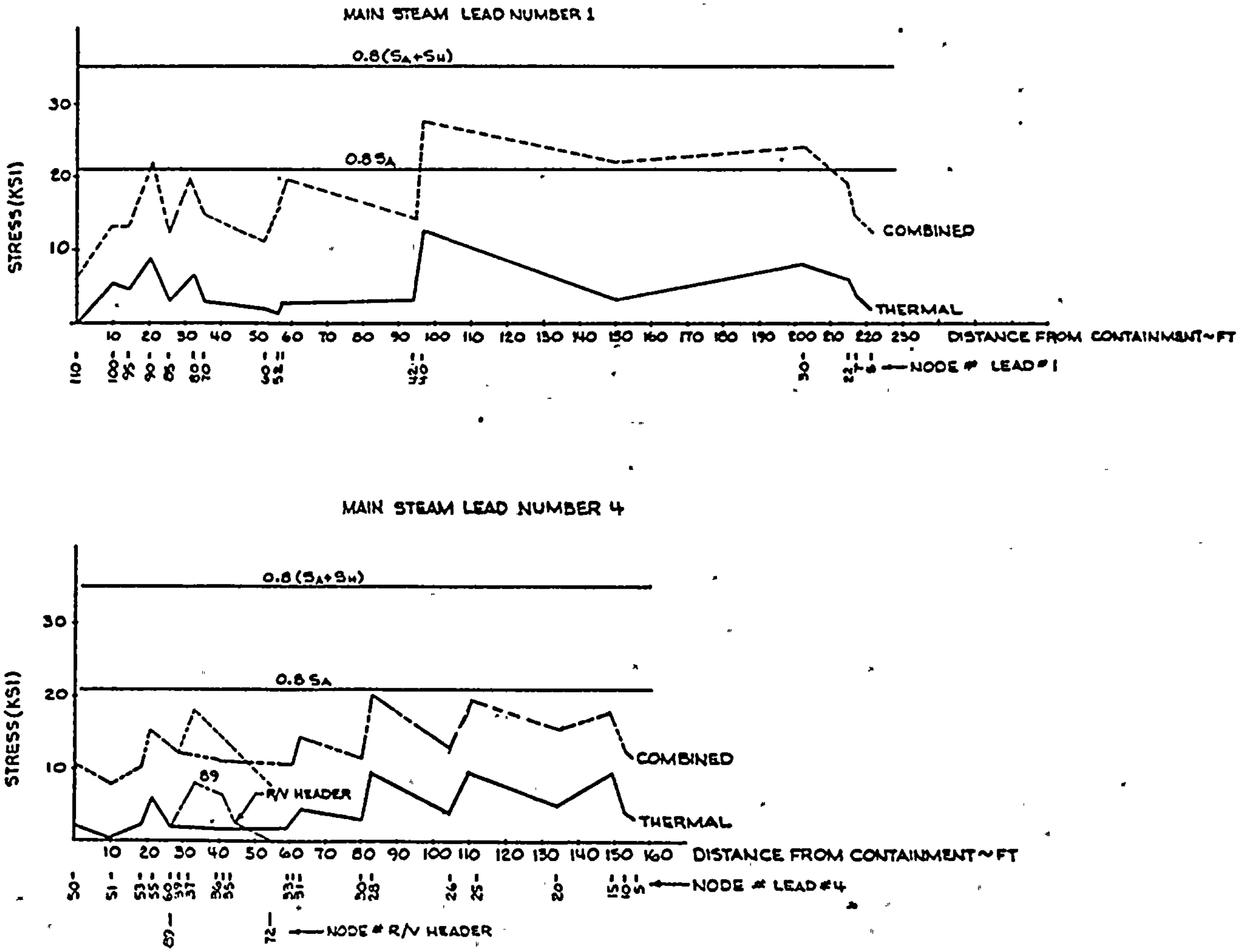
FEEDWATER LINES 3 AND 4, CONTAINMENT TO JUNCTION WITH FW1

Location (NSC Node No.)	Location Type	Reason for Selection
1300	Anchor at Containment	Terminal point
1303	Elbow	High thermal & combined stresses
1305	Elbow	High thermal & combined stresses
1307	3" Aux Feedwater tee	Branch point
1329	Elbow	High thermal & combined stresses
1331	Elbow	High thermal & combined stresses
1347	6" Bypass tee	Branch point
1367	6" Bypass tee	Branch point
1369	Junction of FW4 with FW3	Branch point
1375	Junction of FW3 with FW1	Branch point
1400	Anchor at Containment	Terminal point
1403	Elbow	High thermal & combined stresses
1405	Elbow	High thermal & combined stresses
1407	3" Aux Feedwater tee	Branch point
1429	Elbow	High thermal & combined stresses
1431	Elbow	High thermal & combined stresses
1441	6" Bypass tee	Branch point

MAY 10 1974

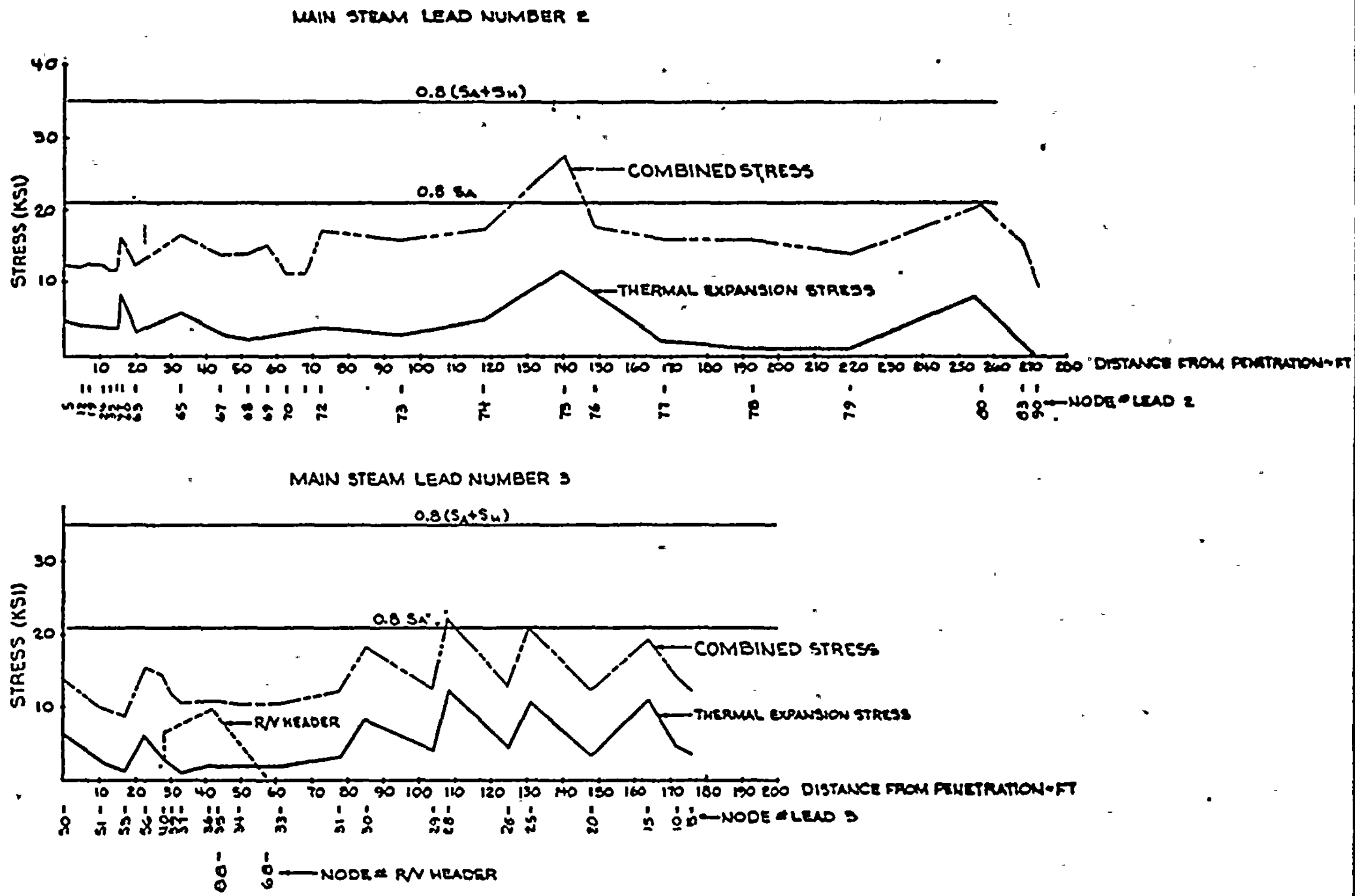
Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE 7-1
DIABLO CANYON UNIT 1
DETERMINATION OF POSTULATED BREAK LOCATIONS
MAIN STEAM PIPING - LINES 1 & 4
CONTAINMENT TO COLUMN LINE G



Nuclear Services Corporation

FIGURE 7-2.
DIABLO CANYON UNIT 1
DETERMINATION OF POSTULATED BREAK LOCATIONS
MAIN STEAM PIPING - LINES 2 & 3
CONTAINMENT TO COLUMN LINE 6



3.6A-73

MAY 10 1974

MAY 10 1974

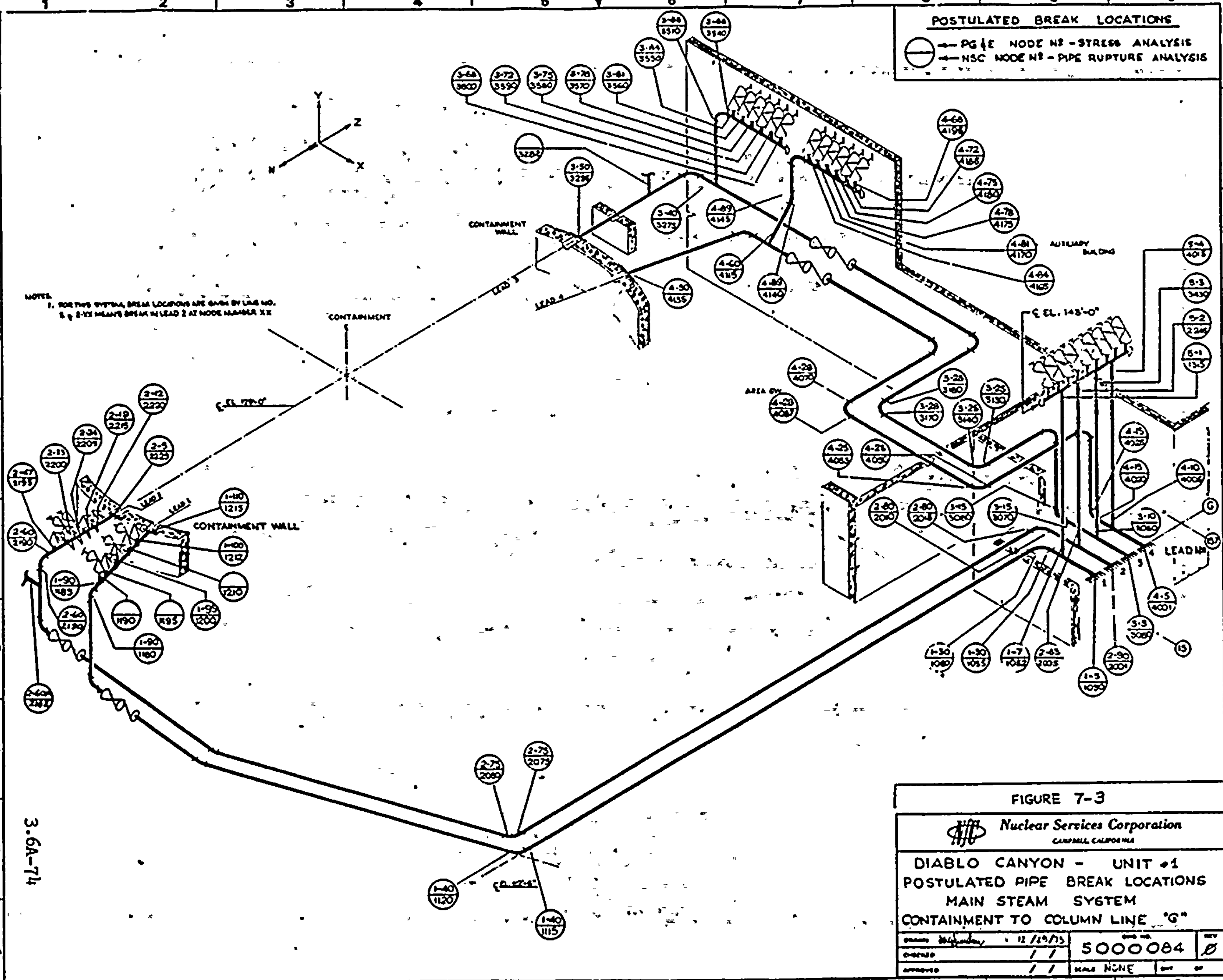


FIGURE 7-3

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

DIABLO CANYON - UNIT #1

POSTULATED PIPE BREAK LOCATIONS

MAIN STEAM SYSTEM

CONTAINMENT TO COLUMN LINE "G"

DRAWN	12/19/73	DWG NO. 5000084	REV Ø
CHECKED	/ /		
APPROVED	/ /	SCALE NONE	SHT OF

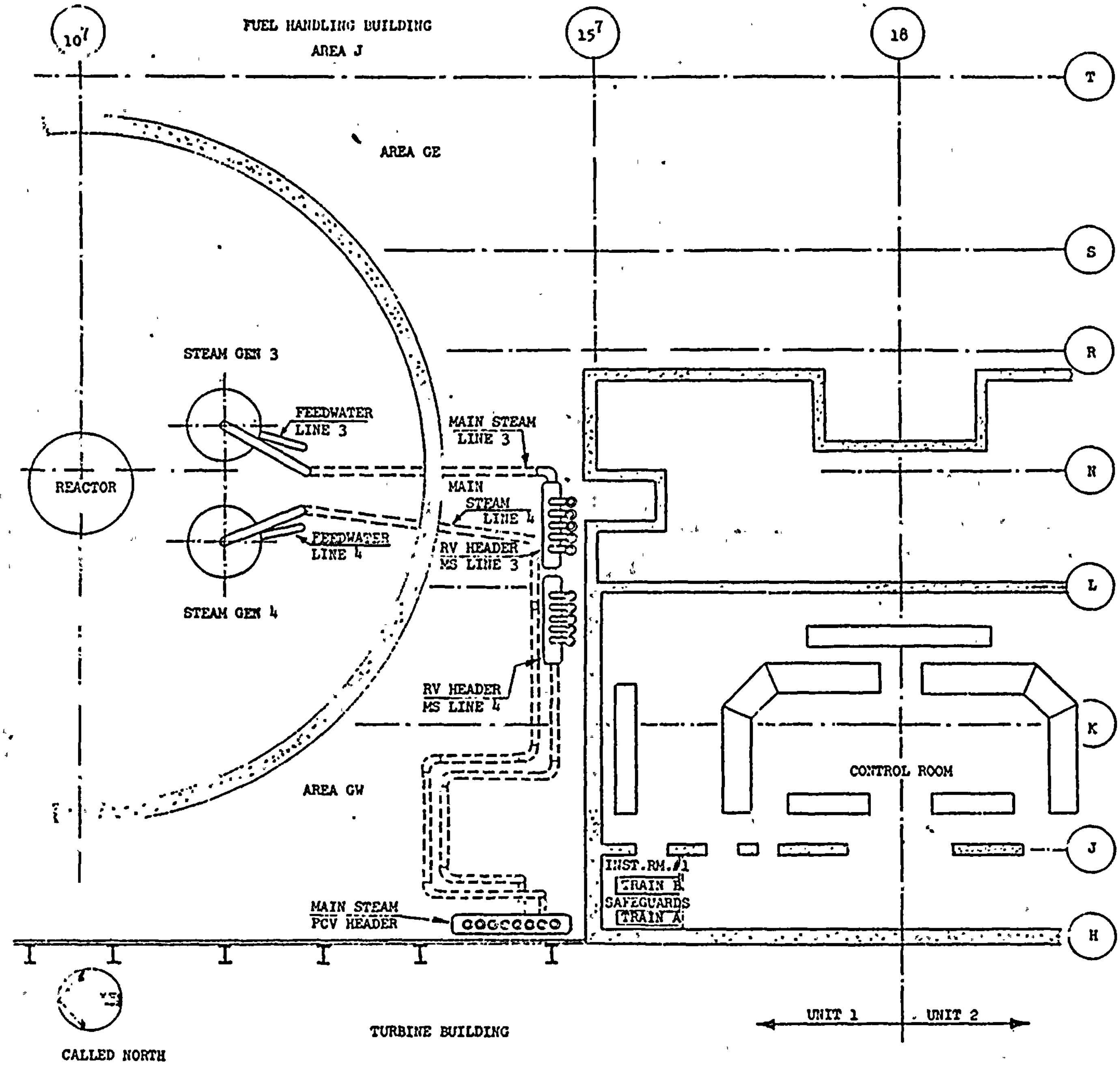


FIGURE 7-4 DIABLO CANYON UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
AUXILIARY BUILDING AREA - 140' ELEVATION

MAY 10 1974

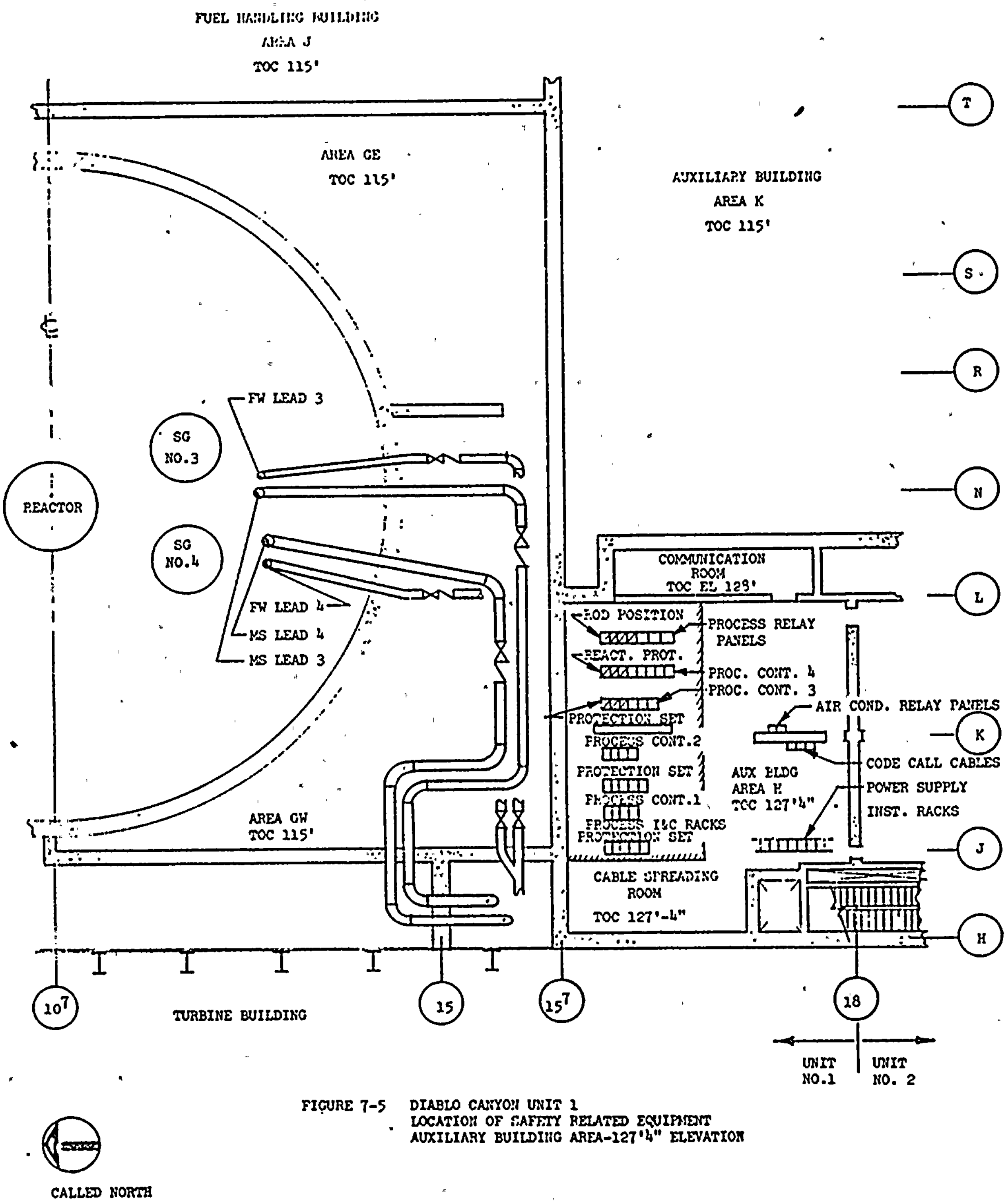


FIGURE 7-5 DIABLO CANYON UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
AUXILIARY BUILDING AREA-127'4" ELEVATION

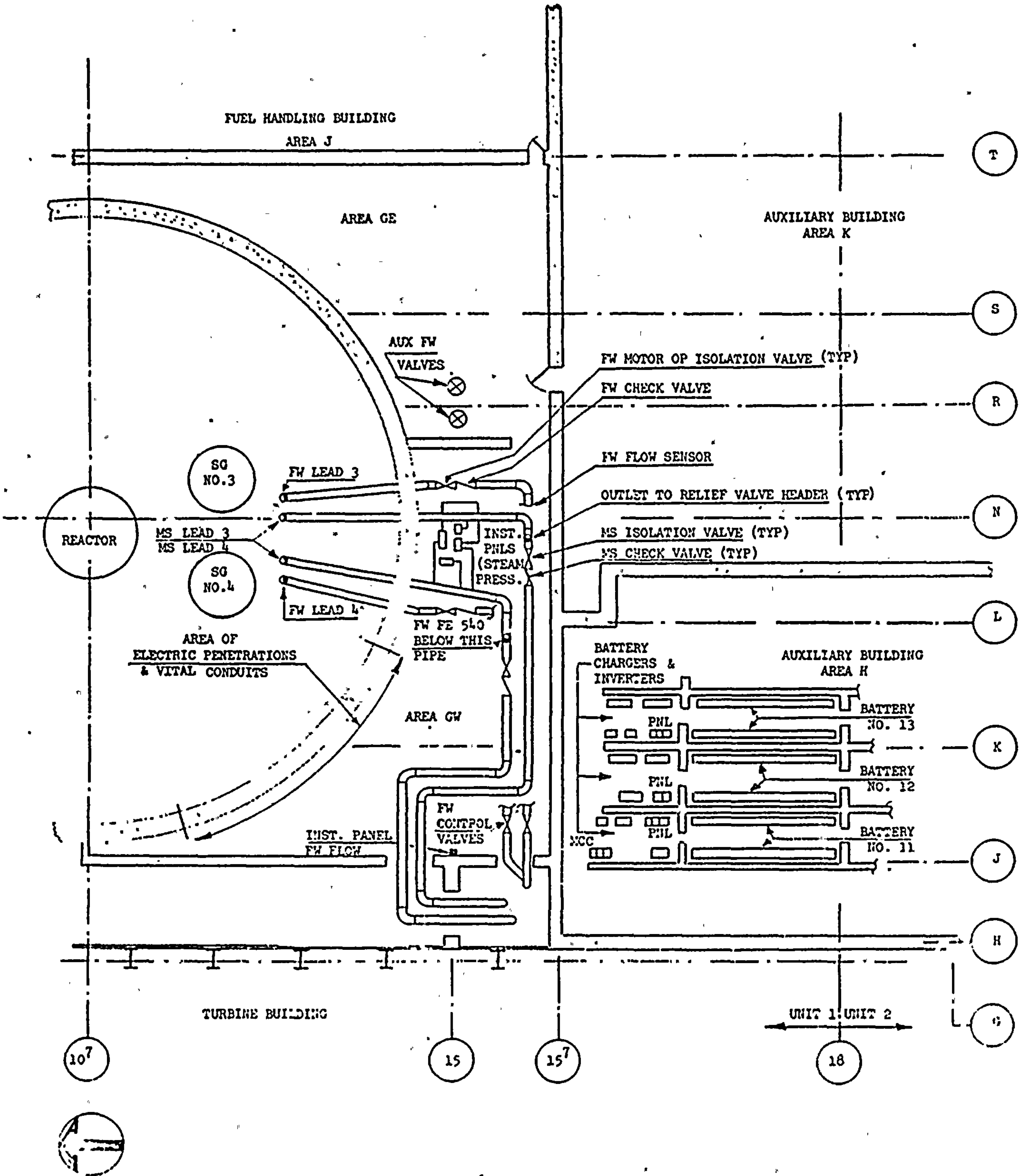


FIGURE 7-6 DIABLO CANYON - UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
AUXILIARY BUILDING AREA - 115' ELEVATION

MAY 10 1974

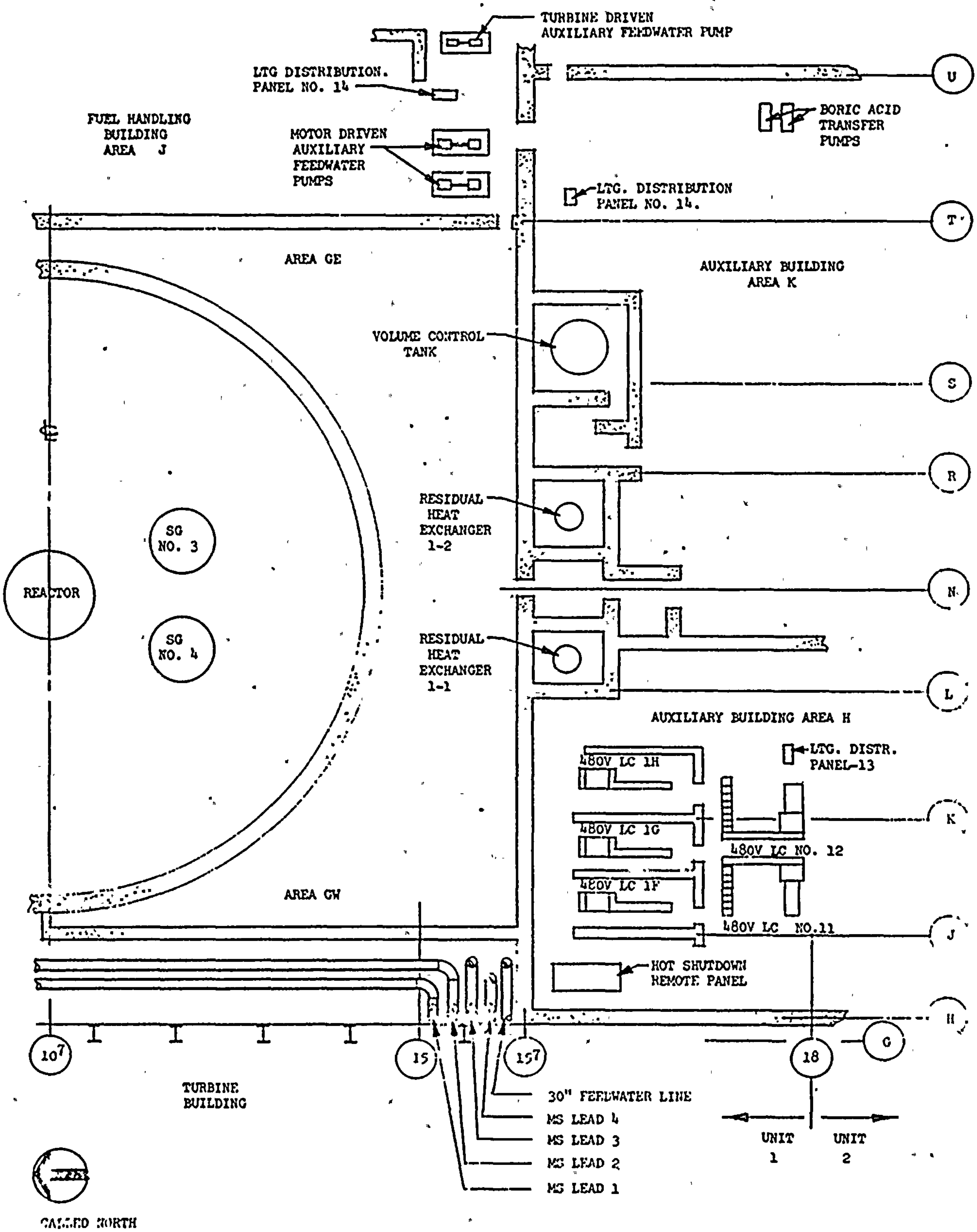


FIGURE 7-7 DIABLO CANYON UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
AUXILIARY BUILDING. 91 FOOT ELEVATION

MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

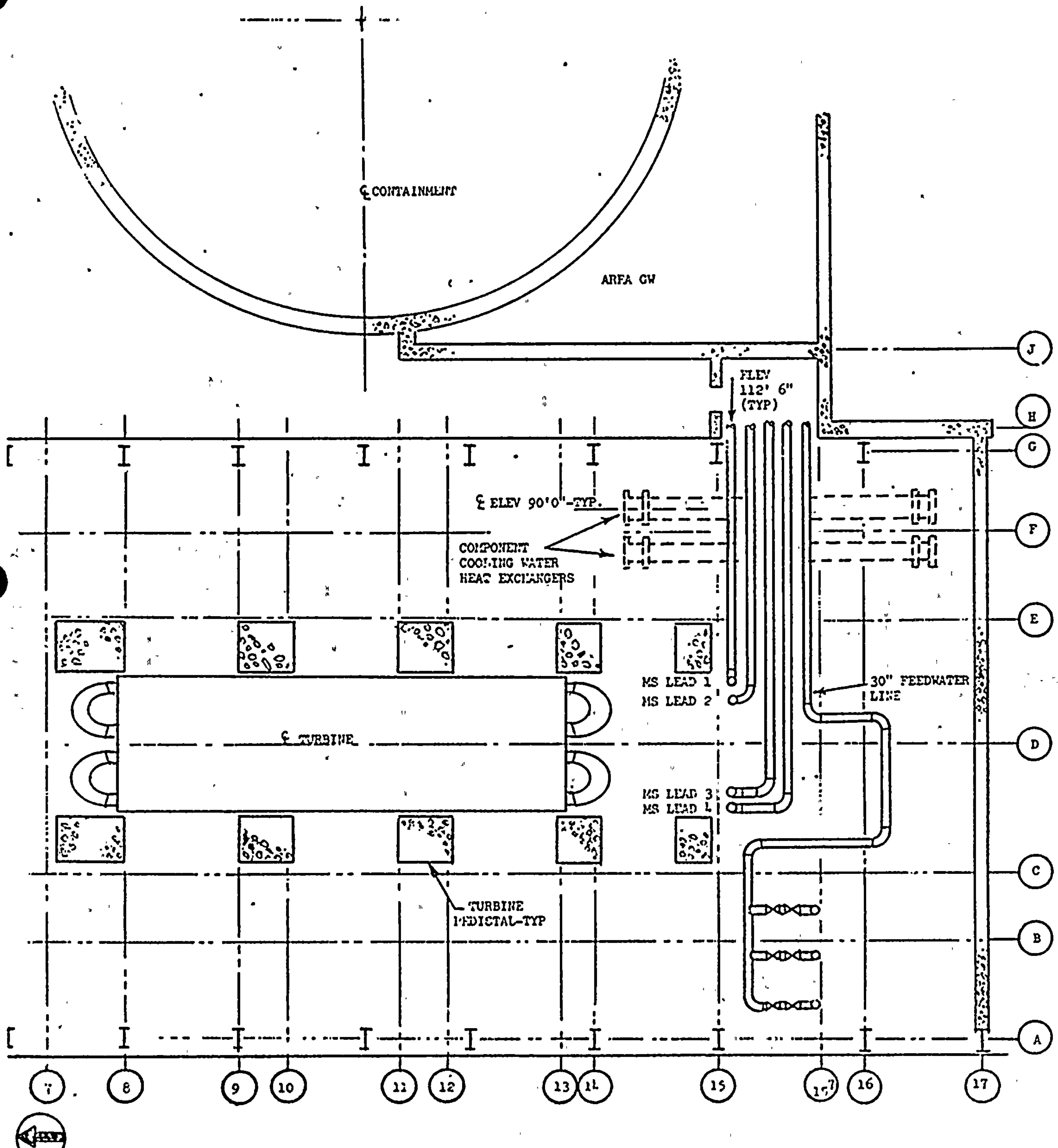


FIGURE 7-8 DIABLO CANYON UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
TURBINE BUILDING - 104' ELEVATION

MAY 10 1974

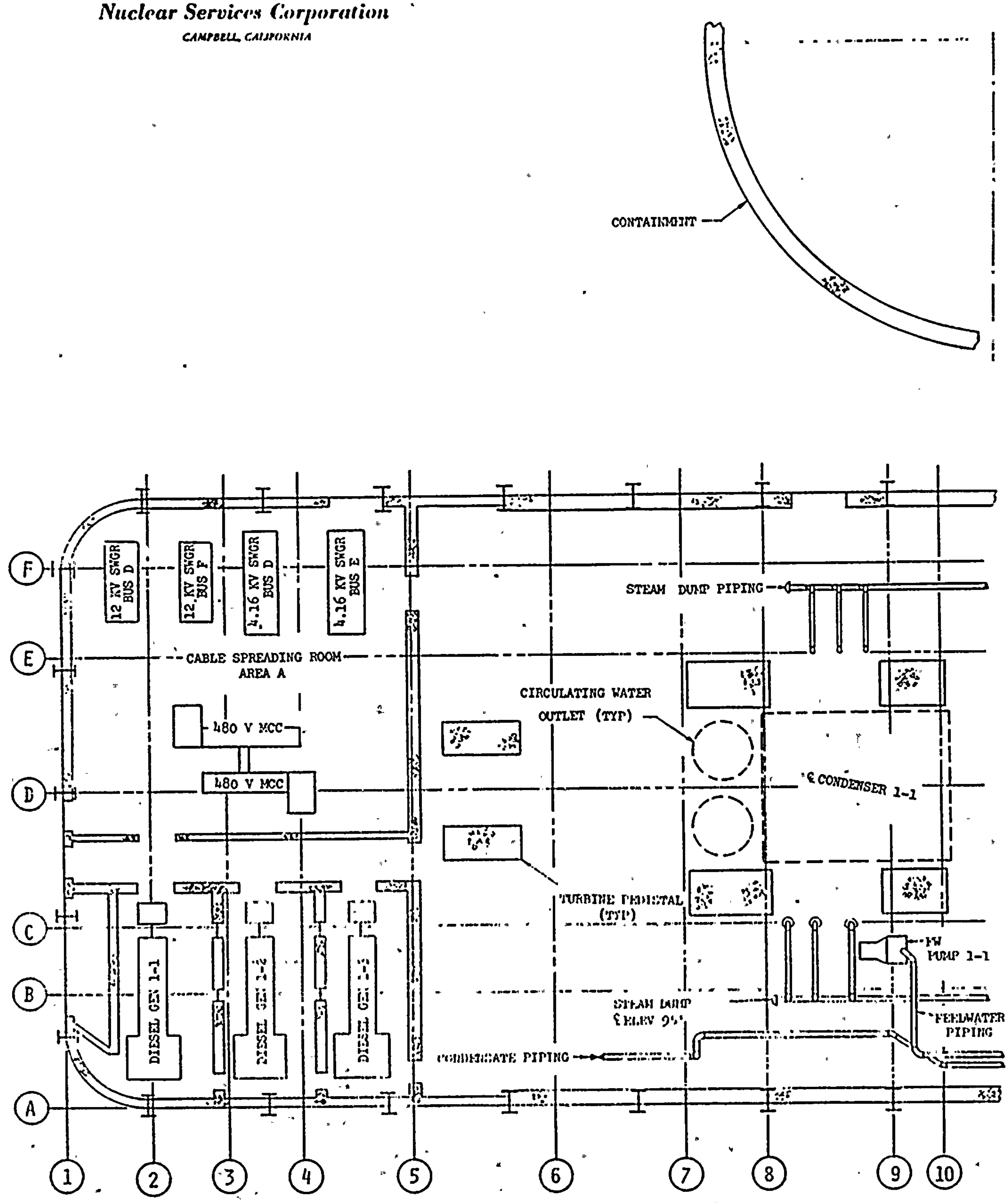


FIGURE 7-9 DIABLO CANYON UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
TURBINE BUILDING - 85' ELEVATION

MAY 10 1974

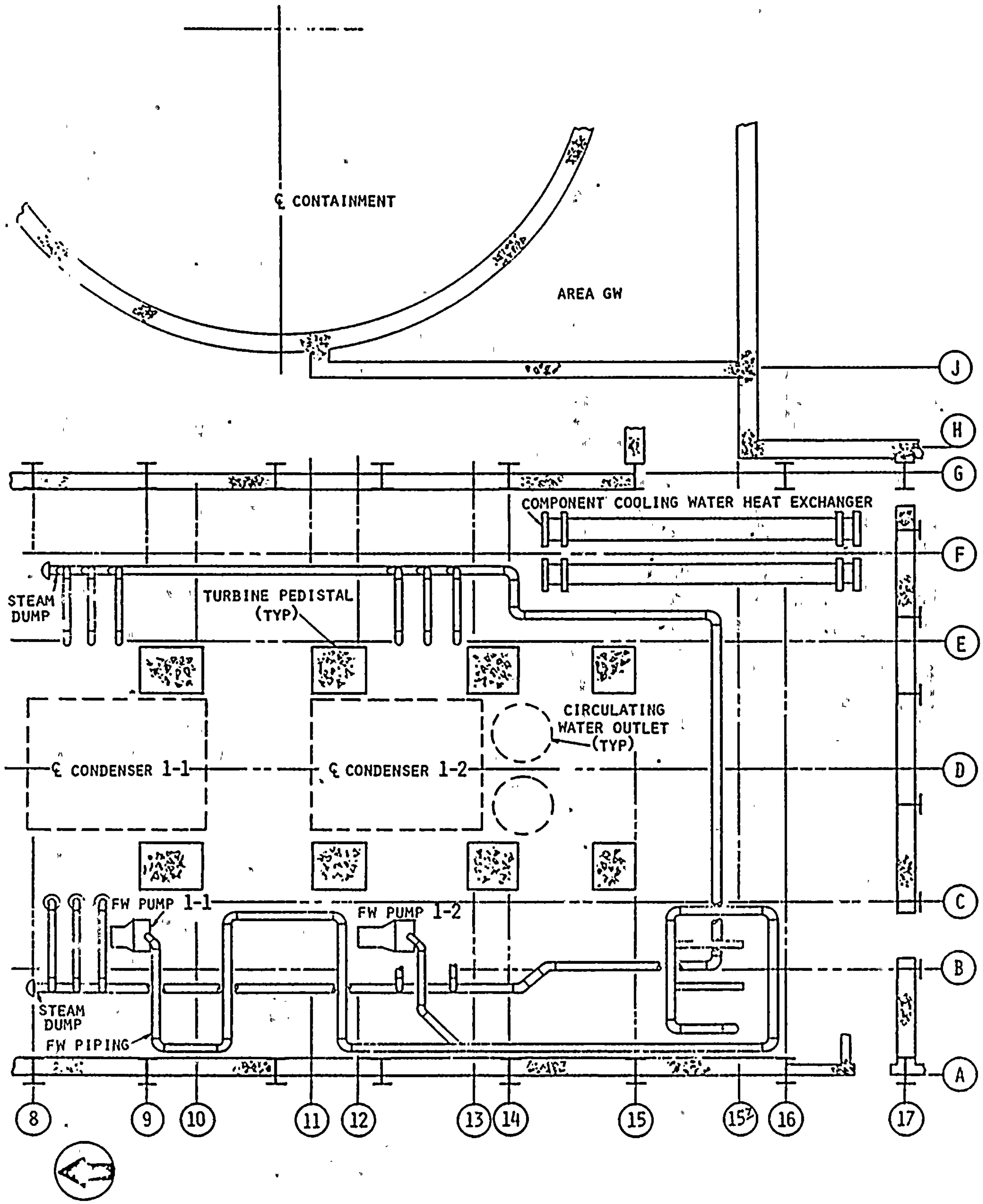


FIGURE 7-10 DIABLO CANYON UNIT 1
LOCATION OF SAFETY RELATED EQUIPMENT
TURBINE BUILDING - 85' ELEVATION

MAY 10 1974

MAY 10 1974

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

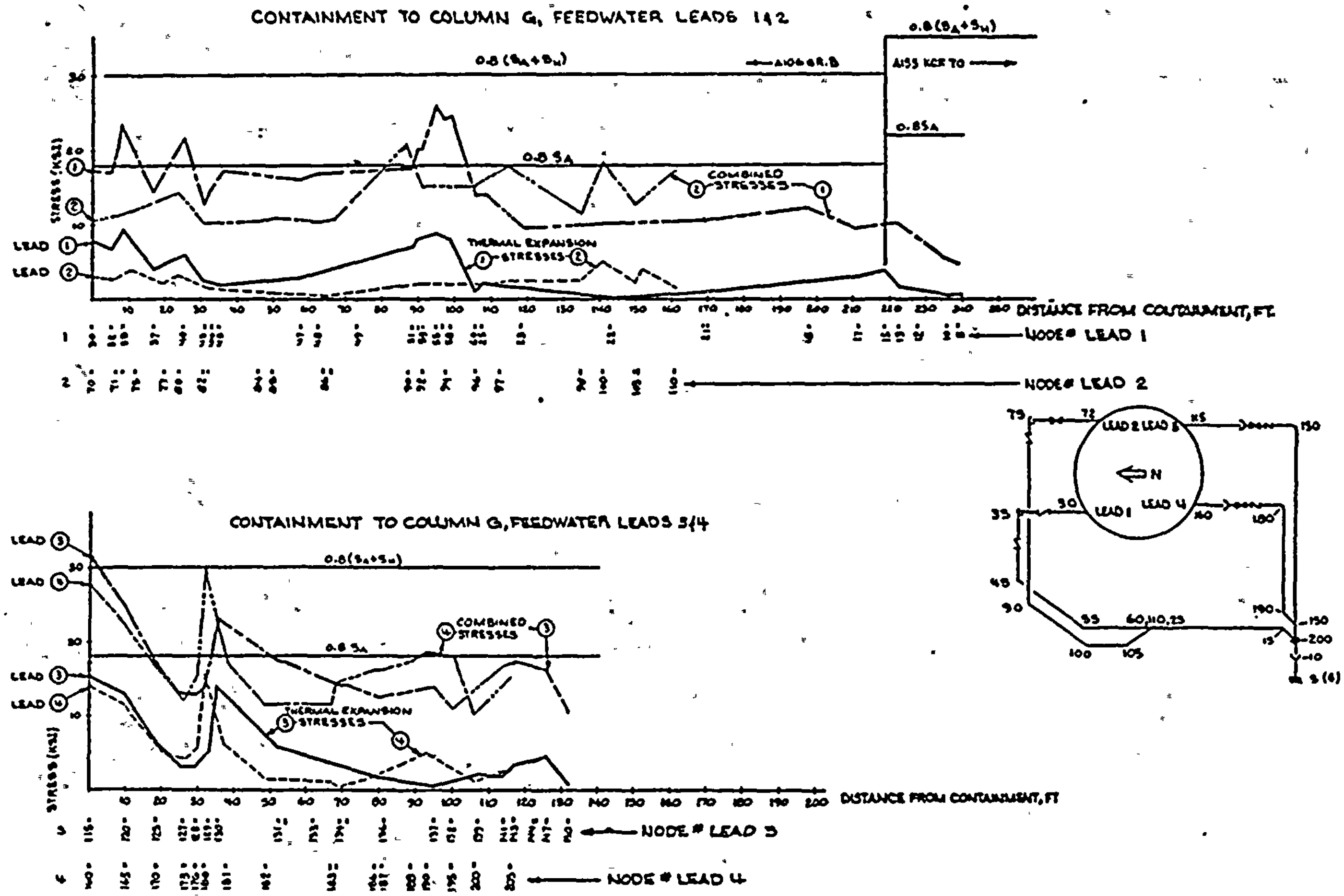
FIGURE 7-11

DIABLO CANYON UNIT 1

DETERMINATION OF POSTULATED BREAK LOCATIONS

FEEDWATER PIPING - LINES 1, 2, 3 & 4

CONTAINMENT TO COLUMN LINE 6



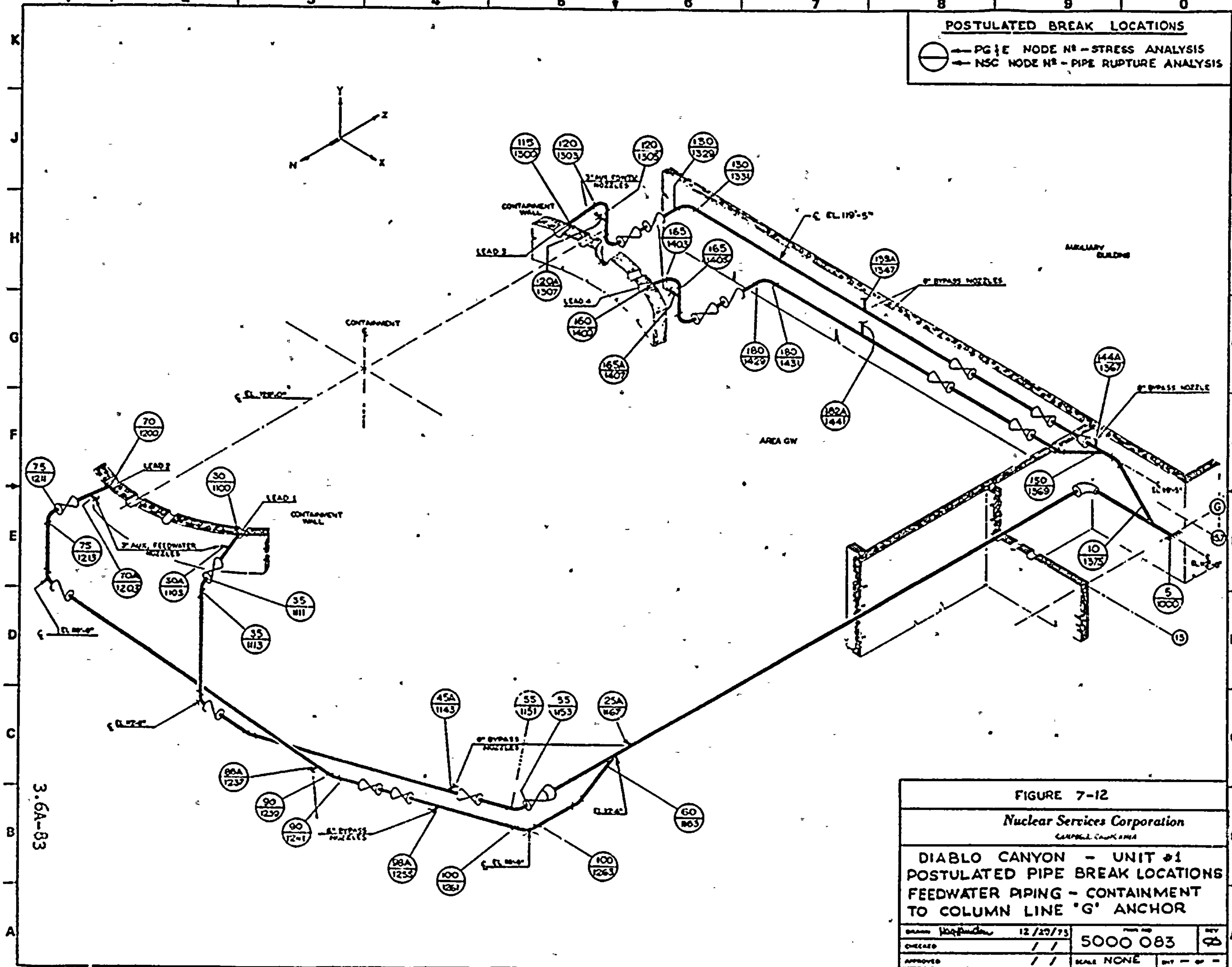


FIGURE 7-12

Nuclear Services Corporation

DIABLO CANYON - UNIT #1
POSTULATED PIPE BREAK LOCATIONS
FEEDWATER PIPING - CONTAINMENT
TO COLUMN LINE 'G' ANCHOR

DRAWN	12/27/73	5000 083	
CHECKED	/ /		
APPROVED	/ /	SCALE NONE	

MAY 10 1974

8.0 REFERENCES

1. Letter (Docket Nos. 50-275 and 50-323) from A. Giambusso of the U.S. Atomic Energy Commission to F. T. Searls of the Pacific Gas and Electric Company, dated December 18, 1972, including the attachment "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment".

2. Letter (Docket Nos. 50-275 and 50-323) from K. Kniel of the U. S. Atomic Energy Comission to F. T. Searls of the Pacific Gas and Electric Company, dated January 29, 1973, including the errata sheet for "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment".

3. Letter (Docket Nos. 50-275 and 50-323) from A. Giambusso of the U.S. Atomic Energy Commission to F. T. Searls of the Pacific Gas and Electric Company, dated August 13, 1973, including enclosures.

4. Final Safety Analysis Report, Diablo Canyon Units 1 and 2, AEC Docket Nos. 50-275 and 50-323, Pacific Gas and Electric Company.

5. Petrokas, R. F., Uffer, R., and Pichai, V., "Criteria for Evaluation of Effects of Postulated Piping Break Outside Containment for Diablo Canyon, Unit 1," Nuclear Services Corporation, Report No. PGE-01-01, September 17, 1973.

6. Piping Stress Analyses:

Thermal -

a. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 4, Diablo Canyon Unit 1, "Main Steam Lead 1-1-cont. to col. G, No RV, Normal T = 536", Dated April 16, 1973.

b. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 8B, Diablo Canyon Unit 1, "Main Steam Lead 2 Cont to Column G No. RV - Normal", dated October 11, 1972.

c. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 9C, Diablo Canyon Unit 1, "Main Steam Lead 3 Cont to Column G No. RV - Normal", dated October 9, 1972.

d. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 24C, Diablo Canyon Unit 1, "Main Steam Lead 4 Cont to Column G No. RV - Normal", dated October 9, 1972.

e. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 29A, Diablo Canyon Unit 1, "Main Steam to Turbine and Steam Dump - Normal", dated April 15, 1971.

f. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 39B, Diablo Canyon Unit 1, "Feedwater From Column G to Cont Anchors - Normal Condition", dated March 9, 1971.

g. Pacific Gas and Electric Co. Piping Flexibility Analysis Number 44, Diablo Canyon Unit 1, "Feedwater System From Feed Pumps to Column G Anchor", dated November 4, 1970.

Seismic -

h. John A. Blume and Associates Analysis, "Diablo Canyon Unit 1, Main Steam Piping Lead 1 - DE Analysis", Revision 7, dated April 20, 1971.

i. John A. Blume and Associates Analysis, "Diablo Canyon Unit 1, Main Steam Piping Lead 4 - DE Analysis, X+Y", Revision 5, dated September 13, 1972.

j. John A. Blume and Associates Analysis, "Diablo Canyon Unit 1, Feedwater Piping Lead 1 + 2 - DE Analysis", Revision 10, dated March 27, 1972.

k. John A. Blume and Associates Analysis, "Diablo Canyon Unit 1, Feedwater Piping Lead 3 + 4 - DE Analysis", Revision 5, dated April 14, 1972.

l. John A. Blume and Associates Analysis, "Diablo Canyon Unit 1, Main Steam Piping - Lead 2", Revision 2, dated September 6, 1973.

m. John A. Blume and Associates Analysis, "Diablo Canyon Unit 1, Main Steam Piping - Lead 3", dated August 23, 1973.

7. Nuclear Services Corporation, "PRTHRUST: Computer Code for Pipe Rupture Thrust Calculation".

8. Nuclear Services Corporation, "PIPERUP: A Computer Program for Analysis of Piping Systems Subject to Pipe Rupture Loads", Rev. No. 1, August 1973.

9. Richardson, L. C., et al, "CONTEMPT - A Computer Program for Predicting the Containment Pressure - Temperature Response to a Loss-of-Coolant-Accident", IDO-17220, June 1967, Phillips Petroleum Company, Idaho Falls, Idaho.

10. Rettig, W. H., et. al,"RELAP 3 - A Computer Program for Reactor Blowdown Analysis", IN-1321 (June 1970), including supplement of June 1971.

11. "Interim Acceptance Criteria for Emergency Core Cooling System For Light-Water Power Reactor", Interim Policy Statement, Atomic Energy Commission Federal Register, June 20, 1971, Page 12217.

MAY 10 1974

12. Meyer, C. A., et al., "1967 ASME Steam Tables", American Society of Mechanical Engineers, 1968.

13. Moody, F. J., "Prediction of Blowdown Thrust and Jet Forces", ASME-AICHE Heat Transfer Conference, ASME paper 69-HT-31, Minneapolis, Minn., August 1969.

14. Shapiro, A. H., "The Dynamics and Thermodynamics of Compressible Fluid Flow", Ronald Press Corp., 1953.

15. Letter from P. Blau, Westinghouse Electric Company, to D. V. Kelley, Pacific Gas and Electric Company, "P.G.&E. Nuclear Plant, Diablo Canyon Site Units Number 1 and 2, Pipe Break Outside Containment", dated October 9, 1973.

16. Pacific Gas and Electric Company, Report Number 300/8, Revision 1, dated March 4, 1970.

17. Pacific Gas and Electric Company, drawing numbers:

438432-7	438296
500032	438297
500029	

18. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture", Journal of Heat Transfer, Transactions, ASME, Vol. 87, p. 134, February 1965.

19. Biggs, J. M. "Introduction to Structural Dynamics", McGraw-Hill Book Co., 1964.

20. Pacific Gas and Electric Company, Contract Drawings for Diablo Canyon Units 1 and 2:

Restraints

049249-2	439522-4	447242-4
150724-5	439599-4	447243-2
438284-6	439600-3	447244-1
438285-6	443476-5	447256-1
438286-6	443489-4	500199-8
439521-4		

Auxiliary Building Structure

438430-12	438458-6	439534-2
438431-10	438460-11	439535-7
438432-7	438461-9	439540-3
438433-6	438462-7	439544-5
438439-12	438464-18	443191-8
438441-10	438468-17	443227-1
438444-10	439526-10	443325-1
438446-14	439528-5	443452-2
438447-11	439532-2	446519-3
438457-9		

MAY 10 1974

Nuclear Services Corporation

Turbine Building Structure

438290-3	438342-4	4394[illegible]3-6
438293-4	438346-13	4395[illegible]-1
438295-7	438351-1	4395[illegible]-1
438296-12	438383-4	43959[illegible]-4
438297-15	438410-2	4396[illegible]-3
438301-4	439441-16	444804-3
438317-7	439442-11	

Cable Tray and Conduit Layout

57556	57571	501724-5
57557-15	57594-9	502078-10
57565	57608-8	502079-
57566	57688-6	502080-3
57570		502081-3

Instrument Tubing and Air Requirements

218352-1
218558-1
218381-2
218382-1

21. Diablo Canyon Unit 1, Containment and Auxiliary Building Structure, Acceleration Response Spectra, Report JABE-PGE-DC-4R.

22. Marc Analysis Corporation, "MARC-CDC: Non-linear Finite Element Analysis Program", Revision E, July 15, 1973.

23. Faupel, J. H.,"Engineering Design",John Wiley & Sons, New York, 1964.

24. Steichen, J. M. and Paxton, M. M., "The Effect of Strain Rate on the Mechanical Properties of Austenitic Stainless Steels", Hanford Engineering Development Laboratory, HEDL-TME 71-56, May 1971.

25. Stress Report for Pacific Gas and Electric Co., by Armco Steel Corporation, Torrance, Calif., May 22, 1972.

26. Annual Book of ASTM Standards (1973), Part 4, April 1973, pp. 684-689.

27. Mechanical and chemical lot test data, submitted by United States Steel Corporation to Pacific Gas and Electric Company per PG&E Spec. No. 8833XR, ABD Orders K-4924-33 and K4928-X-1.

28. USS Steels for Nuclear Applications, Vol. 3 page 31.

29. "Building Code Requirements for Reinforced Concrete", ACI 318-71.

MAY 10 197[illegible]

30. Wood, R. H., "Plastic and Elastic Design of Slabs and Plates", Ronald Press, 1961.

31. Johansen, K. W., "Yield Line Theory", Cement and Concrete Association.

32. Ferguson, P. M., "Reinforced Concrete Fundamentals", J. Wiley and Sons, Inc., July 1972.

33. Beedle, Blackmon, et. al., "Structural Steel Design", The Ronald Press, New York, 1964.

34. "Manual of Steel Construction", American Institute of Steel Construction, 7th Edition, 1970.

35. "Nelson Concrete Anchors", Engineering Design Data, Nelson Stud Welding, Division of Gregory Industries, Lorain, Ohio, 1961.

36. Timoshenko, S., Woinowsky-Krieger, S., "Theory of Plates and Shells", McGraw-Hill, 1959.

37. Transmittal: From I. Sokoloff (PG&E) to Nuclear Services Corporation dated November 6, 1973. Auxiliary building and containment stress analysis data.

38. Bresler, Boris, et al, "Design of Steel Structures", John Wiley and Sons, Inc., New York, 1960.

39. Swartzberg, et. al., "Cryogenic Materials Data Handbook", (Revised) Volume 1, AD713 619, National Technical Information Service, Springfield, Va., July 1970.

40. General Electric, Nuclear Energy Division, Pipe Restraints (Option H.), LA Std., R-P1, Sept. 1973.

41. American Society of Metals, "Metals Handbook", Volume 1, 8th Edition, 1961.

42. Pai, S., "Fluid Dynamics of Jets", Van Nostrand Corp., New York, 1954.

43. Birkhoff and Zarantonello, "Jet Wakes and Cavities", Academic Press, New York, 1957.

APPENDIX A

THERMAL HYDRAULIC ANALYSIS

TABLE OF CONTENTS

		Page
A.1	INTRODUCTION	3.6A-93
A.2	RUPTURED PIPE BLOWDOWN REACTION FORCES	3.6A-94
	A.2.1 Criteria and Assumptions	3.6A-94
	A.2.2 Methods of Analysis	3.6A-96
	A.2.3 Cases Considered and Typical Results	3.6A-102
A.3	COMPARTMENT PRESSURES	3.6A-105
	A.3.1 Areas Affected	3.6A-105
	A.3.2 Criteria and Assumptions	3.6A-105
	A.3.3 Analytical Methods	3.6A-108
	A.3.4 Results and Discussion	3.6A-109
A.4	ENVIRONMENT CONDITIONS	3.6A-111
	A.4.1 Areas Affected	3.6A-111
	A.4.2 Criteria and Assumptions	3.6A-113
	A.4.3 Analytical Methods	3.6A-115
	A.4.4 Results and Discussion	3.6A-116
A.5	BUILDING FLOODING ANALYSIS	3.6A-117
	A.5.1 Areas Affected	3.6A-117
	A.5.2 Criteria and Assumptions	3.6A-118
	A.5.3 Method of Analysis	3.6A-121
	A.5.4 Discussion of Results	3.6A-124

A.1 INTRODUCTION

This appendix contains the thermal hydraulic analyses conducted to establish (1) short term piping blowdown loads, (2) maximum compartment pressure, (3) long term compartment environments, and (4) room flooding.

The short term piping blowdown loads analyses are presented in Section A.2 and were conducted to establish reaction forces on piping systems when pipe rupture occurs. The blowdown reaction forces are then used to determine pipe whip and impact loading on restraints, walls and other obstacles. The blowdown reaction loads are established up to 0.5 seconds after a break, the period in which the major pipe motion and impact loads occur.

Calculation of maximum compartment pressurization due to a pipe rupture is described in Section A.3. It is used in the building structural evaluation (Appendix D) to ensure integrity of that building structure necessary to ensure safe plant shutdown following a postulated pipe rupture.

Safety related equipment must also be capable of surviving high temperature and humidity conditions which might occur due to a pipe rupture. The long term environmental conditions were therefore defined for evaluation of equipment and are contained in Section A.4.

Flooding due to main line rupture was also considered and is presented in Section A.5.

A.2 RUPTURED PIPE BLOWDOWN REACTION FORCES

When a pipe containing high energy fluid ruptures a blowdown force is generated on the pipe which causes the pipe to move. The particular motion and velocities reached are dependent on piping geometry, stiffness characteristics, mass properties and reaction forces. The reaction forces due to a break must therefore be determined in order to establish pipe motion and impact loads into restraints, walls, and other obstacles. The method of analysis used to establish the reaction forces and typical results are presented in this section.

A.2.1 Criteria and Assumptions

Pipe rupture blowdown forces have been determined for the representative break locations defined in Tables 7-1 through 7-6 and are based on the following criteria and assumptions:

A. For design basis circumferential breaks, the flow area is equal to the pipe flow area with a discharge coefficient of 1.0.

B. For design basis longitudinal breaks, flow comes from both upstream and downstream sides of the break. The break length is twice the pipe nominal diameter with a break area equal to the flow area of the pipe. A discharge coefficient of 1.0 is used.

C. The break opens in one millisecond.

D. Initial conditions at time of break is that of the most severe normal plant operating condition.

E. The piping system is modeled taking into account any flow limiters, line restrictions and friction.

F. Failure of the most consequential in line-active component occurs coincidental with the break.

For the short term blowdown analysis the most severe plant operating condition is hot standby. At hot standby conditions, the system is at the highest pressure, 1020 psia. It is also assumed that the plant is at full load immediately prior to the break, thus resulting in the highest possible feedwater temperature of 432°F. The steam generators contain a mixture of steam and water, and the main steam lines contain saturated steam. Under a hot standby condition, the turbine stop valves are also closed thereby maximizing the duration of blowdown forces.

The critical period for blowdown reaction forces is within the first 0.5 seconds, and during this short time the flow control or stop valves cannot respond fast enough to affect short term blowdown forces. Only the isolation check valves can operate within this time, and therefore, the single active component failure is assumed to be a check valve. This assumption will create maximum blowdown forces.

For main steam line breaks, feedwater flow into the steam generators has no effect on the results within the short 0.5 second critical time period. Similarly, a constant pressure feedwater supply from the feedwater pumps can be assumed for the feedwater line breaks.

A.2.2 Methods of Analysis

System blowdown calculations are made using the Nuclear Services Corporation PRTHRUST computer program (Reference 7). PRTHRUST is a modification of RELAP3 (Reference 10), the AEC's presently accepted Loss of Coolant Accident analysis code (Reference 11), for the specific requirements of pipe rupture analysis.

In PRTHRUST, the fluid system is mathematically modeled as an assemblage of control volumes interconnected by flow paths. Characteristics of a control volume include state of the contained fluid and potential energy addition or subtraction. Control volumes are used to model such components as pressure vessels, steam generators, heat exchangers and the piping volumes. Flow paths are used to interconnect control volumes and may include operable control valves, check valves, fills and pumps. A time dependent thermal hydraulic solution is then obtained by integrating a set of differential equations subject to the basic properties of water.

A basic mass and energy balance is performed on each defined volume using the following equations:

$$\frac{dM_i}{dt} = \sum_{j=1}^{N} (W_i)_j \qquad \text{(MASS BALANCE)}$$

$$\frac{dU_i}{dt} = \sum_{j=1}^{N} W_{ij} (h_i)_j + Q_i \qquad \text{(ENERGY BALANCE)}$$

where

$$h_i = \frac{U_i}{M_i} + P_i\left(\frac{V_i}{M_i}\right)$$

M_i = total mass in volume i

W_{ij} = flow rate into volume i through junction j

U_i = energy in volume i

h_{ij} = enthalpy of flowing fluid

Q_i = heat input to volume i

N = number of junctions in volume i

Energy, Q_i, can be added (or subtracted) by defining heat exchangers and/or reactor core regions. Through suitable input parameters and mathematical modeling in the program the energy is calculated and used in the volume energy balance. The program also allows actuation of flow control and energy devices to be triggered at a specified time or based on a physical signal such as pressure or flow at a point in the system. The variation in pump performance under transient conditions is also considered, if desired.

MAY 10 1974

The pressure, P_i, in each volume is determined implicitly by requiring the mass of fluid, M_i, with internal energy, U_i, to fill the control volume, V_i. Through the use of enthalpy, h_i, an estimated pressure, and the 1967 ASME steam tables (Reference 12), the specific volume of the fluid is calculated and compared to the known specific volume V_i/M_i with the equation:

$$h_i = \frac{U_i}{M_i} + P_i\left(\frac{V_i}{M_i}\right) \qquad \text{(THERMODYNAMIC PRESSURE)}$$

Through an iterative process the volume pressure is determined.

Junction flows are calculated from the one-dimensional momentum equation:

$$\frac{1}{144\ g_c}\left(\frac{\ell}{A}\right)\frac{dW_j}{dt} = (P_i - P_{i+1}) + \Delta P_p - \frac{K_j W_j \left|W_j\right|}{\rho_j} + \int_{v_j} \frac{\rho dz}{144}$$

where

g_c = gravitational conversion constant

$\frac{\ell}{A}$ = junction inertia

W_j = average flow from volume i to volume i+1

$(P_j - P_{i+1})$ = thermodynamic pressure differential across the fluid contained in the flow volume

ΔP_p = pump head

$\int_{v_j} \rho dz$ = gravitational head across fluid column

K_j = net friction coefficient including normal friction losses

ρ_j = fluid density in volume j.

A limiting (choked) mass flow is defined in the program by Moody's two-phase choked flow model (Reference 13) where maximum mass flow flux is a function of stagnation pressure and enthalpy. The flow through the junction is chosen as the smaller of the inertial flow (momentum equation) or choked flow. Moody's model also defines the static pressure existing at the throat (limiting flow area).

The discharging fluid thrust (reaction force) is calculated based on the momentum equation applied to the control volume in which the break occurs. From Reference 14, page 18:

$$\sum F = F_R - F_p = \frac{d}{dt} \int_{c.v.} \frac{\rho v.dV}{g_c} + \oint_{c.s.} \frac{\rho(v.dA)v}{g_c}$$

where

$\sum F$ = sum of external forces

F_R = reaction blowdown force

F_p = pressure force

t = time

p = density

v = velocity

c.v. = control volume

V = volume

A = area

g_c = gravitational constant

c.s. = control surface

Transposing terms results in the simplified equation

$$F_R = F_a + F_m + F_p$$

where

F_a = acceleration force = $\frac{d}{dt}\int_{c.v.} \frac{\rho v.dV}{g_c}$

F_m = momentum force = $\oint_{c.s} \frac{\rho(v.dA)v}{g_c}$

F_p = pressure force = $A(P_e - P_a)$

P_e = exit pressure

P_a = ambient pressure

MAY 10 1974

The acceleration force, F_a, component is caused by the change in momentum with time and can be approximated by:

$$F_a = 1/2\ \ell \left(\frac{W_{t+\Delta t} - W_t}{\Delta t . g_c} \right)$$

where

t = time

Δt = calculation time increment

ℓ = length of control volume

W = mass flow rate

The momentum force, F_m, represents momentum from the control volume and can be approximated by:

$$F_m = \frac{\rho A v^2}{g_c} = \frac{Wv}{g_c}$$

The velocity, v, is the exit velocity from the break as determined from either Moody's model or the momentum equation previously discussed.

The pressure force, F_p, is given by the equation:

$$F_p = A(P_e - P_a)$$

If the limiting flow (choked flow) is established, P_e is determined by Moody's model. If inertial or unchoked flow exists $P_e = P_a$ and the pressure force term is equal to zero.

The PRTHRUST computer program incorporates the above equations and solves the time dependent parameters. Reaction force, flow rates, pressures, fluid energy and other pertinent parameters are listed as a function of time.

A.2.3 Cases Considered and Typical Results

Short term blowdown reaction forces resulting from a design basis circumferential or longitudinal break have been calculated for twelve representative locations on the main steam line and twelve representative locations on the feedwater line. The blowdown forces were determined for all locations requiring pipe whip analysis. Each case represents more than one location and was selected as the representative upper bound reaction force for the other break locations. The blowdown reaction forces were established for the nodes summarized in Tables A-1 and A-2. Break location nodes are identified in Figures 7-3 and 7-12 except for nodes 3580 and 1640, located in the turbine building. Node 3580 is located in main steam lead 3 at the high pressure turbine inlet (see Figure B-7). Node 1640 is located in the steam dump line adjacent to the turbine.

Typical models used in the analysis of blowdown reaction forces are given in Figures A-1 and A-2. Figure A-1 represents a circumferential break in the main steam line at node 3050 (case No. 7A). For the hot standby

MAY 10 1974

condition the turbine stop valves are closed thereby stopping steam flow to the turbines. The steam dump line and crosstie header connect all four steam lines together thus feeding steam to both sides of the break. System geometric parameters and initial conditions input to the PRTHRUST computer program for this case are given in Tables A-3 and A-4. The initial pressure and steam generator quality used in the analyses are 1020 psia and 0.04, respectively. These values correspond to a hot standby plant condition.

Figure A-2 represents a feedwater line longitudinal break at the G row anchor, node 1000. As shown in Figure A-2, the flow nozzles, check valves, and pumps have been modeled into the system. Tabular input to the PRTHRUST computer program is given in Tables A-4 and A-5. Similar to the steam runs, the initial pressure and steam generator quality are 1020 psia and 0.04, respectively.

A.2.3.1 General Discussion of Main Steam Line Blowdown Loads

At time zero the pipe rupture is postulated to occur and opens to the full break area in one millisecond. Almost instantaneously the reaction force reaches a value approximately equal to the pipe pressure times the break area. The force increases steadily as the flow accelerates until choking occurs. When the flow chokes the reaction force decreases due to the decreased acceleration force component. The reaction force then fluctuates until a quasi-steady state flow condition is reached in the system. During the quasi-steady state condition the reaction force varies with line pressure adjacent to the break. The reaction force decreases as the piping system depressurizes.

The maximum force for each run is summarized in Table A-1. The maximum force is dependent on break area and ranged from 70,000 to 755,000 pounds.

A.2.3.2 General Discussion of Feedwater Line Blowdown Loads

Similar to the main steam line breaks, the reaction load almost instantaneously reaches a value approximately equal to line pressure times the break area. The force, however, decreases rapidly due to the rapid depressurization of the nearly incompressible fluid. The pressure stabilizes at the saturated water pressure corresponding to the feedwater temperature with the force proportional to the pressure. The reaction force then increases until choking occurs. A quasi-steady state force is then reached and fluctuates with the system line pressure adjacent to the break. Table A-2 summarizes the maximum force calculated for the various break conditions. The maximum force varied from 44,000 to 595,000 pounds.

MAY 10 1974

A.3.0 COMPARTMENT PRESSURES

The objective of the compartment pressurization analysis is to establish structural integrity of that building structure necessary to ensure safe plant shutdown following a postulated pipe rupture.

A.3.1 Areas Affected

A survey of the main steam and feedwater lines through the plant show that a break in area GW at the 115' elevation could pressurize this compartment to a significant level. This region of the auxiliary building must maintain its structural integrity to affect a safe plant shutdown.

A rupture of the main steam or feedwater piping outside of area F (north of containment) would result in the steam being released directly to atmosphere, hence no building compartments would be pressurized. A rupture of the main steam or feedwater piping within the turbine building would not result in any significant level of pressurization due to the enormous size of this compartment.

A.3.2 Criteria and Assumptions

In the event of a main steam line break in area GW (elevation 115') the room would be pressurized by the steam escaping from the break. Pressurization of this room would also occur if a feedwater line broke in this area. In this case, however, the portion of the feedwater flashing into steam would cause the pressurization. It has been determined that the energy and mass blowdown from a double ended circumferential main steam line break is more severe than the blowdown from a longitudinal main steam line break or a circumferential or longitudinal break of a feedwater line. This incident, having the greatest potential for damaging the room due to overpressurization, was therefore selected for analysis.

Compartment GW pressurization was calculated using the following assumptions:

A. The criteria of Section A.2.1 apply except as modified below.

B. The reactor is operating at full load conditions. Reactor trip occurs a maximum of 0.9 seconds after sensing of a high steam line pressure differential (Reference 4, Figure 7.2-1 and Reference 15).

C. The heat transferred from the primary coolant to the steam generator depends on the temprature difference between the primary and the secondary sides of the steam generator and on its overall conductance. The heat added to the primary coolant includes reactor decay heat and heat stored in the primary system and was taken from Reference 15.

D. Steam generator pressure is 1020 psia with a 6% quality, corresponding to a full load plant operating condition, and the feedwater temperature is 432.5°F. These values were obtained from Reference 4.

E. Auxiliary feedwater supply is 675 gallons/minute (Reference 15) to each feed line and is initiated in 2.0 seconds (Reference 4, page 7.3-6).

F. No credit is taken for area reduction in valves.

G. In area GW, at the 115' elevation, credit was taken for venting steam to the atmosphere through the construction opening in J wall.

H. The compartment volumes and venting areas used in evaluation of pressures are as shown in Figure A-11.

I. The walls and floors were modeled as exposed concrete surfaces to simulate their heat sink effect.

J. A relative humidity of 60% exists in the area GW at the time of the break with an initial temperature of 70°F and pressure of 14.7 psia.

K. A discharge coefficient of 0.6 was assumed for all vent areas.

L. A full load plant operating condition with an initial line pressure of 1020 psia with no pressure drop in the piping system is used. This condition represents the worse possible condition conceivable in that the reactor power is maximum, flow control valves are wide open, and pumps are at full load.

A.3.3 Analytical Methods

The compartment pressurizations were determined using two different techniques. In the first method, a long term blowdown analysis was initially performed using Nuclear Services Corporation computer program "PRTHRUST" (program description and capability is discussed in Section A.2.2) to determine the mass flow rate and enthalpy out of the break. No compartments were modeled and the break flow was assumed vented to the atmosphere (14.7 psia) for conservatism. The PRTHRUST model used is given in Figure A-9. The results from this analysis, namely, mass flow rate and enthalpy, were then used as input to a modified version of computer program "CONTEMPT" (Reference 9) to determine the maximum pressures in the compartment.

The CONTEMPT program predicts the pressure-temperature response of a volume (the building compartment) to an energy source (flow from the ruptured steam line). The input conditions for the program are the flow rate and the enthalpy of the leaking system. The building volume is separated into a liquid region and a vapor region (initially air) with each region assumed to have a uniform temperature. The compartment is represented as consisting of several heat-conducting structures whose thermal behaviour can be described by the one-dimensional, multi-region, heat-conducting equation. The program also includes the compartment leakage through venting areas. A schematic of the CONTEMPT model is given in Figure A-10.

MAY 10 1974

The above technique to determine maximum pressures assumes all venting from the compartment is to the atmosphere, including those vents to adjacent rooms. This is due to the program incapability of modeling more than one volume or more than one vent area. The validity of this assumption for the GW compartment was checked with the more representative multi-compartment analysis.

This second method of analysis was run using PRTHRUST (Reference 7), which has a multi-volume capability. The adjacent GE and lower level GW compartments were modeled along with the GW compartment in which the break was postulated. The pressure time history in each of the compartments could therefore be determined. This method assumes an initial steam environment in all compartments. The PRTHRUST compartment model is given in Figure A-11. For conservatism, the higher of the two pressures determined was used in the structural analysis to establish the capability of the wall and floors to withstand pressurization.

A.3.4 Results and Discussion

Analyses indicated that the construction opening in the J wall of compartment GW elevation 115' was required to limit the consequences of a steam line break in this room with regard to pressurization. Accordingly, credit was taken for this additional vent area in all analyses. The predictions of building peak pressure for the GW and GE areas at elevation 115' are shown in Figures A-12 and A-13.

The CONTEMPT analysis gave a slightly higher peak pressure. Pressures resulting from this analysis are shown in Figure A-12. The pressure reaches its maximum value of 23.96 psia in 0.27 seconds and drops down to about 19 psia within 0.9 seconds. The corresponding PRTHRUST pressures are shown in Figure A-13. Here the peak pressure is 20.46 psia which occurs in about 0.15 seconds. Pressurization levels in the adjacent GE and GW (100' elevation) compartments are also shown in Figure A-13. Results indicate that sufficient venting is available in these adjacent compartments to keep the peak pressures down to relatively low levels (<1 psi). This confirms the assumption made in the CONTEMPT analysis that vents to these compartments can be represented as vents to atmosphere. The CONTEMPT analysis, which gave the higher maximum compartment pressure, was used for evaluation of building structural integrity (Appendix D).

A.4 ENVIRONMENT CONDITIONS

The objective of the long term building environment analysis is to establish the temperature and humidity transient response within a building during a main process line break. This is required to verify that no safety related equipment required for safe shutdown of the plant becomes inoperative due to this exposure.

A.4.1 Areas Affected

A review of the main processing lines (feedwater and steam) has been made to establish those building regions where a line break will expose equipment used for safe plant shutdown to a high humidity and temperature environment.

Main processing lines 1 and 2, from the north side of the containment building to the turbine building entry, are routed in such a way that any postulated line break would vent directly to the atmosphere and would create no danger to safety related equipment.

A rupture of main line leads 3 or 4 between the containment building and their entrance into the turbine building (via penetration through J wall) would occur in area GW at an elevation of 115'. Hot steam would mix with air in this compartment with leakage (of the mixture) into the following adjacent areas:

A. Compartment GE at elevation 115'. The venting would occur through the doorway and through the opening between the containment and the wall separating GE from GW.

B. Compartment GW at elevation 100'. Venting would occur through the annular opening between the containment wall and the floor at elevation 115 feet and through small openings in the floor.

C. The turbine building through miscellaneous holes in the J wall.

D. Venting to the atmosphere through the construction opening (assumed to remain), various small holes in the ceiling and through the annular space between the containment wall and the ceiling at an elevation of 140'.

Steam leakage to other areas will be of a very insignificant amount. Steam will be prevented from escaping into the auxiliary building (area K) and the fuel handling building (area J) by maintaining closed doors to the GW and GE areas.

A rupture of the main steam or feedwater lines within the turbine building will not result in any significant temperature or humidity rise due to the large volume and heat sink surface of this building.

A.4.2 <u>Criteria and Assumptions</u>

As discussed above, a rupture of main steam or feedwater lines 1 or 2 on the north side of the containment would result in direct venting to the atmosphere with no resulting consequence.

The highest peak environmental temperature in area GW would be caused by a circumferential break of a 28 inch main steam line. A main steam line break will release more energy and be more severe than any other break. Since main steam lines 3 and 4 in this region are almost identical, only line 4 was selected to be analyzed.

The conditions and assumptions used for this analysis is as follows:

A. The plant is operating at full load. At full load conditions, flow control valves are wide open, pumps are at full flow and reactor power is at maximum. This operating condition gives the highest long term (>2 seconds) environmental temperatures.

B. Off-site power is lost concurrently with the pipe rupture, unless a more severe condition exists when off-site power is not lost.

C. A failure of one active flow control device (flow control valves, check valves, isolation valves etc.) is assumed. The device which is assumed to fail is selected to give the most severe environment. For

MAY 10 1974

the postulated main steam line 3 or 4 breaks it has been established that the most severe condition exists when either the main feedwater control valve or the main steam check valve fails in the line which ruptures.

D. Reactor and safety injection trips, setpoints, and delays used were those established in the Final Safety Analysis Report (Reference 4). Table A-7 defines the setpoints and activation times for the various components in the system. The setpoint values include measurement inaccuracies which will result in the most severe environmental conditions.

E. Initial compartment pressure, temperature, and relative humidity values assumed in the analysis are 14.7 psia, 70°F and 60%, respectively.

F. A total vent area of 338 square feet with a discharge coefficient of 0.6 was used. The vent area includes the construction opening in the J wall which is assumed to remain open.

G. The time after main steam line rupture to reach the various setpoints was established through thermohydraulic modeling of the system with the postulated breaks. Various times were established for the trip signals depending on which assumed active flow component failed.

H. Only heat losses to the compartment walls, ceiling and floor were considered.

A.4.3 Analytical Methods

The methods used in this analysis is the same as the first technique described in Section A.3.2. That is, computer program PRTHRUST was used to model the feedwater-steam-reactor system and to establish leakage rates and energy levels for the postulated breaks and conditions. A model of the system is given in Figure A-9. The mass discharge rate and fluid enthalpy from the leak were then input to the CONTEMPT computer program to establish an energy/mass balance in the compartment volume. Figure A-10 is the model used for the CONTEMPT program. Based on the energy/mass balance, compartment temperature, pressure and relative humidity were established. Reactor trip, safety injection, and valve closure signals were established based on a series of preliminary PRTHRUST runs.

Three main steam pipe rupture conditions were evaluated since each gave different temperature versus time histories. The three runs analyzed were:

A. Break postulated to occur at node 4135 with the feedwater control valve assumed to fail (see Figure B-4).

B. Break postulated to occur at node 4135 with the main steam check valve assumed to fail.

MAY 10 1974

C. Break postulated to occur at node 4070 (underneath the impingement sleeve) with the feedwater control valve assumed to fail.

The postulated break location at node 4115 was not evaluated since a break at node 4135 is a double ended break and more severe.

A.4.4 <u>Results and Discussion</u>

The results of the analysis are given in Figures A-14 and A-15. The maximum temperature reached in compartment GW (outside the main break leakage jet) is 300°F but only exists for approximately two seconds. A semi-stabilized steady state temperature is reached after a relative short period of time. As shown in Figure A-14, both the magnitude and duration of this stabilized temperature are dependent on which flow component is assumed to fail. Since equipment capability of surviving high temperature exposure is dependent on exposure time all three cases are shown. Approximate temperatures and exposure times are summarized below.

Assumed Active Component Failure	Average Temperature (°F)	Exposure Time (Sec)
Main Steam Check Valve Failure	215	50
Feedwater Control Valve Failure	200	120
Postulated Break in Sleeve with Feedwater Control Valve Failure	180	320

The relative humidity of the compartment, Figure A-15, reaches 100% within 10 seconds and remains at this value.

A.5 BUILDING FLOODING ANALYSIS

The purpose of the building flooding analysis is to verify that no safety related equipment becomes submersed in water following a rupture of high energy piping. In addition, no conduits required for safe shutdown of the plant should be exposed to a high temperature environment for such a time as to become inoperative. This evaluation consists of determining which buildings and compartments are affected following a break; the resulting water levels, temperature, and flow paths; and the eventual means of drainage.

A.5.1 Areas Affected

Ruptures of feedwater piping leads 1 and 2 between the containment and their entrance into the turbine building (through column row 15) would result in the water draining directly onto the ground just outside of containment area F. From this location a portion of the water would flash, while the remainder would drain off towards the ocean and be absorbed into the soil.

A rupture of feedwater leads 3 and 4 between the containment and their entrance into the turbine building (via a penetration through J wall) would occur in area GW at the 115' elevation. Water would drain onto the floor and subsequently also flow into area GE. It is proposed that water will be prevented from flowing into the auxiliary building (area K) and the fuel handling building (area J) by closed doors. Drainage is provided by a 13 foot wide opening through J wall in area GW overlooking the pipeway. Water would flow down the side of J wall and subsequently drain towards the ocean. If the water level in area GW (or GE) becomes greater than 4 inches, sleeving around piping floor penetrations would permit water to drain down to areas GW and GE at the 100' elevation. In addition, if the water level becomes greater than 8 inches, water will overflow the curb separating the containment annular space from areas GW and GE and follow the containment side down to the 85' elevation.

A rupture of the feedwater piping between walls 15 and J and the G row anchor would result in the water draining into the turbine building. Thus a rupture in this area or a rupture of the feedwater piping within the turbine building both result in the same areas being affected. The water will drain down to the 85' floor where it will eventually drain into the turbine building sumps.

A.5.2 Criteria and Assumptions

A.5.2.1 Feedwater Ruptures Between the Containment and the Turbine Building

As discussed in Section 5.1, a rupture of the feedwater piping on the north side of the containment, in area F, would result in the water spilling directly onto the ground and hence result in no flooding problem.

In area GW, design basis ruptures of the feedwater piping are postulated to occur at the locations shown in Figure 7-12. At break locations 1303, 1305, 1329, 1331, 1403, 1405, 1429, and 1431, it is proposed that the piping be

MAY 10 1974

sleeved (see Figure B-6); thus water could only flow out of the annular space between the process pipe and the sleeve. The majority of the ejected water would flash, with the resulting water levels on the floor being minimal. For design basis ruptures of the 6 inch feedwater tees (nodes 1347 and 1441 - see Figure B-6) the resulting water level would again be small.

A design basis rupture of the feedwater piping at the containment anchors (nodes 1300 and 1400 for leads 3 and 4, respectively) would thus result in the most severe flooding problem for areas GW and GE. Although the flued head anchors for nodes 1300 and 1400 are actually located inside the containment, a rupture on the feedwater pump side of the anchors would result in water being ejected into area GW. For the worst case flooding analysis, the feedwater is assumed not to flash and flow in its entirety onto the 115 foot elevation floor.

Water from the steam generator side of the break would flow until the water level in the steam generator drops below the sparger inlet. In addition, the piping is assumed to completely drain. The flow rate is determined during the hot standby condition from the short term blowdown analysis (Appendix A, Section A.2) and is taken subsequent to flow choking at the exit plane (this occurs at approximately 0.1 sec). The resulting flow rate of 110 ft^3/sec is assumed to remain constant until the entire water volume of 605 ft^3 is discharged.

Drainage from the feedwater pump side of the break will continue until the isolation valves (nodes 1317 and 1417 for lines 3 and 4, respectively - see Figure B-6) and control valves (nodes 1353 and 1447 for lines 3 and 4, respectively) close. Signal actuation for valve closure is taken as 2 sec, per page 7.3-5 of Reference 4. The feedwater control valve will then close within 5 sec upon receipt of the actuation signal (per page 9, Reference 16). The motor operated isolation valves are assumed to close in 54 seconds.

Postulating the worst case active component failure, the control valve is assumed to fail in the open position. Thus, blowdown from the pump side of the break can occur for a time period of 56 seconds (2 sec actuation time plus 54 sec closure time) providing offsite power is not lost. For the worst case analysis used herein, another 8 seconds of blowdown was added to account for the additional time required to start up the diesel generators (Reference 4, page 8.3-13). For conservatism, flow rate during blowdown is taken from the results of the short term blowdown analysis, following choking at the feedwater flow nozzle. The resulting flow rate is 33.5 ft^3/sec.

Floor areas of compartments GW and GE were determined from Reference 17. Drainage areas around piping penetrations through the 115' elevation floor were determined by on-site measurements. A summary of analysis data and assumptions is presented in Table A-8.

A.5.2.2 <u>Feedwater Line Rupture Within the Turbine Building</u>

A design basis rupture of the feedwater piping within the turbine building would essentially result in the same flooding conditions regardless of the break location. Water would be ejected from the steam generator side of break as a result of line pressure. The quantity of water is conservatively assumed to be that volume contained between the feedwater check valves and the feedwater pumps (2750 ft^3). Water was also conservatively assumed to be discharged from the feedwater pumps until the hotwells (19,000 ft^3, per Section 10.4.1 of Reference 4) and the heater #2 drain tank (1460 ft^3, per Reference 17) are drained. At this time, the condensate pump would be tripped as a result of low pump suction head. Thus the total volume of water available for discharge into the turbine building is 23,210 ft^3.

As discussed in Section A.5.1 this water will drain down to the floor at the 85' elevation. From this location it will drain into the turbine building sump volumes, with perhaps a small quanity flowing into the Unit 2 turbine building. From Reference 17 and as a result of field measurements, it was determined the Unit 1 total sump volume is 62,476 ft^3. Thus the entire quantity of discharge water can be accomodated in the sumps.

A.5.3 <u>Method of Analysis</u>

Flow rate of water into area GW was determined as in Section 5.2.1 above. The transfer rates from area GW to GE (and vice versa) and from GW through

MAY 10 1974

the 13' opening in J wall were determined by application of Bernoulli's and continuity formulas for rectangular channel flow.

$$\dot{Q} = \int_{o}^{h} C_D \, bVdy \qquad \text{(Continuity)}$$

$$V^2 = 2g_c y \qquad \text{(Bernoulli)}$$

where:

$\dot{Q}$ = discharge rate

b = channel width

g = gravitational constant

h = water height

C_D = discharge coefficient = 0.6

y = dy flow element head

dy = flow element thickness

Combining equations and integrating

$$\dot{Q} = \int_{o}^{h} C_D \, b \, (2gy)^{1/2} \, dy$$

$$\dot{Q} = C_D \, b\sqrt{2g} \int_{o}^{h} y^{1/2} \, dy$$

MAY 10 1974

$$\dot{Q} = 2/3\ C_D\ b\ \sqrt{2g_c}\ \ h^{3/2}$$

The above expression is used to determine the volume of water which is discharged through the construction opening out into the pipeway area. Similarly, the flow of water from area GW to area GE can be represented as

$$\dot{Q}_{GW\ to\ GE} = 2/3\ C_D\ b\ \sqrt{2g_c}\ \left(h_{GW}^{3/2} - h_{GE}^{3/2}\right)$$

For water drainage over the 4 inch high piping penetration sleeves, and over the 8 inch curb separating the containment annular space from areas GW and GE, the following formula was used:

$$\dot{Q} = C_D A\ \sqrt{2\ g_c\ (h - h_d)}$$

where:

A = drainage area

h_d = drain height

C_D = discharge coefficient

This equation can be derived from the Bernoulli and continuity equations, similar to channel flow. A discharge coefficient of 0.6 was used in both flow rate equations. The resulting water levels in areas GW and GE were then determined through the integration of the equation:

MAY 10 1974

$$h(t) = \frac{\int_0^t (\dot{Q}_{in} - \dot{Q}_{out})\, dt}{A_{floor}}$$

A.5.4 Discussion of Results

The resulting water levels in compartments GW and GE are shown in Figure A-16. Immediately following the break, water from each section of the severed piping flows into area GW. The water level rises as the input flow exceeds drainage. Water also flows through the contracted doorway into area GE, but its level lags that of GW. Finally at 64 seconds, drainage through the 13' opening in J wall exceeds volumetric input from the severed piping and the water level in GW begins to subside. The water level in GE will continue to rise until it reaches that of GW, after which time it too will subside. Although some water (approximately 820 ft^3) will overflow the 4" high piping floor penetration sleeves and enter GW at the 100' elevation, this water is expected to flash and not result in any significant depths at this lower elevation. The drainage provided by the 13' opening in J wall will result in early all the ejected water spilling onto the ground, where it does not affect any safety related equipment.

A rupture of the feedwater piping within the turbine building will eventually result in all the discharged water draining into the sump volumes below the 85' elevation. Thus the 85' floor will not be covered by any significant height of water for any period of time.

MAY 10 1974

TABLE A-1

MAIN STEAM LINE BLOWDOWN REACTION LOAD SUMMARY

Case No.	Node (PG&E/NSC No.)	Location	Break Type (Area)	Max. Force (kips)
1A	3-50/3295	At containment penetration	S.G. side ($3.687\ ft^2$)	615
1B	3-50/3295	At containment penetration	Turbine side ($3.687\ ft^2$)	670
1C	3-50/3295	At containment penetration	Longitudinal ($3.687\ ft^2$)	616
2	3-40/3273	Branch to pressure relief valves	Branch ($2.655\ ft^2$)	500
3	3-84/3550	In relief valve branch	Branch ($0.330\ ft^2$)	70.0
5	3-28/3170	At elbow close to J wall	Turbine side ($3.687\ ft^2$)	565
7A	3-5/3050	At G wall anchor	S.G. side ($3.687\ ft^2$)	755
7B	3-5/3050	At G wall anchor	Turbine side ($3.687\ ft^2$)	755
8	75/ 3580	At turbine inlet	Longitudinal ($3.687\ ft^2$)	719
9	3-10/3060	Branch to PCV header	Branch ($1.485\ ft^2$)	273
10	--/1640	Dump line break near turbine	S.G. side ($2.028\ ft^2$)	338
10B	--/1640	Dump line break near turbine	Dump side ($3.221\ ft^2$)	493

MAY 10 1974

TABLE A-2

FEEDWATER LINE BLOWDOWN REACTION LOAD SUMMARY

Case No.	Node (PG&E/NSC No.)	Location	Break Type (Area)	Max. Force (kips)
2F	115/1300	At containment penetration	Heater side ($1.118\ ft^2$)	162
2G	115/1300	At containment penetration	S.G. side ($1.118\ ft^2$)	162
4F	130/1329	At elbow upstream of isolation valves	S.G. side ($1.118\ ft^2$)	162
5F	130/1329	At elbow upstream of isolation valves	Heater side ($1.118\ ft^2$)	162
6F	130/1331	At elbow upstream of isolation valves	Longitudinal ($1.118\ ft^2$)	162
7F	139A/1347	6" bypass line	Branch ($0.181\ ft^2$)	44
8F	50/1369	Lead 4 split from Lead 3	Branch ($1.118\ ft^2$)	166
9F	10/1375	Lead 3 & 4 split from Lead 1 & 2	S.G. side ($2.391\ ft^2$)	345
10A	5/1000	At G wall anchor	Heater side ($4.13\ ft^2$)	598
10B	5/1000	At G wall anchor	S.G. side ($4.13\ ft^2$)	598
10C	5/1000	At G wall anchor	Longitudinal ($4.13\ ft^2$)	595
10D	10/1375	Lead 3 & 4 split from Lead 1 & 2	Branch ($2.391\ ft^2$)	344

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1

SUBJECT MAIN STEAM BLOWDOWN ANALYSIS

SHEET________ OF________

TABLE A-3

Operating Conditions: Hot Standby

Break Type & Direction: Circumferential Break at Node 3050
Steam Generator Side of Break - Case 7A

VOLUME DATA

Vol. No.	Description	Volume (ft^3)	Pressure (psia)	Temp. (°F)	Quality
1	Steam Generators 1-1, 1-2, and 1-4	17274	1020	---	0.04
2	Volume from Steam Generators 1-1, 1-2 & 1-4 Up to Header	5761	1020	---	1.0
3	Main Steam Cross Tie Header	244	1020	---	1.0
4	Steam Generator 1-3 Up to Header	1447	1020	---	1.0
5	Steam Generator 1-3	5758	1020	---	1.0
6	Break Volume - From Break to Header	14	1020	---	0.04
7	Volume of Leads 1, 2 & 4 from Header through Dump Cross Tie to Break	422	1020	---	1.0

DD-01-006
MAY 10 1974

TABLE A-4

Operating Conditions: Hot Standby

Break Type & Direction: Circumferential Break at Node 3050
Steam Generator Side of Break - Case 7A

JUNCTION DATA

Inlet Vol. No.	Outlet Vol. No.	Junction Inertia-L/A (ft^{-1})	Minimum Flow Area (ft^2)	Initial Flow (lbm/sec)	Friction Factor(k) $k \times 10^6$	Comments
1	2	14.04	4.185	0.0	1.05	Flow Nozzle in Leads 1, 2 & 4
2	3	21.47	4.454	0.0	1.2	Leads 1, 2 & 4 Connect. to Main Steam Header
3	4	36.015	1.485	0.0	10.66	Lead 3 Connection to Main Steam Header
5	4	36.015	1.395	0.0	10.46	Flow Nozzle in Lead 3
4	6	0.495	1.485	0.0	0.134	---
6	0	0.495	3.687	0.0	0.0	Break at G wall anchor
3	7	15.52	1.485	0.0	6.68	Leads 1, 2 & 4 Connect. to Main Steam Header
7	0	15.52	3.687	0.0	0.0	Opposite Side of Circ. Brk. at G Wall Anchor

PROJECT DIABLO CANYON UNIT 1

SUBJECT FEEDWATER BLOWDOWN ANALYSIS

TABLE A-5

Operating Conditions: Hot Standby

Break Type & Direction: Longitudinal Break at Node 1000 - Case 10C

VOLUME DATA

Vol. No.	Description	Volume (ft^3)	Pressure (psia)	Temp. (°F)	Quality
1	Steam Generator 1-3	5758	1020	-	0.04
2	Steam Generator 1-1	5758	1020	-	0.04
3	Steam Generator 1-2	5758	1020	-	0.04
4	Steam Generator 1-4	5758	1020	-	0.04
5	Steam Generator 1-3 to Tee Node 1375	243	1020	432	---
6	Steam Generator 1-2 to Tee Node 1163	234	1020	432	---
7	Steam Generator 1-1 to Reducer Elbow Node 1191	467	1020	432	---
8	Steam Generator 1-4 to Tee Node 1369	158	1020	432	---
9	Break Volume From Node 1191 to Node 1014	312	1020	432	---
10	Line From Node 1014 to Heater Volume Node 1042	365	1020	432	---
11	Heater Volume	700	1020	432	---
12	Pump Discharge to Heater Volume Nodes 660 & 500 to Node 586	945	1020	432	---
13	Water Supply Reservoir	10000	620	432	---

DD-01-006 MAY 10 1974

PROJECT DIABLO CANYON UNIT 1

SUBJECT FEEDWATER BLOWDOWN ANALYSIS

TABLE A-6

Operating Conditions: Hot Standby

Break Type & Direction: Longitudinal Break at Node 1000 - Case 10C

JUNCTION DATA

Inlet Vol. No.	Outlet Vol. No.	Junction Inertia-L/A (ft^{-1})	Minimum Flow Area (ft^2)	Initial Flow (lbm/sec)	Friction Factor(k) $k \times 10^5$	Comments
1	5	83	0.442	0	180.3	Sparger Area
2	7	72	0.442	0.0	147	Sparger Area
3	6	94	0.442	0.0	152	Sparger Area
4	8	66	0.442	0.0	149	Sparger Area
8	5	66	0.358	0.0	93.8	Flow Nozzle
5	9	83	0.358	0.0	125	Flow Nozzle
9	0	0.61	4.13	0.0	0	Break
6	7	191	0.358	0.0	190	Flow Nozzle
7	9	29	0.358	0.0	0.94	Flow Nozzle
9	10	25.7	4.13	0.0	0.387	---
10	11	16.7	2.391	0.0	0.722	---
11	12	26.5	2.391	0.0	0.664	---
13	12	20.5	2.391	0.0	0.45	---

TABLE A-7

MAXIMUM ACTIVATION TIME/SETPOINT SUMMARY ASSUMED FOR ENVIRONMENT ANALYSIS

	Time to Receive Signal (Sec.)	Time to Complete Function (Sec.)	Total Time (Sec.)	Time Used (Sec.)	Setpoint* Condition
Steam Isolation Check Valve Closure	--	0.05 to 0.20	0.05 to 0.20	0.20	1
Turbine Stop Valve Closure	1.20	>0.10	>1.30	1.30	2
Auxiliary Feedwater Pumps Started					
With off-site power	2.00	Up to 50.	2.0 to 52.0	2.00	2
With loss of off-site power	10.00	Up to 50.	10.0 to 60.0		
Reactor Scram	0.90	Up to 1.8	Up to 2.7	2.70	2
Feedwater Control Valve Closure	2.00	5.50	7.50	7.50	2
Steam Isolation Valve Closure					
High Mass Flow/Low Pressure Setpoint	2.00	5.50	7.50	7.50	3
High Mass Flow/Low Temperature Setpoint	6.00	5.50	11.50	11.50	4
Main Feedwater Pumps Stopped	3.50	Up to 59.0	3.5 to 62.5	62.50	2
Feedwater Isolation Valve Closure					
With off-site power	2.00	54.00	56.00		
With loss of off-site power	10.00	54.00	64.00	64.00	2

*SETPOINT CONDITIONS:

1. Flow reversal in pipe to activate check valve
2. Differential pressure of 148 psi between any one of four lines
3. High mass flow of 1251.8 lb/sec (at full load) in two of four lines plus low pressure of 567 psia in two of four lines
4. High mass flow of 1251.8 lb/sec (at full load) in two of four lines plus low average coolant temperature of 538°F

MAY 10 1974

TABLE A-8

SUMMARY OF FLOODING ANALYSIS DATA

Input Flow Rates:

Steam Generator Side

Flow Rate = 110.0 ft^3/sec

Flow Time = 5.51 sec

Pump Side

Flow Rate = 33.5 ft^3/sec

Flow Time = 64 sec

Floor Areas:

Area of GW = 3641 ft^2

Area of GE = 2684 ft^2

Drain Areas - Area GW:

16.0 ft^2 at 4" above floor level

78.3 ft^2 at 8" above floor level

Additional Drains:

13.0 ft channel from GW through J wall overlooking pipeway

3.375 ft doorway from GW to GE

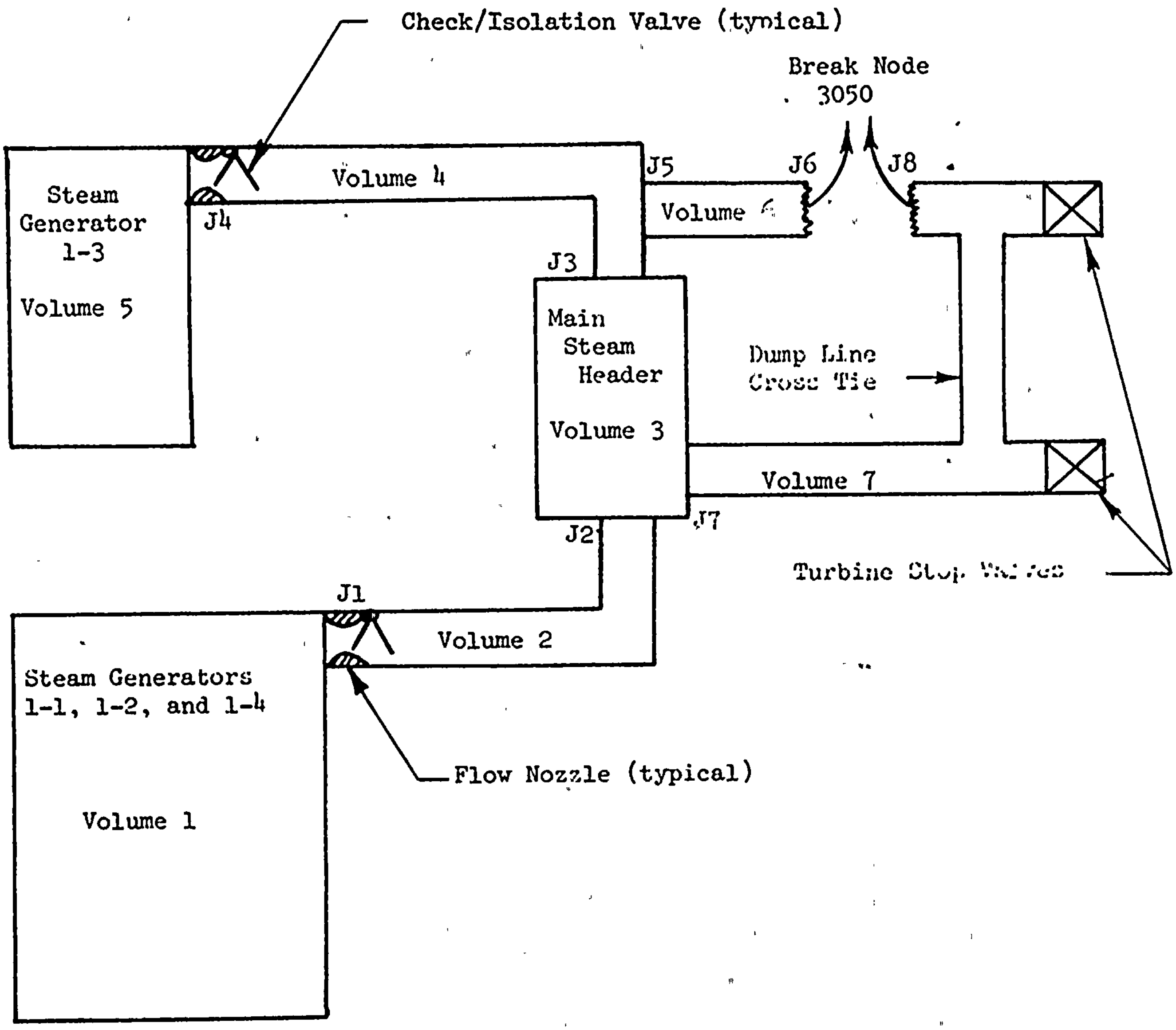
Discharge Coefficients for Drains:

C_D = 0.6 for all drains areas and channels

MAY 10 1974

FIGURE A-1

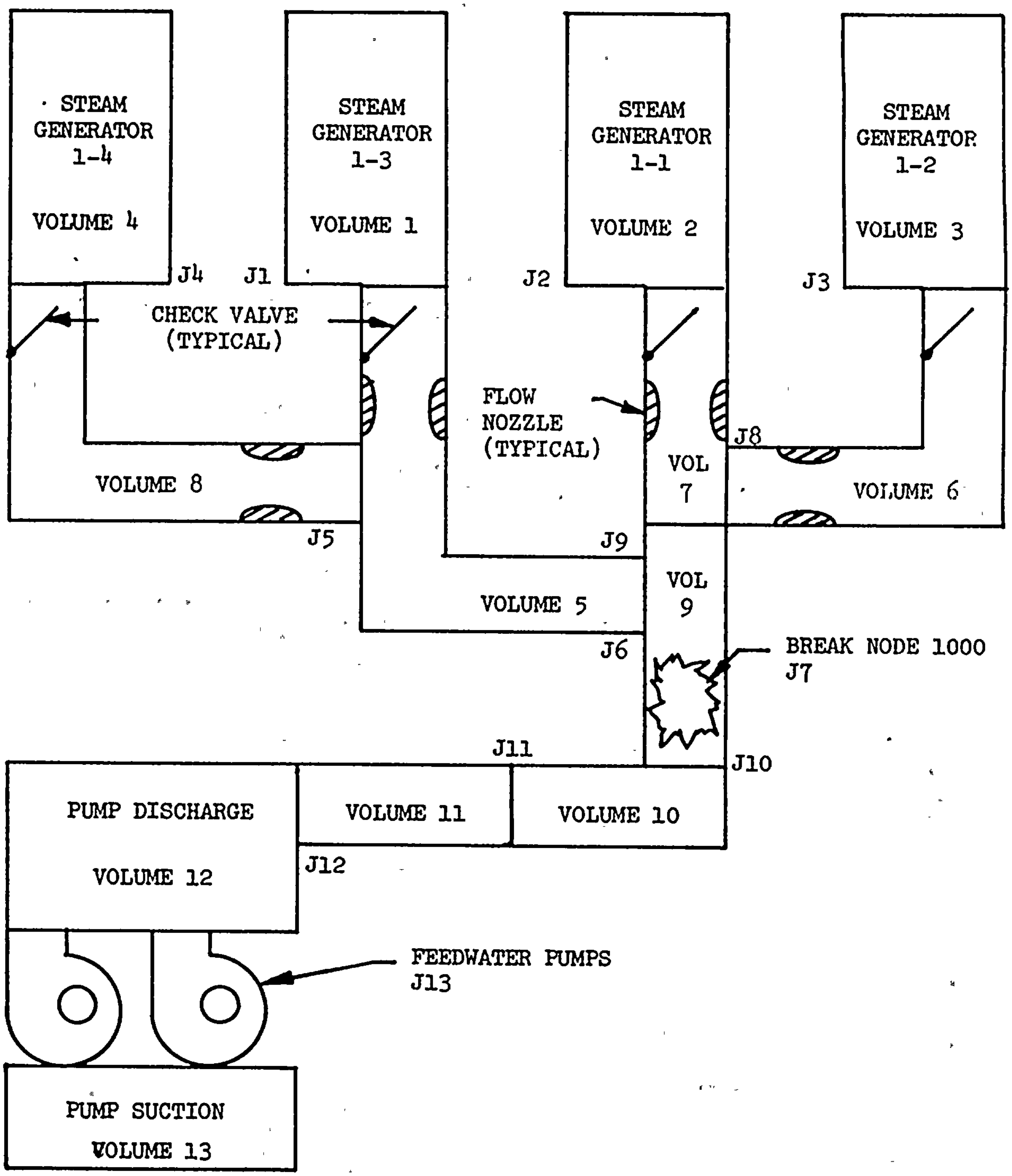
DIABLO CANYON UNIT 1
MAIN STEAM SHORT TERM BLOWDOWN ANALYSIS
CIRCUMFERENTIAL BREAK AT NODE 3050



MAY 10 1974

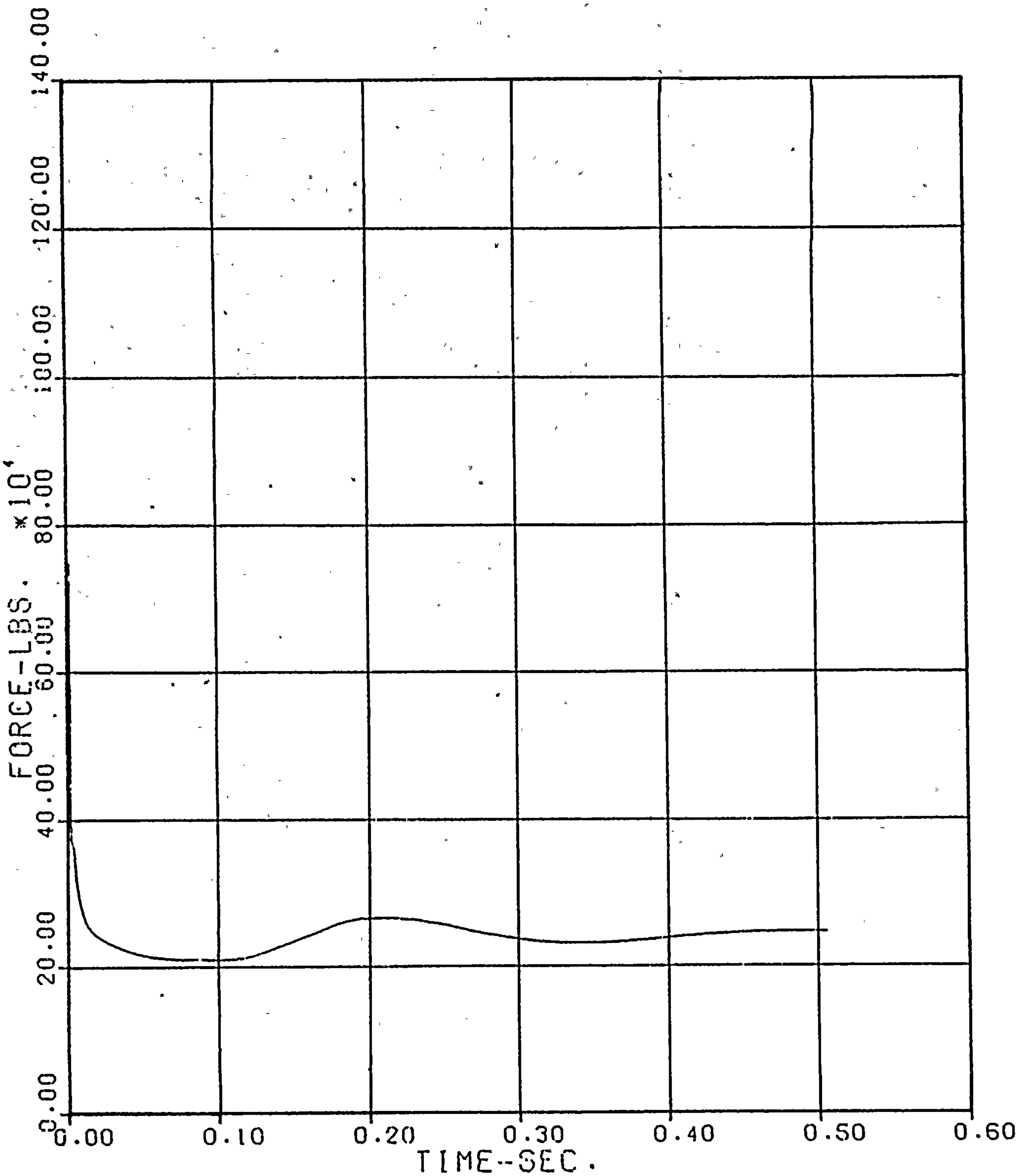
FIGURE A-2

DIABLO CANYON UNIT 1
FEEDWATER SHORT TERM BLOWDOWN ANALYSIS
LONGITUDINAL BREAK AT NODE 1000



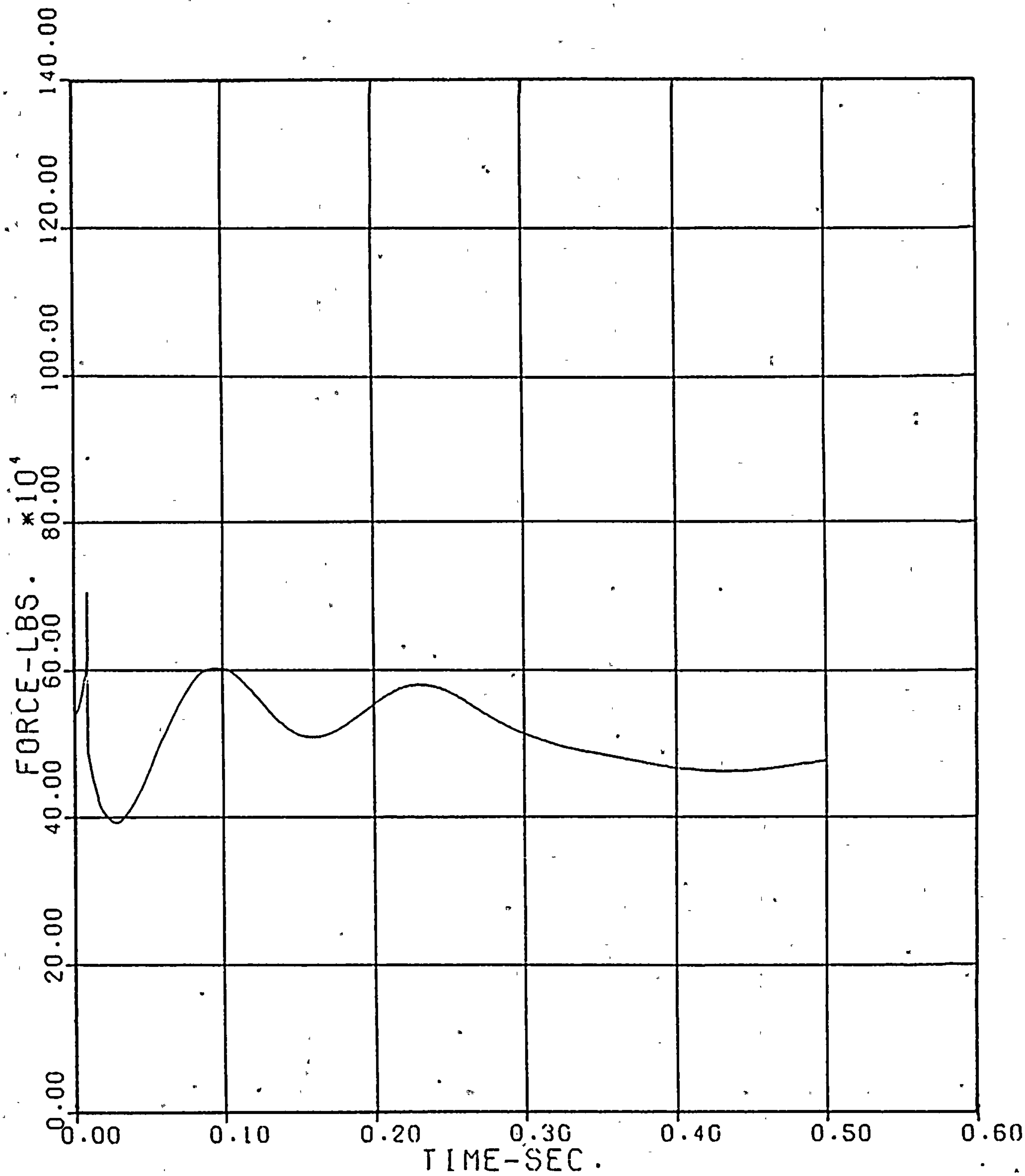
MAY 10 1974

FIGURE A-3. MS HOT STANDBY REACTION LOAD CASE 7A CIRC. BREAK-
NODE 3050 STEAM GENERATOR SIDE



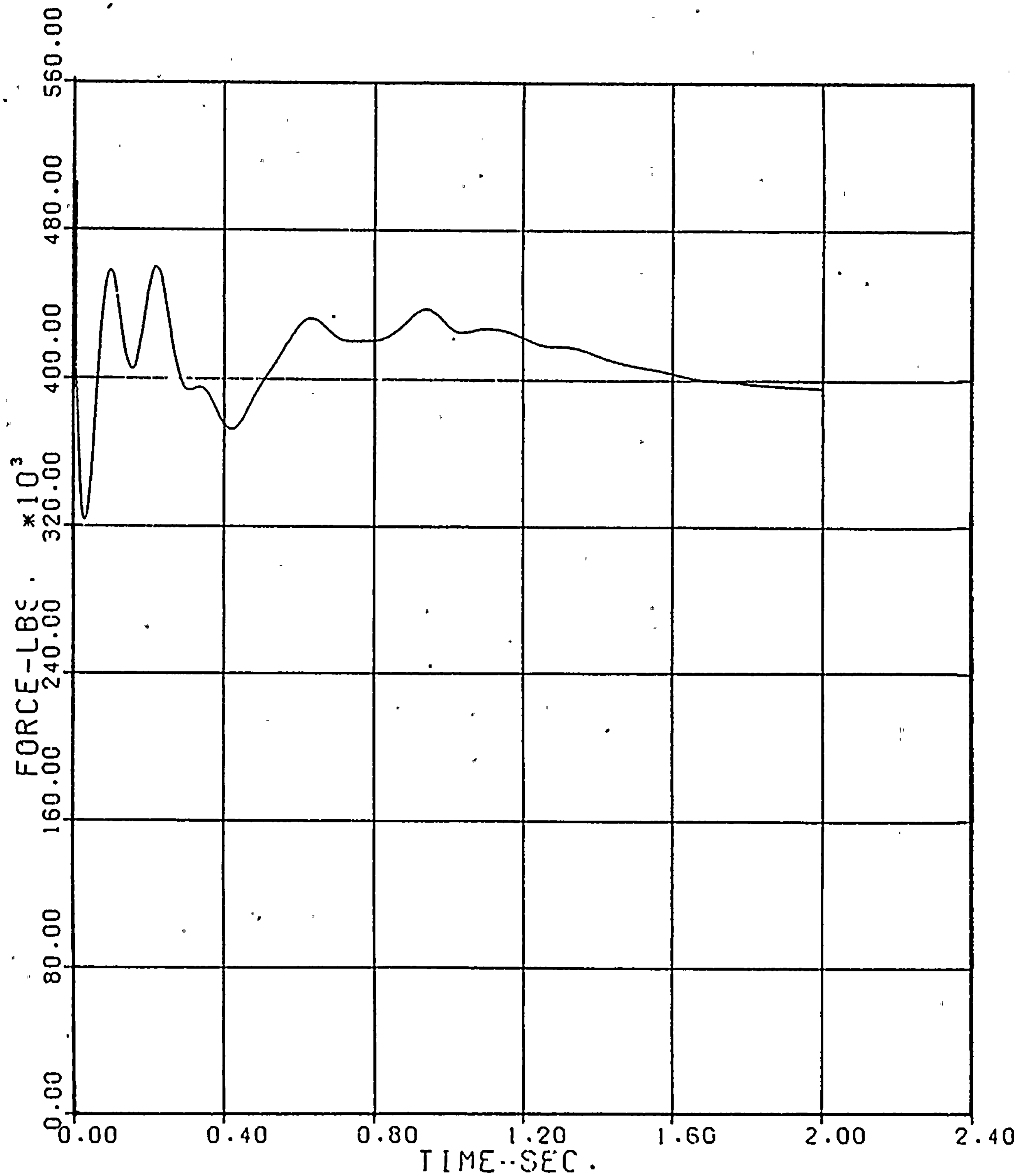
MAY 10 1974

FIGURE A-4. MS HOT STANDBY REACTION LOAD CASE 1C LONG. BREAK - NODE 3295



MAY 10 1974

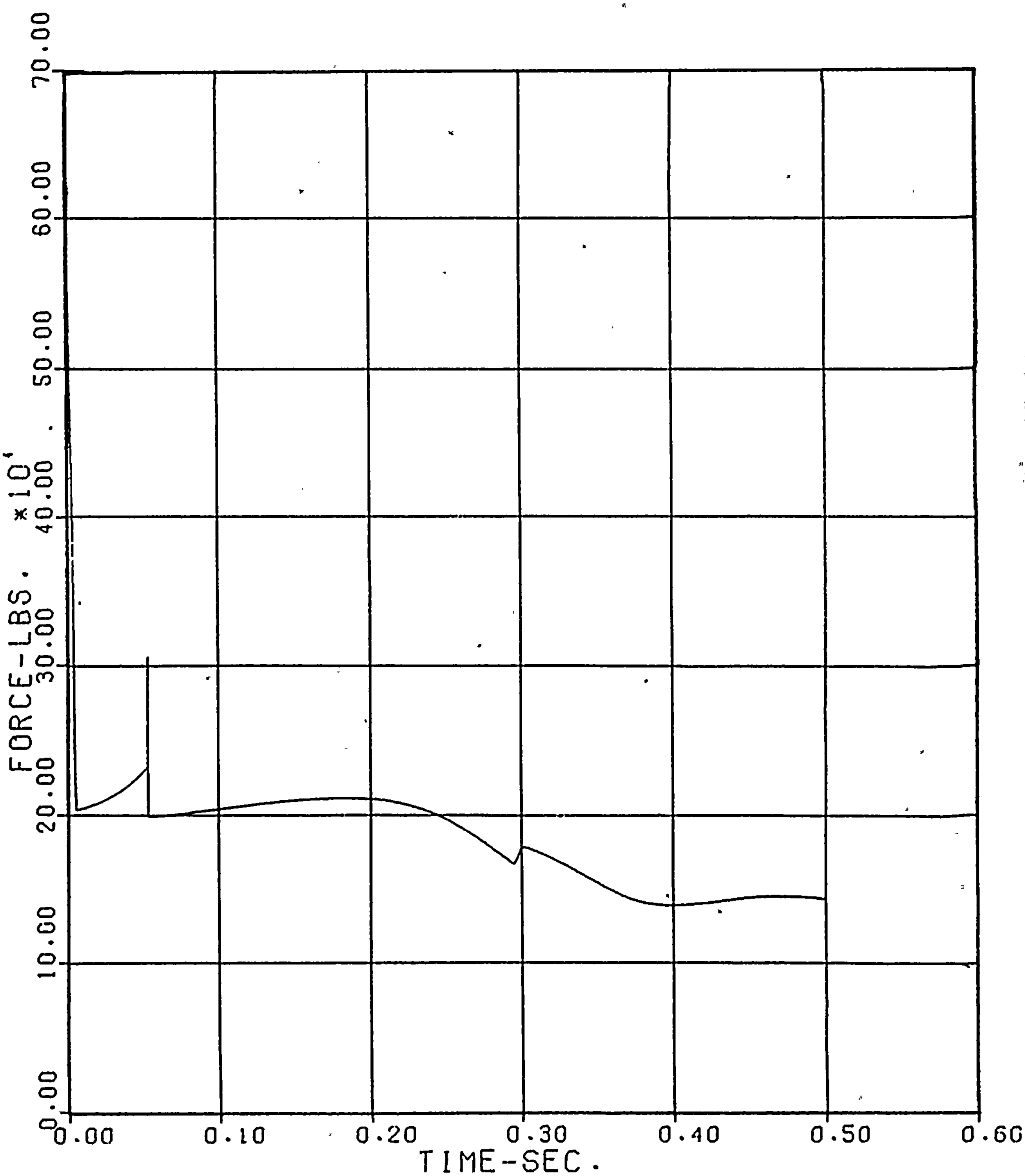
FIGURE A-5. MS HOT STANDBY REACTION LOAD CASE 2 BRANCH BREAK - NODE 3273



MAY 10 1974

Nuclear Services Corporation

FIGURE A-6. FW HOT STANDBY REACTION LOAD CASE 10B CIRC. BREAK-NODE 1000 STEAM GENERATOR SIDE



'MAY 10 1974

FIGURE A-7. FW HOT STANDBY REACTION LOAD CASE 10C LONG. BREAK - NODE 1000

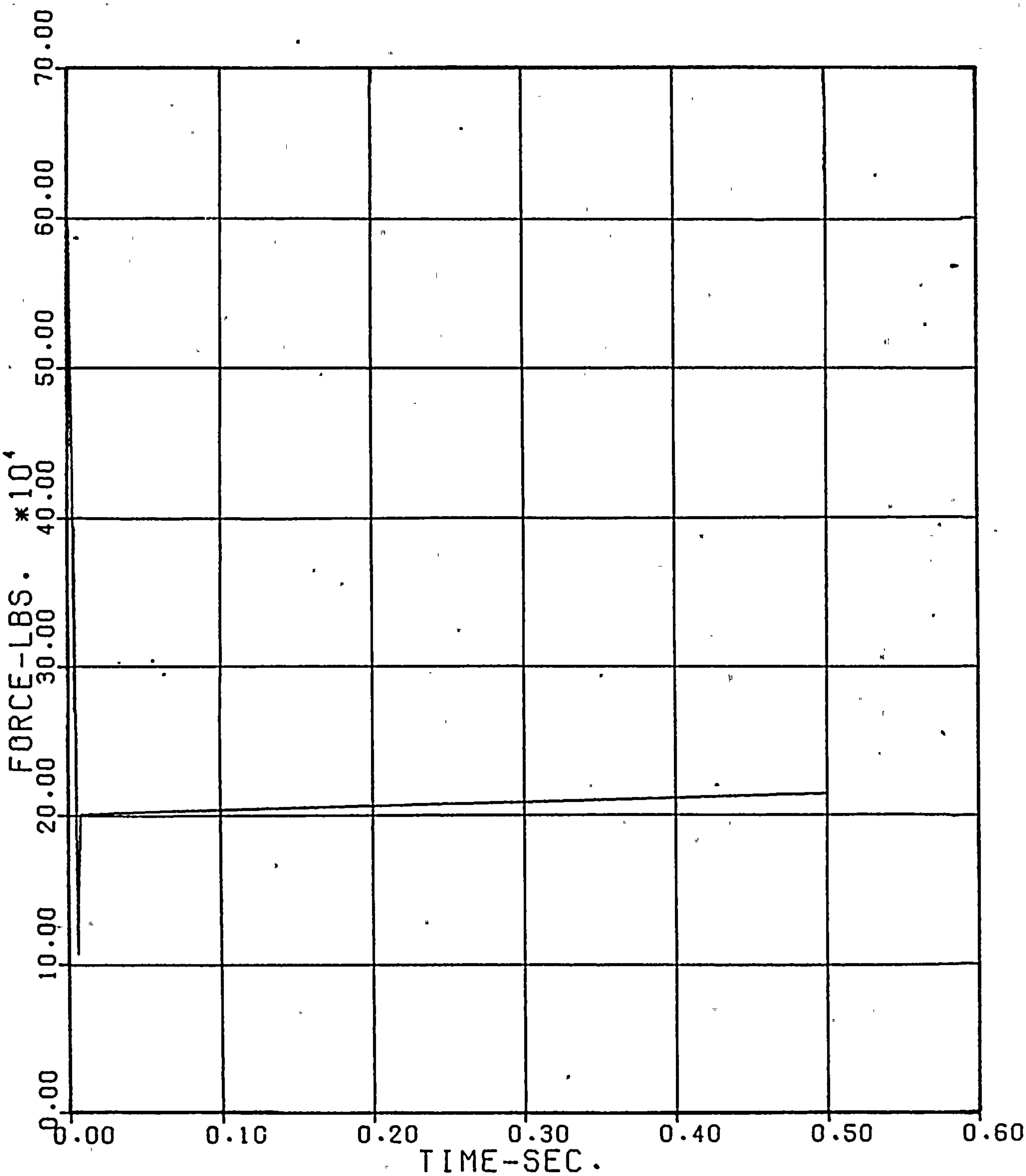
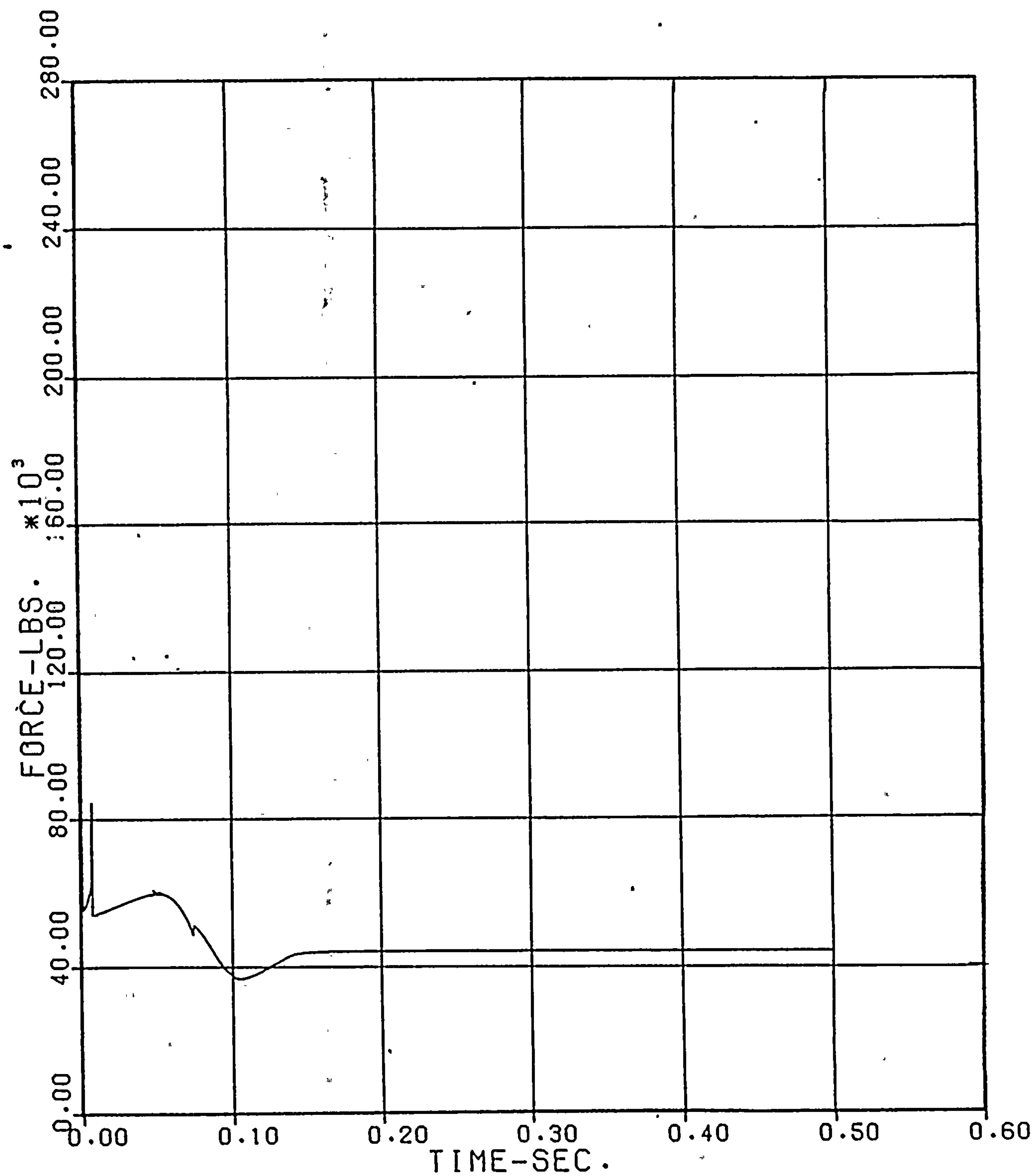


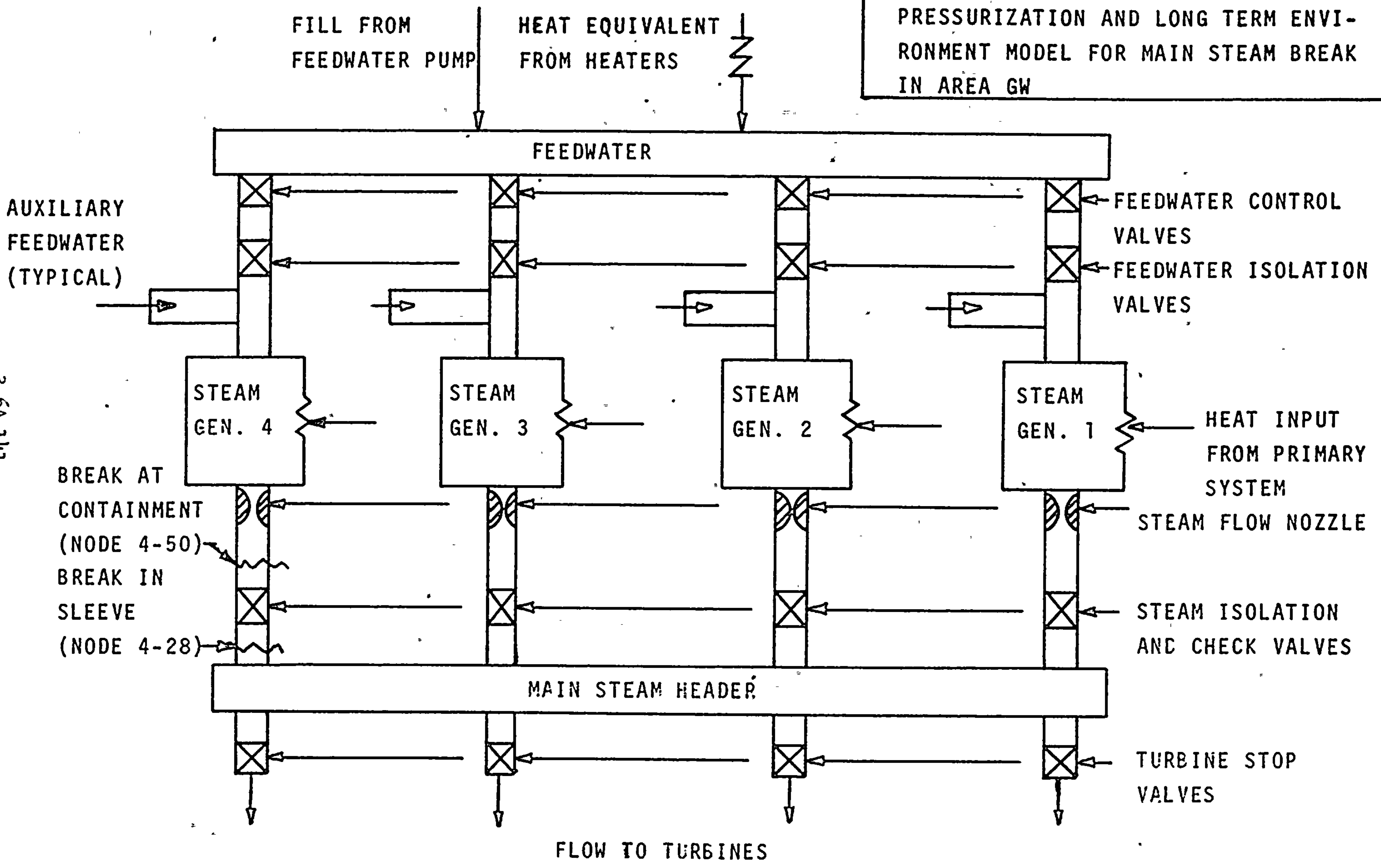
FIGURE A-8. FW HOT STANDBY REACTION LOAD CASE 2F CIRC. BREAK - NODE 1300 HEATER SIDE



MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-9
PRESSURIZATION AND LONG TERM ENVIRONMENT MODEL FOR MAIN STEAM BREAK IN AREA GW

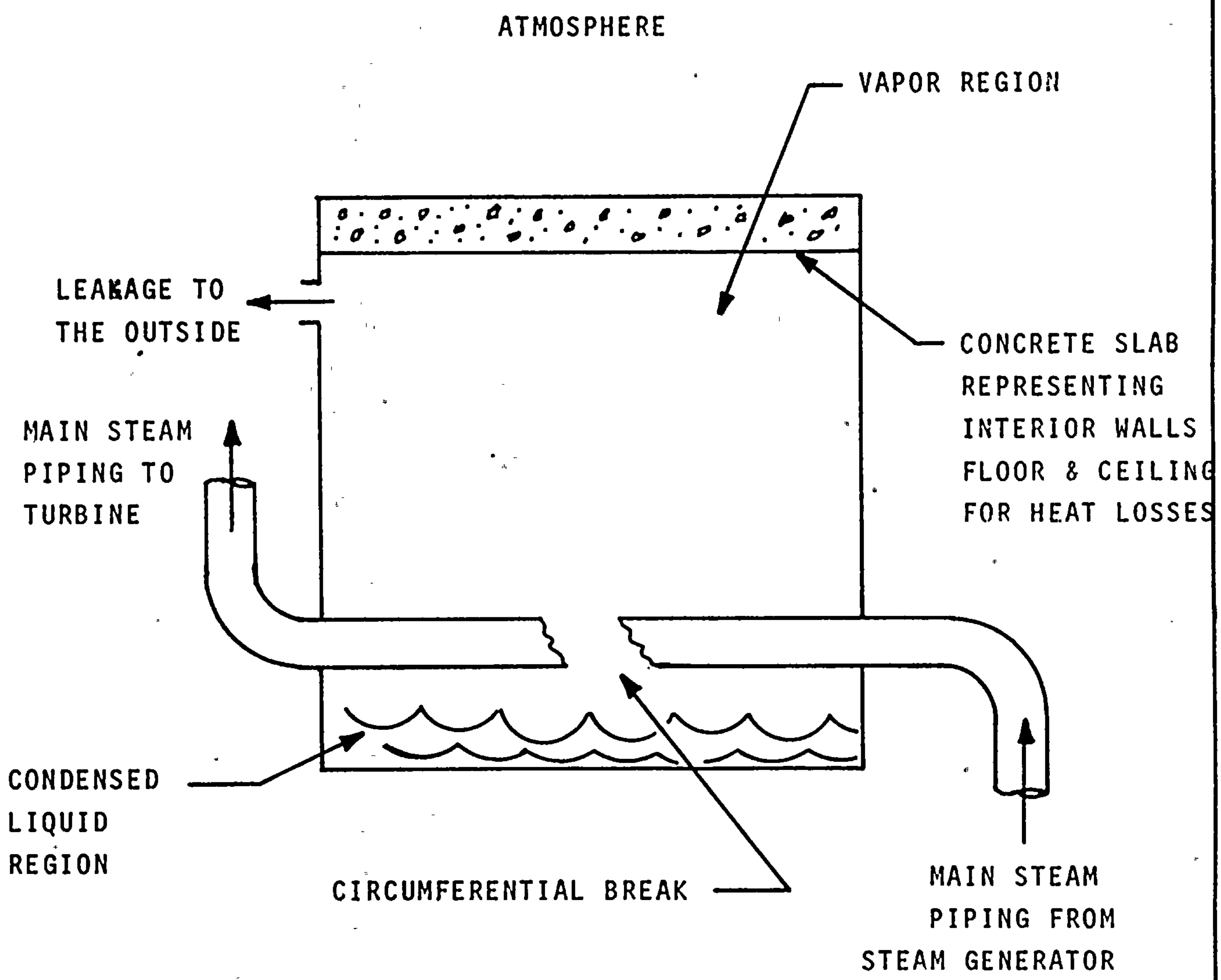


3.6A-141

MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-10
PRESSURIZATION AND LONG TERM ENVIRONMENT MODEL - MAIN STEAM LINE BREAK IN AREA GW AT 115 FOOT ELEVATION

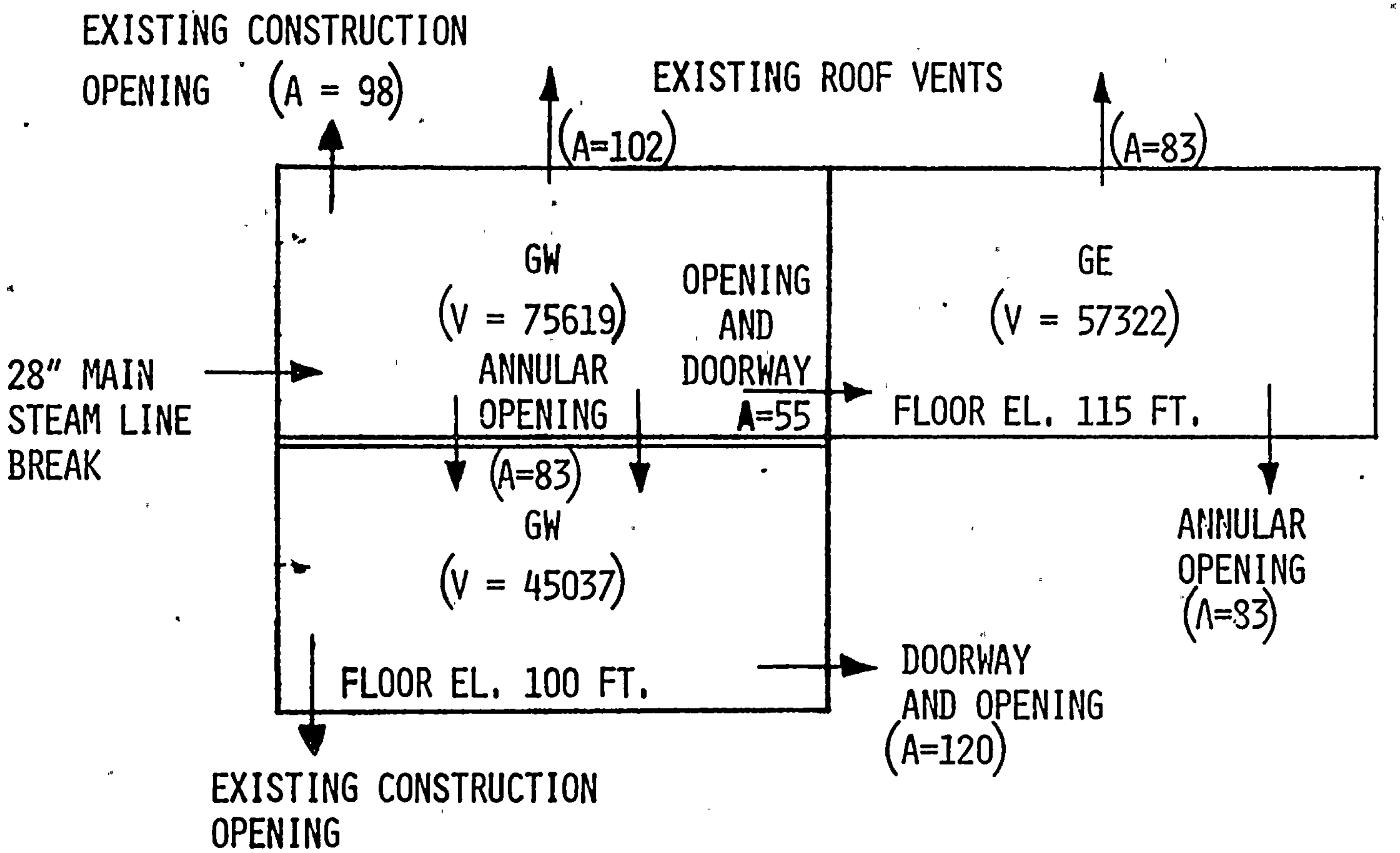


MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-11 COMPARTMENTS MODELED ON PRTHRUST LONG TERM BLOWDOWN RUN

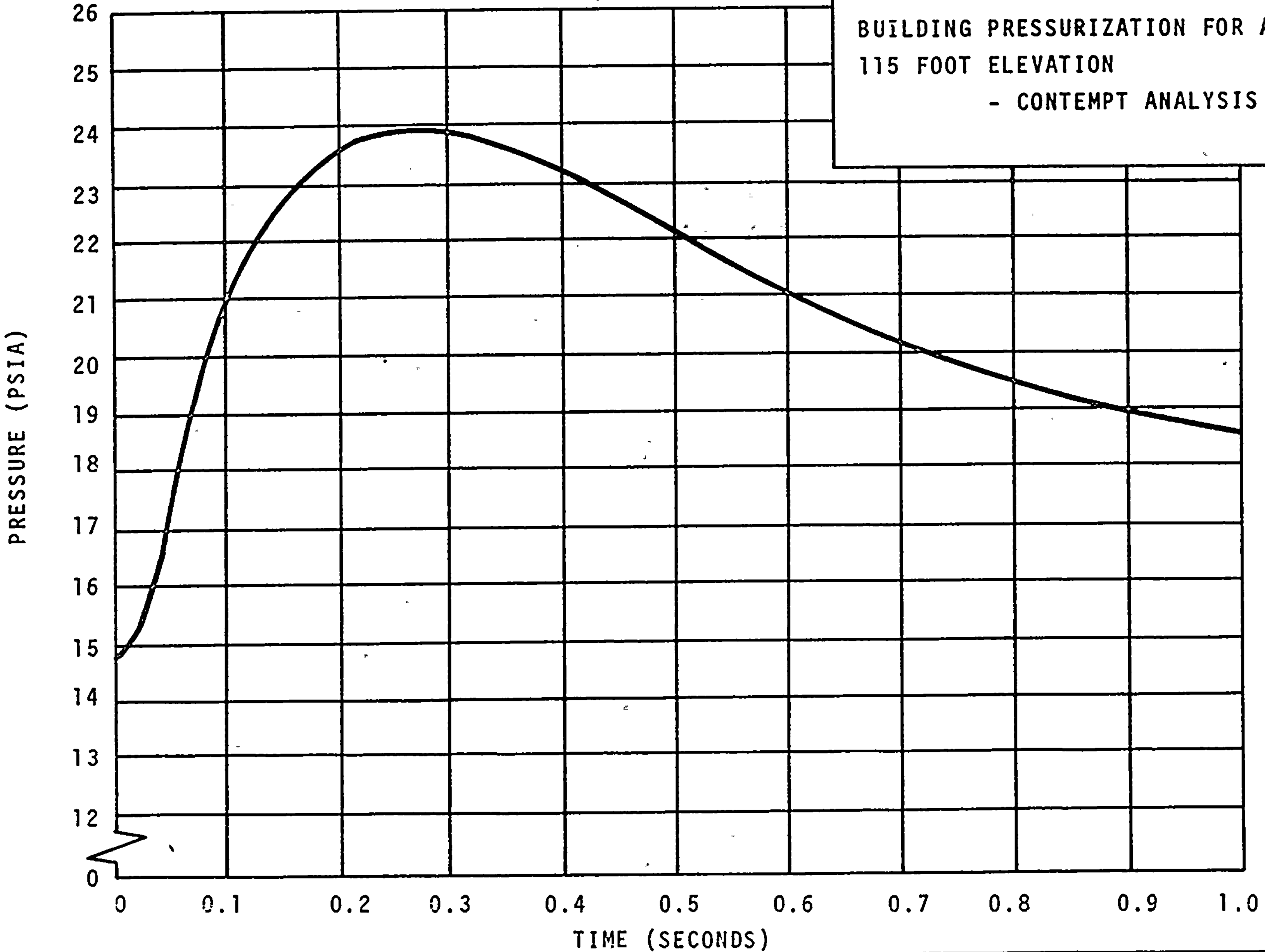
V = VOLUME (CUBIC-FT)
A = AREA (SQ-FT)



MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-12
BUILDING PRESSURIZATION FOR AREA GW AT 115 FOOT ELEVATION
- CONTEMPT ANALYSIS -

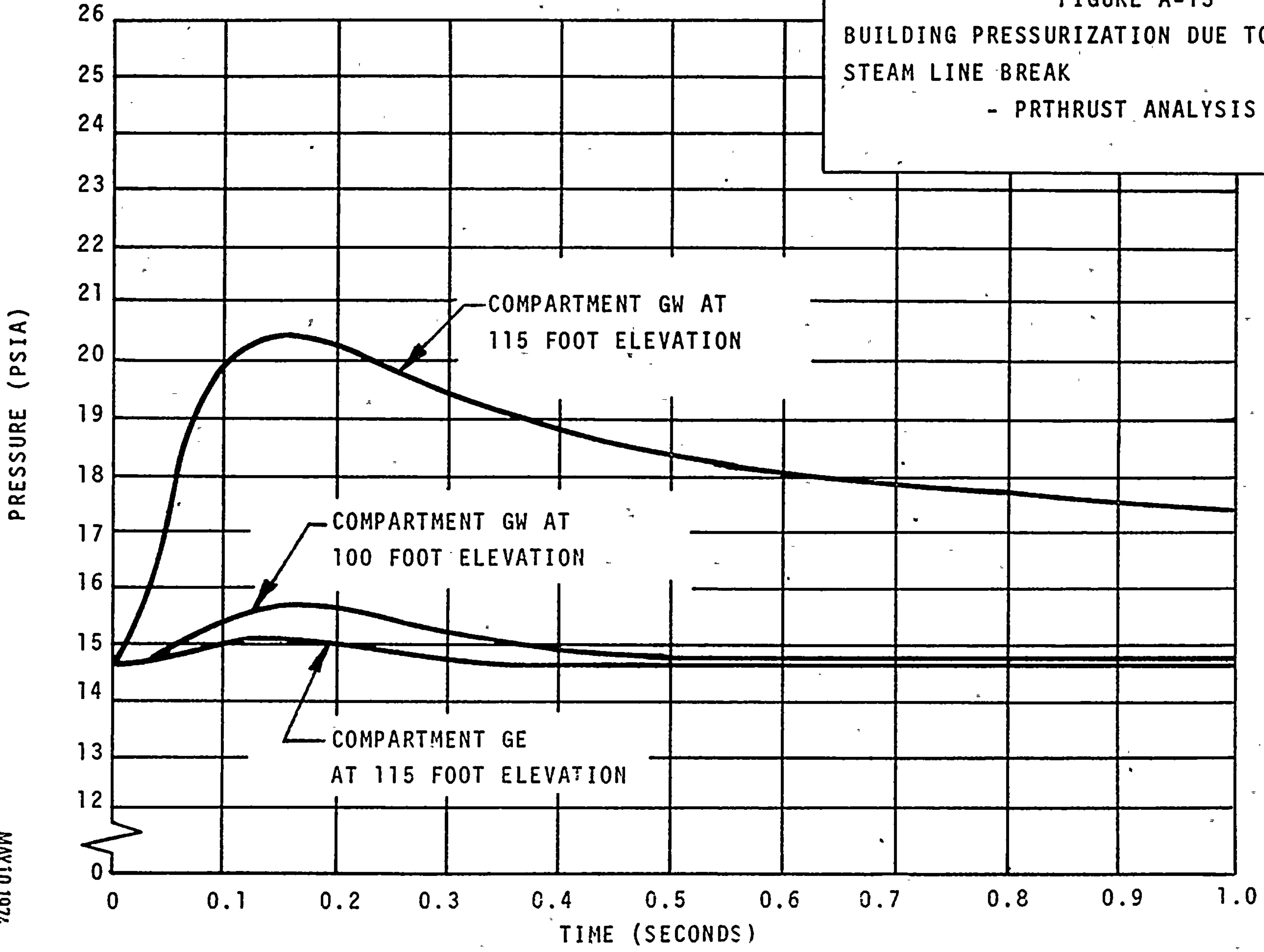


MAY 10 1974

3.6A-144

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-13
BUILDING PRESSURIZATION DUE TO 23" MAIN STEAM LINE BREAK
- PRTHRUST ANALYSIS -



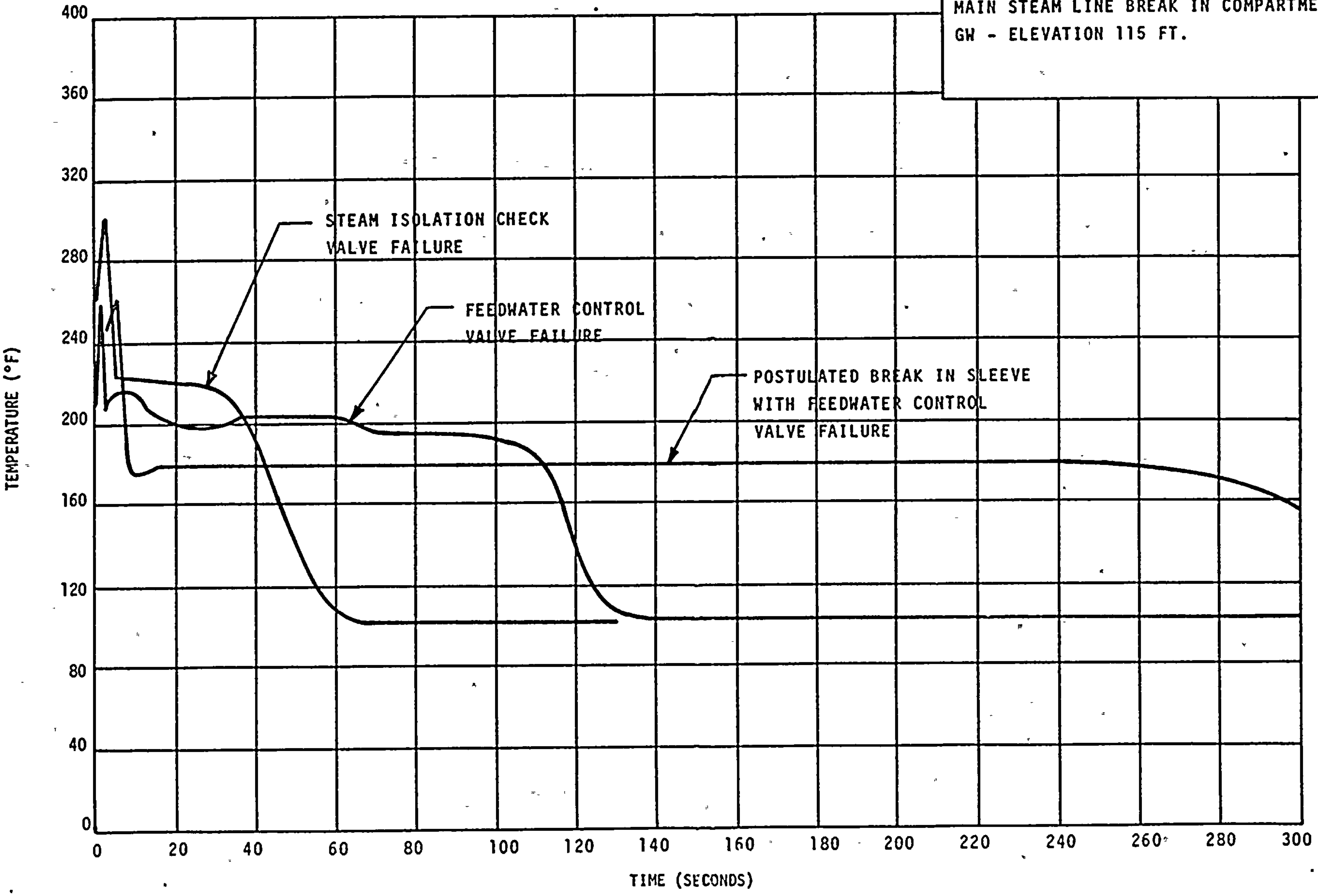
3.6A-145

MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-14
PREDICTED VAPOR TEMPERATURE HISTORY FOR MAIN STEAM LINE BREAK IN COMPARTMENT GW - ELEVATION 115 FT.

MAY 10 1974

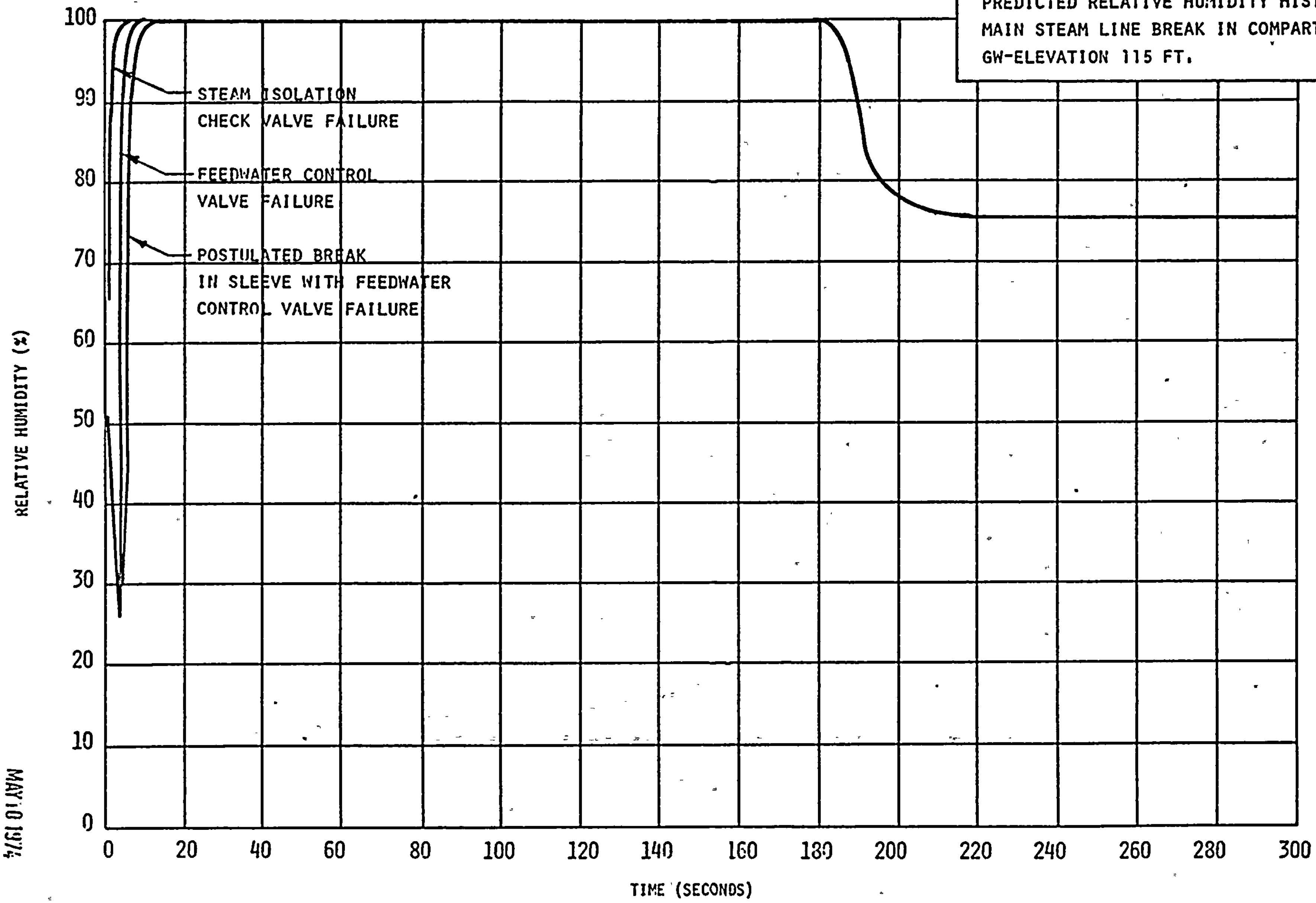


3.6A-146

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE A-15

PREDICTED RELATIVE HUMIDITY HISTORY FOR MAIN STEAM LINE BREAK IN COMPARTMENT GW-ELEVATION 115 FT.

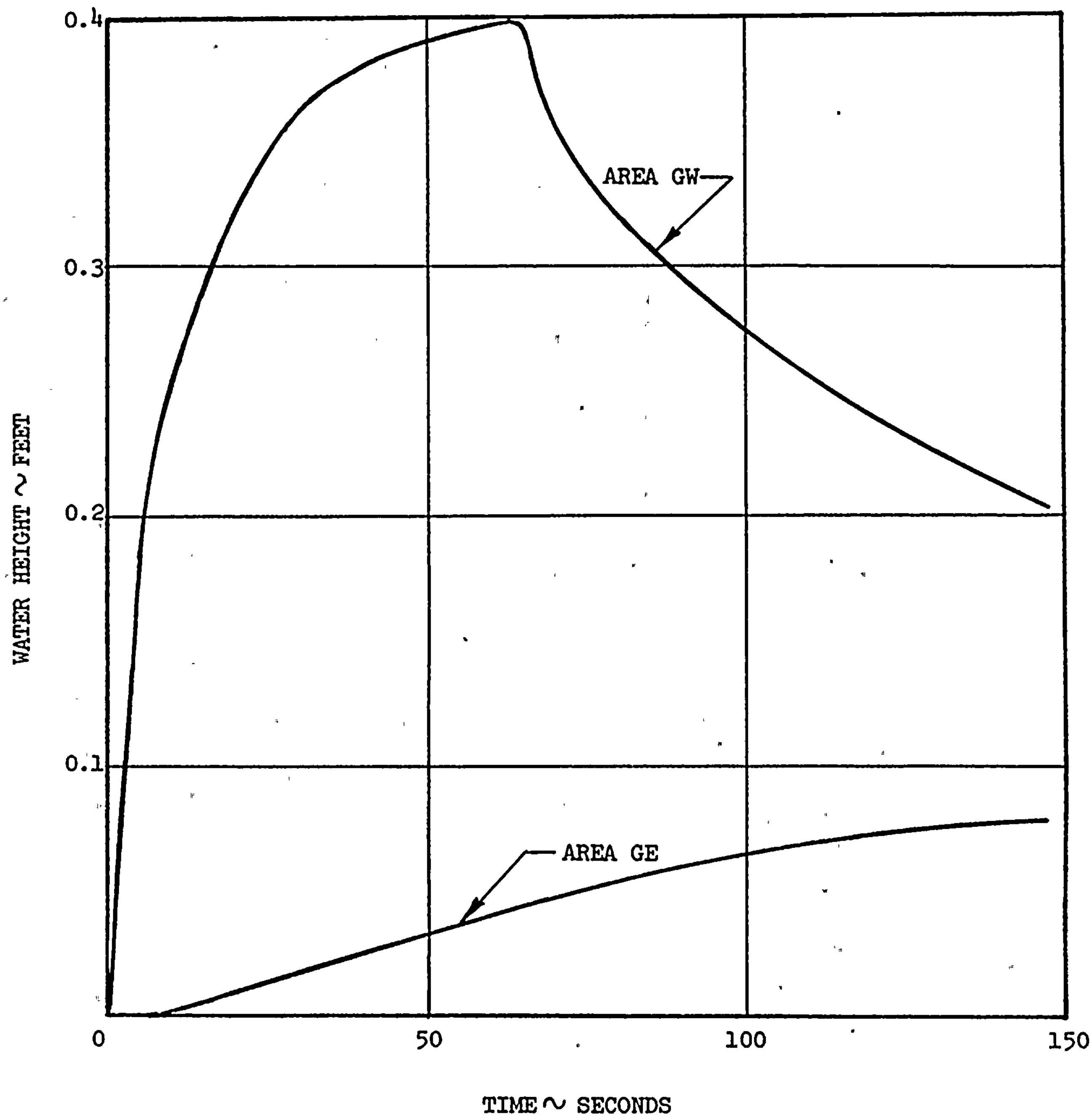


3.6A-147

MAY 10 1974

FIGURE A-16 BUILDING FLOODING ANALYSIS - AREAS GW AND GE, 115' ELEVATION

WATER LEVELS RESULTING FROM A FEEDWATER PIPE RUPTURE



APPENDIX B

PIPE WHIP ANALYSIS

MAY 10 1974

TABLE OF CONTENTS

		Page
B.1	INTRODUCTION	3.6A-153
	B.1.1 Main Steam and Feedwater Piping Containment to G Row Anchor	3.6A-153
	B.1.2 Main Steam and Feedwater Piping Turbine Building	3.6A-154
B.2	METHOD OF ANALYSIS	3.6A-155
	B.2.1 Pipe Whip Analysis	3.6A-155
	B.2.2 Upper and Lower Bound Material Properties	3.6A-158
	B.2.3 Structural Yield/Failure Criterion	3.6A-158
B.3	RESULTS OF PIPE WHIP ANALYSES	3.6A-162

MAY 10 1974

MAY 10 1974

B.1 INTRODUCTION

This section presents the criteria employed, selected input data and typical results of pipe whip analyses conducted on both the main steam and feedwater piping systems.

B.1.1 Main Steam and Feedwater Piping - Containment to G Row Anchor

Analyses were performed for break locations shown on Figures 7-3 and 7-12 for main steam and feedwater lines, respectively. The determination of these break locations is described in Section 7.1.1, and an assessment of the potential damage associated with these break locations is given in Sections 7.1.2 and 7.2.2.

Analyses were run on all feedwater lines and main steam lines 1, 3, and 4. Since all postulated break locations on main steam line 2 coincide with break locations on line 1, the similarity of the lines allows the extrapolation of results from line 1 to be representative of line 2.

The location of piping restraints are shown on Figures B-1 through B-4 for the main steam lines and Figures B-5 and B-6 for the feedwater lines. The restraint locations shown in these figures include both existing restraints (per Reference 20) and those restraints it has been proposed to construct.

In this appendix, detailed input and results are shown for typical analyses. Circumferential breaks at locations 1329 and 1375 on feedwater line 3, and the circumferential break at location 3295 on main steam line 3 are the three representative cases chosen.

B.1.2 Main Steam and Feedwater Piping - Turbine Building

As described in Section 7.1.1.1, locations of design basis breaks within the turbine building are postulated at each location of potential high stress or fatigue.

The purpose of the restraint system within the turbine building is to prevent essential structures and equipment from being damaged to an unacceptable degree as a result of an uncontrolled pipe whip. The procedure to be used for this evaluation was defined in Section 7.1.2.3. Locations where restraints will be required and structural designs for these restraints are currently being evaluated. When determined, the locations and properties of these restraints will be incorporated in the pipe whip mathematical models (Figures B-6 and B-7).

B.2 METHOD OF ANALYSIS

B.2.1 Pipe Whip Analysis

The pipe whip analyses were conducted with the use of the computer program PIPERUP (Reference 8). The PIPERUP computer program performs non-linear elastic-plastic analysis of three dimensional piping systems subjected to concentrated static or dynamic time history forcing functions. These forces result from fluid jet thrust at the location of a postulated longitudinal or circumferential rupture of high energy piping. The program computes and outputs support reactions, internal forces, moments, and system deflections as a function of time. In addition, strains in each section of pipe which have exceeded the yield criterion are also printed.

PIPERUP is an adaptation of the finite element method to the requirements of pipe rupture analyses. The continuous piping is mathematically modeled as an assembly of weightless structural members connecting discrete nodal points. Nodal points are placed in such a manner as to isolate particular types of piping elements, such as straight runs of pipe, valves, elbows, etc., for which force-deformation characteristics can be determined. Nodal points are also placed at all discontinuities, such as piping restraints, branch lines, and changes in cross section. Weight of the system including distributed weight of the piping and concentrated weights (valves, etc.) is lumped at selected system mass points.

An incremental procedure is used to account for the non-linear effects of plastic deformation of the pipe and restraints.

Stress-strain characteristics of the members which connect node points are idealized by three linear segments. The first portion represents linear and perfectly elastic behavior, the second represents the linear strain hardening, with the third portion representing perfectly plastic behavior (Figure B-9).

The method used in the program to account for the change in piping stiffness during plastic deformation is to represent each member connecting two nodes by three subelements in parallel, whose total stiffness equals the elastic stiffness of the pipe. Upon transition from the elastic to the linear strain hardening region, one of the three subelements is hinged, such that it can sustain no increase in load, leaving two subelements which are defined to have a sum stiffness equal to the strain hardening stiffness of the pipe. At the second transition the process is repeated leaving a single subelement with a very small stiffness. Prediction of plastic collapse is based on detection of excessive deflections.

In situations where stress reversal and unloading occurs, an isotropic strain hardening model is used; i.e., unloading is always along the elastic line (Figure B-9).

MAY 10 1974

Pipe restraints are modeled in PIPERUP with an initial gap and a tri-linear stiffness curve. Again, the first stiffness represents linear elastic behavior, the second stiffness models linear strain hardening, and the third stiffness modeling perfectly plastic behavior.

The program uses stiffness and mass proportional damping as follows:

$$(C) = DAMPM * (M) + DAMPK * (K)$$

where (C), (M) and (K) are the viscous damping, mass and stiffness matrices, respectively, and DAMPM and DAMPK are input constants determining the degree of damping desired. These constants can be related to the critical damping system by the following equation:

$$\lambda = \frac{DAMPM}{2\omega i} + DAMPK \frac{\omega i}{2}$$

where λ is the fraction of critical damping and ω is the natural angular frequency of mode i. Inspection of this equation will reveal that DAMPM and DAMPK can be adjusted to meet a given damping criteria, say 2% of critical, at any two desired frequencies, and that a (conservatively) lower damping value would result between these frequencies. In addition, the effect of DAMPM is greater at lower frequencies and the effect of DAMPK is greater at higher frequencies. Experience has shown that for many piping systems the greatest piping response is in the lower frequencies and optimum solution stability is obtained by specifying sufficient damping for these frequencies.

B.2.2 Upper and Lower Bound Material Properties

Consistent with variations in material stress-strain data, pipe whip analyses are performed utilizing both upper and lower bounds of the range of piping material properties. Typically lower bound material properties are conservative for the prediction of pipe hinge formation; while either upper or lower bound properties may yield maximum restraint loads, depending on the size of the restraint gaps. Therefore, at each postulated break location where a pipe whip analysis is to be performed, a judgement evaluation is made as to whether upper bound, lower bound, or both pipe material properties should be used.

B.2.3 Structural Yield/Failure Criterion

The yield and pipe hinge criteria used by PIPERUP is given by:

$$\alpha i^2 M^2 + (\beta/4) T^2 = \sigma_M^2 Z_e^2$$

where

M = Resultant moment in piping

$= \left(M_1^2 + M_2^2\right)^{\frac{1}{2}}$

T = Resultant torque

$\sigma_M = \sigma_y$ = Yield stress for first stage hinge (strain hardening)

$\sigma_M = \sigma_u$ = Ultimate stress for second stage hinge (plastic hinge)

Z_e = Pipe elastic section modulus

$$Z_e = \frac{\pi\left(r_o^4 - r_i^4\right)}{4r_o}$$

r_o, r_i = Pipe outer, inner radius

i = Stress intensification factor for elbows.

α, β = Constants which depend on material properties and pipe geometry

The method used to derive α and β material constants was to assume elastic/linear strain hardening material properties. In this case, failure in bending alone is given by:

$$M_{ult} = \sigma_y Z_p + \left(\sigma_u - \sigma_y\right) Z_e$$

where

Z_p = Plastic section modulus

$= (4/3)\left(r_o^3 - r_i^3\right)$

Substituting into the program failure equation (with $i = 1$) one obtains,

$$\alpha\left(M_{ult}\right)^2 + 0 = \left(\sigma_u Z_e\right)^2$$

Solving,

$$\alpha = \left(\frac{\sigma_u Z_e}{\sigma_y Z_p + \left(\sigma_u - \sigma_y\right) Z_e}\right)^2$$

The above derivation of α is utilized for both upper and lower bound material properties.

MAY 10 1974

The torsional rupture stress, (F_{st}) as a function of pipe geometry and material strength is used to determine failure in torsion alone. The ultimate torsion strength is determined from $T_u = F_{st} Z_{te}$. Substituting into the program failure equation and noting $Z_{te} = 2Z_e$

$$\frac{\beta}{4} T_u^{\;2} = \left(\sigma_u Z_e\right)^2$$

therefore,

$$\beta = \left(\frac{\sigma_u}{F_{st}}\right)^2$$

While the above derivation for β is used with lower bound properties to obtain a minimum torsion strength, an alternate method is used for upper bound. The ultimate torsion strength for upper bound is determined to be:

$$T_u = \left(\frac{\sigma_y}{2}\right) Z_{tp} + \left(\frac{\sigma_u - \sigma_y}{2}\right) Z_{te}$$

where:

$$Z_{tp} = \text{plastic torsion modulus}$$

$$= \frac{2\Pi \left(r_o^{\;3} - r_i^{\;3}\right)}{3}$$

which ignores the effects of torsional buckling for conservatism. Then,

Z_p

$$\frac{\beta}{4} T_u^{\;2} = \left(\sigma_u Z_e\right)^2$$

MAY 10 1974

Substituting $Z_e = Z_{te}/2$, and solving for β

$$\beta = \frac{\left(\sigma_u Z_{te}\right)^2}{\left(\frac{\sigma_y}{2} Z_{tp} + \frac{\sigma_u - \sigma_y}{2} Z_{te}\right)^2}$$

B.3 RESULTS OF PIPE WHIP ANALYSES

Three runs are shown as representative of the analyses performed and are presented in detail here. These three analyses are: feedwater line 3, circumferential break at location 1329, steam generator side of break; feedwater line 3, circumferential break at location 1375, steam generator side of break; and main steam line 3, circumferential break at location 3295, turbine side of break.

The blowdown loads for the circumferential breaks at feedwater location 1375 and main steam line location 3295 are shown in Figures B-10 and B-11, respectively. Because of the proposed sleeve at location 1329 on the feedwater line 3, the blowdown force for this case was taken as shown in Figure B-12 with PA equal to 207 kips.

Locations of existing rupture restraints were taken from the contract drawings of Reference 20. In order to prevent all pipe whips of the main steam and feedwater piping between the containment and G row anchor it was found necessary to specify the addition of several new restraints. The locations of these additional restraints are included in the pipe whip mathematical models shown in Figures B-1 through B-6.

Values of gaps between the rupture restraints (both existing and added) and the piping were chosen on the basis of the piping thermal stress analyses (Reference 6). The values chosen were such that the restraints will not interfere with the piping during any of the normal operating conditions. The gaps thus derived were included in the pipe whip analyses performed.

Force deflection properties of all restraints were determined as part of the structural evaluation. The criteria and methods used for this evaluation are given in Appendix D. Typical results of this evaluation are also given in Appendix D, Tables D-5 and D-6. As shown in these tables, restraint behavior is modeled with a bilinear curve, with both elastic and strain-hardening properties. The restraint characteristics thus derived serve as input data to the PIPERUP program.

The results of the selected pipe whip analyses are presented in Tables B-1 through B-6 and Figures B-13 through B-15. Table B-1 is a table of the maximum restraint reactions developed throughout the analysis for the feedwater break at 1329. It should be noted that a restraint may load and unload several times during the course of an analysis, and the results presented in the tables represent the maximums of these sets of reactions. Table B-2 presents the maximum loads, deflections and strains experienced by the pipe during the duration of the analysis. Tables B-3 through B-6 present the maximum restraint loads and pipe behavior for the other two cases considered.

Along with tabular output, the PIPERUP program also generates plots of reactions and deflections of node points versus time. Examples of these plots are shown in Figures B-13 through B-15. Figure B-13 shows the restraint reaction at 1369 on feedwater line 3 due to the break at node 1375. Figure B-14 displays the restraint reaction at node 1309 on feedwater line 3 due to the break at node 1329. And Figure B-15 shows the deflection in the z direction at node 1329 on feedwater line 3.

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1

SUBJECT FEEDWATER PIPING

SHEET________**OF**________

TABLE B-1

PIPE RUPTURE RESTRAINT LOAD SUMMARY

Piping Run: Feedwater Leads 3 and 4 from Containment to G Row Anchor

Break Location & Direction: Circumferential Break at 1329 - Steam Generator Side

Pipe Properties: Upper Bound

Node	Restraint Type	Load Direction	Maximum Load
1300	Anchor	$\pm X$ Direction	24.6 kips
1300	Anchor	$\pm Y$ Direction	107.3 kips
1300	Anchor	$\pm Z$ Direction	163.5 kips
1300	Anchor	$\pm\theta_x$ Direction	7458.9 in-kips
1300	Anchor	$\pm\theta_y$ Direction	2915.6 in-kips
1300	Anchor	$\pm\theta_z$ Direction	2101.0 in-kips
1309	Frame	+X Direction	145.7 kips
1309	U-Bolt	-Z Direction	395.4 kips
1315	Frame	-Y Direction	755.1 kips
1327	Frame	-Y Direction	671.4 kips

MAY 10 1974

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1
SUBJECT FEEDWATER PIPING SYSTEM

SHEET______ OF______

TABLE B-2

PIPE RUPTURE PIPE LOAD SUMMARY

Piping Run: Feedwater Leads 3 and 4 from Containment to G Row Anchor

Break Location & Direction: Circumferential Break at 1329-Steam Generator Side

Pipe Properties: Upper Bound

PIPE: Node 1315 (Elbow)	
Maximum moment (kips in)	7556
Percent of moment carrying capacity at yield (M_y = 6.39 x 10^3 kips in)	118.2%
Percent of moment carrying capacity at ε_u (M_{ε_u} = 13.4 x 10^3 kip in)	56.2%
Maximum strain level (in/in)	0.0321
Percent of limit strain ($\varepsilon\mu$ = 0.16 in/in)	20.1%
Maximum deflection (in) X deflection at node 1315	2.165
Maximum deflection (in) Y deflection at node 1303	1.992
Maximum deflection (in) Z deflection at node 1329	-9.891

MAY 10 1974

TABLE B-3

PIPE RUPTURE RESTRAINT LOAD SUMMARY

Piping Run: Feedwater Leads 3 and 4 from Containment to G Row Anchor

Break Location & Direction: Circumferential Break at 1375 - Steam Generator Side

Pipe Properties: Upper Bound

Node	Restraint Type	Load Direction	Maximum Load
1300	Anchor	$\pm X$ Direction	29.1 kips
1300	Anchor	$\pm Y$ Direction	27.4 kips
1300	Anchor	$\pm Z$ Direction	16.8 kips
1300	Anchor	$\pm\theta_x$ Direction	2654.8 in-kips
1300	Anchor	$\pm\theta_y$ Direction	3653.4 in-kips
1300	Anchor	$\pm\theta_z$ Direction	692.7 in-kips
1309	Frame	+X Direction	138.2 kips
1327	U-Bolt	+X Direction	288.8 kips
1327	Frame	-Y Direction	54.3 kips
1332	Frame	-Y Direction	280.6 kips
1335	Frame	-Y Direction	338.4 kips
1335	Frame	+Z Direction	166.6 kips
1336	Frame	-Y Direction	237.6 kips
1336	Frame	+Z Direction	109.2 kips
1339	Frame	-Y Direction	149.3 kips
1343	Frame	-Y Direction	307.8 kips
1349	Frame	-Y Direction	153.4 kips
1356	Frame	+Y Direction	323.7 kips

MAY 10 1974

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1

SUBJECT FEEDWATER PIPING SYSTEM

SHEET______ OF______

TABLE B-3 (Cont'd)

PIPE RUPTURE RESTRAINT LOAD SUMMARY

Piping Run: Feedwater Leads 3 and 4 from Containment to G Row Anchor

Break Location & Direction: Circumferential Break at 1375 - Steam Generator Side

Pipe Properties: Upper Bound

Node	Restraint Type	Load Direction	Maximum Load
1356	Frame	+Z Direction	145.6 kips
1369	U-Bolt	+Y Direction	238.9 kips
1400	Anchor	±X Direction	13.1 kips
1400	Anchor	±Y Direction	14.2 kips
1400	Anchor	±Z Direction	12.4 kips
1400	Anchor	$\pm\theta_x$ Direction	1553.6 in-kips
1400	Anchor	$\pm\theta_y$ Direction	1831.8 in-kips
1400	Anchor	$\pm\theta_z$ Direction	519.5 in-kips
1415	Frame	+X, +Z Direction	159.4 kips
1427	U-Bolt	+X Direction	230.9 kips
1427	Frame	-Y Direction	57.4 kips
1433	Frame	-Y Direction	183.2 kips
1437	Frame	-Y Direction	170.9 kips
1437	Frame	+Z Direction	506.3 kips
1443	Frame	-Y Direction	1.7 kips
1443	Frame	+Z Direction	93.7 kips
1450	Frame	+Y Direction	152.3 kips
1450	Frame	+Z Direction	91.0 kips

MAY 10 1977

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1 SHEET______OF______

SUBJECT FEEDWATER PIPING SYSTEM

TABLE B-4

PIPE RUPTURE PIPE LOAD SUMMARY

Piping Run: Feedwater Leads 3 and 4 from Containment to G Row Anchor

Break Location & Direction: Circumferential Break at 1375-Steam Generator Side

Pipe Properties: Upper Bound

PIPE: Node 1363	
Maximum moment (kips in)	4740
Percent of moment carrying capacity at yield (M_y = 6.39 x 10^3 kips in)	74.1%
Percent of moment carrying capacity at ε_u (M_{ε_u} = 13.4 x 10^3 kip in)	35.3%
Maximum strain level (in/in)	0.0
Percent of limit strain ($\varepsilon\mu$ = 0.16in/in)	0.0
Maximum deflection (in) X deflection at node 1371	2.810
Maximum deflection (in) Y deflection at node 1375	5.652
Maximum deflection (in) Z deflection at node 1375	2.903

MAY 10 1974

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1 SHEET______ OF______

SUBJECT MAIN STEAM PIPING

TABLE B-5

PIPE RUPTURE RESTRAINT LOAD SUMMARY

Piping Run: Main Steam Lead 3

Break Location & Direction: Circumferential Break at 3295- Turbine Side

Pipe Properties: Upper Bound

Node	Restraint Type	Load Direction	Maximum Load
3050	Anchor	$\pm X$ Direction	19.9 kips
3050	Anchor	$\pm Y$ Direction	32.2 kips
3050	Anchor	$\pm Z$ Direction	34.3 kips
3050	Anchor	$\pm \theta x$ Direction	3501 in-kips
3050	Anchor	$\pm \theta y$ Direction	5284 in-kips
3050	Anchor	$\pm \theta z$ Direction	3573 in-kips
3245	Frame	+Y Direction	229.4 kips
3245	Frame	-Z Direction	616.5 kips
3250	Frame	+Y Direction	176.3 kips
3275	U-Bolt	+Z Direction	784.3 kips
3285	Frame	+X Direction	362.0 kips
3285	Frame	-Y Direction	81.1 kips
3292	Containment Penetration	+X Direction	1772 kips

MAY 10 1974

Nuclear Services Corporation

PROJECT DIABLO CANYON UNIT 1

SUBJECT MAIN STEAM PIPING

SHEET________ OF________

TABLE B-6

PIPE RUPTURE PIPE LOAD SUMMARY

Piping Run: Main Steam Lead 3

Break Location & Direction: Circumferential Break at 3295- Turbine Side

Pipe Properties: Upper Bound

PIPE: Elbow 3275 - 3280	
Maximum moment (kips in)	26619
Percent of moment carrying capacity at yield (M_y = 25.546 x 10^3 kips in)	104.2%
Percent of moment carrying capacity at ε_u (M_{ε_u} = 60.087 x 10^3 kip in)	44.3%
Maximum strain level (in/in)	.0083
Percent of limit strain ($\varepsilon\mu$ = 0.16 in/in)	5.19%
Maximum deflection (in) X deflection at node 3140	-2.312
Maximum deflection (in) Y deflection at node 3295	-2.998
Maximum deflection (in) Z deflection at node 3295	4.885

MAY 10 1974

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

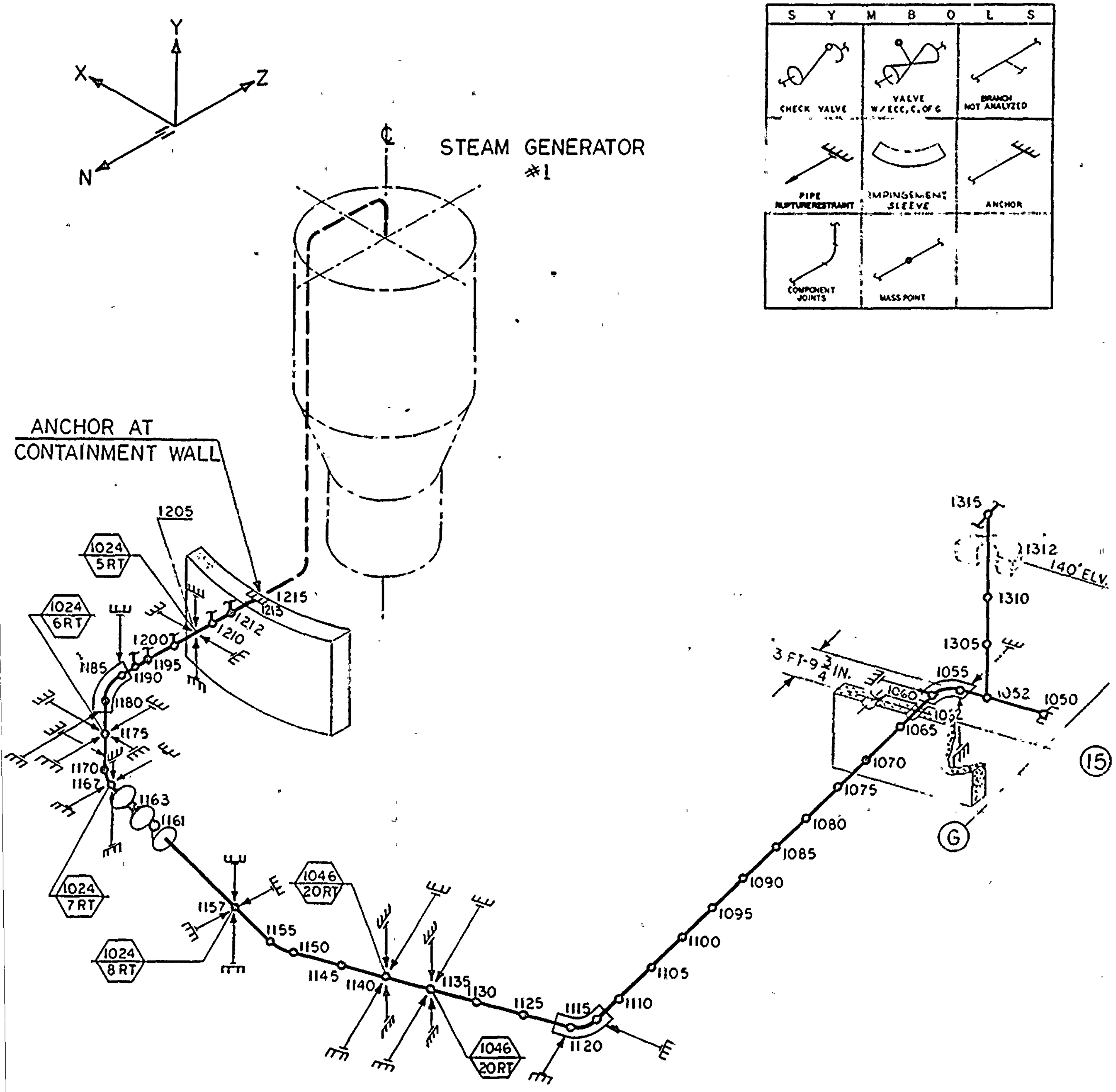


FIGURE B-1 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
MAIN STEAM SYSTEM LINE #1
CONTAINMENT TO COLUMN LINE "G"

MAY 10 1974

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

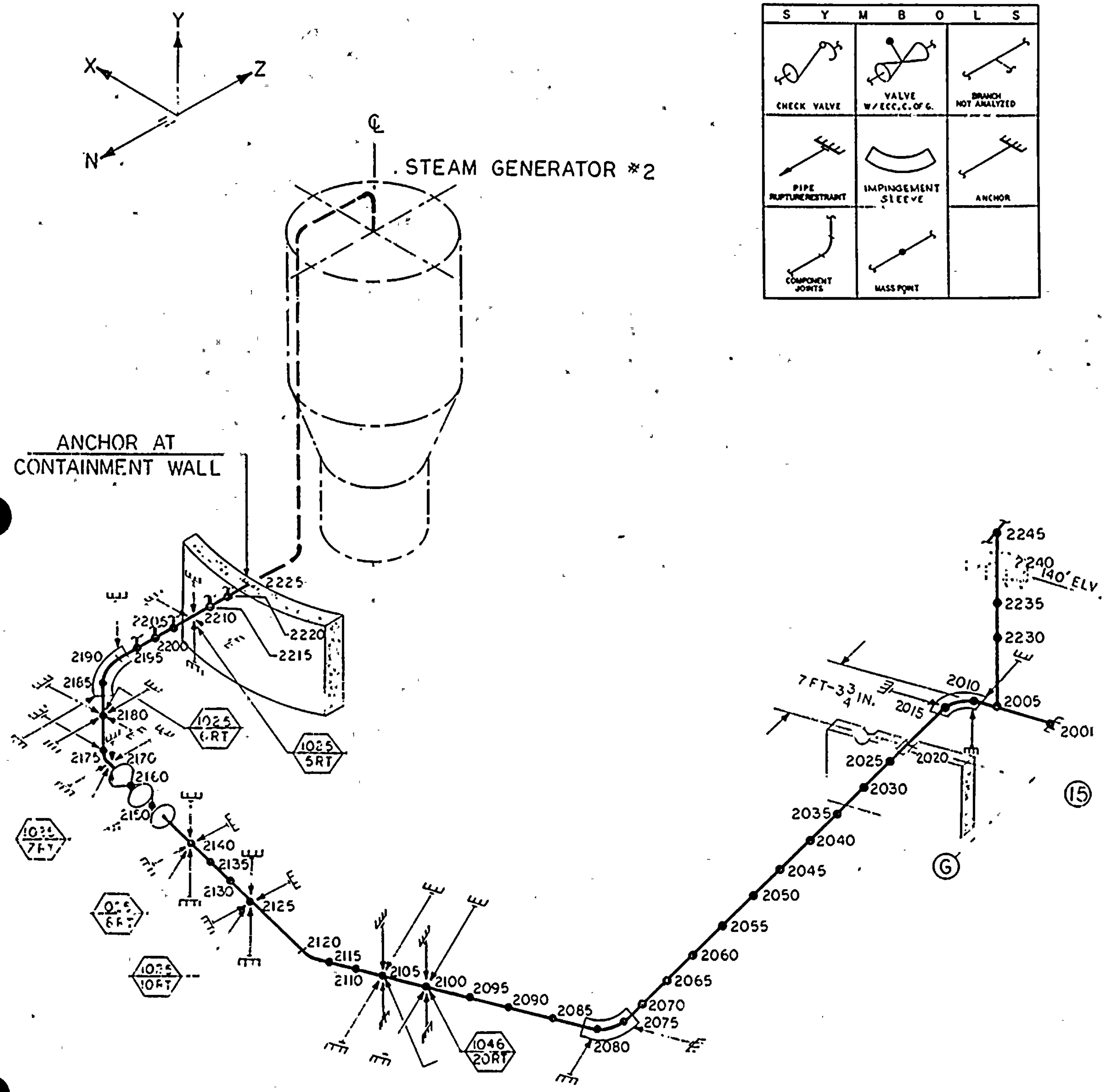


FIGURE B-2 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
MAIN STEAM SYSTEM LINE *2
CONTAINMENT TO COLUM LINE "G"

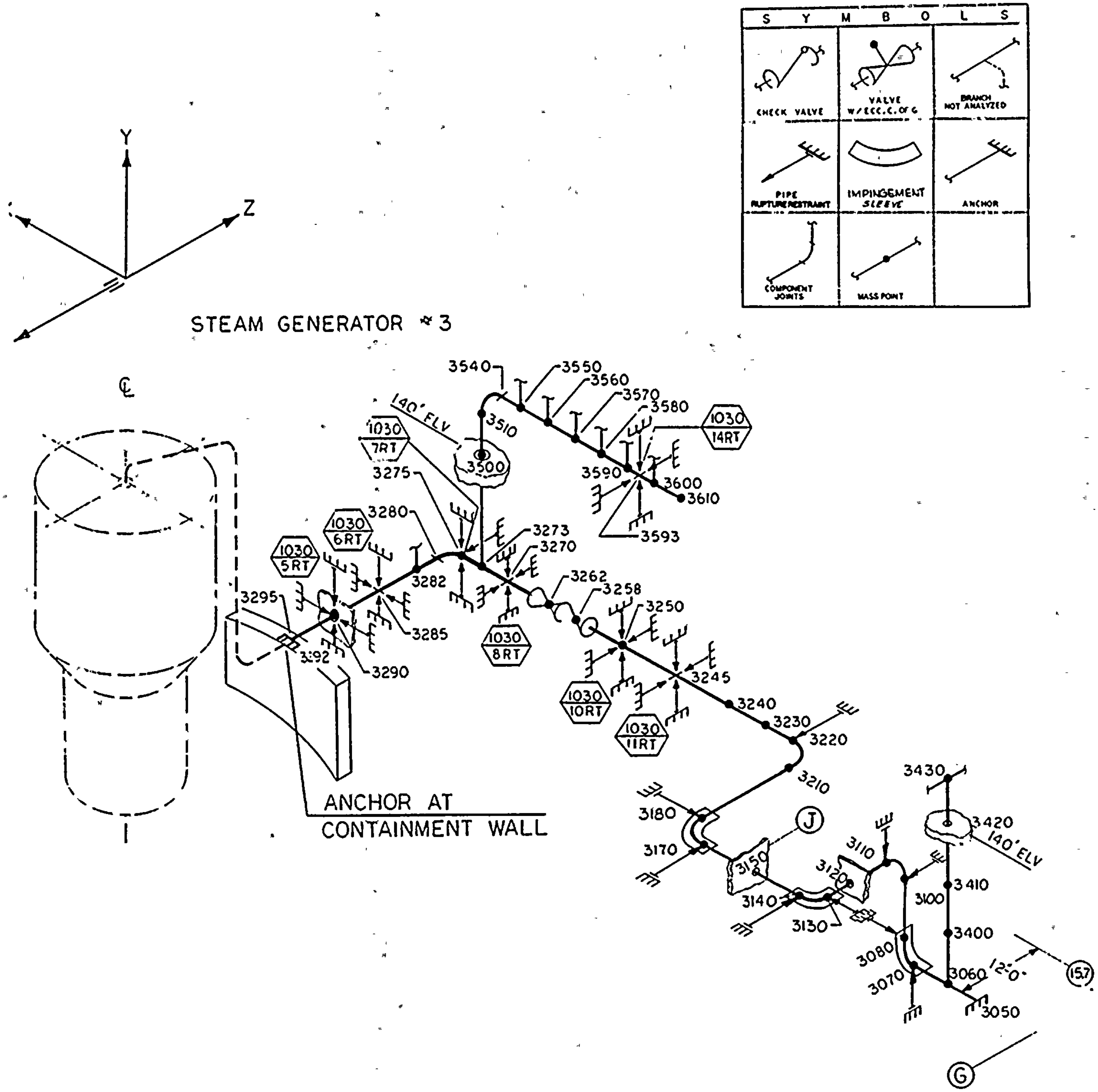


FIGURE B-3 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
MAIN STEAM SYSTEM LINE #3
CONTAINMENT TO COLUMN LINE "G"

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

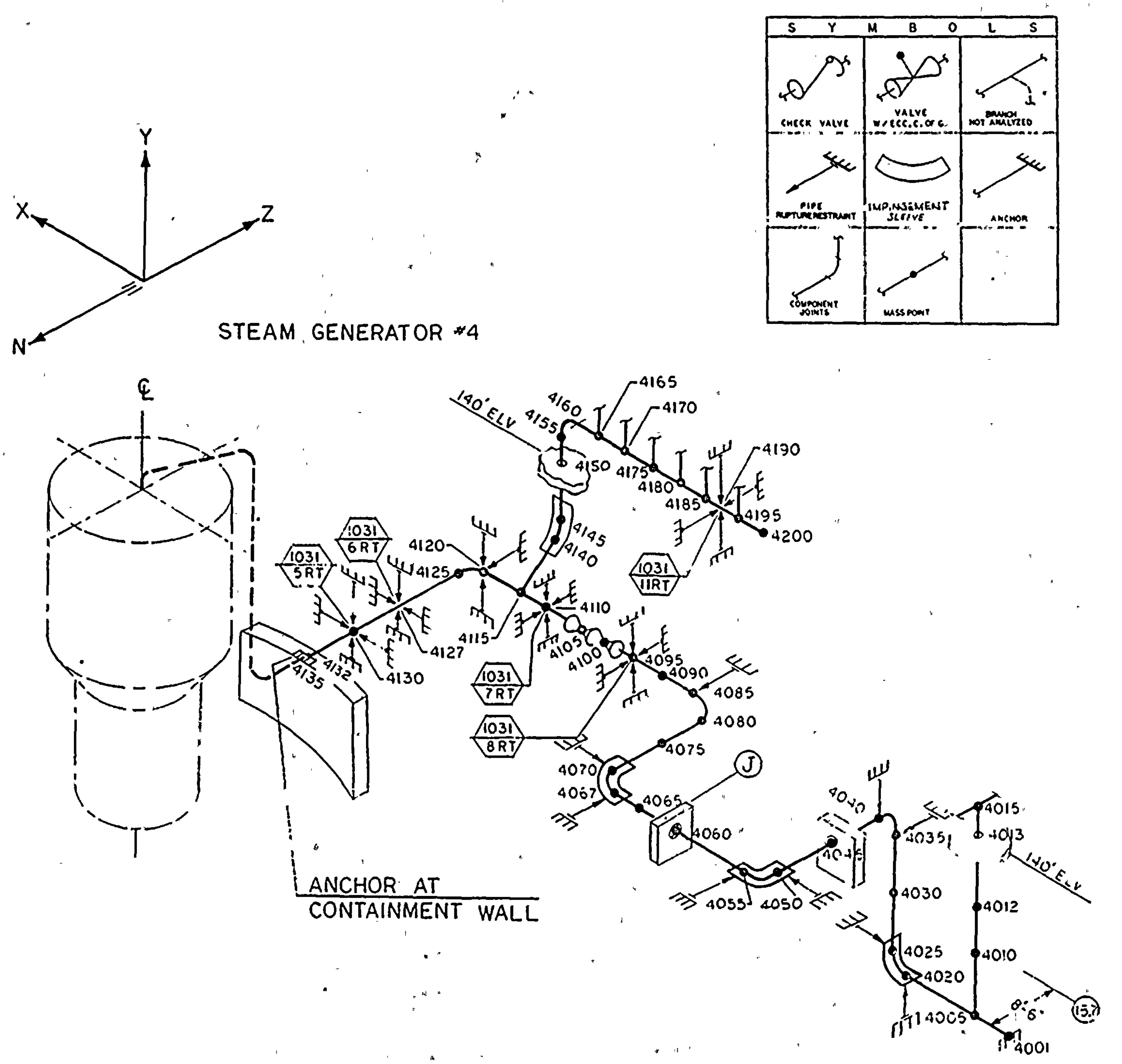


FIGURE B-4 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
MAIN STEAM SYSTEM LINE #4
CONTAINMENT TO COLUMN LINE 'G'

MAY 10 1974

MAY 10 1974

3.6A-176

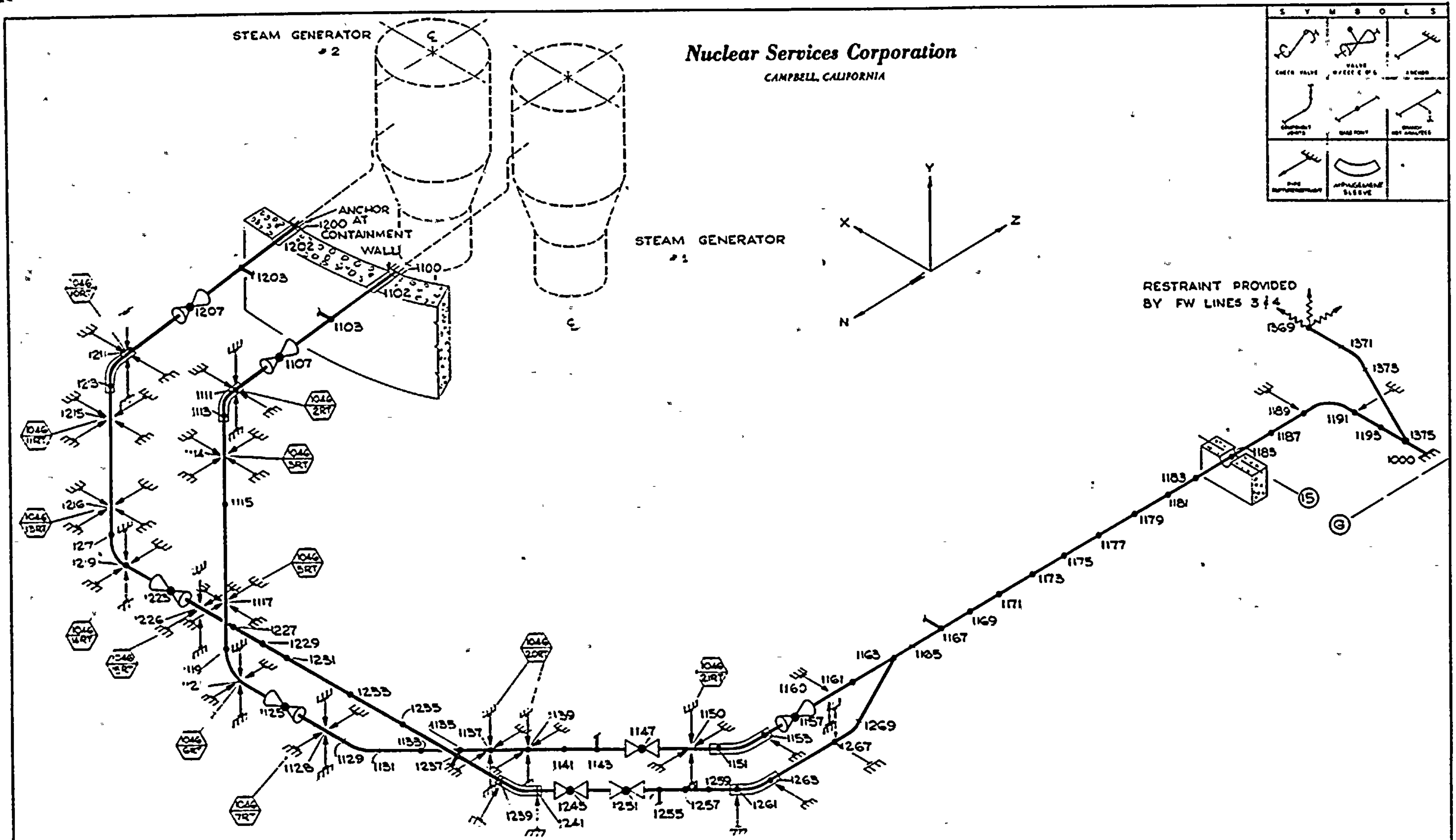


FIGURE B-5 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
FEEDWATER PIPING-LINES 1 AND 2
CONTAINMENT TO COLUMN LINE "G"

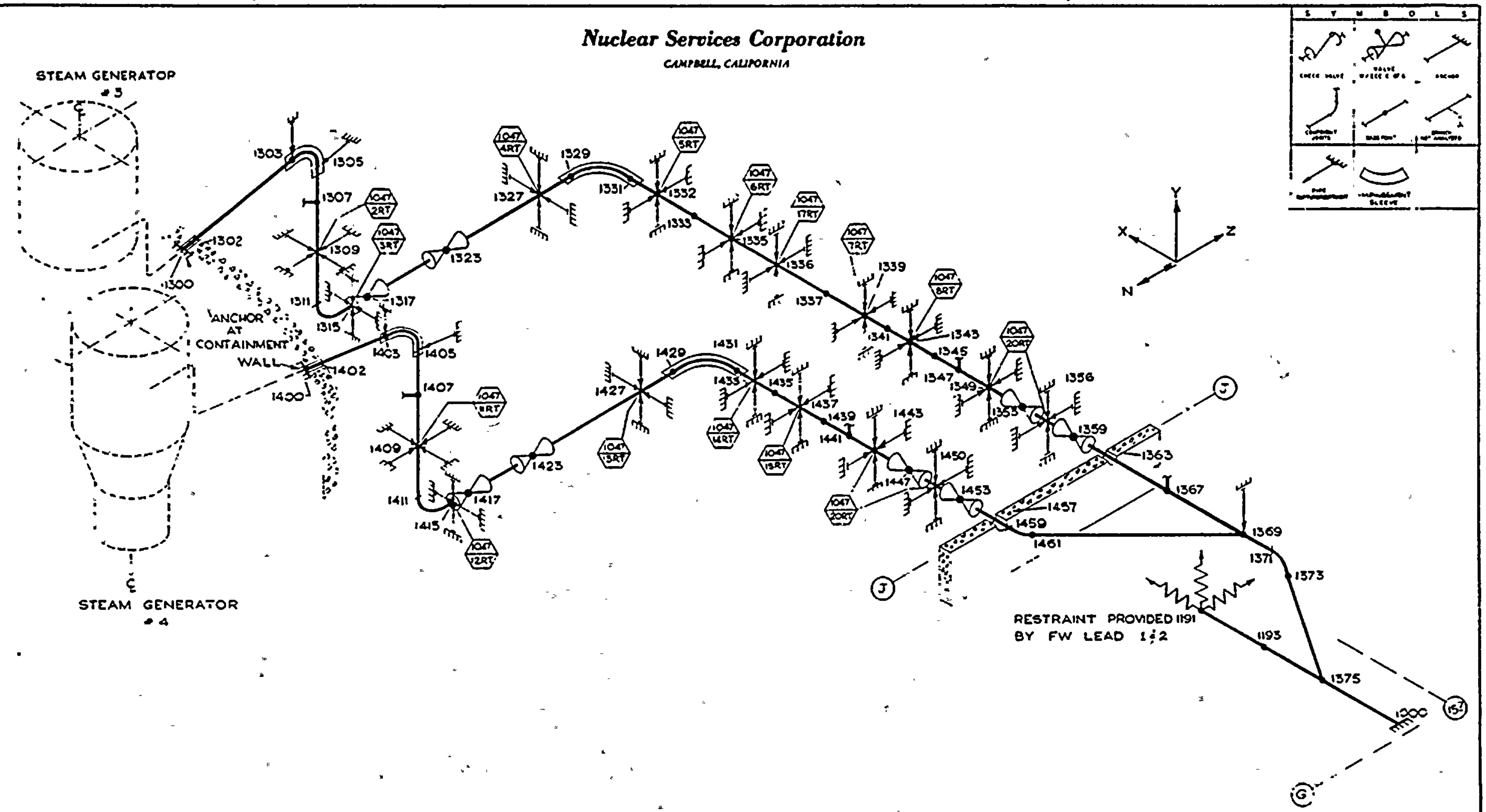


FIGURE B-6 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
FEEDWATER PIPING-LINES 3 & 4
CONTAINMENT TO COLUMN LINE 'G'

3.6A-177

MAY 10 1974

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

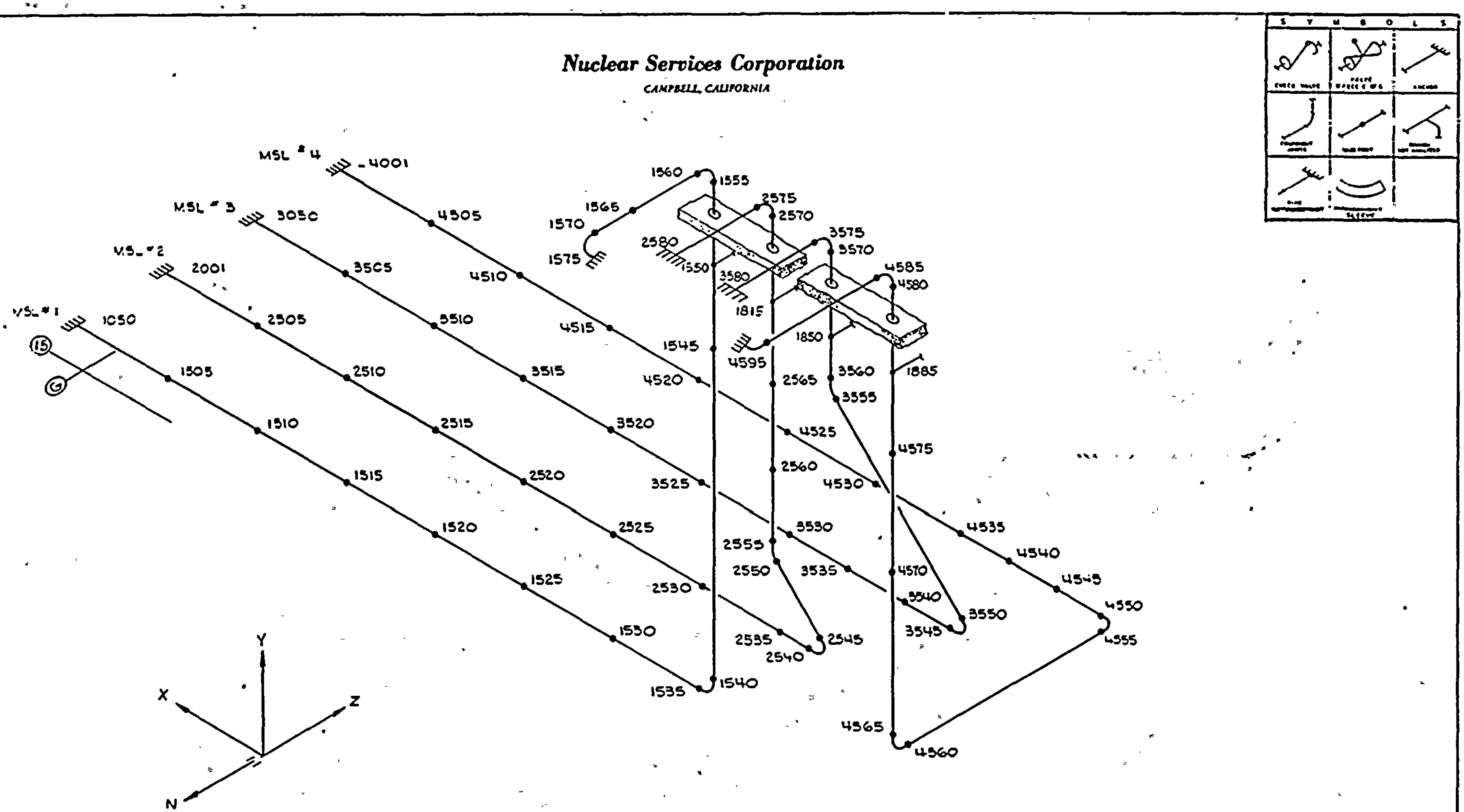


FIGURE B-7 DIABLO CANYON UNIT I
PIPE WHIP MATHEMATICAL MODEL
MAIN STEAM LINES — TURBINE BUILDING

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

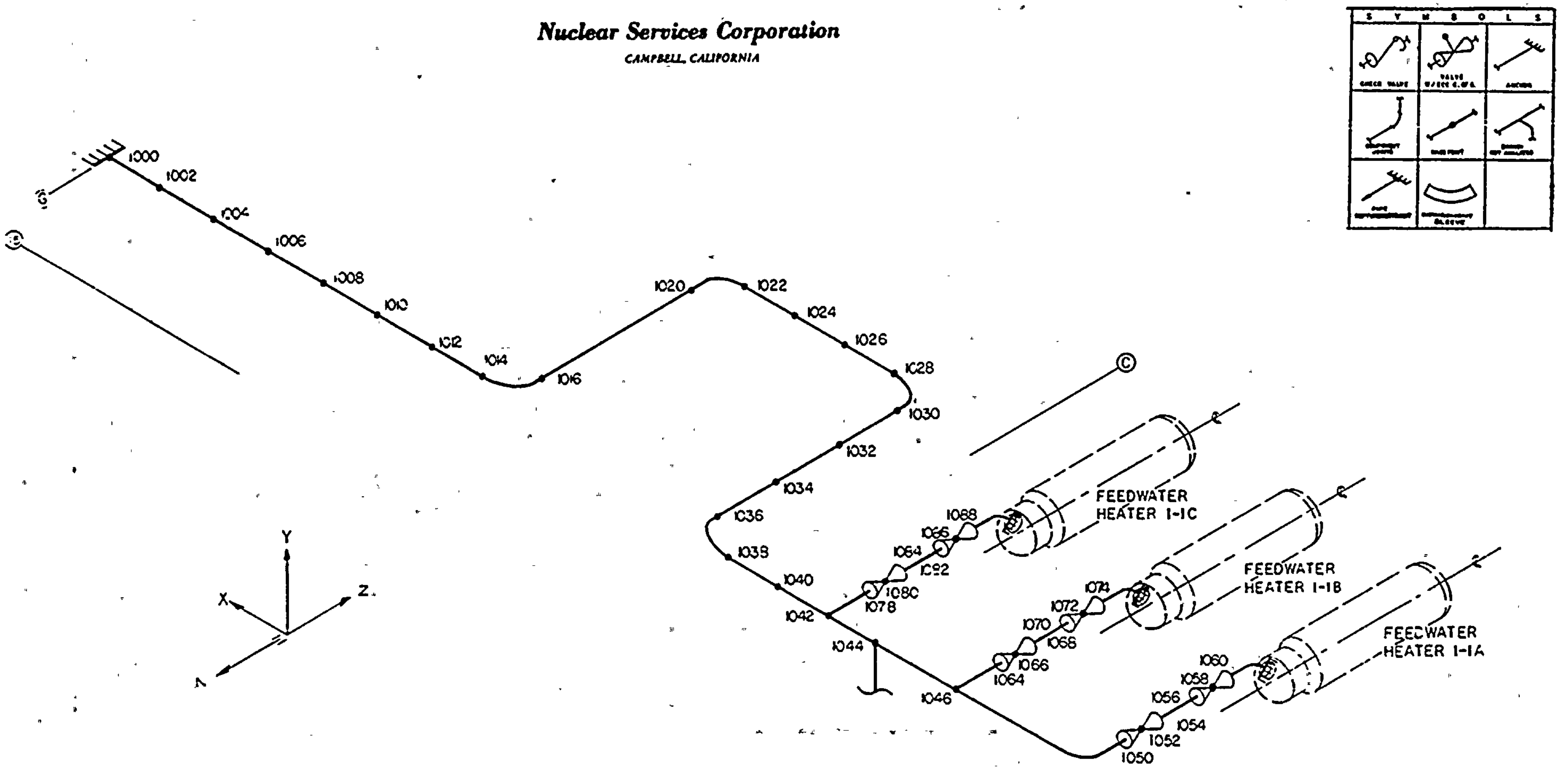


FIGURE B-8 DIABLO CANYON UNIT 1
PIPE WHIP MATHEMATICAL MODEL
FEEDWATER PIPING-COLUMN "G" TO FEEDWATER HEATERS
TURBINE BUILDING

3.6A-179

MAY 10 1974

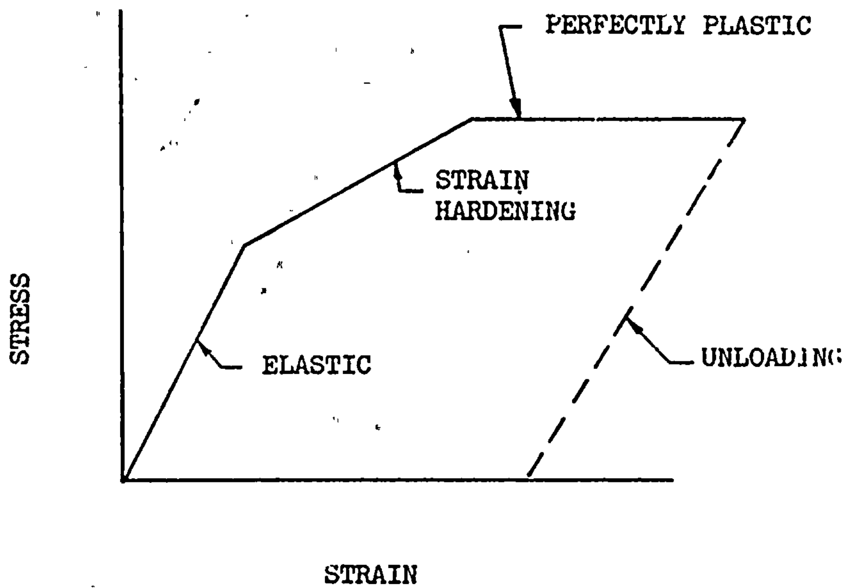
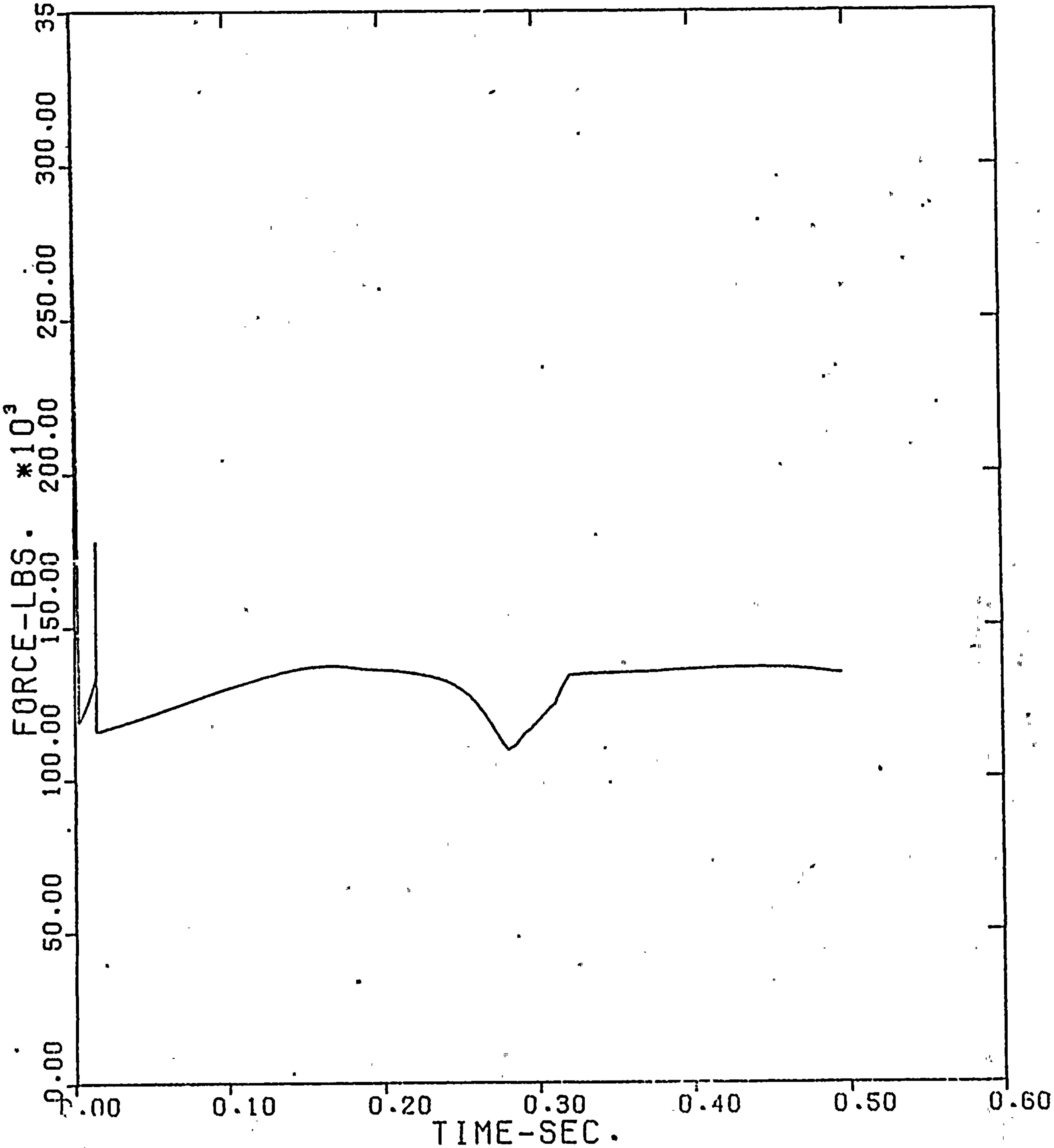


FIGURE B-9

PIPING STRESS-STRAIN CHARACTERISTICS

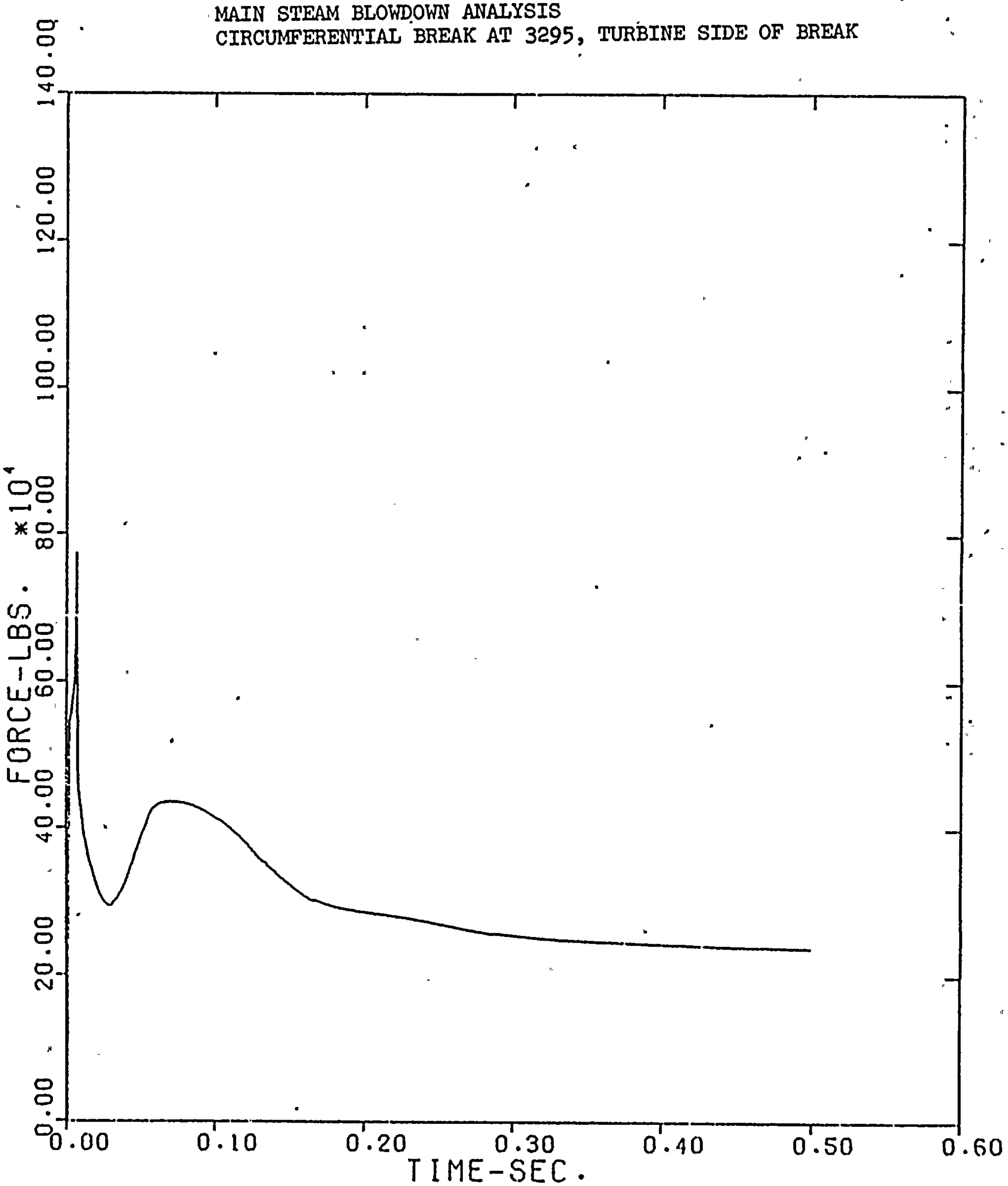
FIGURE B-10 DIABLO CANYON UNIT 1
FEEDWATER BLOWDOWN ANALYSIS
CIRCUMFERENTIAL BREAK AT 1375, STEAM GENERATOR SIDE OF BREAK



MAY 10 1974

Nuclear Services Corporation

FIGURE B-11 DIABLO CANYON UNIT 1
MAIN STEAM BLOWDOWN ANALYSIS
CIRCUMFERENTIAL BREAK AT 3295, TURBINE SIDE OF BREAK



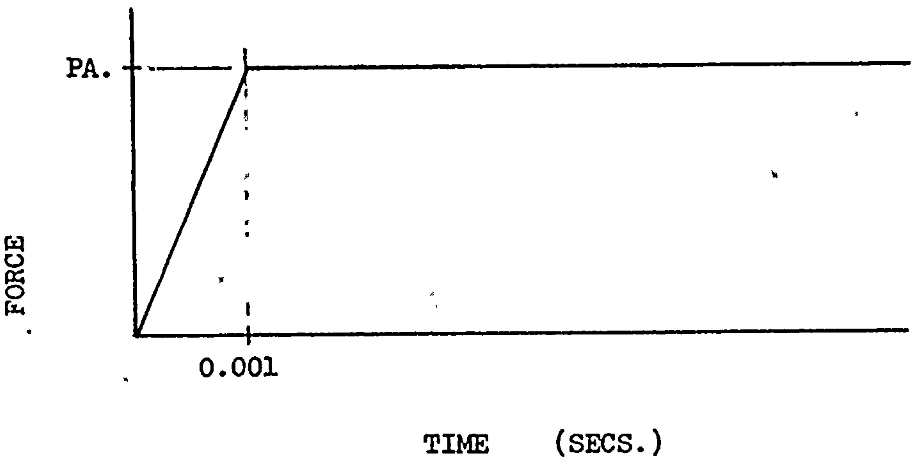
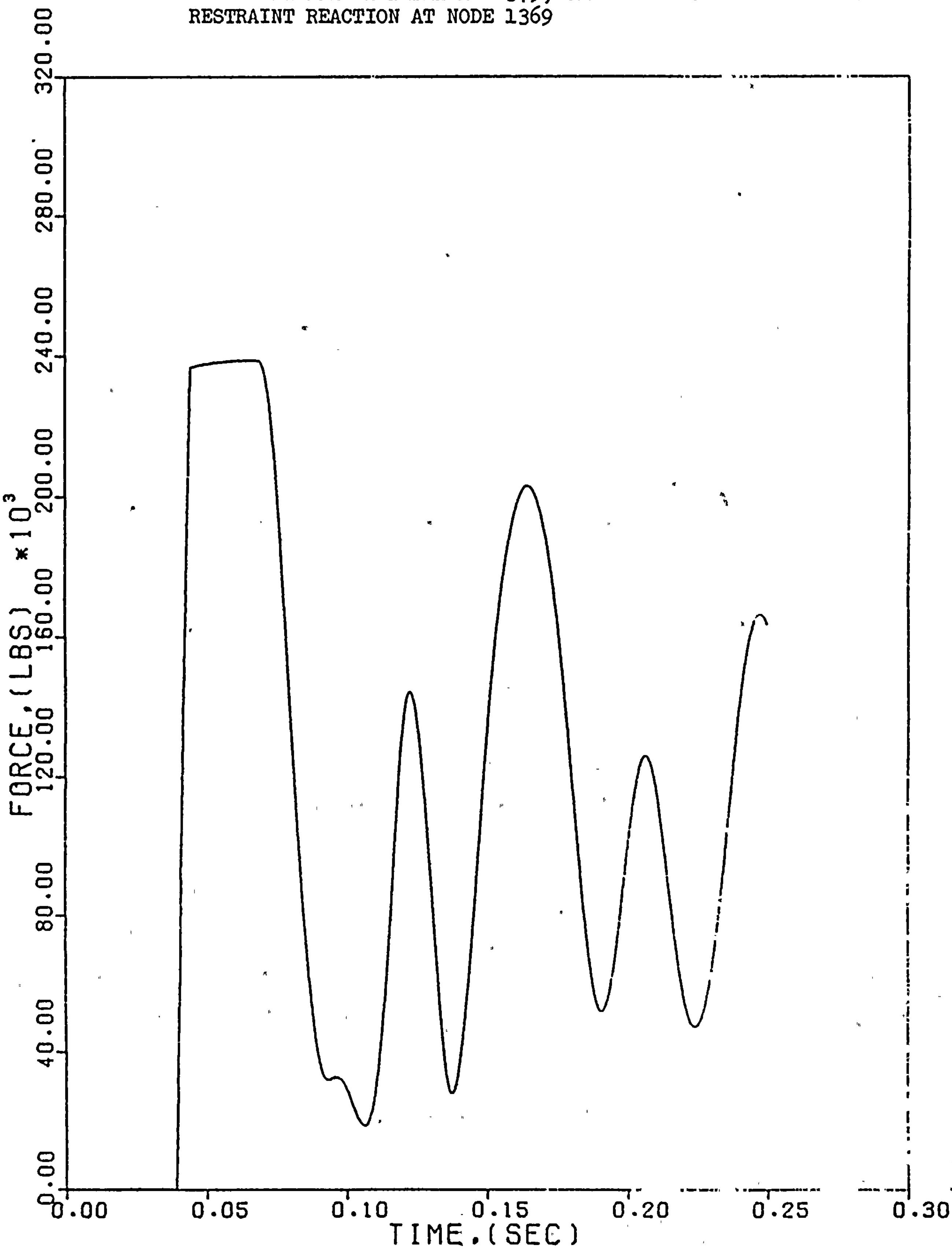


FIGURE B-12

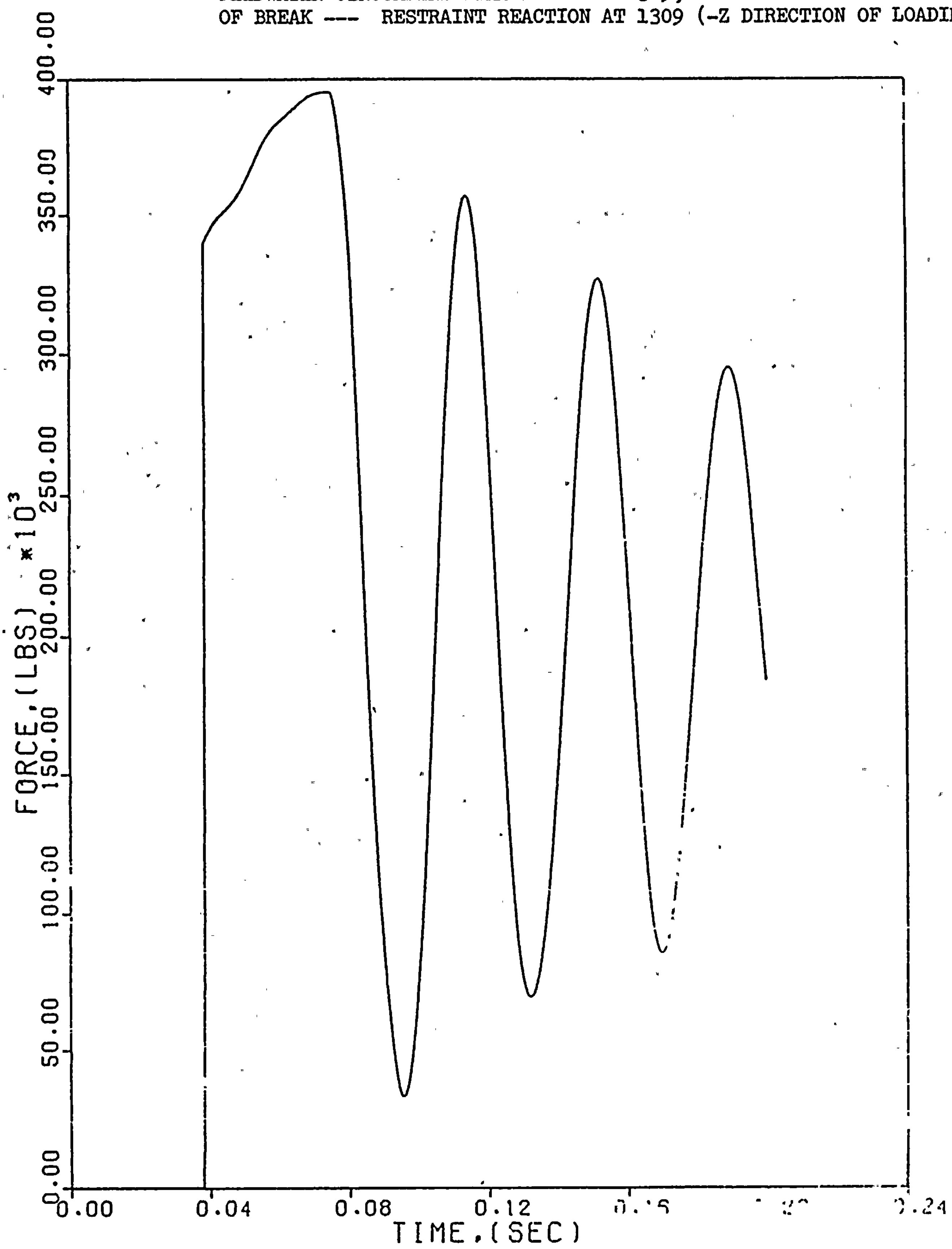
BLOWDOWN FORCE FOR SLEEVED CIRCUMFERENTIAL BREAK

FIGURE B-13 DIABLO CANYON UNIT 1
FEEDWATER BRANCH BREAK AT 1375, STEAM GENERATOR SIDE OF BREAK
RESTRAINT REACTION AT NODE 1369



MAY 10 1974

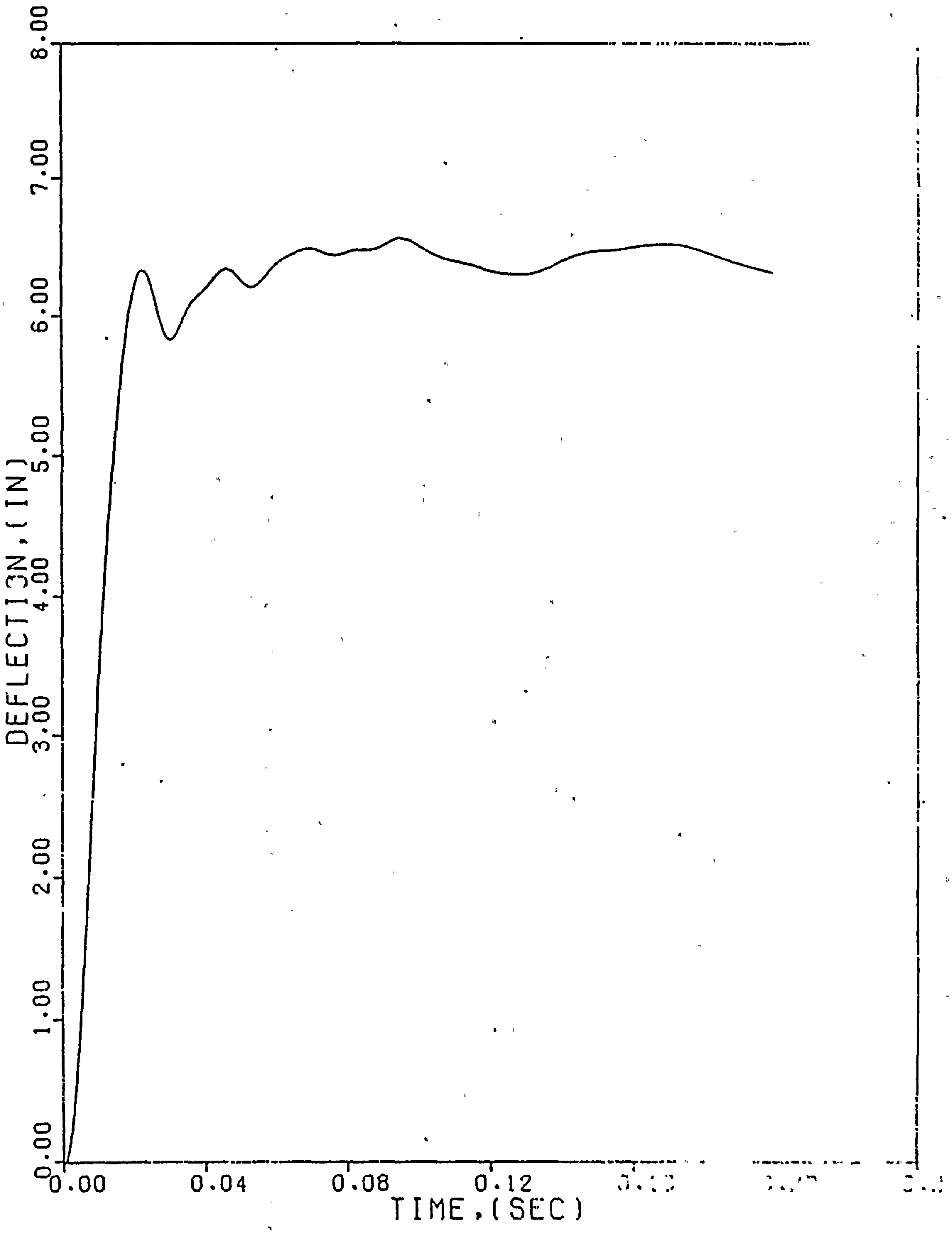
FIGURE B-14 DIABLO CANYON UNIT 1
FEEDWATER CIRCUMFERENTIAL BREAK AT 1329, STEAM GENERATOR SIDE OF BREAK --- RESTRAINT REACTION AT 1309 (-Z DIRECTION OF LOADING)



MAY 10 1974

Nuclear Services Corporation

FIGURE B-15 DIABLO CANYON UNIT 1
FEEDWATER CIRCUMFERENTIAL BREAK AT 1329, STEAM GENERATOR SIDE OF BREAK --- Z DIRECTION DEFLECTION AT 1329



MAY 10 1974

APPENDIX C

JET EFFECTS ANALYSIS

Nuclear Services Corporation

MAY 10 1974

TABLE OF CONTENTS

		Page
C.1	INTRODUCTION	3.6A-191
C.2	CRITERIA AND ASSUMPTIONS	3.6A-192
	C.2.1 Assumptions for Analysis of Design Basis Breaks	3.6A-192
	C.2.2 Assumptions for Analysis of Crack Breaks	3.6A-194
C.3	METHOD OF ANALYSIS	3.6A-195
	C.3.1 Jet Expansion	3.6A-195
	C.3.2 Jet Impingement Target Loading	3.6A-197
	C.3.3 Jet Stagnation Temperature	3.6A-199
	C.3.4 Environmental Effects due to Jets from Crack Breaks	3.6A-201
C.4	RESULTS OF ANALYSES	3.6A-202
	C.4.1 Jet Impingement Analysis	3.6A-202
	C.4.2 Crack Break Analysis	3.6A-203

C.1 INTRODUCTION

This appendix describes the criteria and analytical methods used to evaluate the impingement of a fluid jet issuing from a design basis pipe break. Also included are the criteria and methods used to determine enveloped targets, and environmental effects resulting from crack breaks. Presented are typical results for the analyses of breaks in the main steam piping from the containment to the turbine stop valves and the feedwater piping between the containment and the feedwater pumps.

C.2 CRITERIA AND ASSUMPTIONS

The type, size and shape of pipe breaks which should be considered are given in Reference 1. Cases (a) and (b), below, are postulated to occur at design basis break points and case (c) at most adverse locations along the pipe.

(a) Longitudinal design basis breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The length of the break is equal to twice the pipe diameter with the break area equal to the effective cross-sectional flow area upstream of the break location.

(b) Circumferential design basis breaks are perpendicular to the pipe axis with the break area equal to the internal cross-sectional area of the ruptured pipe.

(c) Crack breaks are postulated at the most adverse locations with regard to essential structures and equipment. The length of the crack is half the pipe internal diameter and the width is half the wall thickness.

C.2.1 Assumptions for Analysis of Design Basis Breaks

A. Fluid discharge is perpendicular to the plane of the break area A. Unobstructed discharge from the break area (or from both ends of the circumferential break) is assumed unless it can be demonstrated that pipe ends of a circumferential break stay together, or when sleeves are installed over break locations and whip restraints prevent movement out of sleeve.

B. Jet flow and impingement force are equivalent to steady break flow and thrust respectively as determined from the PRTHRUST computer program results.

C. The jet force vector for design basis pipe breaks is assumed to be in the direction of flow discharge for an unrestrained pipe as well as for a restrained pipe.

D. A discrete jet boundary defined by a jet divergence half-angle of 10° is assumed.

E. Jet impingement area is identical in shape to the break area (i.e. circular or rectangular).

F. Jet impingement pressure is assumed uniform over the expanded jet cross-sectional area.

G. Target geometry with respect to jet direction is considered in determining projected target area and resulting target impingement loads.

H. The forward jet momentum is assumed to be cancelled during fluid impingement.

C.2.2 <u>Assumption for Analysis of Crack Breaks</u>

A. Fluid discharge is perpendicular to the plane of the break.

B. A discrete boundary is formed bounded by a divergence half-angle of 10°.

C. Jet envelopment area is identical in shape to the break area (i.e. rectangular).

C.3 METHODS OF ANALYSIS

As per Reference 13, the jet impingement force resulting from a design basis break is equivalent to the steady thrust load imposed upon the ruptured piping.

At design basis break locations, the jet impingement force is taken from the time history blowdown analysis performed using PRTHRUST. A typical response is shown in Figure C-1. Maximum steady thrust is taken when fluid discharge reaches the critical flow rate predicted by Moody (Reference 18). During this period, the acceleration thrust force is small and total force consists primarily of the break exit pressure and momentum force terms. Figure C-1 illustrates the selection of maximum steady thrust from the PRTHRUST time history response.

At sleeved design basis break locations, the thrust force can be expressed as (Reference 18):

$$T = 1.26\ PA \text{ for steam, and}$$

$$T = 2.0\ PA \text{ for water}$$

where:

P = the maximum operating pressure inside the pipe before the break occurs

A = the annular area between the sleeve and the pipe.

C.3.1 Jet Expansion

The expanding jet diameter, height or width is assumed to increase linearly with distance with a 10° half angle, as shown in Figure C-2.

The jet dimensions and area at a distance L from the break can be expressed in terms of break area and distance from the jet source as follows:

For Design Longitudinal Breaks:

$$H_j = 0.4432A^{\frac{1}{2}} + 0.3526L \quad (1)$$

$$W_j = 2.2567A^{\frac{1}{2}} + 0.3526L \quad (2)$$

$$A_j = A + 0.9520A^{\frac{1}{2}}L + 0.1243L^2 \quad (3)$$

For Design Circumferential Breaks:

$$D_j = 1.1284A^{\frac{1}{2}} + 0.3526L \quad (4)$$

$$A_j = A + 0.625A^{\frac{1}{2}}L + 0.0976L^2 \quad (5)$$

For Longitudinal Crack Breaks:

$$H_j = \frac{t}{24} + 0.3526L \quad (6)$$

$$W_j = \frac{D}{24} + 0.3526L \quad (7)$$

$$A_j = \frac{Dt}{576} + 0.01469L(D+t) + 0.1243L^2 \quad (8)$$

For Circumferential Crack Breaks:

$$H_j = \frac{D}{24} + 0.3526L \quad (9)$$

$$W_j = \frac{t}{24} + 0.3526L \quad (10)$$

$$A_j = \frac{Dt}{576} + 0.01469L(D+t) + 0.1243L^2 \quad (11)$$

where:

D_j = Diameter of the jet at a distance L from the source (ft)

A_j = Jet cross-sectional area perpendicular to the jet axis at a distance L from the jet source (ft^2)

H_j = Effective jet height perpendicular to the pipe axis (ft)

W_j = Effective jet width parallel to the pipe axis (ft)

A = Break area (ft^2)

L = Distance from the jet source (ft)

D = Piping internal diameter (in.)

t = Piping wall thickness (in.)

The foregoing jet formulas may be used for both steam and subcooled water (feedwater) systems which contain fluid above 240°F and 275°psig. For non-flashing subcooled water systems below 240°F jet area should be calculated using:

$$A_j = A \tag{12}$$

C.3.2 Jet Impingement Target Loading

The normal equivalent static load applied to a target by the jet emanating from a postulated design basis break may be expressed as:

$$\begin{aligned} F &= P_j A_i S_F D_{LF} \\ &= T \frac{A_i}{A_j} S_F D_{LF} \end{aligned} \tag{13}$$

where

T = Total thrust of jet (lbf)

P_j = Jet pressure at the target (psf)

A_i = Cross-sectional area of jet intercepted by target structure (ft^2)

A_j = Total cross-sectional area of jet at the target structure (ft^2)

S_F = Shape factor

D_{LF} = Dynamic load factor

The total thrust, T, has been defined previously. The ratio A_i/A_j represents the proportion of the total mass flow interrupted by the target structure. The dynamic load factor, D_{LF}, accounts for the rapid application of the load. A dynamic load factor of two is recommended unless it can be shown that the rise time of the applied force is significantly longer than the lowest vibration period of the target structure (Reference 19).

The shape factor, S_F, depends on the projected section and orientation of the target struck by the jet, and is a measure of the target's potential for changing the momentum of the jet. The shape factor may be understood by first noting that a plane of area "A" oriented with its normal at an angle ϕ with respect to an impinging jet flow will deflect the flow to a direction parallel to the plane (Figure C-3a). The shape factor for a plane whose normal is oriented at an angle ϕ to the axis of the jet is equal to $\cos \phi$. Shape factors for curved surfaces may be obtained by integration over the area impinged upon by the jet. In this fashion, the following shape factors have been derived.

MAY 10 1974

Jet impinging on a slab at an angle ϕ from perpendicular (Figure C-3a)	$S_F = \cos \phi$
Rectangular jet impinging on a pipe larger than jet (Figure C-3b)	$S_F = 1 - \frac{h}{2D_o}$
Rectangular jet impinging on pipe with h greater than D_o (Figure C-3b)	$S_F = 1/2$
Circular jet impinging on pipe with jet diameter ($D_j = 2r_j$) less than pipe diameter (Figure C-3c)	$S_F = 1 - \frac{D_j}{D_o}(.288)$
Circular jet impinging on pipe with jet diameter greater than pipe diameter (Figure C-3d)	$S_F = .712$

The force 'F' determined using the above formulae and factors represents the integral of a uniform pressure applied normal to the target impinged upon by a jet.

C.3.3 Jet Stagnation Temperature.

Temperature of the jet at enveloped targets is a function of target distance from the break location and decreases with distance due to entrainment and mixing action with the ambient environment surrounding the jet. Conservative assumptions have been made to estimate the stagnation temperature at the target. These assumptions are:

A. An isentropic free expansion of the jet to a maximum plume diameter with a normal shock wave occurring at the maximum diameter. From the maximum diameter, the down stream flow velocity is constant.

B. Conservation of energy through the shock wave.

MAY 10 1974

C. The jet is composed of a single phase fluid with an initial temperature of 520°F assumed for the main steam system. The ambient environment surrounding the jet is also single phase with an initial temperature of 200°F. Fluid properties are assumed constant.

D. A temperature entrainment angle of 6° starting at the normal shock wave.

E. A linear temperature distribution from the jet boundary to the jet centerline.

Based on the entrainment angle and energy balance between the jet and entrainment fluid, the centerline total (stagnation) temperature of the jet may be determined as a function of distance from the break as follows:

Energy at exit plane = energy at any other plane

$$U_o = \dot{W}C_pT_o = A_bV\rho C_pT_o = \int_A \rho VC_p \, TdA \qquad (14)$$

$$= \rho VC_p \int_A TdA$$

but $\int_A TdA = 1/3 \, T_{max}A$ for a linear temperature distribution (15)

thus: $A_bV\rho C_pT_o = 1/3 \, AV\rho C_pT_{max}$

or: $$T_{max} = 3T_o\left(\frac{A_b}{A}\right) \qquad (16)$$

MAY 10 1974

where:

U_o = Internal energy

$\dot{W}$ = Mass flow rate

C_p = Specific heat of fluid

T_o = Stagnation temperature at the break

A_b = Break area

T = Stagnation temperature at distance X from the break

A = Jet enveloped area at distance X from the break

T_{max} = Stagnation temperature at the centerline of the jet at a distance X from the break

ρ = Fluid density

V = Jet velocity

Experimental studies have found the range of temperature entrainment angles to vary between 6.5° (Reference 42) and 25° (Reference 43). The resulting jet stagnation temperatures as a function of distance from the break are shown in Figure C-4 for entrainment angles of 6° and 12°. The more conservative angle of 6° has been used for evaluation of safety related equipment.

C.3.4 <u>Environmental Effects due to Jets from Crack Breaks</u>

As defined in Section 7.1.1.2 crack breaks were postulated to occur in the most adverse locations with regard to safety related equipment. The jet expansion formulas given by equations (6) to (11) were then used to determine the vital target areas enveloped by fluid issuing from the crack. A similar method to that presented in Section C.3.3 is used to determine the resulting fluid temperature at the target.

C.4 RESULTS OF ANALYSES

C.4.1 Jet Impingement Analysis

Typical results of jet impingement analyses of main steam and feedwater pipes outside the containment are presented in Table C-1. Locations of essential equipment and conduits were taken from Reference 20. Whenever the effects of the jet are adverse and unacceptable, it is proposed to mitigate the effects by adopting one or more of the following methods:

A. Redirection of the jet, and also reducing its net force, by installation of sleeves at selected design basis break points. This not only redirects the jet in a zone where there is no vital equipment but also reduces the effective size of the break, thereby reducing the jet force.

B. Installation of jet impingement barriers at appropriate locations so as to avoid impingement on essential equipment. This is proposed for impingements where the jet force or fluid temperature is found to be higher than what the essential equipment can withstand.

C. Relocation of the affected equipment or conduit to avoid the jet.

D. Install backflow dampers to prevent steam from flowing through a damaged ventilation duct into areas containing vital equipment which could not withstand the resulting environment.

C.4.2 <u>Crack Break Analysis</u>

Typical results of enveloped targets due to crack breaks of the main steam and feedwater piping are presented in Table C-2. For that affected vital equipment which cannot withstand the resulting environment, it is proposed to mitigate the effects by adopting one or more of the following methods:

A. Relocation of the equipment or conduit to avoid jet envelopment.

B. Installation of suitable shields to protect the affected equipment.

C. Replacement of equipment which cannot be demonstrated to perform its required function in the post break environment with equipment that is designed for the environment.

MAY 10 1974

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR JET IMPINGEMENT EVALUATION

PROJECT: P. G. & E. - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEADS 1 & 2 AREA F

PIPING RUN: LINE #583 & 584

TABLE C-1

Break Location	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Thrust (lbf)	Jet Area A_j (ft^2)	Jet Pressure T/A_j(psf)(c)	Target Area A_t(ft^2)	Force F(lb)(d)
1120	DC	E	K 5846 (F.W.Isol. Valve Control)	40	57,800	189	282	1.38	540
2080	DC	E	K 5836 (M.S.Isol. Valve Control)	20	57,800	56.3	948	1.18	1378

NOTES: (a) DC - design circumferential
DL - design longitudinal

(b) N - north
S - south
E - east
W - west
U - up
D - down

(d) $F = A_t \times \frac{T}{A_j} \times \varepsilon_F \times D_{LF}$

3.6A-204

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR JET IMPINGEMENT EVALUATION

PROJECT: P. G. & E - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEAD 3

PIPING RUN: LINE #226

TABLE C-1 (con't.)

Break Location	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Thrust (lbf)	Jet Area A_j (ft^2)	Jet Pressure T/A_j(psf)(c)	Target Area A_t(ft^2)	Force F(lb)(d)
3273	DC	U	K 6478 (F.W.Isol. Valve Control Ckt.)	7	460,000	14.5	31,700	1.5	67,700

NOTES: (a) DC - design circumferential
DL - design longitudinal

(b) N - north
S - south
E - east
W - west
U - up
D - down

(d) $F = A_t \times \frac{T}{A_j} \times \varepsilon_F \times D_{LF}$

3.6A-205

MAY 10 1974

MAY 10 1974

3.6A-206

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR JET IMPINGEMENT EVALUATION

PROJECT: P. G. & E - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEAD 4

PIPING RUN: LINE #225 & 585

TABLE C-1 (con't.)

Break Location	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Thrust (lbf)	Jet Area A_j (ft^2)	Jet Pressure T/A_j(psf)(c)	Target Area A_t(ft^2)	Force F(lb)(d)
4115	DC	U	K 6556 (Vital Control Ckt.)		460,000	17	27,000	0.5	19,200
4050	DL	S	K 4319 (Vital Control Ckt.)	5	58,680	13.72	4,234	0.35	2,100

NOTES: (a) DC - design circumferential
DL - design longitudinal

(b) N - north
S - south
E - east
W - west
U - up
D - down

(d) $F = A_t \times \frac{T}{A_j} \times S_F \times D_{LF}$

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR JET IMPINGEMENT EVALUATION

PROJECT: P. G. & E. - DIABLO CANYON #1

PIPING SYSTEM: FEEDWATER LEADS 1 & 2

PIPING RUN: LINE #554 & 555, AREA FW

TABLE C-1 (con't.)

Break Location	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Thrust (lbf)	Jet Area A_j (ft^2)	Jet Pressure T/A_j(psf)(c)	Target Area A_t(ft^2)	Force F(lb)(d)
1113	DC	D	K 5824 (Vital Control Ckt.)	20	53,600	56.3	948	1.77	2385
1113	DC	D	K 5844 (Vital Control Ckt.)	20	53,600	56.3	948	1.06	1424
1239	DC	E	LCV-106	55	53,600	340	157	1	314
			LCV-107	55	53,600	340	157	1	314
			LCV-110	55	53,600	340	157	1	314
			LCV-111 (Above are aux. F.W. Control valves)	55	53,600	340	157	1	314

NOTES: (a) DC - design circumferential
DL - design longitudinal

(b) N - north
S - south
E - east
W - west
U - up
D - down

(d) $F = A_t \times \frac{T}{A_j} \times S_F \times D_{LF}$

3.6A-207

MAY 10 1974

TABLE OF TYPICAL RESULTS FOR JET IMPINGEMENT EVALUATION

PROJECT: P. G. & E - DIABLO CANYON #1

PIPING SYSTEM: FEEDWATER LEAD 3 & 4

PIPING RUN: LINE #556 & 557

TABLE C-1 (con't.)

Break Location	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Thrust (lbf)	Jet Area A_j (ft^2)	Jet Pressure T/A_j(psf)(c)	Target Area A_t(ft^2)	Force F(lb)(d)
1329	DL	N	K 6443 (F.W.Isol. Valve Control)	10	53,600	19	2782	0.51	2,020
1331	DL	N	FCV 440 (F.W. Isol. Valve)	8	53,600	14.1	3785	3	22,710
1429	DL	N	FCV 441 (F.W. Isol. Valve)	7	53,600	11.86	4503	3	27,018
1429	DL	N	K 6460 (F.W. Isol. Valve Pos. Indic.)	9	53,600	16.6	3226	.5	2,300
1441	DC	U	K 6592 (Vital Control Ckt.)	17	39,000	33	1180	.75	1,260

NOTES: (a) DC - design circumferential
DL - design longitudinal

(b) N - north
S - south
E - east
W - west
U - up
D - down

(d) $F = A_t \times \frac{T}{A_j} \times \varepsilon_F \times D_{LF}$

3.6A-208

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEAD 1

PIPING RUN: LINE #228 & 583

TABLE C-2

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
6.4-8 L-N (F)	LC	W	FCV-438 (F.W. Isol. valve)	4.5	3	Note (c)	Note (c)
6.4-8 L-N (F)	LC	W	K 5866 (Vital ckt.)	3.5	0.106	Note (c)	Note (c)
6.4-8 L-N (F)	LC	E	FCV-42 (M.S. Isol. valve)	8	3	Note (c)	Note (c)

NOTES: (a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

3.6A-209

MAY 10 1974

MAY 10 1974

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E. - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEAD 2

PIPING RUN: LINE # 227 & 584

3.6A-210

TABLE C-2 (con't.)

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
6.4-8 L-N (F)	LC	E	FCV-439 (F.W. Isol. valve)	4.5	3	Note (c)	Note (c)
6.4-8 L-N (F)	LC	E	LCV-111 (Aux. F.W. Control Valve)	16	1	Note (c)	Note (c)
6.4-8 L-N (F)	LC	E	K 5907 (Vital ckt.)	4	0.312	Note (c)	Note (c)

NOTES: (a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEAD 3

PIPING RUN: LINE # 226 & 586

TABLE C-2 (con't.)

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
15-15.7 N (GE)	LC	E	FCV 440 (F.W. Isol. valve)	8	3	Note (c)	Note (c)
15-15.7 L-M (GW)	LC	N	FCV 44 (M.S. Isol. valve)	3	3	Note (c)	Note (c)
15-15.7 J-K (GW)	LC	W	KT 6961 (Vital ckt.)	3	0.2	Note (c)	Note (c)

NOTES: (a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

3.6A-211

MAY 10 1974

MAY 10 1974

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E. - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM LEAD 4

PIPING RUN: LINE # 225 & 585

3.6A-212

TABLE C-2 (con't.)

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
15 & M (GW)	LC	W	FCV 441 (F.W. Isol. valve)	10	3	Note: (c)	Note: (c)
14-15 J-H (GW)	CC	U	K 4319 (Vital Circuit)	7	1.5	Note: (c)	Note: (c)
15-15.7 G-J (GW)	CC	U	K 6592 (Vital Circuit)	7	1.0	Note: (c)	Note: (c)

NOTES:
(a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E - DIABLO CANYON #1

PIPING SYSTEM: FEEDWATER LEAD 2

PIPING RUN: LINE #555

TABLE C-2 (con't.)

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
6.6-8 N-R (F)	LC	W	FCV-438 (F.W.Isol. valve)	27	3	Note: (c)	Note: (c)
6.6-8 N-R (F)	LC	D	K 5864 (Vital ckt.)	3	0.2	Note: (c)	Note: (c)
6.6-8 N-R (F)	LC	W	K 5864	11	0.24	Note (c)	Note: (c)

NOTES: (a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

3.6A-213

MAY 10 1974

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E. - DIABLO CANYON #1

PIPING SYSTEM: FEEDWATER LEAD 4

PIPING RUN: LINE #556

TABLE C-2 (con't.)

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
15.2-L (GW)	LC	U	Junction Box BJG 165	6	1	Note: (c)	Note: (c)
15 J-K (GW)	LC	N	K 6123 (Vital ckt.)	15	1	Note: (c)	Note: (c)
15-M (GW)	LC	U	T 648 (Vital ckt.)	6	0.25	Note: (c)	Note: (c)

NOTES: (a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

Nuclear Services Corporation

TABLE OF TYPICAL RESULTS FOR ENVIRONMENTAL EFFECTS OF CRACK BREAKS

PROJECT: P. G. & E. - DIABLO CANYON #1

PIPING SYSTEM: MAIN STEAM

PIPING RUN: LINE #583, 584, 585, 586 (IN TURBINE ROOM)

TABLE C-2 (con't.)

Break Location Col.line & area	Break Type(a)	Jet(b) Direction	Target	Distance L(ft)	Target Envelopment Area A_t (ft^2)	Temperature at Target (°F)	Humidity at Target(%)
15.2 D-E (140')	LC	E	K 5272 (Vital ckt.)	11	0.83	Note: (c)	Note: (c)
15.2 D-E (140')	LC	E	K 5273 (Vital ckt.)	14	1.0	Note: (c)	Note: (c)

NOTES: (a) LC - longitudinal crack
CC - circumferential crack

(b) N - north
S - south
E - east
W - west
U - up
D - down

(c) Environmental capabilities of the equipment and circuits subjected to the jet are being investigated.

3.6A-215

MAY 10 1974

FIGURE C-1

DIABLO UNIT 1 MS BLOWDOWN HOT STANDBY

CASE 5 TURBINE SIDE BR.NODE3-28

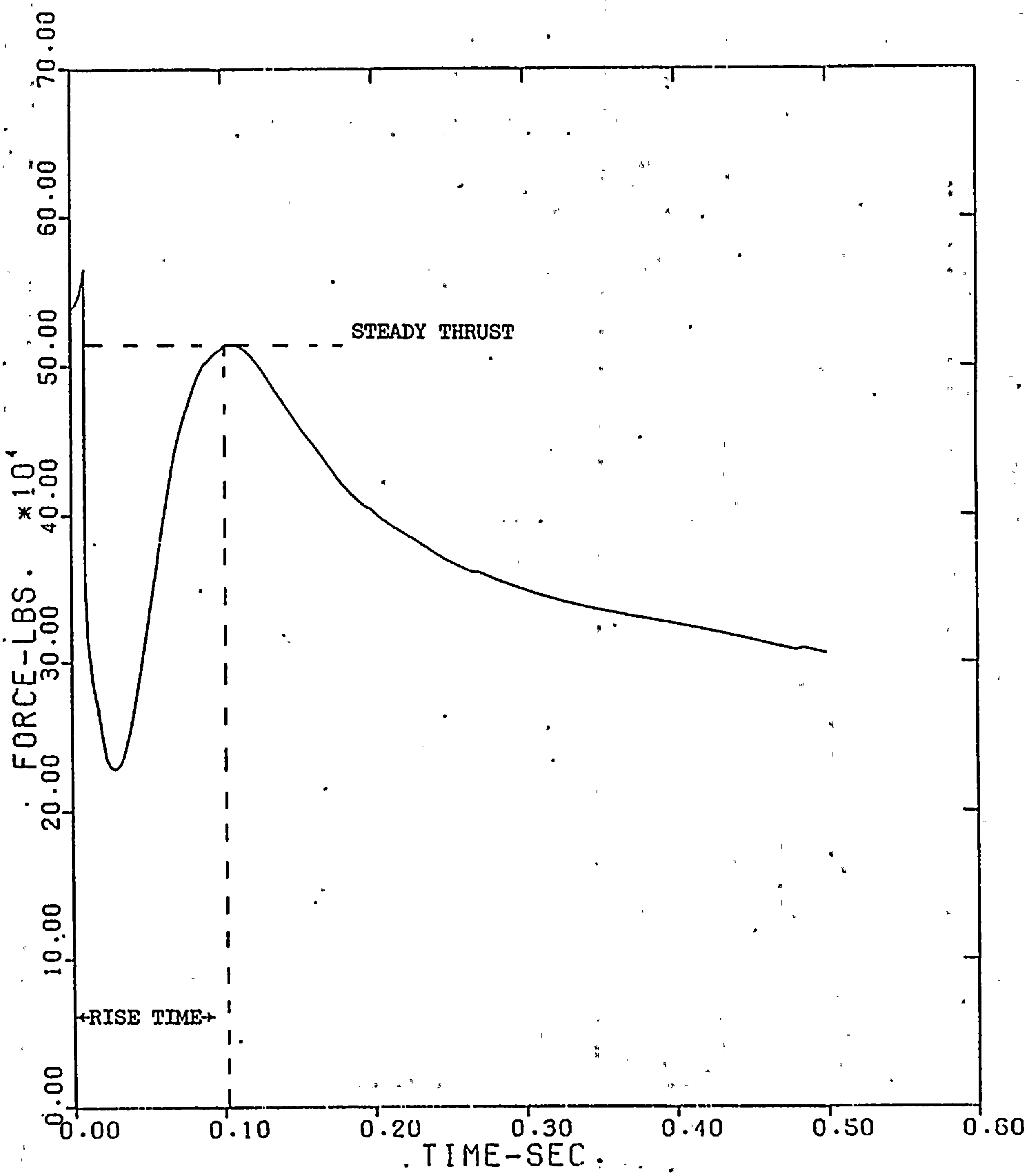
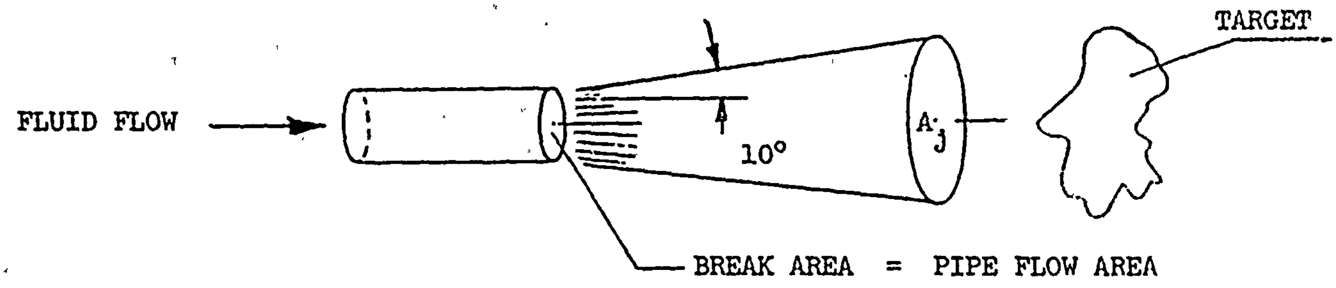


FIGURE C-2 JET IMPINGEMENT EVALUATION

DESIGN BASIS BREAK CONFIGURATIONS

A. CIRCUMFERENTIAL BREAKS

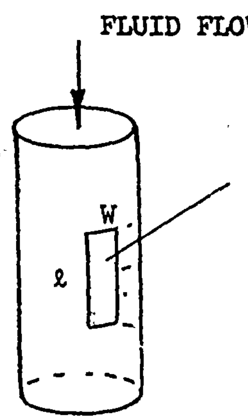


$$A_j = A_o + 0.625 A_o^{1/2} L + 0.09765 L^2$$

where A_o = PIPE FLOW AREA

L = DISTANCE FROM JET SOURCE

B. LONGITUDINAL BREAKS

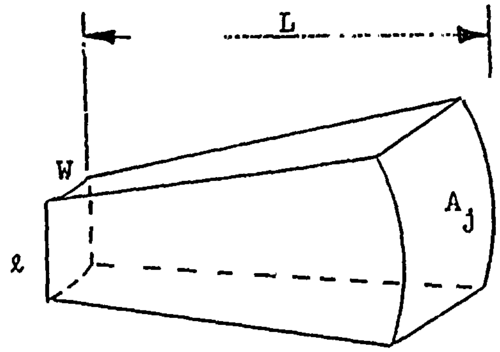


A_o = BREAK AREA

= PIPE FLOW AREA

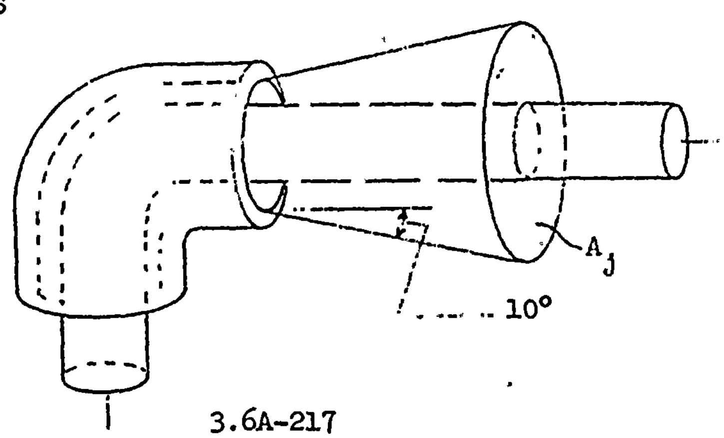
$\ell = 2\ D_o$

$W = \pi\ D_o/8$



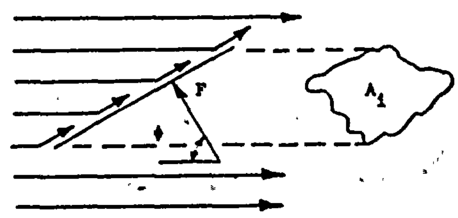
$$A_j = (\ell + 2L \tan 10°) \cdot (W + 2L \tan 10°)$$

C. SLEEVED DESIGN BREAKS



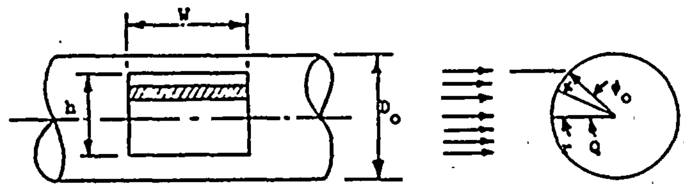
MAY 10 1974

A. SHAPE FACTOR FOR PLANE SURFACE



$S_F = \cos\phi$

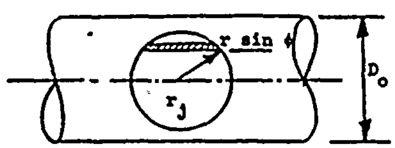
B. RECTANGULAR JET IMPINGING ON CYLINDRICAL TARGET



$S_F = 1 - \frac{h}{2D_o}$ for $h < D_o$

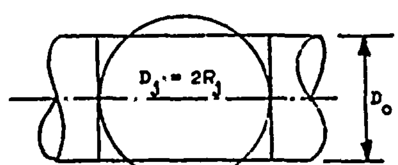
$S_F = 1/2$ for $h \geq D_o$

C. CIRCULAR JET IMPINGING ON PIPE WITH JET DIAMETER LESS THAN PIPE DIAMETER



$S_F = 1 - \frac{.288D_j}{D_o}$

D. CIRCULAR JET IMPINGING ON PIPE WITH JET DIAMETER GREATER THAN PIPE DIAMETER



$S_F = 0.712$

$D_j = 2R_j$

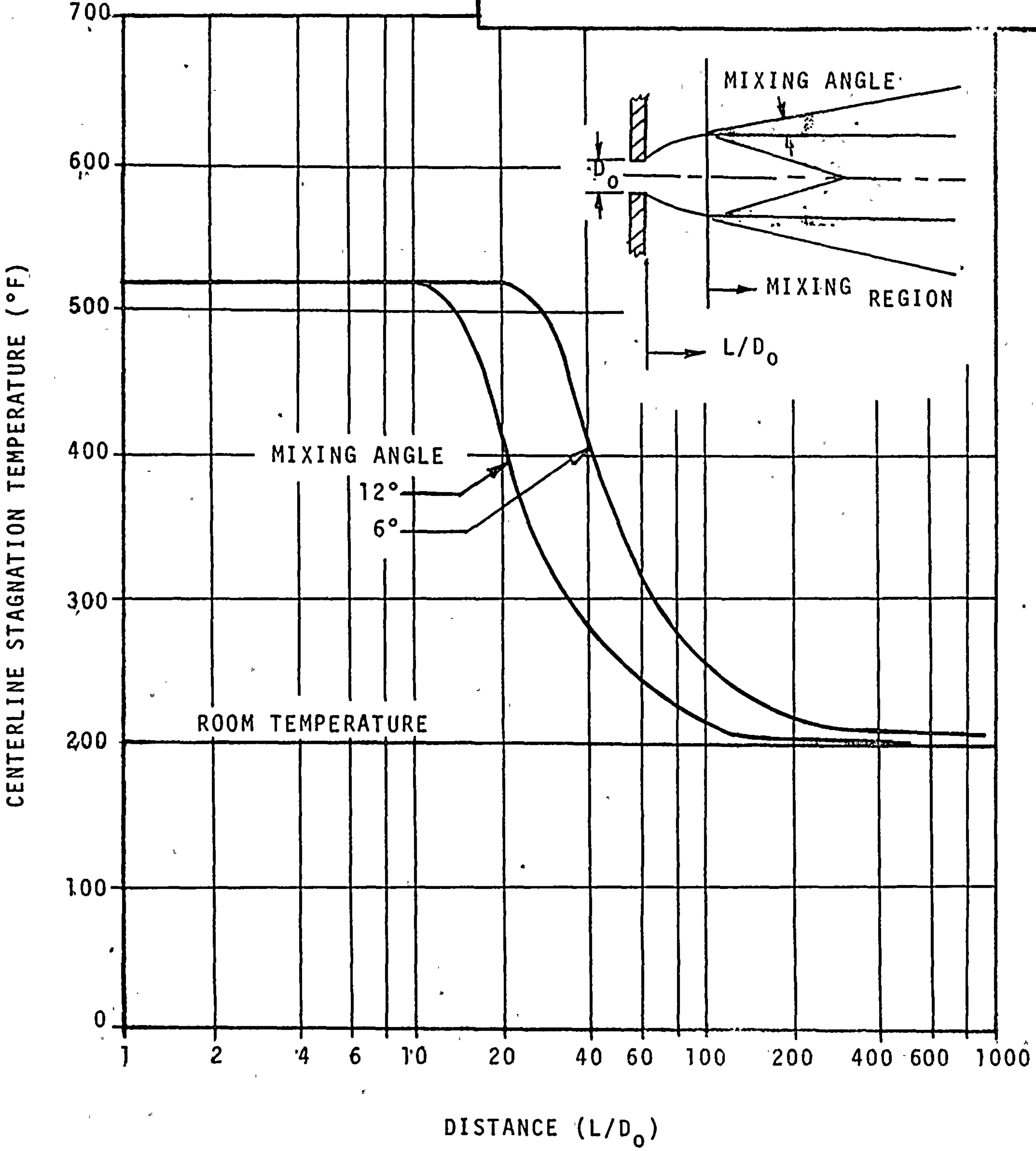
FIGURE C-3 DETERMINATION OF SHAPE FACTORS FOR JET IMPINGEMENT ANALYSIS

MAY 10 1974

Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE C-4

JET CENTERLINE STAGNATION TEMPERATURE



3.6A-219

MAY 10 1974

APPENDIX D

STRUCTURAL ANALYSIS

MAY 10 1974

MAY 10 1974

TABLE OF CONTENTS

		Page
D.1	INTRODUCTION	3.6A-225
D.2	ANALYSES OF PIPE RUPTURE RESTRAINTS	3.6A-227
D.2.1	Description of Restraints	3.6A-227
D.2.2	Load Combinations	3.6A-228
D.2.3	Methods of Analysis	3.6A-231
D.2.4	Material Properties	3.6A-234
D.2.5	Summary of Results	3.6A-235
D.3	IMPINGEMENT SLEEVES	3.6A-237
D.3.1	Description	3.6A-237
D.3.2	Methods of Analysis	3.6A-238
D.4	ANALYSES OF BUILDING STRUCTURE AND CONTAINMENT	3.6A-239
D.4.1	Description of Structures	3.6A-239
D.4.2	Load Combinations	3.6A-240
D.4.3	Methods of Analysis	3.6A-242
D.4.4	Material Properties	3.6A-246
D.4.5	Summary of Results	3.6A-246
D.4.6	Structural Modifications	3.6A-249

MAY 10 1974

D.1 INTRODUCTION

This appendix summarizes the evaluation of the adequacy of essential structures to withstand abnormal loading conditions in the event of a postulated pipe rupture of the main steam or feedwater lines. The evaluation includes comprehensive analyses of pipe rupture restraints, impingement sleeves, auxiliary building structure and the containment. In addition, a preliminary evaluation of high energy lines in the turbine building has been conducted to determine the potential consequences of pipe rupture with respect to the integrity of the component cooling water heat exchangers and other vital components. Design modifications have been proposed when deemed necessary to protect the integrity of structures vital for safe shutdown.

The analyses of pipe rupture restraints includes the determination of the stiffness properties necessary for characterizing dynamic pipe rupture behavior and establishes the ultimate load capacity of all restraints. These analyses also provide the data required to determine reaction loads at structural attachment points. Restraint modifications have been proposed to either reduce loads or prevent excessive pipe whip which would endanger concomitant structures. A description of the restraint systems and the analyses which were performed is provided in Appendix D.2.

In order to prevent unacceptable jet impingement loads at various postulated pipe break locations, the utilization of impingement sleeves has been proposed. A description of the analyses and design of these sleeves is given in Appendix D.3.

By including the modified pipe restraint system in the dynamic pipe whip model (as per Appendix B), predictions for restraint and structural loads were obtained. These loads were combined with the applicable jet impingement, compartment pressurization, seismic, and normal operating loads; and a structural analysis, as described in Appendix D.4, was performed on the affected components. The modified restraint system acts in a manner that prevents unrestrained pipe whip, and therefore, consideration of pipe impact onto primary structures was not necessary. Upon combining all credible loading conditions for the analyses completed to date, it has been established that with some modifications the efficacy of necessary structures will not be impaired beyond an acceptable level.

All of the proposed design modifications were incorporated into the analyses. For those changes related to the piping in area GW and along the pipeway, sufficient analysis has been completed to allow for detailed designs to be prepared. In all other areas, feasibility has been established and conceptual designs are being generated. The proposed structural modifications are itemized and described in Section D.4.6.

D.2 ANALYSES OF PIPE RUPTURE RESTRAINTS

D.2.1 Description of Restraints

The pipe rupture restraint system for the main steam and feedwater lines between the containment anchor and the G-line anchor consists of two subsystems: the pipeway on the northwest side of containment (area F), and the frame restraints in area GW (north auxiliary building). Locations of all restraints are shown, schematically, on the isometric piping drawings in Appendix B, Figures B-1 through B-6.

Restraint in the pipeway area is provided by several inter-connected space frames which are attached to the containment outer wall with anchor bolts. This arrangement results in a relatively stiff structure which develops high loads when subjected to dynamic load conditions. In order to attenuate the dynamic loads, the addition of annealed stainless steel (A304) rods at selected locations has been proposed as a design modification. Tables D-1 and D-2 provide a summary of restraint type and location (NSC node number as designated in the mathematical model) for each significant pipeway restraint.

The restraint system in area GW (feedwater and main steam lines 3 and 4) is comprised of a series of discrete, wide flange beam frames which carry loads generally independent of each other. These frames are attached with base plates and anchor bolts to adjacent concrete structure (floors, walls, and columns). Since these frame structures have high stiffnesses, rods have been proposed where necessary to reduce structural loads. Restraint types and specific locations within area GW are summarized in Tables D-3 and D-4.

Three typical frame restraints located in area GW are shown on Figures D-1, D-2 and D-3, and a portion of the pipeway space frame is depicted in Figures D-4 and D-5. Complete details of all restraints are available from the Pacific Gas and Electric (PG&E) contract drawings of Reference 20. The basic rod (U-bolt) design which has been proposed as a design modification is given in Figure D-6. The detailed design of connection points to existing or proposed frame structure has yet to be completed.

Additional elements of the main restraint systems are provided by the containment and G-line pipe anchors. The containment anchors (both feedwater and main steam) consist of an axisymmetric collar, flued head, which acts integrally with the pipe and is welded to a containment penetration pipe sleeve. Restraint at the G-line anchor is provided by a large welded plate girder. Locations of the pipe anchors are shown, schematically, on the isometric piping drawings in Appendix B, Figures B-1 through B-6.

D.2.2 Load Combinations

The frame restraints in area GW and along the pipeway were subjected to the load combination criteria of Reference 3.

MAY 10 1974

The area GW restraints actively function only during a postulated break, thus the only significant loadings are derived from the pipe rupture reactions.

Since the pipeway space frame in addition to providing rupture restraint also acts as a pipe support, it must be subjected to load combinations which include dead and seismic loads. The seismic loads consist of components due to horizontal and vertical response of both the pipe and the space frame. Applying the criteria of Reference 3 and utilizing the acceleration response spectra of Reference 21, the following load combination equation was derived for determining maximum beam attachment loads at the containment:

$$X = D + L + T_a + R_a + P_a + Y_j + Y_r + Y_m + Feqs$$

where

X = Design capability

D = Dead load

L = Live load

T_a = Thermal load generated by postulated break and including T_o

T_o = Thermal load under normal operating conditions

R_a = Pipe reactions under thermal conditions

P_a = Pressure equivalent static load

Y_j = Jet impingement loads

Y_r = Restraint reaction due to postulated break

Y_m = Missile impact loads

Feqs = Loads generated by the double design earthquake

Since the live, thermal, pressure, and jet impingement loads are negligible or nonexistent, and since the pipe is restrained in a manner which prevents generation of missiles, the above load combination reduces to

$$X = D + Y_r + Feqs.$$

The pipe anchors are subjected to loadings derived from normal conditions, severe and extreme environmental conditions, and postulated pipe rupture. For the analyses performed on the penetration sleeve reaction upon the containment, the loading types were combined according to the three criteria from Reference 3 as given in Section D.4.2. For the analyses performed on the steel portion of the anchors, the loadings were combined according to Section F of Reference 3 as follows:

$$Z = D + L + T_a + Y_r + Y_j + Y_m + Feqs$$

where Z = Section design capability based on 50% of ultimate strain.

The loads generated by the double design earthquake were conservatively assumed to be equal to two times the loads generated by the design earthquake. In addition, T_a was conservatively assumed to be less than T_o. Dead, live and jet impingement loads are negligible or nonexistent, thus the Section F criterion becomes

$$Z = T_o + Y_r + 2.0\ Feqo.$$

D.2.3 Methods of Analysis

The frame restraints in area GW were analyzed for a plane stress condition with the aid of the computer code MARC-CDC. MARC-CDC is a non-linear, finite element program developed at the MARC Analysis Corporation. It is a recognized program in the public domain, and is available and supported at the CYBERNET Centers of Control Data Corporation. A description of the software is provided in a three volume user manual (Reference 22).

Since the frames can be realistically considered as uncoupled plane structures, the plane stress analysis was conducted so that the effects of shear deformation would be accurately accounted for. Constitutive parameters were adjusted for WF beam flanges in consideration of the non-plane properties of the individual beams. All frame attachment points were considered as fixed boundaries unless initial analysis indicated a need to reduce moment transfer. Reduction of moment at attachments was accomplished by locally deleting a portion of the beam flanges from the model.

Due to the high degree of coupling between major restraint points in the pipeway, it was considered appropriate to conduct segmented three dimensional analyses of the space frame in order to establish the pertinent characteristics. These analyses were accomplsihed through the utilization of MARC-CDC beam elements (shear deformation neglected) joined as fixed members. Attachments to the containment wall were modeled as pinned joints in consideration of the relatively high flexibilities at these points.

For most of the load cases, the above analyses were completed within the elastic regime; however, in isolated instances significant yielding occurred. In such cases the isotropic strain hardening option of MARC-CDC was automatically invoked and a non-linear analysis was thus obtained. Typical finite element models used for the analyses of area GW and pipeway restraints are depicted in Figures D-1 through D-5.

The A304 rod restraints (Figure D-6) were analyzed as one-dimensional, purely extensional members with material strain hardening included in the plastic region. Yield load and elastic and strain hardening stiffnesses were based on the gross rod cross-section. Ultimate load was established by the tensile stress area of the threaded portion in conjunction with the maximum allowable strain. The following conventional formulations were utilized for the analysis.

$$K_e = \frac{A_r E}{L}$$

$$K_{st} = \frac{A_r E_{st}}{L}$$

$$P_y = \sigma_y A_r$$

$$P_{ma} = \sigma_{ma} A_s$$

MAY 10 1974

where

K_e = Elastic stiffness

K_{st} = Strain hardening stiffness

A_r = Gross cross sectional area of rod

A_s = Tensile stress area

E = Elastic modulus

E_{st} = Strain hardening modulus

L = Effective length of rod

P_y = Yield load

P_{ma} = Maximum allowable load (ultimate)

σ_y = Yield stress

σ_{ma} = Maximum allowable stress

In general, the pipe anchors were analyzed with classical structural analysis methods. However, in order to establish ultimate anchor capabilities, strain hardening effects for bending were incorporated by the methods defined in Reference 23.

D.2.4 Material Properties

Typical stress-strain curves for the steels used in the frame restraints of area GW and the pipeway are provided in Figures D-7 and D-8, respectively. Median material parameters, as noted, were used with the finite element models for characterization of the restraints. Ultimate load capability was based on a maximum strain allowable established as 50 percent of the minimum strain associated with ultimate stress.

Material property variations for annealed A304 stainless steel are shown on the stress-strain curves of Figure D-9. The median material properties as indicated were utilized for the pipe rupture analyses. However, the structural loads that could result from rods with upper bound properties were also determined. An increase of 15% in the median load was established from the data provided in Figure D-9 for the upper bound case. Amplification of loads due to high strain rates for annealed A304 are insignificant per Reference 24.

The material allowables utilized for the pipe anchor analyses are summarized as follows:

- Containment Pipe Anchor

  Concrete (bearing adjacent to penetration sleeve)

  Allowable design bearing stress = .75 f'_c

  where, f'_c = 3800 psi (Reference 4)

MAY 10 1974

Pipe Penetration Sleeve and Flanges (ASTM A-333 Gr 1, Reference 25)

Room Temp.	600°F
σ_y = 30. ksi	22.2 ksi
σ_u = 55. ksi	55.0 ksi

- G Row Pipe Anchor

Bolts (A490)

σ_y = 130. ksi (Reference 26)

σ_u = 150. ksi

Welded Plate Girder (A-516 Gr 70)

σ_y = 50. ksi (Reference 27)

Shear Lugs welded to pipe (A204 Gr C, Reference 20)

Room Temp.	600°F	
σ_y = 43. ksi	37.75 ksi	(Reference 28)
σ_u = 75. ksi	81.40 ksi	

D.2.5 Summary of Results

From the analyses, the force-deflection behavior of the restraints was compiled in a manner which defined a bilinear curve. That is, yield and ultimate loads and elastic and inelastic stiffnesses were determined. This data was then employed in the dynamic pipe whip model (Appendix B) to simulate restraints.

Tabulations of the characteristics for selected restraints along feedwater and main steam lines for area GW and the pipeway are provided in Tables D-5 and D-6. The characteristics for restraints which have not been listed are of similar magnitudes.

Analyses of the pipe anchors and adjacent structure for the predicted pipe rupture, thermal, and seismic loads indicate that the design criteria is satisfied assuming the following proposed modifications:

A. The main steam and feedwater containment anchors associated with lines 1 and 2 be modified to increase the torsional capability of the penetration sleeves. This modification can be implemented by providing a load path through additional beams on the outside of the containment. These beams would be attached between the exterior flange of the penetration sleeve and the space frame connection points on the containment surface.

B. The shear lugs on the G-row anchor should be modified so that the contact point is located within 1.0 inch of the attachment point. The moments developed by the pipe rupture loads (from main steam lines 1 and 2 and the feedwater line) at the G-line anchor cause yielding of the shear lugs. This has been incorporated into the dynamic analysis and it has been demonstrated that no pipe whips result from the lack of moment resistance at G-row anchor.

D.3 IMPINGEMENT SLEEVES

D.3.1 Description

The impingement sleeves (Figure D-10) which have been proposed for incorporation at selected pipe elbows throughout the feedwater and main steam lines act to prevent jet impingement upon critical structures and equipment which would occur as a result of design circumferential or longitudinal pipe breaks. In the event of longitudinal pipe rupture within the sleeve, the thrust is reacted fully by internal pressurization with the exception of two small jets which escape at the sleeve termination points.

The capability of the proposed sleeves to withstand the longitudinal and circumferential break jet impingement forces has been investigated and a detailed analysis has established the design requirements. In addition, the sleeves have been analyzed and designed to transfer rod restraint (U-bolt) loads which will be imposed at some sleeve locations (Figure D-6).

In order to retain the capability for field inspection of the pipes, the sleeves are designed to allow for field disassembly. This is accomplished by utilizing two half-sleeves bolted together through flange elements as shown schematically in Figure D-10. More detailed designs are being developed and will be presented in a future report.

D.3.2 Methods of Analysis

The impingement sleeves are designed to withstand internal pressurization resulting from circumferential or longitudinal pipe breaks and to transfer the bearing loads developed by the proposed rod restraints. Material allowables corresponding to fifty percent of ultimate strain were utilized for design values. Each sleeve is designed for a single event, and the strains and deflections associated with this event are allowed to exceed the elastic limit but not to exceed values associated with ultimate stresses.

Away from the bolted flanges, the state of stress in the sleeve under internal pressurization is essentially that described by shell membrane theory, and the sleeve in these regions was analyzed accordingly. The load transfer between the half-sleeves is accomplished by bolted flanges reinforced with gussets. The load transfer through these flanges generates local bending moments which are carried by the flange-gusset assembly. The analysis of this region was predicated upon the assumption of the material straining beyond the elastic limit as stated above.

D.4 ANALYSIS OF BUILDING STRUCTURE AND CONTAINMENT

D.4.1 Description of Structures

For the piping between containment and G-line anchors, the Category I structures of primary interest are the containment outer wall and area GW between elevations 115' and 140'. The major components which bound area GW are reinforced concrete walls, floors and interior columns. Major structural walls in this area act as shear members for reacting seismic loadings. Floors and interior columns carry dead loads and design live loads. The components which experience significant loading are listed below:

a. 3' concrete wall at column line 15^Z; separates area GW (and GE) from main auxiliary building

b. 2' concrete wall at column line J; separates area GW from turbine building

c. 2' concrete wall 17' east of column line N

d. 2' concrete floor slab at elevation 115'

e. 2'-8" concrete floor slab at elevation 140'

f. 2' interior concrete columns, numbers 4 and 5

g. containment outer wall, 3'-8" concrete shell with 3/8" steel interior liner

Figure D-11 provides a general layout for location of column lines. The complete structural details are provided in the PG&E contract drawings listed in Reference 20.

For postulated pipe ruptures occurring west of the G-line anchor and in the turbine building, the major structural considerations are dictated by the need to protect the component cooling water heat exchangers and related equipment from unacceptable damage levels. Methods of achieving this protection were discussed in Section 7.2.1.3. In order to substantiate the retention of structural integrity for the turbine building during a pipe rupture event, the following structure must be evaluated:

a. main exterior columns and vertical cross-bracing
b. primary shear walls (e.g. wall at column line 17)
c. turbine pedestal
d. concrete floor slab at elevation 140' and related steel frame superstructure.

Figure D-12 provides a general layout for location of column lines. Pertinent turbine building structural details are provided in the PG&E contract drawings listed in Reference 20.

D.4.2 Load Combinations

The structures described in the previous section are subjected to various loadings derived from normal conditions, severe and extreme environmental conditions, and postulated pipe rupture. Structural design loadings include dead, live, thermal and seismic; and rupture loads include differential compartment pressures, pipe restraint reactions and jet impingement. For the analyses performed, all loading types were combined according to the three criteria from Reference 3 as follows:

<u>Criterion 1.</u> $U = D + L + T_a + R_a + 1.5\ P_a$

<u>Criterion 2.</u> $U = D + L + T_a + R_a + 1.25\ P_a + Y_r + Y_j + Y_m + 1.25\ Feqo$

<u>Criterion 3.</u> $U = D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + Feqs$

where:

U = Section design capability

D = Dead load

L = Live load

T_a = Thermal loads due to postulated break

R_a = Pipe reactions under thermal conditions due to postulated break

Y_r = Restraint reactions due to postulated break

Y_j = Jet impingement loads

Y_m = Missile impact loads

Feqo = Loads generated by design earthquake

Feqs = Loads generated by double design earthquake.

All credible combinations within each individual criterion were considered in order to define the worst condition. For example, the floor slab in area

GW at elevation 140' developed the highest stresses when the design live loads (7.85 psi) were neglected. Furthermore, since <u>maximum</u> values for normal, seismic and abnormal loads were superimposed in all instances, credit was not taken for the potentially beneficial consequences of load time phasing. Restraint attachment points were analyzed for the load combination conditions described in Section D.2.2.

D.4.3 <u>Methods of Analysis</u>

Section design capability for reinforced concrete members was established by employing ultimate strength methods as defined in Chapter 10 of ACI-318 (Reference 29). Capacity reduction factors were applied to the calculated failure capability as per Chapter 9 of ACI-318. A value of 0.90 was utilized for members subjected to localized loading with bending, and a value of 0.70 was used for members acting as beam-columns.

In general, yield line theory (References 30, 31, and 32) was utilized to determine the ultimate capability of floors, walls and columns. Analyses performed in this manner assume yielding of the reinforcing steel and cracking of the concrete accompanied by a redistribution of moments. In order to ascertain the limiting load various crack patterns, yield lines defining a collapse mechanism must be considered. Once the critical pattern is determined, a virtual work analysis is performed and the ultimate load is established.

Before the use of the yield line method of analysis can be justified, compliance with criteria related to the reserve ductility of the structure must be established. Since concrete is essentially a brittle material, the ductility of reinforced concrete is derived from the capability of the steel reinforcing to sustain large deformations. However, effective "plastic hinges" cannot be developed to redistribute moments unless the steel controls the ultimate section capability. This is the case when the flexural member is underreinforced; that is, the steel yields well in advance of concrete crushing ($p < 0.50p_b$). As the member design approaches balanced ($p = p_b$) or overreinforced ($p > p_b$) conditions, crushing of the concrete becomes critical and an attendant loss of ductility is experienced. Therefore, the utilization of yield-line analysis is predicated on the necessary condition that the member be underreinforced. Furthermore, the ductility criterion is also a sufficient condition when other modes of failure (e.g. shear as opposed to bending) are determined to be secondary.

Since the primary floor and wall slabs and columns associated with the building structure are very lightly reinforced, the rotational capacity, ductility, of the members is sufficient to justify the use of yield-line analyses. For example, pertinent data for the walls at column line 15[7] and J can be summarized as follows:

	Wall @ 15[7]	Wall @ J
p_b : Balanced steel ratio	0.058	0.058
p : Actual steel ratio	0.0019	0.0042
$\therefore$ P/P_b	.033	.072

Thus, it is apparent that more than sufficient rotational capacity is available to sustain ultimate loads without concrete crushing.

The accuracy of yield-line analysis has been verified by comparison to test data (Reference 32). Available test results indicate that values obtained from yield line analysis are 80% to 90% of the actual capacity, thus the theoretical results are conservative. The conservation of the theory is derived from (1) the strain-hardening capability of the reinforcement, (2) membrane action of the slab near failure, and (3) the presence of compressive membrane stresses during bending. Test results of characteristic slabs also serve to alleviate uncertainties due to effects of triaxial states of stress at slab boundaries.

Yield line patterns resulting in minimum capability were established for the respective structures for each type of abnormal loading (impact, jet impingement and differential pressure). These loads were then conservatively superimposed with normal and seismic loads and ultimate structural capabilities were thus established for moment and shear resistance.

Frame restraint column and beam attachments were analyzed elastically by conventional methods available in numerous text books (e.g., Reference 33) and in accordance with the working stress method per AISC Specifications Part I (Reference 34). All column (beam) base connections were assumed to be bearing type and thus were subjected to tension and shear interaction. Interaction relationships were taken from References 34 and 35.

Two methods were utilized to characterize the restraint provided by wall and floor penetrations. When an applicable structural reaction boundary could be defined, the allowable loads and stiffnesses were determined by using an elastic-plastic, plane stress, closed form solution of a point load applied to a semi-infinite plate. This solution was based on the derivation given in Reference 36, which has been modified to include the nonlinear behavior of concrete. Conventional linear structural analysis methods were utilized for penetration characterization for instances where an applicable boundary could not be defined. In all cases the loads resulting from data calculated by the above techniques are conservative since no consideration was given to the kinetic energy absorbed locally by the impacting pipe.

D.4.4 Material Properties

The allowable design stresses for the materials as specified by PG&E contract drawings (Reference 20) are summarized as follows:

Concrete:

Allowable design stress = $0.85f'_c$

where, f'_c = 3000 psi for containment and turbine building

f'_c = 5000 psi for area GW

Reinforcing Steel (ASTM, A615, Reference 25)

Grade 60 (f_y = 60000 psi) for containment and some area GW columns

Grade 40 (f_y = 40000 psi) all other structure

Framing Steel - A36, f_y = 36000 psi (turbine building).

Base Plates - A36, f_y = 36000 psi (Reference 34)

Anchor Bolts - A490 (Reference 34)

Allowable working stress = 54000 psi (based on gross cross-sectional area)

D.4.5 Summary of Results

This section provides a summary of the results of the structural analyses of the pipe rupture restraint attachments and building structures and documents the comparisons between structural capabilities and predicted loads, including load combination requirements. The predicted pipe rupture loads are based on results found elsewhere in this report, and the normal operating and seismic loads are from data provided by PG&E, References 4, 6 and 37.

In Tables D-7 and D-8, load comparisons are made at individual pipe rupture restraint locations along feedwater lines 1 and 2, and 3 and 4, respectively. Load comparisons are provided for main steam lines 1 through 4 in Tables D-9 and D-10. In each case the overall allowable load was determined by selecting the weakest applicable component as critical. For example, at restraint 1047-2RT(-Z) there are four components which are loaded; that is, rod, frame, column attachments, and floor; the rod being critical in this example.

In general, the predicted applied loads represent values obtained for "median" restraint material properties. In a few instances the loads resulting from upper bound properties precipitate a change in the critical component thus reducing the available margin of safety. When this condition is present, it has been noted by indicating the upper bound load in parenthesis and the minimum margin of safety has been shown. The allowable load for all structural components includes the effects of load combinations when applicable. Results of the analyses show that all components meet the loading criteria specified herein assuming the proposed modifications are incorporated.

Table D-11 provides a comparison of predicted peak differential compartment pressures to allowable pressures as established by load combination 1 which is critical for pressure. With the exception of the containment outer wall for which elastic bending theory was employed, the ultimate pressure capability values were derived from yield line analyses. The wall at column line J, margin of safety of 4%, has been determined to be the critical component with respect to pressure.

Evaluation of load combinations 2 and 3 indicates that the minimum allowable loads, without exception, result from combination 2 because the structural capability is more sensitive to 1.25 P_a than to 1.0 Feqs plus 1.0 P_a. Furthermore, the dead and live loads and seismic stresses, as part of the load combination for structures in bending, had negligible effect on the capabilities of walls and columns and significant effect on floor capabilities. Table D-12 summarizes the loads induced into reactive structure of area GW by the pipe rupture restraints and compares this load to the allowable reactive load. In all cases the structural capability exceeds the applied load.

For the pipeway area the maximum restraint reactions experienced by the containment are less than 1000 kips while the capability is in excess of 3000 kips. Therefore, in all cases the containment wall has greater capability than the attachments which are associated with the pipeway structure.

MAY 10 1974

Table D-13 compares the predicted jet impingement forces at unsleeved breaks to the allowable loads. All potentially impinged structures in area GW can withstand the predicted forces without modifications.

For all of the above tables the allowable load entries with "greater than" quantities imply that the particular value can be refined upward with additional analyses. Loads associated with the proposed structural modifications, which are required to provide rod attachment points, are not presented since final design values have not been established.

Insofar as the results have been completed, it is noted that no structural failures will result from the predicted loadings. However, implementation of a number of structural modifications has been proposed and was assumed throughout the analyses. The proposed modifications are discussed in the following section.

D.4.6 Structural Modifications

As indicated in Appendix D.2, it is proposed that a number of stainless steel rods be added to the pipe restraint system in order to maintain the loads at an acceptable level. Additional frame structure will also be required at selected points either to enhance load distribution or to provide attachment points for the rods. Furthermore, it is proposed that minor modifications to some existing restraints be implemented to increase load capability and ensure structural integrity.

In particular, it is proposed that the following modifications be incorporated:

Feedwater and Main Steam Lines 3 and 4

- 1047-2RT* and 1047-11RT - Extend two beams to the ceiling at El. 140'. Add eight, 1 3/4" diameter bolts to the base plates of Detail 3E/464 (Dwg. No. 438464) to increase ±Z capability.

- 1047-4RT - Remove short length of flanges of the W 36x300 at El. 115' in order to redistribute loading.

- 1030-7RT and 4120 - Remove short length of flange of horizontal beam at El. 126'-9" adjacent to the wall at 15^7. Strengthen existing wall attachment to increase shear capacity.

- 3220, 3170, 4085 and 4067 - Add a steel frame restraint between floors at El. 115' and El. 140' and between column lines 15 and 15^7 and rows J and K. The frame will provide attachment points for rod restraints necessary to prevent unrestrained pipe whip.

Feedwater Lines 1 and 2

- 1239 - Add vertical beam at El. 114'-9 1/2" for rod attachment.

* See Figures B-1 through B-6 for location of restraints and piping node numbers.

MAY 10 1974

- 1153 and 1263 - Add vertical beam between Els. 109'-4" and 118'-2". Modify horizontal beams from wide flange to box section. Add a horizontal beam at El. 118'-2".

- 1160 - Add short vertical and horizontal beams for rod attachment.

Main Steam Lines 1 and 2

- 1190 and 2190 - Modify the two horizontal WF beams at El. 119'-1" to box sections to increase capability.

- 1180 and 2185 - Attach a WF 36x300 to the five space frame attachment points at El. 123'-0" in order to redistribute loads.

- 1172 and 2172 - Add vertical beam between Els. 119'-1" and 114'-9 1/2" and modify existing WF beams to box sections.

- 1060 and 2015 - Remove vertical beam in penetration of wall at 15 to allow rod restraint to function effectively.

- Selected bolted connections within the pipeway space frame must be welded in order to transfer the applied loads.

G-Line Anchor Area

- Feedwater nodes 1369, 1189 and 1191; and main steam nodes 3080, 4025, 3110, 3070, 4040, 4020, 3140, 4055, 2010 and 1055.

- Add frames and rod restraints in the area between column lines 15 and 15[7] and between G-row anchor and J row to prevent unrestrained pipe whip.

Further modification of selected frame restraints in the form of structural shims is proposed and considered necessary to control the amount of kinetic energy generated during a rupture event. The shims act to reduce the gap through which the pipe moves prior to impact and thus are effective in easing the dynamic load amplification at restraints.

As a result of the need for additional venting area to reduce GW compartment pressures (see Appendix A.3), it has been proposed that the construction opening in the wall at column line J remain open. This additional opening will influence the strength and stiffness of the shear wall in terms of the capability to resist seismic loading. Analyses of this condition indicate that shear stresses are increased by 30% but remain well within the allowable; therefore replacement structure is not required.

Analysis of modifications that are associated with high energy lines within the turbine building are currently in progress.

TABLE D-1

RESTRAINT SUMMARY FEEDWATER LINES 1 AND 2

PG&E Restraint Number	NSC Node Number	Direction*					
		+X	-X	+Y	-Y	+Z	-Z
	1100	P	P	P	P	-	-
1046-2RT	1111	F	F	R	F	-	-
1046-3RT	1114	F	F	-	-	F	R
1046-5RT	1117	R	F	-	-	F	F
1046-6RT	1121	-	-	F	R	F	F
1046-7RT	1128	-	-	F	F	F	R
1046-20RT	1137	-	-	F	F	F	R
1046-20RT	1139	-	-	F	F	F	F
1046-21RT	1150	-	-	F	F	F	R
	1153	-	R	-	-	-	-
	1160	R	-	-	R	-	-
	1185	P	P	P	P	-	-
	1189	R	-	-	-	-	-
	1191	-	-	-	-	R	-
	1200	P	P	P	P	-	-
1046-10RT	1211	F	F	R	F	-	-
1046-11RT	1215	F	F	-	-	F	R
1046-13RT	1216	R	F	-	-	F	F
1046-14RT	1219	-	-	F	R	F	F
1046-15RT	1226	-	-	F	F	F	R
	1239	-	-	-	-	-	R
	1241	-	R	-	-	-	-
	1261	-	R	-	-	-	-
	1263	-	R	-	-	-	-
	1267	-	R	R	-	-	-

* Some restraints are not parallel to global coordinate system.

F - Frame

P - Wall or floor penetration

R - Rod

MAY 10 1974

TABLE D-2

RESTRAINT SUMMARY MAIN STEAM LINES 1 AND 2

PG&E Restraint Number	NSC Node Number	Direction*					
		+X	-X	+Y	-Y	+Z	-Z
	1213	P	P	P	P	-	-
	1185	-	-	R	-	-	-
	1178	-	-	-	-	-	R
1024-6RT	1175	F	F	-	-	F	F
	1172	R	-	-	-	-	-
1024-7RT	1167	-	-	F	R	R	F
1024-8RT	1157	-	-	F	R	F	R
1046-20RT	1140	-	-	F	R	F	F
1046-20RT	1135	-	-	R	F	F	R
	1121	-	-	-	-	-	R
	1112	-	R	-	-	-	-
	1062	-	P	P	P	-	-
	1060	R	-	-	-	-	-
	1055	-	-	-	R	R	-
	1312	P	P	-	-	P	P
	2225	P	P	P	P	-	-
1025-5RT	2205	F	F	F	F	-	-
	2190	-	-	R	-	-	-
	2185	-	-	-	-	-	R
1025-6RT	2180	F	F	-	-	F	F
	2177	R	-	-	-	-	-
1025-7RT	2170	-	-	F	R	R	F
1025-8RT	2140	-	-	F	F	F	F
1025-10RT	2125	-	-	F	R	F	R
1046-20RT	2105	-	-	F	R	F	F
1046-20RT	2100	-	-	R	F	F	R
	2080	-	-	-	-	-	R
	2075	-	R	-	-	-	-
	2020	-	P	P	P	-	-
	2015	R	-	-	-	-	-
	2010	-	-	-	R	R	-
	2240	P	P	-	-	P	P

* Some restraints are not parallel to global coordinate system.

F - Frame

P - Wall or floor penetration

R - Rod

MAY 10 1974

TABLE D-3

RESTRAINT SUMMARY FEEDWATER LINES 3 AND 4

PG&E Restraint Number	NSC Node Number	Direction					
		+X	-X	+Y	-Y	+Z	-Z
	1302	P	P	P	P	-	-
	1303	-	-	R	-	-	-
	1305	-	-	-	-	R	-
1047-2RT	1309	F	W	-	-	F	R
1047-3RT	1315	F	F	F	F	-	-
1047-4RT	1327	R	F	F	F	-	-
1047-5RT	1332	-	-	F	F	R	F
1047-6RT	1335	-	-	F	F	W	F
1047-17RT	1336	-	-	F	F	F	F
1047-7RT	1339	-	-	F	F	W	F
1047-8RT	1343	-	-	F	F	W	F
1047-20RT	1349	-	-	F	W	W	F
1047-20RT	1356	-	-	F	W	W	F
	1363	-	-	P	P	P	P
	1369	-	-	R	-	-	-
	1402	P	P	P	P	-	-
	1403	-	-	R	-	-	-
	1405	-	-	-	-	R	-
1047-11RT	1409	W	F	-	-	F	R
1047-12RT	1415	F	F	F	F	-	-
1047-13RT	1427	R	F	F	F	-	-
1047-14RT	1433	-	-	F	F	R	F
1047-15RT	1437	-	-	F	F	F	F
1047-20RT	1443	-	-	F	W	F	F
1047-20RT	1450	-	-	F	W	F	F
	1457	-	-	P	P	P	P

F - Frame

P - Wall or floor penetration

R - Rod

W - Wall or floor

TABLE D-4

RESTRAINT SUMMARY MAIN STEAM LINES 3 AND 4

PG&E Restraint Number	NSC Node Number	Direction					
		+X	-X	+Y	-Y	+Z	-Z
	3292	P	P	P	P	-	-
1030-5RT	3290	P	P	P	P	-	-
1030-6RT	3285	F	F	F	F	-	-
1030-7RT	3275	-	-	F	F	R	-
1030-8RT	3270	-	-	F	F	W	F
1030-10RT	3250	-	-	F	F	W	F
1030-11RT	3245	-	-	F	F	W	F
	3220	-	-	-	-	R	-
	3180	R	-	-	-	-	-
	3170	-	-	-	-	-	R
	3150	-	-	P	P	P	P
	3140	-	-	-	-	-	R
	3130	-	R	-	-	-	-
	3120	P	P	P	P	-	-
	3110	-	-	R	-	-	-
	3100	-	-	-	-	R	-
	3080	R	-	-	-	-	-
	3070	-	-	-	R	-	-
	3420	P	P	-	-	P	P
	3500	P	P	-	-	P	P
1030-14RT	3593	-	-	F	F	F	F

F - Frame

P - Wall or floor penetration

R - Rod

W - Wall

MAY 10 1974

Nuclear Services Corporation

TABLE D-4 (Cont'd)

RESTRAINT SUMMARY MAIN STEAM LINES 3 AND 4

PG&E Restraint Number	NSC Node Number	Direction +X	-X	+Y	-Y	+Z	-Z
	4132	P	P	P	P	-	-
1031-5RT	4130	F	W	F	F	-	-
1031-6RT	4127	F	F	F	F	-	-
	4120	-	-	F	F	R	-
1031-7RT	4110	-	-	F	F	F	F
1031-8RT	4095	-	-	F	F	F	F
	4085	-	-	-	-	R	-
	4070	R	-	-	-	-	-
	4067	-	-	-	-	-	R
	4060	-	-	P	P	P	P
	4055	-	-	-	-	-	R
	4050	-	R	-	-	-	-
	4045	P	P	P	P	-	-
	4040	-	-	R	-	-	-
	4035	-	-	-	-	R	-
	4025	R	-	-	-	-	-
	4020	-	-	-	R	-	-
	4013	P	P	-	-	P	P
	4150	P	P	-	-	P	P
1031-11RT	4190	-	-	F	F	F	F

F - Frame

P - Wall or floor penetration

R - Rod

W - Wall

MAY 10 1974

TABLE D-5
SUMMARY OF TYPICAL RESTRAINT PROPERTIES FOR FEEDWATER AND MAIN STEAM 1 AND 2

PG&E Restraint Number	Node	Load Direction	Restraint Type	Elastic Deflection (in.)	Plastic Deflection (in.)	Elastic Stiffness lbs/inx10^{-6}	Strain Hardening Stiffness lbs/inx10^{-6}	Yield Load (kips)	Maximum Allowable Load (kips)
-	1100	±X, +Y	Containment Penetration	0.0142	0.0235	60.0	44.2	852.0	1893.0
-	1100	-Y	Containment Penetration	0.0144	0.0222	59.0	46.8	852.0	1893.0
1046-2RT	1111	+Y	Rod	0.107	5.81	2.69	0.013	286.3	362.0
1046-3RT	1114	-Z	Rod	0.088	4.67	3.26	0.0162	286.3	362.0
1046-5RT	1117	+X	Rod	0.088	5.22	3.88	0.019	340.7	440.0
1046-6RT	1121	-Y	Rod	0.088	5.22	3.88	0.019	340.7	440.0
1046-7RT & 1046-20RT	1128 & 1137	-Z	Rod	0.09	4.43	1.72	0.0085	151.3	189.0
1046-21RT	1150	-Z	Rod	0.121	6.76	1.96	0.0097	236.7	302.3
-	1153	-X	Rod	0.096	5.38	2.45	0.0122	236.7	302.6
-	1160	+X	Rod	0.088	5.22	3.88	0.019	340.7	440.0
-	1213	±X, +Y	Containment Penetration	0.0230	0.0391	64.8	46.6	1490.	3312.
-	1213	-Y	Containment Penetration	0.0218	0.0352	68.3	51.8	1490.	3312.
-	1185	+Y	Rod	0.26511	14.870	1.749	0.008683	463.68	592.80
-	1172	+X	Rod	0.129	16.079	4.28	0.02124	552.41	893.94
1024-7RT	1167	-Y	Rod	0.1962	24.44	2.641	0.01311	518.25	838.66
1024-7RT	1167	+Z	Rod	0.103	5.99	5.86	0.0291	605.7	780.0

Nuclear Services Corporation

TABLE D-6

SUMMARY OF TYPICAL RESTRAINT PROPERTIES FOR FEEDWATER AND MAIN STEAM 3 AND 4

PG&E Restraint Number	Node	Load Direction	Restraint Type	Elastic Deflection (in.)	Plastic Deflection (in.)	Elastic Stiffness lbs/inx10^{-6}	Strain Hardening Stiffness lbs/inx10^{-6}	Yield Load (kips)	Maximum Allowable Load (kips)
-	1303	+Y	Rod	0.103	5.80	2.29	0.0113	236.8	302.3
-	1305	+Z	Rod	0.103	5.48	2.772	0.0138	286.3	362.0
1047-2RT	1309	+X	Frame	0.225	-	3.155	-	710.0	-
1047-2RT	1309	-X	Column	0.04422	0.0	39.8	39.8	1760.	1760.
1047-2RT	1309	+Z	Frame	0.272	-	1.75	-	476.5	-
1047-2RT	1309	-Z	Rod	0.079	4.636	4.30	0.0214	340.7	439.9
1031-5RT	4130	+X	Frame	0.0695	-	20.08	-	1395.0	-
1031-5RT	4130	-X	Wall	0.0187	0.0696	23.8	4.9	444.	785.
1031-5RT	4130	+Y	Frame	0.0374	-	26.67	-	997.0	-
1031-5RT	4130	-Y	Frame	0.0449	-	21.07	-	945.	-
-	4120	+Z	Rod	0.142	9.3	5.37	0.0266	766.6	1014.0
1030-8RT	3270	+Y	Frame	0.0391	-	15.94	-	623.9	-
1030-8RT	3270	-Y	Frame	0.0494	-	11.98	-	592.2	-
1030-8RT	3270	+Z	Wall	0.03251	0.0	9.5	9.5	3088.0	3088.0
1030-8RT	3270	-Z	Frame	0.0740	-	11.53	-	853.6	-
1030-10RT & 1030-11RT	3250 & 3245	+Y	Frame	0.0463	-	16.36	-	758.0	-

3.6A-259

MAY 10 1974

TABLE D-7

COMPARISON OF COMBINED RESTRAINT

LOADS TO ALLOWABLE RESTRAINT LOADS

FEEDWATER LINES 1 & 2(1)

PG&E Restraint Number	NSC Node Number	Dir.	Applied Load (2) (kips)	Allowable Load (3) (kips)	Critical Component	Margin of Safety
1046-2RT	1111	+Y	305	362	Rod	0.19
1046-3RT	1114	-Z	323	362	Rod	0.12
1046-5RT	1117	+X	381	440	Rod	0.15
1046-6RT	1121	-Y	411	440	Rod	0.07
1046-7RT	1128	-Z	188	189	Rod	0.01
1046-20RT	1137	-Z	160	189	Rod	0.18
1046-21RT	1150	-Z	258	302	Rod	0.17
	1153	-X	240	302	Rod	0.26
	1160	-Y	156	189	Rod	0.21
1046-10RT	1211	+Y	313	362	Rod	0.17
1046-11RT	1215	-Z	323	362	Rod	0.12
1046-13RT	1216	+X	391	440	Rod	0.13
1046-14RT	1219	-Y	391	440	Rod	0.13
1046-15RT	1226	-Y	942	1164	Attachments	0.24
1046-15RT	1226	-Z	155	189	Rod	0.22
	1239	-Z	158	189	Rod	0.20
	1241	-Z	163	189	Rod	0.16
	1261	-X	251	302	Rod	0.20
	1263	-X	250	302	Rod	0.21
	1267	-X	197	238	Rod	0.21
	1267	+Y	196	238	Rod	0.21

(1) Only those restraints with significant loading are enumerated.

(2) Predicted combined restraint loads (See Section D.2.2)

(3) Based on analysis of restraint frame, rod, attachments, and containment or building structure.

MAY 10 1974

TABLE D-8

COMPARISON OF PIPE BREAK RESTRAINT

LOADS TO ALLOWABLE RESTRAINT LOADS

FEEDWATER LINES 3 & 4 (1)

PG&E Restraint Number	NSC Node Number	Dir.	Applied Load (2) (kips	Allowable Load (3) (kips)	Critical Component	Margin of Safety
	1302	+Y	1862	1893	Containment wall	0.02
1047-2RT	1309	-Z	396(455)	475(4)	Floor attachments	0.04
1047-2RT	1309	+X	264(304)	313	Floor attachments	0.03
1047-3RT	1315	-Y	1060	1800	Frame	0.70
1047-3RT	1315	+Y	373	700	Floor attachments	0.88
1047-4RT	1327	+X	342	362(4)	Rod	0.06
1047-4RT	1327	-Y	671	>824(5)	Frame	>0.23
1047-5RT	1332	+Z	325	362	Rod	0.11
1047-5RT	1332	+Y	343	>514(5)	Frame	>0.50
1047-6RT	1335	-Y	425	2310	Floor at El. 115'	High
1047-6RT	1335	+Z	687	> 975	Wall at 15[7]	>0.42
1047-7RT	1339	-Z	940	975	Wall at 15[7]	0.04
1047-7RT	1339	+Z	522	> 975	Wall at 15[7]	>0.87
1047-8RT	1343	-Y	308	2310	Floor at El. 115'	High
1047-8RT	1343	-Z	436	> 975	Wall at 15[7]	>1.24
1047-11RT	1409	+X	259	849	Column #4	High
1047-11RT	1409	-Z	396	440	Rod	0.11
1047-13RT	1427	+X	288	362	Rod	0.26
1047-15RT	1437	+Z	565	> 975	Wall at 15[7]	>0.73
1047-20RT	1356	+Y	324	443	Wall attachments	0.37
1047-20RT	1450	-Z	347	375	Wall attachments	0.08
1047-20RT	1443	+Z	204	1058	Wall at 15[7]	High
	1305	+Z	252	362(4)	Rod	0.44
	1303	+Y	257	302(4)	Rod	0.18
	1457	-Z	1543	2397	Wall at J	0.55

(1) Only those restraints with significant loading are enumerated.

(2) Predicted restraint loads from pipe rupture analysis using median material properties. Upper bound loads shown in parenthesis if the margin of safety is affected.

(3) Based on analysis of restraint (frame or rod), restraint column or beam attachment, and wall (floor) structure.

(4) Values reflect capability of proposed redesign (see Appendix D.4).

(5) Yield point of frame restraint.

MAY 10 1974

TABLE D-9

COMPARISON OF COMBINED RESTRAINT

LOADS TO ALLOWABLE RESTRAINT LOADS

MAIN STEAM LINES 1 & 2(1)

PG&E Restraint Number	NSC Node Number	Dir.	Applied Load (2) (kips)	Allowable Load (3) (kips)	Critical Component	Margin of Safety
	1185	+Y	515	593	Rod	0.15
	1178	-Z	870	1014	Rod	0.17
	1172	+X	920(1058)	1101	Attachment	0.04
1024-7RT	1167	-Y	704	839	Rod	0.19
1024-7RT	1167	+Z	619	780	Rod	0.26
1024-8RT	1157	-Y	343	440	Rod	0.28
1024-8RT	1157	-Z	405	440	Rod	0.09
1024-20RT	1140	-Y	340	440	Rod	0.29
1024-20RT	1135	-Z	345	440	Rod	0.28
	1121	-Z	673	780	Rod	0.16
	1112	-X	423	440	Rod	0.04
	1062	+Y	298	1858	Wall @ 15	High
	1312	+X	726	2473	Floor @ El. 140'	High
	2190	+Y	515	593	Rod	0.15
	2185	-Z	870	1014	Rod	0.17
	2177	+X	920	932	Rod	0.01
1025-7RT	2170	-Y	704	839	Rod	0.19
1025-7RT	2170	+Z	619	780	Rod	0.26
1025-10RT	2125	-Y	343	440	Rod	0.28
1025-10RT	2125	-Z	405	440	Rod	0.09
1025-20RT	2105	-Z	340	440	Rod	0.29
1025-20RT	2100	-Z	345	440	Rod	0.28
	2080	-Z	673	780	Rod	0.16
	2075	-X	423	440	Rod	0.04

(1) Only those restraints with significant loading are enumerated.

(2) Predicted combined restraint loads (See Section D.2.2) using median material properties. Upper bound loads shown in parenthesis if the margin of safety is affected.

(3) Based on analysis of restraint frame, rod, attachments and containment or building structure.

MAY 10 1974

Nuclear Services Corporation

TABLE D-10

COMPARISON OF PIPE BREAK RESTRAINT

LOADS TO ALLOWABLE RESTRAINT LOADS

MAIN STEAM LINES 3 & 4 (1)

PG&E Restraint Number	NSC Node Number	Dir.	Applied Load (2) (kips)	Allowable Load (3) (kips)	Critical Component	Margin of Safety
	3295	+X	1772	3312	Containment wall	0.87
1030-6RT	3285	+X	361	657	Floor attachments	0.82
1030-6RT	3285	-Y	384	>1282(5)	Frame	High
1030-7RT	3275	-Y	(4)	(4)	Wall attachment	
1030-7RT	3275	+Z	783(900)	975	Wall at 15[7]	0.08
1030-8RT	3270	-Y	556	>592(5)	Frame	>0.06
1030-10RT	3250	-Y	626	>715(5)	Frame	>0.14
1030-10RT	3250	+Y	757	>758(5)	Frame	>0.00
1030-11RT	3245	+Y	464	>758(5)	Frame	>0.63
1030-11RT	3245	-Z	616	>975	Wall @ 15[7]	>0.58
	3245	-Y	298	>714(5)	Frame	>1.39
	3500	+X	2676	3846	Floor @ Elev. 140'	0.44
	4120	-Y	(4)	(4)	Wall attachment	
1031-6RT	4127	-X	431	657	Floor attachments	0.52
1031-6RT	4127	-Y	1059	>1282(5)	Frame	>0.21
1031-7RT	4110	-Y	448	>714(5)	Frame	>0.59
1031-8RT	4095	-Y	415	>714(5)	Frame	>0.72
1031-8RT	4095	-Z	421	1119	Wall attachments	1.66

(1) Only those restraints with significant loading are enumerated.

(2) Predicted restraint loads from pipe rupture analysis using median material properties. Upper bound loads shown in parenthesis if the margin of safety is affected.

(3) Based on analysis of restraint (frame or rod), restraint column or beam attachment, and wall (floor) structure.

(4) Redesign proposed to reduce applied load and increase capability (see Section D.4.6).

(5) Yield point of frame restraint.

MAY 10 1974

TABLE D-11

RESULTS OF ANALYSIS OF

LOAD COMBINATION 1(1)

AUXILIARY BUILDING-AREA GW

Structural Component	Ultimate Pressure Capability(psi)	Allowable Pressure(psi)(2)	Predicted Peak Pressure(psi)(3)
Floor slab at Elev. 115'	19.8	13.2	9.26
Wall at column line 15	17.4	11.6	9.26
Wall at column line "J"	14.5	9.6	9.26
Wall at 17' east of column line "N"	19.9	13.3	<9.26
Floor slab at Elev. 140'	>19.8	>13.2	9.26
Containment	>13.9	> 9.3	9.26

(1) Load Combination 1: $U \geq D + L + T_a + R_a + 1.5\ P_a$

where U = Section design capability (psi)

P_a= Predicted peak pressure (psi)

(2) Allowable pressure = $\frac{U^*}{1.5}$

where $U^* = U - (D + L + T_a + R_a)$ = Ultimate pressure capability

(3) Based on results presented in Appendix A.3.

MAY 10 1974

TABLE D-12

RESULTS OF ANALYSIS OF

LOAD COMBINATIONS 2 AND 3[1] FOR RESTRAINT REACTIONS

AUXILIARY BUILDING-AREA GW

Structural Component	Allowable Y_r (kips) [2]	Maximum Predicted Y_r (kips) [3]
Floor slab at Elev. 115'-West of Col. line 'K'	384	192
Floor slab at Elev. 115'-East of Col. line 'K'	2310	744
Wall at column line 15^{7} between pilasters	975	765
Wall at column line 15^{7} at pilaster	975	764
Floor slab at Elev. 140'	963	757
Interior Column #4	849	259

(1) See discussion of Appendix D.4.2 for definition of terms and constraint equations.

(2) Load combination 2 is critical in all cases since the 1.25 P_a (11.58 psi) term has a greater effect on the capability than does the double design earthquake (1.0 Feqs).

(3) Predicted restraint loads (Y_r) based on pipe whip analysis (Appendix B).

MAY 10 1974

TABLE D-13

RESULTS OF ANALYSIS OF LOAD

COMBINATIONS 2 AND 3(1) FOR JET IMPINGEMENT

FEEDWATER AND MAIN STEAM LINES 1, 2, 3 & 4

Structural Component	Break Locations(2)	Allowable Load (Y_j)-kips(3)	Maximum Predicted Load-kips(4)
Feedwater			
Containment	1303, 1403	High	120
Wall @ Col. Line 15⁷	1303,1369,1429, 1403	> 975	120
Wall @ Col. Line J	1000	> 716	440
Floor @ Elev. 107'	1375,1000	359	200
Main Steam			
Containment	1185,2190	High	1200
Column No. 5	3295	609	549
Floor @ Elev. 115'	3273,4115	1703	920
Floor @ Elev. 140'	3273,3540,1052, 2005,3060,4005	1903	920
Floor @ Elev. 107'	1052,2005,3060, 4005,1050,2001, 3050,4001	359	270
Wall @ Col. Line J	1050,2001,3050, 4001	> 716	540
Wall @ Col. Line 15⁷	4115,4135	> 975	869

(1) See Appendix D.4.2 for definition of terms and constraint equations.

(2) Design breaks without sleeves.

(3) Allowable total jet impingement force when load combination 2 or 3 is invoked.

(4) Predicted maximum loads based on jet impingement analysis (Appendix C), utilizing a dynamic load factor equal to 2.0. (Reference 19).

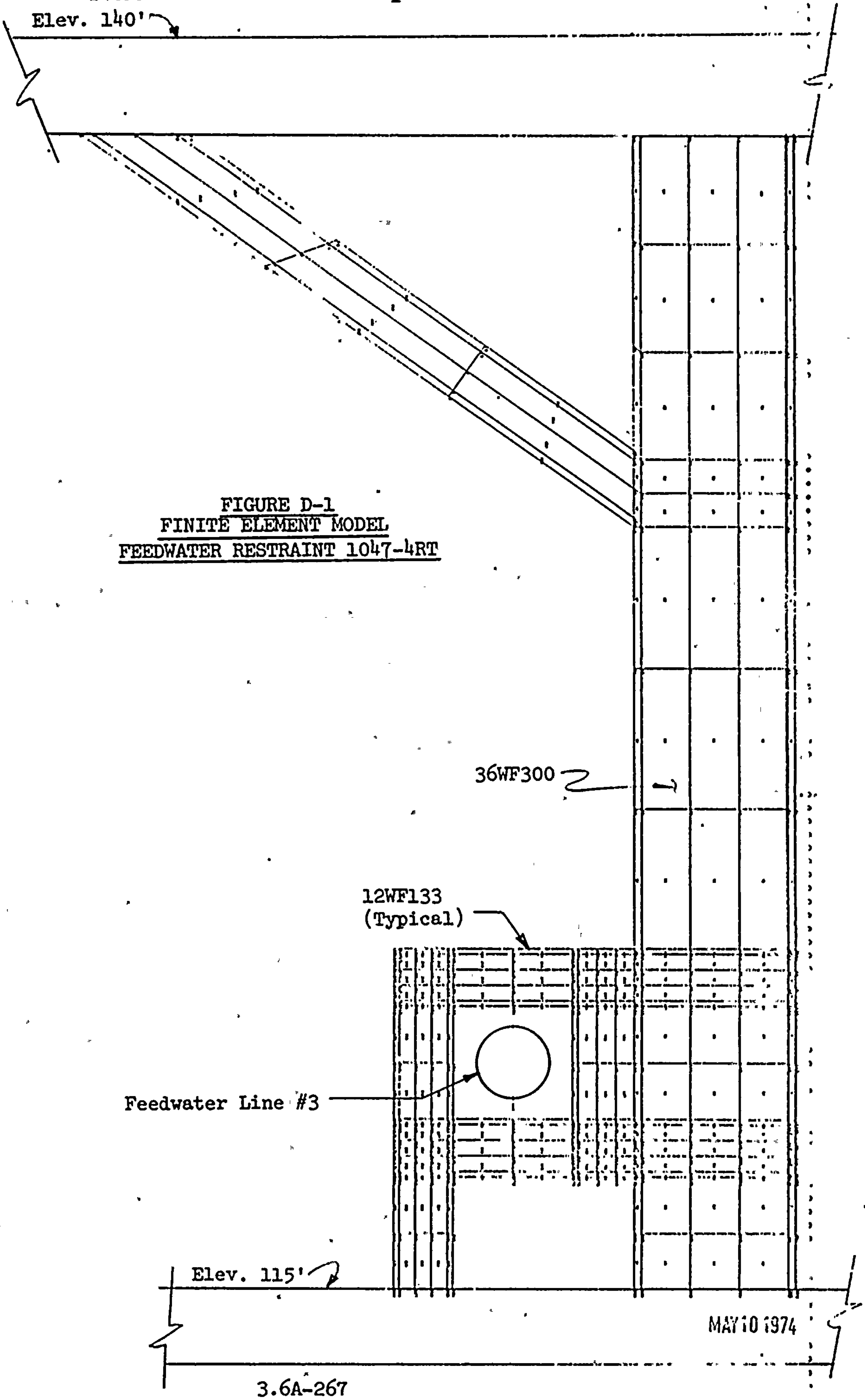


FIGURE D-1
FINITE ELEMENT MODEL
FEEDWATER RESTRAINT 1047-4RT

MAY 10 1974

FIGURE D-2
FINITE ELEMENT MODEL
FEEDWATER RESTRAINT 1047-12RT

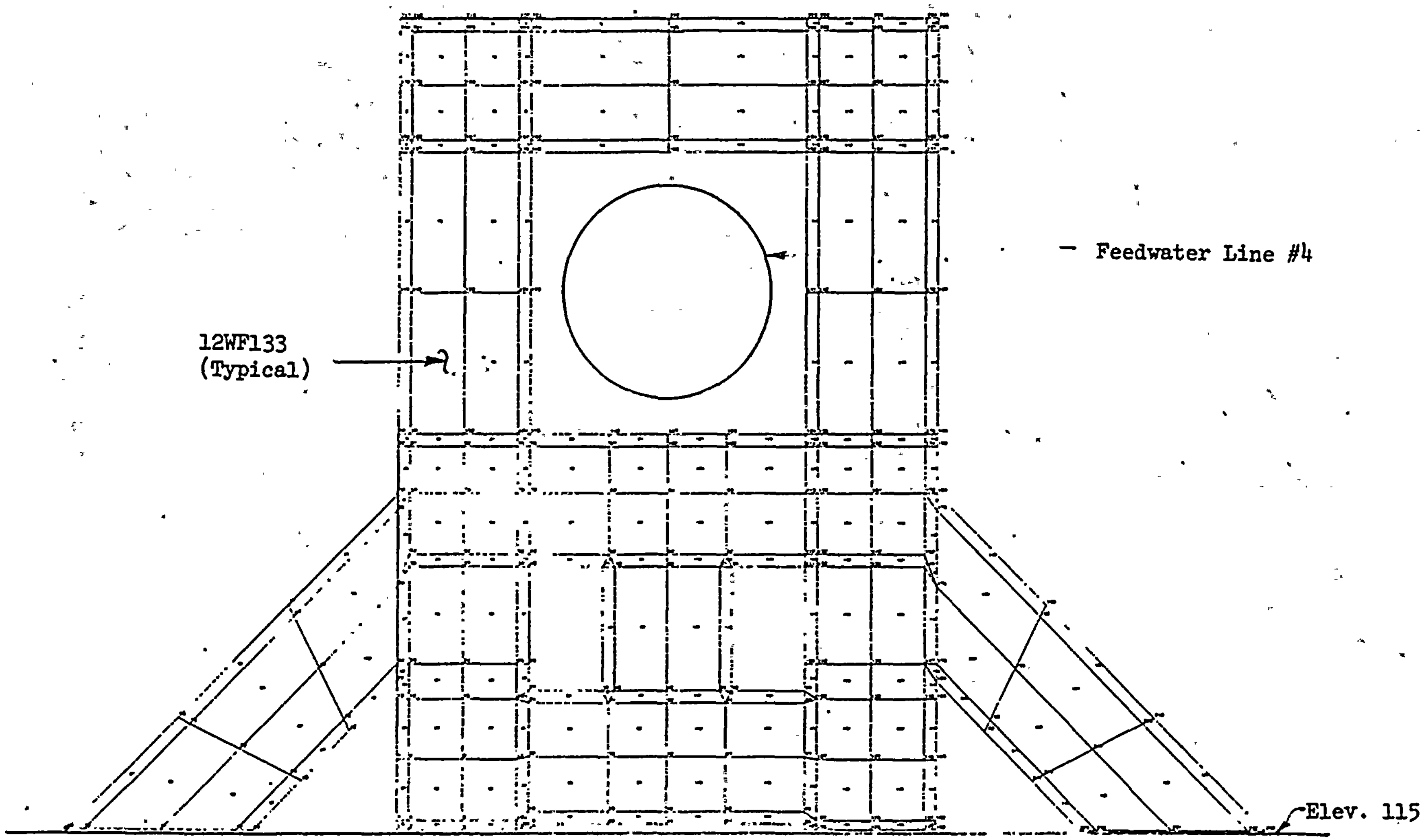
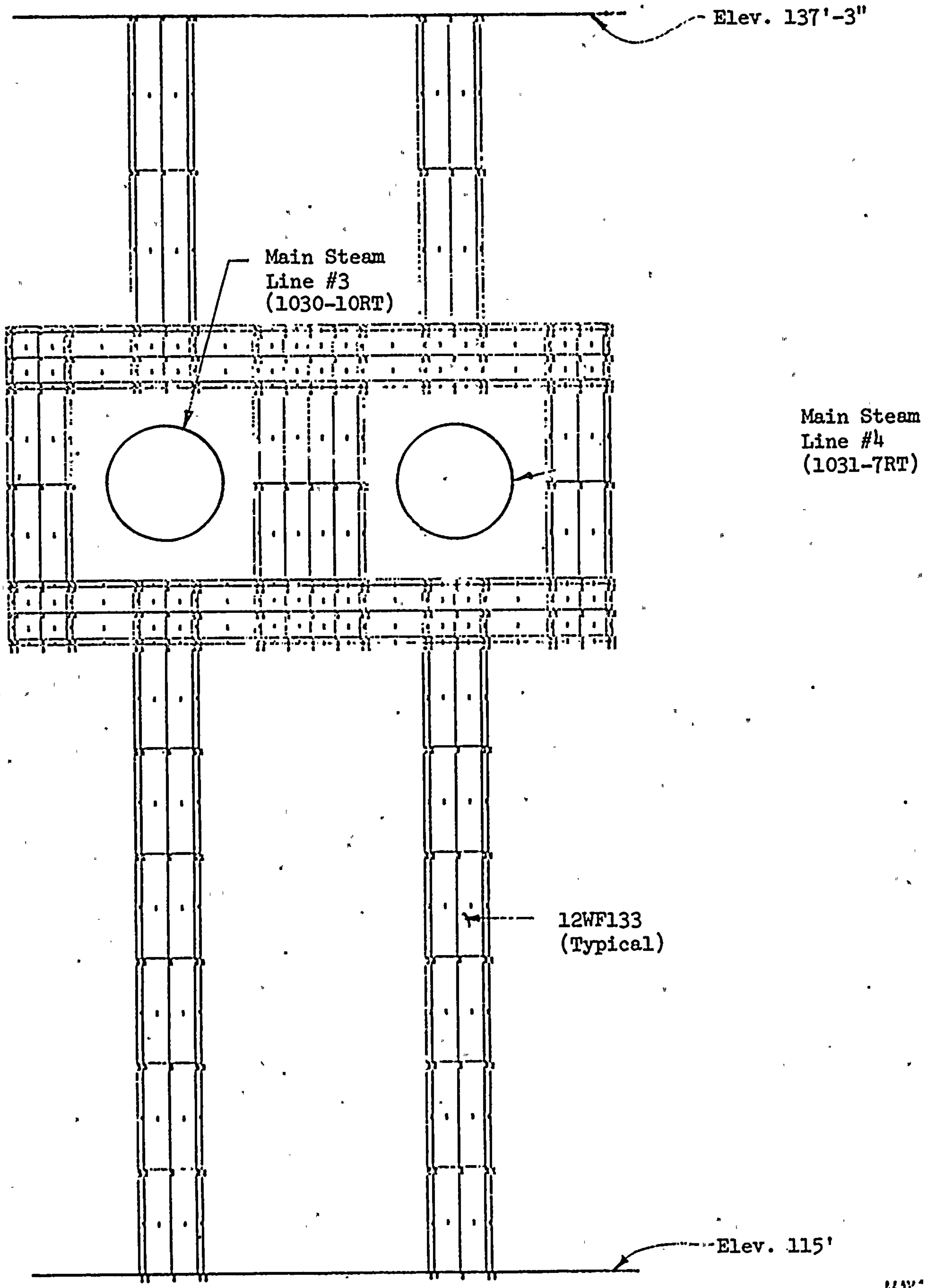


FIGURE D-3
FINITE ELEMENT MODEL
MAIN STEAM RESTRAINTS 1030-10RT AND 1031-7RT



MAY 10 1974

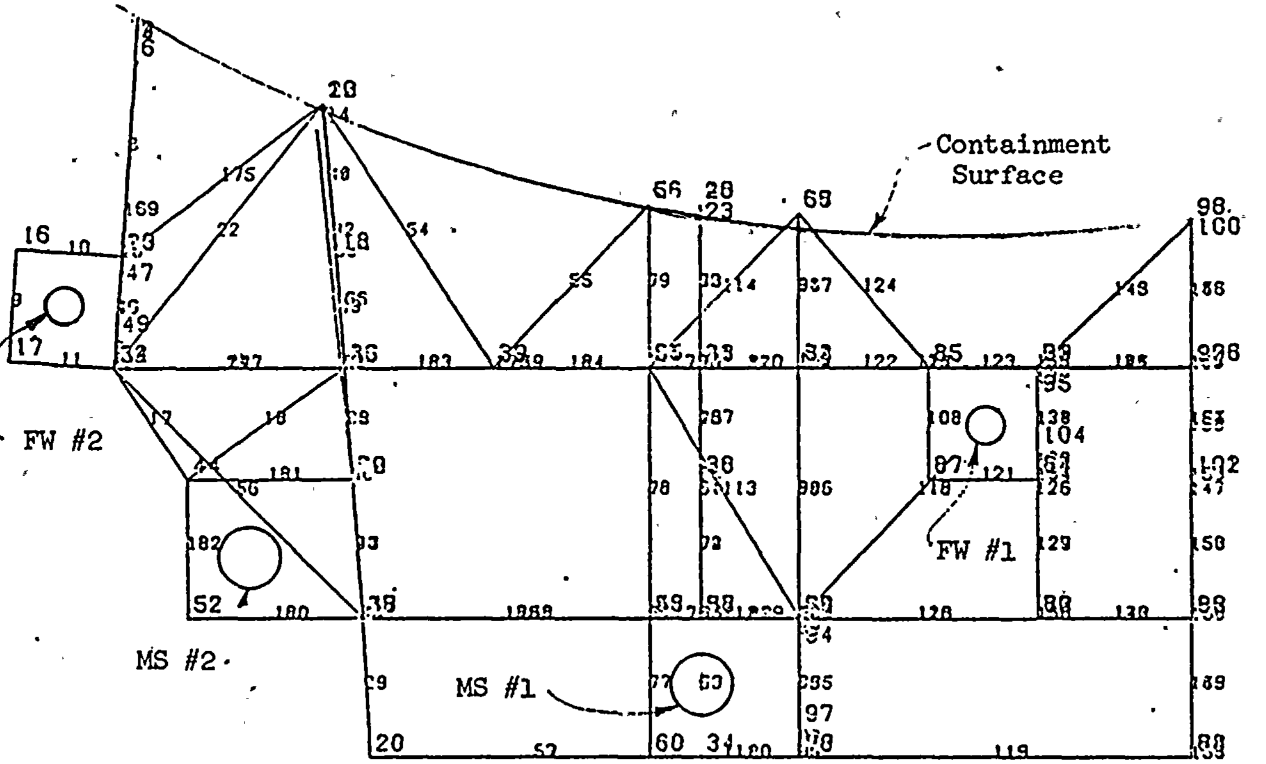
Nuclear Services Corporation

FIGURE D-4

FINITE ELEMENT MODEL

PIPEWAY SPACE FRAME

NORTH SIDE OF CONTAINMENT



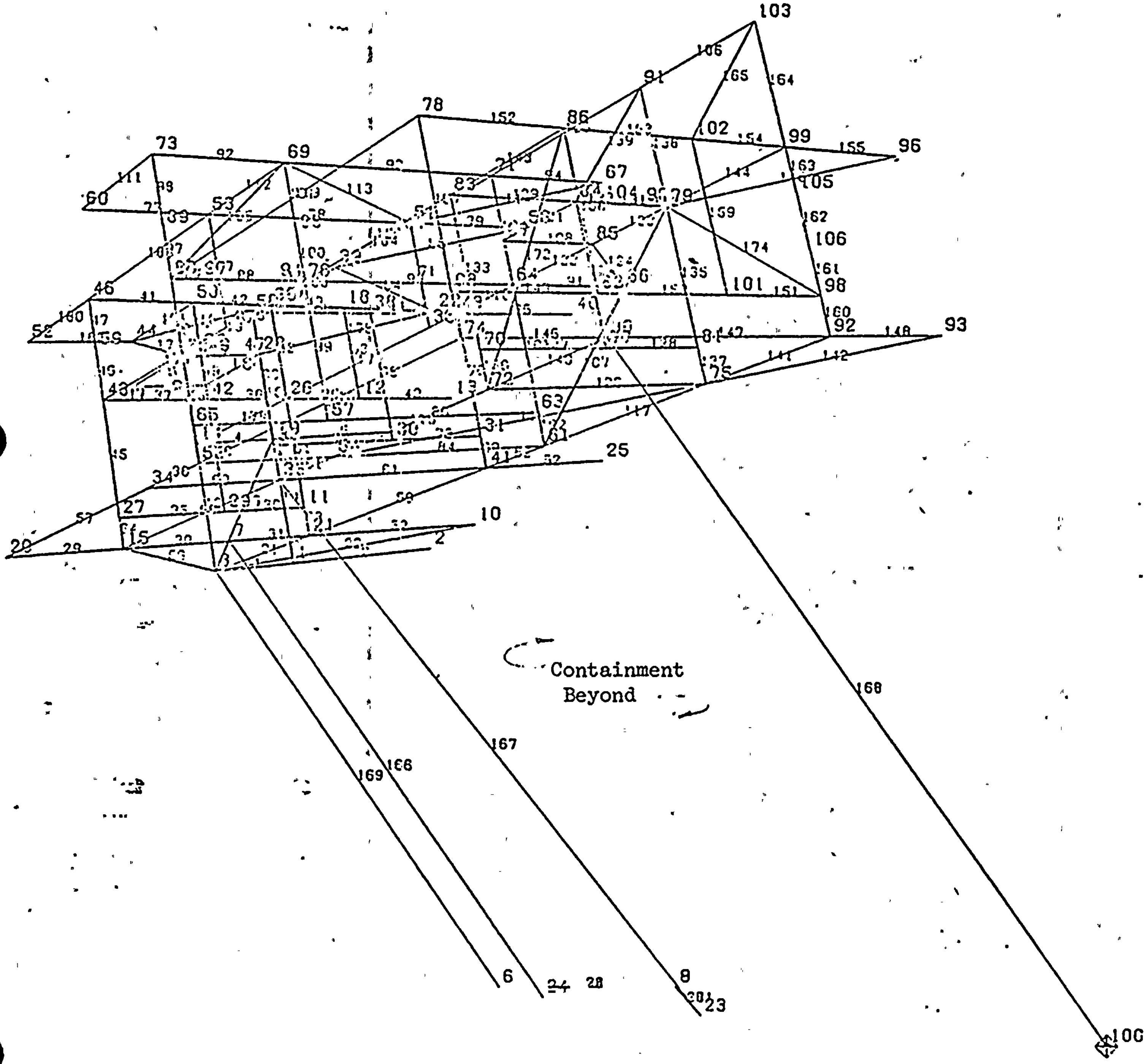
PLAN VIEW BELOW EL. 119'-1"

MAY 10 1974

Nuclear Services Corporation

FIGURE D-5

FINITE ELEMENT MODEL
PIPEWAY SPACE FRAME
NORTH SIDE OF CONTAINMENT



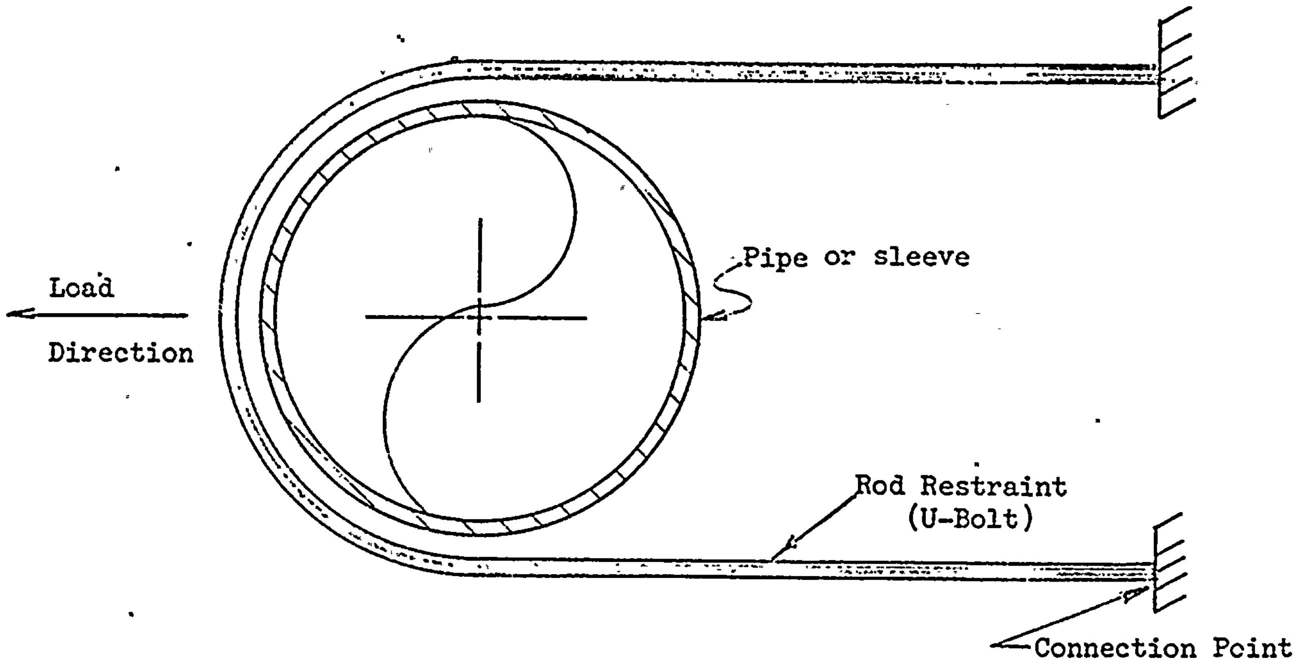
OBLIQUE VIEW

MAY 10 1974

Nuclear Services Corporation

FIGURE D-6

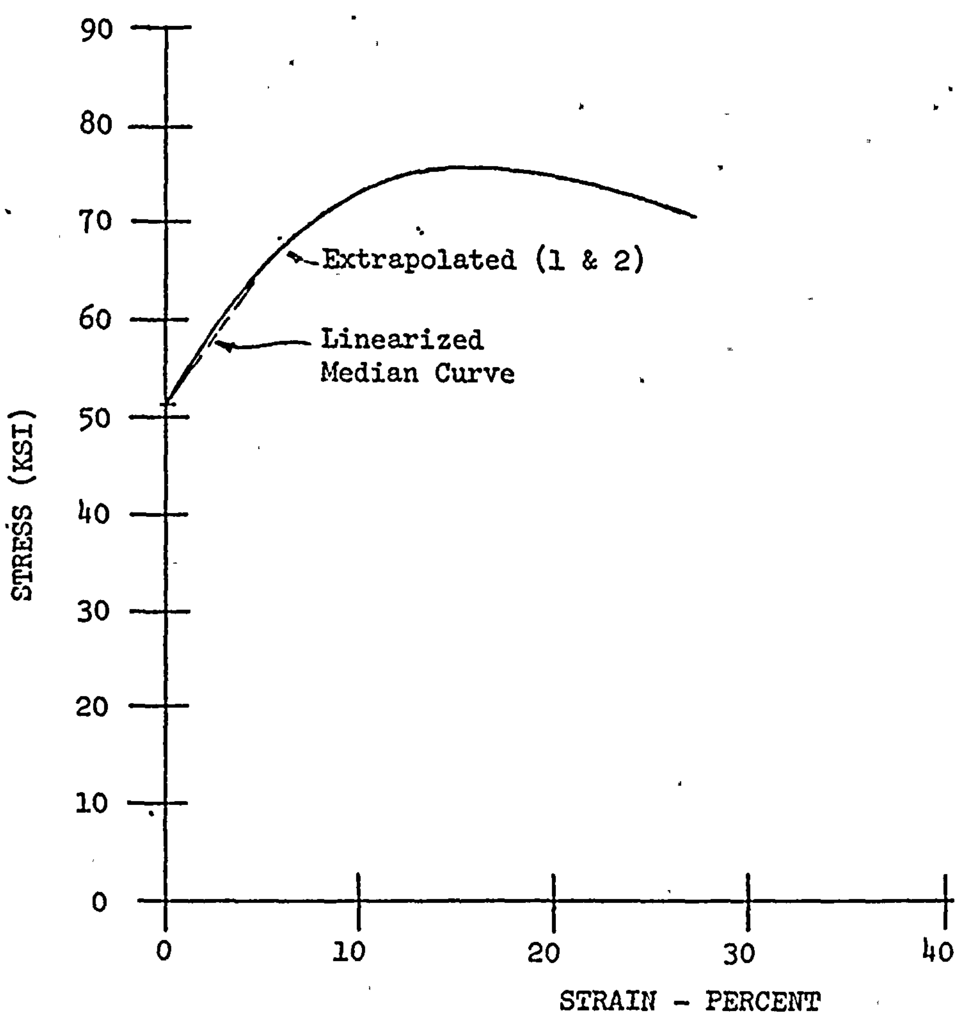
BASIC ROD RESTRAINT DESIGN



Nuclear Services Corporation

FIGURE D-7

MEDIAN STRESS - STRAIN CURVE
FOR A 516 GRADE 70 STEEL
AREA GW RESTRAINTS



Elastic Modulus
E = 29 x 10^6 psi

Median
σ_y = 51.8 ksi
σ_u = 75.5 ksi
ϵ_u = 27.0 %
σ_{ma} = 65.0 ksi
ϵ_{ma} = 5.0 %

Lower Bound (1)
σ_y = 47.3 ksi
σ_u = 70.0 ksi
ϵ_u = 34.0 %

Upper Bound (1)
σ_y = 56.4 ksi
σ_u = 80.9 ksi
ϵ_u = 20.0 %

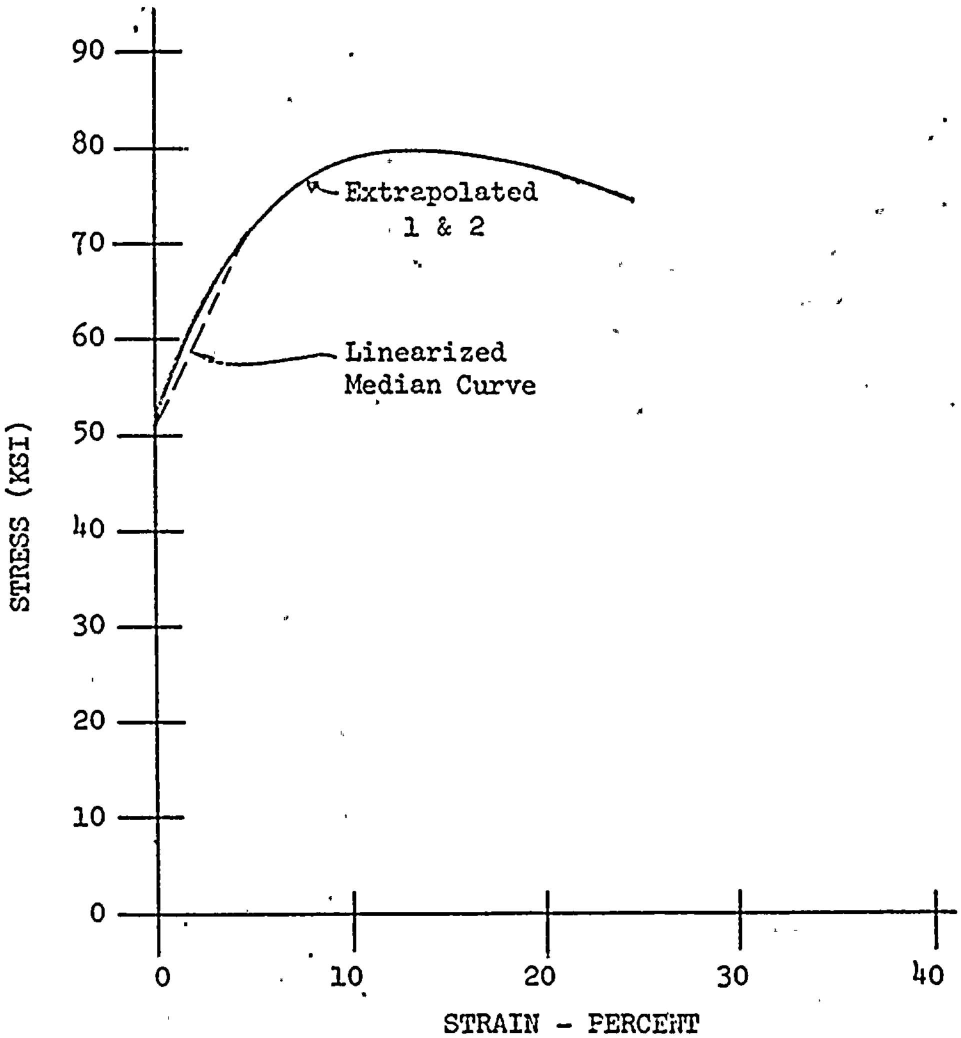
1) Reference 27
2) Reference 38

MAY 10 1974

Nuclear Services Corporation

FIGURE D-8

MEDIAN STRESS - STRAIN FOR
A 441 STEEL PIPEWAY RESTRAINTS



Elastic Modulus
$E = 29 \times 10^6$ psi

Median
$\sigma_y = 53.8$ ksi
$\sigma_u = 79.3$ ksi
$\varepsilon_u = 24.55$ %
$\sigma_{ma} = 71.0$ ksi
$\varepsilon_{ma} = 4.25$ %

Lower Bound (1)
$\sigma_y = 43.7$ ksi
$\sigma_u = 70.8$ ksi
$\varepsilon_u = 32.1$ %

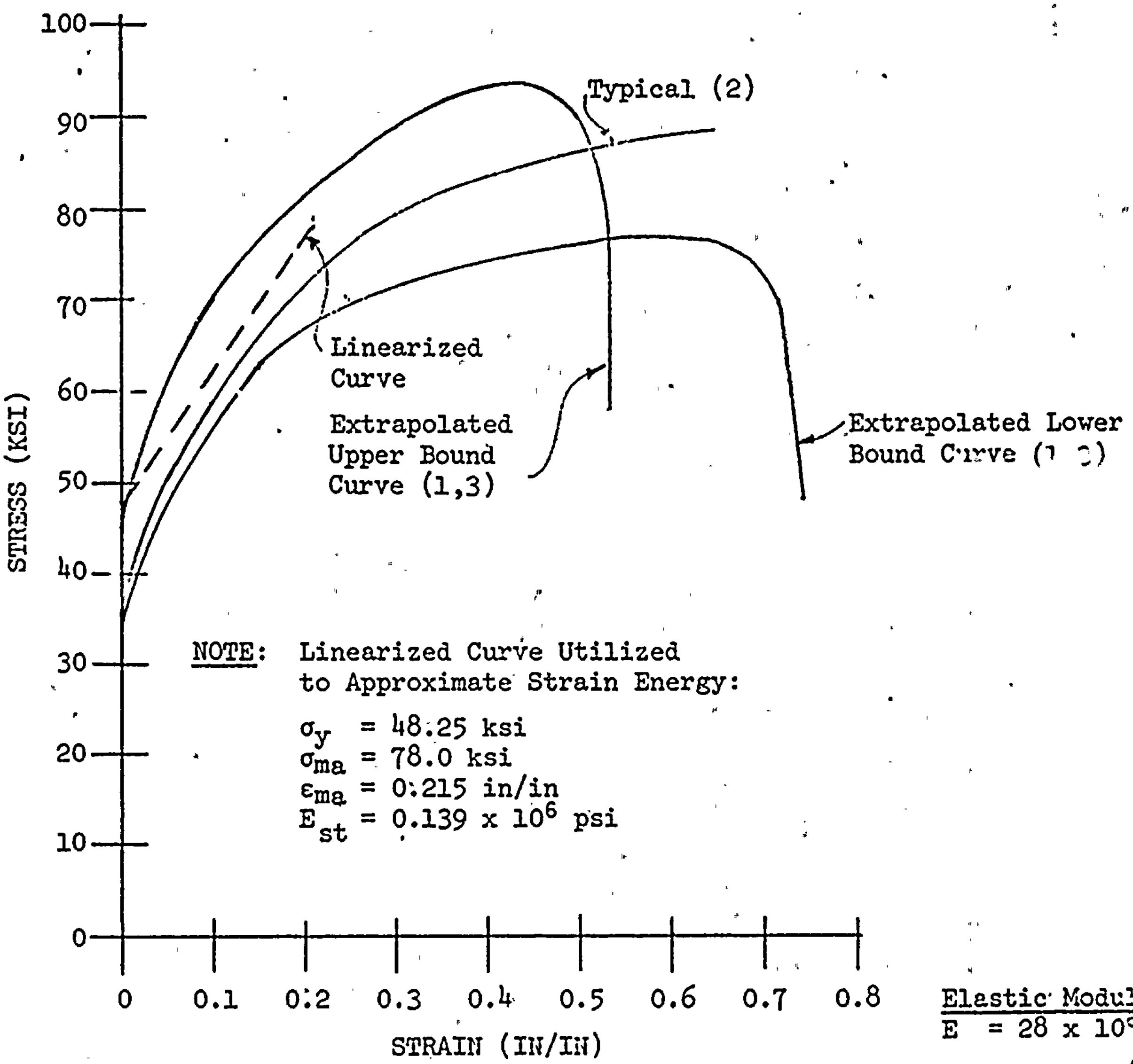
Upper Bound (1)
$\sigma_y = 64.0$ ksi
$\sigma_u = 87.8$ ksi
$\varepsilon_u = 17.0$ %

1) Reference 27
2) Reference 38

MAY 10 1974

FIGURE D-9

ROOM TEMPERATURE STRESS - STRAIN CURVES
FOR ANNEALED A 304 STAINLESS STEEL ROD



Elastic Modulus
$E = 28 \times 10^6$ psi

Lower Bound (3)
$\sigma_y = 34$ ksi
$\sigma_u = 76$ ksi
$\varepsilon_u = .74$ in/in

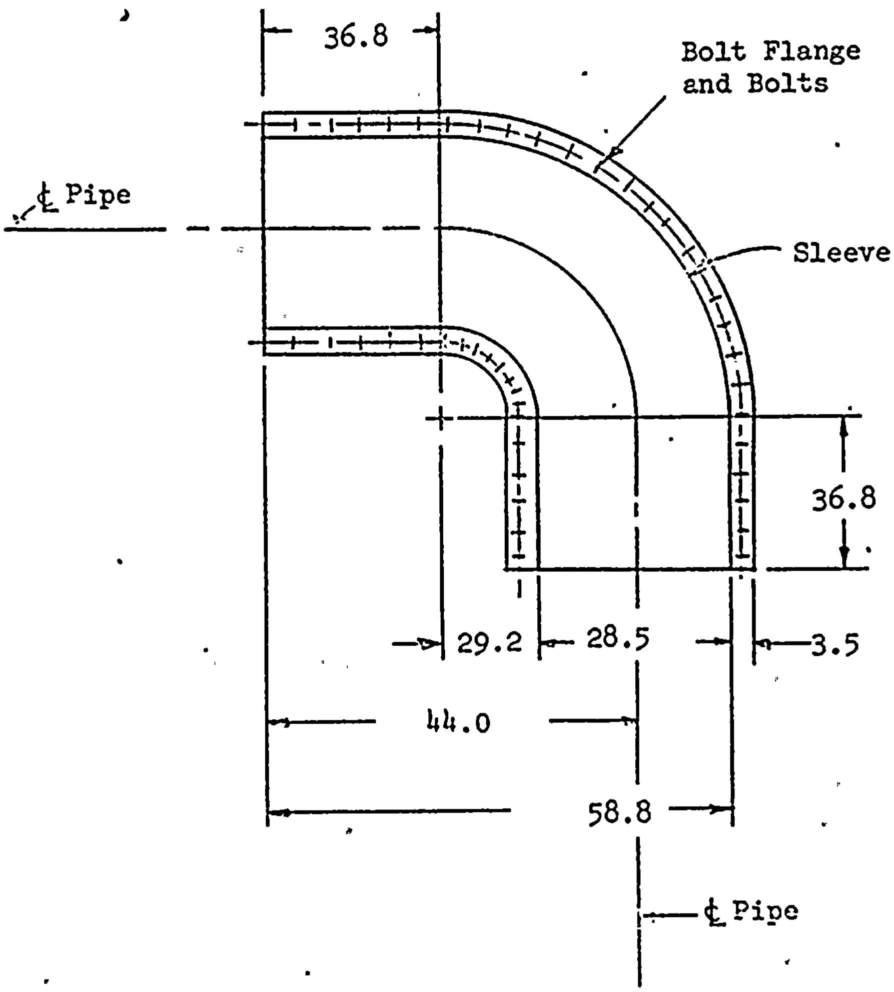
Upper Bound (3)
$\sigma_y = 46$ ksi
$\sigma_u = 94$ ksi
$\varepsilon_u = .52$ in/in

1) Reference 39
2) Reference 40
3) Reference 41

MAY 10 1974

FIGURE D-10

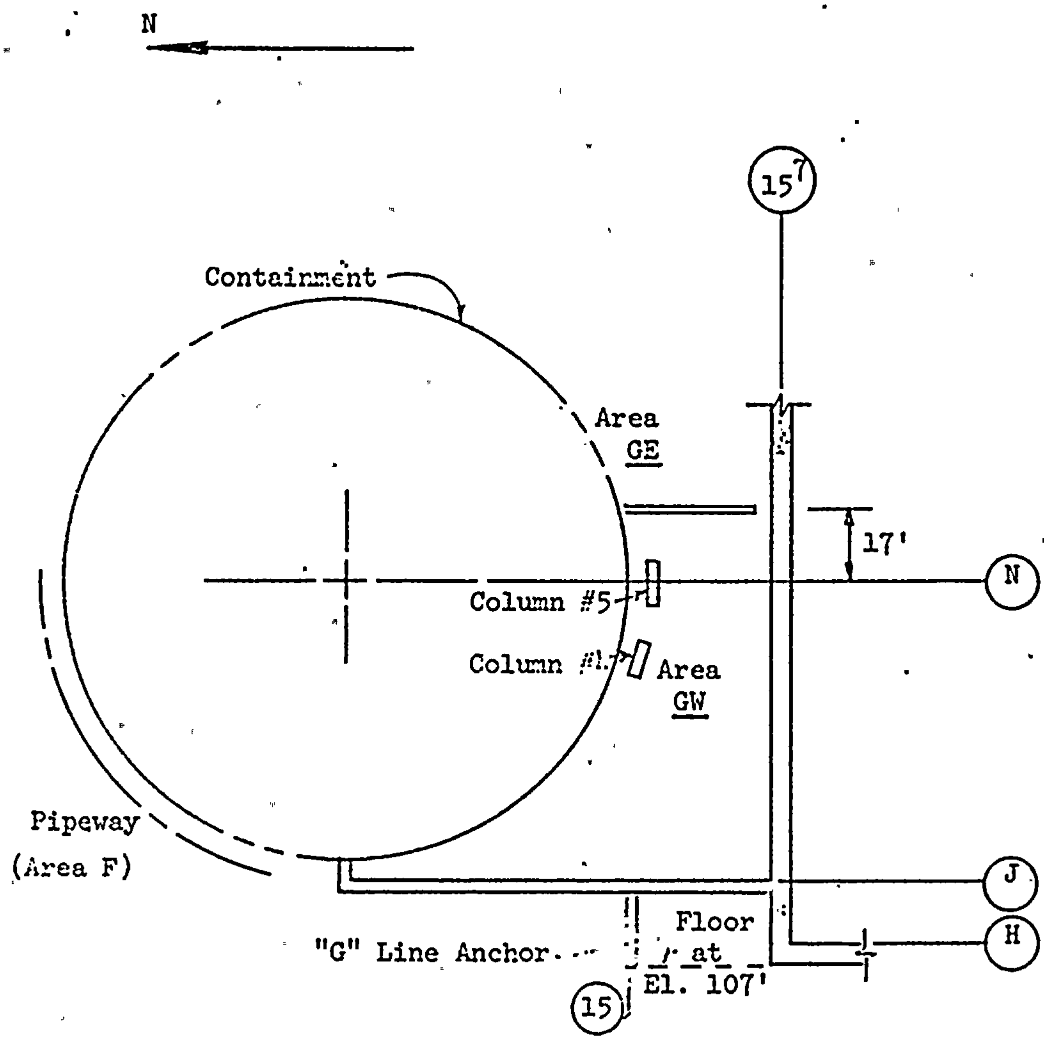
TYPICAL SLEEVE DIMENSIONS



MAY 10 1974

FIGURE D-11

LOCATION PLAN

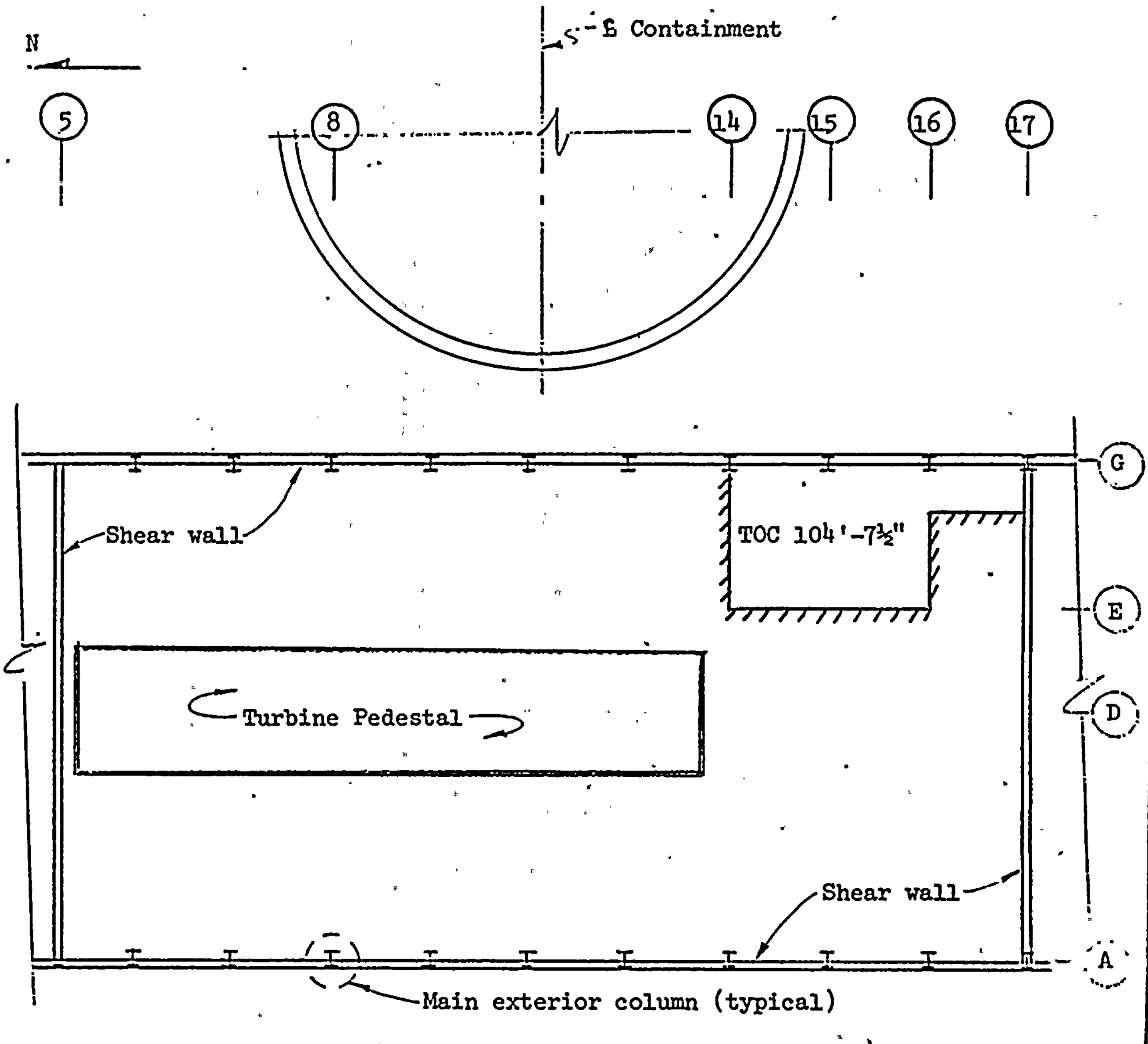


PLAN @ ELEVATION 115'-0"

Nuclear Services Corporation

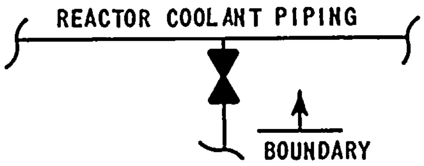
FIGURE D-12

TURBINE BUILDING - LOCATION PLAN



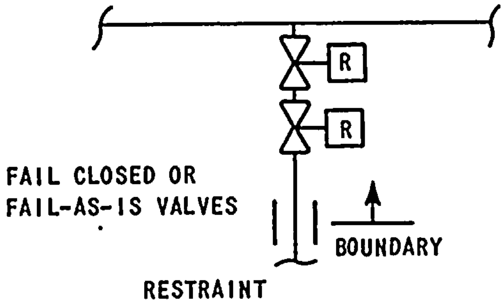
MAY 10 1974

CASE I — OUTGOING LINES WITH NORMALLY CLOSED VALVE



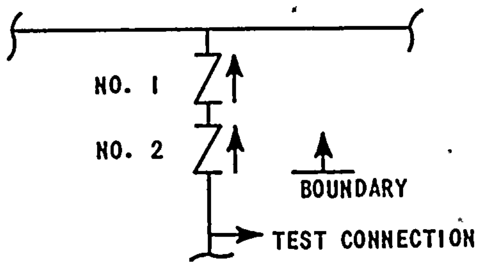
NOTE: PRESSURIZER SAFETY VALVES ARE INCLUDED UNDER THIS CASE.

CASE II — OUTGOING LINES WITH NORMALLY OPEN VALVES

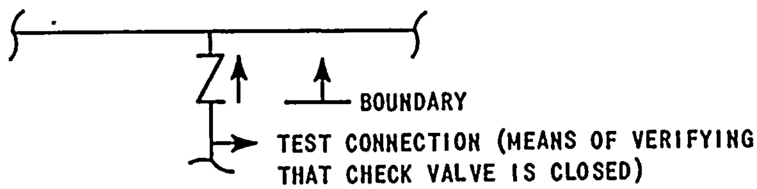


NOTE: THE REACTOR COOLANT PUMP NO. I SEAL IS ASSUMED TO BE EQUIVALENT TO FIRST VALVE

CASE III — INCOMING LINES NORMALLY WITH FLOW



CASE IV — INCOMING LINES NORMALLY WITHOUT FLOW

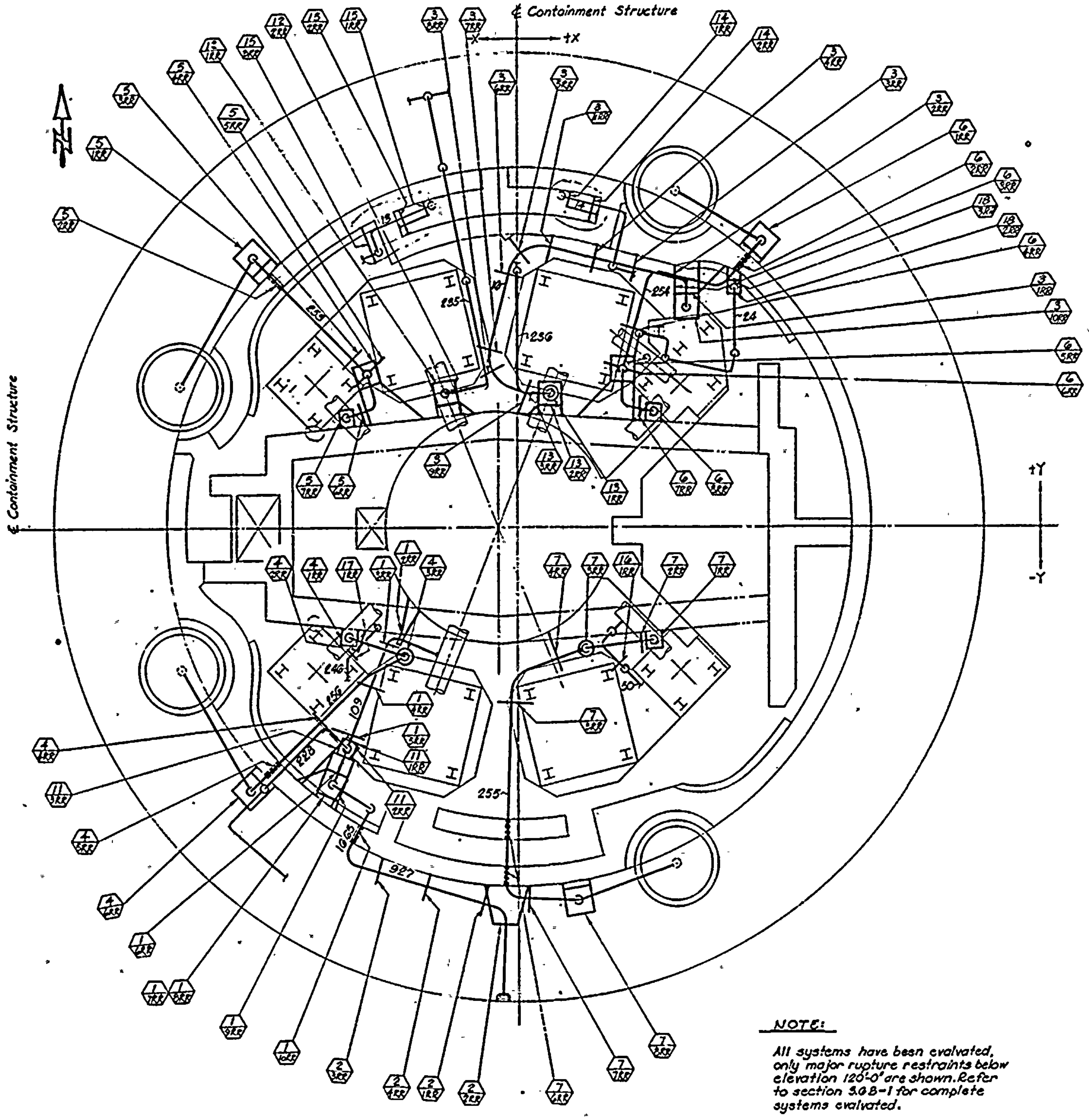


CASE V — ALL INSTRUMENTATION TUBING AND INSTRUMENTS CONNECTED DIRECTLY TO THE REACTOR COOLANT SYSTEM IS CONSIDERED AS A BOUNDARY. HOWEVER, A BREAK WITHIN THIS BOUNDARY RESULTS IN A RELATIVELY SMALL FLOW WHICH CAN NORMALLY BE MADE UP WITH THE CHARGING SYSTEM.

UNITS I AND 2
DIABLO CANYON SITE

FIGURE 3.6-1

LOSS OF COOLANT ACCIDENT BOUNDARY LIMITS



NOTE:

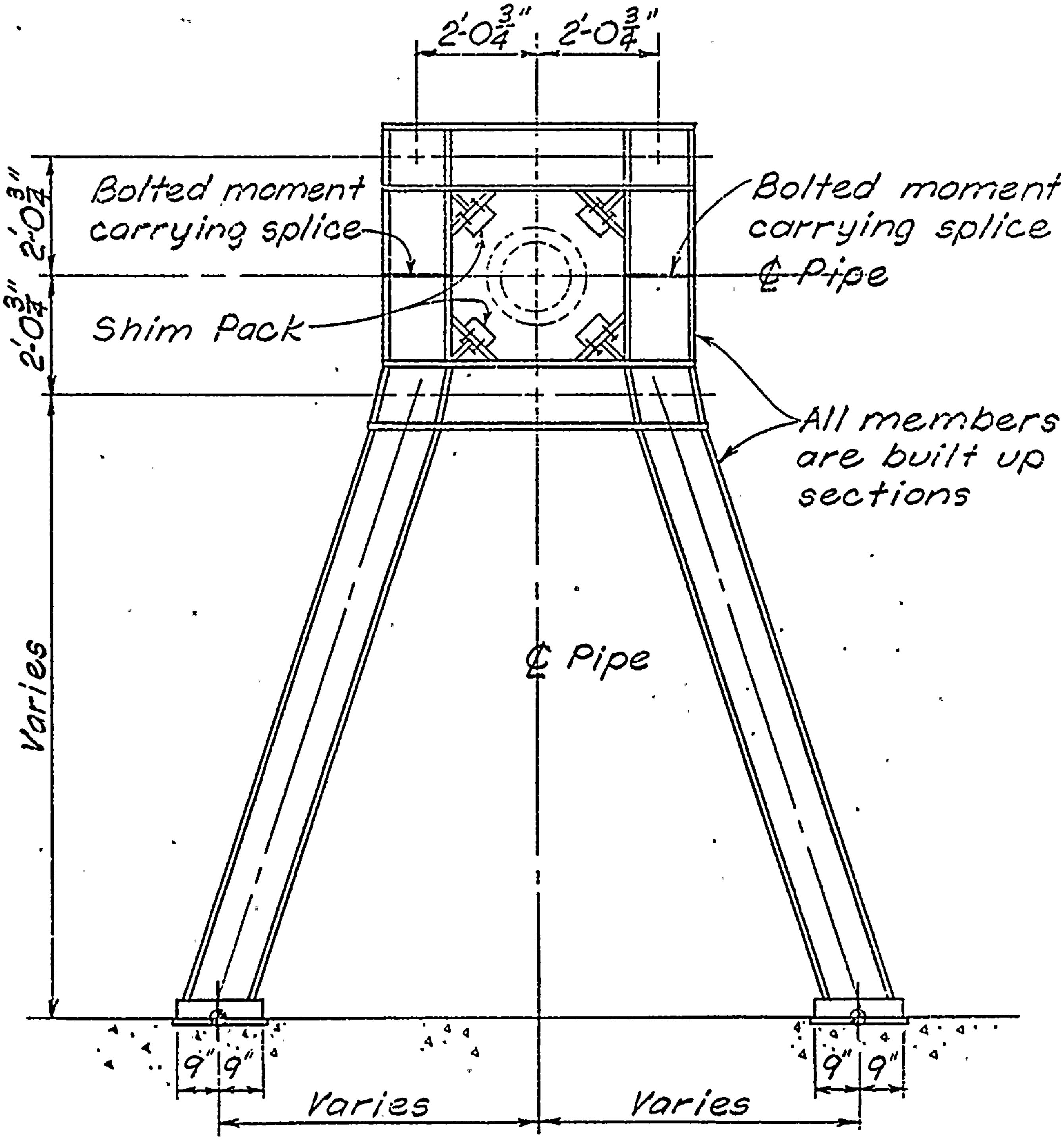
All systems have been evaluated, only major rupture restraints below elevation 120'-0" are shown. Refer to section 3.6B-1 for complete systems evaluated.

Line Description	RESTRAINT No.	Pipe No.	Pipe Size
Residual Heat Removal Supply	1/1RR		
	1/2RR	109	14φ
	1/3RR	•	•
	1/4RR	•	•
	1/5RR	•	•
	1/6RR	•	•
	1/7RR	•	•
	1/8RR	•	•
	1/9RR	•	•
Loop 4 Hot Leg after Valve 8702	1/10RR	1665	•
Loop 4 Hot Leg to Res. Ht. Rem. PP.	2/1RR	927	14φ
	2/2RR	•	•
	2/3RR	•	•
	2/4RR	•	•
Pressurizer Surge Line	3/1RR	16	14φ
	3/2RR	•	•
	3/3RR	•	•
	3/4RR	•	•
	3/5RR	•	•
	3/6RR	•	•
	3/7RR	•	•
	3/8RR	•	•
	3/9RR	•	•
	3/10RR	•	•
	3/11RR	•	•
Accumulator Injection Loop 4	4/1RR	256	10φ
	4/2RR	•	•
	4/3RR	•	•
	4/4RR	•	•
	4/5RR	•	•
	4/6RR	•	•
Accumulator Injection Loop 1	5/1RR	253	10φ
	5/2RR	•	•
	5/3RR	•	•
	5/4RR	•	•
	5/5RR	•	•
	5/6RR	•	•
	5/7RR	•	•
Accumulator Injection Loop 2	6/1RR	254	10φ
	6/2RR	•	•
	6/3RR	•	•
	6/4RR	•	•
	6/5RR	•	•
	6/6RR	•	•
	6/7RR	•	•
	6/8RR	•	•
Accumulator Injection Loop 3	7/1RR	255	10φ
	7/2RR	•	•
	7/3RR	•	•
	7/4RR	•	•
	7/5RR	•	•
	7/6RR	•	•
	7/7RR	•	•
	7/8RR	•	•
Safety Injection Loop 4	11/1RR	238	6φ
	11/2RR	•	•
	11/3RR	•	•
Safety Injection Loop 1	12/1RR	235	6φ
	12/2RR	•	•
Safety Injection Loop 2	13/1RR	236	6φ
	13/2RR	•	•
	13/3RR	•	•
Loop 2 Pressurizer Spray Line	14/1RR	14	4φ
	14/2RR	•	•
Loop 1 Pressurizer Spray Line	15/1RR	13	4φ
	15/2RR	•	•
	15/3RR	•	•
Charging Line Loop 3	16/1RR	50	3φ
Charging Line Loop 4	17/1RR	246	3φ
Let Down Line Loop 2	18/2RR	24	3φ
	18/3RR	•	•

70 restraints

UNITS 1 AND 2
DIABLO CANYON SITE

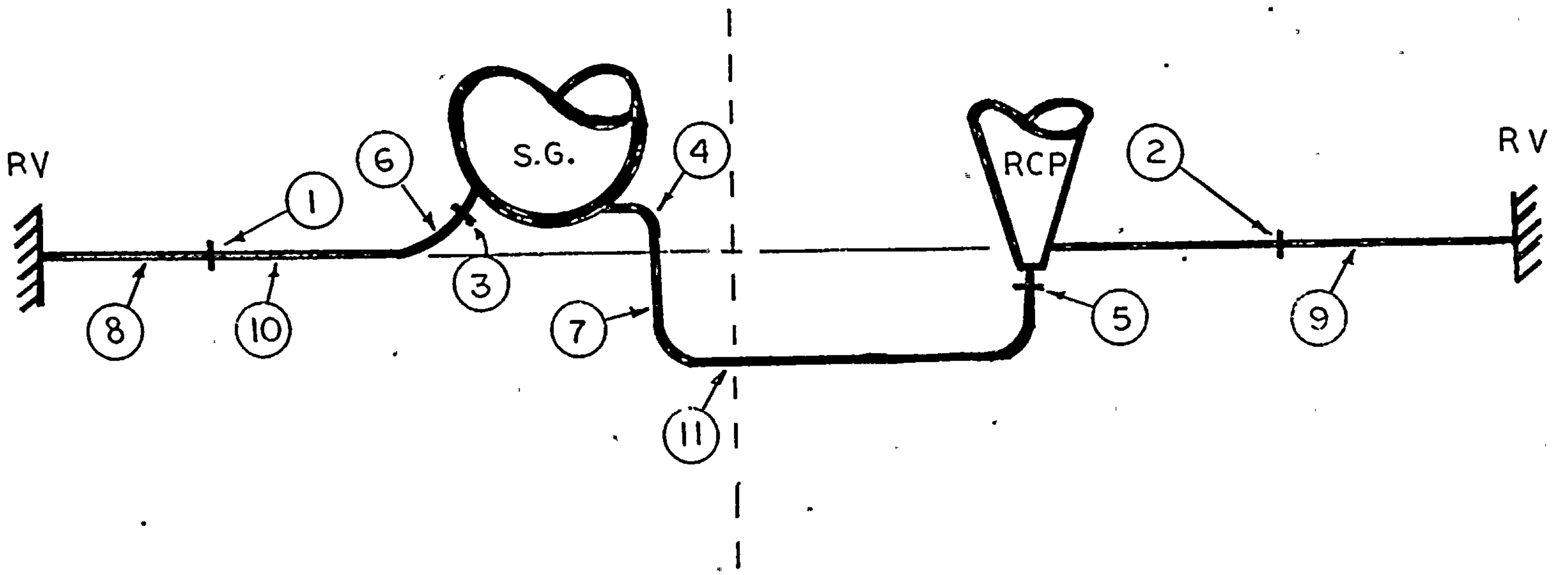
FIGURE 3.6-2
CONTAINMENT STRUCTURE
PIPE RUPTURE RESTRAINTS



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.6-3

TYPICAL PIPE RUPTURE RESTRAINT



RV = REACTOR VESSEL
S G = STEAM GENERATOR
RCP = REACTOR COOLANT PUMP

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.6-4

PRIMARY COOLANT LOOP BREAKS

3.7 SEISMIC DESIGN

3.7.1 INPUT CRITERIA

Design Response Spectra

Section 2.5.2 provides a discussion of the earthquakes postulated for the Diablo Canyon plant site, and the effects of these earthquakes in terms of maximum freefield ground motion accelerations and corresponding response spectra at the plant site. The maximum vibratory accelerations at the plant site would result from either Earthquake D-modified or Earthquake B, depending upon the natural period of the vibrating body. Response acceleration spectra curves for horizontal free field ground motions at the plant site from Earthquake D-modified and Earthquake B are presented in Figures 2.5-17 and 2.5-18, respectively.

For design purposes, the response spectra for each damping value from Earthquake D-modified and Earthquake B are combined to produce an envelope spectrum. The acceleration value for any period on the envelope spectrum is equal to the larger of the two values from the Earthquake D-modified spectrum and the Earthquake B spectrum. Vertical free field ground accelerations are assumed to be two-thirds of the corresponding maximum horizontal accelerations, and the vertical free field ground motion response spectra are assumed to be two-thirds of the corresponding horizontal spectra. The hypothetical earthquake which would produce these horizontal and vertical vibratory accelerations is termed the Design Earthquake (DE).

In order to assure adequate reserve energy capacity, the design of Design Class I structures and equipment is reviewed for accelerations twice those of the DE. The hypothetical earthquake which would produce these double vibratory accelerations is termed the Double Design Earthquake (DDE).

Design Acceleration - Time History

The free field ground motion acceleration time histories used in the dynamic analyses of the Containment Structure and Auxiliary Building are developed by the following procedure: The response spectra for 2% damping for Earthquake D-modified and Earthquake B are enveloped to produce a single response spectrum (DE intensity). A time history is then developed that produces a spectrum with no significant deviation from the smooth DE envelope spectrum. This procedure eliminates undesirable peaks and valleys that exist in the response spectrum calculated directly from the earthquake D-modified and earthquake B records. A similar procedure is used to obtain a free field ground motion acceleration time history for the DDE. The free field ground motion acceleration-time histories for the DE and DDE are shown in Figures 3.7-1 and 3.7-2 respectively. Comparison of the response spectrum computed from time history with the smoothed envelope spectrum is shown in Figure 3.7-3 (2% damping) and in Figure 3.7-4 (5% damping) for the DE and DDE respectively. These spectra are calculated at period intervals of 0.01 seconds, which adequately define the spectra.

Damping Values

The specific percentages of critical damping used for Design Class I structures, systems, and components are as follows:

Type of Structure	% of Critical Damping DE	DDE
Containment structures and all internal concrete structures	2.0	5.0
Other conventionally reinforced concrete structures above ground, such as shear walls or rigid frames	5.0	5.0
Welded structural steel assemblies	1.0	1.0
Bolted or riveted steel assemblies	2.0	2.0
Vital piping systems	0.5	0.5
Foundation rocking*	5.0	5.0

* Five percent of critical damping is used for structures founded on rock for the purpose of computing the response in the rocking mode, and seven percent of critical damping is used for the purpose of computing the response in the translation mode.

Bases for Site Dependent Analysis

Site conditions are used to develop the shape of site seismic design response spectra as described in Section 2.5.2.

Soil-Supported Structures

All Design Class I plant structures are founded on rock with the exception of the following, which are located on compacted engineered fill:

Condensate Water Storage Tank (one for each unit)

Primary Water Storage Tank (one for each unit)

Firewater and Transfer Tank (one for both units)

The depth of fill varies from zero to twenty feet. To account for effects of the fill, the free field ground motion response spectra were increased by 50% in accordance with Reference 1.

Soil-Structure Interaction

Soil-structure interaction effects are considered as described in Section 3.7.2 under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

3.7.2 SEISMIC SYSTEM ANALYSIS

In accordance with the proposed STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS
3 REPORTS FOR NUCLEAR POWER PLANTS, paragraphs under the first two headings below, "Seismic Analysis Methods" and "Description of Seismic Analyses," apply to all seismic analysis performed, i.e. both seismic system analysis and seismic subsystem analysis. Paragraphs under subsequent headings in this section (3.7.2) provide discussion of specific topics applicable to seismic system analysis. Discussion of specific topics applicable to seismic subsystem analysis is provided in Section 3.7.3.

The seismic analysis of Design Class I structures, systems, and components is based on input free field ground motions of the DE and DDE described in Section 3.7.1.

Seismic Analysis Methods

Four dynamic methods of seismic analysis are used for Design Class I structures, systems, and components: time history modal superposition; response spectrum modal superposition; response spectrum, single-degree-of-freedom, and the method for rigid equipment and piping. The concept of modal analysis and each of the
3 four methods of seismic analysis are discussed in subsequent paragraphs.

Modal Analysis

The structure, system, or component is represented as a mathematical model which is in the form of lumped masses interconnected by springs or finite elements. The mathematical model typically has one, two, or three degrees of freedom for each lumped mass or node point, but could have as many as six degrees of freedom for each lumped mass or node point.

Each multiple-degree-of-freedom (multidegree) system has the same number of normal modes as it has degrees of freedom. The characteristic of a normal mode of vibration is that, under certain conditions, the multidegree system could vibrate freely in that mode alone, and during such vibration, the ratio of displacements of any two masses is constant with time. These ratios define the characteristic shape of the mode. For any vibration of the multidegree system, the motion in any of the individual normal modes can be treated as an independent single-degree-of-freedom system, and the complete motion of the multidegree system can be obtained by superimposing the independent motions of the individual modes. The normal mode method is limited to linear, elastic systems.

The natural frequencies and characteristic shapes are determined by solution of the equations of motion for free vibrations.

Time History Modal Superposition

The time history of response in each mode is determined from the acceleration time history input by integration of the equations of motion. The modal responses are combined by algebraic sum to produce an accurate summation at each step.

Response Spectrum Modal Superposition

The response spectrum is a plot, for all periods of vibration, of the maximum acceleration experienced by a single-degree-of-freedom vibrating body during a particular earthquake. The response spectrum modal superposition method of analysis applies to multidegree systems and is based on the concept of modal analysis. The modal equation of motion for a multidegree system is analogous to the equation of motion for a single degree of freedom. The maximum response in each mode is calculated, and modal responses (displacements, accelerations, shears, moments, etc.) are combined by the square root of the sum of the squares method. For closely spaced modes, the combined total response is obtained by taking the absolute sum of the closely spaced modes and then taking the square-root of the sum of the squares with all other modes. (See Subsection 3.7.3 under the heading "Procedures for Combining Modal Responses for an exception to the procedure for closely spaced modes). 3

Response Spectrum, Single-Degree-of-Freedom

Many components can be accurately represented by a single-degree-of-freedom mathematical model. The response spectrum method of analysis is applicable, and the concept of modal analysis is not required.

Rigid Equipment or Piping

When a mechanical component or piping run is rigid ($T \leq .05$ seconds), it is designed for the maximum acceleration experienced by the supporting structure (or ground if the component is located on the ground) at that location.

Application

All Design Class I structures, systems, components, and piping are designed by time history modal superposition, response spectrum modal superposition, response spectrum single-degree-of-freedom, or the method for rigid equipment and piping, except the following:

1. Containment spray pumps and motors which are designed by the pseudo-dynamic method described in this Section (3.7.2) under the heading "Description of Seismic Analyses" subheading "Design Class I Mechanical Equipment."

2. Certain Design Class I piping less than 6 inches in diameter that is restrained according to criteria described in this Section (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Piping."

3. Mechanical equipment whose seismic adequacy is verified by testing as described in Section 3.9.

4. Electrical and instrumentation equipment whose seismic adequacy is verified by testing as described in Section 3.10.

Description of Seismic Analyses

Design Class I Structures

Dynamic analyses by the time history modal superposition method were performed for the Containment Structure and the Auxiliary Building by John A. Blume & Associates, Engineers. Acceleration time histories were obtained at specific points in the structures, and from these, response spectra were calculated. In order to provide for possible variations in the parameters used in the dynamic analyses, such as mass values, material properties, and material sections, the calculated spectra were modified. It is estimated that the calculated periods of the structure could vary by approximately 10%, and to account for this, the peaks of the spectra were correspondingly widened. The modified spectra, known as "smooth spectra," are used in the design of Design Class I equipment and piping located in the Containment Structure and Auxiliary Building.

Allowable stresses for Design Class I structures are presented in Section 3.8.

Containment Structure

Model

The Containment Structure calculations were performed with a computer program for analysis of axisymmetric structures by the finite element method. The foundation rock mass and the Containment Structure are modeled as one structure system to consider the effect of rock-structure interaction, as shown in Figure 3.7-5. The boundary dimensions of the model are selected such that they do not have a significant effect on the response of the structure. The exterior shell and internal structure are modeled using shell elements with four degrees of freedom at each nodal point. There is a total of 156 nodal points and 140 elements in the model. The weight of mechanical equipment in the structure is included in the calculation of equivalent mass density for the structure elements. Values of elastic constants for the rock mass and their variation with depth are based on field measurements made at the plant site (see Section 2.5).

Input Boundary Motions

In the seismic analysis of the finite element model, the motions at the boundary of the rock mass are required as input. These boundary motions are derived using procedures described in the following steps:

1. The finite element model of the rock mass only (without the structure) is subjected to a unit impulse acceleration acting at the rock mass boundaries. As a result, the acceleration time-history (impulse response which reflects the rock mass properties) is obtained at the center nodal point on the surface of the rock mass.

2. The impulse response function, together with the desired free field ground motion are used as input to a deconvolution program. The required boundary motion is obtained as the output. This boundary motion, when used as input to the rock mass model, produces a time history at the center nodal point on the surface of the model that is equivalent to the free field motion. To check the accuracy of the boundary motion, the rock mass without the structure is analyzed using this motion as input, and the computed free field ground motion at the center nodal point on the surface of the rock mass is obtained. The computed free field spectrum is calculated for this surface motion and compared with the DE (or DDE) smoothed spectrum. Due to approximations involved in the analytical methods used to derive the boundary motions, the computer spectra show slight deviations from the desired smoothed spectra. To account for these deviations, the structural response results are then conservatively scaled upward by appropriate correction factors.

The boundary motions derived from the procedure described above are used to complete the analysis of the Containment Structure.

Natural Periods and Mode Shapes

The natural periods for all significant modes of the Containment Structure are listed in Table 3.7-1. The corresponding mode shapes are shown in Figure 3.7-6.

Response Values

The analysis provides acceleration time histories, maximum absolute accelerations, displacements, shell forces and moments, total shears, and total overturning moments. These maximum response values are listed in Tables 3.7-2 through 3.7-7 for the nodal points indicated in Figure 3.7-5.

Acceleration response spectra are calculated from the acceleration time histories, and corresponding smooth spectra are prepared. Typical smooth spectra are shown in Figures 3.7-8 through 3.7-12.

Auxiliary Building

Model

The dynamic time history analysis of the Auxiliary Building is performed with a computer program for analysis of a spring and lumped mass model. The model, shown in Figure 3.7-13, consists of six lumped masses with three degrees of freedom at each mass point; one translational degree of freedom in each horizontal direction and one rotational degree of freedom. The weights of the mass points include an appropriate live load on each floor to account for the effect of small pieces of equipment, concrete pads for equipment, tanks, pumps, and incidental weight not considered otherwise. Weights of cranes, storage tanks, and other large pieces of equipment are added to the appropriate mass point weights.

Locations of the centers of masses and rigidities are calculated to consider torsional modes of vibration. Mass moments of inertia and torsional rigidities are calculated by conventional structural analysis methods.

To simulate the foundation soil interaction with the structure, foundation springs are introduced as shown in Figure 3.7-13. The stiffnesses of these foundation springs are derived by using the equations developed for the case of a rigid plate on a semi-infinite elastic half-space.[2,3,4] The Auxiliary Building is a broad-based and comparatively low-rise structure, and therefore rocking is insignificant.

Natural Periods

The natural periods for all significant modes of the Auxiliary Building are listed in Table 3.7-8.

Response Values

The analysis provides acceleration time histories for each degree of freedom at each mass point. Maximum absolute accelerations, relative displacements, story shears, overturning moments, and torsional moments are also calculated. These response quantities are listed in Tables 3.7-9 through 3.7-13.

Acceleration response spectra are calculated from the acceleration time histories at the mass points, and corresponding smooth spectra are prepared. Typical smooth spectra are shown in Figures 3.7-14 through 3.7-19.

Turbine Building and Intake Structure

The Turbine Building and the Intake Structure are Design Class II. However, Design Class I equipment is located inside: Component cooling water heat exchangers, 4160 volt vital switchgear, and emergency diesel generators in the Turbine Building; and auxiliary saltwater (ASW) pumps, ASW piping, and ASW instrumentation in the Intake Structure. In order to provide assurance that the function of Design Class I equipment will not be adversely affected even in the unlikely event of a Double Design Earthquake, these structures are reviewed for that earthquake to assure that they would not collapse. The analysis for the Turbine Building shows that some yielding would occur in the vertical X-bracing along column lines A and G, shown in Figure 3.7-19A, from such an
8 earthquake in the North-South direction. The analysis was based on conservative assumptions and showed that the yielding would be limited to safe values. The yielding occurs only in the X-bracing and does not extend to the columns or spandrel beams. The analysis for the Intake Structure showed that all stresses would be less than yield.

Design Class I Mechanical Equipment

Reactor Coolant Loop

Analysis of the reactor coolant loops and support systems for seismic loads is based on a three dimensional, multi-mass elastic dynamic model, discussed in Section 5.2. The model is shown in Figure 5.2-5.

Reactor Vessel Internals

A standard containment building with the reactor vessel support, the reactor vessel, and the reactor internals are included in the multi-mass mathematical model used to determine the dynamic response of the reactor internals. The mathematical model of the building, attached to the ground, is similar to that used to evaluate the building structure. The reactor internals are modeled as a single degree of freedom system for vertical earthquake analysis because previous analyses have shown that this is its behavior. The reactor internals are mathematically modeled by beams, concentrated masses, and linear springs for horizontal earthquake analysis.

All masses, water and metal are included in the mathematical model. All beam elements have the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components are attached uniformly their mass is included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements is also included as a distributed mass. Horizontal components are considered as a concentrated mass acting on the barrel. This concentrated mass also includes components attached to the horizontal members, since these are the media through which the reaction is transmitted. The water near and about these separated components is considered as being additive at these concentrated mass points. The modulus of elasticity is chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross sectional area is selected along with a value for Poisson's ratio. The fuel assembly moment of inertia is derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provide stiffness values for use in this analysis. The fuel assemblies are assumed to act together and are represented by a single beam. Figure 3.7-20 shows the mathematical model used.

The evaluation is performed for the simultaneous occurrence of horizontal and vertical seismic input motions. The total seismic response is obtained by adding the responses for vertical excitation absolutely to the separate results for the N-S and E-W directions. The larger of the two values so determined at each point in the model is considered as the earthquake response. The response spectrum method of analysis is used.

It is readily observed from the mode shapes how a particular component constitutes a natural mode of the system. The damping values are assigned accordingly. For example, since the first mode obtained from the eigenvalue-eigenvector solution using the model of Figure 3.7-20 is found to be determined by the fuel assemblies, as shown in Figure 3.7-21, the conservative damping value of 1 percent of critical is used for that mode. The damping values shown for concrete structure in Section 3.7.1 are used whenever the Containment Structure determines the system mode.

The response to horizontal excitation is conservatively determined by adding the absolute responses for all modes. The results obtained from the linear analysis indicate that during an earthquake, particularly the DDE, the relative displacements between the components may close the gaps and consequently the structures can impinge on each other. It is clear that linear analysis does not provide information about the impact forces generated when components impinge each other, but has the advantage of simplicity and provides information about the natural frequency of the system. Therefore, for those cases where components would be expected to impinge each other, linear analysis is applied but the gaps are conservatively treated as being closed. Reference 5 provides further details.

The criterion for normal plus DE loadings is that the stresses are limited to those given by the ASME Nuclear Power Plant Components Code for upset condition. These limits are intended to assure that the reactor will be able to continue or resume operation. For the normal plus DDE and the normal plus DDE plus DBA loading conditions, the criteria for acceptability in regard to mechanical integrity analyses is that adequate core cooling and core shutdown must be assured. This implies that the deformation of the reactor internals must be sufficiently small so that the geometry remains substantially intact. Consequently, the limitations established on the internals are concerned principally with the maximum allowable deflections and/or stability of the parts in addition to a stress criterion to assure integrity of the components. The deflections and stresses caused by the DDE are small in comparison to those caused by the DBA. Accordingly, faulted limits for the internals are covered in Section 3.9.3.

Other Design Class I Mechanical Equipment

Design Class I mechanical equipment is grouped into two categories: equipment purchased directly by the Applicant; and equipment supplied by Westinghouse Electric Corporation.

Equipment Purchased Directly by the Applicant

Design Criteria

Equipment is considered rigid if all natural periods are equal to or less than 0.05 second. Rigid equipment is designed for the maximum acceleration of the supporting structure at the equipment location. Flexible equipment (natural period greater than 0.05 seconds) is analyzed by response spectrum methods. Hydrodynamic analysis of tanks is performed using the methods described in Chapter 6 of Reference 6.

Allowable Stresses

Load combinations and allowable stresses for Design Class I equipment are given in Section 3.9.

Equipment Supplied by Westinghouse Electric Corporation

The seismic response of Design Class I piping and components is determined as part of a multi-degree of freedom model which includes the support characteristics. This model is a multi-mass mathematical representation of the system. A sufficient number of masses are included to assure an accurate determination of the dynamic response. A single mass model is used to determine vertical response loads for the seismic design when justified by the equipment design characteristics and/or the conservatism of the assigned loadings.

Horizontal and vertical seismic umbrella spectra are prepared which encompass the floor response spectra at the elevations where the piping system attaches to the building structure. The system is evaluated for the simultaneous occurrence of these horizontal and vertical seismic input motions. For each mode, the results for the vertical excitation are added absolutely to the separate results for the N-S and E-W directions. The larger of the two values so determined at each point in the model is considered as the earthquake response. The umbrella spectra are compared with the horizontal and vertical floor response spectra developed from the building time history analyses to assure conservatism of the spectra used.

The effect of differential seismic movement of interconnected components between floors are considered in the analysis. This effect is generally negligible.

In a coupled system with different structural elements, either the lowest damping value of the system was used for all modes, or the damping was determined for each mode according to the energy distribution. Whenever the latter method was used, a natural mode was found to have nearly all of its energy in a particular structural element so that the damping for that element could be used to give an accurate determination of the respose. The materials employed in systems are standard. The material properties which can effect a variation in modal period are well known, and the known variation in these properties does not account for any measurable or significant shift in period or increase in seismic loads.

Details of the response spectrum analysis are as follows:

1. If a component falls within one of the many categories which has been previously analyzed using a multi-degree-of-freedom model and shown to be relatively rigid, then the equipment specification for that component is checked to assure that the equivalent static g-values specified are larger than the building floor response spectrum values and therefore are conservative. Equipment is considered to be rigid relative to the building if its natural frequencies are all greater than twice the building's natural frequency for all building frequencies that have accelerations greater than the ground acceleration.

2. If the component cannot be categorized as similar to a previously analyzed component that has been shown to be relatively rigid, then an analysis is performed as described below.

Design Class I mechanical equipment including heat exchangers, pumps, tanks, and valves are analyzed using a multi-degree-of-freedom modal analysis. Appendages, such as motors attached to motor operated valves, are included in the models. The natural frequencies and normal modes are obtained using analytical techniques developed to solve eigenvalue-eigenvector problems. A response spectrum analysis is then performed using horizontal and vertical umbrella spectra that encompass the appropriate floor response spectra developed from the building time history analysis. The simultaneous occurrence of horizontal and vertical motions are included in the analyses. These response spectra are combined with the modal participation factors and the mode shapes to give the structural response for each mode from which the modal stresses are determined. The combined total seismic response is obtained by adding the individual modal responses utilizing the square root of the sum of the squares method. Combined total response for closely spaced modal frequencies whose eigenvectors are orthogonal are handled in the above mentioned manner. In the rare event when two significantly closely spaced modal frequencies occur and their eigenvectors are parallel, the combined total response is obtained by adding the square root of the sum of the squares of all other modes to the absolute value of one of the closely spaced modes.

Under certain conditions, the natural frequency of the equipment is not calculated. Under those conditions, using the appropriate damping value, the peak value of acceleration response curve is used to calculate the inertia forces. This method of calculation is termed the pseudo-dynamic method.

Hydrodynamic analysis of tanks is performed using the methods described in Chapter 6 of Reference 6. Bridge and trolley structures are designed so that restraints prevent derailing due to the DDE. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper under DDE.

Components and supports of the Reactor Coolant System are designed for the loading combinations given in Section 5.2. Components are designed in complete accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels and the USAS Code for Pressure Piping. The allowable stress limits for these components and supports are also given in Section 5.2.

In addition to these limits, the reactor vessel radial movement for normal plus the DDE plus the Design Basis Accident loading condition is limited to two inches.

The loading combinations and stress limits for components and supports are given in Section 3.9.

Design Class I Piping

Criteria

The following criteria determine the type of seismic analysis performed for Design Class I piping:

Six Inches in Diameter and Larger

Seismic analysis is performed by the response spectrum, modal superposition method.

Less than Six Inches in Diameter

Where feasible, the piping is restrained according to criteria which limit the periods of free vibration to values that assure only moderate amplification of support accelerations. For example, the maximum period for straight runs of piping is 0.067 seconds. The supports are conservatively designed for a seismic acceleration of 2g. In order to verify the adequacy of the restraint criteria and support loads, the most flexible combination of piping and support arrangement was analyzed by response spectrum, modal superposition using the largest applicable spectrum. The results of this analysis showed all piping stresses less than the allowable values given in Section 3.9 and all support loads less than the corresponding values calculated from the 2g seismic acceleration.

Where the piping can not be restrained to meet the criteria of the preceeding paragraph, the seismic analysis is performed by response spectrum, modal superposition.

Model

The response spectrum modal superposition analyses are performed either by John A. Blume & Associates, Engineers or by the Company. Three dimensional mathematical models are used. A typical mathematical model is shown in Figure 3.7-22. Valves and valve operators are included in the piping models as eccentric masses. Where Design Class II piping connects to Design Class I piping between anchor points, the Design Class II piping is included in the model. 26

Allowable Stresses

Load combinations and allowable stresses for Design Class I piping are given in Section 3.9.

Procedures Used to Lump Masses

Structures

The mass of the structure is assumed to be concentrated at particular locations on the model. These locations coincide with either floor levels, significant points where dynamic response is required as input for piping and equipment, nodal points in the finite element model, or any other points required to accurately define the natural frequencies and mode shapes for the significant modes.

Equipment and Piping

The mass of the equipment and piping systems is assumed to be concentrated at particular locations on the model. These locations coincide with either actual masses such as pumps, motors, valves, etc.; restraints and anchors; or any other points required to accurately define the natural frequencies and mode shapes for the significant modes.

Rocking and Translational Response Summary

Methods used to consider soil-structure interaction are described in this Section (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

Methods Used to Couple Soil With Seismic System Structures

The procedures used to represent the Containment Structure and surrounding rock mass as a finite element model and the procedures used to derive the stiffnesses of foundation springs for the Auxiliary Building are described in this Section (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

Development of Floor Response Spectra

Floor response spectra are developed using time history modal superposition analyses as described in this Section (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

Differential Seismic Movement of Interconnected Components

Components and supports of the Reactor Coolant System are designed for the loading combinations and stress limits given in Section 5.2. The loading combinations and stress limits for other components and supports are given in Section 3.9.

Effects of Variations on Floor Response Spectra

Consideration of the effects on floor response spectra of possible variations in the parameters used for the structural analysis is discussed in connection with the development of smooth spectra in this Section (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

Use of Constant Vertical Load Factors

The Category I structures are heavy, massive, reinforced concrete rigid-type structures and are founded on competent hard rock. For such structures, insignificant amplification of vertical motions can be expected - the critical factor in design being the response of the structures due to horizontal earthquake motions. The Containment Structure and Auxiliary Building are designed for a vertical static coefficient equal to two-thirds of the peak horizontal ground motion in accordance with the criteria approved in the PSAR.

Method Used to Account for Torsional Effects

The Containment Structure is essentially axisymmetric and therefore has insignificant torsional response. The torsional response of the Auxiliary Building is calculated by use of a combined translational and torsional mathematical model in the seismic system time history modal superposition analysis, as described in this Section (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

Comparison of Responses

Time history analyses only are performed for Design Class I structures. Response spectrum analyses are not performed because the time history produces spectra that accurately represent the criteria response spectra.

Methods for Seismic Analysis of Dams

There are no dams included in the Diablo Canyon plant.

Methods to Determine Overturning Moments

The maximum overturning moments for Design Class I structures are determined as part of the time history, modal superposition analyses. Vertical earthquake is considered to act concurrently with the maximum horizontal overturning moments.

Analysis Procedure for Damping

Structures are analyzed using modal superposition techniques, and element or material-associated damping ratios are given in Section 3.7.1. "Composite" or modal damping ratios in structural systems comprised of different element material types are selected based on an inspection of the significant mode shapes and on the assumption that the contribution of each material to the composite effective modal damping is proportional to the elastic energy induced in each material. The following criteria and procedures are applied on a mode by mode basis to evaluate and conservatively determine composite damping values.

1. Where a particular mode primarily indicates response of a single element type, the damping ratio corresponding to that element type is assigned to that mode. Where all but a negligible amount of the elastic energy is induced in, for example, concrete or rock, the damping ratio appropriate to these materials is applied. Similarly, where a lightly damped material exhibits a major portion of the elastic energy of the mode, a conservative choice is made to use the damping ratio of that material for that mode. In most cases for this plant, the modes are well defined according to material types and composite damping values can be selected on the basis of a visual inspection of mode shapes and no additional numerical computations are required.

2. In a few instances, the above criteria cannot be applied because a particular mode indicates response of several element types. The damping ratio for that mode is conservatively estimated based on the degree of participation of the different elements. The elastic energy induced in the different elements is estimated and the composite damping values assigned in proportion to the elastic energy.

The approach described above is consistent with currently accepted techniques, and in all cases, the damping values are selected conservatively. The use of this approach results in a design that can conservatively resist the seismic motions postulated for the Diablo Canyon site.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

Number of Earthquake Cycles

Where fatigue is a criterion, it is assumed that there are 20 occurrences of the DE, each producing 20 cycles of maximum response.

Basis for Selection of Forcing Frequencies

Design Class I equipment and piping is analyzed by the response spectrum method or the pseudo-dynamic method, using floor response spectra, unless it can be shown to be rigid, as discussed in Section 3.7.2 under the heading "Seismic Analysis Methods." Accordingly, a special procedure to avoid certain frequencies is not needed.

Root Mean Square Basis

The method for combining modal responses is defined in Subsection 3.7.2 under the heading "Seismic Analysis Methods."

Procedure for Combining Modal Responses

For closely spaced modes, the combined total response is obtained by taking the absolute sum of the closely spaced modes and then taking the square root of the sum of the squares with all other modes.

An exception to the procedure for closely spaced modes is Design Class I piping analyzed by the response spectrum modal superposition method where all modal responses are combined by the square root of the sum of the squares (SRSS) method to obtain total response.

A study was conducted to evaluate the effects of combining modes with closely spaced modal frequencies by the absolute sum method. A total of 29 piping systems were studied, which represented approximately 20 percent of the total number of piping systems analyzed.* Of these 29 piping systems, 8 systems had 3
no closely spaced frequencies and 8 systems had closely spaced frequencies which were in the rigid period range and therefore required no further study.

The remaining 13 systems had some modal frequencies in the flexible range which could be termed closely spaced. Of these, 5 systems had low seismic stresses with an adequate margin of safety so that any possible increase in seismic stresses due to combination of closely spaced frequencies by the absolute sum method would not affect the safety of the piping systems. In addition, 6 systems had closely spaced frequencies, but study of the mode shapes revealed that the seismic stresses would not be significantly affected by absolute sum of these modal responses.

*As of August 21, 1973

For the 2 remaining systems, it was not possible to positively conclude that the effects of combining the modes with closely spaced frequencies by absolute sum would be minimal by inspecting the stresses or mode shapes. Therefore, these 2 systems were reanalyzed by computer and it was found that if the seismic
3 responses of the modes with closely spaced frequencies were combined by the absolute-sum method, the increase in stress would be less than 1 percent.

It was therefore concluded that the combination of modal responses of piping systems by the SRSS method is adequate an- conservative.

Significant Dynamic Response Modes

The containment spray pump and motor are analyzed by the pseudo-dynamic method which involves the multiplication of the total weight of the equipment or component member by a seismic acceleration coefficient based on the peak value of the acceleration response curve. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as a single-degree-of-freedom system are considered to have a modal participation of one. Seismic acceleration coefficients for multi-degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation.

Design Criteria and Analytical Procedures for Piping

Stresses induced in Design Class I piping from relative movement of anchor points (points where all degrees of freedom are fixed) are combined with stresses calculated in the piping response spectrum modal superposition analyses.

When both anchor points are attached directly to a structure, the relative displacement is determined directly from the time history modal superposition analysis. When an anchor point is located on a component, the contribution of the component deflection to the deflection of the mathematical model for the piping analysis is evaluated and, if appropriate, the component is included in the model.

Basis for Computing Combined Response

As a minimum, mechanical equipment is designed for a vertical static coefficient equal to two-thirds of the peak horizontal ground motion in accordance with the criteria approved in the PSAR. Horizontal and vertical responses are combined by direct addition.

Equipment supplied by Westinghouse Electric Corporation is reviewed for a vertical force determined from a response spectrum, single degree-of-freedom analysis, as described in Subsection 3.7.2 under the heading "Description of Seismic Analyses," subheading "Design Class I Mechanical Equipment."

The horizontal and vertical response of Design Class I piping is determined from the response spectrum, modal superposition analyses described in Subsection 3.7.2 under the heading "Description of Seismic Analyses," subheading "Design Class I Piping." Horizontal and vertical responses are combined by direct addition on a mode by mode basis. Modal responses are combined by the square root of the sum of the squares method. 30

Use of Simplified Dynamic Analysis

All methods of seismic analysis used for Design Class I structures, systems, components, and piping are described in Subsection 3.7.2.

Two methods of dynamic seismic analysis are used for Design Class I components and piping which are different than multiple-degree-of-freedom, modal analysis methods. The first of these is the response spectrum, single-degree-of-freedom method used for components whose dynamic behavior can be accurately represented by a single-degree-of-freedom mathematical model. The second of these is the method for rigid components and piping where the component or piping is designed for the maximum acceleration experienced by the supporting structure at the location of support, if all natural periods of the component or piping are less than or equal to 0.05 second.

The "pseudo-dynamic" method of analysis is used for certain items of mechanical equipment as described in Subsection 3.7.2. The basis for this method is described in that Subsection (3.7.2) under the heading "Description of Seismic Analyses," subheading "Design Class I Mechanical Equipment."

Certain Design Class I piping less than six inches in diameter is restrained according to criteria described in Section 3.7.2 under the heading "Description of Seismic Analyses," subheading "Design Class I Piping." The results of a response spectrum, modal superposition analysis performed to verify the adequacy of the restraint criteria are also described at that location.

Modal Period Variation

Consideration of the effects on floor response spectra of possible variations in the parameters used for structural analysis is discussed in connection with the development of smooth spectra in Section 3.7.2 under the heading "Description of Seismic Analyses," subheading "Design Class I Structures."

Torsional Effects of Eccentric Masses

Valves and valve operators are included as eccentric masses in the mathematical models for piping seismic analysis, as described in Section 3.7.2 under the heading "Description of Seismic Analyses," subheading "Design Class I Piping."

Piping Outside Containment Structure

The procedures used to determine piping stresses resulting from relative movement between anchor points (points where all degrees of freedom are fixed) are discussed in this Section (3.7.3) under the heading "Design Criteria and Analytical Procedures for Piping." The forces exerted by piping on anchor points including the Containment Structure penetrations, are included in the evaluation of stresses for Design Class I structures.

Buried Design Class I piping is confined by sand backfill in a rock trench. The piping material is ASTM A 53 carbon steel. The ductility of this material provides assurance that the integrity of the pipe would be maintained in the unlikely event of a localized stress or strain concentration.

Interaction of Other Piping With Design Class I Piping

Mathematical models for Design Class I piping seismic analyses normally originate and terminate at anchor points. Where Design Class II piping connects to Design Class I piping between anchors, the Design Class II piping is included in the mathematical model.

Field Location of Supports and Restraints

Seismic supports and restraining devices for Design Class I piping are located as follows:

Two Inches In Diameter and Less

38 Field-routed and vendor-furnished piping two inches and less in diameter is supported by the piping installation contractor's field personnel in accordance with criteria supplied by the Company's Department of Engineering on Approved for Construction drawings. These criteria specify size, type, spacing, and permissable locations for seismic supports and restraining devices. Prior to fuel loading, the completed installation of this piping will be viewed by an experienced piping engineer from the Company's Department of Engineering to assure compliance with the criteria and the observance of good design practice.

Larger Than Two Inches in Diameter

The size, type, and location of each support or restraining device on each line is shown on Approved for Construction drawings.

The procedures followed during development of the Approved for Construction drawings provide assurance that the field location and the seismic design of supports and restraining devices are consistent with the assumptions made in the seismic analysis. These procedures are:

1. The locations of supports and restraining devices are established on preliminary drawings.

2. The locations shown on the preliminary drawings are used to develop the mathematical model for the seismic analysis, and the seismic analysis is performed. If the results show piping stresses higher than allowable, adjustments are made in the location or type of support or restraining device and the seismic analysis is repeated.

3. The reactions calculated as part of the seismic analysis, combined with other loads, are used for final design of piping supports and restraining devices.

4. When the design is complete, drawings are issued Approved for Construction to the piping installation contractor. Installation of supports and restraining devices is in accordance with Approved for Construction drawings.

3.7.4 SEISMIC INSTRUMENTATION PROGRAM

The seismic instrumentation installed at the Diablo Canyon consists of Strong Motion Triaxial Accelerographs, Multielement Seismoscopes, and Peak Acceleration Recorders, as described below. This instrumentation is installed to fulfill all requirements of Regulatory Guide 1.12.

Location and Description of Instrumentation

Seismic instrumentation is provided in the free field, on the Unit 1 Containment Structure, and on the Auxiliary Building to completely define free field ground motions and to sample structural response. All strong motion triaxial accelerographs are rigidly attached to the structures so that their records can be related to movement of the structures, and all are accessible for periodic servicing and for obtaining readings.

Strong Motion Triaxial Accelerographs

A strong motion triaxial accelerograph provides a time history of acceleration for each of three orthogonal directions. These histories are recorded at a remote location on magnetic tape. The instrument starts recording on receipt of a signal from a seismic trigger which has an adjustable threshold. Three strong motion triaxial accelerographs are provided at Diablo Canyon: two on the Unit 1 Containment Structure (one on the base slab and the other at Elevation 231), and one rigidly mounted on the foundation rock far enough from the plant structures that the recordings are representative of free field ground motions. The recording equipment is located in the control room.

Multielement Seismoscopes

A multielement seismoscope consists of twelve singe-degree-of-freedom masses arranged in such a way that each mass scribes its maximum displacement on a steel plate. Each of the twelve masses has a different natural frequency of
3 vibration, with the natural frequencies selected to cover the range of interest for the design response spectrum. The maximum displacement of each mass can be directly related to its maximum acceleration, thus providing a point on the response spectrum. One set of three multielement seismoscopes, each aligned on one of three orthogonal axes, will be installed on the base slab of the Unit 1 Containment Structure.

Peak Acceleration Recorders

Peak acceleration recorders are located as follows:

1. On the internal structure and exterior shell of the Unit 1 Containment Structure.

2. At Elevation 140 in the Auxiliary Building.

3. On the foundation slab of the Turbine Building.

4. Adjacent to the auxiliary saltwater pumps in the Intake Structure.

Control Room Operator Notification

Operation of the time history recording strong motion triaxial accelerograph will activate an annunciator in the control room. The accelerograph record will be available to the operator within a few minutes using rapid playback equipment supplied by the accelerograph manufacturer.

Evaluation

In the event of an earthquake that produces significant free field ground motions, all seismic instruments will be read and the readings compared to the corresponding design values. This comparison, together with information provided by other plant instrumentation and an inspection of safety-related systems will form the basis for a judgement on the effects of the earthquake.

3.7.5 SEISMIC DESIGN CONTROL MEASURES

Equipment Purchased Directly by the Applicant

The Pacific Gas and Electric Company engineering department is described in Chapter 17 as are the procedures for specifying technical requirements in purchase orders and specifications.

The design seismic accelerations developed from the structure seismic system analysis are included in the purchase order or specification for Design Class I equipment. The purchase order or specification requires that the manufacturer submit seismic calculations for review by the Pacific Gas and Electric Company responsible engineer. The calculations are approved only when all seismic design criteria are met.

Equipment Supplied By Westinghouse Electric Corporation

The following procedure is implemented for Design Class I mechanical equipment that falls within one of the many categories which have been analyzed as described in Section 3.7.2 and has been shown to be rigid relative to the building.

1. Equivalent static acceleration factors for the horizontal and vertical directions are included in the equipment specification. The vendor must certify the adequacy of the equipment to meet the seismic requirements as described in Section 3.7.2.

2. When the floor response spectra are developed the cognizant engineer responsible for the particular component checks to assure that the acceleration factors are less than those given in the equipment specification.

All other Design Class I equipment is analyzed or tested as described in Sections 3.7.2 and 3.10.

Design control measures and design documentation for all Design Class I systems, structures and components are in accordance with formalized quality assurance procedures. These procedures are presented in Chapter 17 "Quality Assurance," Section 17.1.

3.7.6 REFERENCES

1. Blume, John A. & Associates, Engineers, "Diablo Canyon Unit No. 1, Tanks on Compacted Fill," Letter report summarizing earlier recommendations, April 1973.

2. Timoshenko and Goodier, "Theory of Elasticity", McGraw-Hill, 1951, Second Edition.

3. Barkan, D.D., "Dynamics of Bases and Foundations", McGraw-Hill, 1961.

4. Richart, F.E. Jr.; Hall, J.R. Jr.; and Woods, R.D., "Vibrations of Soils and Foundations", Prentice-Hall, 1970.

5. Bohm, G., "Seismic Analysis of Reactor Internals for Pressurized Water Reactors", First National Congress of Pressure Vessel and Piping Technology, ASME Panel on Seismic Analysis & Design of Pressure Vessel and Piping Components, San Francisco, May 10-12, 1971.

6. U.S. Atomic Energy Commission (Division of Reactor Development) Publication TID - 7024, "Nuclear Reactors and Earthquakes".

APPENDIX 3.7A

Diablo Canyon Seismic Reevaluation for a Magnitude 7.5 Hosgri Earthquake

Introduction

The information contained in this appendix presents a program of seismic reevaluation of the Diablo Canyon Design Class I structures, systems and components for the seismic conditions stipulated in Supplement No. 4 to the Safety Evaluation Report issued by the U.S. Nuclear Regulatory Commission on May 11, 1976. Briefly, the program considers a hypothetical magnitude 7.5 earthquake on the Hosgri fault causing an effective peak horizontal free field ground acceleration at the plant site of 0.75g and develops appropriate elastic response spectra for input to the plant structures. The spectra presented are site specific and account for the filtering effect of large foundations on the ground motion waves. Where justified, inelastic behavior using appropriate ductility factors will be used in applying these curves. Damping values are in accordance with Regulatory Guide 1.61 or as otherwise accepted by the Nuclear Regulatory Commission for seismic analysis at the S. S. E. level. Torsion and tilting due to the lack of synchronism of the ground waves beneath structures are to be accounted for in the analysis. The sequence of steps in verifying seismic adequacy together with the acceptance criteria to be adopted, including the use of documented actual material strengths, are presented. Preliminary results and conclusions, as known, are given.

Development of Spectral Response

The 0.75g peak acceleration was used as a normalizing value for time histories of strong motion for selected earthquakes recorded on rock close to the epicenters, thus providing the best available models for the Diablo Canyon conditions relative to the Hosgri fault zone. The eight earthquake records used are as follows.

Earthquake	M	Depth (km)	Recorded at	Epicentral Distance (km)	Component	Peak Acceleration (g)
Helena 1935	6	5	Helena	3 to 8	EW	0.16
Helena 1935	6	5	Helena	3 to 8	NS	0.13
Daly City 1957	5.3	9	Golden Gate Park	8	N80W	0.13
Daly City 1957	5.3	9	Golden Gate Park	8	N10E	0.11
Parkfield 1966	5.6	7.0	Temblor 2	7	S25W	0.33
Parkfield 1966	5.6	7.0	Temblor 2	7	N65W	0.28
San Fernando 1971	6.6	13	Pacoima Dam	3	S14W	1.17
San Fernando 1971	6.6	13	Pacoima Dam	3	N76W	1.08

The magnitudes are the greatest recorded thus far close in on rock stations and range from 5.3 to 6.6. Adjustments were made subsequently in the period range of the final response spectrum above 0.40 sec. for the greater long period energy expected in a 7.5M shock as compared to the model magnitudes.

The procedure followed was to develop seven percent damped response spectra for each of the eight records normalized to 0.75g and then to treat the results statistically according to period bands to obtain the mean, the median and the standard deviations of spectral response. At this stage, no adjustments for the size of the foundation or for ductility were made. (Figure 3.7A-1 for free field $\tau = 0$).

Foundation Size

It is known that a large foundation is subjected to reduced motion from high frequency waves, as compared to particle motion or to motion of a small foundation[1, 2, 3, 4]. The record of large buildings in major earthquakes has generally been good[5]. Using the same time histories of motion as listed above, the acceleration was averaged over the time required for the waves to pass the foundations. New spectra were developed, statistically treated, and used as before to obtain curves for analysis. These curves were then smoothed and adjusted as before for magnitude effects in the long period range.

The parameters employed for the high frequency averaging procedure are as follows:

	Tau = Effective Length of Foundation / Shear Velocity
Containment and Intake Structure	0.04 Sec.
Auxiliary Building	0.06 Sec.
Turbine Building	0.08 Sec.
Minor Structures	0 Sec.

Proposed Response Spectra

Figure 3.7A-1 provides a comparison of the 7.5M Hosgri elastic spectra at seven percent damping for the various buildings to the 6.5M Hosgri and DDE spectra at seven percent and five percent damping, respectively.

Figures 3.7A-2 through 3.7A-13 show curves for the 7.5M Hosgri spectra for various tau and damping values. The zero period and very high frequency portions are rounded to values slightly less than obtained from tau alone for a flat foundation at the surface, in view of the contiguous foundations and also a very conservative allowance for embedment and wave scattering phenomena[4]. On this basis, the curves shown in Figures 3.7A-2 through 3.7A-13 for damping values from two to seven percent and tau values of 0, 0.04, 0.06, and 0.08 represent elastic response spectra applicable to the containment and intake structures, auxiliary building, and turbine building respectively. These are all elastic curves with no reduction for ductility. Inelastic response, when considered, will be accounted for by appropriately adjusting the elastic response quantities.

Application of Response Spectra

In using the response spectra in the process of seismic design re-evaluation, the analyses may be spectral (by hand or computer) or time history providing the time history is modified so as to closely duplicate the response spectrum of interest. In all cases, at least three modes (if there are three or more modes) and all modes with periods greater than 0.05 seconds, shall be considered and combined as the square root of the sum of the squares. Modal and horizontal and vertical component combinations of motion shall be treated as in the original design. The rock base may be considered as rigid where allowed under current NRC procedures for sites with shear velocity of 3,500 fps or more.

Torsion and Tilting

If a foundation is long relative to the length of the ground waves, torsion about the vertical axis is induced even if the structure is symmetrical[1, 2]. Tilting about a horizontal axis may also occur. For the Diablo Canyon structures, torsion generally represents a relatively small contribution to the total seismic load and is typically resisted by shear in the same structural elements which resist translational seismic loads. Torsion was specifically considered in the design of the auxiliary building, and the equipment located therein, to account for the asymmetry of the structure, thus providing it with inherent torsional resistance. The torsional stresses induced in the containment shell are low. In general, for all structures, the torsional response increases with height above the base and distance from the center of mass.

To account for the torsional phenomenon, where significant, an allowance for torsionally induced tangential acceleration at the extremities of a foundation shall be made in the amount of $n\,\Upsilon$ where n is as shown in Table 1. The factor n varies with the ratio of plan dimensions a and b of the foundation.[6] An equivalent foundation diameter equal to $\sqrt{ab}$ will be used for rectangular structures and the actual diameter for circular structures. Thus:

$\alpha_o = n\tau$, in g units at base

wherein: α_o = The tangential acceleration at the periphery from torsion (only) at the base, g

$$\tau = \frac{\sqrt{ab}}{V_s}, \text{ or } = \frac{dia}{V_s}, \text{ Sec, as per Table 1}$$

V_s = Shear velocity of Rock, fps.

a and b = Plan dimensions of foundation ($a \geq b$) feet

The torque T_o, applied to the base is:

$$T_o = 0.5\alpha_o k^2 W_s \sqrt{ab}, \text{ ft - kip}$$

wherein: W_s = The seismic weight of the structure and equipment, kips

k = Ratio of radial distance to center of mass (above the base) of an average differential circular sector to $\frac{\sqrt{ab}}{2}$

TABLE 1

Structure	a (ft)	b (ft)	$\sqrt{ab}$ (ft)	t (sec)	n	α_o (g)	T_o/W_s (ft)
Containment	148(dia)		148(dia)	0.04	6	0.24	17.8
Auxiliary	250	173	208	0.06	5	0.30	31.2
Turbine Building	742	139	321	0.08	4	0.32	51.4
Intake	240	104	158	0.04	4	0.16	12.6

The tangential acceleration at any point of interest in the structure is resolved into vectors (parallel to the translational mode analyses) and the effects of each vector on stress or stress parameters are determined. The effects of the torsional vector are then combined with the effects of each translational mode in the same direction, in turn, on the basis of the square root of the sum of the squares.

Investigations thus far indicate that, for most of the points in the structures, the increase in stress due to torsion will be a very small percentage of the stress due to translational modes. The reasons for this include some or all of the following factors: Tau is small, the height above the base may be small, the radius from the center of mass may be small, and the horizontal spectral accelerations are generally large compared to the tangential accelerations due to torsion. In any case where the stress increase due to torsion is significant, torsional analysis shall be conducted under the assumption that the torque decreases linearly with height from T_o at the base to zero at the top. The analysis shall take into account the inertial effects and the fundamental natural mode of torsional vibration.

Where two or more structures are founded on a common base, such as the containment exterior shell and the interior structure, the torsional input to the base will be the same for each structure, but the response of the structure shall be varied in accordance with its polar moment of inertia, equivalent radius, seismic weight, and shear modulus.

Figure 3.7A-14 is an example showing absolute tangential acceleration in the containment exterior shell. In view of the fact that the translational spectral accelerations (which also produce shear in the concrete of the shell) are several times the tangential accelerations and are operative at different frequencies, it can be seen that the net effect of torsion in such a case is a small fraction of the translational stresses.

Tilting from some waves is only possible where the foundations are relatively deep and even then there is some normally neglected restoring value from the sides. Other waves can theoretically induce a tilting moment without embedment. However, the effects in this case are grossly reduced by the competency and stiffness of the rock and the effect of gravity.

In view of the above, plus the fact that only high frequency waves can cause tilting by interaction whereas the translational fundamental modes are generally at medium or low frequencies, it is not considered necessary to review further for this particular phenomenon.

Material Properties and Allowable Stresses

Allowable stress under severe earthquake conditions depends upon the actual properties of the materials in a structure. Although it is well known that such properties vary over a controlled range of values, and an individual member or connection must be designed for stresses below the average, in a whole structure, there is available the spectrum of values, most or all of which are effective in emergencies. It is therefore appropriate to utilize average actual material properties in analysis for severe seismic events rather than highly conservative and inappropriate design values intended for individual members; alternatively, the problem can be treated probabilistically.[7]

Treatment of Ductility

Slight excursions beyond the yield point shall be allowed under certain conditions when checking the plant for the effects of a hypothetical 7.5M earthquake nearby on the Hosgri fault zone. The yield point for this purpose shall be determined under the assumption of idealized elasto-plastic behavior as shown in Figure 3.7A-15. The "effective" yield point, u_y, shall be determined so that Area A equals Area B, and Area D equals the sum of Areas C and E.

The ductility factor, μ, is defined by:

$$\mu = \frac{u_m}{u_y}$$

wherein u_m and u_y are the maximum displacement and the displacement at first yielding, respectively, as determined by the properties of the materials subject to the following limits:

		$\dfrac{u_m}{1.2\ u_y}$
Where Containment is Involved	....	$1.2\ u_y$
Category I Other Than Containment	....	$1.3\ u_Y$
Equipment and/or Its Supports	....	Up to $1.3\ u_y$ Where This Value or More Can Be Documented

The allowable ductility, μ, is thus the constant shown in the above table. The value of μ in absorbing energy can be transformed into a reduction of the spectral demand as follows:

$$S'_a = \rho S_a$$

wherein: S'_a = The Elastic Spectral Acceleration Reduced for Ductility

S_a = The Elastic Spectral Acceleration

ρ = A Ductility Reduction Factor from Figure 3.7A-16

Non-Category I structures, such as the turbine building superstructure with its steel frame, shall be checked with story ductility factors up to three. Local ductility factors in individual steel elements or portions thereof shall not exceed six. The applicable equations from Figure 3.7A-16 shall be used to reduce spectral accelerations.

Strain criteria shall be compatible with the ductility factor used in determining inelastic response values and the need to assure that the associated deformation will not affect safety functions of plant systems and structures.

<u>Analysis Program and Scope</u>

The analysis shall consider the following plant areas.

1. Containment
2. Auxiliary Building
3. Turbine Building
4. Intake Structure
5. Outdoor Tanks

The program shall be designed to verify adequacy of an item as early in the analysis path as possible by comparing the Hosgri response to that for which the structure or component has been, or can be, qualified. Analysis shall be carried as far as is necessary to verify adequacy or to identify any necessary modifications to structures or components. Primary attention in the analysis shall be directed at structures or components important to safety in the following systems:

1. Reactor Coolant Pressure Boundary
2. Those Necessary to Accomplish and Maintain Safe-Shutdown
3. Those Necessary for Heat Rejection to the Ultimate Heat Sink
4. ECCS

REFERENCES

1. Yamahara, H., "Ground Motions During Earthquakes and the Input Loss of Earthquake Power to an Excitation of Buildings", Soils and Foundations, Volume 10, No. 2, 1970, Pages 145-161, Tokyo.

2. Ambraseys, N., "Characteristics of Strong Ground Motion in the Near Field of Small Magnitude Earthquakes", Invited Lecture, Fifth Conference European Committee for Earthquake Engineering, Istanbul, September 1975.

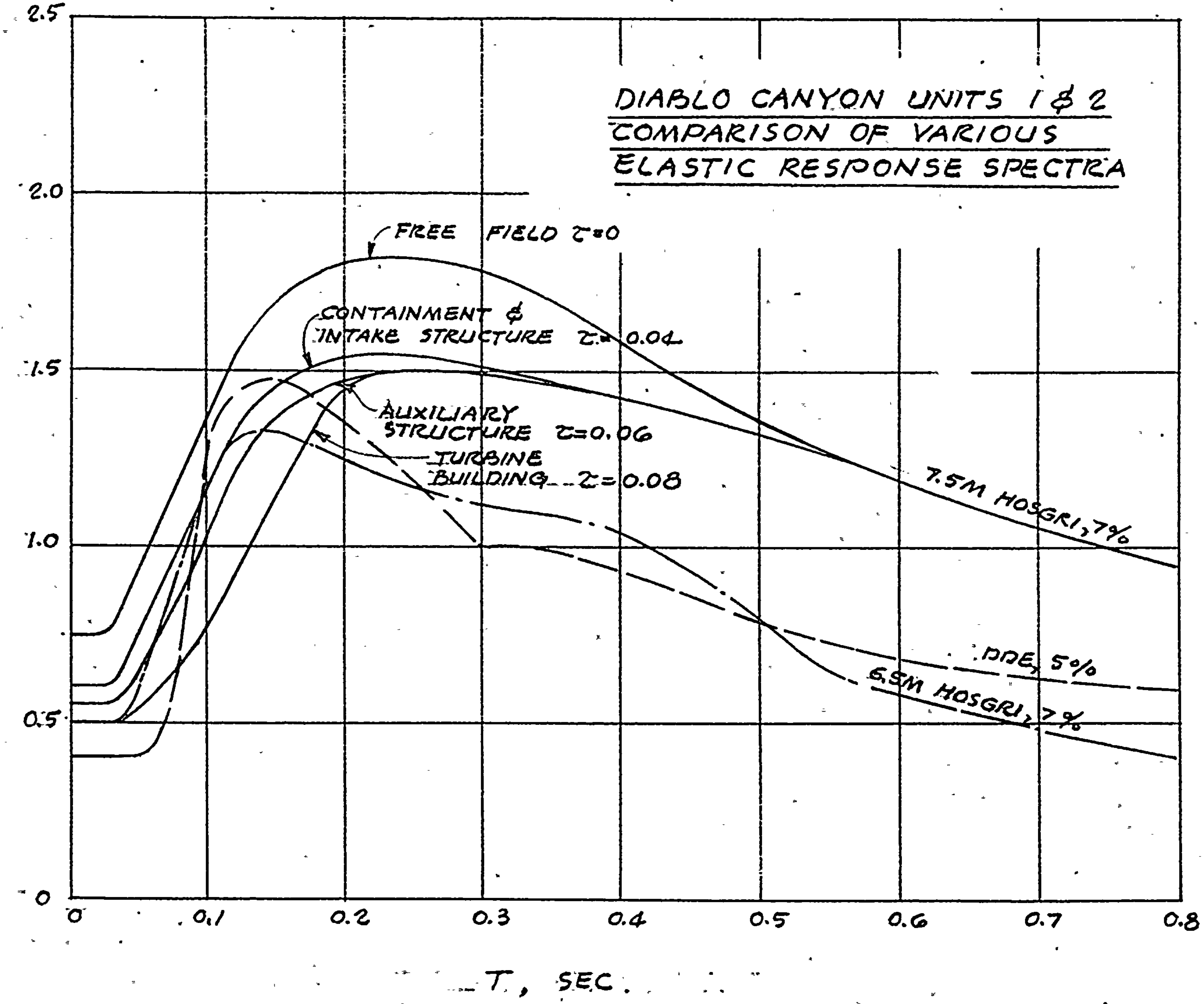
3. Scanlan, R. H., "Seismic Wave Effects on Soil-Sturcture Interaction", Earthquake Engineering and Structural Dynamics, Volume 4, 1976, Pages 379-388.

4. Wong, H. L., and M. D. Trifunac, "Interaction of a Shear Wall With the Soil for Incident Plane SH Waves: Elliptical Rigid Foundation", Bulletin of the Seismological Society of America, Volume 64, No. 6, 1974.

5. Page, Robert A., John A. Blume, and William R. Joyner, "Earthquake Shaking and Damage to Buildings", Science, Volume 189, No. 4203, August 1975.

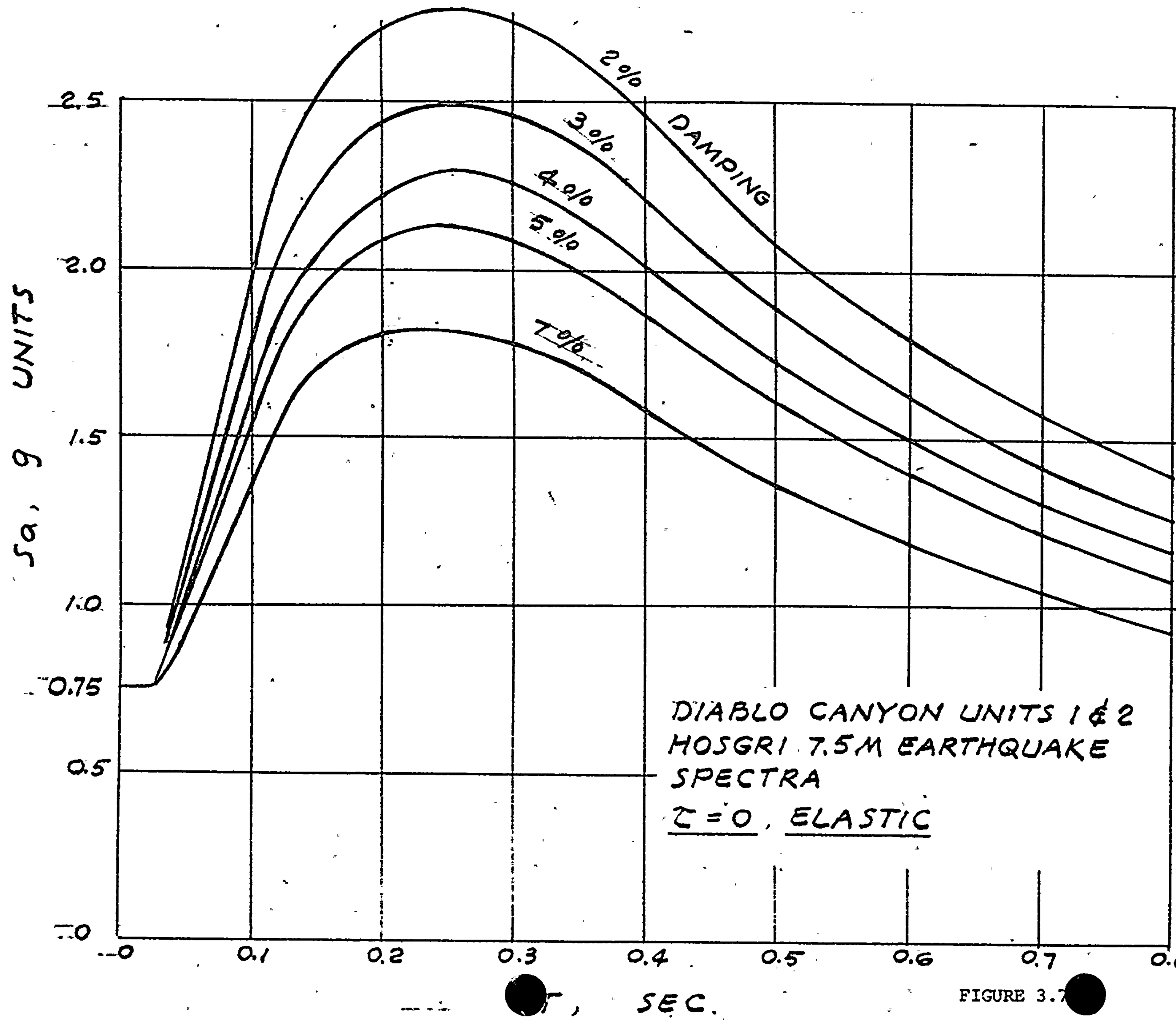
6. Newmark, Nathan M., "Torsion in Symmetrical Buildings", Proceedings, Fourth World Conference on Earthquake Engineering, 1969.

7. Blume, John A., "Allowable Stresses and Earthquake Performance", Sixth World Conference on Earthquake Engineering, New Delhi, India, 1977.



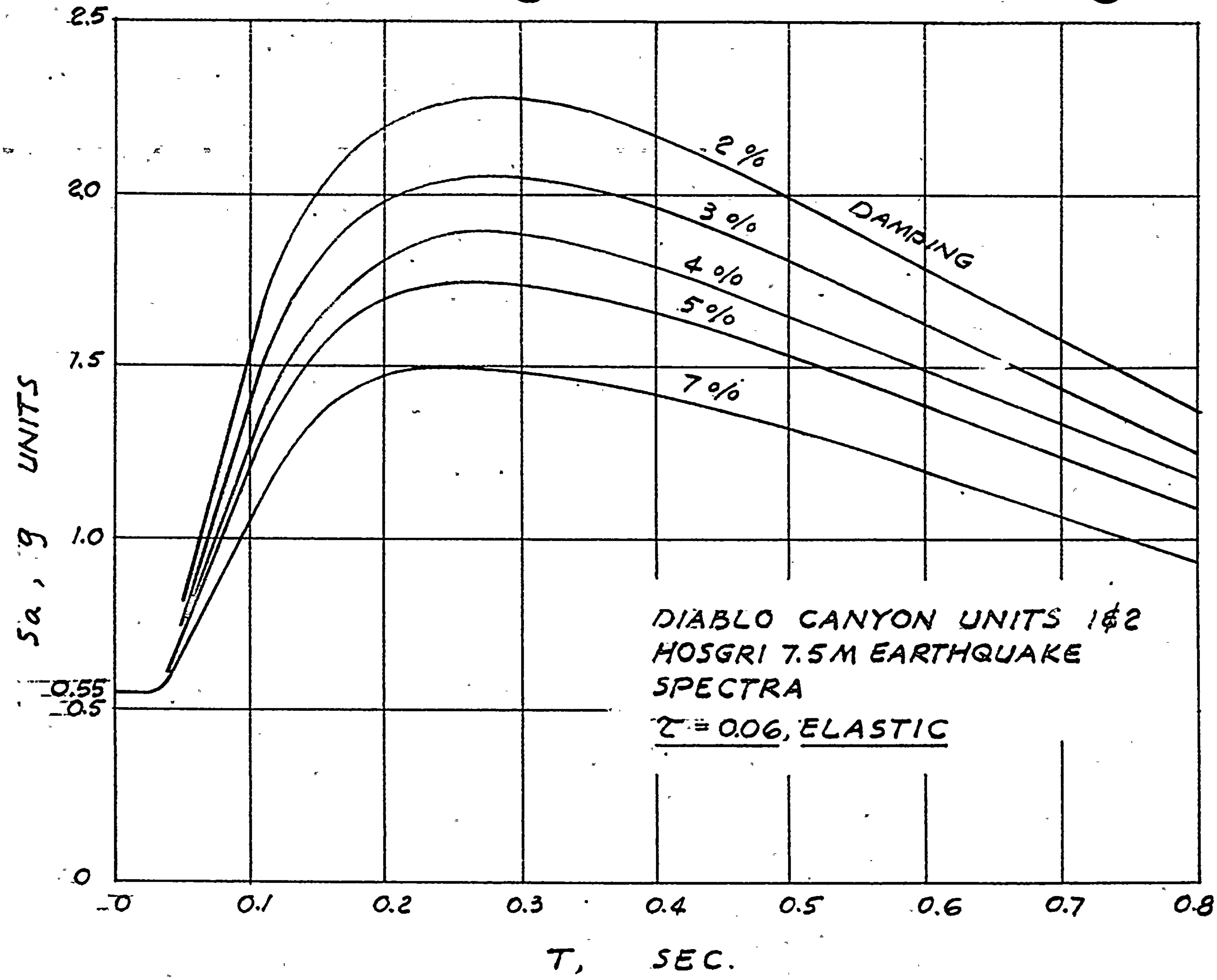
DIABLO CANYON UNITS 1 & 2
COMPARISON OF VARIOUS
ELASTIC RESPONSE SPECTRA
FREE FIELD τ=0
CONTAINMENT &
INTAKE STRUCTURE τ=0.04
AUXILIARY
STRUCTURE τ=0.06
TURBINE
BUILDING τ=0.08
7.5M HOSGRI, 7%
DDE, 5%
6.5M HOSGRI, 7%
Sa, g UNITS
T, SEC.
2.5
2.0
1.5
1.0
0.5
0
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8

FIGURE 3.7A-1



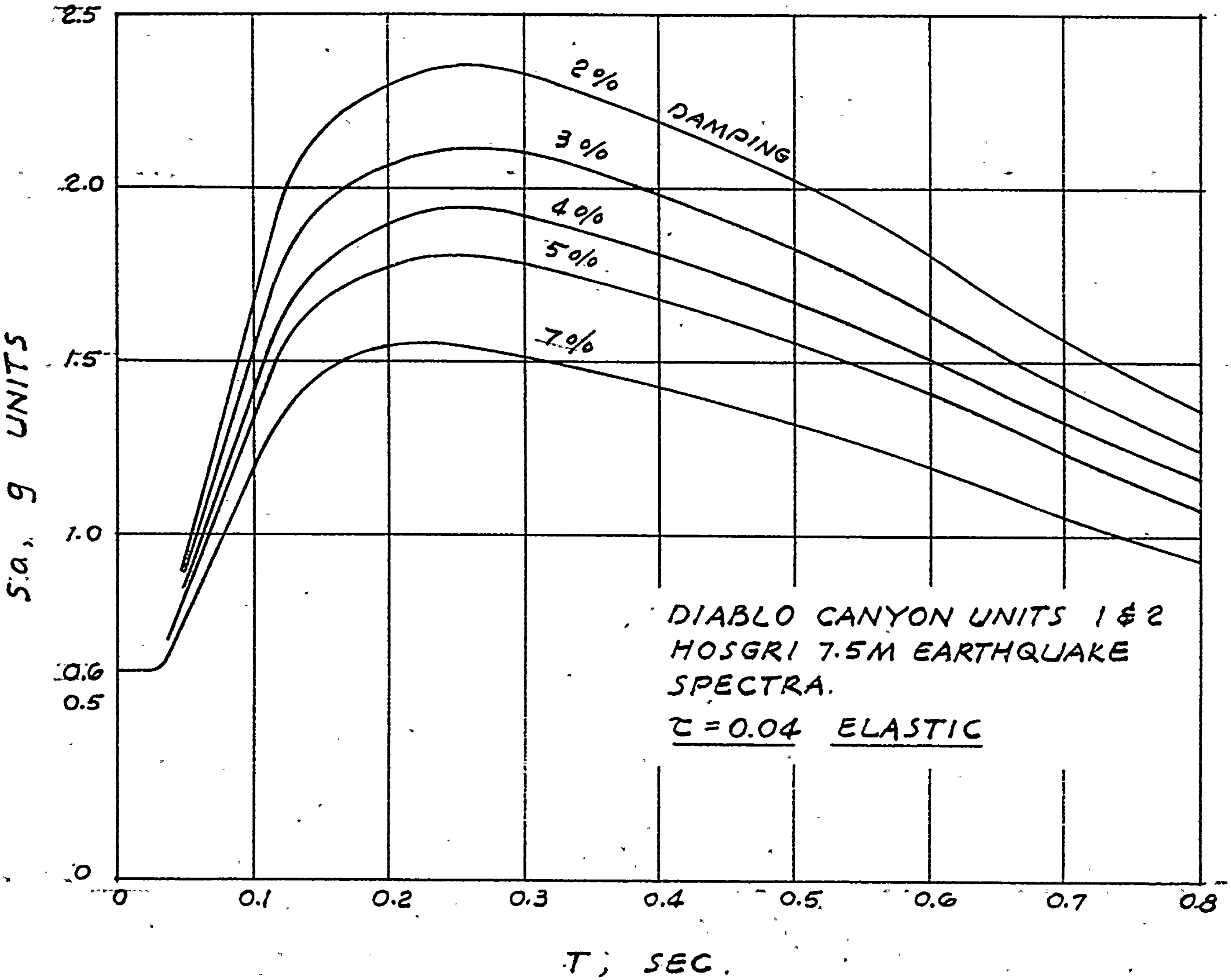
2%
3%
4%
5%
7%
DAMPING
2.5
2.0
1.5
1.0
0.75
0.5
0
Sa, g UNITS
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
T, SEC.
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0, ELASTIC

FIGURE 3.7



2.5
2.0
1.5
1.0
0.55
0.5
0
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
Sa, g UNITS
T, SEC.
2 %
3 %
4 %
5 %
7 %
DAMPING
DIABLO CANYON UNITS 1&2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0.06, ELASTIC

FIGURE 3.7A-4



2 %
DAMPING
3 %
4 %
5 %
7 %
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE
SPECTRA.
τ = 0.04 ELASTIC
S.a., g UNITS
2.5
2.0
1.5
1.0
0.6
0.5
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
T; SEC.

FIGURE 3.7A-3

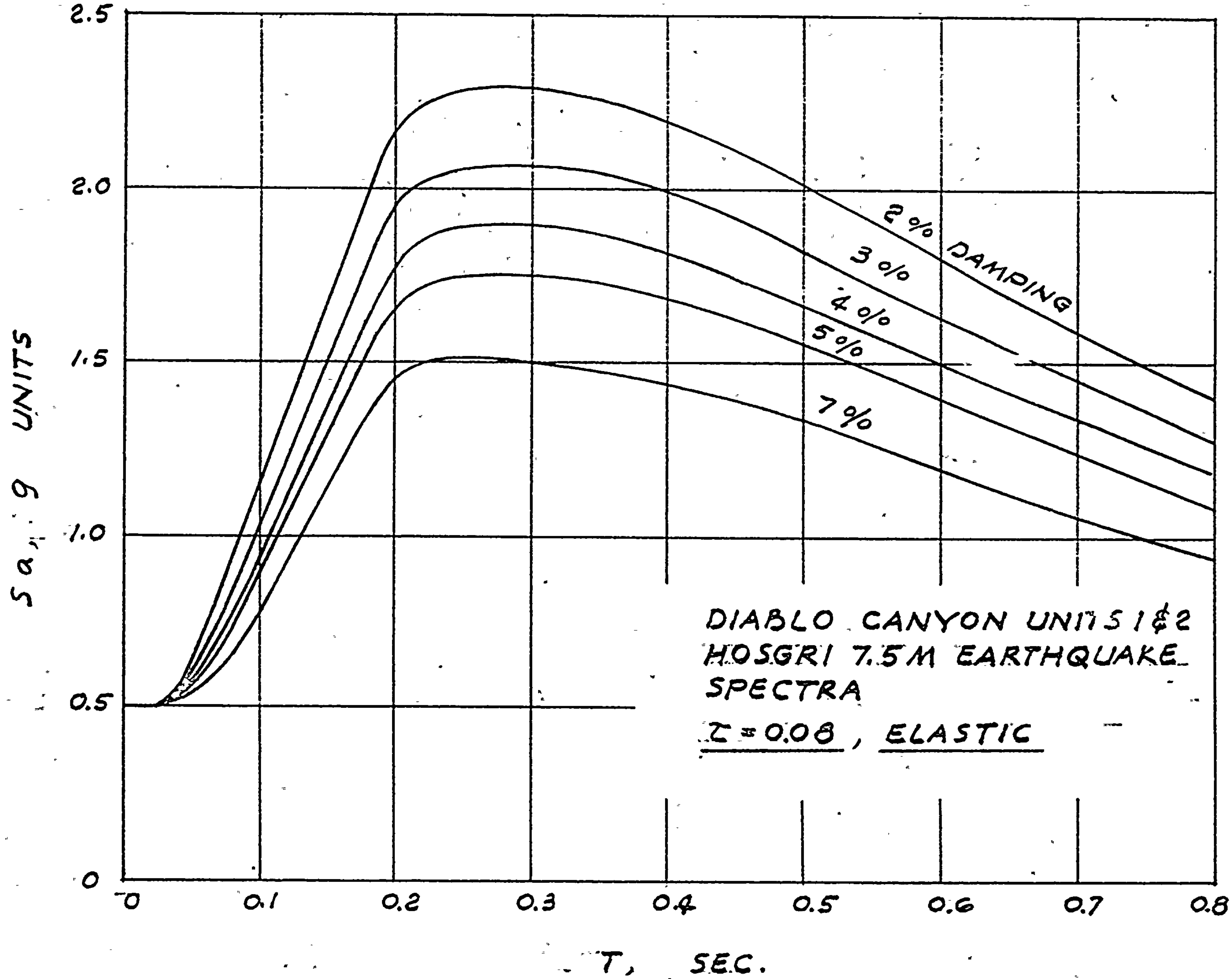

2.5
2.0
1.5
1.0
0.5
0
Sa, g UNITS
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
T, SEC.
2 % DAMPING
3 %
4 %
5 %
7 %
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0.08 , ELASTIC

FIGURE 3.7A-5

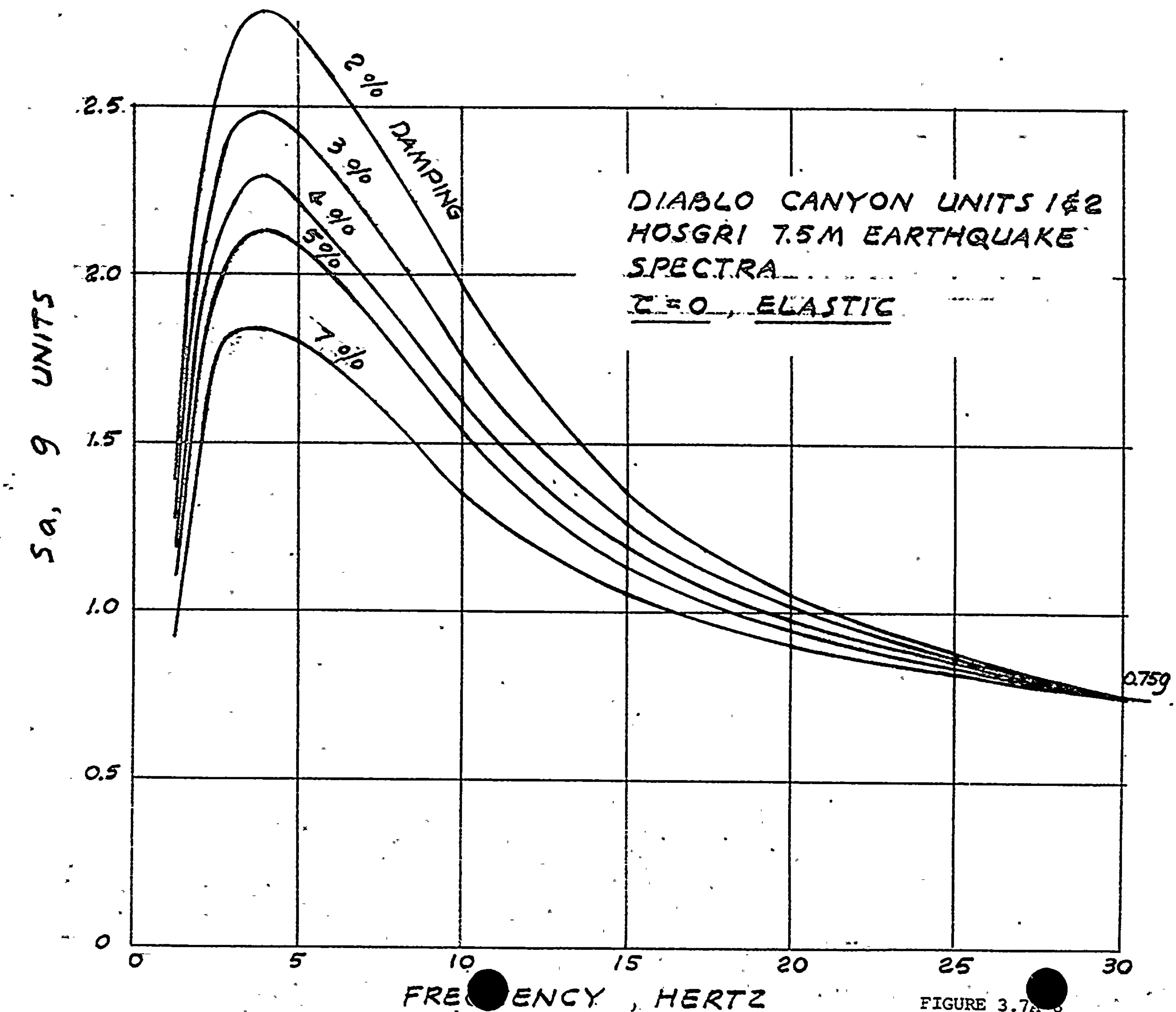

DIABLO CANYON UNITS 1&2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0 , ELASTIC
2 % DAMPING
3 %
4 %
5 %
7 %
0.75g
Sa, g UNITS
0
0.5
1.0
1.5
2.0
2.5
0
5
10
15
20
25
30
FREQUENCY , HERTZ

FIGURE 3.7A-8

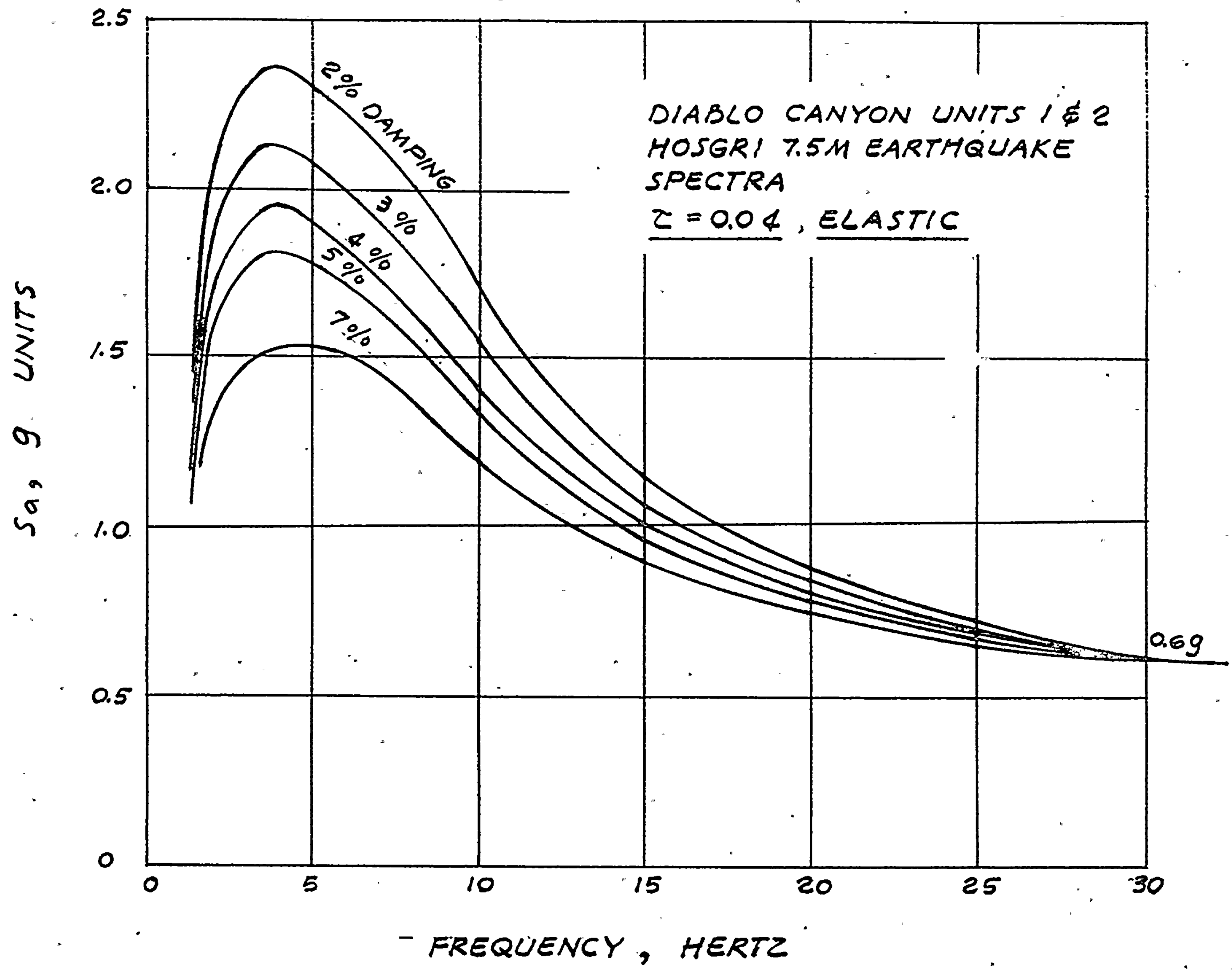

DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0.04 , ELASTIC
2% DAMPING
3 %
4 %
5 %
7 %
0.6g
Sa, g UNITS
FREQUENCY , HERTZ
2.5
2.0
1.5
1.0
0.5
0
5
10
15
20
25
30

FIGURE 3.7A-7

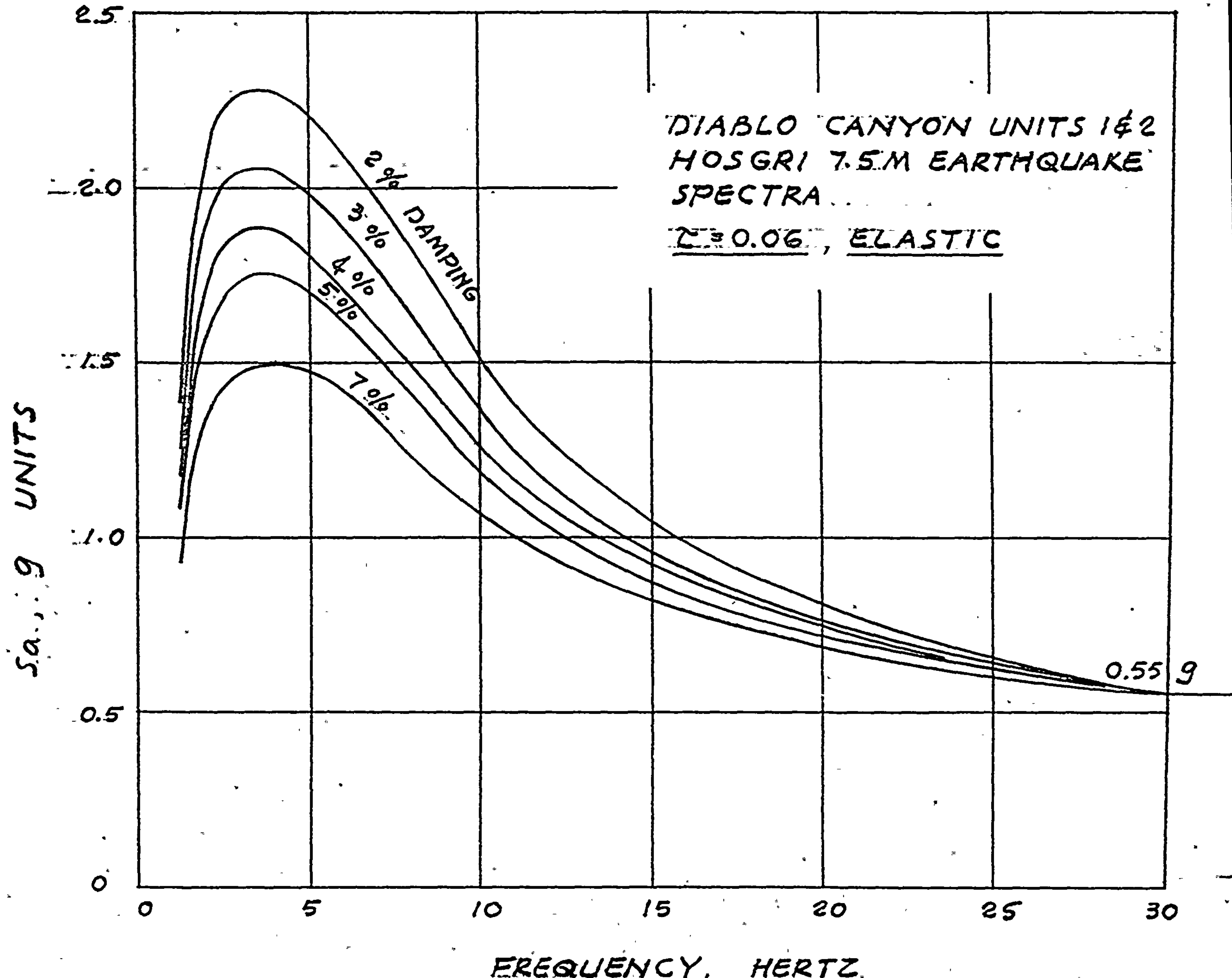

DIABLO CANYON UNITS 1&2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0.06 , ELASTIC
2%
3%
4%
5%
7%
DAMPING
0.55 g
Sa., g UNITS
2.5
2.0
1.5
1.0
0.5
0
0
5
10
15
20
25
30
FREQUENCY, HERTZ

FIGURE 3.7A-8

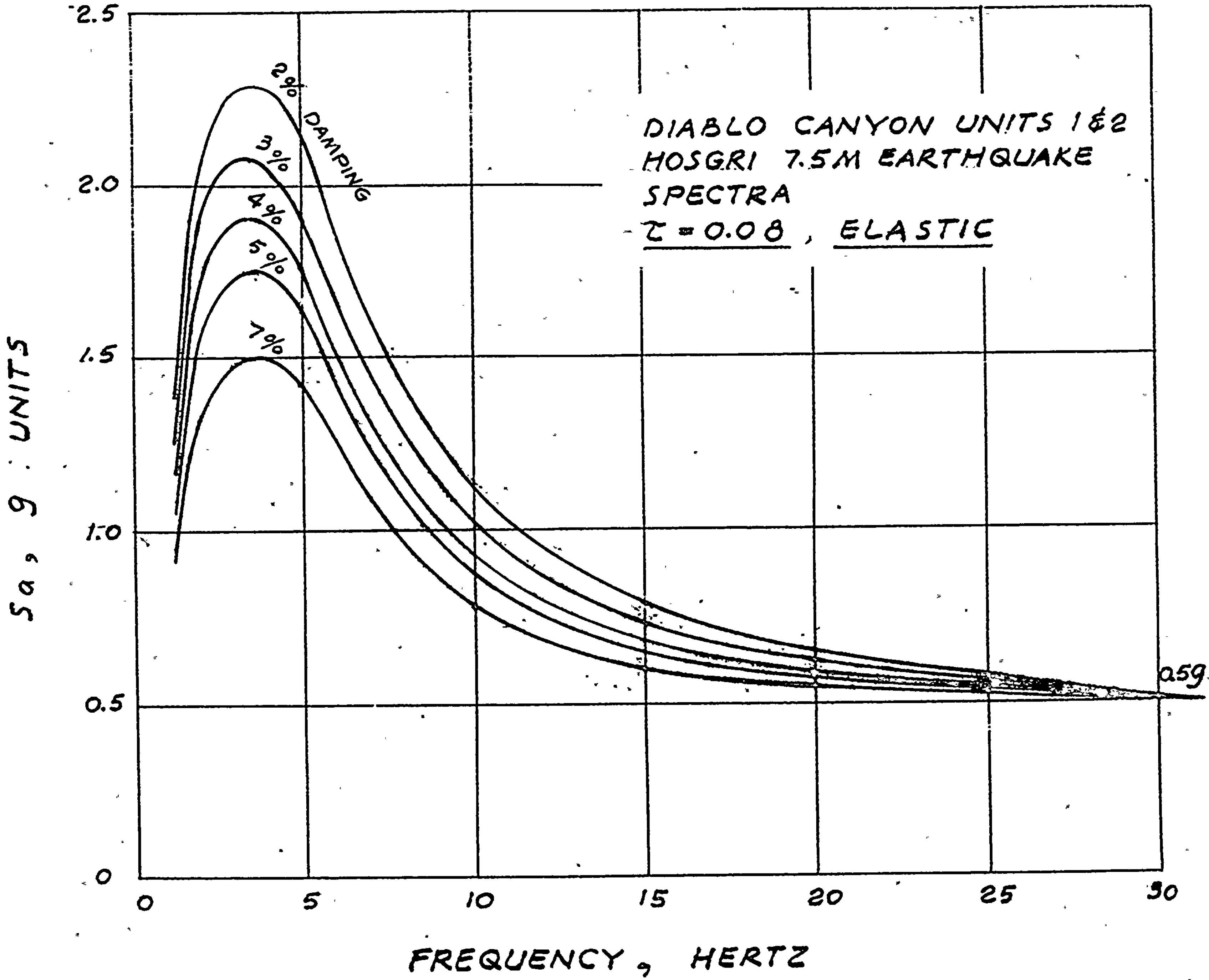
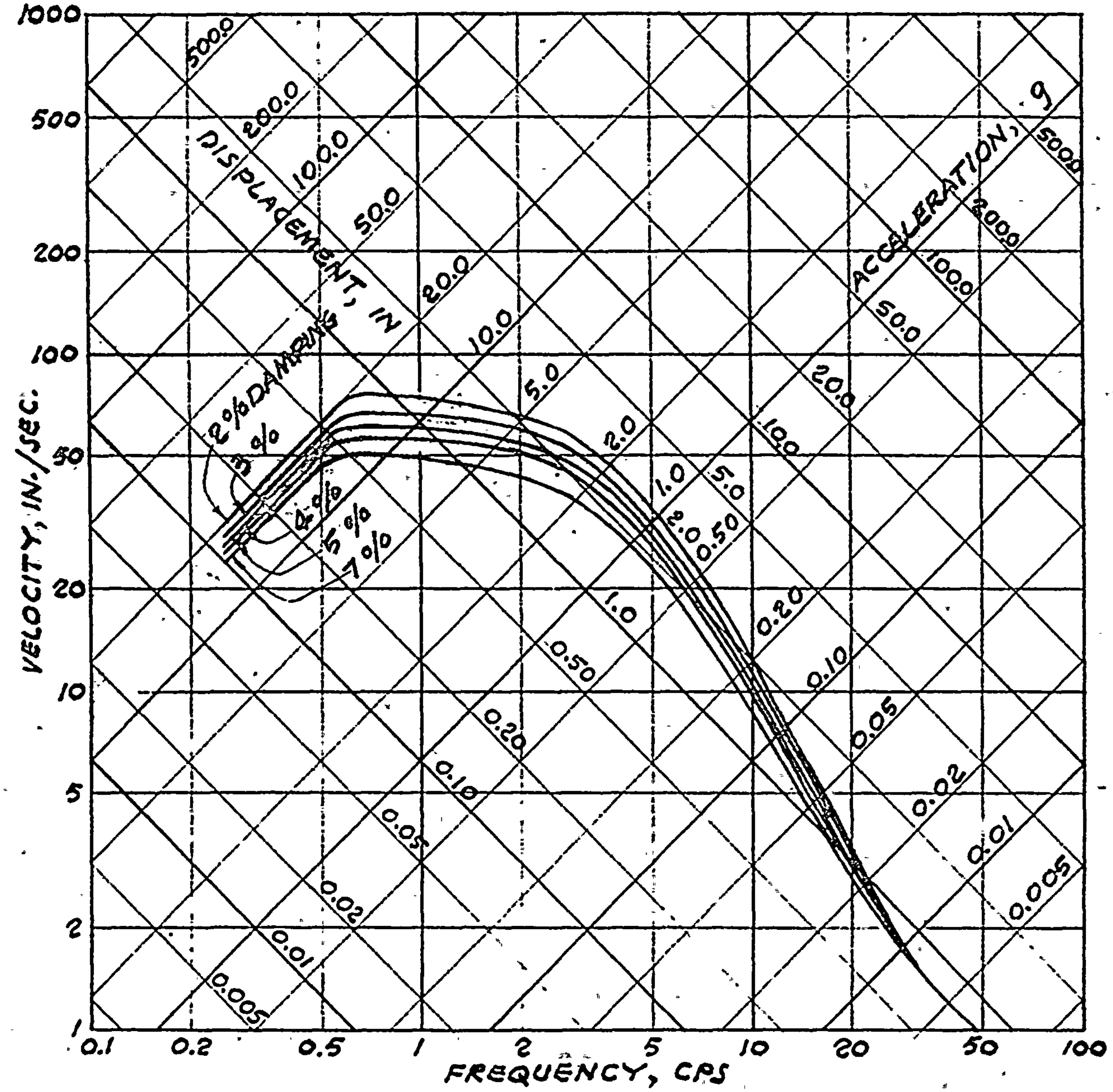
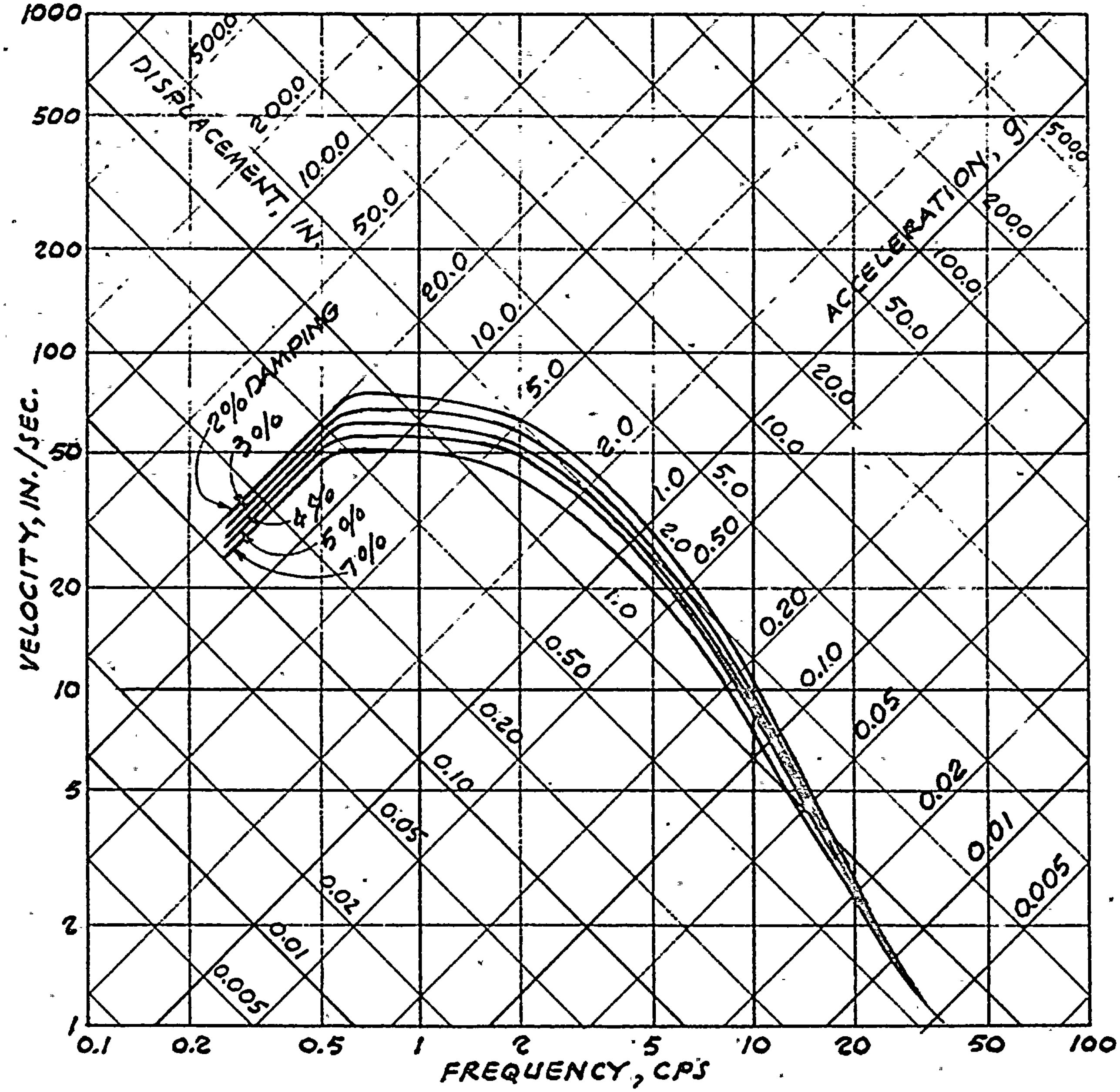

DIABLO CANYON UNITS 1&2
HOSGRI 7.5M EARTHQUAKE
SPECTRA
τ = 0.08 , ELASTIC
2% DAMPING
3%
4%
5%
7%
0.59
Sa, g UNITS
FREQUENCY, HERTZ
2.5
2.0
1.5
1.0
0.5
0
0
5
10
15
20
25
30

FIGURE 3.7A-9



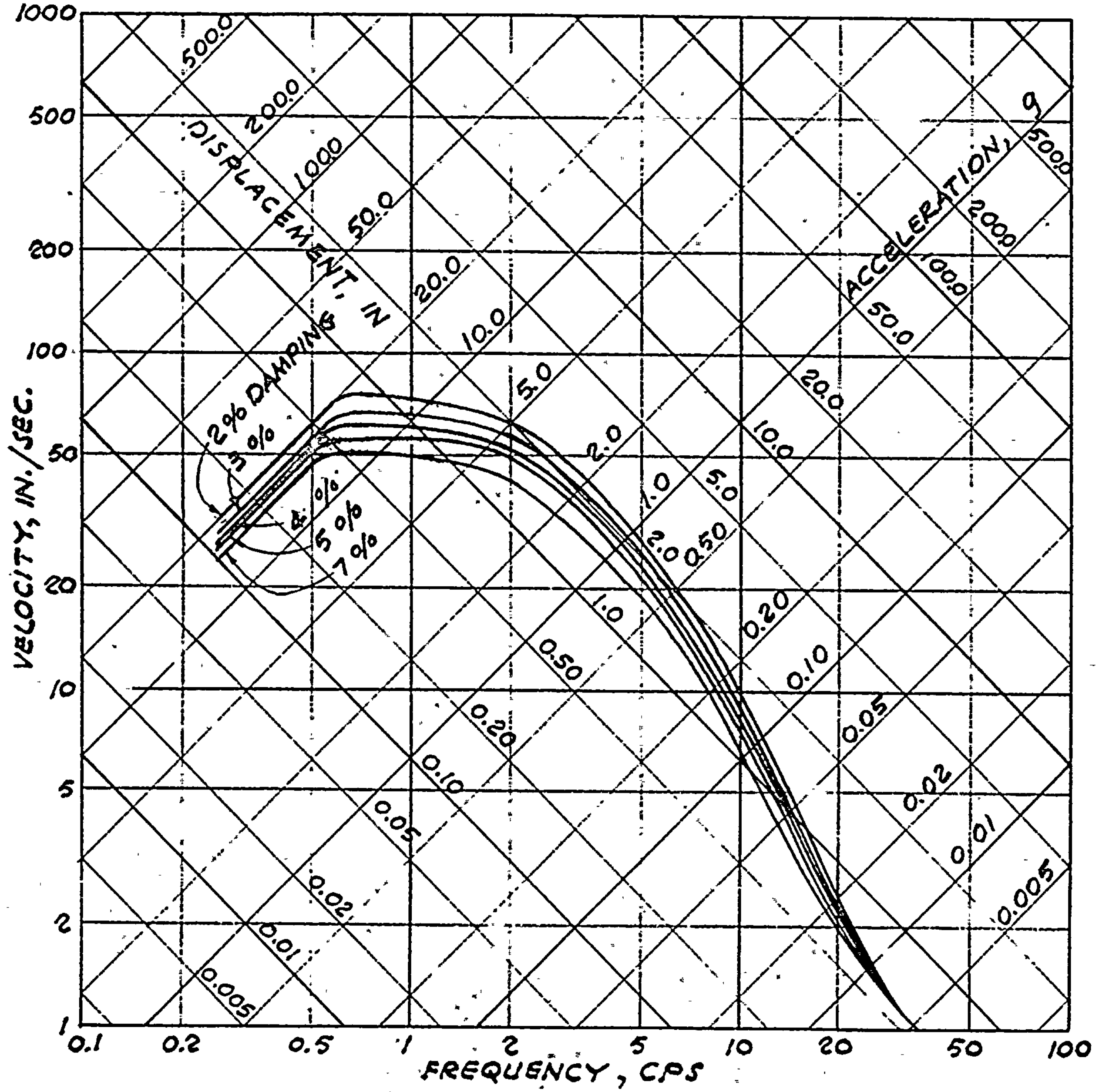
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE SPECTRA
τ=0 , ELASTIC

FIGURE 3. 10



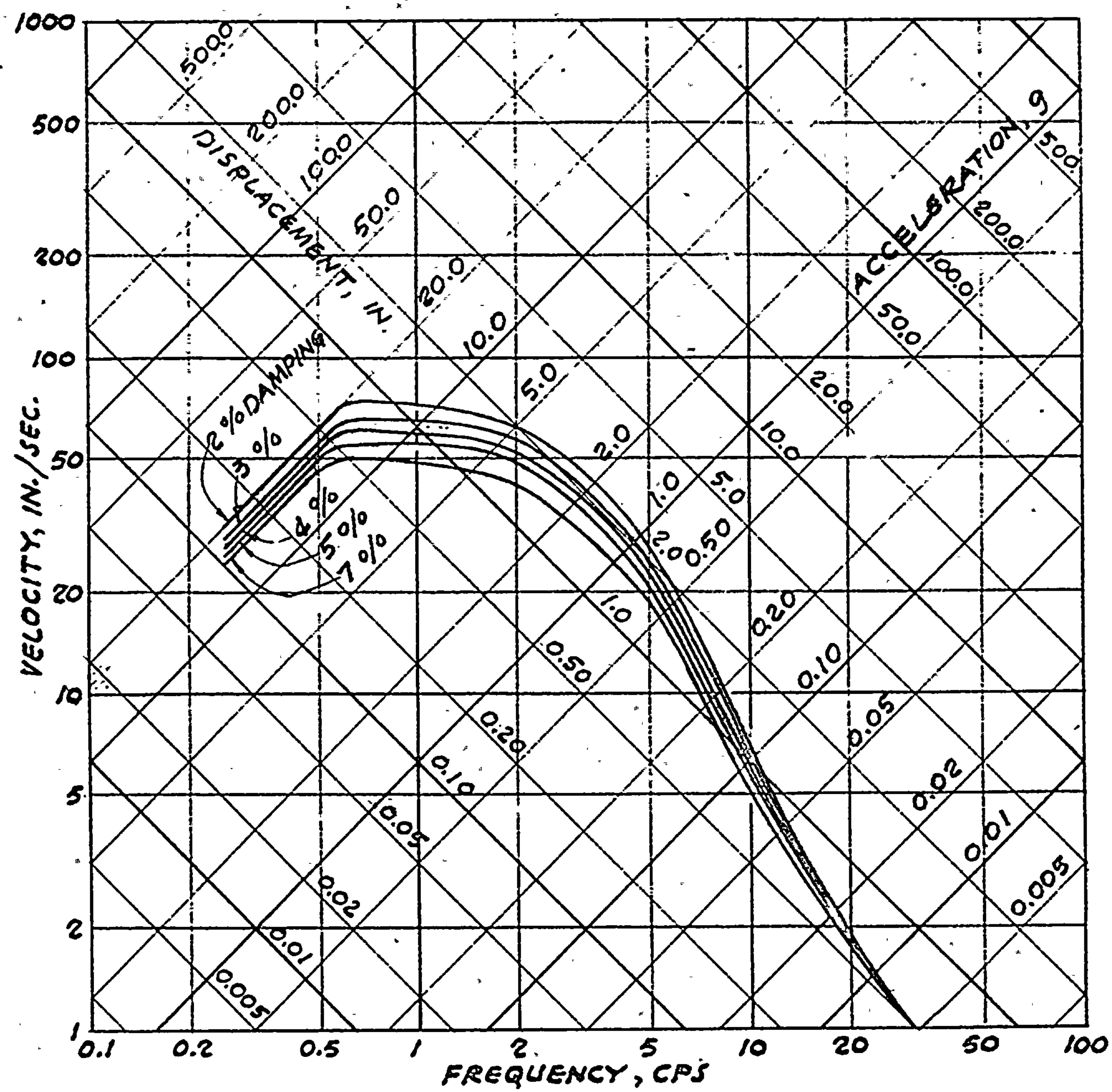
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE SPECTRA
$\tau = 0.04$, ELASTIC

FIGURE 3.7A-11



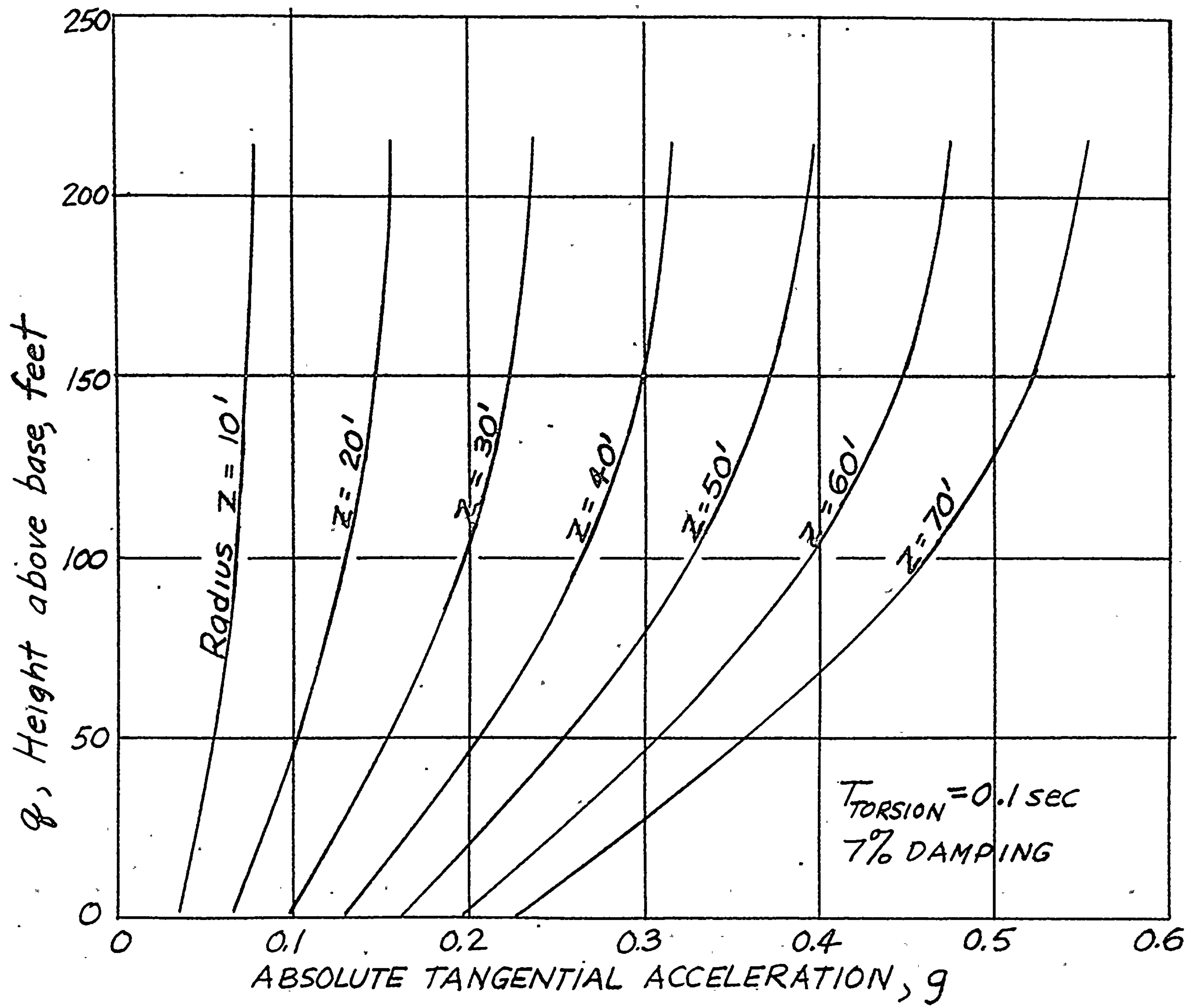
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE SPECTRA
$\tau = 0.06$, ELASTIC

FIGURE 3.[illegible]12



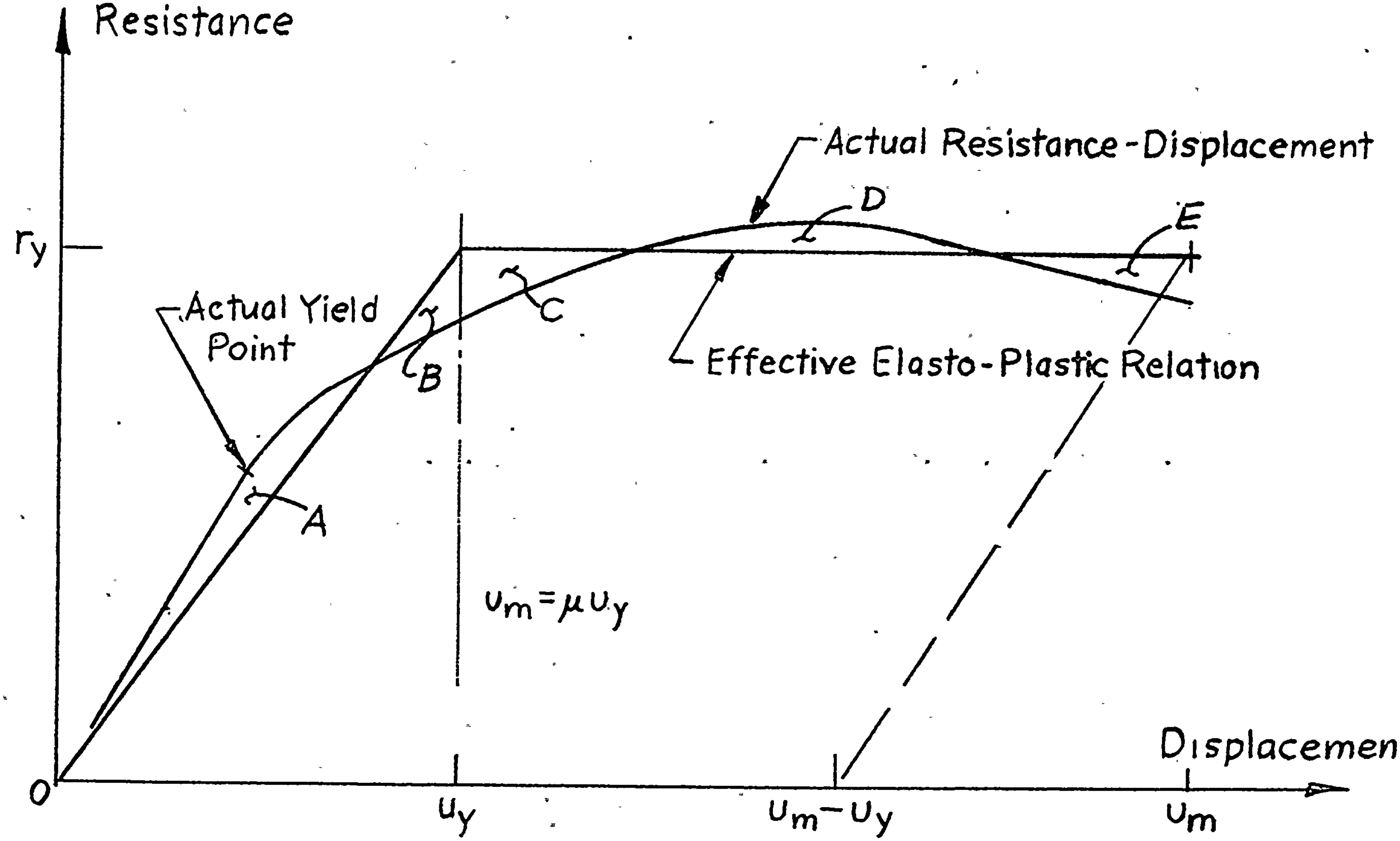
DIABLO CANYON UNITS 1 & 2
HOSGRI 7.5M EARTHQUAKE SPECTRA
τ=0.08 , ELASTIC

FIGURE 3.7A-13



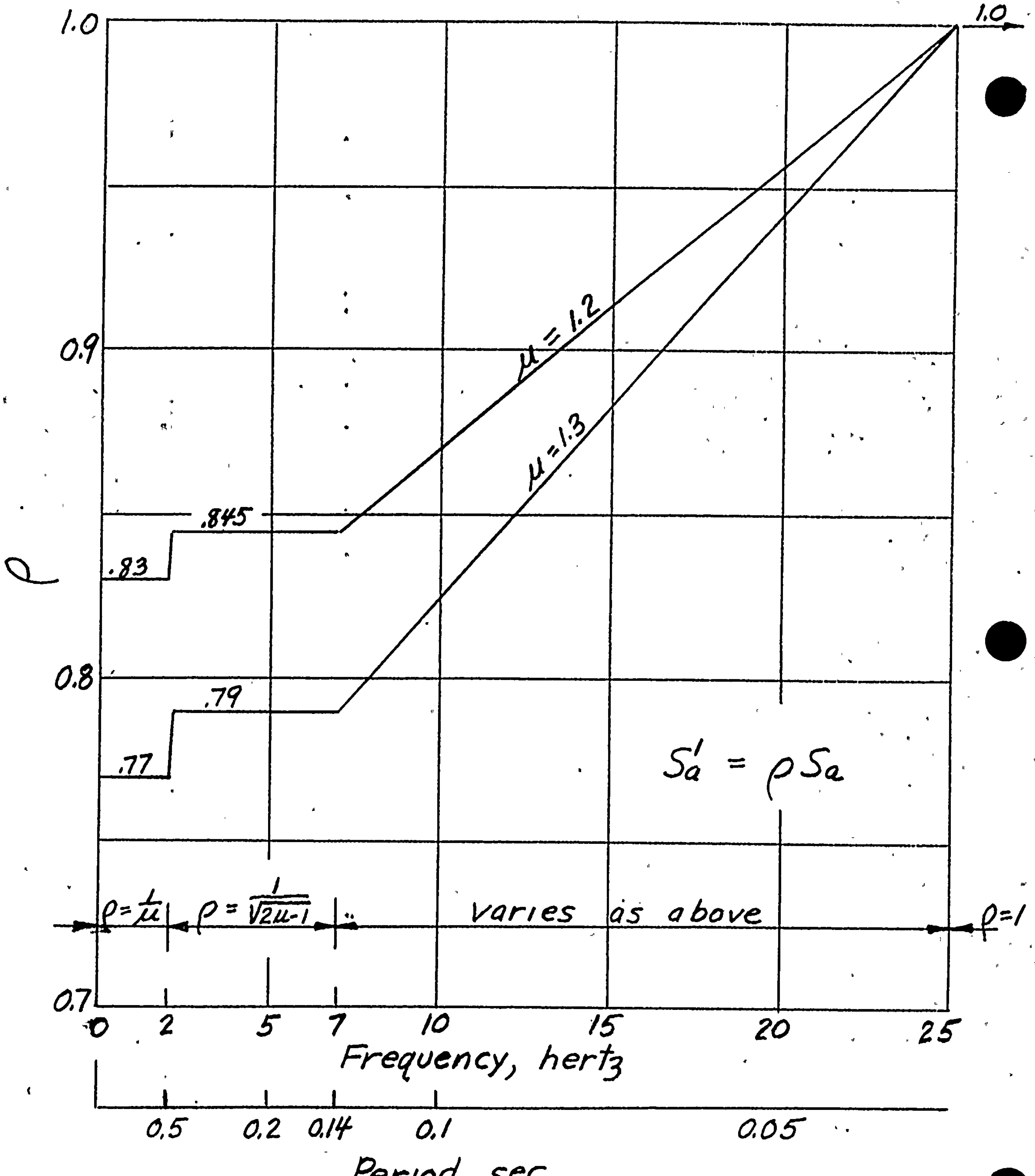
CONTAINMENT STRUCTURE TORSIONAL MOTION

FIGURE 3.[illegible]-14



RESISTANCE-DISPLACEMENT RELATIONSHIP

FIGURE 3.7A-15



DUCTILITY REDUCTION FACTOR

(July 1976)

FIGURE 3.7A-16

TABLE 3.7-1

CONTAINMENT STRUCTURE
PERIODS OF VIBRATION

Mode No.	Period, T in sec.	
1	0.255	3.92
2	0.093	10.7
3	0.088	11.4
4	0.073	13.7
5	0.060	16.7
6	0.058	17.2
7	0.057	17.5
8	0.051	19.6
9	0.051	19.6

TABLE 3.7-2

CONTAINMENT STRUCTURE

MAXIMUM ABSOLUTE ACCELERATIONS

Structure	Nodal Point*	Elevation, Ft.	Maximum Absolute Acceleration, g DE Analysis	DDE Analysis
Exterior Structure	2	301.64	1.275	2.083
	8	274.37	1.032	1.736
	10	258.27	0.907	1.567
	14	231.00	0.743	1.177
	17	205.58	0.837	1.358
	23	181.08	0.911	1.369
	26	155.83	0.866	1.292
	34	130.58	0.713	1.080
	37	109.67	0.492	0.793
Interior Structure	19-22	140.00	0.735	1.195
	24	127.00	0.597	0.982
	27-30	114.00	0.478	0.773
	32	110.00	0.455	0.726
	38	102.00	0.384	0.601
Base Slab	47-58	88.58	0.291	0.483

*See Figure 3.7-5.

TABLE 3.7-3

CONTAINMENT STRUCTURE

MAXIMUM DISPLACEMENTS

Structure	Nodal Point*	Elevation, Ft.	Maximum Displacements, inches DE Analysis	DDE Analysis
Exterior	2	301.64	0.666	1.063
Structure	8	274.37	0.602	0.967
	10	258.27	0.562	0.911
	14	231.00	0.480	0.807
	17	205.58	0.389	0.695
	23	181.08	0.314	0.587
	26	155.83	0.248	0.459
	34	130.58	0.180	0.327
	37	109.67	0.115	0.212
Interior	19-22	140.00	0.083	0.139
Structure	24	127.00	0.069	0.114
	27-30	114.00	0.056	0.090
	32	110.00	0.053	0.084
	38	102.00	0.043	0.068
Base Slab	47-58	88.58	0.030	0.050

*See Figure 3.7-5.

TABLE 3.7-4

CONTAINMENT STRUCTURE

MAXIMUM SHELL FORCES AND MOMENTS* - DE ANALYSIS

Nodal Point**	Elevation, Ft.	Shell Moments, kip-ft/ft			Shell Forces, kips/ft		
		M_{SS}	M_{TT}	M_{ST}	F_{SS}	F_{TT}	F_{ST}
2	301.64	0.21	0.21	28.99	2.74	3.84	3.75
8	274.37	0.33	0.44	2.96	14.47	32.07	23.85
10	258.27	1.76	0.91	1.63	21.04	40.91	32.80
14	231.00	9.17	2.94	0.36	37.68	42.73	48.97
17	205.58	5.74	1.26	0.27	63.59	33.27	66.44
23	181.08	7.58	2.54	0.31	91.25	37.79	79.59
26	155.83	5.69	1.49	0.50	110.72	36.31	91.43
34	130.58	4.31	1.01	0.27	151.69	31.20	108.65
37	109.67	8.26	2.75	0.19	174.13	18.99	122.66
57	88.58	1.01	0.14	2.23	209.79	63.73	127.22

*See Figure 3.7-7.

**See Figure 3.7-5.

TABLE 3.7-5

CONTAINMENT STRUCTURE

MAXIMUM SHELL FORCES AND MOMENTS* - DDE ANALYSIS

Nodal Point**	Elevation, Ft.	Shell Moments, kip-ft/ft			Shell Forces, kips/ft		
		M_{SS}	M_{TT}	M_{ST}	F_{SS}	F_{TT}	F_{ST}
2	301.64	0.36	0.37	47.17	4.30	6.33	6.04
8	274.37	0.62	0.76	4.77	22.00	53.37	39.37
10	258.27	2.71	1.46	2.63	32.58	67.71	54.58
14	231.00	15.29	4.92	0.50	60.01	71.93	83.06
17	205.58	8.14	1.64	0.37	103.39	53.31	110.30
23	181.08	11.39	3.96	0.45	154.79	56.72	132.95
26	155.83	8.27	2.21	0.77	190.50	54.24	162.53
34	130.58	6.07	1.36	0.42	251.35	46.24	195.36
37	109.67	15.95	5.31	0.34	282.88	30.75	217.34
57	88.58	1.74	0.23	4.18	338.73	110.90	220.62

*See Figure 3.7-7.

**See Figure 3.7-5.

TABLE 3.7-6

CONTAINMENT STRUCTURE

MAXIMUM TOTAL SHEARS

Structure	Associated Nodal Points*	Elevation, Ft.	Maximum Shears, kips x 10^3 DE Analysis	Maximum Shears, kips x 10^3 DDE Analysis
Exterior	2	301.64	0.19	0.66
Structure	8	274.37	5.81	9.38
	10	258.27	8.49	13.91
	14	231.00	11.39	19.55
	17	205.58	15.00	25.02
	23	181.08	17.95	29.98
	26	155.83	20.63	36.66
	34	130.58	24.53	44.18
	37	109.67	27.83	49.42
	57	88.58	29.55	51.39
Interior	19 & 22	140.00	8.06	13.23
Structure	27 & 30	114.00	10.27	16.87
	49 & 54	88.58	18.85	30.96
Total Base Shear	49, 54 & 57	88.58	35.05	59.99

*See Figure 3.7-5.

TABLE 3.7-7

CONTAINMENT STRUCTURE

MAXIMUM TOTAL OVERTURNING MOMENTS

Structure	Associated Nodal Points*	Elevation, Ft.	Maximum Overturning Moment, kip-ft x 10^6 DE Analysis	DDE Analysis
Exterior Structure	2	301.64	0.00	0.00
	8	274.37	0.12	0.18
	10	258.27	0.27	0.41
	14	231.00	0.61	0.97
	17	205.58	1.03	1.67
	23	181.08	1.48	2.50
	26	155.83	1.79	3.08
	34	130.58	2.45	4.07
	37	109.67	2.82	4.58
	57	88.58	3.39	5.48
Interior Structure	19 & 22	140.00	0.06	0.10
	27 & 30	114.00	0.20	0.33
	49 & 54	88.58	0.76	1.24
Total O.T.M. at Base	49, 54 & 57	88.58	3.48	5.62

*See Figure 3.7-5.

TABLE 3.7-8

AUXILIARY BUILDING
PERIODS OF VIBRATION

Mode No.	N-S Direction Period, T in sec.	E-W Direction Period, T in sec.
1	0.454 2.2	0.516 1.9
2	0.355 2.8	0.105 9.5
3	0.106 9.4	0.053 18.9
4	0.088 11.4	0.035 28.6
5	0.049 20.2	--

Note: The first five modes of vibration in the N-S direction and first four modes of vibration in the E-W direction were considered in the analysis. The higher modes, not considered in the analysis, are insignificant because the periods of vibration corresponding to these modes are in the rigid category.

TABLE 3.7-9

AUXILIARY BUILDING

MAXIMUM ABSOLUTE ACCELERATIONS--DE ANALYSIS

Mass Point*	Elevation, Ft.	Maximum Absolute Accelerations: N-S Direction: Horizontal Acceleration, g	N-S Direction: Rotational Acceleration, rad/sec^2	E-W Direction: Horizontal Acceleration, g
6	188.00	0.69	0.009	0.55
1	163.00	0.98	0.090	1.20
2	140.00	0.58	0.079	0.80
3	115.00	0.42	0.054	0.54
4	100.00	0.31	0.041	0.37
5	85.00	0.27	0.022	0.27

*See Figure 3.7-13

TABLE 3.7-10

AUXILIARY BUILDING

MAXIMUM RELATIVE DISPLACEMENTS--DE ANALYSIS

		Maximum Relative Displacements		
		N-S Direction		E-W Direction
Mass Point*	Elevation, Ft.	Translation, in.	Rotation, radians x 10^{-5}	Translation, in.
6	188.00	1.344	0.907	1.378
1	163.00	0.112	2.467	0.138
2	140.00	0.061	2.213	0.086
3	115.00	0.038	1.548	0.052
4	100.00	0.022	1.182	0.030
5	85.00	0.014	0.560	0.014

*See Figure 3.7-13.

TABLE 3.7-11

AUXILIARY BUILDING

MAXIMUM STORY SHEARS--DE ANALYSIS

Member*	Maximum Story Shears, kips x 10^3 N-S Direction	Maximum Story Shears, kips x 10^3 E-W Direction
5	1.75	1.39
1	10.36	13.53
2	45.35	57.59
3	71.42	92.41
4	49.42	48.28

*See Figure 3.7-13

TABLE 3.7-12

AUXILIARY BUILDING
MAXIMUM OVERTURNING MOMENTS--DE ANALYSIS

Member*	Maximum O.T. Moments, kip-ft x 10^6 N-S Direction	E-W Direction
5	0.084	0.067
1	0.243	0.318
2 (top)	0.236	0.323
2 (bottom)	1.370	1.763
3	2.441	3.149
4	3.183	3.873

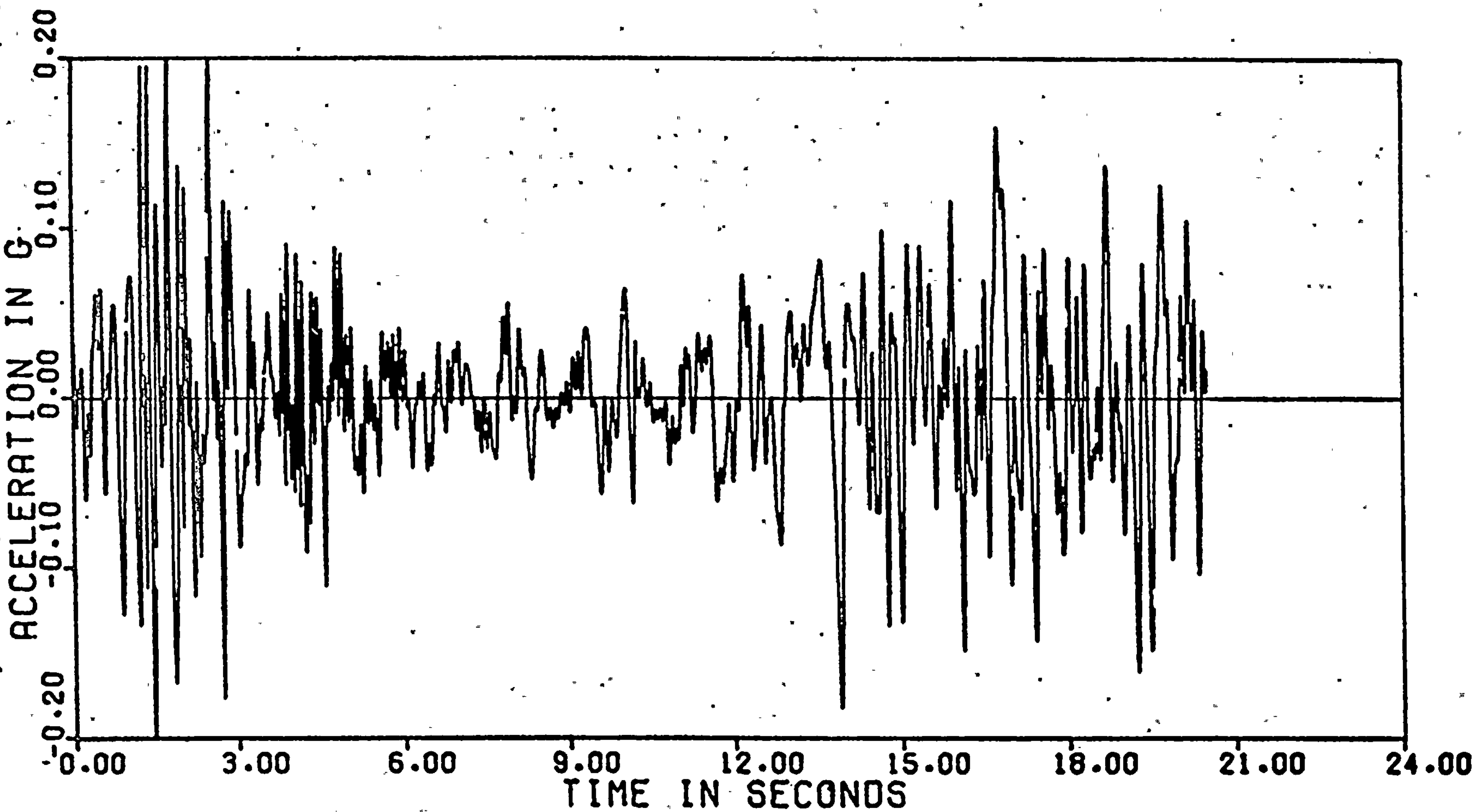
*See Figure 3.7-13

TABLE 3.7-13

AUXILIARY BUILDING
MAXIMUM TORSIONAL MOMENTS DUE TO
EARTHQUAKE IN N-S DIRECTION--DE ANALYSIS

Member*	Maximum Torsional Moments, kip-ft x 10^5
5	0.086
1	0.869
2	40.891
3	63.017
4	48.944

*See Figure 3.7-13.

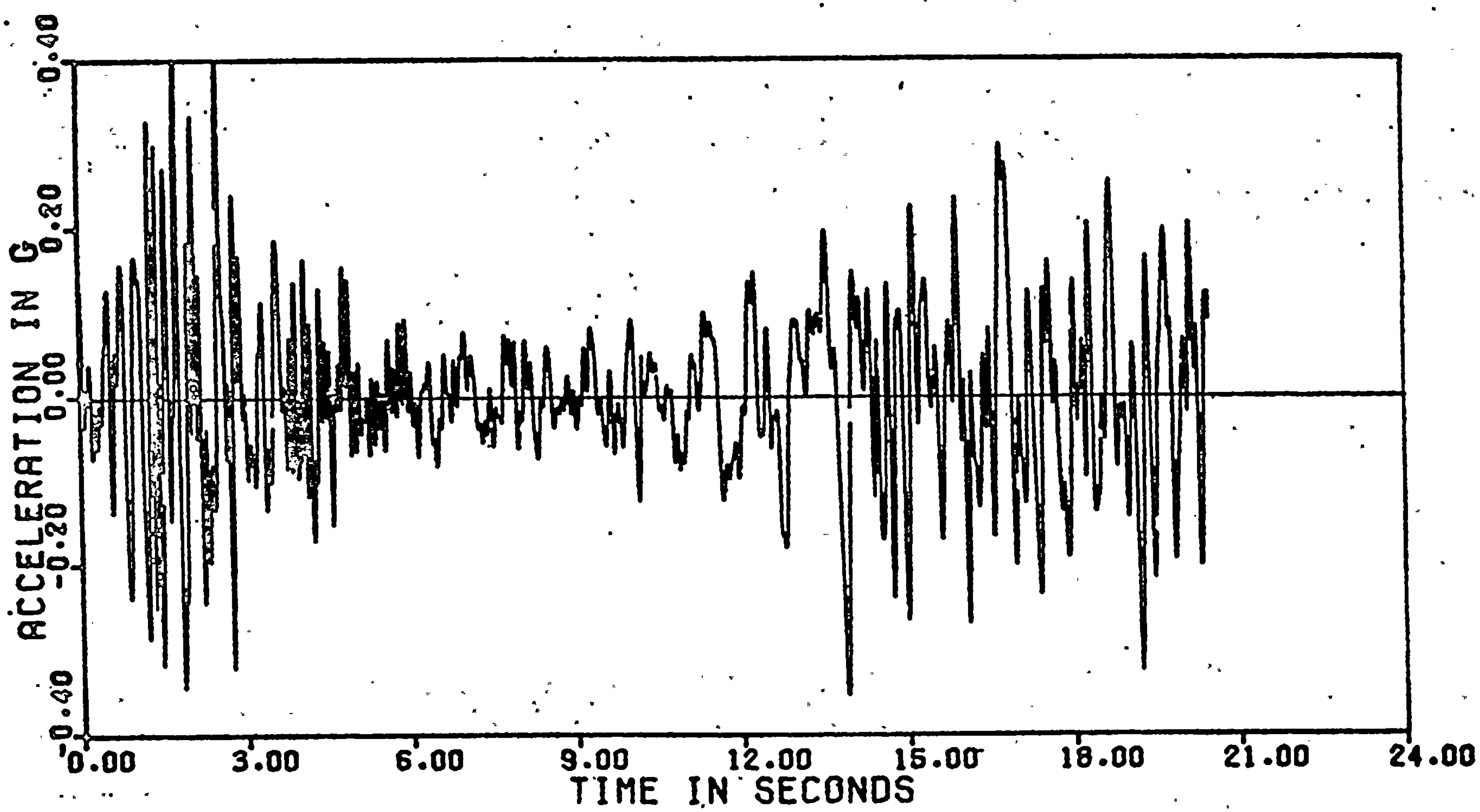


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-1

FREE FIELD GROUND MOTION

DE ANALYSIS

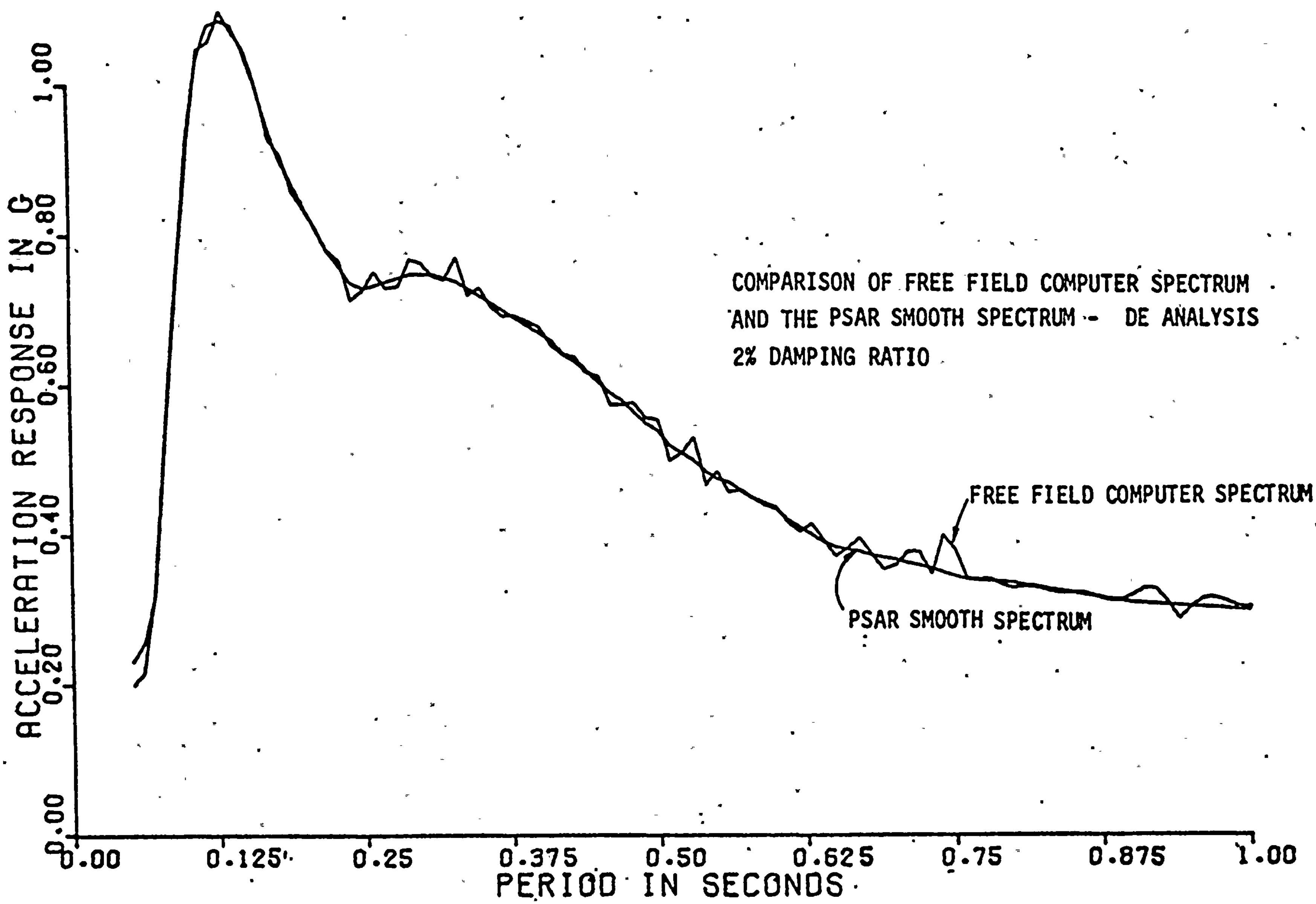


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-2

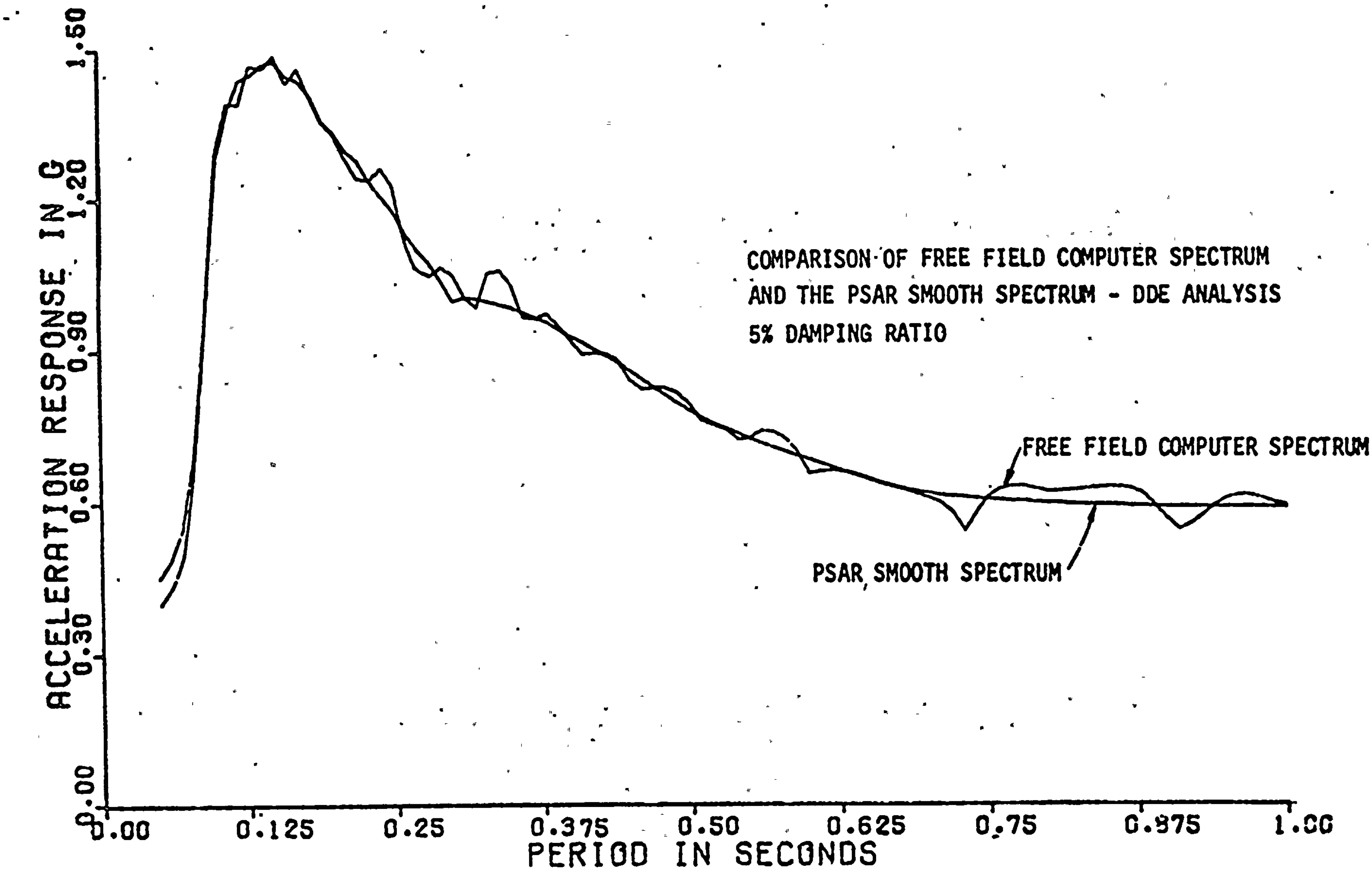
FREE FIELD GROUND MOTION

DDE ANALYSIS



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-3
COMPARISON OF SPECTRA
DE ANALYSIS

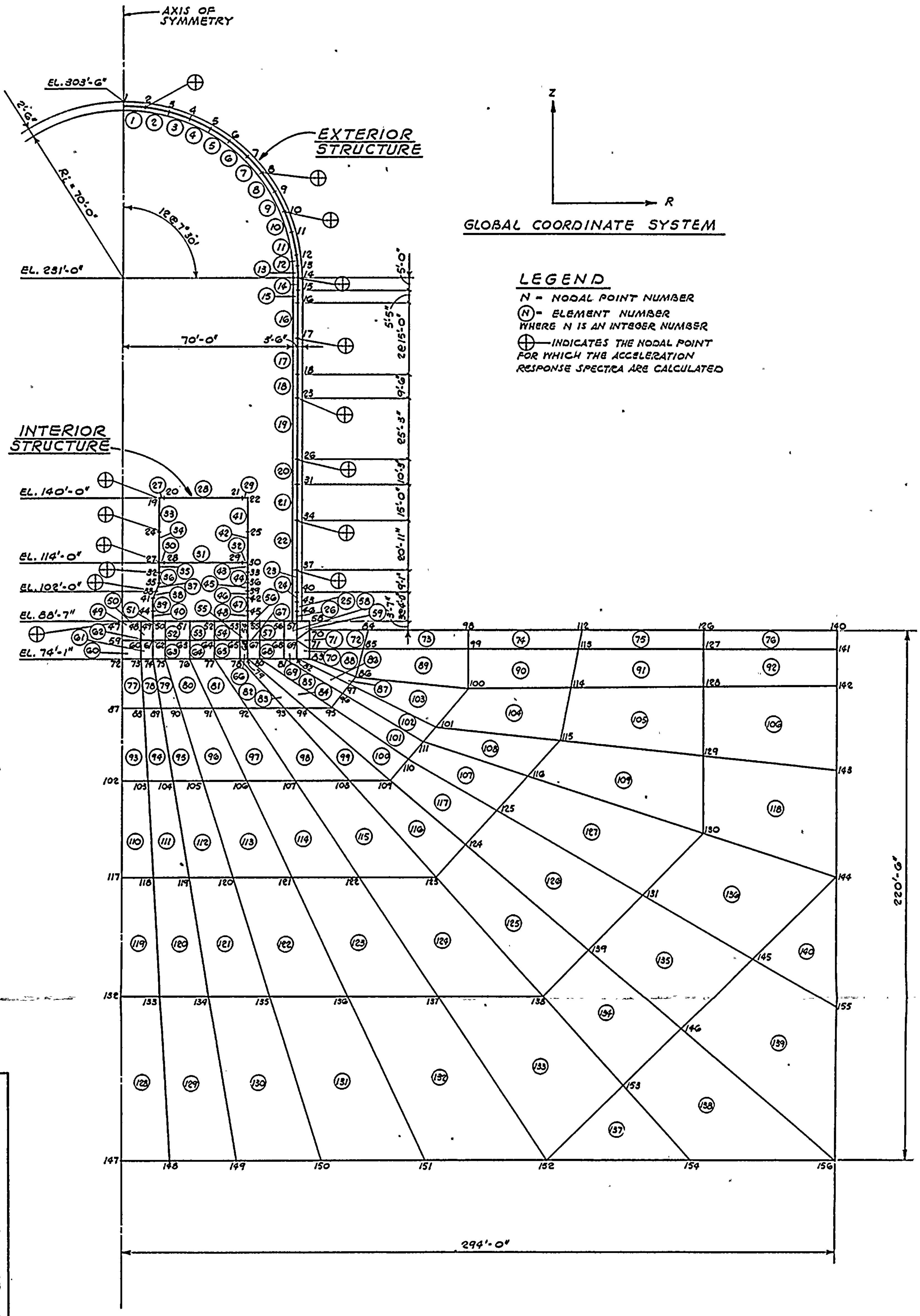


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-4

COMPARISON OF SPECTRA

DDE ANALYSIS

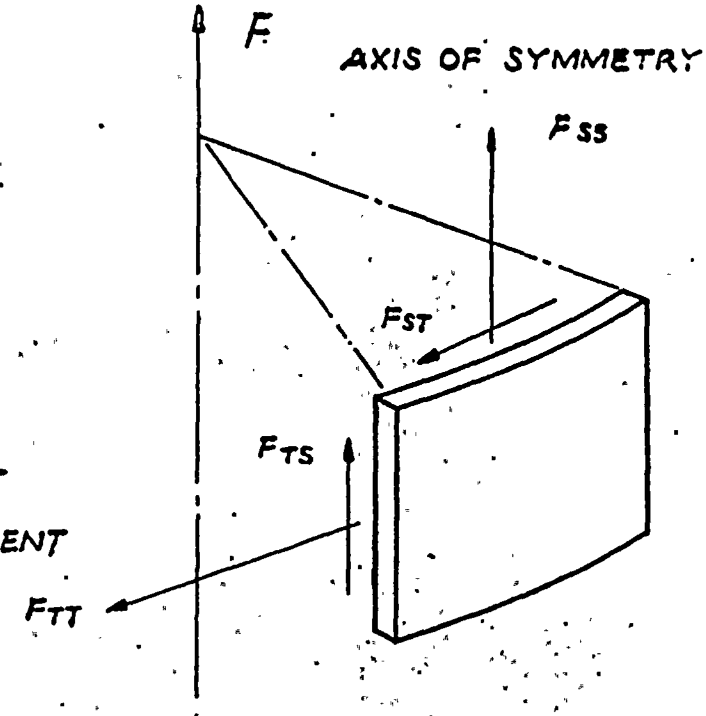


UNITS 1 AND 2
DIABLO CANYON SITE
FIGURE 3.7-5
CONTAINMENT STRUCTURE
FINITE ELEMENT MODEL

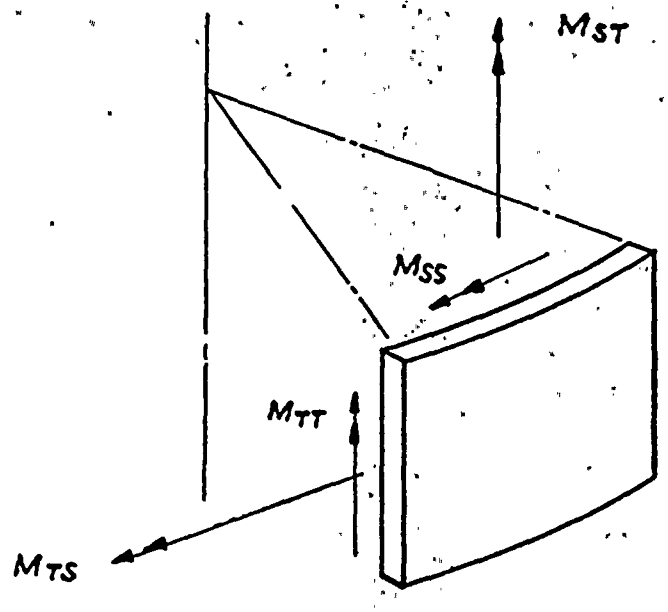
CONTAINMENT STRUCTURE
FINITE ELEMENT MODEL
SHELL FORCES AND MOMENTS

LEGEND:

- F_{SS} – LONGITUDINAL FORCE
- F_{TT} – HOOP FORCE
- F_{ST} – SHEAR FORCE
- M_{SS} – LONGITUDINAL MOMENT
- M_{TT} – CIRCUMFERENTIAL MOMENT
- M_{ST} – CROSS MOMENT

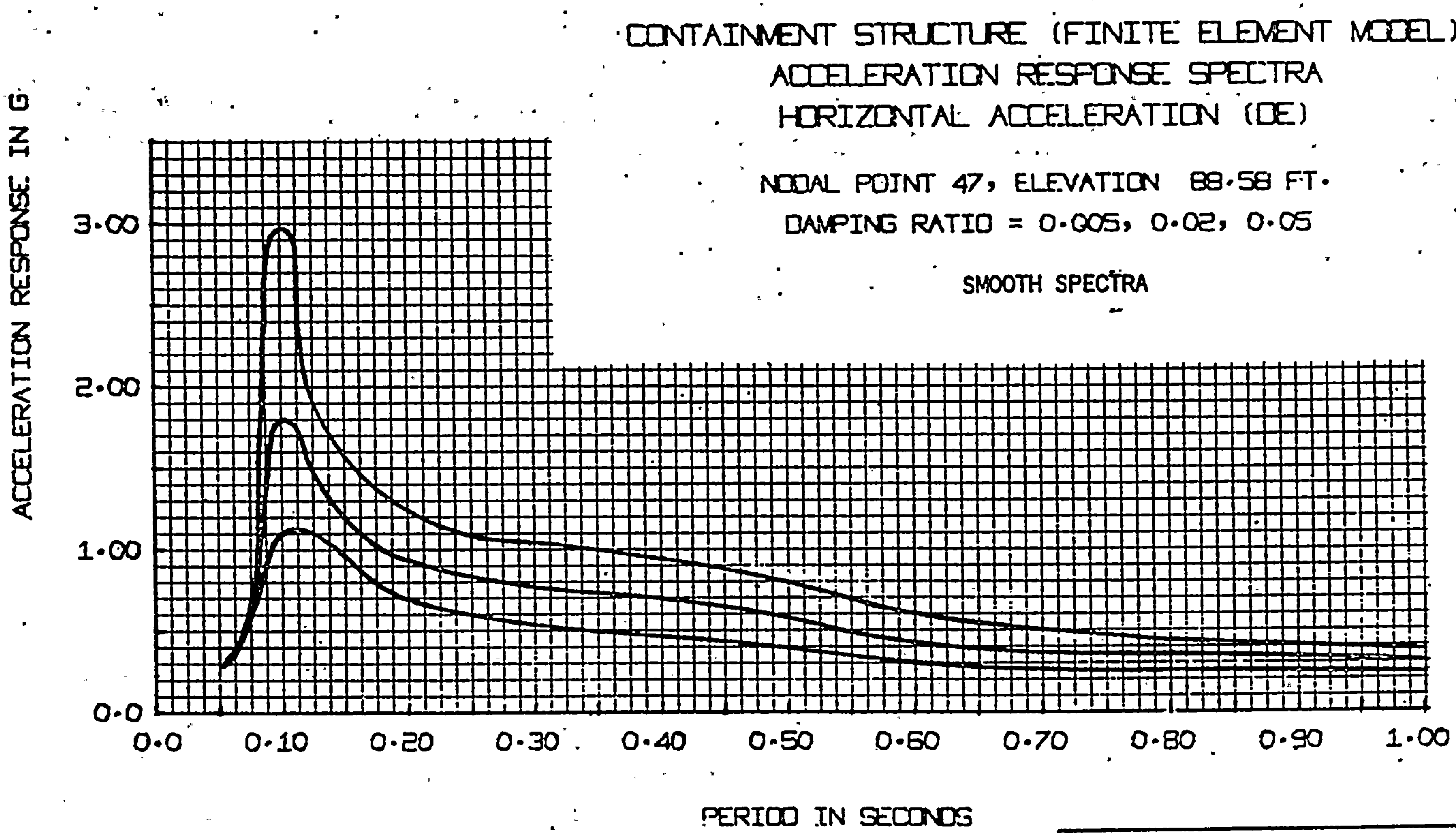


(a) SHELL FORCES



(b) SHELL MOMENTS

UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 3.7-7 CONTAINMENT STRUCTURE SHELL FORCES AND MOMENTS

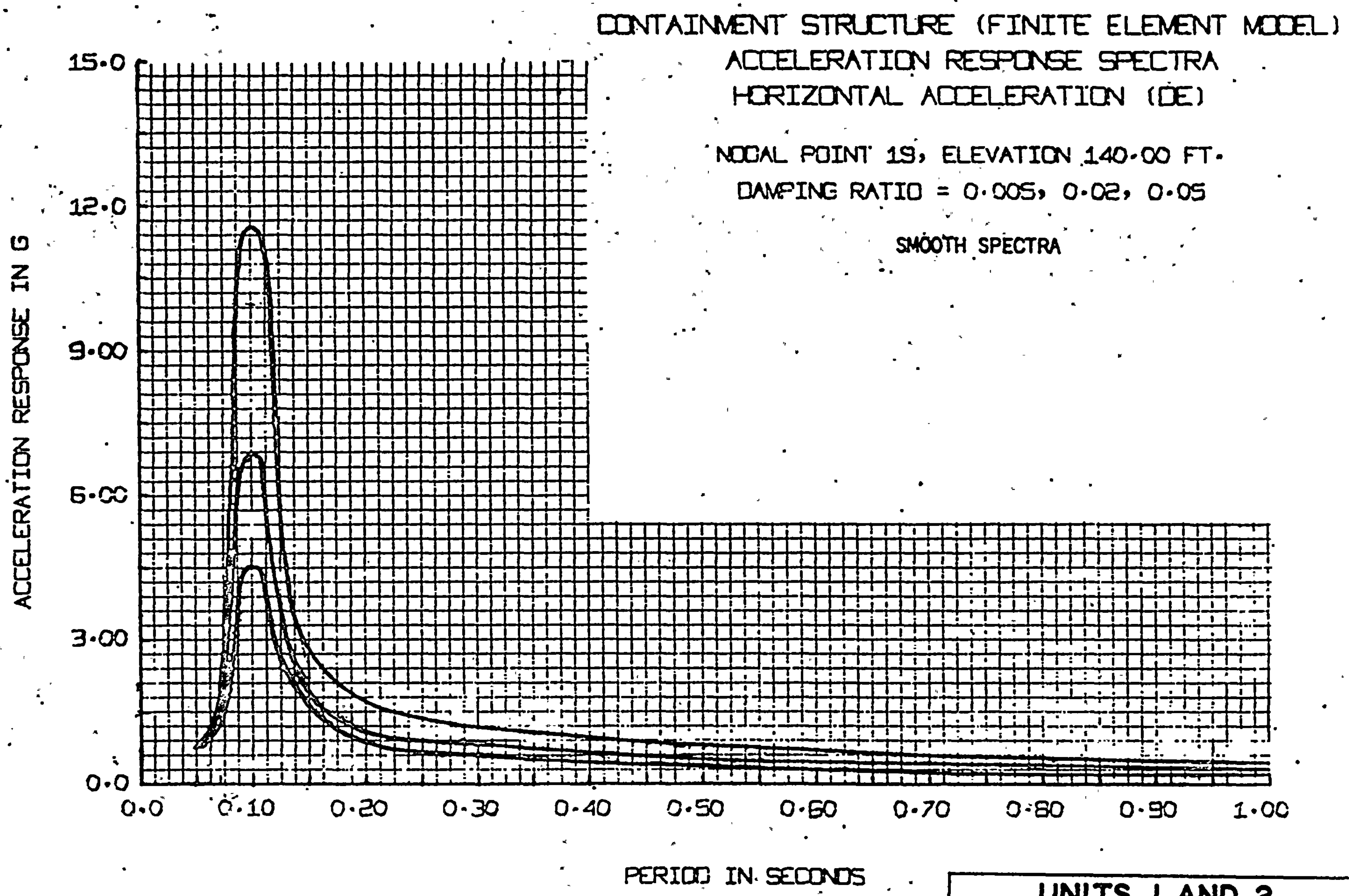


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-8

CONTAINMENT STRUCTURE

TYPICAL SPECTRA

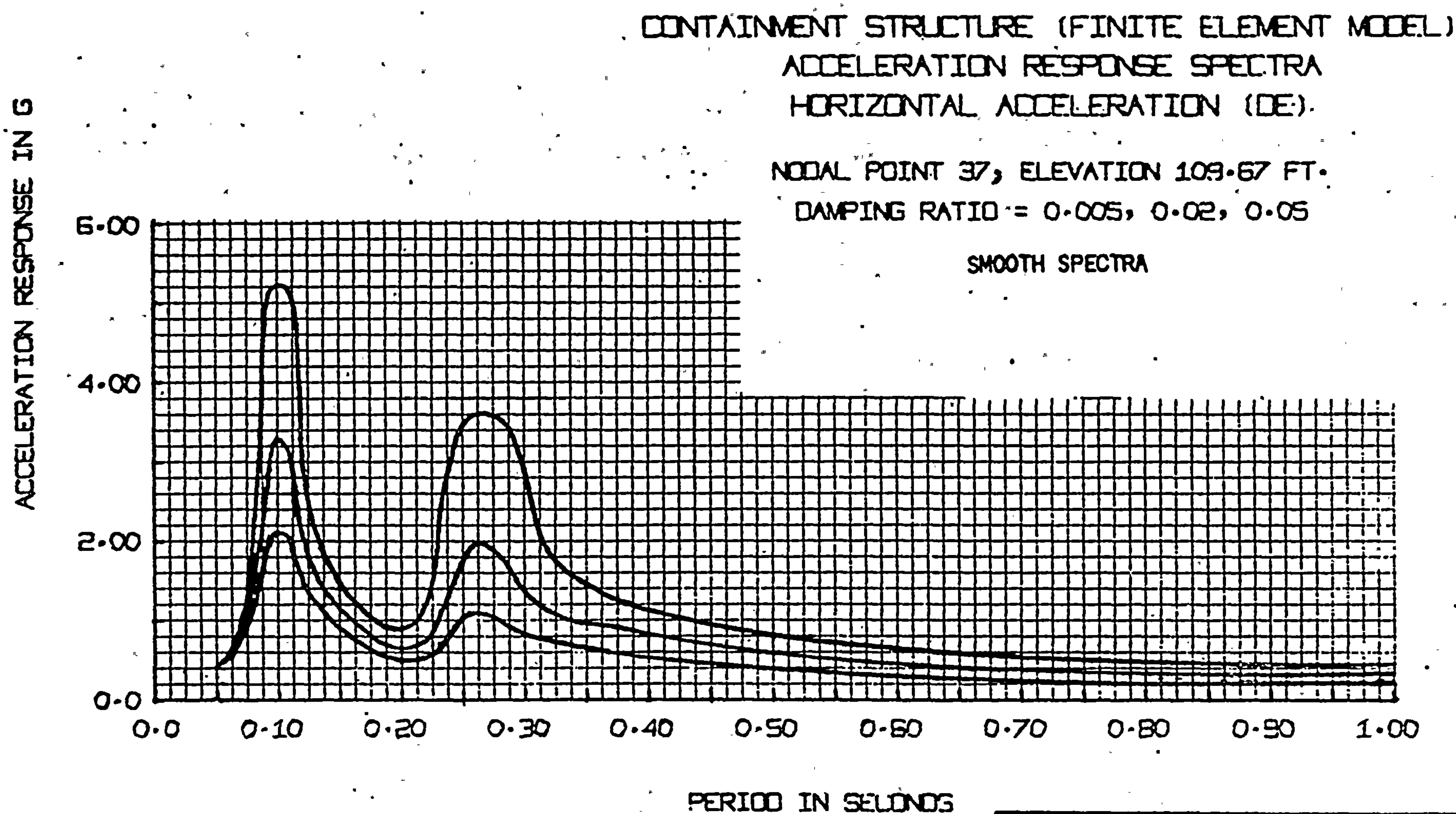


UNITS 1 AND 2 DIABLO CANYON SITE

FIGURE 3.7-9

CONTAINMENT STRUCTURE

TYPICAL SPECTRA

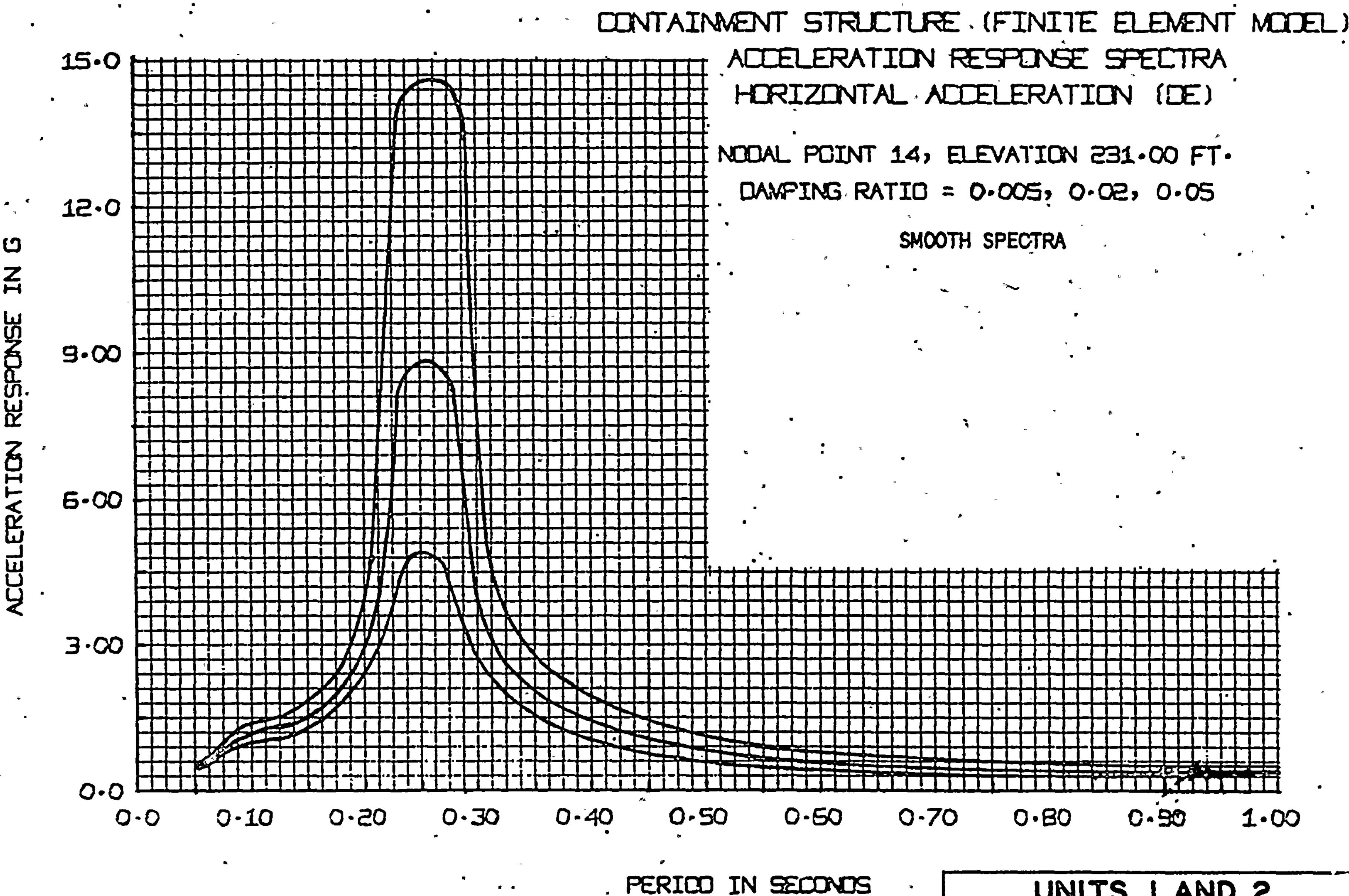


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-10

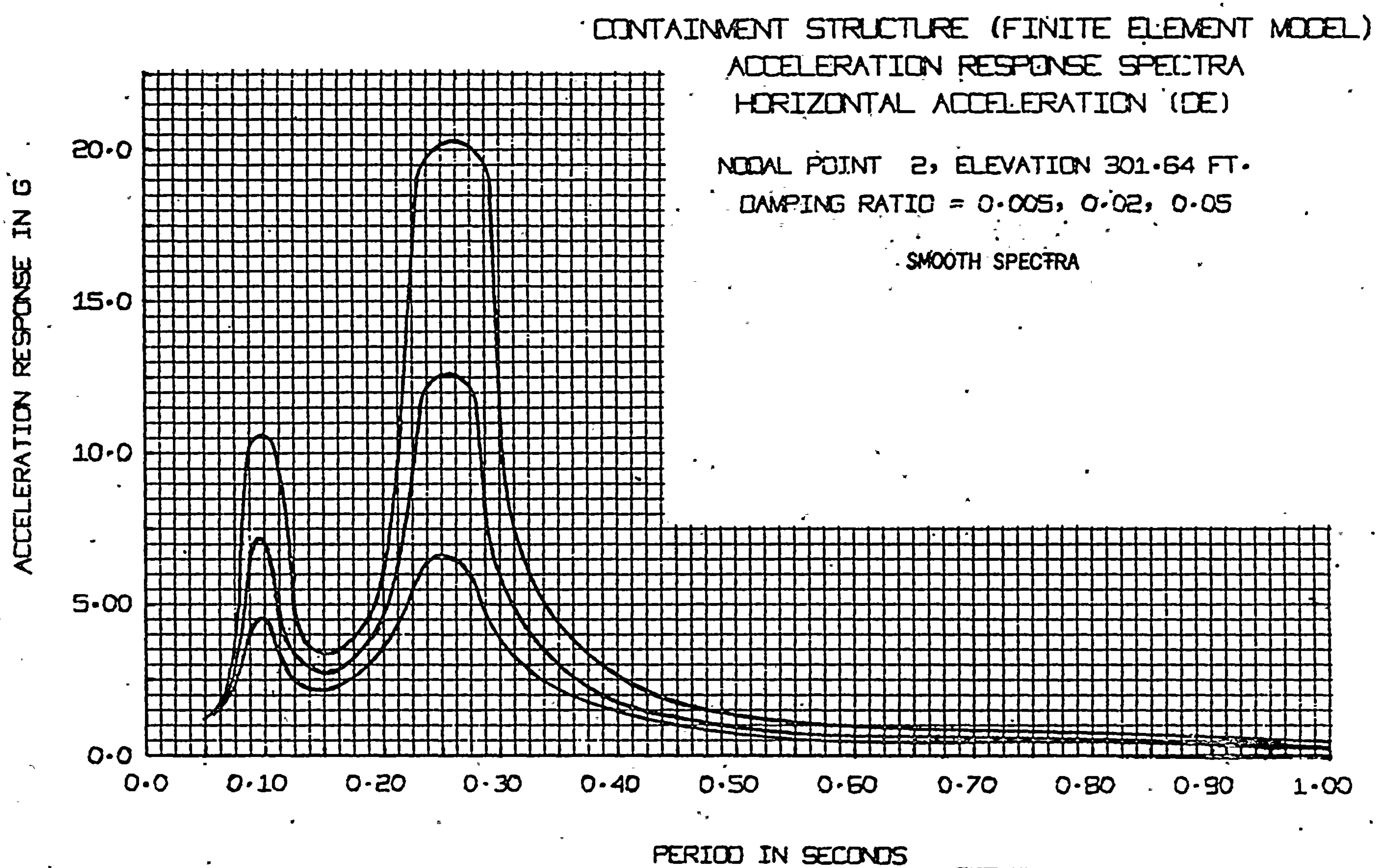
CONTAINMENT STRUCTURE

TYPICAL SPECTRA



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-11
CONTAINMENT STRUCTURE
TYPICAL SPECTRA

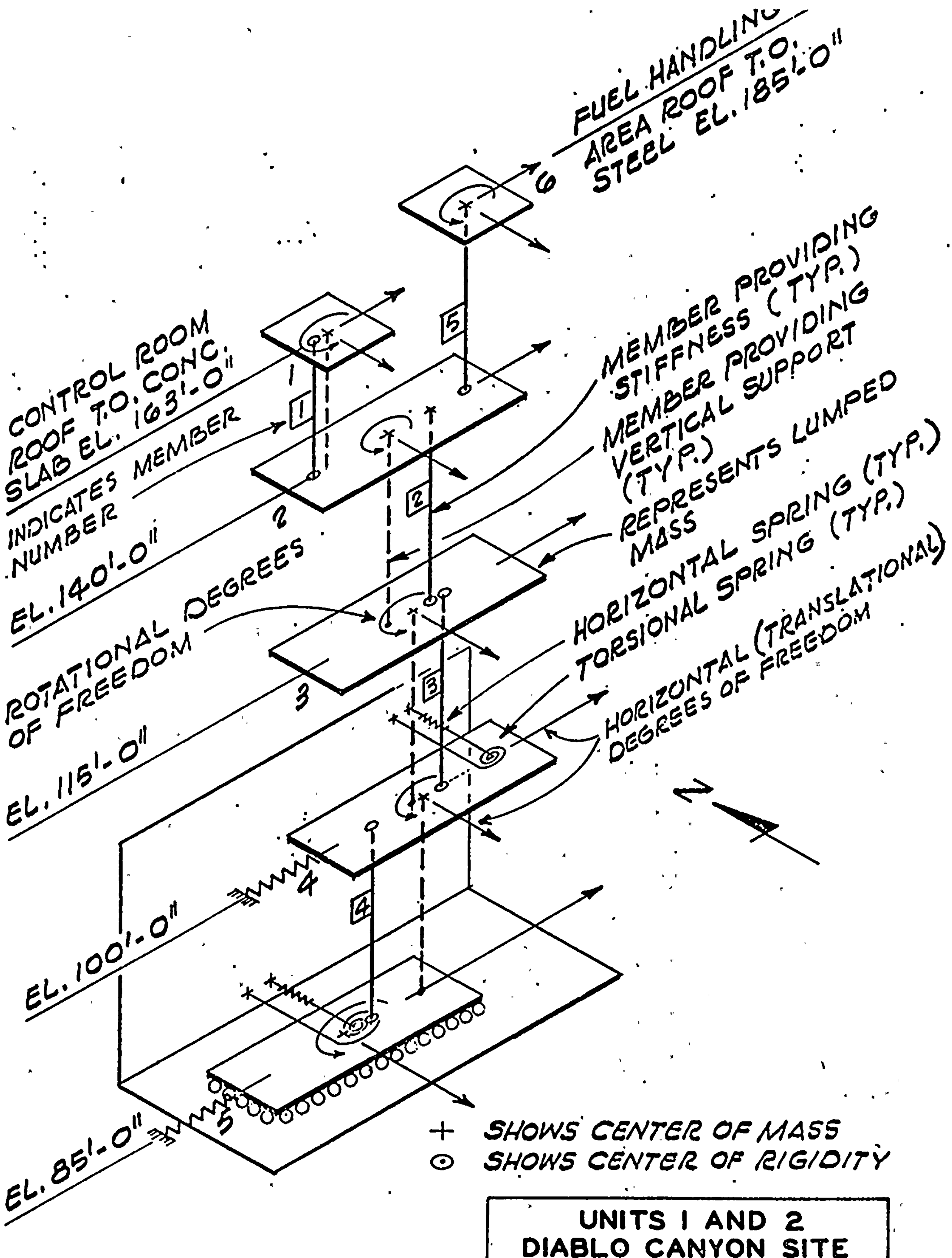


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-12

CONTAINMENT STRUCTURE

TYPICAL SPECTRA

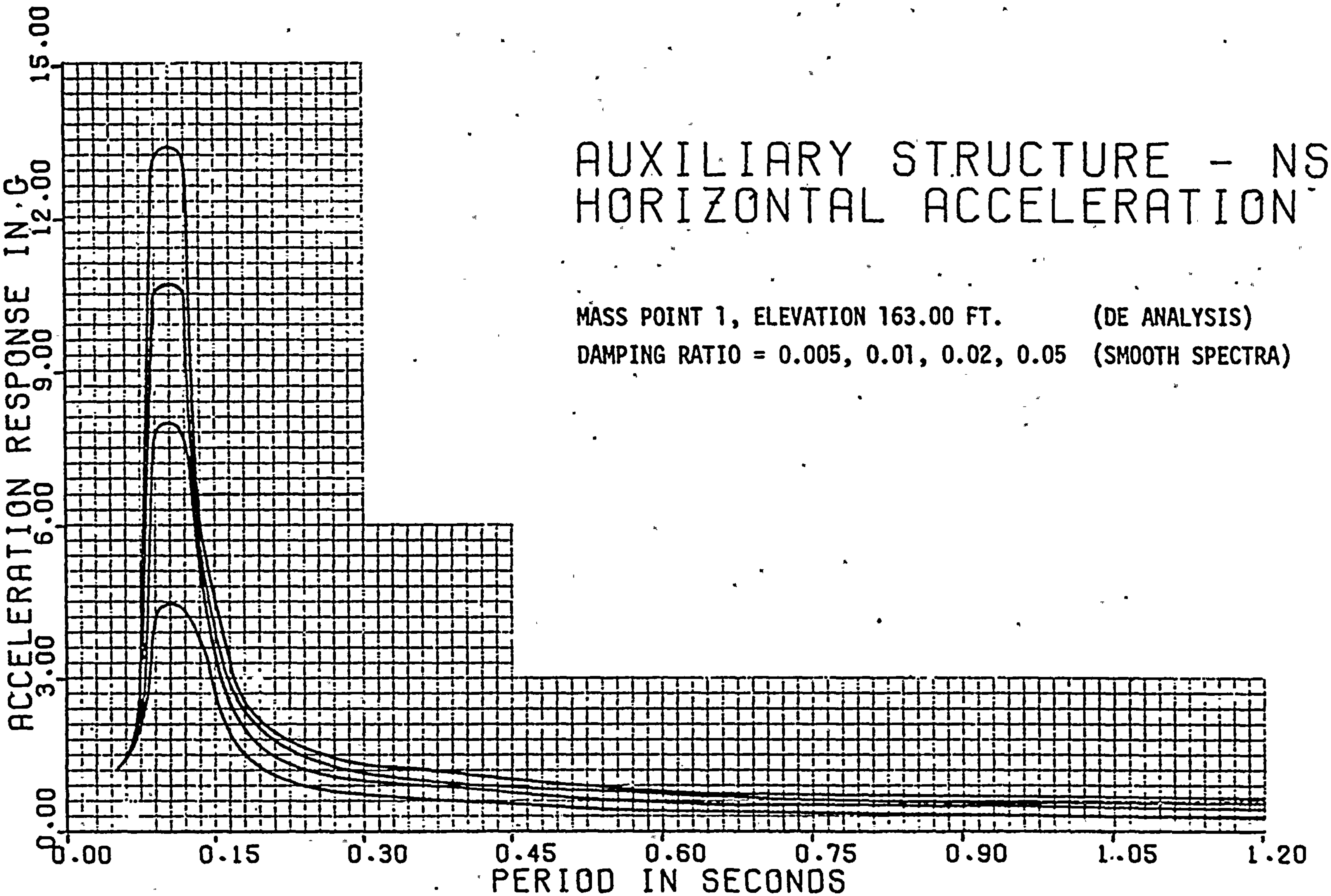


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-13

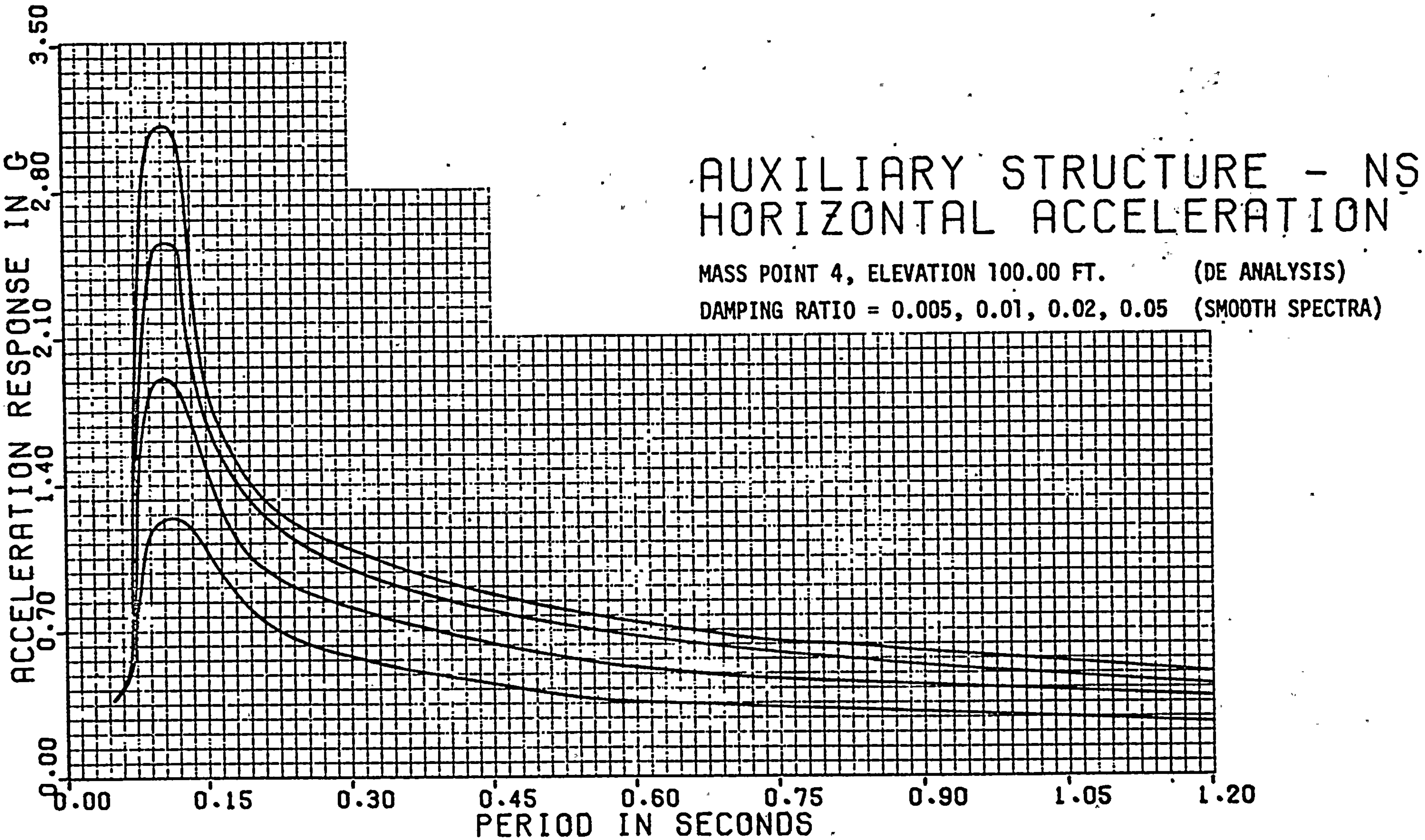
AUXILIARY BUILDING

MATHEMATICAL MODEL



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-14
AUXILIARY BUILDING
TYPICAL SPECTRA

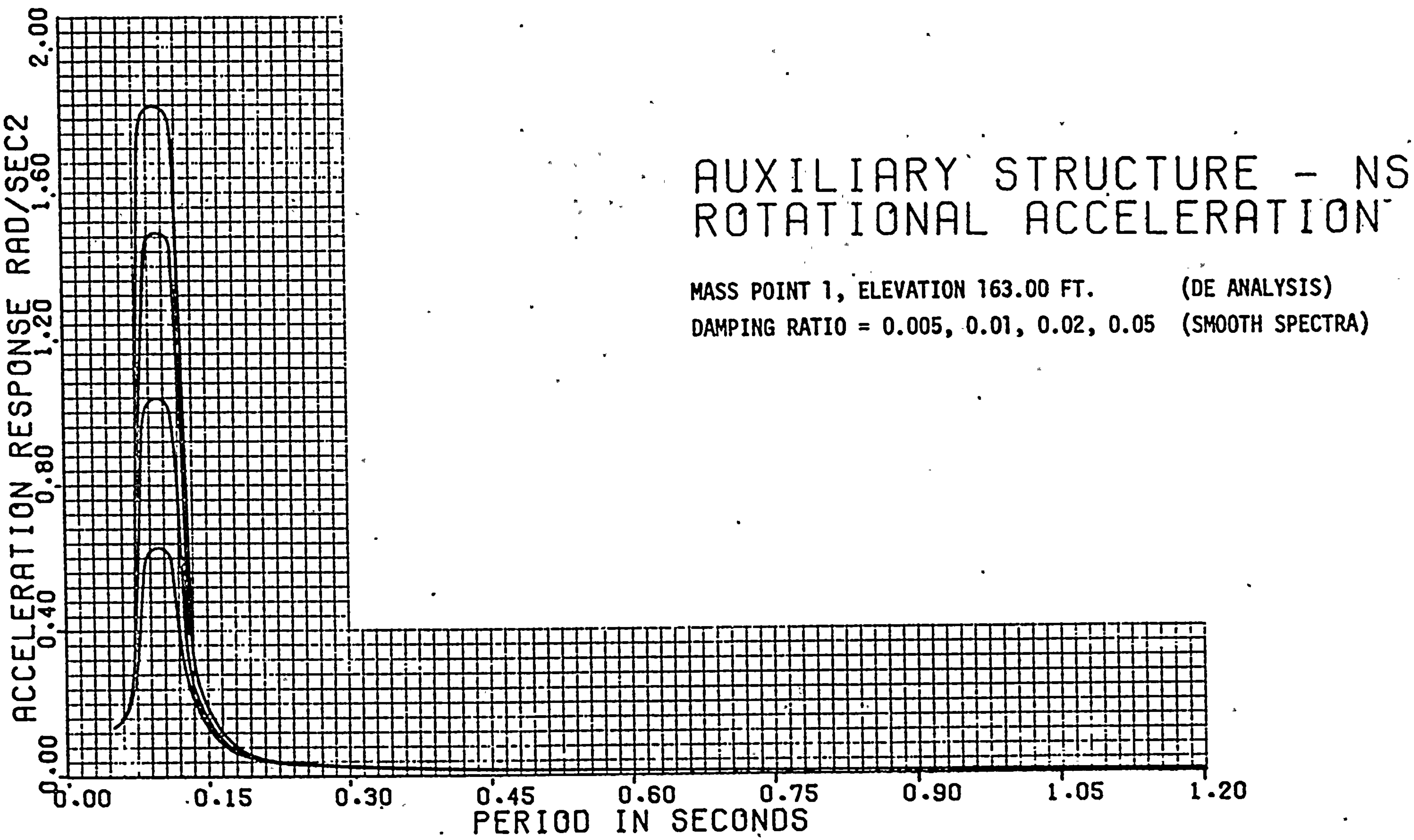


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-15

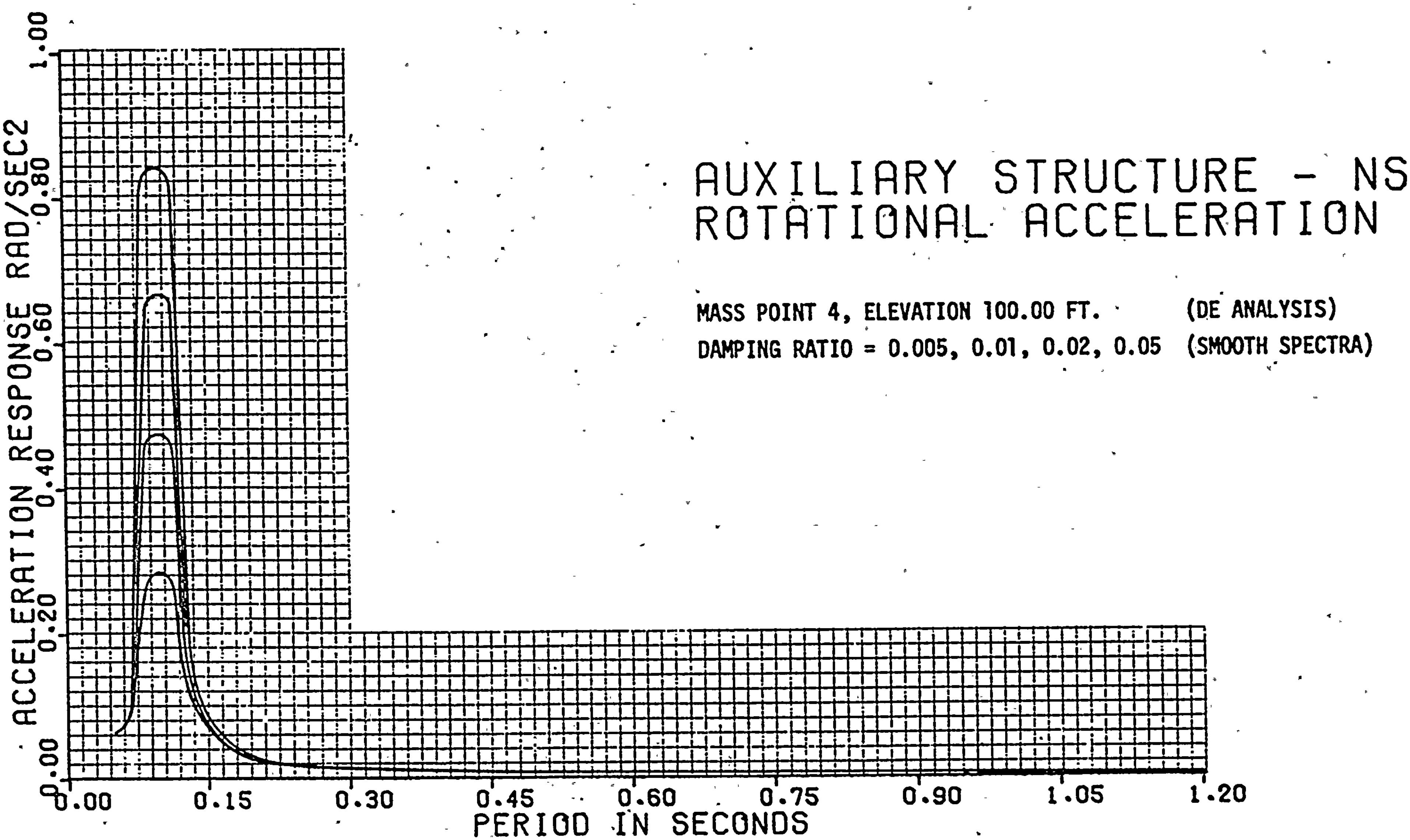
AUXILIARY BUILDING

TYPICAL SPECTRA



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-16
AUXILIARY BUILDING
TYPICAL SPECTRA

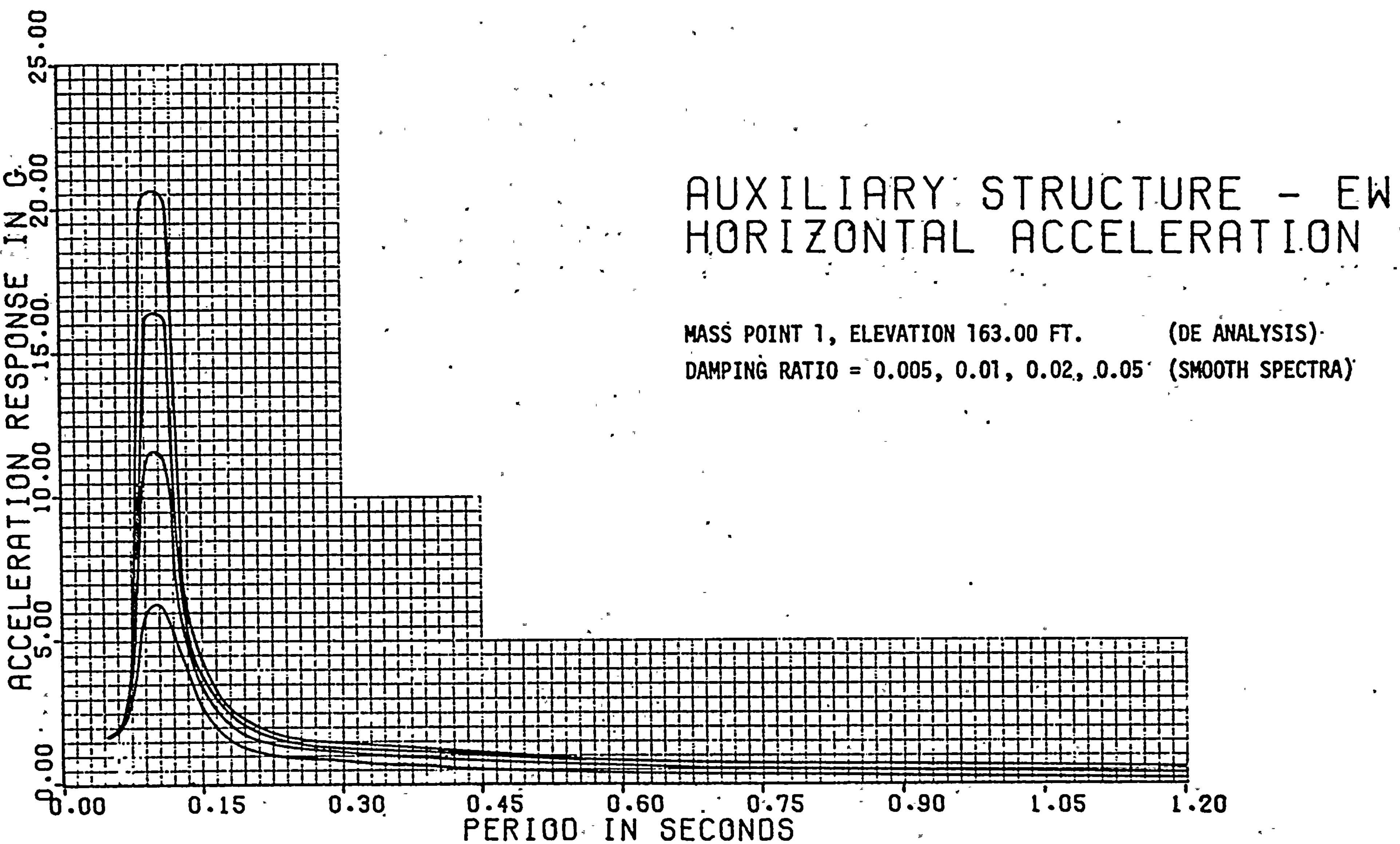


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-17

AUXILIARY BUILDING

TYPICAL SPECTRA

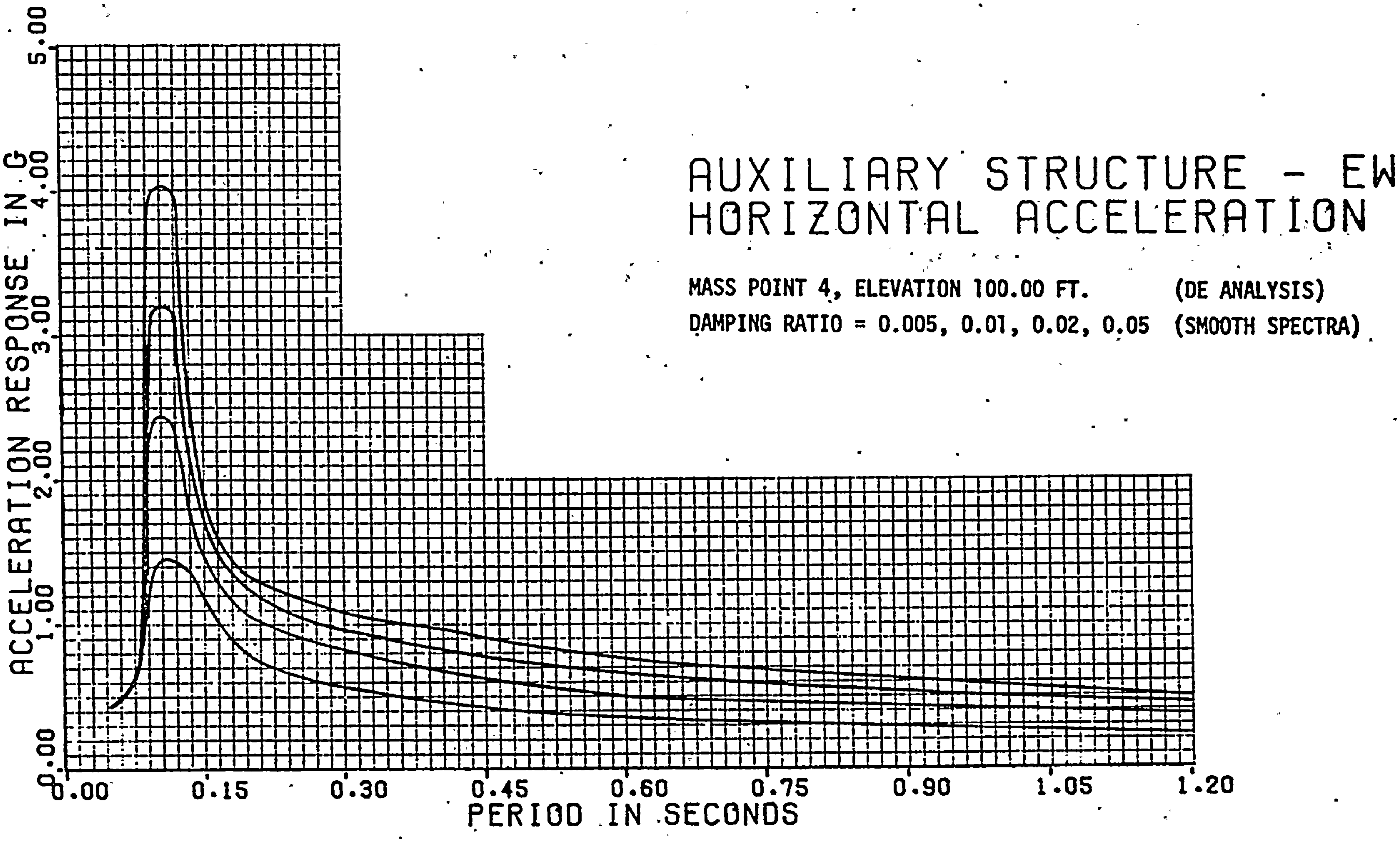


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-18

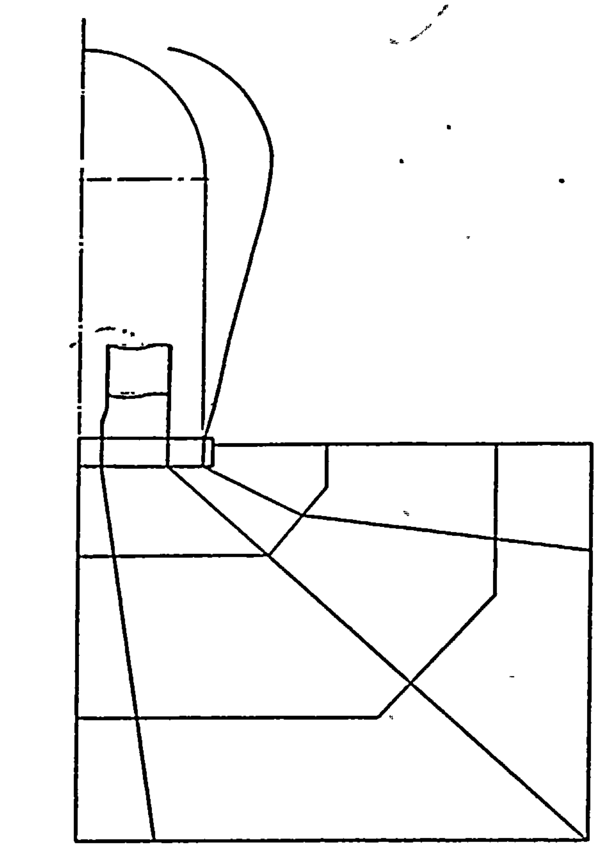
AUXILIARY BUILDING

TYPICAL SPECTRA

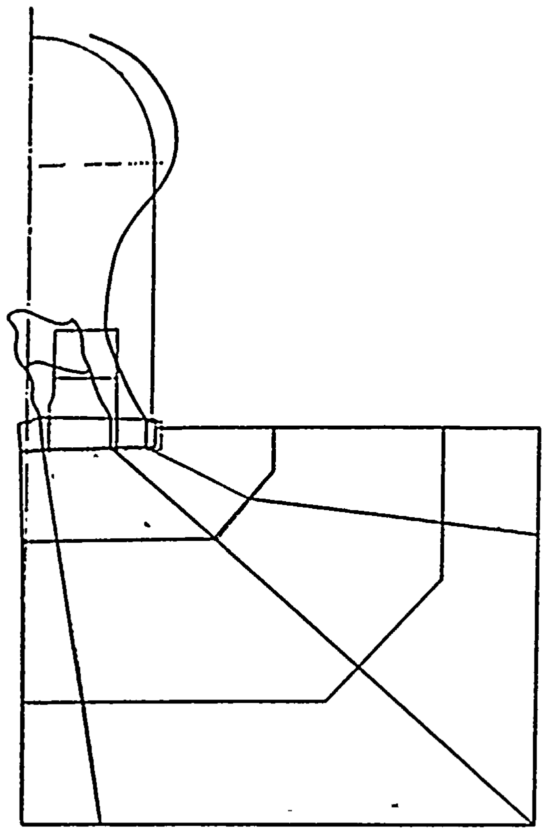


UNITS 1 AND 2
DIABLO CANYON SITE

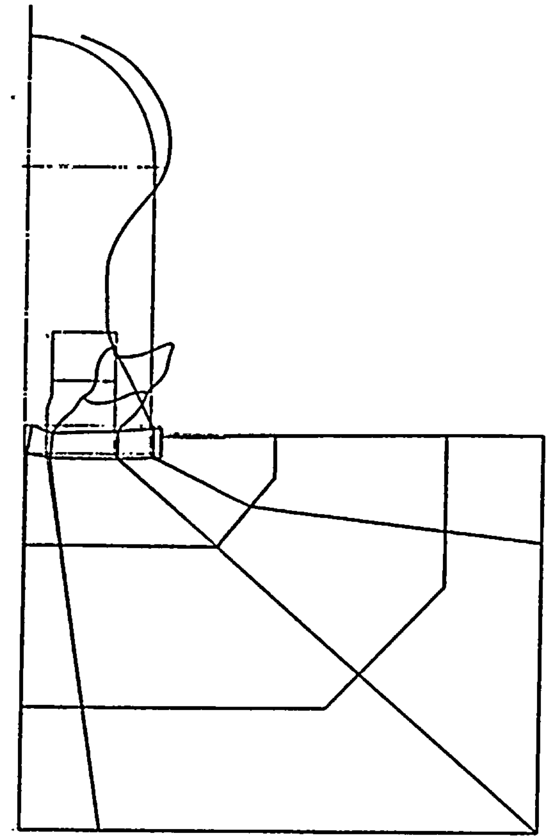
FIGURE 3.7-19
AUXILIARY BUILDING
TYPICAL SPECTRA



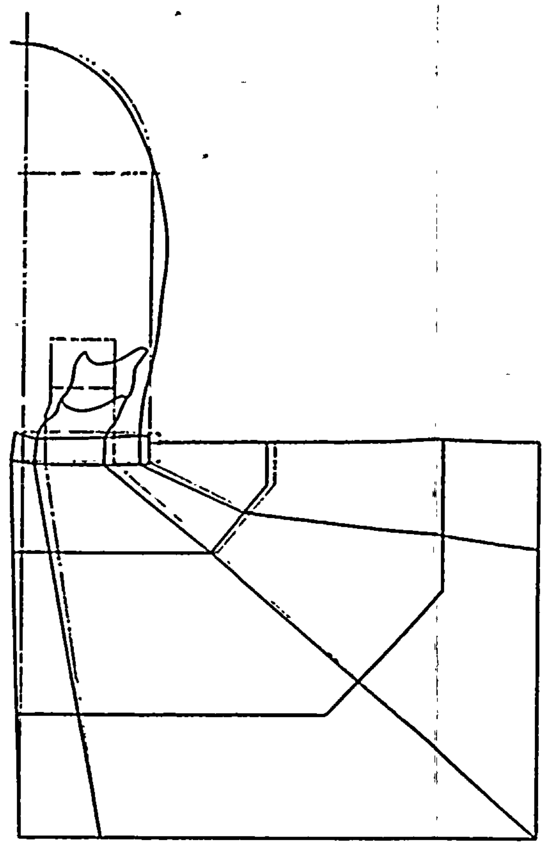
MODE 1
PERIOD = 0.255 SEC.



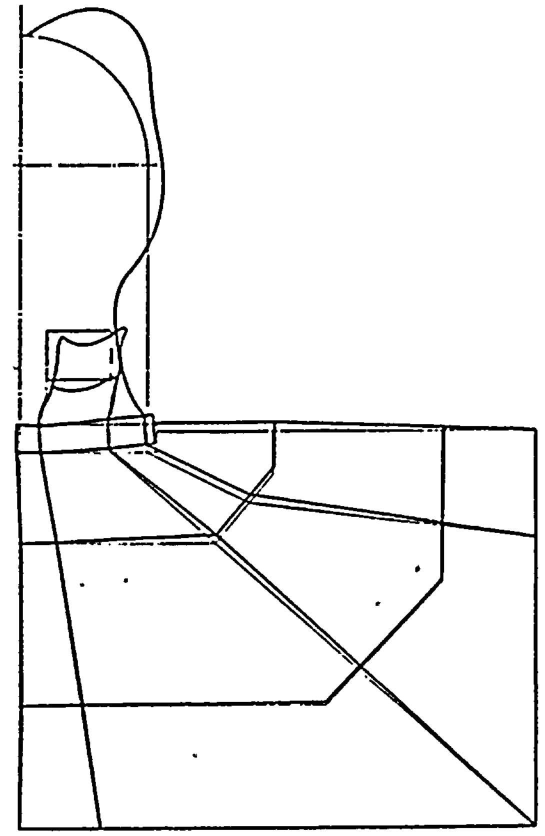
MODE 2
PERIOD = 0.093 SEC.



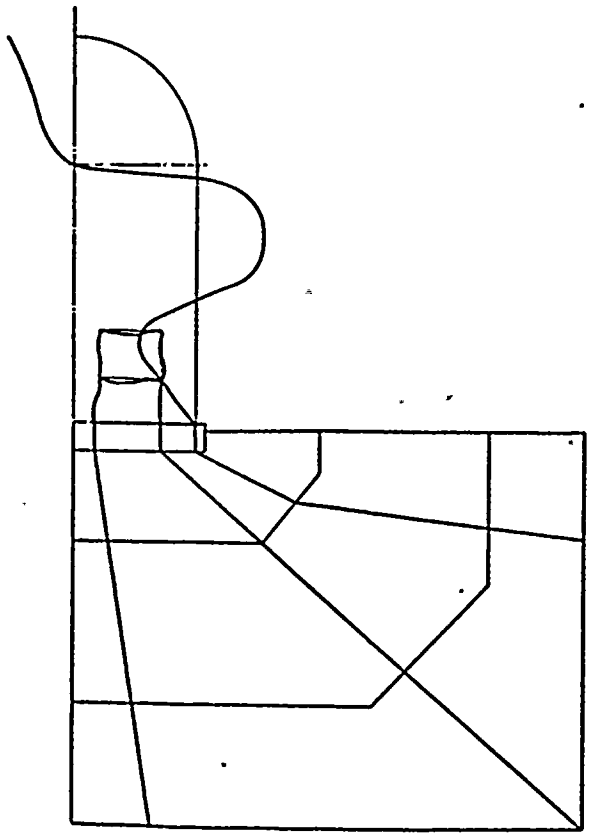
MODE 3
PERIOD = 0.088 SEC.



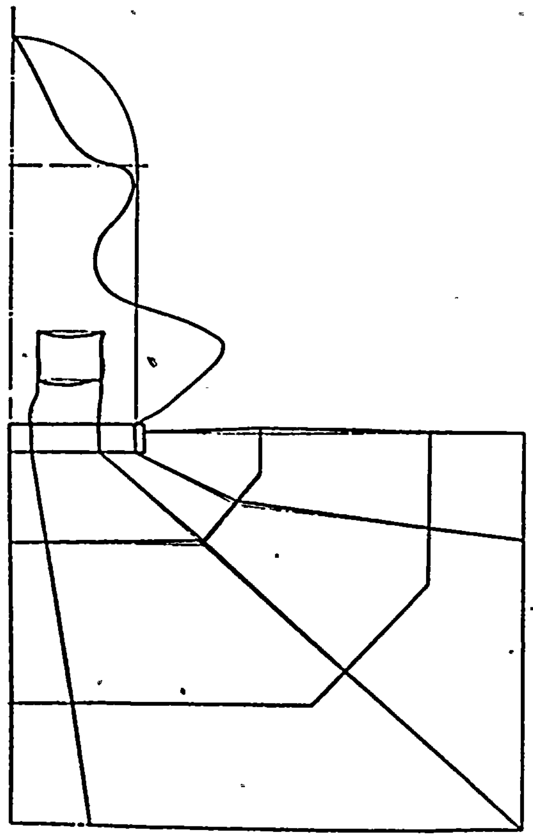
MODE 4
PERIOD = 0.073 SEC.



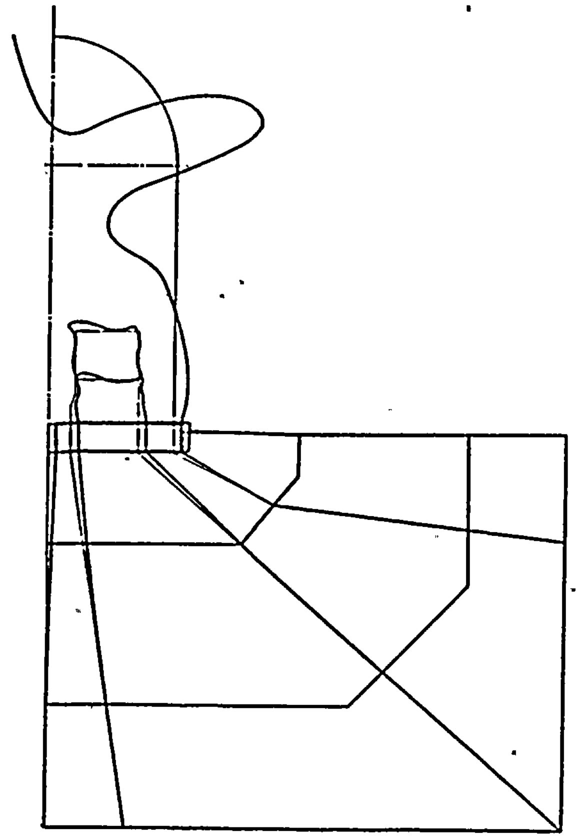
MODE 5
PERIOD = 0.060 SEC.



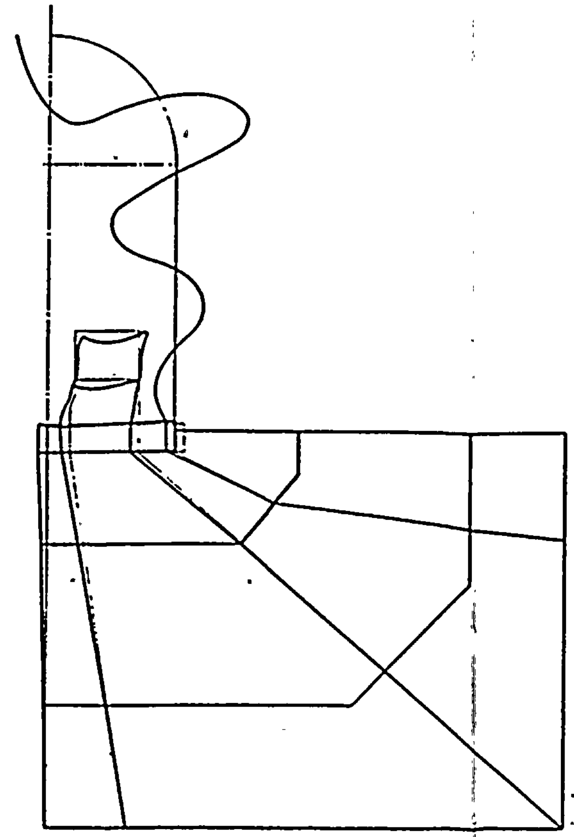
MODE 6
PERIOD = 0.058 SEC.



MODE 7
PERIOD = 0.057 SEC.



MODE 8
PERIOD = 0.051 SEC.



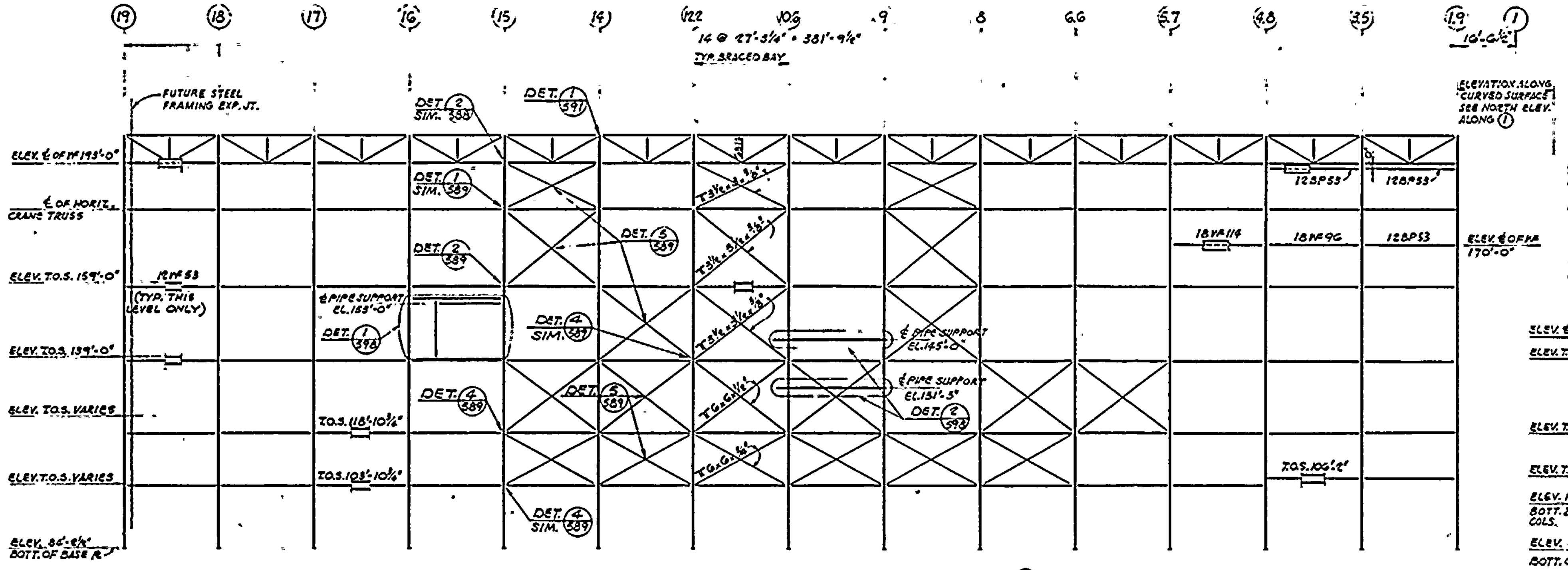
MODE 9
PERIOD = 0.0509 SEC.

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-6

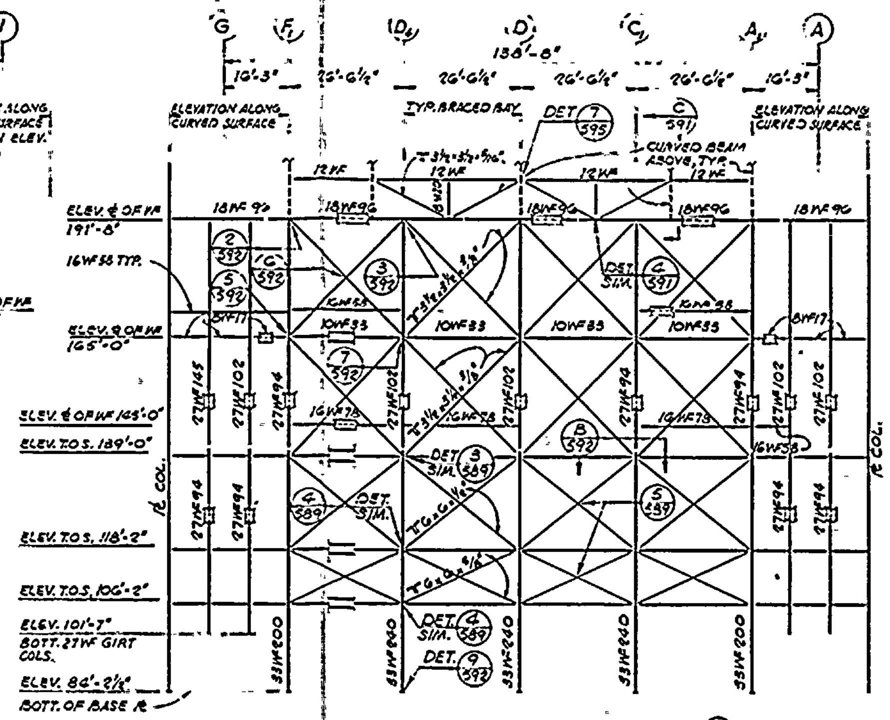
CONTAINMENT STRUCTURE

MODE SHAPES



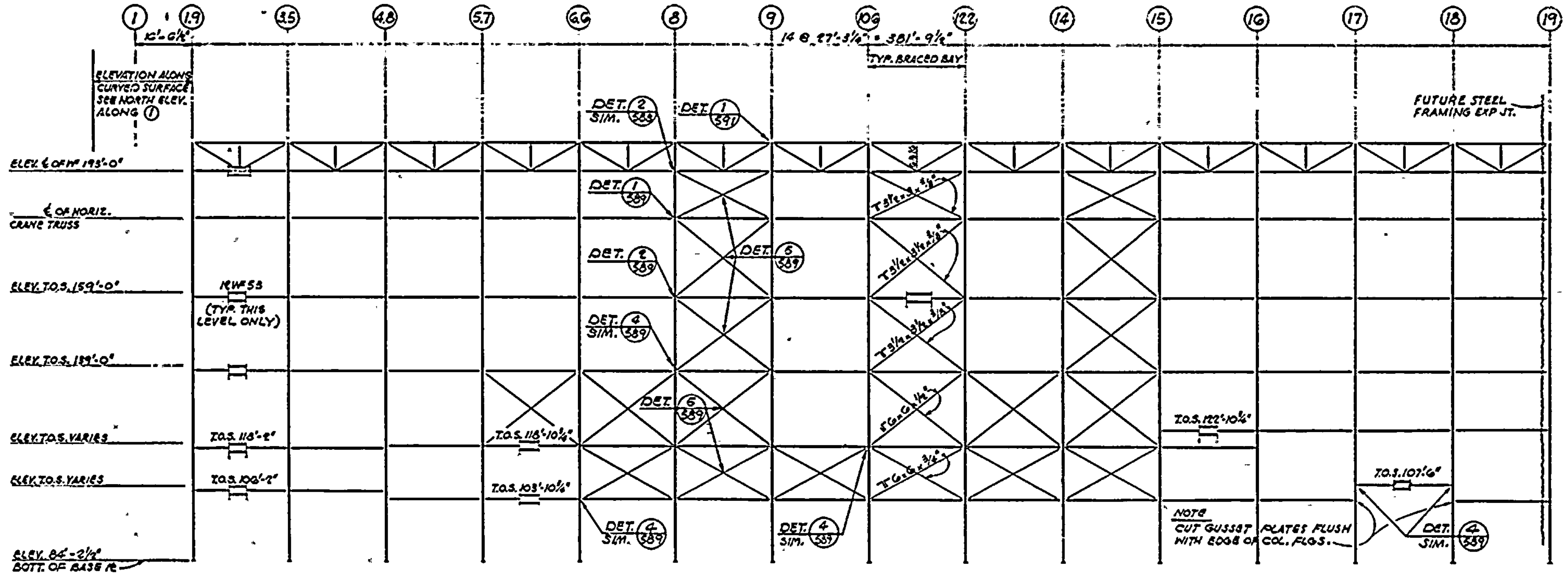
EAST ELEVATION ALONG (G)

SCALE: 1/16"=1'-0"



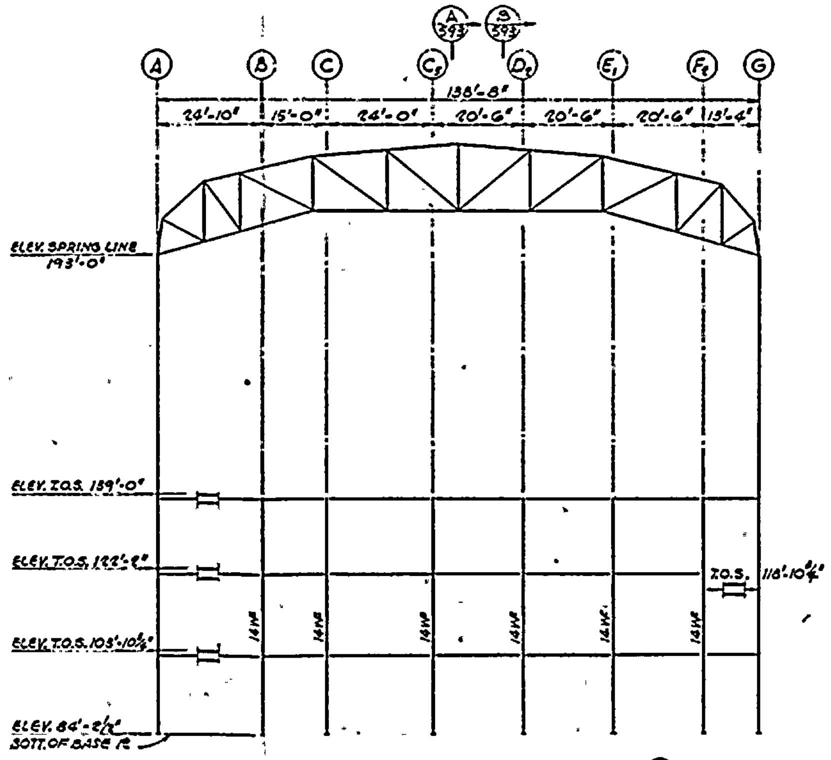
NORTH ELEVATION ALONG (1)

SCALE: 1/16"=1'-0"



WEST ELEVATION ALONG (A)

SCALE: 1/16"=1'-0"



SOUTH ELEVATION ALONG (19)

SCALE: 1/16"=1'-0"

NOTE:

ERECTION OF STRUCTURAL STEEL ON THE NORTH SIDE OF TRUSS T-3 BETWEEN COLUMN LINES (1) & (19) & LINES (A) & (G) ABOVE EL. 140' WILL HAVE TO BE DELAYED UNTIL HEAVY COMPONENTS OF THE GENERATOR ARE MOVED INTO THE TURBINE BUILDING.

UNITS 1 AND 2
DIABLO CANYON SITE

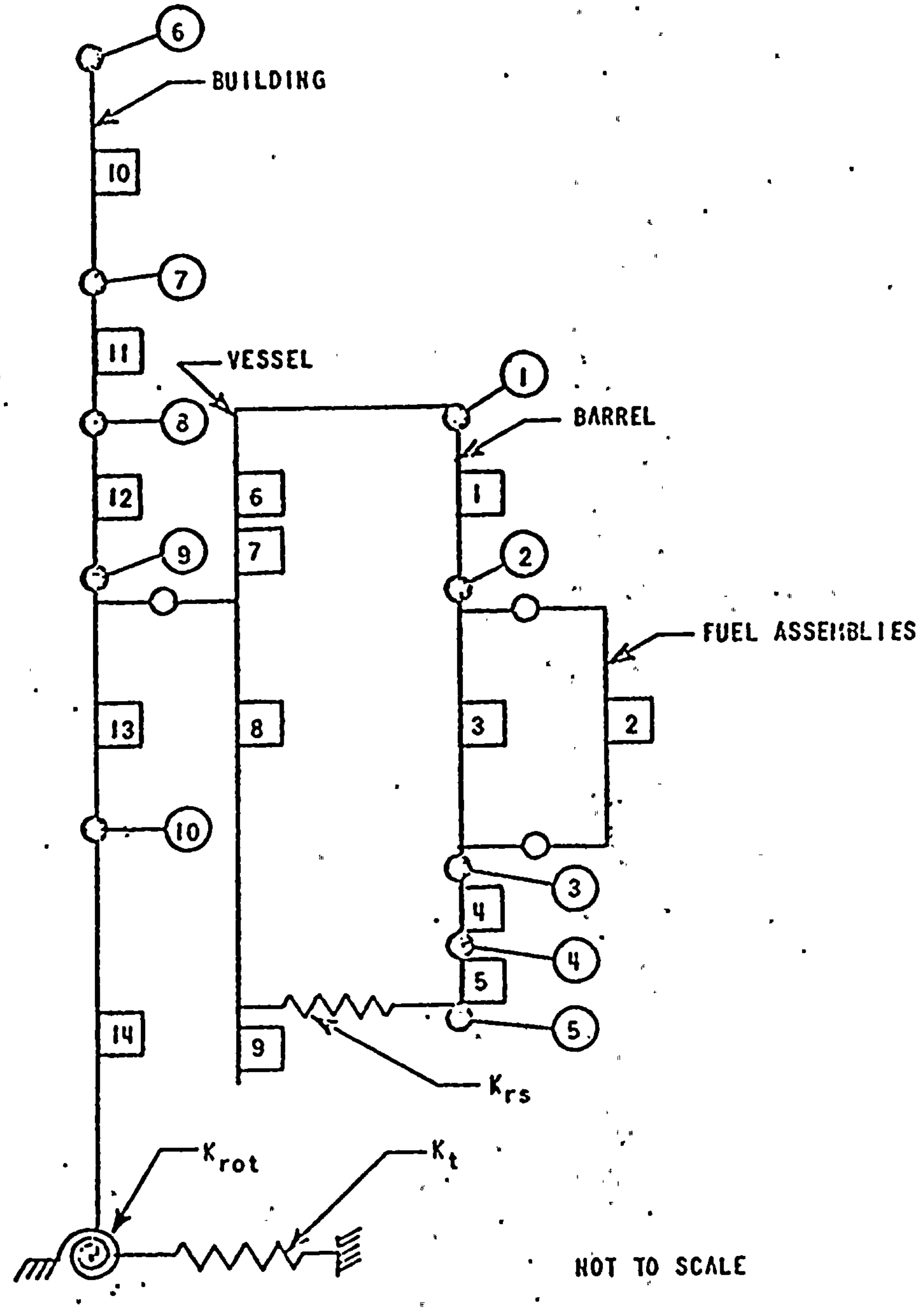
FIGURE 3.7-19A
TURBINE BUILDING
STRUCTURAL STEEL EXTERIOR
STRUCTURAL FRAMING ELEVATIONS

Amendment 8 May 1974

K_{rs} = RADIAL SUPPORT SPRING CONSTANT

K_{rot} = ROTATIONAL GROUND SPRING CONSTANT

K_t = TRANSLATIONAL GROUND SPRING CONSTANT

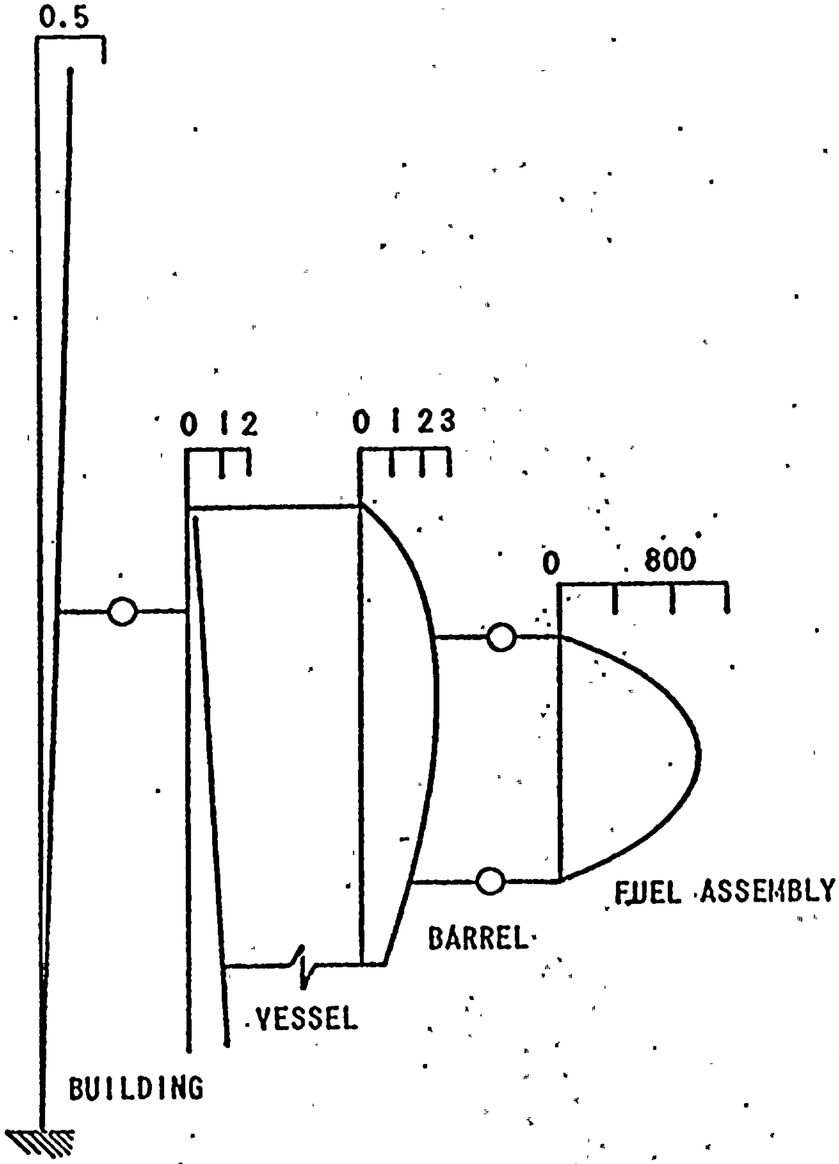


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-20

REACTOR INTERNALS

MATHEMATICAL MODEL

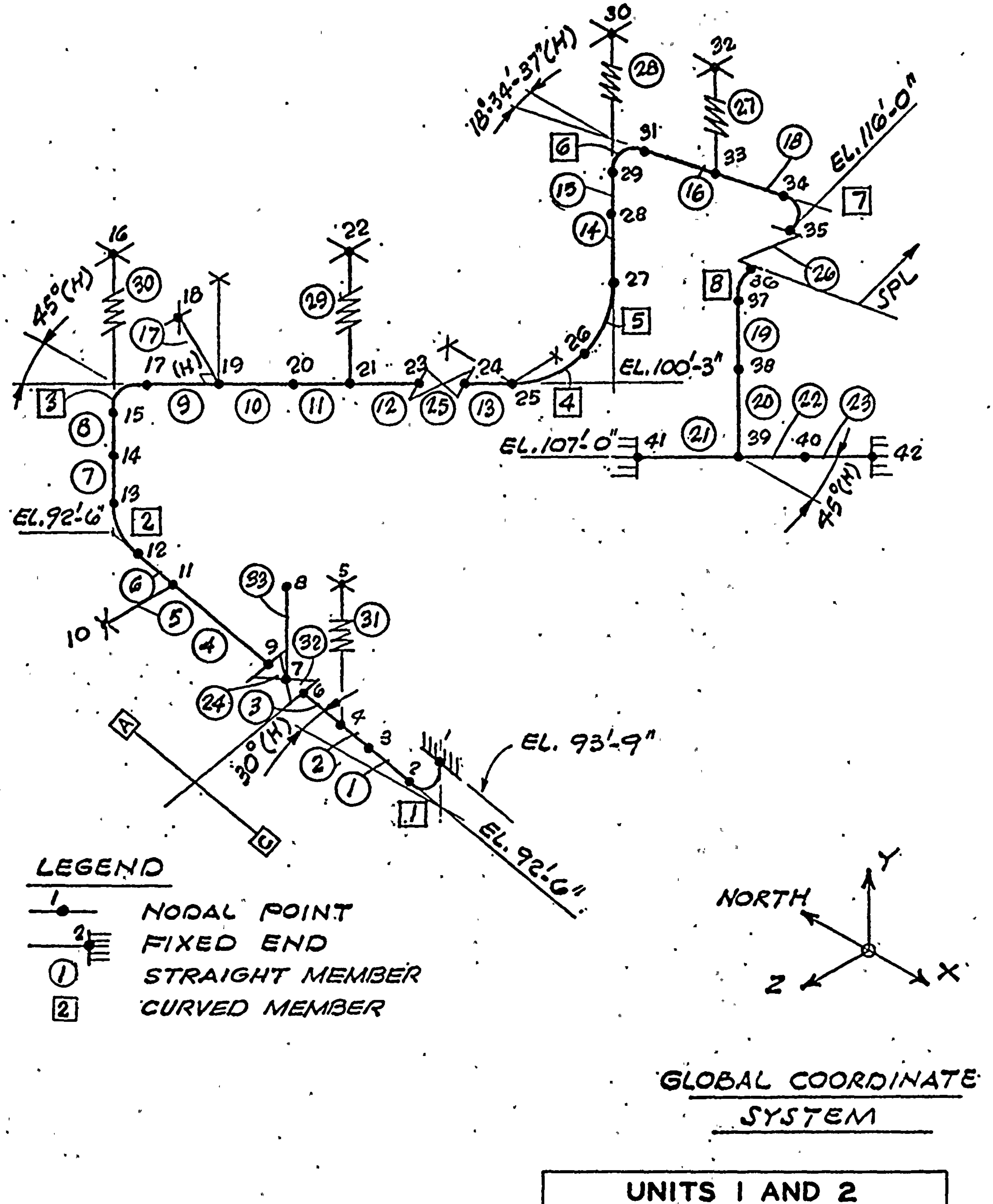


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-21

REACTOR INTERNALS

FIRST MODE OF VIBRATION



GLOBAL COORDINATE SYSTEM

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.7-22

ACCUMULATOR LOOP NO. 1

MATHEMATICAL MODEL

3.8 DESIGN OF DESIGN CLASS I STRUCTURES

Figure 1.2-2 shows the location of all structures for the Diablo Canyon plant, Units 1 and 2. The design classification of plant structures is given in Table 3.2-4, "Classification of Structures, System and Components for Diablo Canyon, Section I - Structures, Buildings, and Facilities." In that table, the following Design Class I structures are shown:

Containment Structure

Auxiliary Building

The design of these structures is discussed in this section, the Containment Structures in Subsection 3.8.2 and the Auxiliary Building in Subsection 3.8.1. In the Diablo Canyon plant, the Auxiliary Building includes the control room, which serves both units, and a refueling area for each unit. In addition to the discussion of the Auxiliary Building, Section 3.8.1 contains a discussion of the Design Class I foundations for the Condensate Water Storage Tanks, the Refueling Water Storage Tanks, the Firewater Tank and the Transfer Tank, all of which are located on reinforced concrete slabs adjacent to the east side of the Auxiliary Building, and a discussion of the analyses performed on Design Class II structures which contain Design Class I equipment.

3.8.1 STRUCTURES OTHER THAN CONTAINMENTS

Auxiliary Building

Description of the Auxiliary Building

The Auxiliary Building is located between the Unit 1 and Unit 2 Containment Structures. It contains the control room which includes consoles for each unit and a fuel handling area for each unit. In addition, the Auxiliary Building contains equipment for the Chemical and Volume Control Systems, the Safety Injection Systems, the Residual Heat Removal Systems, the Component Cooling Water Systems, the Liquid Radwaste Systems, the Gaseous Radwaste System, and others.

The main floor levels in the Auxiliary Building are at elevations 60, 73, 85, 100, 115, and 140. Elevations 60 and 73 are below ground level, which is at elevation 85 except for the east side of the building where ground level is at elevation 115.

The general arrangement of equipment in the Auxiliary Building is shown in Figures 1.2-4 through 1.2-11, Figures 1.2-21 through 1.2-27, and Figures 1.2-29 and 1.2-30. Generally speaking, one half of the Auxiliary Building is a mirror image of the other, with each half of the structure containing equipment for one unit. The control room is located at elevation 140. The two fuel handling areas which contain the spent fuel pools, the fuel handling cranes, fuel racks, and related equipment are located on the east side of the Auxiliary Building with the top of the spent fuel pools at elevation 140.

2 The Auxiliary Building is a reinforced concrete, shear wall structure except for the fuel handling area crane support structure which is structural steel. The shear walls are generally 3 feet thick, with a minimum thickness of 2 feet. Slabs are generally 2 feet thick. The walls of the spent fuel pools are a minimum of 6 feet thick except for local areas around the fuel transfer tubes. The foundation slabs under the spent fuel pits have a minimum thickness of 5 feet. The spent fuel pits sides and bottoms are lined with stainless steel, 1/4 inch thick on the bottoms and 1/8 inch nominal on the sides. Representative concrete outlines, reinforcing steel arrangements, and structural steel details for the Auxiliary Building are shown in Figures 3.8-0A through 3.8-00.

The 125 ton overhead crane in the fuel handling area shown in Figure 3.8-00, is equipped with restraints which prevent derailing from motions associated with an earthquake. The detail of these restraints is shown in Figure 3.8-ON, Section F and Detail 5.

The only connections between the Auxiliary Building and other structures are the Fuel Transfer Tube and miscellaneous piping. The Fuel Transfer Tube is fitted with expansion bellows which allow relative movement between the Auxiliary Building, the Containment Structure exterior shell, and the internal structure of the Containment Structure. The design of the expansion bellows considers the maximum axial and lateral relative deflection that could occur during the Double Design Earthquake. Piping systems are analyzed for the maximum relative displacements of the Auxiliary Building and other structures, and the piping anchor points in the structures are designed to withstand the resulting forces.

Codes and Standards

The following codes and standards were used in the design, construction, inspection and testing of the Auxiliary Building.

1. ACI Standard Building Code Requirements for Reinforced Concrete (ACI 318-63), except that design loading combinations were as described in this Subsection 3.8.1.

2. Manual of Standard Practice for Detailing Reinforced Concrete Structures (ACI 315-65).

3. Recommended Practice for Evaluation of Compression Test Results of Field Concrete (ACI 214-65).

4. Inspection of the Cadweld Rebar Splice (Erico Products, Inc., RB-5M768).

5. Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction, American Welding Society AWS D12.1-61.

6. AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Sixth Edition.

7. Code for Welding in Building Construction, AWS D1.0-69. Work performed prior to December 12, 1969 is in accordance with the earlier edition AWS D1.0-66.

8. Stud welding is in accordance with the Supplement to American Welding Society Specifications AWS D1.0-66 and AWS D2.0-66 on Requirements for Stud Welding.

9. Materials and the quality control tests for materials conform to ASTM standards.

Regulatory Guide 1.15, Testing of Reinforcing Bars for Category I Concrete Structures (dated December 28, 1972) and Regulatory Guide 1.55, Concrete Placement in Category I Structures (dated June 1973) were issued after construction at the Diablo Canyon plant was nearly complete. A comparison of the program used for the Diablo Canyon plant with the Regulatory Position of Regulatory Guide 1.15 is presented in Table 3.8-1. The quality assurance program for the Diablo Canyon plant meets the requirements of Regulatory Guide 1.55. In regard
8 to Regulatory Guide 1.55, the references used for guidance are those listed in Appendix A as they existed at the time of the PSAR.

<u>Design Loads</u>

The following loads are considered in the design of the Auxiliary Building:

Dead Loads

These consist of the weight of concrete, reinforcing steel, structural steel, and permanent equipment loads. Equipment loads are supplied by the manufacturer.

Live Loads

These consist of temporary equipment loads and a uniform load to account for the miscellaneous temporary loadings that may be placed on the structure.

Earthquake Loads

Earthquake loads are based on a time history modal superposition analysis of the Auxiliary Building by John A. Blume & Associates, Engineers. This analysis is described in Subsection 3.7.2.

Wind Loads

Wind loads are determined in accordance with the criteria presented in Section 3.3. However, the forces due to wind are much less than those due to earthquake; consequently, seismic loads, rather than wind, are entered in the load combination equations.

Thermal Loads

Thermal loads are loads induced by local increases in temperature. Thermal loads result from normal operating conditions and from postulated accident conditions.

Pipe Reaction Loads

Pipe reactions which result from hydraulic forces, thermal expansion, and seismic events, are transferred to the structure through pipe supports. Pipe reaction loads result from normal operating conditions and postulated accident conditions.

Jet and Missile Loads

These are localized forces on structures in the immediate vicinity of a postulated pipe break. Jet forces result from the impingement of high energy fluid on a structure. Missile forces result when a part possessing kinetic energy strikes a structure.

Loading Combinations

Normal Conditions

Dead load, live load, loads from the Design Earthquake, thermal loads, and pipe reactions are considered in all possible combinations. Inasmuch as working stress design is used for normal operating loads, the factored load approach is not used. For each structural member, the combination of these loads that produces the maximum stress is used for design. Stated in equation form:

$$C = D + L + DE + T_o + R_o$$

Where C = Required load capacity of section.

D = Dead load of structure and equipment loads.

L = Live load.

DE = loads resulting from the Design Earthquake.

T_o = Thermal loads during normal operating conditions.

R_o = Pipe reactions during normal operating conditions.

Abnormal Conditions

Dead load, live load, Double Design Earthquake loads, and loads associated with accidental pipe rupture are considered in the following combinations. For each structural member, the combination that produces the maximum stress is used for design.

Concrete Structural Elements

$$U = D + L + T_A + R_A + 1.5\ P_A$$

$$U = D + L + T_A + R_A + 1.25\ P_A + 1.0(Y_R + Y_J + Y_M) + 1.25\ DE$$

$$U = D + L + T_A + R_A + 1.0\ P_A + 1.0(Y_R + Y_J + Y_M) + DDE$$

Where T_A = Thermal loads on structure generated by a postulated pipe break including T_o.

R_A = Pipe reactions on structure from unbroken pipe generated by a postulated pipe break conditions including R_o.

P_A = Pressure load within or across a compartment and/or building generated by a postulated pipe break and including an appropriate dynamic factor (DLF) to account for the dynamic nature of the load.

Y_R = Reaction on structure from broken pipe generated by a postulated pipe break, including an appropriate DLF.

Y_J = Jet load on structure generated by a postulated pipe break, including an appropriate DLF.

Y_M = Missile impact load on a structure generated by or during a postulated pipe break, such as a whipping pipe, including an appropriate DLF.

U = Strength required to resist design loads based on the methods described in ACI 318-63.

DDE = Loads resulting from the Double Design Earthquake.

Steel Structural Elements

Where elastic working stress design methods are used:

$$1.6\ S^{**} = D + L + T_A^{*} + R_A + P_A$$
$$1.6\ S^{**} = D + L + T_A^{*} + R_A + P_A + 1.0(Y_J + Y_R + Y_M) + DE$$
$$1.6\ S^{**} = D + L + T_A^{*} + R_A + P_A + 1.0(Y_J + Y_R + Y_M) + DDE$$

81

Where plastic design methods are used:

$$.90\ Y^{**} = D + L + T_A^{*} + R_A + 1.5\ P_A$$
$$.90\ Y^{**} = D + L + T_A^{*} + R_A + 1.25\ P_A + 1.0(Y_J + Y_R + Y_M) + 1.25\ DE$$
$$.90\ Y^{**} = D + L + T_A^{*} + R_A + 1.0\ P_A + 1.0(Y_J + Y_R + Y_M) + DDE$$

Where S = required section strength based on elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specifications for the Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

Y = required section strength based on plastic design methods described in Part 2 of AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

---

* Thermal loads are neglected when it can be shown that they are secondary and self-limiting in nature and where the material is ductile.

** For existing structures, the 1.6 factor applied to the required section strength S and the 0.90 reduction factor applied to the required section strength Y, are increased to 1.7 and 1.0 respectively. On such situations, however, it is verified that deflections will not result in the loss of function of any safety-related system. 81

Concrete or Steel Structural Elements

Both cases of L having its full value present during the postulated pipe rupture or being completely absent are checked.

Design and Analysis Procedures

Structural analysis of the Auxiliary Building is performed by the traditional methods of engineering analysis for structural steel and reinforced concrete structures. These methods are based on the principles of equilibrium, compatability of deformations, and predictions of material strength by the methods of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" (AISC Code) and the "ACI Standard Building Code Requirements for Reinforced Concrete" (ACI Code).

For the Auxiliary Building, lateral loads from seismic exceed those from wind. Thus, lateral loads are determined from the time history, modal superposition analysis described in Section 3.7.

The specific design methods used for the structural steel and concrete parts of the Auxiliary Building are as follows:

Structural Steel

The refueling crane support structure consists of roof trusses in the East-West direction supported on structural steel columns. The columns are connected to the trusses with moment-resistant connections and are pinned at the base. The overhead refueling crane is supported on the same columns as the roof trusses. The bents formed by the roof trusses and columns are braced in the North-South direction by structural steel members.

The analysis of the structural steel crane support structure was performed using the STRUDL program for space frame analysis. Member sizes and connection design is determined in accordance with the AISC code.

Concrete

Reinforced concrete slabs are supported on shear walls or, in some locations, on columns. For vertical loads, the slabs are designed as two way slabs except when the ratio of length-to-width is such that the slab action is one way. All slab design is in accordance with the methods presented in the ACI Code.

The seismic forces from the time history, modal superposition analysis are distributed to individual structural members using the methods outlined in References 1 and 2.

Missile Forces, Jet Forces, and Pipe Reactions

Missile forces are calculated by the methods described in Section 3.5. Jet forces and pipe reactions from a postulated broken pipe are calculated as described in Section 3.6.

Computer Programs

The following computer programs were utilized for structural analysis of the Auxiliary Building:

1. The DYBOX 2 program was developed to calculate structural properties of a
shearwall structure based on the methods of Reference 1. The program was
used to calculate the following properties of the Auxiliary Building at 3
each floor level: weight of structure and equipment, mass moment of
inertia, torsional rigidity, areas and moments of inertia in two orthogonal
directions, coordinates of the center of mass and the center of rigidity,
and the total rigidity in each direction.

2. The SHEARWALL 4 program was developed to calculate the individual member forces from the output of the time history, modal superposition analysis. The program is based on the methods of References 1 and 2 and the ACI code and was used to calculate shear forces, overturning stresses, and area of reinforcing steel required for each shear wall.

3. The STRUDL program of the Massachusetts Institute of Technology Integrated Civil Engineering System (ICES) was used to analyze the fuel handling area crane support structure.

4. Dynamic analysis programs described in Section 3.7 were used for the time history, modal superposition analysis of the Auxiliary Building.

3

Verification of these computer programs was accomplished as follows:

1. Test problems were performed for the DYBOX 2 program and the SHEARWALL 4 program. For each test problem, the calculations were performed both by hand and by use of the computer program. The test problems were selected so that they are representative of the actual calculations performed for the Auxiliary Building. The test problem for DYBOX 2 is shown in Figure 3.8-0Q, and a comparison of the results is shown in Table 3.8-7. The test problem for SHEARWALL 4 is shown in Figure 3.8-0R, and a comparison of the results is shown in Table 3.8-8.

Description	Std. No.
Power Circuit Breakers	NEMA SG-4
Power Switchgear Assemblies	NEMA SG-5
Industrial Control Equipment	NEMA ICS
Switchgear Assemblies including metal-enclosed buses	USA C37.20
Electrical Indicating Instruments	USA C39.1
Instrument Transformers	USA C57.13
DC and AC Fractional Hp Motors	USA C50
AC Power Circuit Breaker	USA C37.4
Power Circuit Breaker Control	USA C37.11
Relays and Relay Systems Associated with Electric Power Apparatus	ANSI C37.90

47

The manufacturer has made design and production tests on this switchgear according to ANSI Standards C37.09 and C37.09a, Test Procedures for AC High Voltage Circuit Breakers. These standards provide the test plans, set up, procedures, and acceptability requirements. Test results for this standard production equipment are kept by the manufacturer.

The design tests were made to determine the adequacy of the design of the particular type, model, and size of circuit breaker to meet its assigned ratings at the time this model was developed. Applicable portions of these design tests as well as test data of previous similar equipment may have been used to evaluate current designs.

The maximum design hot spot temperature rise above ambient allowed in this equipment while carrying rated current is 65°C. With a 40°C ambient, the maximum allowable design hot spot temperature is therefore 105°C. This is the temperature which could be tolerated continuously for normal life by material of the lowest temperature classification used in this switchgear, such as Class A insulation. A long experience record of satisfactory performance for this equipment is evidence of the conservatism of this design limit. 47

Additional conservatism results from the following considerations:

(1) The ambient temperature does not stay continuously at 40°C (See page 3.11-4).

(2) The equipment is not loaded continuously to its rated value. The maximum current rating of the 4,160 volt safety related switchgear (Boards 1F, 1G,
47 and 1H) is 1,200 amps; yet the largest motor draws less than 100 amps when operating at rated load, and the total load on any bus section is always less than 500 amps.

(3) The steady-state temperature rise of the hottest spot on this equipment during test at rated load was 46°C above an ambient temperature of 22°C. This rise is 19°C below the allowable value.

Production tests also were made on the actual equipment delivered to check the quality and uniformity of the workmanship and materials used. The tests and inspection described below are part of the manufacturer's regular manufacturing procedure.

Tests are made to check the wiring for continuity and to assure that it conforms to the wiring diagrams. A master breaker fixture is used to assure interchangeability of circuit breakers and to check primary and secondary contact alignment. Instruments, relays, and contactors are tested to assure proper operation and function, and compliance with the equipment specification.

Resistance measurements were made of the breaker's operating coils to assure conformance with design specifications. No-load operation tests were made, including checks of breaker timing at maximum, normal, and minimum control voltage ratings.

In addition to the manufacturer's tests, the equipment has been given thorough tests in the field before operation. Included are high potential and insulation resistance tests and a complete check of the operation of the equipment.

2. Electric Motors - All Class 1E electric motors for the balance of plant are located outside the containment where the ambient does not exceed 40°C, except as noted on Page 9.4-1, and the radiation is low.

These motors have been designed, manufactured, and tested as required by NEMA. The motor manufacturer performed tests as described in NEMA Standard MG-1, Parts 12 and 20, and according to procedures described in IEEE Standard No. 101 and 112A to verify the design, construction and performance of the motor. 47

Included in these tests are:

a. No load current and speed at rated power frequency and voltage.

b. Current input at rated frequency with motor at standstill.

c. A high potential test applied to the winding for one minute with an a.c. voltage of twice rated plus 1000 volts. This test is performed after all other tests.

d. A dynamic balance of the rotating parts.

In addition, each motor is performance tested coupled to its load by the supplier of the driven equipment to verify performance of the combination. These tests are performed in accordance with the Hydraulic Institute Standards. The overall efficiency at the design point flow of each motor-pump unit is determined to verify that it is greater than the product of the motor efficiency (at the design point brake horsepower) and the pump efficiency at the design point flow.

The motors are also given thorough tests after they are installed in the plant. These include high potential tests, dynamic balance tests of motor and driven load, and tests to determine the starting currents and times and the full load current of the motor.

The following analysis of environmental capability of the auxiliary saltwater pumps is presented as an example to illustrate the margins available. The 400 horsepower, 4,000 volt motors for these pumps have Class B insulation, which has a maximum continuous total temperature rating of 130°C. In the case of a motor, NEMA Standard MG-1 allows a temperature rise of 80°C (measured by resistance) above an ambient of 40°C, for a total temperature of 120°C. Above this temperature, another 10°C rise is allowed for the hottest spot, for a total overall temperature limit of 130°C - the rating of the material.

47

Actual tests on these motors at rated load show a steady-state temperature rise by resistance between 50 and 52°C for 6 to 7 hours, with the test ambient temperature of 27°C. With an ambient air temperature of 40°C, the total motor winding temperature would be 40 + 52 = 92°C maximum. Even if the ambient air temperature were 50°C (122°F), the total temperature of 102°C would still be well below the allowable 120°C.

This analysis, along with the one given on page 3.11-4, show that these motors will operate within their environmental capabilities.

3. <u>Motor Control Centers</u> - The Class 1E motor control centers are 480 volt, 3 phase, and are located in a section of the Auxiliary Building where the environment is clean, dry, ventilated, and free of radiation. The ambient air temperature near this equipment does not exceed 40°C, except as noted on page 9.4-1.

This equipment has been designed, manufactured, and tested to NEMA standard, ICS-1970, Industrial Controls and Systems, and also tested according to IEEE Standard No. 74, Test Code for Industrial Control. Both of these standards contain test plans, set up, procedures, and acceptability requirements. Included are tests for temperature, dielectric strength, durability, operating performance, and short circuit capability. Additional standards with test codes are applied for the critical components.

4. <u>Electrical Cable</u> - Insulated electrical conductors were qualified by tests made by manufacturer to the requirements of PG&E specification and the standards of IPCEA and AEIC.

A typical example of design specification and test requirements for electrical cable condensed from PG&E purchase specifications follows:

ETHYLENE-PROPYLENE INSULATED LOW VOLTAGE POWER CABLE

(TYPICAL SPECIFICATION)

1. Design Specification Requirements

A. General

1. Cables comply with AEIC Specifications, applicable requirements of IPCEA-NEMA Standards publications.

2. Cables shall be rated 90°C continuous conductor temperature.

B. Conductors

1. Conductor metal shall be copper with class B conductor stranding. Conductor shall be tin or alloy coated and shall be in accordance with ASTM B3.

C. Insulation

1. Insulation to be of high quality, high dielectric strength, ethylene-propylene insulation compound, flame retardant, resistant to heat, moisture, ozone, and corona and suitable for operation in wet and dry locations at a maximum conductor temperature of 90°C for normal operation, 130°C for emergency overload conditions, and 250°C for short-circuit conditions. As a minimum, the insulation shall meet the requirements specified in Table I below.

2. Insulation minimum average insulation thickness shall be equal to or greater than that specified in Table II. The minimum thickness at any point shall not be less than 90 percent of the minimum average thickness.

2. The STRUDL program is a recognized program in the public domain, and has had sufficient history of use to justify its applicability and validity without further demonstration. The dated program version, the software or operating system, and the computer hardware configuration are as follows: 3

Massachusetts Institute of Technology
Integrated Civil Engineering System
STRUDL II, Version 1, Modification 1
IBM 360/85, OS release 20.6

Structural Acceptance Criteria

Normal Loads

For normal loads the Auxiliary Building is designed for the allowable stresses of ACI 318-63 and the AISC specification for the Design, Fabrication, and Erection of Structural Steel for Buildings except that the increase in allowable stress usually allowed for load combinations involving earthquake forces is not used.

Abnormal Loads

For abnormal loads the Auxiliary Building is designed for overall elastic behavior. The capacity of the various structural elements is based on the yield strength of the material, reduced by a factor, Ø, which provides for the possibility that small, adverse variations in material strengths, workmanship, dimensions and control, while individually within required tolerances and the limits of good practice, occasionally may be additive.

The Ø factors used are:

Reinforced concrete	Ø factors in accordance with ACI 318-63
Structural steel	Ø = 1.00

For load combinations involving Y_R, Y_J, and Y_M, local stresses due to those concentrated loads may exceed the allowables provided there is no loss of function.

Factors of Safety

The factors of safety for the structural elements of the Auxiliary Building are at least as great as indicated by the load factors given in this Subsection 3.8.1 under the heading "Loading Combinations." The calculated stresses for the Auxiliary Building are given in Table 3.8-4. In most cases, these stresses are less than the allowable value indicated in the table and the actual factor of safety is therefore greater than indicated by the load factor equation.

The gap between the Auxiliary Building and the Containment Structure is 5 inches. As discussed in Subsection 3.8.2, this provides a factor of safety of 2.38 against the structure impacting during a Double Design Earthquake.

The gap between the Auxiliary Building and the Turbine Building is 8 inches. As discussed in Section 3.7, the Turbine Building, although Design Class II, has been analyzed dynamically for the Double Design Earthquake. The maximum relative displacement between the Auxiliary Building and the Turbine Building is 1.38 inches, thus providing a Factor of Safety of 8.0/1.38 = 5.84 against impact of the structures during a Double Design Earthquake.

Materials and Quality Control

The Concrete and Reinforcing Steel sections of Materials and Quality Control for the Containment Structure also apply to the Auxiliary Building except as superseded by information in the following paragraphs.

Concrete

The concrete strengths used in the Auxiliary Building are:

Above elevation 85 and all columns	5,000 psi
Other	3,000 psi

The average strengths and coefficients of variation of representative mixes as of December 14, 1972 are:

Design Strength	Average 28 Day Strength	Coefficient of Variation	Number of Tests
3,000 psi	3,960 psi	9.1%	103
5,000 psi	5,710 psi	7.5%	166

These coefficients of variation represent "excellent control," as defined in Table 2 of ACI 214-65.

Reinforcing Steel

Reinforcing steel is ASTM A 615, Grade 40 except in some columns Grade 60 is used.

The average and minimum properties of representative bar sizes, as of December 12, 1972, are as follows:

	#8		#11	
	Grade 40	Grade 60	Grade 40	Grade 60
Average Yield Strength, psi	49,655	66,189	48,302	68,582
Minimum Yield Strength, psi	41,200	60,250	42,950	61,710
Average Tensile Strength, psi	82,236	102,403	81,074	105,822
Minimum Tensile Strength, psi	74,392	96,500	72,940	94,390
Average Elongation, %	18.9	13.94	14.82	14.41
Minimum Elongation, %	13.0	11.00	8.5	9.4
Total Number of Heats	67	18	91	56

Splices

The majority of splices in the Auxiliary Building are lap splices made in accordance with ACI 318-63.

Cadweld splices are used in some locations in the Auxiliary Building. The quality control procedures described for Cadweld splices in the Containment Structure also apply to Cadweld splices in the Auxiliary Building.

Butt welded splices are used where a section of wall has to be temporarily left open for access, and in certain other locations. Butt welded splices are made in accordance with ACI 318-63 and the American Welding Society's Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction using the "short-arc" process or low hydrogen stick electrodes by the shielded arc process. Both processes have minimum preheat and interpass temperatures of 400°F.

Completed welds are wrapped with a protective blanket of insulating material to avoid rapid cooling.

Procedure qualification and welder qualification are as follows:

1. A welding procedure qualification test is made for each position and for each grade and size of bar. The test consists of two tension tests and one nick break test. Bars may not be rolled during welding.

2. Welder qualification tests are made for each position, type of electrode, grade and size of bar, and joint design. Qualification for one size of bar is considered qualification for all smaller sizes. Each test consists of one tension and one nick break test. Bars may not be rolled during welding.

3. Tension specimens are tested to failure and must comply with the minimum tensile requirement for the grade of reinforcing steel.

4. The nick break specimen is broken and visually examined for soundness. The specimen must exhibit the following: The sum of the longest dimension of all inclusions visible in any one joint must not exceed ½ inch; no inclusion may be closer to the weld surface than a distance equal to the largest dimension of the inclusion; there must be no incomplete fusion or lack of penetration or cracks in the weld or bare metal.

Testing percentages applicable to butt-welded splices for each welder, position, and grade of bar are as follows:

Two out of the first ten splices.
Six out of the next 90 splices.
Four out of second and subsequent 100 splice units.

3 Qualification for one size of bar is considered as qualification for all smaller sizes.

Structural Steel

Structural steel is ASTM A 36; ASTM A 441; ASTM A 516, Grade 70; or ASTM A 572, Grade 42. Charpy impact tests were performed on all structural steel at the following temperatures:

1.	Framing for pipe rupture restraints	40°F
2.	Structural steel embedded in concrete	20°F
3.	Structural steel exposed to atmosphere, other than described in item 1.	0°F

Foundations For Design Class I Tanks

The following Design Class I tanks are located adjacent to the east side of the Auxiliary Building on reinforced concrete foundation slabs:

Condensate Water Storage Tank (one for each unit)
Refueling Water Storage Tank (one for each unit)
Firewater Tank (one for both units)
Transfer Tank (one for both units)

The design of these tanks above the foundation slab is discussed in Section 3.9. The foundation slab design is discussed in this section as follows:

Description of the Foundation Slabs

Each of the Condensate Water Storage Tanks and Refueling Water Storage Tanks has a separate, circular foundation slab. The Firewater Tank and the Transfer Tank, which serve both units, are concentric tanks on a common circular
3 foundation slab. Each of the foundation slabs is as shown in Figure 3.8-0P and consists of a 1'-0" thick reinforced concrete slab with an integral edge beam. Each of the tanks, except for the Firewater Tank, is anchored to its

foundation with 63 anchor bolts which are ASTM A 193, Grade B7. The bolt diameters are 1 1/4 inch for the Condensate Water Storage Tanks and the Transfer Tank and 1 3/8 inch for the Refueling Water Storage Tanks. The wall of the Firewater Tank is welded to an insert plate in the foundation. The tank foundation slabs rest on engineered fill which varies in depth from ten to twenty feet. The engineered fill is placed in horizontal layers of 8 inch maximum depth and compacted to 95% of maximum density.

Codes and Standards

The foundation slabs for the Design Class I tanks listed are designed and constructed in accordance with the ACI Standard Building Code Requirements for Reinforced Concrete (ACI 318-63).

Loads and Loading Combinations

The foundation slabs are designed for dead load, including weight of the water, and Double Design Earthquake. The load combination used is

C = D + DDE

where C = total load on foundation
D = dead load of tank including weight of water
DDE = load resulting from the Double Design Earthquake.

Design and Analysis Procedures

The size of the foundation slabs is selected so that the soil bearing pressure determined from an analysis based on equilibrium considerations, is limited to a safe value.

The anchor bolts attaching the tank to the slab are designed to develop the yield strength of the bolts.

Structural Acceptance Criteria

Stresses in the reinforced concrete slabs are limited to the allowable values in ACI 318-63.

Materials and Quality Control

The quality control measures discussed in the Concrete and Reinforcing Steel sections of Materials and Quality Control for the Containment Structure also apply to the Design Class I tank foundations.

Material strengths for the Design Class I tank foundations are:

Concrete

Concrete strength is 3,000 psi.

Reinforcing Steel

Reinforcing steel is ASTM A 615, Grade 40.

Anchor Bolts

Anchor bolts are in accordance with ASTM A 193, Grade B7.

Design Class II Structures Containing Design Class Equipment

The Turbine Building and the Intake Structure are Design Class II structures that contain Design Class I equipment. The Turbine Building contains the Component Cooling Heat Exchangers, the Emergency Diesel Generators and the 4.16 kv vital switchgear. The Intake Structure contains the Auxiliary Salt Water Pumps and conduits. In order to assure that the Design Class I equipment would not be affected by failure of the Design Class II structures, both the Turbine Building and the Intake Structure are analyzed for the Double Design Earthquake using dynamic analysis methods. These analyses are discussed in Section 3.7.

3.8.2 CONTAINMENT STRUCTURE

Description of the Containment

The reactor containment for each unit is a cylindrical, reinforced concrete structure that completely encloses the reactor and Reactor Coolant System. It assures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the Reactor Coolant System were to occur simultaneously with an earthquake of an intensity twice the maximum postulated.

The Containment Structures for Units 1 and 2 are essentially identical, except for orientation. The following discussion applies to either unit.

The concrete outline and equipment locations are shown in Chapter 1. The exterior shell consists of a 142 foot high cylinder, topped with a hemispherical dome. The cylinder wall is 3 feet, 8 inches thick, and the dome is 2 feet, 6 inches thick. Both have an inside diameter of 140 feet. The base is a circular slab 153 feet in diameter and 14 feet 6 inches thick, with the reactor cavity near the center. The inside of the dome, cylinder, and base slab is lined with welded steel plate which forms a leaktight membrane. The liner is 3/8-inch thick on the wall and dome and 1/4-inch thick on the base slab.

The internal concrete structure approximates a 106 foot diameter, 51 foot high cylinder, with a slab on top. However, there are multiple openings and walls, such as the reactor support and the stainless steel lined refueling canal, which complicate the shape. The walls and top slab are generally 3 feet thick. This structure provides support for the reactor and components of the Reactor Coolant System, provides radiation shielding, and provides protection for the liner from postulated missiles originating in the Reactor Coolant System.

A polar crane is mounted on top of the internal concrete cylinder wall. The support of the polar crane, its connection to the concrete, and provisions to 3
resist seismic forces are shown in Figure 3.8-15I and described in Subsection 9.1.4.

The piping and electrical connections between equipment inside the Containment Structure and other parts of the plant are made through specially designed, leaktight penetrations. In addition to the piping and electrical penetrations, other penetrations are the 18-foot 6-inch diameter equipment hatch, the 9-foot 7-inch diameter personnel hatch, the 5-foot 6-inch diameter emergency personnel hatch, and the fuel transfer tube.

The 6-foot 7-inch by 13-foot ventilation duct is attached to the outside of the structure, extending from an elevation 25 feet above the base slab to the top of the dome. The duct is fabricated from steel plate with stiffeners.

A system of lightning rods is installed on the dome to protect against lightning damage.

The plant is located on rock which is described as "a stratified sequence of fine- to very-fine-grained sandstone which is deeply weathered to a depth of 15 feet." This rock is of sufficient strength to carry all foundation loads imposed by plant structures as described in Section 2.5.

The following paragraphs describe the various parts of the structure.

Exterior Shell

Reinforcing Steel

The reinforcing steel arrangement is designed to provide continuous reinforcement for tensile and shear membrane forces in the cylinder and dome. The reinforcing in the cylinder wall consists of hoop bars, which are horizontal, and inclined bars, which are oriented 30° from the vertical. In Figure 3.8-1, layers (4) and (6) are the No. 18 hoop bars, spaced at 8½ inches center-to-center vertically, and layers (3) and (5) are the inclined No. 18 bars spaced at 8½ inches center-to-center, measured normal to the bars.

The dome reinforcing is accomplished by extending the inclined bars past the springline and over the dome. After crossing the dome, the same bar once again becomes an inclined bar in the cylinder. A layer (3) bar becomes a layer (5) bar after crossing the dome, as shown in Figure 3.8-2. No inclined bars are terminated at the springline or in the dome.

The dome steel layout is based on the division of a sphere into 20 equilateral spherical triangles as shown in Figure 3.8-3. At the springline, two sides of the triangles make an angle of 30 degrees with the vertical. Thus, an inclined cylinder bar is parallel to the sides of the triangles at the springline. The inclined cylinder bars are extended into the dome so that they are always parallel to one side of a spherical triangle. Figure 3.8-4 shows the five types of bars in the dome. When these five types are superimposed, there are three layers of reinforcing steel at every point above the pentagon ABCDE in Figure 3.8-3. Below pentagon ABCDE, the inclined bars make up two layers at every point, and bars similar to the cylinder hoop bars are used to provide reinforcing in the third direction.

Layers (1) and (2) (Figure 3.8-1) are inclined at 30 degrees to the vertical and extend from the base slab to elevation 172. These bars, which are spaced at 17 inches center-to-center, provide additional capacity for earthquake forces. Above elevation 170, No. 4 bars are spaced at 12 inches center-to-center horizontally and vertically.

Splices

All No. 18 bars are spliced by Cadwelding.

The Cadweld process employs a steel sleeve with grooves on the inside. The sleeve is placed over the ends of the bars to be spliced, and the space between the bars and the sleeve is filled with molten metal alloy. The force in the reinforcing bar is transferred to the Cadweld sleeve by shear in the metal alloy. Past experience, testing, and evaluation have shown that the use of the Cadweld process results in consistent, acceptable splices which have ductility very similar to that of the reinforcing steel.

All Cadweld splices in the No. 18 bars are made with "T-Series" sleeves which are designed to develop the full tensile strength of the bar.

As a general rule, splices are staggered a minimum of 3'-0". In a very few instances, staggering of splices is reduced because of local conditions.

Liner

All seams are full penetration butt-welded, and are covered with welded steel channels on the inside of the structure. These "leak chase" channels are designed to provide a sensitive and accurate means of detecting leakage. They are arranged in zones so that one zone at a time can be pressurized to test the leaktight integrity of the liner plate welds. These leak chase zones can be retested at any time in the future.

The liner in the dome and cylinder wall is anchored by welded studs which extend into the concrete wall past the innermost layers of reinforcing steel. Three types of studs are used: "L" shaped, 3/8-inch diameter, with an 8½-inch shaft and a 4-inch arm; threaded, 3/8-inch diameter with an 8½-inch shaft; and threaded, 1/2-inch diameter with an 11-inch shaft. All threaded studs are provided with an anchorage, and provide resistance to pullout that is equal to or better than the 3/8-inch stud with a 4-inch arm. The studs are spaced a maximum of 19.6 inches on center (plus a placement tolerance of 1/2 inch) in a pattern that is compatible with the reinforcing steel. See Figure 3.8-5.

Penetrations

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the Containment Structure liner. The pipe, electrical conductor cartridge, duct, or access hatch passes through the embedded sleeve and one or both ends of the resulting annulus are closed off by welded end plates, bolted flanges, or flued heads. Typical electrical and piping penetrations are shown in Figures 3.8-6 through 3.8-10 and the fuel transfer tube penetration is shown in Figure 3.8-11. The penetrations are designed to maintain the same high degree of leaktight integrity afforded by the Containment Structure itself.

Electrical Penetrations

"Cartridge" type penetrations are used for all electrical conductors passing through the containment boundary. Cast epoxy-to-metal seals are used to provide a double pressure barrier for each conductor as shown in Figures 3.8-8 and 3.8-9. The penetrations are provided with a connection to allow periodic testing of the volume between the end plates, and the weld connecting the sleeve to the liner plate has a leak chase channel. There are a total of 45 electrical penetrations, some of which have more than one conductor.

Piping Penetrations

Piping penetrations are provided for all piping passing through the containment boundary. Typical piping penetrations are shown in Figures 3.8-6 and 3.8-7. Several small pipes may pass through a single embedded sleeve to minimize the number of penetrations required. Welded end plates or flued heads are used to provide end closure. The welded joints are covered with a leak chase channel to allow periodic testing. The weld connecting the sleeve to the liner plate also has a leak chase channel.

Pipes carrying hot fluids through penetrations are designed to maintain the temperature of the concrete adjacent to the sleeve below 200°F under normal operating conditions.

Pipes and penetrations are anchored as required to resist the forces and movements incident at the penetration under normal and accident conditions and to limit the loads imposed on the Containment Structure liner. Piping loads are transferred to the penetration sleeve and thence to anchors in the concrete wall rather than to the Containment Structure liner.

Equipment and Personnel Access Hatches

The equipment hatch is furnished with a double-gasketed flange and bolted dished door. Equipment up to a diameter of approximately 18 feet can be transferred into and out of the Containment Structure through this hatch.

The hatch barrel is embedded in the Containment Structure wall and welded to the liner. Provision is made for pressurizing the space between the double gaskets of the door flanges and the weld seam channels at the sleeve-to-liner joint. The two personnel hatches are double door, hydraulically-latched, welded steel assemblies. A quick-acting type equalizing valve connects each personnel hatch with the interior of the containment vessel for the purpose of equalizing pressure in the two systems when entering or leaving. The personnel hatch doors are interlocked to prevent simultaneous opening. Remote indicating lights and annunciators situated in the control room indicate the door operational status. Provision is made to permit bypassing the door interlocking system to allow doors to be left open during a plant cold shutdown. Each door hinge is capable of independent three-dimensional adjustment to assist proper seating. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior, or to the outer door, from outside, is possible by the use of special door unlatching tools. All doors on the personnel hatches are double-gasketed and provided with fittings to allow pressurization of the space between the double gaskets.

Special Penetrations

1. Fuel Transfer Tube Penetration

A fuel transfer tube penetration is provided for fuel movement between the refueling canal in the Containment Structure and the spent fuel pool. The penetration consists of a 20-inch diameter stainless steel pipe installed inside a 24-inch diameter pipe sleeve as shown in Figure 3.8-11. The inner pipe acts as the transfer tube and is fitted with a double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring to permit pressure testing all welds essential to the integrity of the penetration. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures.

2. Containment Supply and Exhaust Purge Ducts

The ventilation system purge duct is equipped with two quick-acting tight-sealing valves (one inside and one outside the containment) to be used for isolation purposes. These valves are normally closed during reactor operation. They are manually opened for containment purging but are automatically closed upon a signal of high containment pressure or high containment radiation level. The space between the valves can be pressurized to check the integrity of the penetration. In addition, the shaft seals of the purge valves are equipped with double seals with provision for testing the space between.

3. Spare Penetrations

Capped spare penetrations are provided. The welds between the sleeve and the liner and between the sleeve and the cap are covered with leak chase channels.

General

All spaces that are equipped for pressurization on penetrations and penetration sleeves are included in the same system of pressurization zones as the liner seam leak chase channels.

For all penetrations except the equipment and personnel hatches, the No. 18 reinforcing bars are deflected around the opening. For the equipment and personnel hatch openings, a 2½-inch thick structural steel, hexagonal collar (hex collar) is provided to transfer the reinforcing bar forces around the opening as shown in Figures 3.8-12 and 3.8-13. The reinforcing bars are Cadwelded to special studs threaded into the 4-inch thick edge of the hex collar.

For all penetrations in the exterior shell, a thickened insert plate is welded into the liner.

Base Slab and Shell-Base Slab Connection

The seams on the base slab and reactor cavity liner are full penetration butt-welded and are covered with leak chase channels. The leak chase channels are arranged in zones in the same manner as those on the exterior shell liner.

There are two penetrations through the base slab for recirculation lines. These are similar to penetrations used in the exterior shell. Weld seams between the liner and the penetration sleeve and between the penetration sleeve and internal are covered with leak chase channels. The volume in the end of the penetration internal has a fitting for pressurization. These leak chase channels and the volume in the end of the penetration internal are connected in the zones of pressurization used for liner leak chase channels.

The detail of the shell-base slab connection is shown in Figure 3.8-14. The vertical wide flange steel beams provide a gradual transition of load carrying elements between the base slab and the cylinder, and resist the radial bending moments and shears. The beams are keyed and grouted in a groove at the base slab and extend approximately 20 feet up the wall. They do not participate in resisting either uplift due to pressure or shear and tension forces due to earthquake.

The 3-foot 8-inch thick cylinder wall is designed to offer minimum bending resistance at the junction with the base slab. To achieve this, the wall is divided into three layers, with the contact surface between the layers designed as a slip surface. The 12-inch inner layer, next to the liner plate, provides stiffness to the liner plate. The L-shaped stud anchors, welded to the liner plate, and layers (1) and (2) of the wall reinforcing bars are in this layer. The middle layer is the wide flange steel beams. The voids between the beam webs are filled with concrete. The outer layer is 20 inches thick. Layers (3) through (6) of the wall reinforcing bars are in this layer. The slip surface between layers is provided by covering both flanges of the steel beams with two sheets of Johns-Manville #60 asbestos sheet packing. This packing is graphite coated on one side, and the two sheets are placed with the graphite coated sides in contact. The Company has successfully used this means of providing sliding supports on penstock piers for over 25 years. The inert nature of the material, and the fact that it will be completely isolated from the atmosphere by a minimum of 20 inches of concrete combine to assure that it will be fully effective throughout the lifetime of the plant.

The detail at the bottom of each of these three layers is shown in Figure 3.8-14. The innermost and outermost layers have a 1-inch neoprene pad to allow slight rotation without crushing of the concrete. The center layer, consisting of the beams, has a 5-inch deep pocket in which the beams are placed and grouted.

The diagonal wall reinforcing extends to the bottom of the base slab for anchorage, as shown in Figure 3.8-15. The base slab bars are bent up at 45 degrees and passed through the diagonal bars. The ends of the base slab bars are provided with a mechanical anchorage consisting of a Cadweld sleeve and a steel plate.

The shell liner is anchored to the base slab by No. 14 rebar welded to the bottom course liner plate, which is 3/4-inch thick. These rebars are embedded 7.5 feet in the base slab concrete.

<u>Internal Structure</u>

The internal structure which is shown in the Figures 3.8-15A through 3.8-15G consists of the following parts: 3

1. Lower operating floor at elevation 91 is a 2-foot thick concrete slab placed over the Containment Structure base slab liner.

2. Circular crane wall is a 3-foot thick, 106-foot OD reinforced concrete wall, concentric with the exterior shell, and extending vertically from the Containment Structure base slab liner at elevation 89 to the main operating floor at elevation 140. The runway for the 200-ton polar gantry crane is located on top of the circular crane wall. This wall is anchored to the Containment Structure base slab by No. 18 reinforcing bars. This

anchorage is developed through the Containment Structure base slab liner by means of Cadweld sleeves welded to each side of the liner at the same locations.

3. Reactor shield wall is a 34-foot OD, 17-foot ID reinforced concrete wall. This wall is anchored to the Containment Structure base slab in the same manner as the circular crane wall.

4. Fuel transfer canal is a stainless steel lined cavity which can be filled with water during refueling. The vertical walls of the fuel transfer canal are 4 feet thick.

5. Main operating floor at elevation 140 is a 3-foot thick concrete slab supported by the circular crane wall and the fuel transfer canal walls. This slab is thickened locally up to 7 feet near openings.

6. Main steam line restraint towers are reinforced concrete buttresses extending from the main operating floor at elevation 140 to elevation 184.

7. Annulus platforms are structural steel platforms at elevations 117 and 140 located between the circular crane wall and the exterior shell. Steel framing is also provided at elevations 106'-8" and 101'-5" for support of piping.

Codes and Standards

The following codes and standards are used, insofar as they are applicable in the design and/or construction of the Containment Structure:

1. ACI Standard Building Code Requirements for Reinforced Concrete (ACI 318-63).

2. Manual of Standard Practice for Detailing Reinforced Concrete Structures (ACI 315-65).

3. Recommended Practice for Evaluation of Compression Test Results of Field Concrete (ACI 214-65).

4. Inspection of the Cadweld Rebar Splice (Erico Products, Inc., RB-5M 768).

5. Recommended Practices for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction, American Welding Society, AWS D 12.1-61.

6. AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, Sixth Edition.

7. Construction of the Containment Structure liner conforms to the applicable parts of Part UW, "Requirements for Unfired Pressure Vessels Fabricated by Welding," Section VIII, ASME Boiler and Pressure Vessel Code, 1968 Edition including addenda through Summer 1968.

8. Those parts of penetration insert plates, penetration sleeves, airlocks, and access hatches, which form part of the pressure boundary conform to Class B requirements of Section III, ASME Boiler and Pressure Vessel Code, 1968 Edition, including addenda through Summer 1968.

9. Code for Welding in Building Construction, AWS D 1.0-69. Work performed prior to December 12, 1969 is in accordance with the earlier edition AWS D 1.0-66.

10. Stud welding is in accordance with the Supplement to American Welding Society Specifications AWS D 1.0-66 and AWS D 2.0-66 on Requirements for Stud Welding.

11. Materials and the quality control tests for materials conform to ASTM standards.

12. Pressure tests of the Containment Structure, leak chase channels, double penetration volumes, volumes between double seals, and volumes between double isolation valves are in accordance with the requirements of ANS 7.60 Standard for Leakage Rate Testing of Containment Structures for Nuclear Reactors, dated April 29, 1970.

13. Safety Guide 12 Instrumentation for Earthquakes, dated March 10, 1971.

14. Safety Guide 18 Structural Acceptance Test for Concrete Primary Reactor Containments, dated October 27, 1971.

8 Regulatory Guides

The following Regulatory Guides were issued after construction at the Diablo Canyon plant was partially completed:

1. Safety Guide 10, Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments, dated March 10, 1971.

2. Regulatory Guide 1.15, Testing of Reinforcing Bars for Category I Concrete Structures, dated December 28, 1972.

3. Safety Guide 19, Nondestructive Examination of Primary Containment Liner Welds, dated August 11, 1972.

4. Regulatory Guide 1.55, Concrete Placement in Category I Structures, dated June 1973.

Inasmuch as the corresponding programs for the Diablo Canyon plant were conservatively formulated, the inspection provided essentially equals, and in many cases exceeds, that provided by the regulatory position in the guides. Detailed comparisons of the program used for Diablo Canyon plant with the regulatory position of Safety Guide 10, Regulatory Guide 1.15, and Safety Guide 19 are presented in Tables 3.8-1 through 3.8-3, respectively. The quality assurance program for the Diablo Canyon plant meets the requirements of Regulatory Guide
8 1.55. In regard to Regulatory Guide 1.55, the references used for guidance are those listed in Appendix A as they existed at the time of the PSAR.

Proposed ACI-ASME Code for Containments

The Proposed Standard Code for Concrete Reactor Vessels and Containments (proposed code) was issued in the spring of 1973 for Trial Use and Comment as the Proposed Section III, Division 2, of the ASME Boiler and Pressure Vessel Code. The Trial Use and Comment period ended November 28, 1973 and it is expected that the proposed code with the completed addenda will be forwarded to the ACI and ASME for approval in the first half of 1974. The approved code should be available by September 1974.

The technical requirements of the proposed code are derived from the Building Code Requirements for Reinforced Concrete (ACI 318-71), from Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, and from other codes and standards commonly applied to containment structure design, fabrication, and examination. Likewise, the technical requirements for the Diablo Canyon containment structures are based on those same codes and standards, except that in many cases an earlier edition was applied to Diablo Canyon in accordance with the PSAR. As a result, the corresponding requirements for the Diablo Canyon containment structures are essentially equal to the technical requirements of the proposed code. In addition, there are some significant areas such as the amount of liner radiography where the Diablo Canyon requirements are more stringent than those of the proposed code.

Tables 3.8-1, 3.8-2, and 3.8-3 compare the Diablo Canyon programs for reinforcing steel, Cadweld splices, and nondestructive examination of the liner with the regulatory position in Regulatory Guide 1.15, Safety Guide 10, and Safety Guide 19, respectively. Inasmuch as the technical requirements of the proposed code are identical to the requirements of those Safety and Regulatory Guides, the comparisons also apply to the proposed code. These comparisons are considered representative of the degree to which the corresponding requirements for the Diablo Canyon containment structures are equal to the technical requirements of the proposed code. 3

The general requirements of the proposed code require third party inspection for all containment structure fabrication and construction. For Diablo Canyon, third party inspection was provided for fabrication and installation of all containment structure penetrations in accordance with the Class B requirements of Section III, ASME Boiler and Pressure Vessel Code.

Design Loads

The following loads were considered in the design of the Containment Structure:

Dead Loads

These consist of the weight of concrete, reinforcing steel, steel liner, structural steel, and permanent equipment loads. Equipment loads are supplied by the manufacturers.

Live Loads

These consist of temporary equipment loads and a uniform load to account for the miscellaneous temporary loadings that may be placed on the structure.

Internal Pressure Due to Loss of Coolant Accident

In the hypothetical loss of coolant accident used for design of the Containment Structure, the water in the Reactor Coolant System is assumed to be released through a double-ended break in a reactor coolant pipe, thereby creating a rapid rise in internal pressure. The variation of internal pressure with time used for design purposes is shown in Figure 3.8-16. The design peak pressure is 47 psig, which is greater than any of the peak pressures calculated in the detailed analysis reported in Chapter 6. Transients for the 1.25 times and 1.50 times design pressure are shown in Figures 3.8-17 and 3.8-18.

The hypothetical loss of coolant accident results in a pressure differential between the volume within the circular crane wall and the surrounding Containment Structure volume. For design purposes, the maximum pressure differential was taken as 15 psi. This value is greater than the value calculated in the detailed analysis reported in Chapter 6.

Loads Due to Thermal Expansion

These are loads resulting from the internal temperatures associated with normal operation and the hypothetical loss of coolant accident. The maximum internal atmospheric temperature during normal operation is 120°F. The temperature transients, associated with the hypothetical loss of coolant accident, which were used for design of the Containment Structure are shown in Figures 3.8-16, 3.8-17, and 3.8-18 corresponding to design pressure, 1.25 times design pressure, and 1.50 times design pressure, respectively. The maxima of the temperature transients are 246°F on the transient corresponding to design pressure, 285°F on the transient corresponding to 1.25 times design pressure, and 306°F on the transient corresponding to 1.50 times design pressure.

Loads Due to Postulated Pipe Ruptures and Missile Impact

Design of the internal structure includes calculation of the effects of forces from postulated pipe ruptures transmitted through pipe restraints and equipment supports, jet forces from postulated pipe ruptures, and forces resulting from postulated missile impact. The forces from postulated pipe ruptures are calculated as described in Section 3.6. The forces from postulated missile impact are calculated as described in Section 3.5.

Earthquake Loads

Earthquake loads are based on a time history modal superposition analysis of the Containment Structure and surrounding rock mass performed by John A. Blume and Associates, Engineers, as described in Section 3.7.2.

Wind Loads

Wind loads are determined in accordance with the criteria presented in Section 3.3. However, the forces due to wind are much less than those due to earthquake; consequently, seismic considerations, rather than wind, control the design of the Containment Structure.

Test Pressure

Internal pressure is applied to test the structural integrity of the vessel up to 115 percent of the design pressure. For this structure, the test pressure is 54 psig.

Negative Pressure

This consists of loading from an internal negative pressure of 3.5 psig. A pressure of this magnitude would result from the combined effects of cooling of the containment volume 70°F below the temperature at which the containment was sealed, a rise in external barometric pressure of 1 psi and the burning of hydrogen evolved from a metal-water reaction.

Loading Combinations

The following loading combinations are used in design of the Containment Structure elements:

Operating Conditions

Exterior Shell and Base Slab

Dead load, thermal load, Design Earthquake, and negative pressure are considered as follows:

$$C = D + T_o + DE + NP$$

where C = Required load capacity of section

D = Dead load of structure and equipment loads

T_o = Load due to operating temperature

DE = Load due to Design Earthquake

NP = Negative Pressure

Internal Structure

Dead load, live load, thermal load, and load from the Design Earthquake are considered as follows:

$$C = D + L + T_o + DE$$

where L = Live load

Accident Conditions

Exterior Shell and Base Slab

1. $C = 1.0D \pm 0.05D + 1.5P + 1.0T''$

2. $C = 1.0D \pm 0.05D + 1.25P + 1.0T' + 1.25DE$

3. $C = 1.0D \pm 0.05D + 1.0P + 1.0T + 1.0DDE$

Symbols used in these formulas are defined as follows:

C = Required load capacity of section.

P = Accident pressure as shown in Figure 3.8-16.

T = Load due to maximum temperature associated with 1.0P.

T' = Load due to maximum temperature associated with 1.25P.

T" = Load due to maximum temperature associated with 1.5P.

DDE = Loads resulting from the Double Design Earthquake.

Internal Structure

Dead load, live load, load due to Double Design Earthquake, compartment pressurization, pipe reactions associated with a postulated pipe rupture, jet forces, and missile loads are combined as follows:

$$C = D + L + DDE + CP + R + J + M$$

where CP = compartment pressurization associated with a loss-of-coolant accident.

R = pipe reactions associated with a postulated pipe rupture.

J = jet impingement load

M = missile impact load

Design and Analysis Procedures

Membrane Stress in Cylinder and Dome

For the accident loading conditions, the exterior shell is subjected to tensile membrane forces in the horizontal and vertical directions. Thus, the concrete is assumed cracked and its contribution to the strength of the exterior shell is neglected. The stress analysis is performed with two sets of assumptions: first, the effect of the liner is neglected and the stress
2 analysis is performed assuming that the reinforcing steel alone resists the tensile, compressive, and shear membrane forces; second, in order to investigate the states of stress in the liner, the analysis of the exterior shell is

performed assuming that tensile, compressive, and shear membrane forces are shared by the reinforcing steel and the liner. Where the liner is 2
considered, the strain in the reinforcing steel is made equal to the corresponding strain in the liner for dead load, pressure, and earthquake forces. The membrane forces from temperature are internal in nature and are calculated by considering that the compressive force in the liner is balanced by an equal tensile force in the reinforcing steel. At any point in the exterior shell, the sum of the membrane forces in the liner and the reinforcing steel is equal to the total membrane force from dead load, pressure, and earthquake. Thus, the liner is not relied upon for strength, but the stresses in the liner are 2
fully determined.

Since the thicknesses of the cylinder and dome are small in comparison with their radii of curvature, they are analyzed as a thin walled shell structure.

Internal Pressure and Dead Load

Membrane forces due to axisymmetric loads such as internal pressure and dead load are calculated as follows:

Vertical or meridional force, $N\phi$

$$N\phi = \frac{V}{2\pi\, r_o \sin\phi}$$

where V = sum of vertical loads above section considered.

r_o = distance of section considered from shell centroid.

ϕ = angle between the shell surface and a horizontal plane.

Horizontal or hoop force, No

$$\frac{N\phi}{r_1} + \frac{No}{r_2} = p$$

where r_1, r_2 = radius of curvature of shell in meridional and hoop directions, respectively.

p = unit load normal to the shell surface.

This equation becomes

$N\phi + No = pR$ for the dome and,

$No = pR$ for the cylinder

where R = radius of cylinder and dome.

The membrane forces for dead load and internal pressure are shown in Figure 3.8-19.

Earthquake

Membrane forces for the Design Earthquake and Double Design Earthquake are from the finite element, time history modal superposition analysis described in Section 3.7.2. A plot of these membrane forces in the cylinder and dome is shown in Figure 3.8-20.

Wind

Membrane forces from wind are shown in Figure 3.8-21. These are less than the membrane forces due to earthquake.

Temperature

Membrane forces from temperature are shown in Figure 3.8-22.

The combined membrane forces for the three accident loading conditions are shown in Figures 3.8-23, 3.8-24, and 3.8-25.

Liner Anchors

The liner anchors are designed so that they have sufficient strength and flexibility to withstand any combination of liner stress and deformation that can be reasonably assumed to occur under the accident loading conditions. The following conditions have been considered in design of the liner anchors:

1. Variations in anchor spacing including the possibility of a defective or missing anchor.
2. Variations in anchor stiffness.
3. Variations in liner plate curvature.
4. Liner plate thicker than nominal due to rolling tolerances.
5. Yield strength in excess of ASTM minimums.
6. Liner plate seam offset.

Equipment and Personnel Hatch Openings

Membrane forces are transferred around the equipment hatch and personnel hatch openings by means of hexagon-shaped steel collars to which the reinforcing steel is attached.

The analyses of the equipment hatch and personnel hatch openings take into account the following:

1. Direct stresses in the collar.
2. Stress concentrations in the reinforcing bars adjacent to the collar.
3. Bending stresses in the collar.

The analytical procedure for determining stresses is as follows:

1. For the first phase of analysis, a large area of the shell adjacent to the opening, including the steel "hex collar," is subjected to basic design membrane stresses, $N\phi$, No, $N\phi o$ (vertical, hoop, and tangential shear, respectively), as shown in Figure 3.8-26. This area extends beyond the opening far enough to make the effects of the opening negligible. The area is represented as a finite element mesh, consisting of bar and plate elements, and is analyzed using a plane stress finite element program. The objective of this analysis is to determine the primary stresses in the collar and reinforcing steel.

2. After completing the first phase of analysis, bending moments and shears are calculated in the region surrounding the opening. This is accomplished

by examining the equilibrium of the cylindrical shell. For a cylindrical shell, the basic equation of equilibrium is:

No = pR where No = hoop stress
p = internal pressure
R = radius of cylinder

Reinforcing steel and plate forces from the first phase of analysis are transformed into hoop forces which are then used in the above equation to solve for pressure. Where the calculated pressure exceeds the actual pressure, a hypothetical inward pressure, q, is applied as shown in Figure 3.8-26. This pressure is zero when No = pR. The total resultant of q is equal and opposite to the total resultant of the internal pressure acting over the area of the opening. Thus, overall equilibrium is maintained. The area subjected to the pressure q is treated as a flat plate supported, but free to rotate, at the edges of the circular opening. This area is represented as a finite element mesh, and a finite element plate bending program is used for the analysis, resulting in evaluation of bending moments $M\phi$ and Mo and torsional moments $M\phi o$ and $Mo\phi$, as defined in Figure 3.8-26.

Juncture of Cylinder and Base Slab

At the base of the cylinder, radial expansion from internal pressure is prevented by the base slab. In this region, the cylinder undergoes a transition from zero radial displacement at the base slab to full membrane displacement a short distance up from the base slab. This displacement results in longitudinal
3 curvature in the cylinder.

Because only limited data on the shear strength of concrete subjected to biaxial tension were available when the Unit 1 construction permit application was submitted, a system was developed to provide radial shear strength independently of the concrete. This system consists of structural steel, wide flange beams, embedded in the bottom 20 feet of the concrete cylinder wall and keyed into the base slab, as shown in Figure 3.8-14. These structural steel beams are continuous around the circumference of the cylinder and provide known bending and shear strength adequate to assure the integrity of the wall in the transition region.

For analysis, the bending and shear strength of the reinforced concrete parts of the wall are conservatively neglected. The displacement, slope, bending moment, and shear in the structural steel beams is calculated using the method outlined on page 46B of Reference 3. An analogy between this system and a homogeneous cylinder is assumed and expressed by the relationship:

$$\frac{d^4w}{dx^4} + 4\beta^4 w = 0$$

Where: w = radial displacement of cylinder

x = vertical coordinate measured from top of a rigid base

β^4 = coefficient representing the elastic properties and radius of cylinder 3

The above relationship represents the interaction of hoop forces and bending moments. Since the concrete is ignored in this analysis, the elastic properties of the cylinder are determined by the area (A_H) and spacing (s) of the hoop reinforcing and the moment of inertia (I) and spacing (b) of the steel beams. Thus,

$$\beta^4 = \frac{A_H\, b}{4R^2\, Is}$$

where R is radius to the centroid of the hop reinforcing bars.

Constants for the differential equiation are evaluated for boundary conditions of zero moment at the base and displacement at the base equal to the membrane displacement of the exterior shell. Solution of the differential equation results in an equation for the displacement w of the wide flange beams in terms of Q_o the shear at the base. The appropriate derivatives give the moments and shears. The resulting displacement, slope, moment, and shear are shown in Figure 3.8-27.

. Special provisions, described in this subsection in a previous paragraph entitled <u>Base Slab and Shell-Base Slab Connection</u> were made to enable the reinforced concrete wall to accept the calculated deformations at the transition region. The calculated deformations, and the related slopes, bending moments and shears are overestimates of the actual values to the extent that the concrete may add stiffness to the steel beams. The added stiffness should be minimal, however, because of the above special provisions and because the wall is subjected to vertical membrane tension while the maximum curvature of 3
the steel beams is not sufficient to develop compressive stress in the concrete. The calculated slope at the bottom of the cylinder wall is 1° 6'. This value is small enough that the reinforcing steel can readily accommodate the rotation.

The design of this transition region has not changed significantly since it was specifically reviewed and approved as part of the construction permit applications for Unit 1 and Unit 2.

Base Slab

For preliminary analysis, the base slab is treated as a uniform circular plate of radius, R = 73.0 ft. The analytical model is different for each of the three loading conditions as described below:

For the first accident loading condition, C = 1.0D ± 0.05D + 1.5P + 1.0T", the base slab is analyzed as a plate supported on a rigid non-yielding foundation. This assumption is considered reasonable since the foundation material is rock. The analytical procedure is given on Page 309 of Reference 3. The plate is subjected to a uniform load of pressure and dead weight (q) acting downward and a load distributed over the periphery (Q) acting upward due to internal pressure. This causes the base slab to "dish" so that the whole structure is supported at the center by a flat area of radius "x" as shown in Figure 3.8-28. The radial and tangential moments, and radial shear for this condition are also shown in Figure 3.8-28.

For the second accident loading condition, C = 1.0D ± 0.05D + 1.25P + 1.0T' + 1.25DE, there is a vertical and a lateral load due to earthquake in addition to the other loads. The lateral load results in an overturning moment which shifts the flat area away from center. For analysis, the overturning moment is represented as a couple consisting of a concentrated downward load at the edge, and an upward load at the center of the plate. This upward load, together with the upward earthquake force is considered equivalent to a peripheral load as shown in Figure 3.8-28. The same analytical procedure is used as for the first accident loading condition. Thus, on one-half of the plate, the distance x is reduced, and the plate moments increase correspondingly. The radial and tangential moments, and radial shear for this condition are shown in Figure 3.8-28.

For the third accident loading condition, $C = 1.0D \pm 0.05D + 1.0P + 1.0T + 1.0DDE$, the earthquake overturning moment is large enough that the flat portion in the symmetrical analysis becomes very small. For analysis, it is conservatively assumed that the flat portion disappears ($X = 0$, $Q = qR/2$). In this case, the slab acts as a simply supported circular slab as shown in Figure 3.8-28. The radial and tangential moments and radial shear for this condition are also shown in Figure 3.8-28.

The preliminary analysis shows that the third accident loading condition is the most severe. A more detailed final analysis is then performed for this loading condition in which the effect of the rigidity of the reactor cavity walls on the bending behavior of the plate is considered. The base slab is represented as a two dimensional finite element mesh, with reactor pit walls represented as very stiff elements. The loading consisted of the internal pressure, q, the equipment weight and the dead weight of the slab. A finite element plate bending program is used for the analysis.

Internal Structure

The internal structure consists of structural elements which are well defined by the "ACI Standard Building Code Requirements for Reinforced Concrete" such as "Beams," "T-Beams," "Deep Beams," "One and Two-way Slabs and Walls," or AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings." These methods are based on well established principles of equilibrium and strain compatibility as well as results of many years of experience by steel and concrete construction industries. Following are principal structural features and design methods of the Internal Structure:

Seismic forces are calculated as outlined in Section 3.7.

The operating deck at El. 140.0 is supported by 3-foot thick 106 foot OD circular crane wall, 4-foot thick fuel transfer canal walls and structural steel columns placed on the periphery next to the containment wall.

The slab within the circular crane wall is in general 3 feet thick considered as a two-way slab. Because of irregular shape, it is represented by approximate models with negative moments based clamped edges while positive moments are based on hinged edges.

Because of large openings it was necessary to thicken parts of the slab to 7 feet and these parts are treated as beams spanning between the circular crane wall and fuel transfer canal walls.

Outside the circular crane wall the operating deck consists of 1'-6" thick concrete slab supported on the circular crane wall and on steel beams on periphery; steel grating is placed over steel beams.

Lateral forces are transmitted to the circular crane wall through diaphragm action of concrete slabs.

The circular crane wall provides support for the operating floor at elevation 140'-0". It is also a primary system transmitting lateral loads into the base.

Compartment pressurization loads are carried primarily by tensile membrane hoop stress in the wall.

Computer Programs

The following computer programs were utilized for structural analysis of the Containment Structures:

1. A plane stress finite element program, developed at the University of California, was used in the analysis of the equipment hatch and personnel
3 hatch openings.

2. A plate bending finite element program, developed at the University of California, was used in the analysis of the equipment hatch and personnel hatch openings and the analysis of the Containment Structure base slab.

3. The STRAN program was developed to calculate membrane stresses in the reinforcing steel and liner of the Containment Structure exterior shell. The program calculations are based on equilibrium and compatibility of strains.

4. Dynamic analysis programs, described in Section 3.7, were used for the time history, model superposition analysis of the Containment Structure.

Verification of these computer problems was accomplished as follows:

Test problems were calculated to verify the two finite element programs and the STRAN program. For each test problem, the calculations were performed both by hand and by use of the computer program. The test problems were selected so that they are representative of the actual calculations performed for the Containment Structure. The test problem for the STRAN program and the plane stress finite element program are shown in Figure 3.8-28F, and a comparison of the results is shown in Table 3.8-9. The test problem for the plate bending finite element program is shown in Figure 3.8-28G. The comparison of results is shown graphically also in Figure 3.8-28G. 3

Structural Acceptance Criteria

The structural acceptance criteria for the Containment Structure exterior shell and internal structure are as follows:

Operating Conditions

For operating conditions the Containment Structure is designed for the allowable stresses of the applicable code such as ACI 318-63, AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, and ASME Boiler and Pressure Vessel Code, except that the increase in allowable stress or decrease in load factor usually allowed for load combinations involving earthquake or wind forces is not used.

Accident Conditions

For accident conditions the Containment Structure is designed for overall elastic behavior under all load combinations. The capacity of the various structural elements is based on the yield stress of the material, reduced by a factor, ϕ, which provides for the possibility that small, adverse variations in material strengths, workmanship, dimensions and control, while individually within required tolerances and the limits of good practice, occasionally may combine. The ϕ factors used are:

Exterior shell reinforcing steel, structural steel, and liner in tension	$\phi = 0.95$

Other structural steel	$\phi = 0.90$*
Reinforced concrete in base slab and internal structure	ϕ factors in accordance with ACI 318-63

For structural steel hex collars and liner plate which are stressed bi-axially, yield is determined by either the Tresca criterion used in Section III, ASME Boiler and Pressure Vessel Code or the Huber-Hencky-von Mises criterion.

Strain in the liner is limited to 0.005 in compression and 0.003 in tension.

Factors of Safety

The factors of safety for the exterior shell and internal structure of the Containment Structure are at least as great as indicated by the load factors given in this Subsection 3.8.2 under the heading "Loading Combinations." The calculated stresses for the exterior shell are given in Figures 3.8-28A through 3.8-28E, and the calculated stresses for the internal structure are given in table 3.8-5. In most cases, the calculated stresses are less than the allowable value and the actual factor of safety is therefore greater than indicated by the load combination equations.

The maximum relative displacement between the Containment Structure and the Auxiliary Building has been calculated as 2.1 inches. This includes the effects of accident pressure and temperature and Double Design Earthquake. Since the gap between the structures is 5 inches, the factor of safety against impact is 5/2.1 = 2.38.

The maximum relative displacement between the Containment Structure exterior shell and internal structure has been calculated as 0.79 inches. This includes thermal effects and deflections for the Double Design Earthquake. Since the minimum gap between the internal structure and the exterior shell is 2 inches, the factor of safety against impact is 2/0.79 = 2.54.

*See footnote in discussion of loading combinations in Section 3.8.1. 81

Materials and Quality Control

During the first 16 months of construction, a Civil Engineer from the Company's Engineering Department was assigned to the construction site on a full-time basis. This engineer was familiar with and had participated in the design of the Containment Structure. For the period he was on-site, he was part of the Quality Assurance Department (described in Chapter 17) and his responsibilities included performing audits on the various construction quality assurance programs. This engineer was qualified as ASNT Level II for radiographic, magnetic particle, ultrasonic, and dye penetrant methods of nondestructive testing. In addition other engineers from the Company's Engineering Department who were involved in the design of the Containment Structure maintained daily contact with the site by telephone calls and made periodic visits to the site during construction.

Inspectors from the Company's Engineering Department performed regularly scheduled shop inspections on materials and components for the Containment Structure.

The Company's General Construction Department provided a complete staff of Resident Engineers, Field Engineers, Quality Control Engineers, and Inspectors for supervision and inspection of contruction operations at the site. Their responsibilities for quality control of the Containment Structure were as follows:

1. To inspect materials delivered to the job site and examine supplier's certified test reports of physical and chemical properties.

2. To inspect handling and placing of concrete, reinforcing bars, embedded items, and forms.

3. To maintain an adequate force of qualified supervisory personnel at all times.

4. To maintain, as a part of its field engineering force, qualified personnel to perform a thorough inspection of each significant construction operation.

5. To supervise and be fully responsible for the quality of work performed by contractors.

6. To maintain records of inspections which were performed.

Many of the Company General Construction personnel at the site attended a formal course of instruction in radiographic, magnetic particle, ultrasonic, and dye penetrant methods of nondestructive testing.

Company technicians staffed the on-site materials laboratory where tests on cement, aggregate, concrete, and reinforcing steel were performed.

Concrete

Concrete is a dense, durable mixture of sound aggregate, cement, water, and such admixtures as may be found advantageous. The concrete strengths used in the Containment Structure are:

Exterior Shell	3,000 psi
Base Slab	5,000 psi
Internal Structure	5,000 psi

Concrete construction meets, as a minimum, the requirements of ACI 318-63, "Building Code Requirements for Reinforced Concrete."

Cement

Cement is clean, fresh, Type II, low alkali, moderate heat, portland cement conforming to the specifications of ASTM C 150, except that the Company specification is more stringent in requiring that the compressive strengths for any mill-run or bin be not less than 1,700 psi at three days, 2,700 psi at seven days, and 4,000 psi at 28 days, and that the loss on ignition be less than 2.0 percent. In addition, the following Optional Chemical Requirements of ASTM C 150 are required by the Company specifications:

1. Total alkalies of the cement, calculated as the percent of Na_2O + 0.658 times the percent of K_2O is limited to 0.60 percent.

2. The sum of tricalcium silicate and tricalcium aluminate was limited to 58 percent.

During manufacture, samples of cement were taken once each shift or at the rate of one sample for every 2,000 barrels. After the quality history was established in accordance with Section 5 of the Federal Test Method Standard No. 158a, testing was performed at the reduced testing rate specified in that standard. A report of the tests made on each sample was sent to the Company Department of Engineering Research in Emeryville, California. In addition, each shipment of cement was accompanied by a mill certificate and a report of the average of all the individual tests was sent with the initial delivery from each new lot or grind.

Cement shipped to the batch plant was not placed in a plant bin unless it had been accepted by the Company.

In addition to the tests the cement manufacturer performed, the Company made the following tests on each new lot to assure conformance with ASTM C 150:

ASTM C 109	Compressive Strength of Hydraulic Cement Mortars (using 2-in. cube specimens).
ASTM C 114	Chemical Analysis of Hydraulic Cement.
ASTM C 151	Autoclave Expansion of Portland Cement.
ASTM C 191	Time of Setting of Hydraulic Cement by Vicat Needle.
ASTM C 204	Fineness of Portland Cement by Air Permeability Apparatus.

The tests prescribed in ASTM C 114 were also performed periodically during storage to check for any effect on cement characteristics. These tests supplemented visual inspection during storage.

Aggregates

Aggregates consist of inert materials that are clean, hard, durable, free from organic matter, not coated with clay or dirt, and conforming to American Society for Testing Materials Designation C 33, "Standard Specification for Concrete Aggregates." In addition to the requirements of ASTM C 33, the Company specification requires that:

1. Sodium Sulfate Test for Soundness (ASTM C 88). For fine aggregate, the portion retained on a No. 50 screen be limited to a weighted average loss of no more than 8% after 5 cycles. For coarse aggregate, the weighted average loss after 5 cycles be no more than 10%.

2. Sand Equivalent Test (California Division of Highways Test Method No. California 217). Sand Equivalent value be at least 75.

3. The fineness modulus be within the limits of 2.6 to 2.9.

4. Los Angeles Rattler Test (ASTM C 131) for coarse aggregate. Loss by weight using Grading A, be a maximum of 10% by weight at 100 revolutions and 40% by weight at 500 revolutions.

5. Cleanness Value (California Division of Highways Test Method No. California 227-B) for coarse aggregate. Cleanness Value be at least 75.

6. Specific Gravity (ASTM C 127) for coarse aggregate. Specific Gravity on a saturated surface dry basis be at least 2.60.

7. The chloride content of aggregate be no more than 440 ppm.

The following tests were performed by the aggregate supplier at the frequency indicated:

Test	ASTM Designation	Frequency
Screen Analysis and Fineness Modulus	C 136	B
Clay Lumps and Friable Particles	C 142	D
Minus 200 Mesh	C 117	D
Organic Impurities	C 40	D
Soft Particles	C 235	D
Lightweight Particles	C 123	D
Specific Gravity	C 127 & 128	C
Absorption	C 127 & 128	C
Unit Weight	C 29	C
Los Angeles Abrasion (coarse)	C 131	E
Soundness	C 88	E
Effect of Organic Impurities on Fine Aggregate	C 87	F
Petrographic	C 295	F
Sand Equivalent Test	California Test Method 217	A
Cleanness Value	California Test Method 227-B	C

Frequency:

A. Once each 100 tons but not more than 10 nor less than one per day of production.

B. Once each 2,000 tons but not less than one test per week during production.

C. Every 10,000 tons or once every 10 days of production.

D. Every 20,000 tons or once every 20 days of production.

E. Once for initial source approval, then once per 30,000 tons.

F. Once per deposit.

All tests except the Soundness Test, ASTM C 88 and Soft Particles, ASTM C 235, were also performed by the Company on a periodic basis. Samples were taken at the place where the aggregate entered the batch bin.

Admixtures

Admixtures conformed to the following ASTM standards:

1. Pozzolan ASTM C 618
2. Air Entraining Agent ASTM C 260
3. Water Reducing Agent ASTM C 494 1.1.1, Type A

A certificate of compliance accompanied each load of admixture delivered to the construction site.

Water

Water is clean and free from deleterious amounts of silt, oil, acids, alkali, salts, and organic substances. Chlorides, calculated as Cl are limited to 1,000 ppm and sulfates, calculated as SO_4, are limited to 1,000 ppm.

Concrete Mixing, Placing, and Testing

The contractor was required to submit concrete mix designs meeting the Company specification requirements. The mixes were designed in accordance with Method 2, Section 308 of ACI 301. The Company's material testing laboratory made sample batches of the proposed mixes and tested them according to:

ASTM C 192	Making and Curing Concrete Test Specimens in the Laboratory.
ASTM C 39	Compressive Strength of Molded Concrete Cylinders.
ASTM C 143	Slump of Portland Cement Concrete by the Pressure Method.

For each design mix, 7-day and 28-day compressive strength tests were made on 6 x 12-inch cylindrical samples in the laboratory.

The contractor was required to submit lift drawings, which showed the location of all construction joints and embedded items, for approval by the Company. The lift drawings were approved prior to concrete placement.

At construction joints in all structural concrete, the surface of the hardened concrete was roughened to expose the coarse aggregate by either bush hammering, wet sandblasting, or cutting with an air-water jet. Prior to placing the next lift of concrete the surface of the hardened, cleaned concrete was wetted and given a 1/2-inch coat of bonding mortar on all horizontal joints. The bonding mortar had the same sand-cement ratio as the concrete mix and had a water-cement ratio such as to make a thick slurry but, at most, no greater than that for the concrete. Vertical joints in walls were provided with shear keys.

Vertical joints were staggered by at least 6 inches.

The concrete was batched and mixed in an automatic batching and mixing plant located at the construction site. Approved concrete mixes were punched on cards, and the appropriate card was inserted into the control console to initiate batching. The console automatically printed out the quantities of each material in the batch, the time, date, batch number, and mix identification for each batch. Prior to startup of the plant, all weighing equipment was certified. This equipment was periodically checked to assure continuing accuracy.

A full-time Company inspector checked the batching and mixing operation.

The maximum temperature of concrete at placement was as follows:

$55^{\circ}F$	Base slab
$70^{\circ}F$	Internal structure and exterior shell

The concrete was placed within 45 minutes after introduction of water to the mix.

Concrete placement was inspected by Company inspectors. The concrete was either maintained in a moist condition for seven days by approved methods or coated with an approved curing compound.

Concrete was sampled at the frequency required by ACI 301-66. Sampling concrete and making, curing, and testing specimens was in accordance with:

ASTM C 172	Sampling Fresh Concrete.
ASTM C 31	Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field.
ASTM C 39	Compressive Strength of Molded Concrete Cylinders.
ASTM C 143	Slump of Portland Cement Concrete.
ASTM C 231	Air Content of Freshly Mixed Concrete by the Pressure Method.

All taking and testing of concrete samples was done by qualified Company personnel. Compressive strength tests were evaluated in accordance with ACI 214. Company specifications required that 95 percent of all cylinders tested meet or exceed the specified strength for 5,000 psi concrete, and 90 percent meet or exceed the specified strength for 3,000 psi concrete. The correlation between field specimens and design strengths was evaluated continuously during construction.

The average strengths and coefficients of variations as of December 4, 1972 were:

Mix	Design psi	Cement* Sacks/Yds.	Average Strength psi	Coefficient of Variation	Number of Tests
Unit 1					
7AP	5000	7.5	6500	4.3%	11
8	5000	7.5	6400	6.5%	134
8A	5000	7.0	6220	8.3%	18
8AP	5000	6.6	6120	6.4%	43
9BP	3000	6.0	3800	7.0%	87

Mix	Design psi	Cement* Sacks/Yds.	Average Strength psi	Coefficient of Variation	Number of Tests
Unit 2					
8A	5000	7.0	6680	6.7%	40
8AP	5000	6.6	6200	7.1%	101

These coefficients of variation represent "excellent control" as defined in Table 2 of ACI 214-65.

Concrete in Unit 1 and 2 containments is Class AP for base slab and interior concrete and Class BP for cylinder and dome. Mixes designated 7AP, 8, 8A, and 8AP are Class AP. Mixes 9BP and 10BP are Class BP.

Reinforcing Steel

Reinforcing steel is deformed billet-steel bar conforming to ASTM designation A 615. All reinforcing bars in the Containment Structure are Grade 60, except for the following which are Grade 40:

1. Liner anchorages in the base slab.
2. Anchorages on the structural steel beams embedded at the base of the Containment Structure wall.

Table 3.8-1 compares the program for testing of reinforcing bars at the Diablo Canyon plant to the requirements of Regulatory Guide 1.15, which was issued after construction at Diablo Canyon was partially complete. Table 3.8-1 also indicates those areas where the Company specification is more stringent than ASTM A 615.

Heat number identification was maintained on reinforcing steel from the start of manufacture through placement in the structure.

Physical and chemical test results were sent to the construction site with the first load of steel from each heat. Test values were checked by Company inspectors or quality control engineers.

*Cement and pozzolan

Detailing was in accordance with ACI Standard 315-65, Manual of Standard Practice for Detailing Reinforced Concrete Structures. Bars to be bent were cold bent around pins of the following minimum diameters:

1. Stirrups and ties - four times the bar diameter.
2. No. 8 bars or smaller - six times the bar diameter.
3. Nos. 9, 10, and 11 - eight times the bar diameter.
4. Nos. 14 and 18 - ten times the bar diameter.

Fabrication tolerances were as follows:

1. Cut length:

No. 14 and No. 18 bars	+0 inch, -3/8 inch
All other bars	±1 inch

2. Depth of truss bars:

No. 18 bars	±2 inches
All other bars	+0 inch, -1/2 inch

3. Stirrups, ties, and spirals:

   ±1/2 inch

4. All other bends:

No. 14 and No. 18 bars	±1/2 inch
All other bars	±1 inch

Placement tolerances were as follows:

1. Concrete cover to formed surfaces:

No. 14 and No. 18 bars	-1/2 inch, +2 inches
All other bars	±1/2 inch

2. Longitudinal location of bends:

No. 14 and No. 18 bars	±2 inches
All other bars	±1 inch

3. Depth of bars in slabs:

8 inches or less in thickness	±1/4 inch
Over 8 inches in thickness	±1/2 inch

4. Lateral location in the plane of reinforcing:

   ±2 inches

Occasionally, reinforcing steel bars had to be moved to avoid interferences. In this situation, a bar could be moved, within the plane of the reinforcing layer or curtain, up to one half the specified spacing. If this was not sufficient, the resulting arrangement was submitted to the Company for approval. Also, if the bar had to be moved out of the reinforcing layer or curtain to avoid an interference by more than one bar diameter or the above tolerances, whichever was greater, the resulting arrangement was submitted to the Company for approval.

Tack welding to reinforcing bars was not permitted.

Reinforcing steel placement was inspected by contractor quality control inspectors and by Company inspectors.

As of December 4, 1972 the average and minimum properties of No. 18 bars in the Containment Structure were as follows:

Yield - minimum	61,750 psi
- average	62,552 psi
Tensile - minimum	93,750 psi
- average	106,065 psi
Elongation - minimum	7.0%
- average	9.4%

Splices

Cadweld Splices

Cadweld splices were used at all locations for primary reinforcing in the exterior shell and base slab. Cadweld splices were used in a few locations in the internal structure.

Quality control procedures for Cadweld splices are described in Table 3.8-2 which compares the program used at the Diablo Canyon plant to that required by Safety Guide 10. Safety Guide 10 was issued after construction at Diablo Canyon was partially complete.

As of December 4, 1972 the average and minimum strengths of Cadweld tensile samples are:

Minimum tensile strength	85,000 psi
Average tensile strength	97,725 psi*
Number of tests	641
Number of Cadwelds placed	19,068

Butt Welded Splices

Butt welded splices were used in a few locations where there was insufficient room to properly mount the Cadweld crucible. As of January 1, 1974, six butt welded splices had been made on No. 18 bars in the exterior shell of the Unit 1
3 Containment Structure. The quality Control measures applied are the same as those described in Subsection 3.8.1 for butt welded splices in the Auxiliary Building.

Lap Splices

Lap splices are in accordance with ACI 318-63.

Liner, Penetration Sleeves, and Penetration Internals

The Containment Structure liner is carbon steel conforming to ASTM A 516, "Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service," Grade 70. This steel has a minimum yield strength of 38,000 psi, a minimum tensile strength of 70,000 psi, and a minimum elongation of 17% in an 8-gauge length at failure. Charpy V-notch impact tests were performed at +20°F in accordance with ASTM A 370.

Penetration sleeves conform to one of the following three material specifications:

1. ASTM A 106, "Seamless Carbon Steel Pipe for High Temperature Service," Grade B, with the additional requirement that Charpy V-notch impact tests be performed at 0°F.

2. ASTM A 333, "Seamless and Welded Steel Pipe for Low Temperature Service," Grade 1, except that Charpy V-notch impact tests were performed at $0^{o}F$.

3. ASTM A 516, "Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service," Grade 70, to ASTM A 300, except that Charpy V-notch impact tests were performed at $0^{o}F$.

For all three material specifications, the Charpy impact tests were in accordance with the requirements of paragraph N-330 of ASME Section III, 1968 edition.

Penetration internals conform to the following material specifications:

1. Equipment and personnel hatches are ASME SA 516, Grade 70 to SA 300 with Charpy impact values at $0^{o}F$ in accordance with paragraph N-330 of ASME, Section III, 1968 edition.

2. Carbon steel flued heads are ASME SA 105, Grade II with Charpy impact tests at $0^{o}F$ in accordance with paragraph NB-2300 of Section III, ASME B&PV Code, 1971 edition. Ultrasonic and magnetic particle inspections are performed in accordance with paragraphs NB 2542 and NB 2545, respectively.

3. Stainless steel flued heads are ASME SA 182, Grade F 304. Ultrasonic and liquid penetrant inspections are performed in accordance with paragraphs NB 2542 and NB 2546, respectively.

Welded studs attached to the liner meet the requirements of ASTM A 108, Grade 1015-1018.

Mill Test Reports certifying the physical and chemical properties of the liner plate delivered to the job site were required from the steel supplier. The average and minimum properties of liner plate are as follows:

Reactor Pit and Floor Plates	Unit 1	Unit 2
Yield Strength - minimum	43,800 psi	39,800 psi
- average	51,400 psi	55,100 psi
Tensile Strength - minimum	71,000 psi	74,000 psi
- average	76,500 psi	78,900 psi
Elongation - minimum	19%	17%
- average	25%	24%
Total number of heats	16	11

	Unit 1	Unit 2
Total number of slabs	58	62
Total number of tests	58	62
Cylinder and Dome		
Yield Strength - minimum	41,900 psi	38,100 psi
- average	48,800 psi	46,100 psi
Tensile Strength - minimum	70,200 psi	70,100 psi
- average	74,900 psi	73,681 psi
Elongation - minimum	19%	18%
- average	26.5%	25.4%
Total number of heats	23	22
Total number of slabs	251	255
Total number of tests	251	255

2 Fabrication of the Containment Structure liner conforms to the applicable parts of Part UW, "Requirements for Unfired Pressure Vessels Fabricated by Welding," Section VIII, ASME Boiler and Pressure Vessel Code.

All of the welds were visually examined by contractor quality control inspectors. All field welds were also visually examined by Company inspectors.

Table 3.8-3 compares the program for nondestructive testing of Containment Structure liner welds, including penetration sleeves and inserts, used on the Diablo Canyon plant to that required by Safety Guide 19, which was issued after construction at Diablo Canyon was partially complete.

Erection tolerances for the liner were as follows:

The liner of the completed structure shall be substantially round. At points not more than 4 inches above the base, the radius of the 3/4-inch liner shall be 69 feet 11-13/16 inches plus or minus 1/2-inch. The maximum diameter of the 3/8-inch liner shall not exceed 140 feet 4 inches and the minimum diameter shall not be less than 139 feet 8 inches.

The liner shall be erected true and plumb. At any point the out-of-plumb shall not exceed 1/240 of the height of the point above the base. For any plate (10 feet ± in height) the out-of-plumbness shall not exceed 1/120.

Flat spots or local out-of-roundness shall not exceed 2 inches in 15 feet of arc.

The base liner shall not deviate from a plane surface between anchorages by more than 1/240.

Stud welding was in accordance with Supplement to AWS D1.0-66. The tolerance on the location of each stud was ±1/2-inch. Each welder, at the beginning of each work day, attached at least two test studs which were then tested by bending the stud approximately 45 degrees toward the plate to demonstrate the integrity of the stud to plate weld. If failure occurred in the weld, the welding procedure or technique was corrected and two successive studs successfully welded and tested before further studs were attached to the liner plate. These test studs were allowed to remain in place but are not considered as a part of the regular stud pattern required by the design. A 100 percent visual inspection of liner stud anchors was made prior to pouring concrete.

Structural Steel

Hexagonal collars at equipment hatch and personnel hatch meet the requirements of ASTM A 516, Grade 70, and ASTM A 300, except that Charpy "V" notch impact tests were performed at +20°F.

Structural steel in the internal structure and the wide flange beams embedded in the lower part of the exterior shell meet the requirements of ASTM A 36, ASTM A 441, or ASTM A 572 Grade 42, with Charpy "V" notch impact tests in accordance with ASTM A 370 at +20°F.

The following quality control procedures were followed in the fabrication of the hexagonal steel collars at the equipment hatch and personnel hatch openings:

1. The 4-inch thick plate for the edge pieces was ultrasonically examined in accordance with ASTM A 435, except that scanning covered 100% of the surface.

2. Fabrication conformed to the applicable parts of Part UW "Requirements for Unfired Pressure Vessels Fabricated by Welding" of Section VIII of the ASME Boiler and Pressure Vessel Code. All welds are full penetration butt welds and were 100% radiographed in accordance with Paragraph UW-51.

3. The reinforcement plates were heat treated after fabrication in accordance with Paragraph UCS-56, "Requirements for Postweld Heat Treatment" of Section VIII of the ASME Boiler and Pressure Vessel Code.

Testing and In-Service Surveillance Requirements

Testing

8 After each Containment Structure is completed, with liner, concrete, and all electrical and piping penetrations, equipment hatch and personnel locks in place, the following tests are performed:

8

Structural Integrity Test

The structural integrity test is performed by pressuring the Containment Structure with air up to 115 percent of design pressure, or 54 psig. During this test, structural deflections are measured, crack patterns in the concrete are measured and photographed, and strains in the liner and reinforcing steel are measured electrically and recorded. The deflections, crack patterns, and strains are compared to the theoretical predictions to verify the structural integrity of the Containment Structure. The structural integrity test of each Containment Structure meets the requirements of Regulatory Guide 1.18, Structural Acceptance Test for Concrete Primary Reactor Containments. The Unit 1 Containment Structure is a prototype concrete primary reactor containment as
8 defined in Regulatory Guide 1.18.

For the structural integrity test, the pressure is increased in increments to the maximum of 54 psig. Measurements are made at 0, 15, 25, 35, 47, and 54 psig during pressurization and again during depressurization. At each pressure level, the deflection and strain gauge readings are made after a one hour wait to allow adjustment of strains. The crack patterns are recorded both before and immediately after the test and at the maximum pressure level achieved during the test.

The planned instrumentation for each unit is as follows:

Unit 1

The radial and longitudinal growth are measured by means of calibrated targets attached to the exterior shell and sighted by means of high magnification theodolites. Radial deflections are measured at three points on each of six equally spaced meridians; at the springline, at mid-height of the cylinder, and at the top of the base slab. Vertical deflections are measured at the springline and at the top of the dome.

The radial and tangential deflections of the Containment Structure wall are measured at twelve locations adjacent to the equipment hatch, which is the largest opening.

The pattern of cracks that exceed 0.01 inch in width is mapped or photographed near the base-wall intersection, at mid-height of the wall, at the springline of the dome, and around the equipment hatch, which is the largest opening. At each location, an area of at least 40 square feet is mapped or photographed.

Strain measurements are made at the following locations in accordance with the requirements for prototype Containment Structures:

1. In the wall at the top of the base mat.

2. In the wall at the equipment hatch, which is the largest opening, with at least one gauge located approximately 0.5 times the wall thickness from the edge of the opening.

3. In the wall at the level of the springline.

4. In the wall where pure membrane stress is anticipated, i.e., where there are no discontinuities.

Inasmuch as the concrete is assumed cracked, and the strength of the concrete is neglected, strain measurements are made on the reinforcing steel and liner, rather than in the concrete. At the equipment hatch, additional strain measurements are made on the structural steel hex collar. In the wall at the top of the base slab, additional strain measurements are made on the structural steel wide flange beams.

The method used for attaching strain gauges to No. 18 reinforcing bars is shown in Figure 3.8-29.

In evaluating the results of the structural integrity test, the deflection measurements will be considered the most reliable result.

Unit 2

The deflection measurement and crack mapping program for Unit 2 is identical to that for Unit 1.

Preoperational Leakage Rate Tests

Overall Integrated Leakage Rate Tests

63 During the depressurization phase of the structural integrity test the sequence is stopped at 47 psig to conduct an overall integrated leakage rate test at design pressure.

During the overall integrated leakage rate tests, the double penetration and weld channel zones are open to the atmosphere inside the Containment Structure.

Sensitive Leakage Rate Tests

Following the successful completion of the overall integrated leakage rate tests and the structural integrity test, a sensitive leakage rate test is performed with only the volume of the weld channels and double penetrations included in the test. The sensitive leakage rate test is performed with penetrations and weld channels at 47 psig and with the Containment Structure at atmospheric pressure. 8

General

All leakage rate tests are conducted and evaluated in accordance with Appendix J of 10 CFR 50.

In-Service Surveillance Requirements

Periodic leakage rate testing will be performed in accordance with the requirements of Appendix J of 10 CFR 50.

3.8.3 REFERENCES

1. Portland Cement Association, "Analysis of Small Reinforced Concrete Buildings for Earthquake Forces," Chicago, Illinois, 1955.

2. Blume, Newmark, and Corning, "Design of Multistory Reinforced Concrete Buildings for Earthquake Motions," Portland Cement Association, Chicago Illinois, 1961.

3. Timoshenko and Woinowsky - Krieger, "Theory of Plates and Shells," McGraw-Hill, Inc., New York, 1959, Second Edition.

TABLE 3.8-1

(Sheet 1 of 2)

TESTING OF REINFORCING BARS FOR DESIGN CLASS I CONCRETE STRUCTURES
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH REGULATORY GUIDE 1.15

DIABLO CANYON PLANT	REGULATORY GUIDE 1.15
The number of test specimens required for acceptance is in accordance with ASTM A 615, Deformed Billet-Steel Bars for Concrete Reinforcement, American Society for Testing and Materials. Additional samples were tested as part of the splice testing program. The requirements for acceptance testing are more stringent than ASTM A 615 in that all tests must be conducted using the full section of the bar.	At least one full-diameter specimen from each bar size should be tested for each 50 tons or fraction thereof of reinforcing bars that are produced from each heat and used in Category I structures.
Test procedures are in accordance with ASTM A 615-68.	The test procedures should be in accordance with ASTM A 370-68, Standard Methods and Definitions for Mechanical Testing of Steel Products, American Society for Testing and Materials.
Acceptance standards are in accordance with ASTM A 615-68 using full sections of the bars as rolled. Bend test requirements described in Item 3 below are more stringent than those in Supplemental Requirements (S-1) of ASTM A 615-72.	The acceptance standards should be in accordance with ASTM A 615-72, Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement, American Society for Testing and Materials, including Supplemental Requirement (S-1),* using full sections of the bars as rolled.

*Supplemental Requirement (S-1) is for a 90° bend test, using a pin diameter 10 times the bar diameter, on No. 14 and No. 18 bars.

TABLE 3.8-1

(Sheet 2 of 2)

TESTING OF REINFORCING BARS FOR DESIGN CLASS I CONCRETE STRUCTURES
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH REGULATORY GUIDE 1.15

DIABLO CANYON PLANT	REGULATORY GUIDE 1.15
In addition to the requirements of ASTM A 615, the Company specification requires the following: 1. Grade 60 bars be limited in carbon and manganese content to a maximum of 0.45 percent, respectively. 2. Performance of a check analysis, which is listed as an option in ASTM A 615. 3. No. 14 and No. 18 bars be subjected to a 90° bend test using a pin having a diameter eight times the diameter of the bar.	Where any material property such as yield strength to tensile strength ratio, ductility, weldability or other similar property is relied upon by the designer or constructor, then the reinforcing bar chemistry should be controlled to the extent required to achieve the desired material property, and confirmatory testing should be performed.
Deformations were inspected during production to assure conformance with ASTM A 615.	Deformations of the reinforcing bars should be inspected to assure their compliance with ASTM A 615-72 and with the licensee's specifications pertinent to bonding and other purposes which are dependent on the deformation characteristics.
Adequacy of deformations for splicing was demonstrated by the tensile tests of the Cadweld splices. See Table 3.8-2.	Adequacy of deformations for splicing will be demonstrated by the tensile tests of the mechanical splice. See Safety Guide 10, "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures.

TABLE 3.8-2

(Sheet 1 of 5)

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CONCRETE CONTAINMENTS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 10

DIABLO CANYON PLANT	SAFETY GUIDE 10
Prior to production splicing, each operator was instructed by a representative of the manufacturer. Each operator (a crew consisted of an operator and a helper) prepared one qualification splice for each of the splice positions for which he was qualified. The qualification splice was made using the same materials as those used in the structures. The completed qualification splices had to pass visual inspection and develop the minimum tensile strength of the reinforcing steel. A manufacturer's representative was present for at least the first 20 production splices for each crew to verify that proper procedures were being used and quality splices obtained.	1. <u>Crew Qualification</u> - Each member of the splicing crew (or each crew if the members work as a crew) should prepare two qualification splices for each of the splice positions (e.g., horizontal, vertical, diagonal) to be used. The qualification splices should be made using the same materials (e.g., bar, sleeve, powder) as those to be used in the structure. The completed qualification splices should meet the requirements specified by the designer of the containment structure and approved by the licensee, pass visual inspection as provided by Paragraph 2 below, and meet the tensile tests as provided by Paragraph 3 below.
All completed splices were visually inspected in accordance with the recommendations of the Erico Co. inspection manual RB-5M 768, Inspection of the Cadweld Rebar Splice. This visual inspection included both ends of the sleeve, the tap hole, and measurement of void area. In addition, at least twice daily for each Cadweld crew, an inspector observed the entire splicing operation including cleaning of rebar ends, spacing of rebar, centering rebar in sleeve, loading the crucible, and firing the charge. The Cadweld procedure specified for the Diablo Canyon Plant includes placing a mark 12 inches ±1/4 inch back from the	2. <u>Visual Inspection</u> - All completed mechanical splices should be inspected at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve in accordance with the requirements specified by the designer of the containment structure and approved by the licensee. Among the items which should be included in these specifications are longitudinal centering of sleeve on the spliced ends, allowable voids in filler metal, extent of leaking of filler metal, permissible gap between rebar ends, cartridge size, gas blowout, amount of packing and

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CONCRETE CONTAINMENTS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 10

DIABLO CANYON PLANT	SAFETY GUIDE 10
end of the bar. This line was used as a reference to determine if the bar ends are centered in the sleeve.	slag at the tap hole. Splices that fail to pass visual inspection should be discarded and replaced, and should not be used as tensile test samples.
Acceptance criteria for splice tensile tests is as follows: No splice in the test series may have a tensile value below 125 percent of the specified yield point stress, and no more than 5 percent of the splices tested may have an ultimate tensile strength less than 85 percent of that specified. The average tensile strength of all splices in the test series must equal or exceed the ASTM specified minimum ultimate strength.	3. Tensile Testing - Splice samples may be production splices (i.e., those cut directly from in-place reinforcing) or sister splices (i.e., those removable splices made in-place next to production splices and under the same conditions). Splice samples should be subjected to tensile tests in accordance with the sampling frequency specified in Paragraph 4a or Paragraph 4b below, to determine conformance with the following acceptance standards: a. The tensile strength of each sample tested should be equal or exceed 125 percent of the minimum yield strength specified in the ASTM standard appropriate for the grade of reinforcing bar using loading rates set forth in ASTM Specification A 370 dated August 15, 1968. b. The average tensile strength of each group of 15 consecutive samples should equal or exceed the guaranteed ultimate tensile strength specified for the reinforcing bar.

TABLE 3.8-2

(Sheet 3 of 5)

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CONCRETE CONTAINMENTS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 10

DIABLO CANYON PLANT	SAFETY GUIDE 10
	If any sample tested fails to meet the provisions of Paragraph 3a above, the procedure of Paragraph 5a below should be followed. If the average tensile strength of the 15 samples tested fails to meet the provisions of Paragraph 3b above, the procedure of Paragraph 5b below should be followed.
Testing frequency for each crew, position, and grade of bar was as follows: One out of the first 10 splices. This splice must be a production splice for No. 18, Grade 60 bars and a sister splice for other sizes and grade of bar. Three out of the next 90 splices for No. 18, Grade 60 bars and one out of the next 90 splices for all other sizes and grades of bar. Three out of second and subsequent 100 splice units for No. 18, Grade 60 bars and one out of second and subsequent 100 splice units for all other sizes and grades of bar. At least 25 percent of the total number of No. 18, Grade 60 test splices must be made by cutting out production splices on a random basis. The remaining test splices may be made by having test bars	4. Tensile Test Frequency - Separate test cycles should be established for mechanical splices in horizontal, vertical, and diagonal bars, for each bar size, and for each splicing crew as follows: a. Test Frequency for Production Splice Test Samples. If only production splices are tested, the sample frequency should be: 1 of the first 10 splices 1 of the next 90 splices 2 of the next and subsequent units of 100 splices b. Test Frequency for Combinations of Production and Sister Splices. If production and sister splices are tested, the sample frequency should be:

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CONCRETE CONTAINMENTS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 10

DIABLO CANYON PLANT	SAFETY GUIDE 10
tie wired alongside the production bars and spliced in sequence with those bars. The minimum length of the spliced bars is 3 feet.	1 production splice of the first 10 production splices 1 production and 3 sister splices, for the next 90 production splices 3 splices, either production or sister splices, for the next and subsequent units of 100 splices. At least 1/4 of the total number of splices tested should be production splices.
In the event a splice should fail the tensile test criteria, the specimen was to be examined by a testing laboratory. Based on the results of this investigation, additional splices by the crew responsible as directed by the Engineer were to be taken from the structure to insure that there are no other defective splices. The procedures of the crew responsible for making the failed splice were to be reviewed, and if necessary, the crew retrained and requalified.	5. Procedure for Substandard Tensile Test Results a. If any production or sister splice tested fails to meet the tensile test specification of Paragraph 3a and the observed rate of splices that fail the tensile test at that time does not exceed 1 for each 15 consecutive test samples, the sampling procedure should be started anew. If any production or sister splice used for testing fails to meet the tensile test specification in Paragraph 3a, and the observed rate of splices that fail the tensile test exceeds 1 for each 15 consecutive test samples, mechanical splicing should be stopped. In addition, the adjacent production splices on each side of the last failed splice and 4 other splices distributed uniformly throughout the balance of

TABLE 3.8-2

(Sheet 5 of 5)

MECHANICAL (CADWELD) SPLICES IN REINFORCING BARS OF CONCRETE CONTAINMENTS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 10

DIABLO CANYON PLANT	SAFETY GUIDE 10
	the 100 production splices under investigation should be tested, and an independent laboratory analysis should be made to identify the cause of all failures. The results of these tests should be evaluated by the designer of the containment structure and the licensee to determine the required corrective action. The designer and the licensee should specify the extent of repairs necessary and the actions required to prevent further failures from the identified causes.
	b. If two or more splices from any of these 6 additional splice samples fail to meet the tensile test specification of Paragraph 3a, the balance of the 100 production splices under investigation should be rejected and replaced.
	When mechanical splicing is resumed, the sampling procedure should be started anew.
	If the average tensile strength of the 15 consecutive samples fails to meet the provisions of Paragraph 3b above, the designer of the containment structure and the licensee should evaluate and assess the acceptability of the reduced average tensile strength with respect to the required strength at the location from which the samples were taken.

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
	1. Nondestructive Examination of Liner Seam Welds
For each welder and welding position, the first 10 feet of weld was examined radiographically. Thereafter, a minimum of 10 percent of the welding (to at least include all intersections of joints) was progressively examined radiographically as welding was performed. This was done on a random basis with the location specified in such a manner that an approximately equal number of radiographs were taken from the work of each welder. The techniques of radiographic examination of welds were in accordance with Paragraph UW-51 of Section VIII, ASME Boiler and Pressure Vessel Code (ASME B&PV Code). See Notes 1 and 2.	a. For each welder and welding position (flat, horizontal, and overhead), the first 10 feet of weld, and one spot (not less than 12 inches in length) in each additional 50 foot increment of weld (weld test unit) or fraction thereof should be examined radiographically in accordance with the techniques prescribed in Section V, "Nondestructive Examination," of the ASME Boiler and Pressure Vessel Code (ASME B&PV Code). In any case, a minimum of 2 percent of all liner seam welds should be examined by radiography.
Where radiographic examination of liner seam welds was not feasible, a minimum of 10 percent of the welding (to at least include all locations where there are welded backing strip splices and intersections) was examined by magnetic particle or liquid penetrant testing. Magnetic particle testing was in accordance with Appendix VI of Section VIII, ASME B&PV Code. Liquid penetrant testing was in accordance with Appendix VIII of Section VIII, ASME B&PV Code. See Notes 1 and 2.	b. Where radiographic examination of liner seam welds is not feasible or where the weld is located in areas which will not be accessible after construction, the entire length of weld should be examined by the magnetic particle method or by the ultrasonic method in accordance with the techniques prescribed in Section V of the ASME BP&V Code for such examination methods.

TABLE 3.8-3

(Sheet 2 of 10)

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
All liner seam welds were tested for leaktightness in accordance with the following method:	c. All liner seam welds should be tested for leaktightness in accordance with the following method (or other methods of equivalent sensitivity):
Immediately preceding the test, a soap solution is applied to the weld. The application of the soap solution must not precede the vacuum box by more than 3 minutes. The vacuum box, which contains a viewing window, is placed over the area to be tested and evacuated to a 5 psi differential with the atmospheric pressure.	Immediately preceding the test, a soap solution (or other appropriate solution) should be applied to the weld. A vacuum box containing a viewing window should be placed over the area to be tested and evacuated to produce at least 5 psi differential with the atmospheric pressure. Leaks in welds, if present, should be detected by formation of bubbles. The solution used for the test should have bubble formation properties adequate for identification of leaks. The test solution should be checked every hour, with a suitable test leak to verify the bubble formation property of the solution used.
Leak-chase-channels are installed over the liner welds. Upon completion of one zone of leak-chase-channels, the zone was tested at the containment structure design pressure of 47 psi. The acceptance criteria is that there be no loss of pressure within 2 hours as indicated by a pressure gauge.	d. Where leak-chase-system channels are installed over liner welds, channel-to-liner-plate welds should be tested for leaktightness by pressurizing the channels to containment design pressure. If any indicated loss of channel test pressure occurs within 2 hours, as evidenced by a test gauge, the channel-to-liner welds should be soap bubble tested in accordance with the above procedure.

TABLE 3.8-3

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
	2. Nondestructive Examination of Penetration, Airlock, and Access Opening Welds
All welds in penetration, airlocks, and access openings that are not backed by concrete were fully examined in accordance with Class B requirements of Section III, ASME B&PV Code. See Notes 1, 2, 3, and 4. All welds between flued heads and pipelines were fully examined in accordance with the Class II requirements of ANSI B 31.7, Nuclear Power Piping.	a. All welds in penetrations, airlocks, and access openings that are not backed by concrete, such as welds between penetrations and flued fittings and pipelines, should be fully examined in accordance with examination methods of NE-5120 of Section III of the ASME B&PV Code employing the techniques prescribed in Section V of that code.
Welds backed by concrete in the vicinity of penetrations were examined as follows: 1. Welds between the penetration sleeve and insert plate were fully examined in accordance with the Class B requirements of Section III, ASME B&PV Code. See Notes 1, 2 and 4. 2. Welds between the insert plate and the liner were examined under the same criteria as liner seam welds. All welds backed by concrete in the containment structure are carbon steel.	b. All welds in the vicinity of penetrations and access openings that are backed by concrete, such as welds between penetration and reinforcing plate,* penetration and liner, reinforcing plate and liner, liner insert and liner, reinforcing plate and frames for airlocks and access openings, and liners and frames for airlocks and access openings, should be fully examined (1) in accordance with Paragraph 2a above or (2) by magnetic particle, or liquid penetrant when a nonmagnetic weld is used, in accordance with the techniques prescribed in Section V of the ASME B&PV Code.

*Thickened liner insert which provides local reinforcement.

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
Examination of welds in penetration assemblies and in the vicinity of penetrations is described in the preceding paragraphs.	c. All welds in bellow type expansion joints provided in penetration assemblies or appurtenances to the containment vessel should be magnetic- particle or liquid-penetrant tested when a nonmagnetic weld is used, in accordance with the techniques prescribed in Section V of ASME B&PV Code for such examination methods.
	3. Qualification of Welders and Welding Procedures
The qualification of welders, welding machine operators, and welding procedures was in accordance with Section IX, "Welding Qualifications," of the ASME B&PV Code. See Note 2.	The qualification of welders, welding machine operators, and welding procedures should be in accordance with Section IX, "Welding Qualifications," of the ASME B&PV Code.
	4. Qualification of Nondestructive Examination Personnel
Nondestructive examinations were performed by personnel qualified in accordance with the appropriate parts of the ASME B&PV Code. See Notes 1 and 2.	Nondestructive examination should be performed by personnel designated by the licensee or his agent and qualified in accordance with the provisions of Section V of the ASME B&PV Code.
	5. Selection of Spots for Radiographic Examination
The spots of liner seam welds to be radiographically examined were selected on a random basis with the locations selected such that all intersections of joints were examined, and an approximately equal number of radiographs were taken from the work of	The spots of liner seam welds to be radiographically examined should be randomly selected, but no two spots in adjacent weld test units should be closer than 10 feet and their locations should be recorded.

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
each welder. The location covered by each radiograph was recorded.	
	6. Time of Examination
Nondestructive examinations were done progressively as welding was performed.	All examinations should be performed as soon as practicable after the lineal increment of weld to be examined is completed.
	7. Acceptance Standards
	a. Containment Liner Seam Welds Examined by Radiography
Where a spot in the seam weld is judged acceptable in accordance with Paragraph UW-51 of Section VIII, ASME B&PV Code, the entire weld test unit represented by this spot radiograph is considered acceptable. See Notes 2 and 3.	Where a spot in the seam weld is judged acceptable in accordance with the referenced standards of NE-5120 of Section III of the ASME B&PV Code, the entire weld test unit represented by this spot radiograph is considered acceptable.
	b. Containment Liner Seam Welds Examined by Ultrasonic or Magnetic Particle
Where a spot in the seam weld examined by magnetic particle or liquid penetrant method is judged acceptable in accordance with the acceptance criteria referenced in Section VIII, ASME B&PV Code, the entire weld seam represented by the examination is considered acceptable. See Notes 2 and 3.	Seam welds examined by ultrasonic or magnetic particle methods are considered acceptable provided the examinations meet the acceptance standards referenced for such examination methods in NE-5120 of Section III of the ASME B&PV Code.

TABLE 3.8-3

(Sheet 6 of 10)

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
	c. Soap Bubble Leak Tests of Containment Liner Welds
The acceptance criterion for the vacuum box test is that no leaks be detected.	Liner welds are considered acceptable provided no leakage is detected by soap bubble tests (or by other methods of equivalent sensitivity).
	d. Penetration, Airlock, and Access Opening Welds
Penetration, airlock, and access opening welds that are not backed by concrete are considered acceptable provided the examinations meet the acceptance standards referenced for Class B vessels in Section III, ASME B&PV Code. See Notes 2, 3 and 4. Welds between flued heads and pipelines are considered acceptable provided the examinations meet the acceptance standards referenced for Class II piping in ANSI B 31.7, Nuclear Power Piping. Welds between the penetration sleeve and insert plate are considered acceptable provided the examinations meet the acceptance standards referenced for Class B vessels in Section III, ASME B&PV Code. See Note 2.	Penetration, airlock, and access opening welds are considered acceptable provided the examinations meet the acceptance standards referenced in NE-5120 of Section III of the ASME B&PV Code. Welds in bellows type expansion joints are considered acceptable if the examinations meet the acceptance standards referenced in magnetic particle and liquid penetrant methods in NE-5120 of Section III.

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
	8. Repair and Reexamination a. Containment Liner Seam Welds Examined by Radiography
If a radiographed spot failed to meet the specified acceptance standards, two additional spots of the same length were radiographically examined in the same weld seam at locations away from the original spot but in welds performed by the same welder or welder operator. The locations of these additional spots were determined as provided for the original spot examination.	When a radiographed spot fails to meet the specified acceptance standards, two additional spots should be radiographically examined in the same weld test unit at locations at least one foot removed (on each side) from the original spot. The locations of these additional spots should be determined by the examiner using the same procedure followed in the selection of the original spot for examination and the examination results should determine the following corrective actions:
If the two additional spots examined showed welding which meets the specified acceptance standards, the entire weld represented by the three radiographs is judged acceptable. The defective welding disclosed by the first of the three radiographs was removed and repaired.	(1) If the two additional spots examined meet the specified acceptance standards, the entire weld unit represented by the three spot radiographs is considered acceptable. However, the defective welding disclosed by the first of the three radiographs should be repaired by welding.
If either of the two additional spots examined showed welding which does not comply with the specified acceptance standards, the entire portion of the seam represented was considered unacceptable or	(2) If either of the two additional spots examined fails to meet the specified acceptance standards, the entire weld test unit is considered unacceptable.

TABLE 3.8-3

(Sheet 8 of 10)

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
optionally the entire weld represented was completely radiographed and defective welding corrected to meet the specified acceptance standards.	The entire weld should be removed and the joint should be rewelded or, optionally, the entire weld unit may be completely radiographed and defective welding only need be repaired.
Repair welding was performed using a qualified procedure. The rewelded joints or weld repaired areas were completely reradiographed and meet the specified acceptance standards.	(3) Repair welding should be performed using a procedure as specified under regulatory position 3. above. The weld repaired areas in each weld test unit should be spot radiographed at one selected location to meet the acceptance criteria specified in regulatory position 7.a. or 8.a. (1). above.
	b. Containment Liner Seam Welds Examined by Ultrasonic or Magnetic Particle
If a weld which had been examined did not comply with the specified acceptance standards, additional examination was performed to the same extent as required for radiography. The weld was repaired and reexamined in accordance with the provisions of Section VIII of the ASME B&PV Code. See Notes 2 and 3.	When a weld which has been examined does not comply with the specified acceptance standards, the weld should be repaired and reexamined in accordance with the provisions of Section III of the ASME B&PV Code.

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

DIABLO CANYON PLANT	SAFETY GUIDE 19
	c. Soap Bubble Tests of Containment Liner Welds
If a weld was judged unacceptable because leakage is detected by the soap bubble test, the weld was repaired. Repair welding was performed using a procedure qualified as specified for production welds. The weld repaired areas were reexamined by soap bubble leakage retesting.	Welds judged unacceptable because leakage is detected by the soap bubble test (see regulatory position 7.c. above) should be repaired. Repair welding should be performed using a qualified procedure as specified under regulatory position 3. above. The weld repaired areas should be reexamined by soap bubble leakage retesting.
	d. Penetration, Airlock and Access Opening Welds
If a weld was judged unacceptable on a penetration sleeve airlock, or access opening, the weld was repaired and reexamined in accordance with the provisions for Class B vessels of Section III of the ASME B&PV Code. See Notes 2 and 3.	Welds judged acceptable in accordance with regulatory position 7.d. should be repaired and reexamined in accordance with the provisions of Section III of the ASME B&PV Code.
	9. Records
Retention of records is discussed in Chapter 17.	Records of radiographs and other nondestructive examinations including those for repaired defective welds should be retained by the licensee in compliance with the provisions of Section XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

TABLE 3.8-3

(Sheet 10 of 10)

NONDESTRUCTIVE EXAMINATION OF PRIMARY CONTAINMENT LINER WELDS
COMPARISON OF PROGRAM USED ON DIABLO CANYON PLANT WITH SAFETY GUIDE 19

Notes:

1. Section V, ASME B&PV Code, which provides techniques for nondestructive examination applicable to all sections of the ASME B&PV Code, was first published in July 1971. Although it may eventually replace the corresponding parts of other sections of the ASME B&PV Code, the individual sections still contain techniques for nondestructive examinations.

2. References in the table to ASME B&PV Code for the Diablo Canyon plant refer to 1968 Edition including addenda through Summer 1968.

3. NE-5120, Section III, ASME B&PV Code requires examination technique and acceptance criteria in accordance with Section VIII, ASME B&PV Code (Paragraph UW-51 for radiography).

4. Class B requirements of Section III, ASME B&PV Code specify radiographic examination and acceptance criteria in accordance with Paragraph UW-51, Section VIII, ASME B&PV Code.

STRESS IN AUXILIARY BUILDING STRUCTURAL MEMBERS
TABLE 3.8-4

Description of Member	Location of Member	Load Combination	Total Calculated Stress	Allowable Stress Ksi	Remarks
3'-0" x 3'-6 concrete beam	From 2'-0" wall - Col. #7 - Col. #8, El. 137	D + L + DE	742 k-ft.	890 k-ft.	Bending moment
3'-0" x 3'-0" concrete beam	Between col. lines T and U, 13' south of col. line 15, El. 137	D + L + DE	998 k-ft.	1078 k-ft.	Bending moment
1'-6" roof slab	Between col. lines L and R, 15^{7} and 17^{4}-, El. 166	D + L + DDE	59.7 k-ft.	84.6 k-ft.	Bending moment
2'-0" concrete T-beam	At col. line T, between col. line 17^{4}- and 18, El. 115	D + L + DE	183 k-ft.	283 k-ft.	Bending moment
2'-6" concrete wall	Liquid hold-up tank enclosure wall along col. line 15^{7}- between El. 55 and 85	D + L + DDE	812 k	6950 k	Shear

TABLE 3.8-4

(Sheet 2 of 5)

Description of Member	Location of Member	Load Combination	Total Calculated Stress	Allowable Stress Ksi	Remarks	
2'-0" concrete wall	East wall of liquid hold-up tank enclosure wall along col. line V from El. 55	D + L + DDE	3674 k	29000 k	Shear	
2'-0" deep concrete	Floor of control room along col. line K between col. lines $16\overset{6}{-}$ and $17\overset{3}{-}$	D + L + DE	372 k-ft.	395 k-ft.	Bending moment	8
2'-0" concrete slab	Floor of control room between col. lines J and K	D + L + DE	41 k-ft.	42.5 k-ft.	Bending moment	
W36 x 170 composite beam	Roof of control room between col. lines $15\overset{7}{-}$ and $20\overset{3}{-}$ and H and L	D + L + DE	2.07 ksi 21.48 ksi	2.25 ksi 22 ksi	Compression in concrete Tension in steel	

TABLE 3.8-4

Description of Member	Location of Member	Load Combination	Total Calculated Stress	Allowable Stress Ksi	Remarks
1'-8" x 3'-0" curved concrete girder	Floor between col. lines S and T and $9\frac{7}{-}$ and $6\frac{4}{-}$, El. 140	D + L + DE	173 k-ft. 73 k-ft.	240 k-ft. 163 k-ft.	Bending moment Torsional moment
1'-6" concrete wall	45' north of col. line 18 37' west of col. line R above El. 140	D + L + DDE	98 k	1100 k	Shear force
1'-0" concrete wall	29' north of col. line 18 20' west of col. line R above El. 140	D + L + DDE	739 k	3200 k	Shear force
Concrete wall	Column line V	D + L + DDE	3674 k	29000 k	Shear force
2'-0" concrete wall	Column line H	D + L + DDE	3578 k	21500 k	Shear force
3'-0" concrete wall	Column line $15\frac{7}{-}$	D + L + DDE	444 k	2250 k	Shear force

TABLE 3.8-4

Description of Member	Location of Member	Load Combination	Total Calculated Stress	Allowable Stress Ksi	Remarks
Concrete column-wall	Column line K	D + L + DDE	1819 k	16500 k	Shear force
2'-0" concrete wall	Column line J	D + L + DDE	1494 k	17500 k	Shear force
Build-up crane column	Column line 16^{7} V^{1}	D + L + DDE	0.800	1.0	*
W 14 x 74	Top and bottom chords	D + L + DDE	0.048	1.0	*
Angles 3 x 3 x 3/8	Truss bracing	D + L ± DDE	0.210	1.0	*
Angles 6 x 6 x 1/2	Diagonal bracing	D + L + DDE	0.240	1.0	*
Build-up section	Runway girder	D + L + DDE	0.320	1.0	*
2 - C 15 x 50	Horizontal strut	D + L ± DDE	0.180	1.0	*

TABLE 3.8-4

Description of Member	Location of Member	Load Combination	Total Calculated Stress	Allowable Stress Ksi	Remarks
Angles 8 x 8 x 7/8	Top and bottom chords	D + L + DDE	0.070	1.0	*
Angles 6 x 4 x 3/8	Truss bracing	D + L + DDE	0.980	1.0	*
Angles 6 x 6 x 1/2	Cord bracing	D + L + DDE	0.380	1.0	*
Angles 6 x 6 x 3/8	Sway frame - bottom chord	D + L + DDE	0.300	1.0	*
Angles 5 x 5 x 3/8	Sway frame - bracing	D + L + DDE	0.680	1.0	*
W 8 x 35	Sway frame - purlin	D + L + DDE	0.230	1.0	*

D = Dead load DE = Load from design earthquake
L = Live load DDE = Load from double design earthquake
* = Allowable is AISC interaction formula percentage

STRESS IN INTERNAL STRUCTURE OF CONTAINMENT BUILDING

Table 3.8-5

(Sheet 1 of 2)

Description of Member	Location of Member	Load Combination	Calculated Stress	Allowable Stress	Remarks
Rebar @ 3 ft. Concrete Wall	Crane Wall:				
	1. Vertical Bar	D + L + DDE + CP + R + J + M	22 Ksi	51 Ksi	
	2. Hoop Bar		38 Ksi	54 Ksi	
4 ft. Concrete Wall	Fuel Transfer Canal:				
	1. Wall @ N & S from El. 113'-1 1/2" to El. 140'-0"	D + L + DDE + CP + R + J + M	403 K.ft.	403 K.ft.	Allowable is Bending Moment
	2. Wall @ W from El. 113'-1 1/2" to El. 140'-0"	D + L + DE + T	113 K.ft.	161 K.ft.	"
2 ft. Concrete Wall	Fuel Transfer Canal Wall @ W from El. 88'-0" to El. 113'-1 1/2"	D + L + DE + T	50 K.ft.	121 K.ft.	"
3 ft. Concrete Slab	Fuel Transfer Canal Floor @ El. 113'-1 1/2	D + L + DE + T	89 K.ft.	112 K.ft.	"
4 ft. Concrete Slab	Fuel Transfer Canal Floor @ El. 104'-0	D + L + DE + T	133 K.ft.	161 K.ft.	"
6 ft. Concrete Wall	Reactor Cavity Wall:				
	1. Vertical Bar	D + L + DE + T	443 K.ft.	624 K.ft.	"
	2. Hoop Bar	D + L + DDE + CP + R + J + M	249 K.ft.	252 K.ft.	Allowable is Tension Force
3 ft. Concrete Slab	Floor @ El. 140'-0	D + L + DE + T	69 K.ft.	93 K.ft.	Allowable is Bending Moment

STRESS IN INTERNAL STRUCTURE OF CONTAINMENT BUILDING

Table 3.8-5

(Sheet 2 of 2)

Description of Member	Location of Member	Load Combination	Calculated Stress	Allowable Stress	Remarks
4 ft. 6 in. Concrete Slab	Floor @ El.140'0	D + L + DDE + CP + R + J + M	464 K.ft.	524 K.ft.	Allowable is Bending Moment
5 ft. Concrete Slab	Floor @ El.140'0	D + L + DDE + CP + R + J + M	474 K.ft.	528 K.ft.	"
10 in. Concrete Slab	Annulus Platform @ El. 130'0	D + L + DE + T	15 K.ft.	16 K.ft.	"
1 ft. 6 in.	Annulus Platform @ El. 140'0	D + L + DE + T	32 K.ft.	43 K.ft.	"
21*73	Annulus Platform @ El. 130'0	D + L + DE + T	22 Ksi	24 Ksi	"
21*62	"	"	19 Ksi	24 Ksi	"
12*40	"	"	17 Ksi	24 Ksi	"
12*65	Annulus Platform Column	"	254 K	268 K	Allowable is Axial Load
12*99	"	"	110 K	366 K	"
21*55	Annulus Platform @ El. 140'-0	"	20 Ksi	24 Ksi	"
21*82	"	"	11 Ksi	24 Ksi	"
21*68	"	"	20 Ksi	24 Ksi	"
21*96	"	"	19 Ksi	24 Ksi	"

* - Wide Flange

TABLE 3.8-7

TEST PROBLEM

COMPUTER PROGRAM VERIFICATION

DYBOX 2 PROGRAM

	Computer Output	Hand Solution	
Weight (KIPS)	3667	3667	
Mass Moment of Inertia (KIP-Ft-Sec^2)	110,691	110,759	
Torsional Rigidity (ft-K)	10166×10^{10}	10288×10^{10}	
Area, X - X (ft^2)	168	168	
Area, Y - Y (ft^2)	133	133	
Moment of Inertia, X - X (ft^4)	24,696	24,696	3
Moment of Inertia, Y - Y (ft^4)	12,578	12,578	
Center of Mass from X - X axis (ft)	19.7	19.7	
Center of Mass from Y - Y axis (ft)	54.2	54.2	
Center of Rigidity from X - X axis (ft)	20.0	20.0	
Center of Rigidity from Y - Y axis (ft)	44.2	43.5	
Total Rigidity, X - X Direction (K/Ft)	1,614,000	1,614,000	
Total Rigidity, Y - Y Direction (K/Ft)	1,183,781	1,247,000	

TABLE 3.8-6

BLANK

TABLE 3.8-8

TEST PROBLEM

COMPUTER PROGRAM VERIFICATION

SHEARWALL 4 PROGRAM

A. Shear Stress (KSI), for 0.20 g Lateral Force

	Computer Output	Hand Solution
Wall 1, Pier 1	.006	.006
Wall 2, Pier 1	.007	.007
Wall 3, Pier 1	.007	.007
Wall 4, Pier 1	.002	.002
Wall 5, Pier 1	.008	.008
Wall 6, Pier 1	.013	.013

B. Overturning Stress, (KSI), for 0.20 g Lateral Force 3

Wall 1, Pier 1	-.004	-.004
Wall 2, Pier 1	-.003	-.003
Wall 3, Pier 1	.001	.001
Wall 4, Pier 1	.003	.003
Wall 5, Pier 1	-.003	-.003
Wall 6, Pier 1	-.004	-.004

C. Area of Steel ($in.^2$)

Wall 1, Pier 1	.12	.12
Wall 2, Pier 1	.09	.09
Wall 3, Pier 1	.09	.09
Wall 4, Pier 1	.09	.09
Wall 5, Pier 1	.09	.09
Wall 6, Pier 1	.12	.12

TABLE 3.8-9

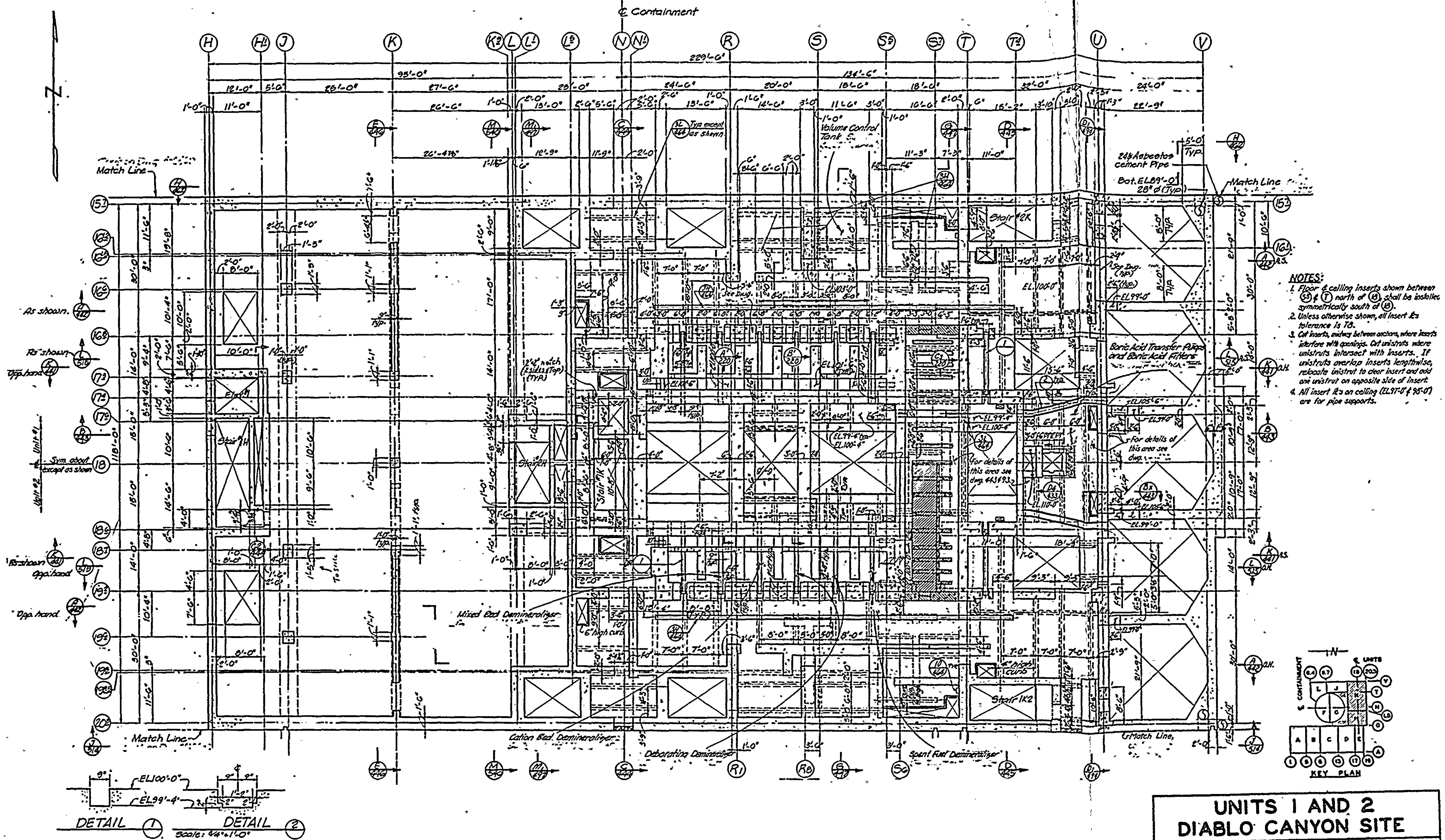
TEST PROBLEM
COMPUTER PROGRAM VERIFICATION

STRAN PROGRAM
PLANE STRESS FINITE ELEMENT PROGRAM

	Hand Solution*	STRAN Output*	Finite Element Output*
f_1	25.6	25.6	25.9
f_2	44.7	44.7	45.2
f_3	46.9	46.9	47.4
σ_x	-1.6	-1.6	-1.0
σ_y	-10.5	-10.5	-10.0
τ	9.8	9.8	9.7

* Stress, in KIPS per square inch

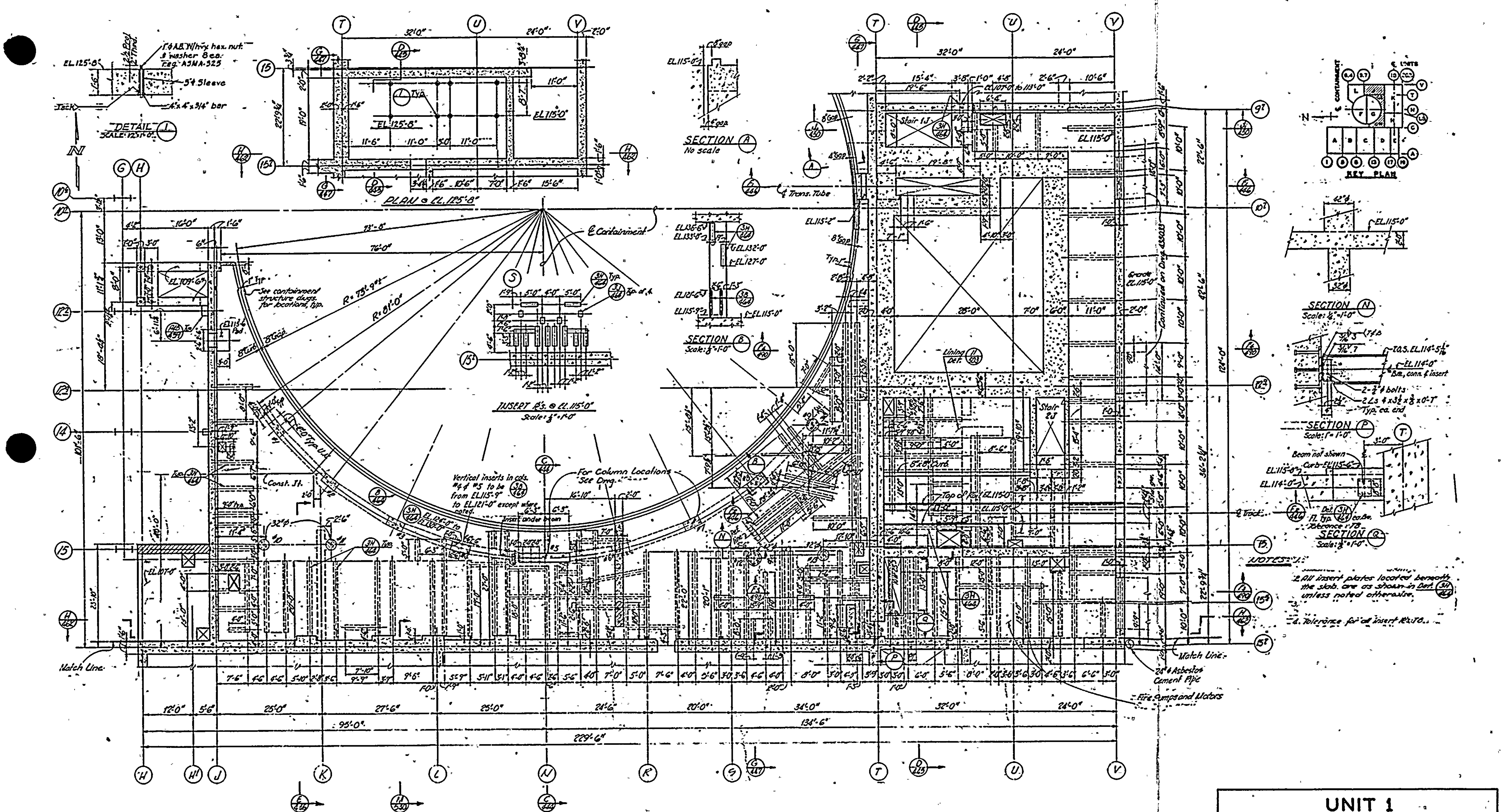
3



UNITS 1 AND 2
DIABLO CANYON SITE

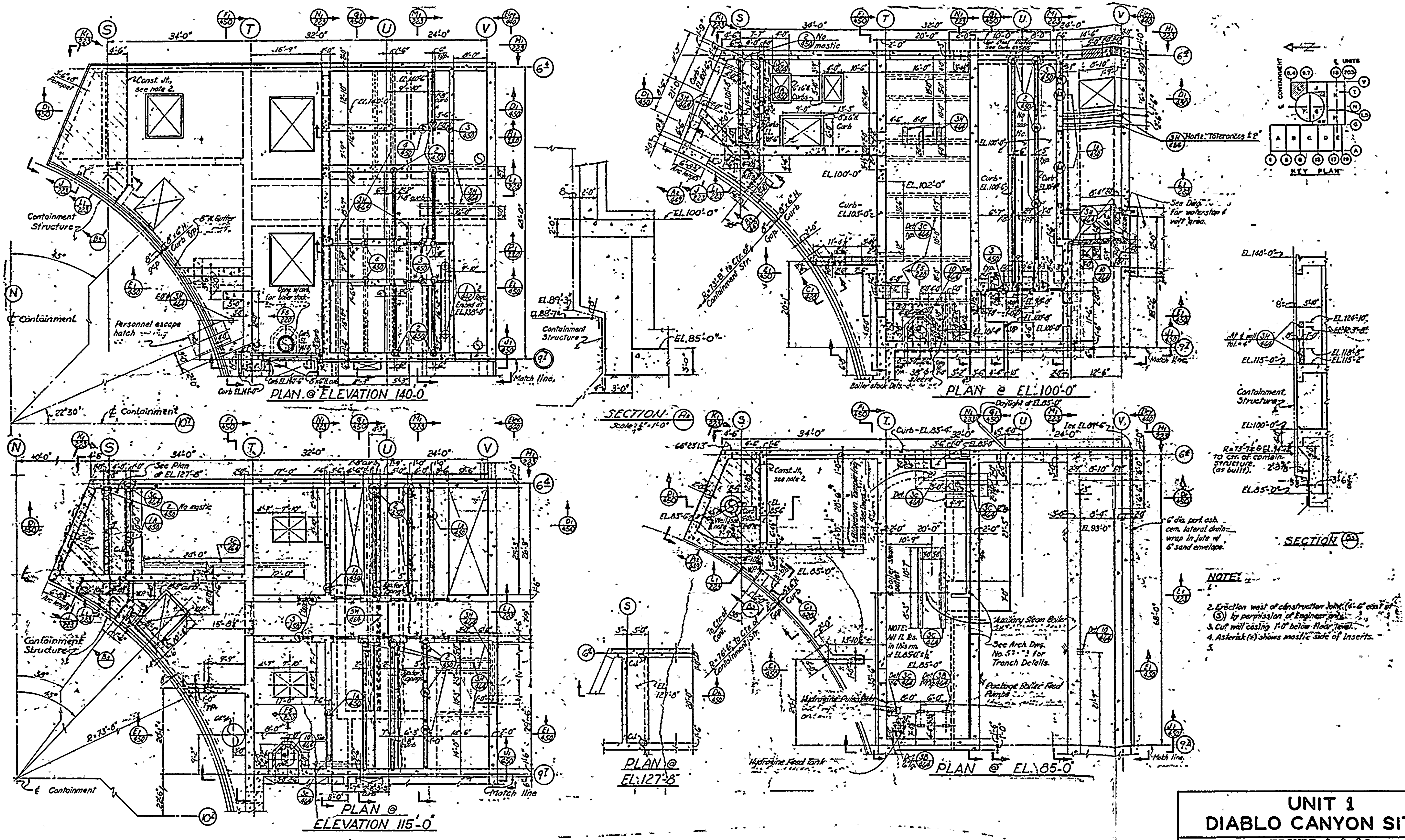
FIGURE 3.8-0A

AUXILIARY BUILDING
CONCRETE OUTLINE
PLAN AT ELEVATION 100
AREAS H AND K



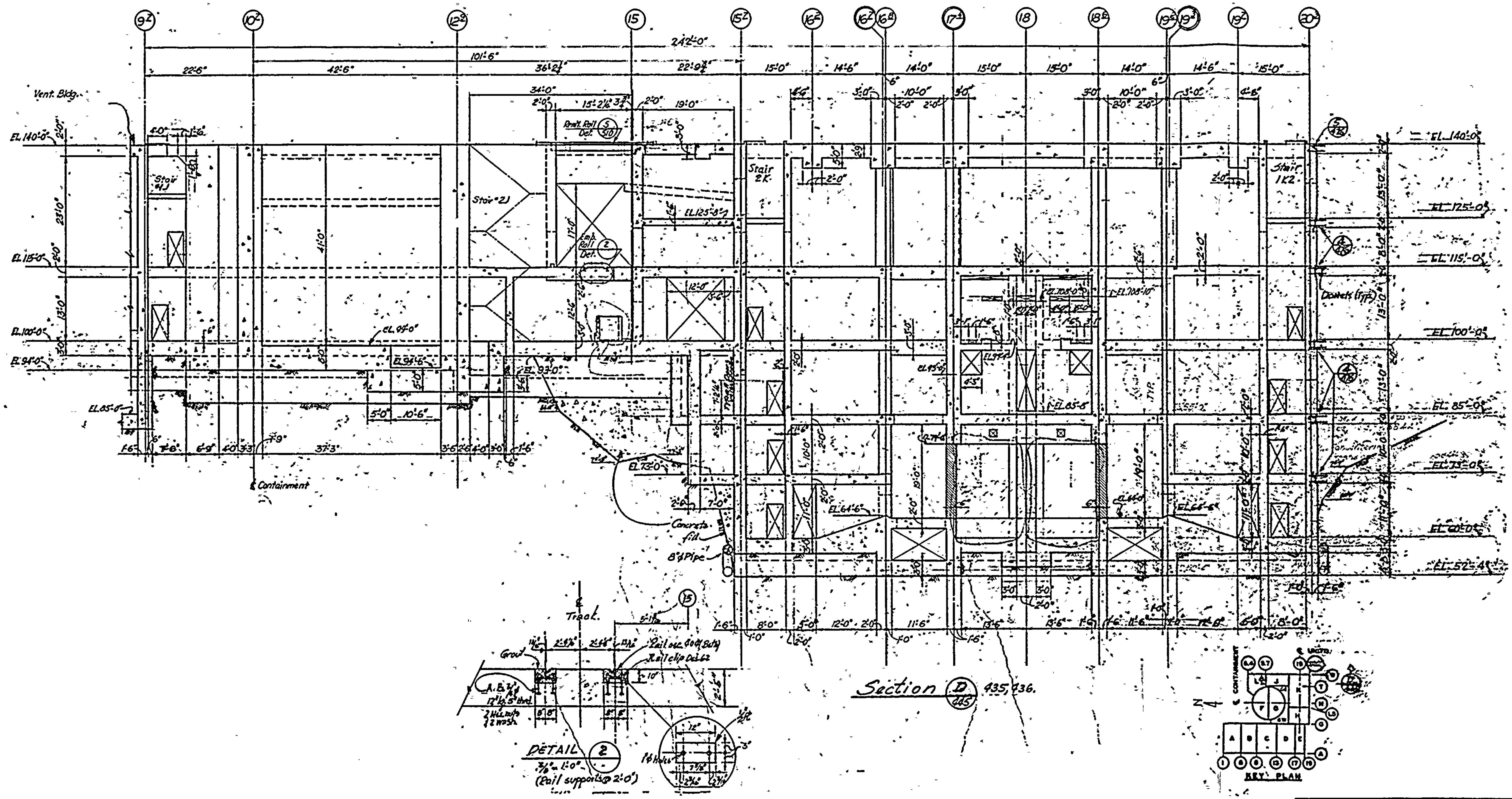
UNIT 1
DIABLO CANYON SITE

FIGURE 3.8-0B
AUXILIARY BUILDING-CONCRETE OUTLINE
PLAN AT ELEVATION 115
AREAS J, GE, GW



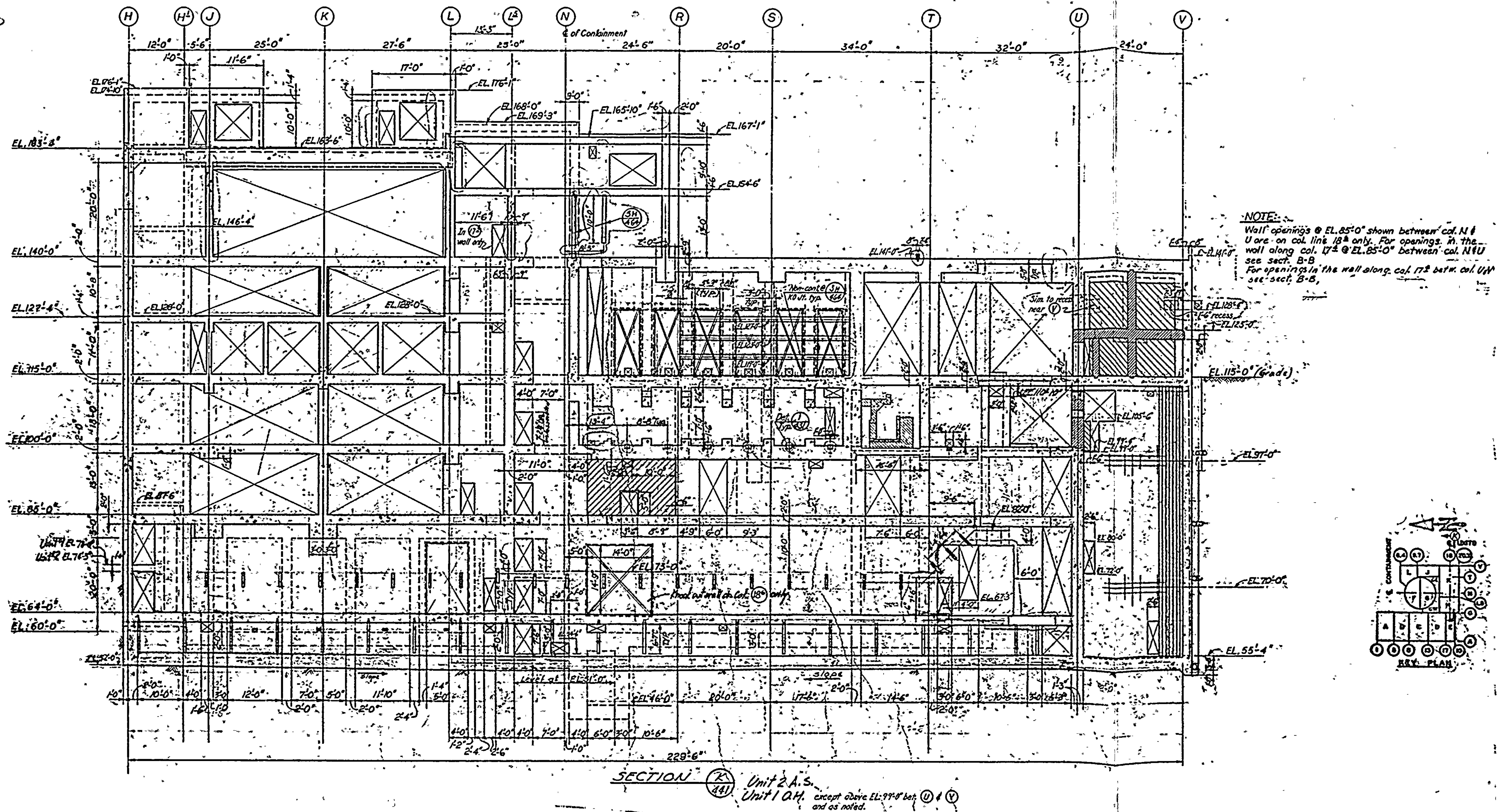
UNIT 1
DIABLO CANYON SITE

FIGURE 3.8-0C
AUXILIARY BUILDING-CONCRETE OUTLINE PLANS AT ELEVATIONS 85, 100, 115 AND 140 - AREA L



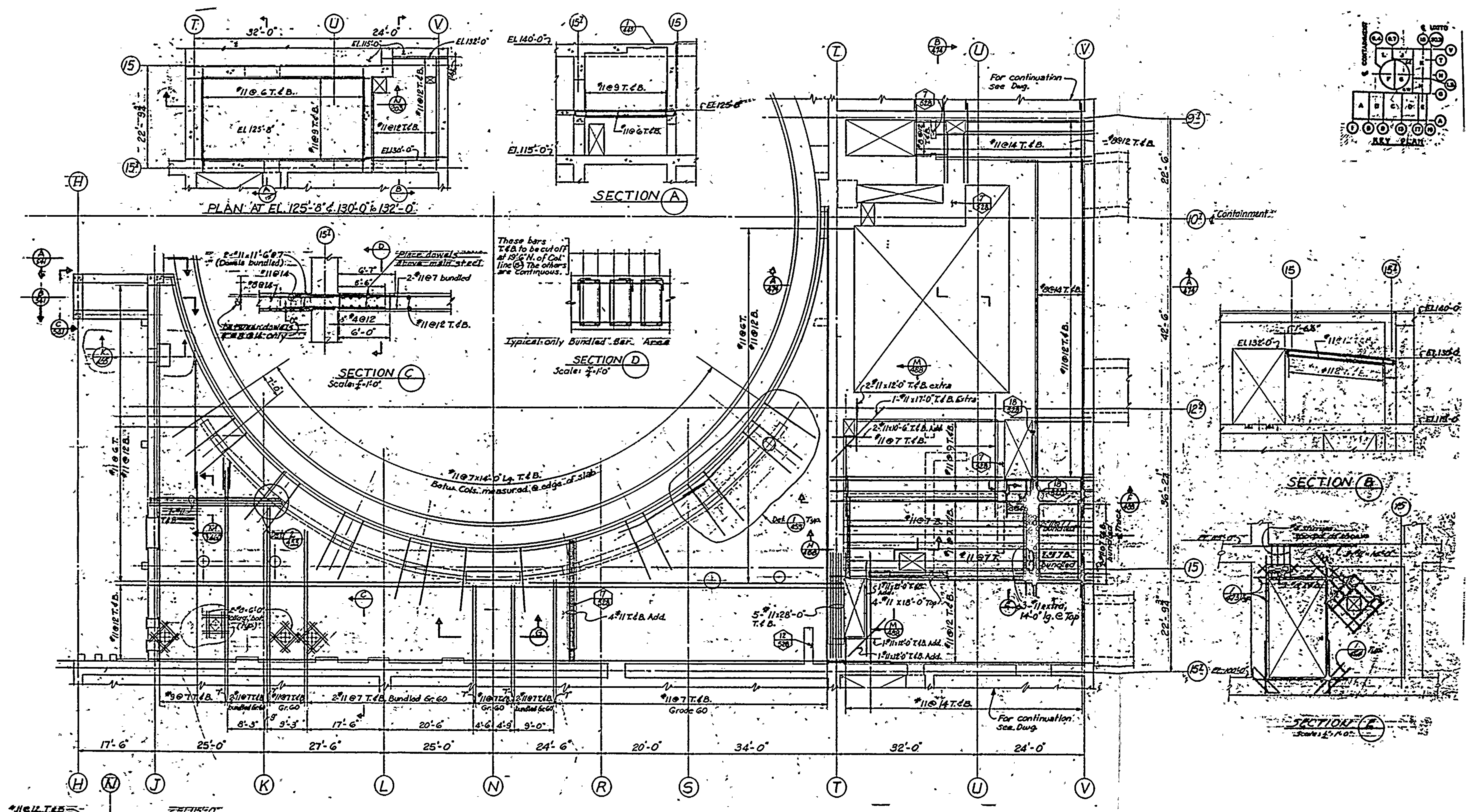
UNITS 1 AND 2
DIABLO CANYON SITE

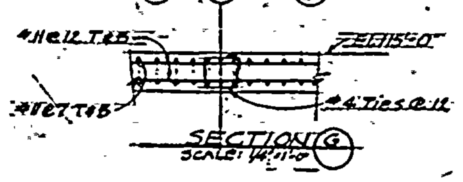
FIGURE 3.8-0D
AUXILIARY BUILDING
CONCRETE OUTLINE
SECTION AREAS J AND K



UNITS 1 AND 2
DIABLO CANYON SITE

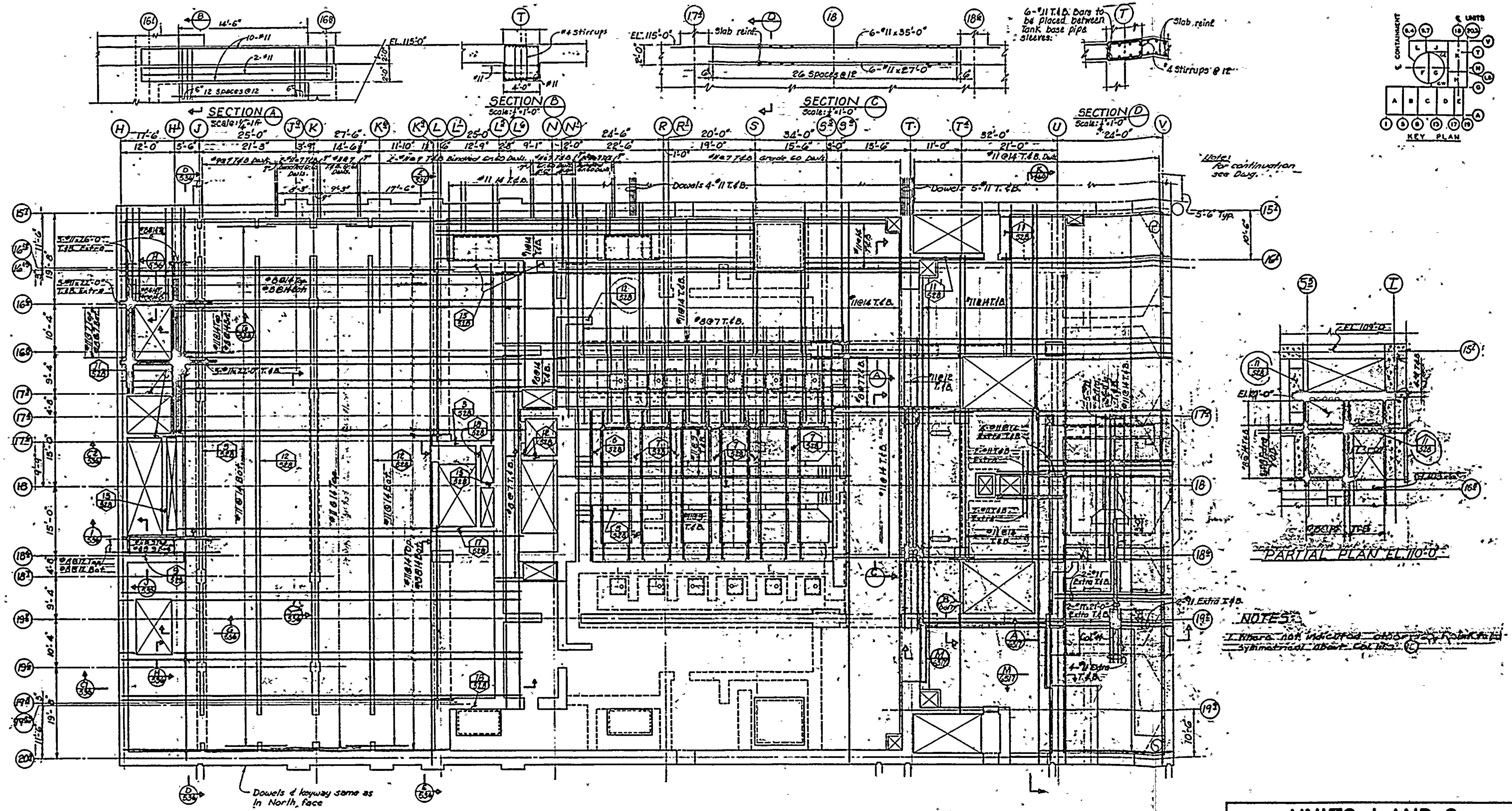
FIGURE 3.8-0E
AUXILIARY BUILDING
CONCRETE OUTLINE
SECTION AREAS H AND K





UNIT 1
DIABLO CANYON SITE

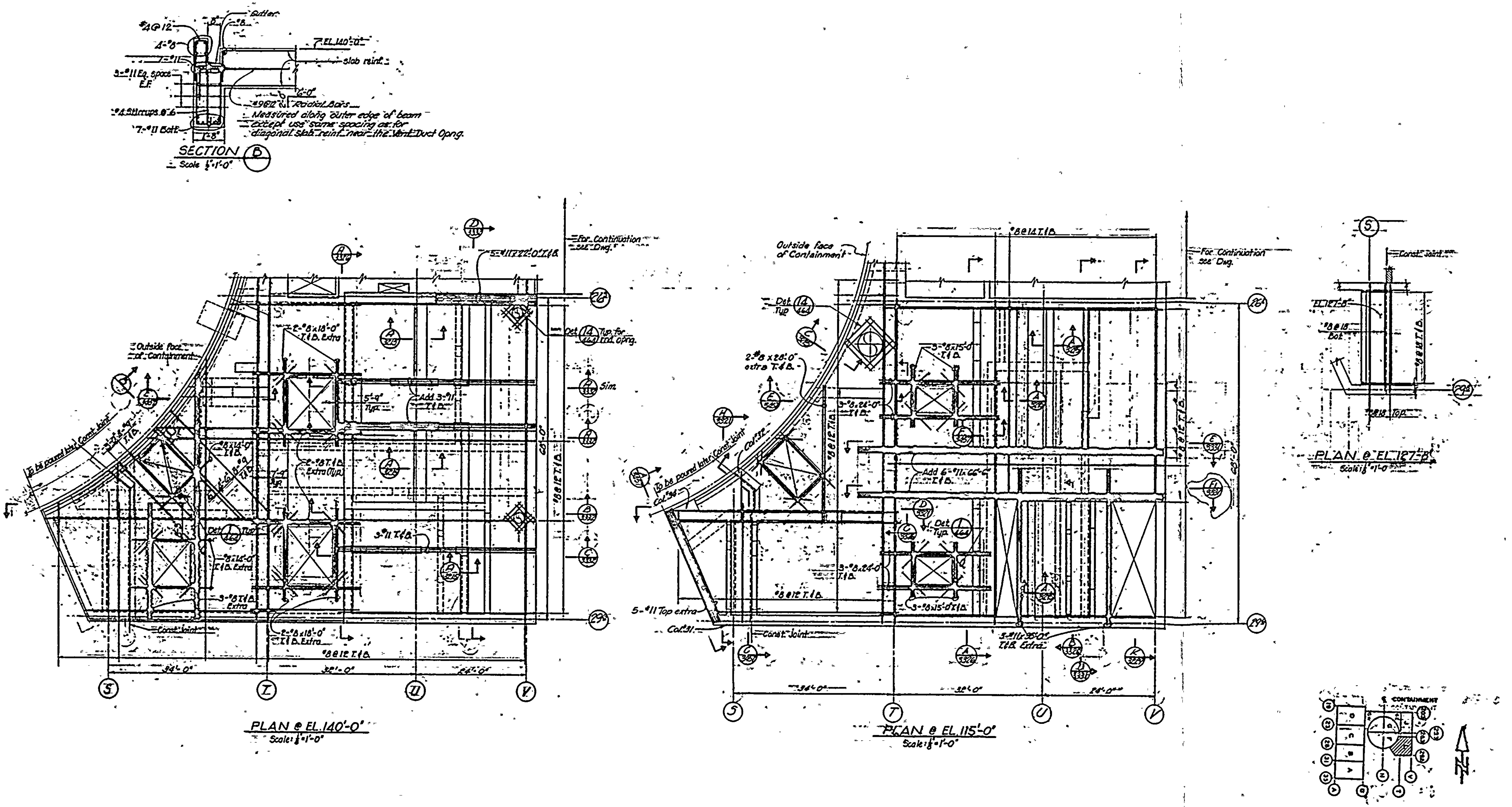
FIGURE 3.8-OF
AUXILIARY BUILDING
CONCRETE REINFORCING - PLAN AT
ELEVATION 115-AREAS J, GE, AND GW



UNITS 1 AND 2
DIABLO CANYON SITE

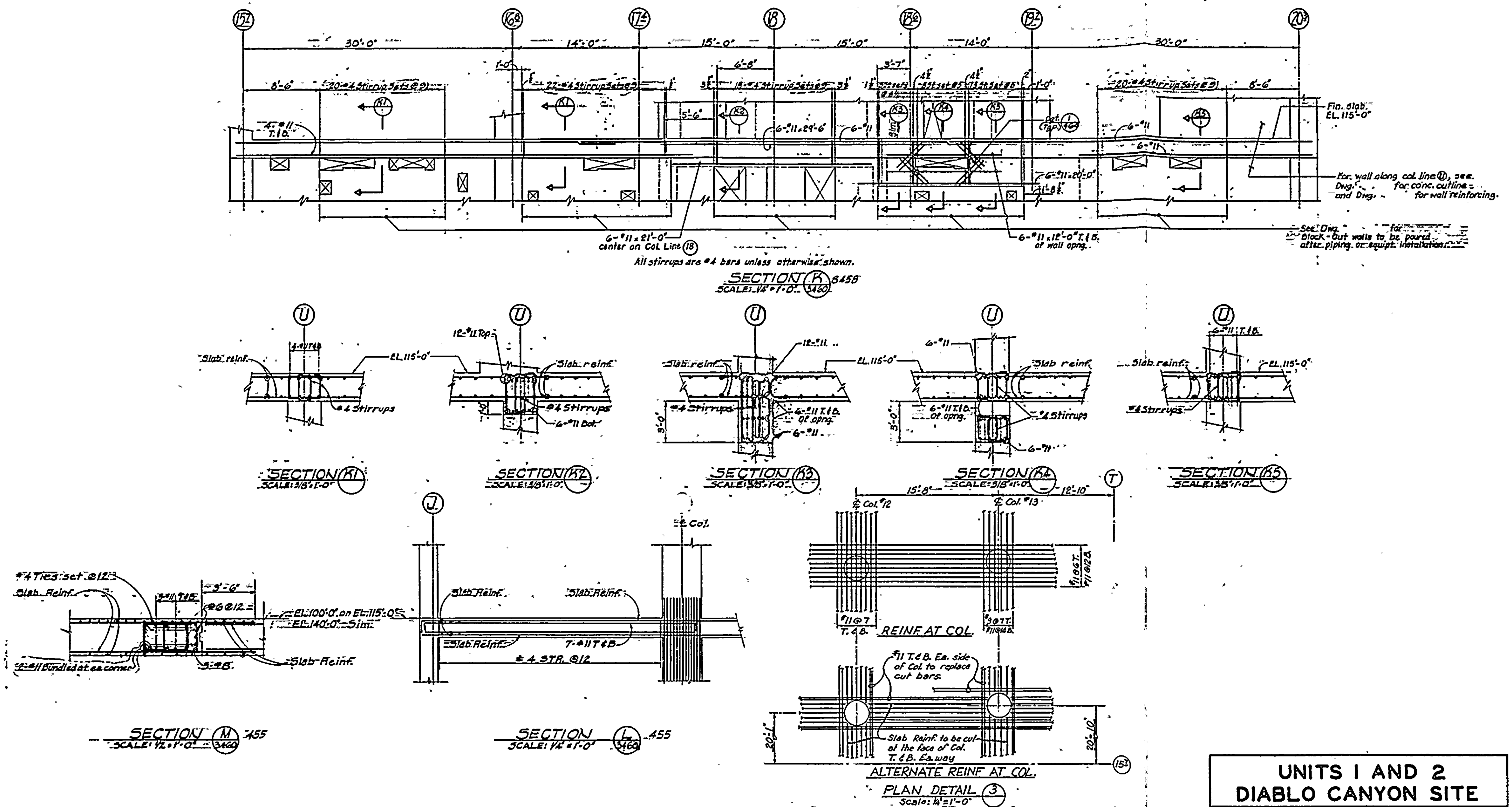
FIGURE 3.8-0G

AUXILIARY BUILDING
CONCRETE REINFORCING
PLAN AT ELEVATION 115
AREAS H AND K



UNIT 2
DIABLO CANYON SITE

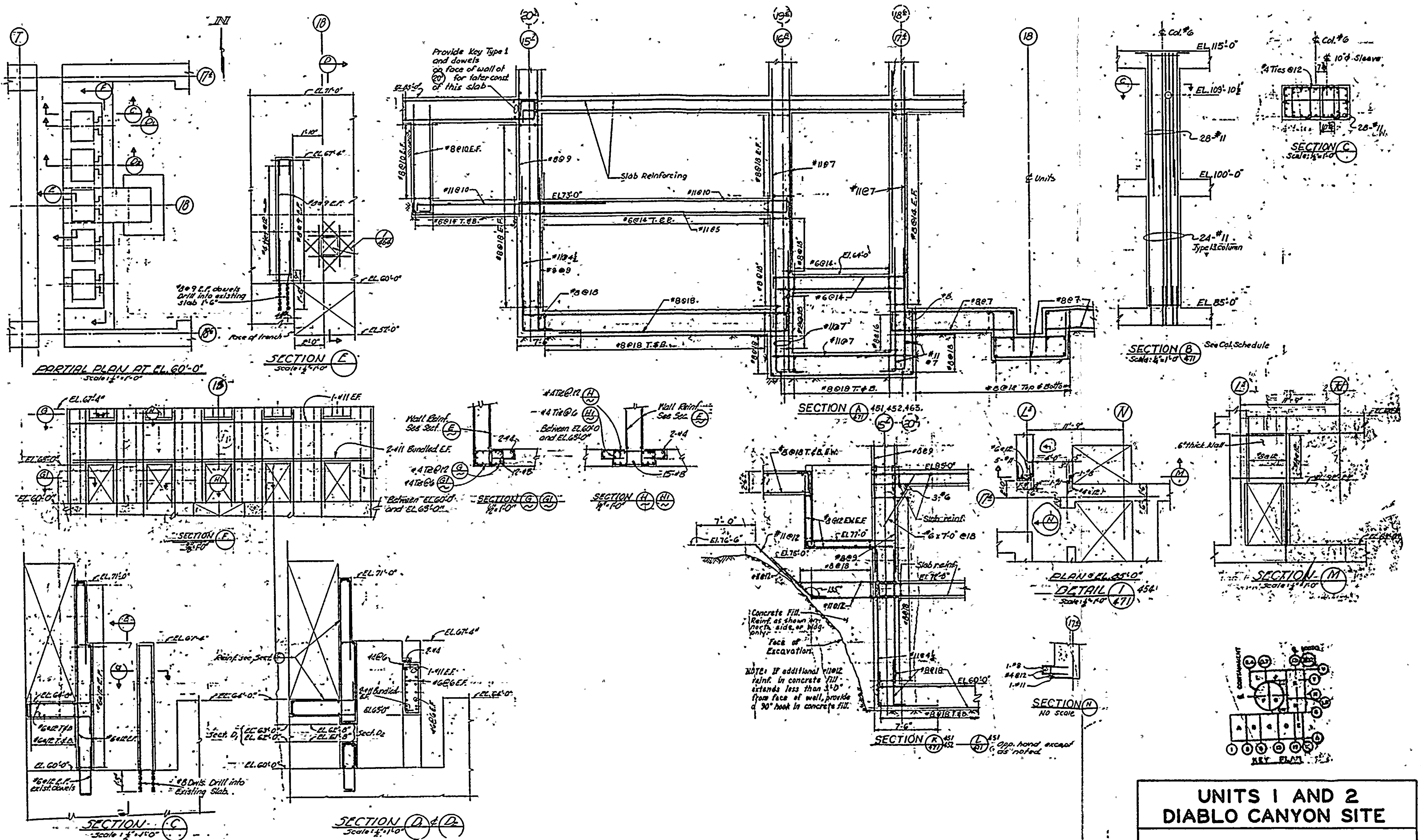
FIGURE 3.8-0H
AUXILIARY BUILDING
CONCRETE REINFORCING - PLANS AT
ELEVATION 115 AND 140 - AREA L



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-01

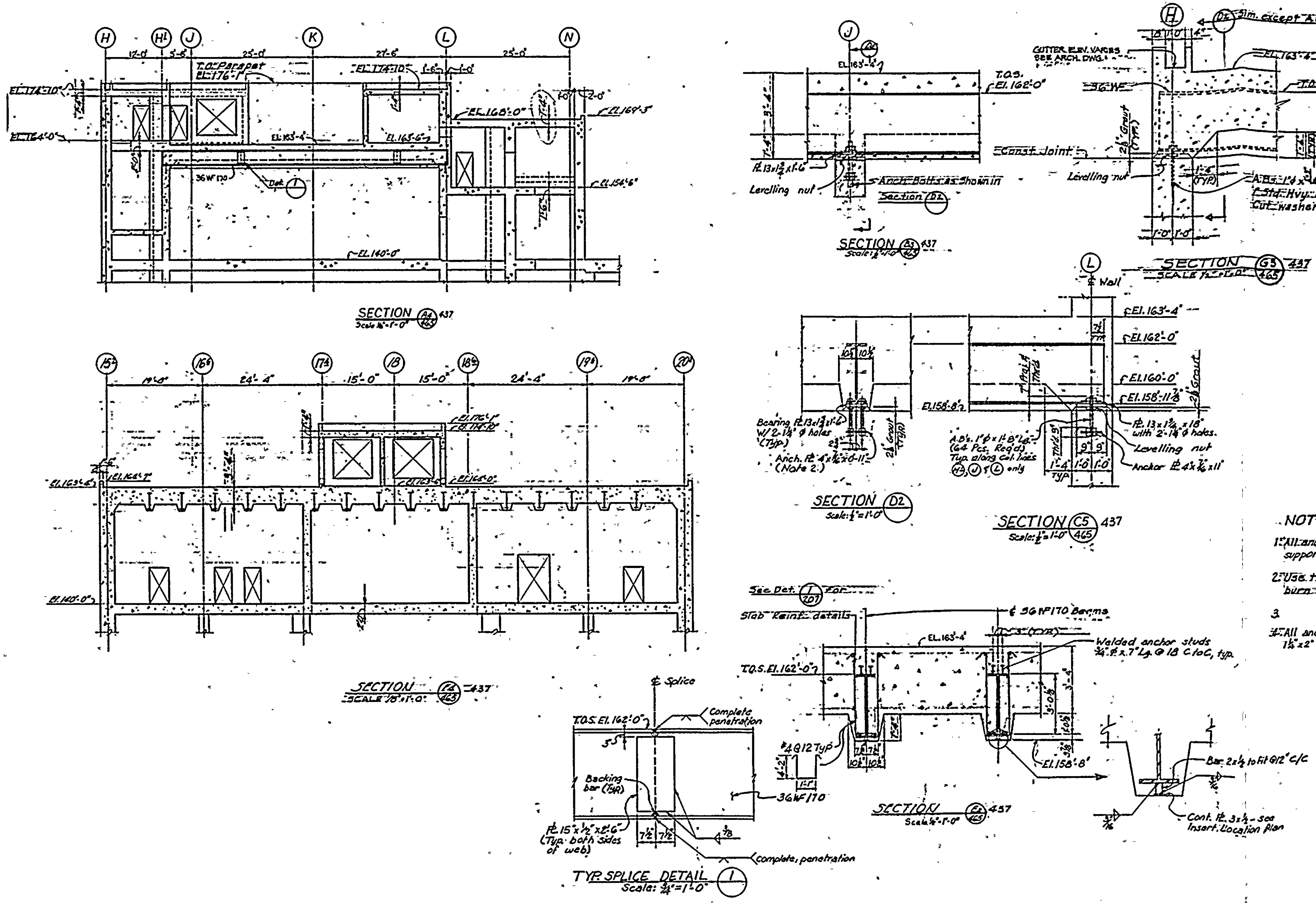
AUXILIARY BUILDING
CONCRETE REINFORCING
MISCELLANEOUS SECTIONS
AREA K

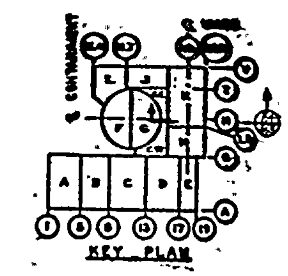


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-0J

AUXILIARY BUILDING
CONCRETE REINFORCING
SECTIONS
AREAS H, K, AND GE





NOTES

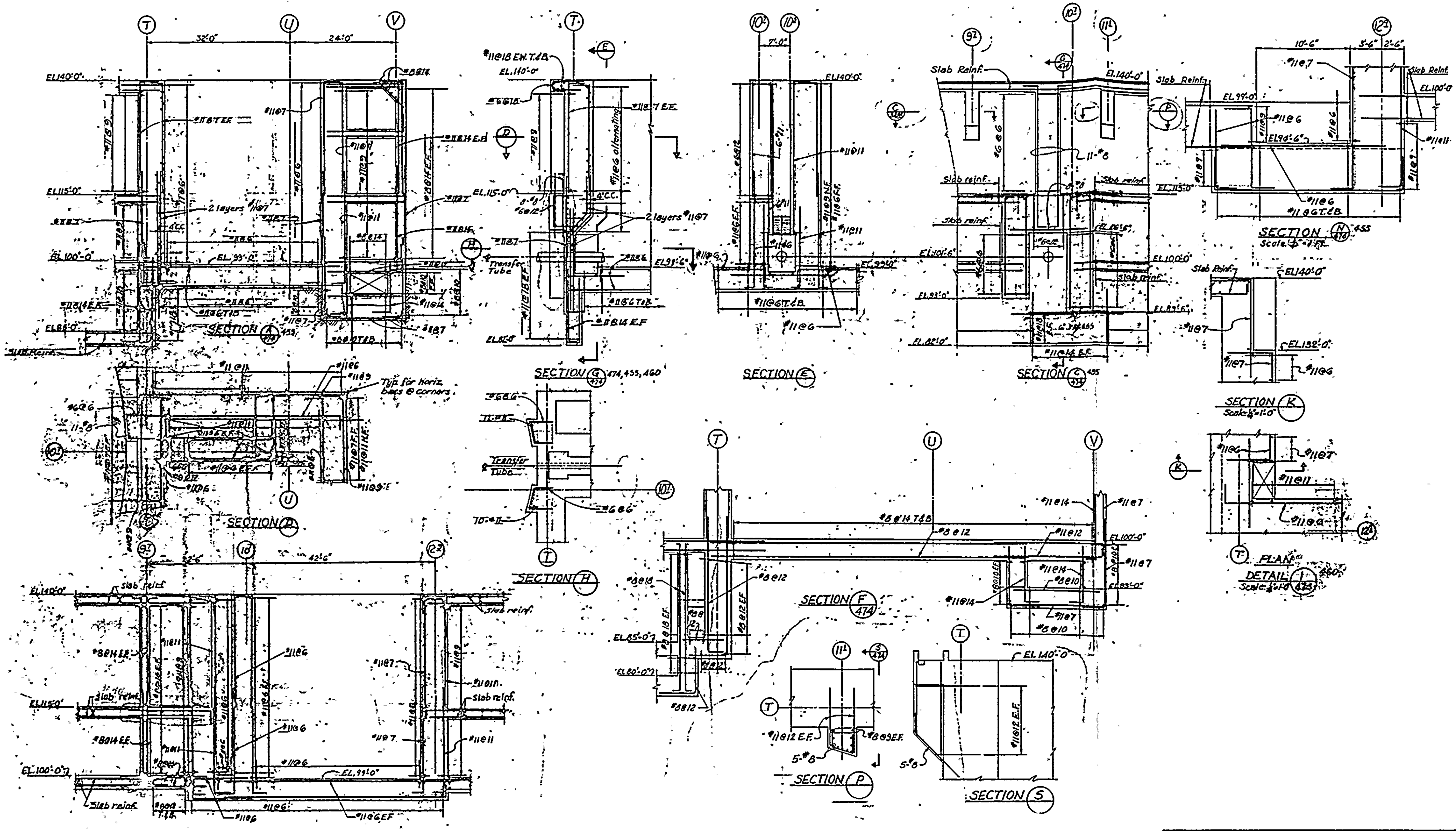
1. All anchor bolts shall be set at the centerline of supporting walls. Tolerance T2

2. Use the same anchor Pl.s as fabricated except field burn another hole to suit dim. shown in Sect. D2.

3. All anchor bolt holes in bottom flange of beams shall be 1½"x2" slots, long axis parallel to longit. ℄ of Beam.

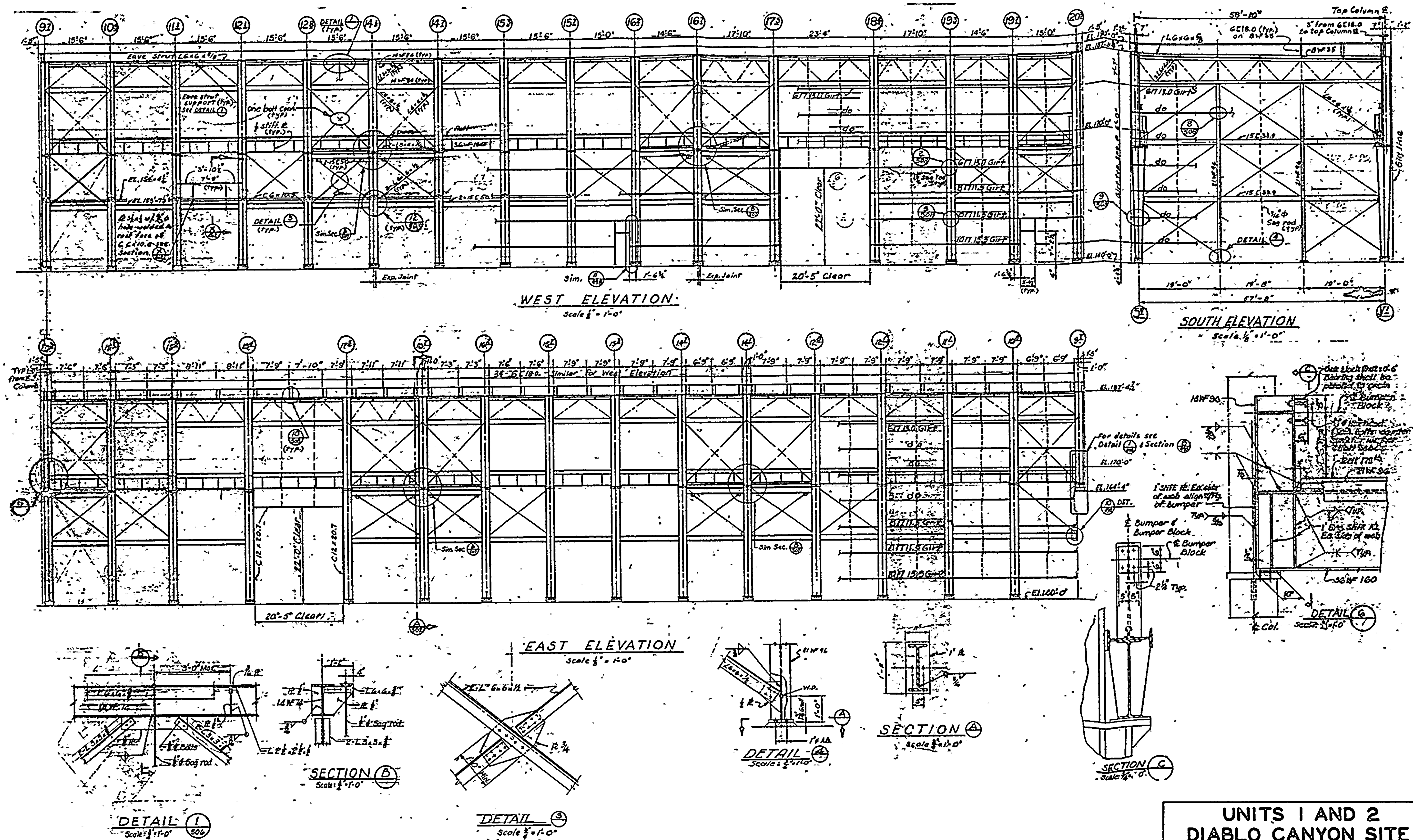
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-0K
AUXILIARY BUILDING
CONTROL ROOM
SECTIONS



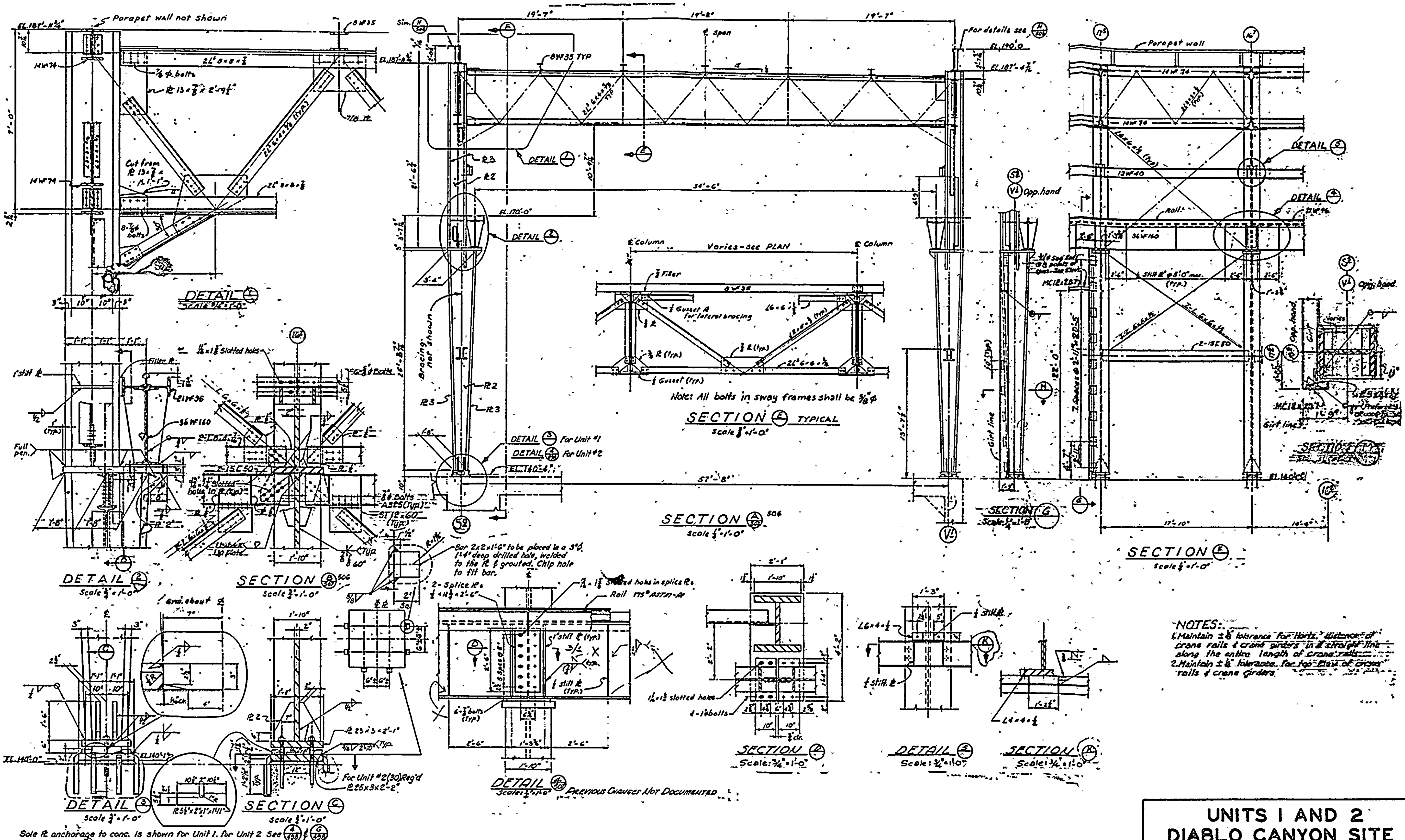
UNIT 1
DIABLO CANYON SITE

FIGURE 3.8-0L
AUXILIARY BUILDING
SPENT FUEL POOL
CONCRETE REINFORCING



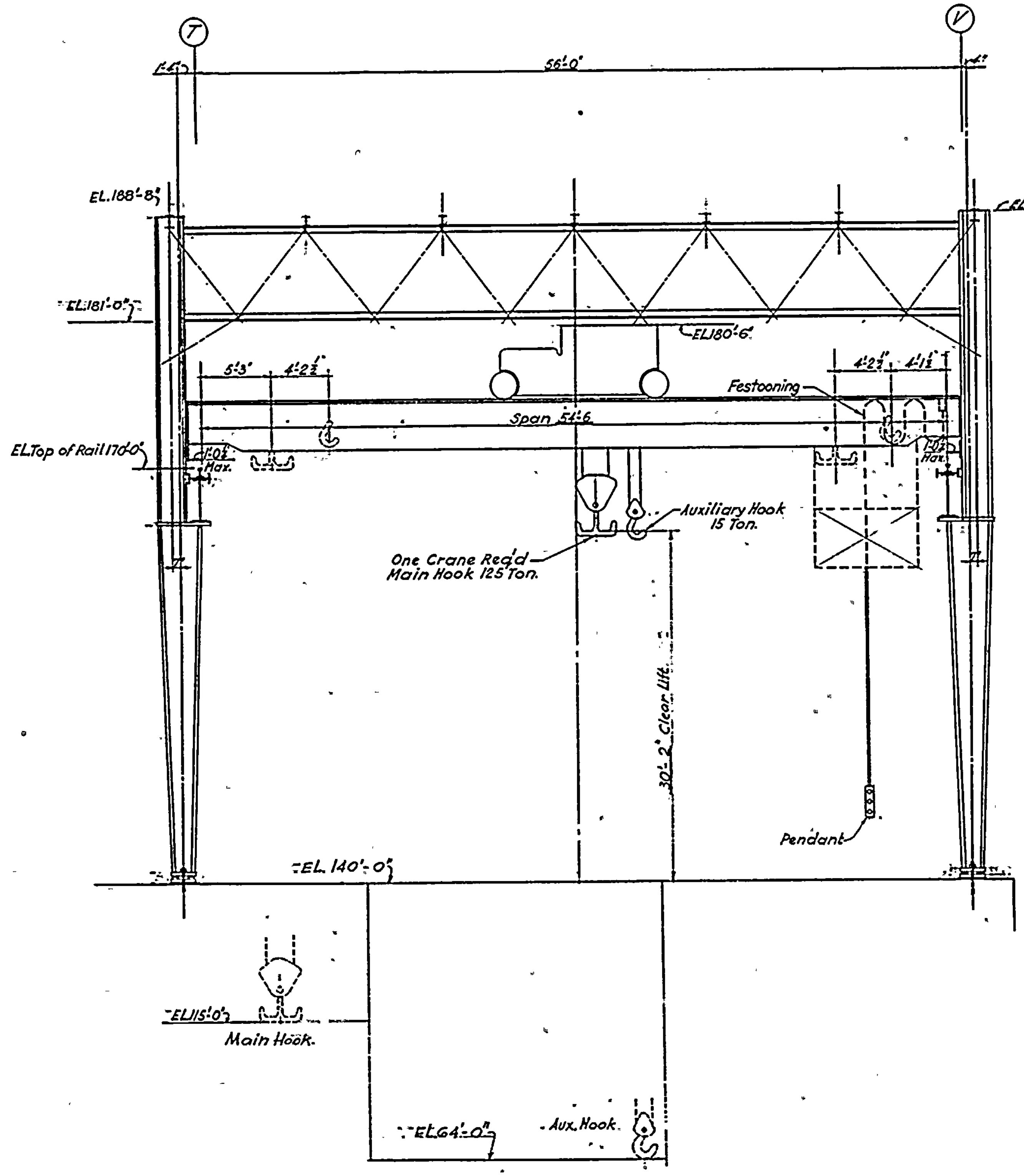
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-0M
AUXILIARY BUILDING
CRANE SUPPORT STRUCTURE
ELEVATIONS AND DETAILS

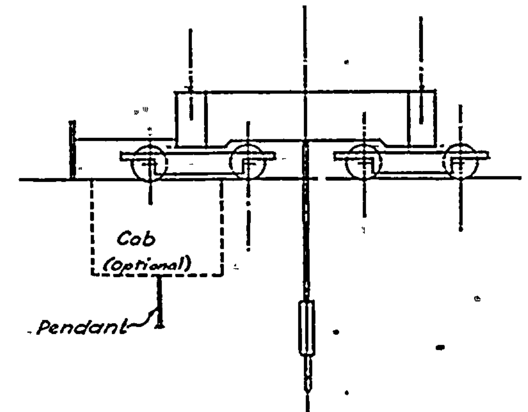


UNITS 1 AND 2
DIABLO CANYON SITE

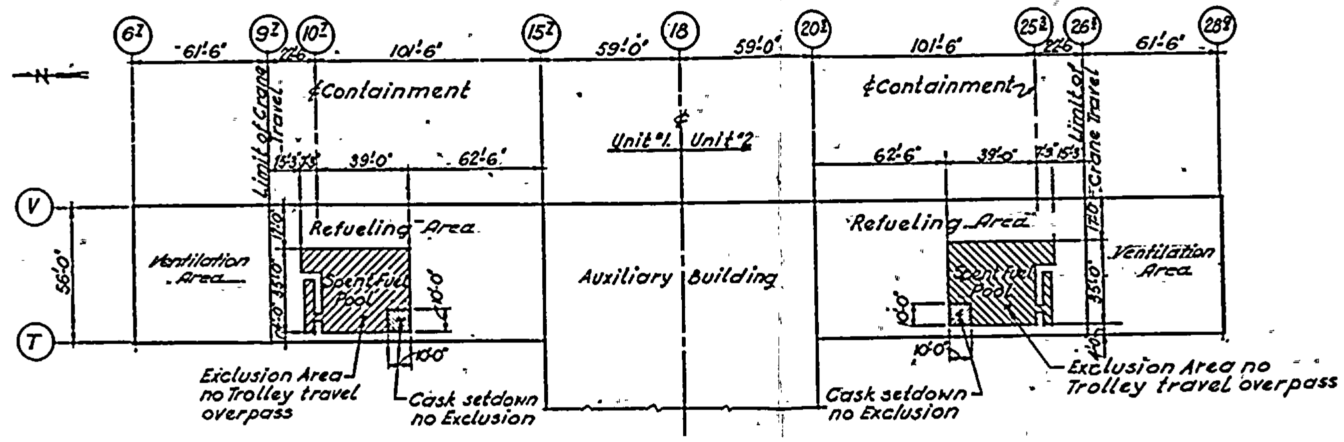
FIGURE 3.8-0N
AUXILIARY BUILDING
CRANE SUPPORT STRUCTURE
SECTION AND DETAILS



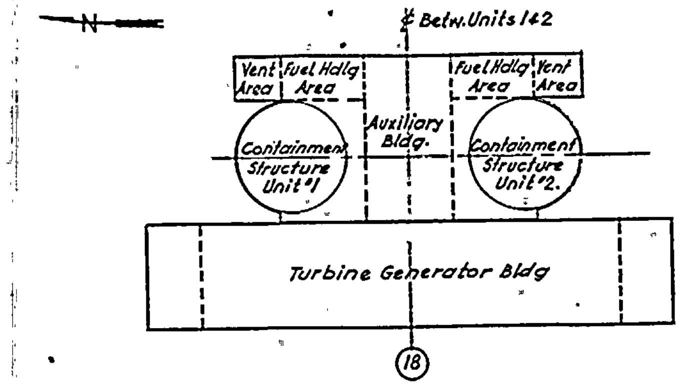
TYPICAL CROSS SECTION



SIDE VIEW



PLAN - TRAVEL LIMITS & EXCLUSION AREA
No Scale

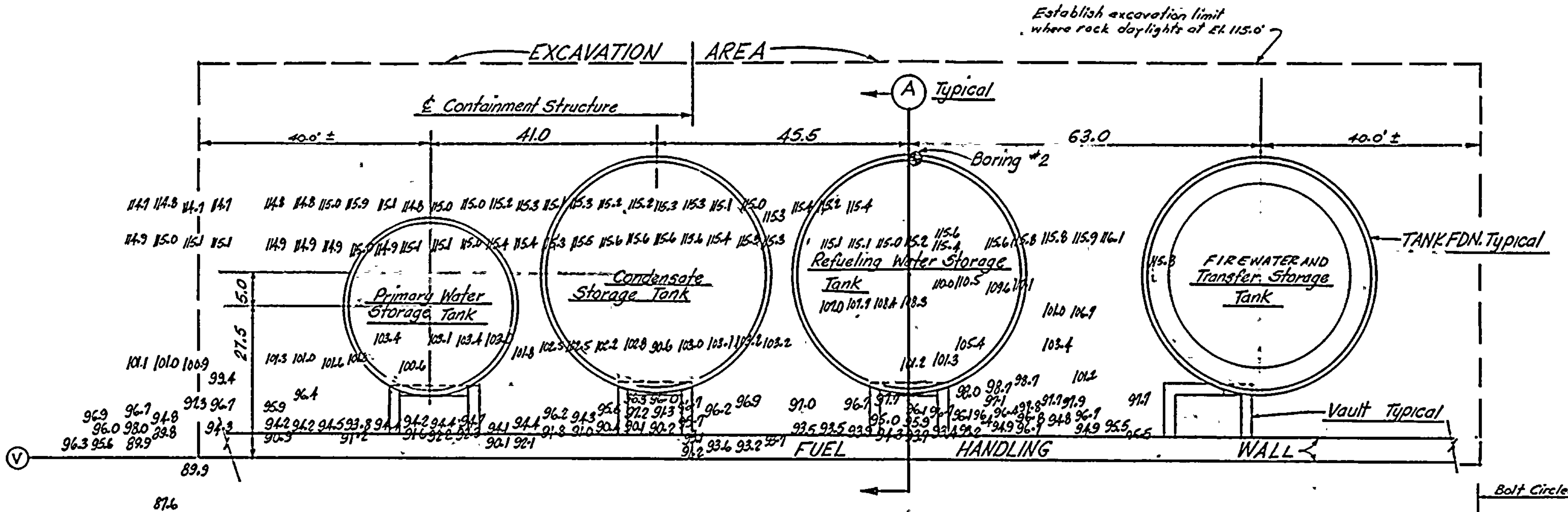


KEY-PLAN

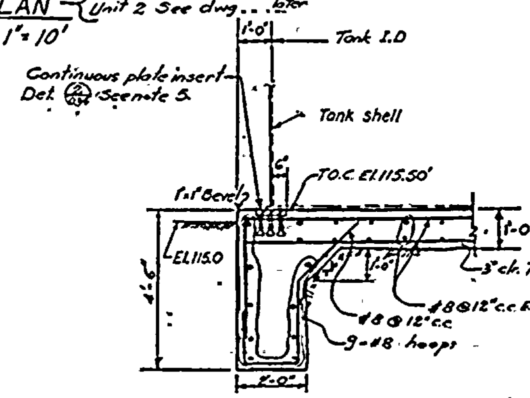
NOTE:
Crane data and clearances from Whiting Corp. handbook

UNITS 1 AND 2
DIABLO CANYON SITE

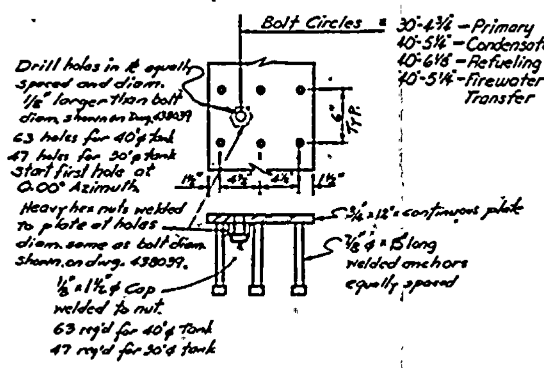
FIGURE 3.8-00
AUXILIARY BUILDING
REFUELING AREAS
OVERHEAD CRANE



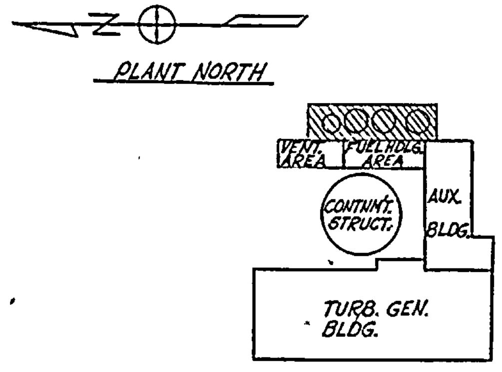
PLAN
{Unit 1 shown / Unit 2 See dwg. ...}
Scale 1" = 10'



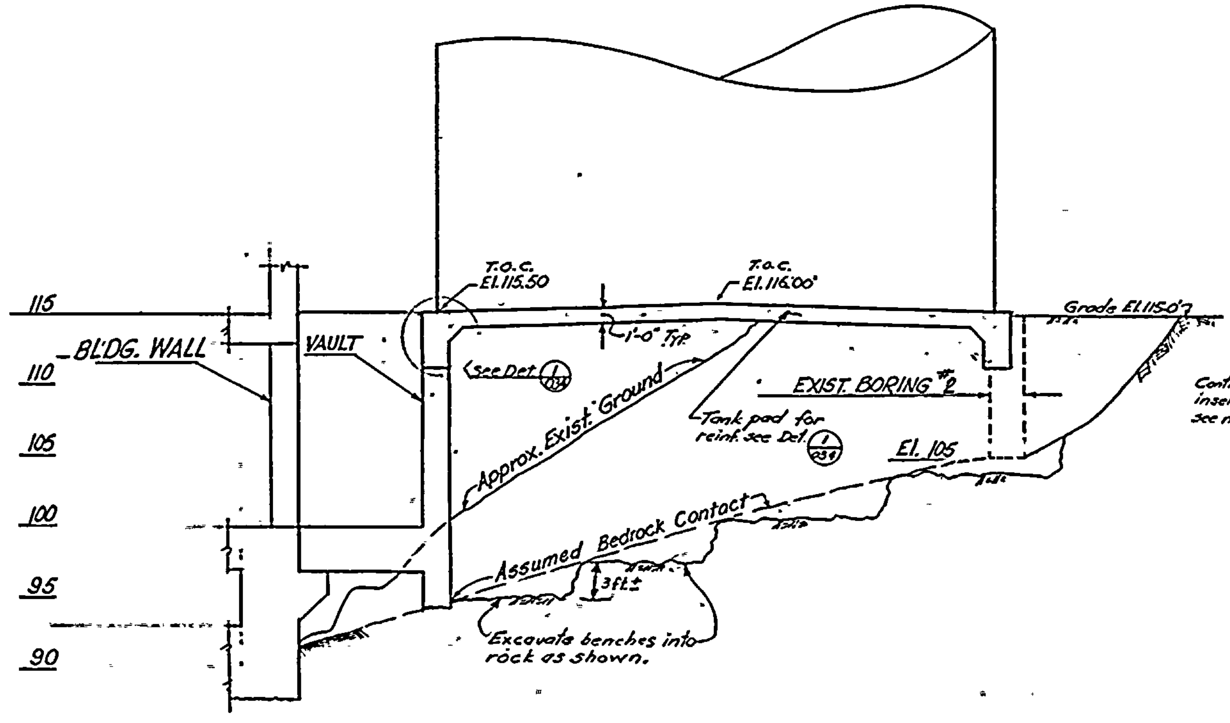
TYPICAL TANK FOUNDATION DETAIL 1/034
no scale



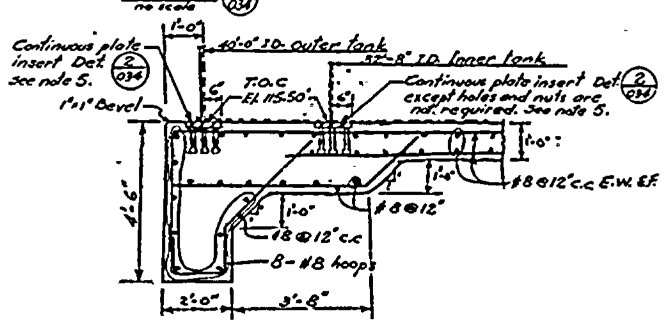
DETAIL 2/034
no scale
by Spec. 8831



KEY PLAN
NO SCALE



TYPICAL SECTION A
Scale 1" = 5'



FIREWATER AND TRANSFER TANK FOUNDATION DETAIL 3/034
no scale

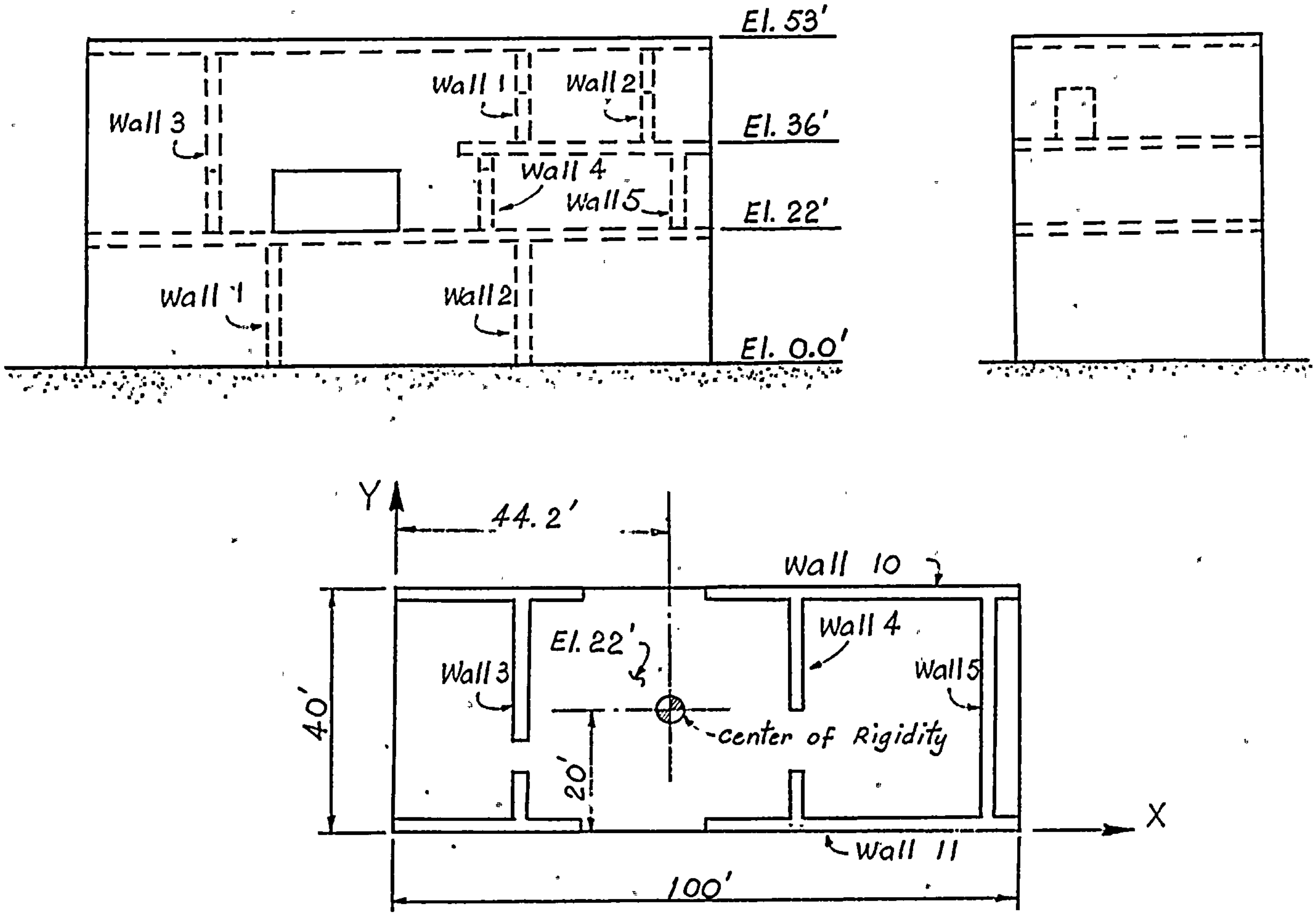
NOTES:

1. Excavation:
   a. Remove all the natural clays, & expose the sandstone or siltstone bedrock in the area shown in plan.
   b. After the entire area has been excavated & before any backfill is placed it shall be inspected by the engineer.
2. Backfill:
   a. Excavation shall be backfilled as soon as possible with gravelly, sandy silt from the plant site borrow area.
   b. The backfill shall be placed in 8 inch (max.) horiz. layers and compacted to 95% of the max. density obtainable by the ASTM D1557-66T (C) method of compaction.
3.
4. Numbers shown as 115.8 on plan, are existing ground elevations.
5. Material for plate inserts, bolts, nuts etc. (except anchor bolts and nuts) shall be stainless steel for Primary water and Refueling Water Tanks, Carbon Steel for Condensate and Transfer Tanks. Stainless Steel shall be ASTM A240 Type 304. Anchor bolts and nuts per Note 7.
6. Provide a 6" x 6" blockout in tank fdn. for tank bottom drains. For location see Tank Requirement dwgs. Fill block out with grout after tank installation.
7. Anchor bolts and nuts shall meet requirements of ASTM A-193 Gr. B7.

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-0P

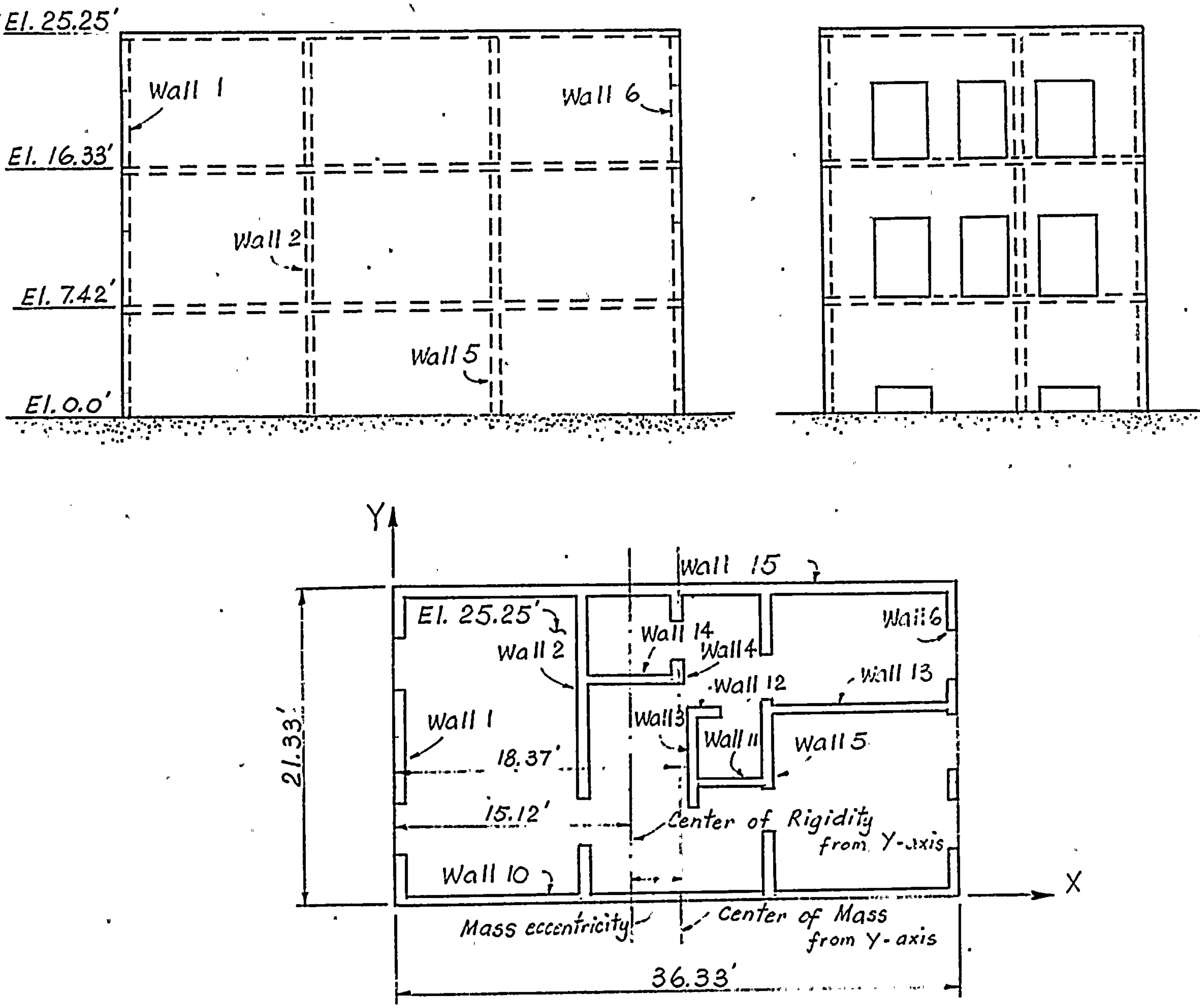
DESIGN CLASS I TANKS
CONCRETE FOUNDATIONS



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-0Q
COMPUTER PROGRAM VERIFICATION
DYBOX 2

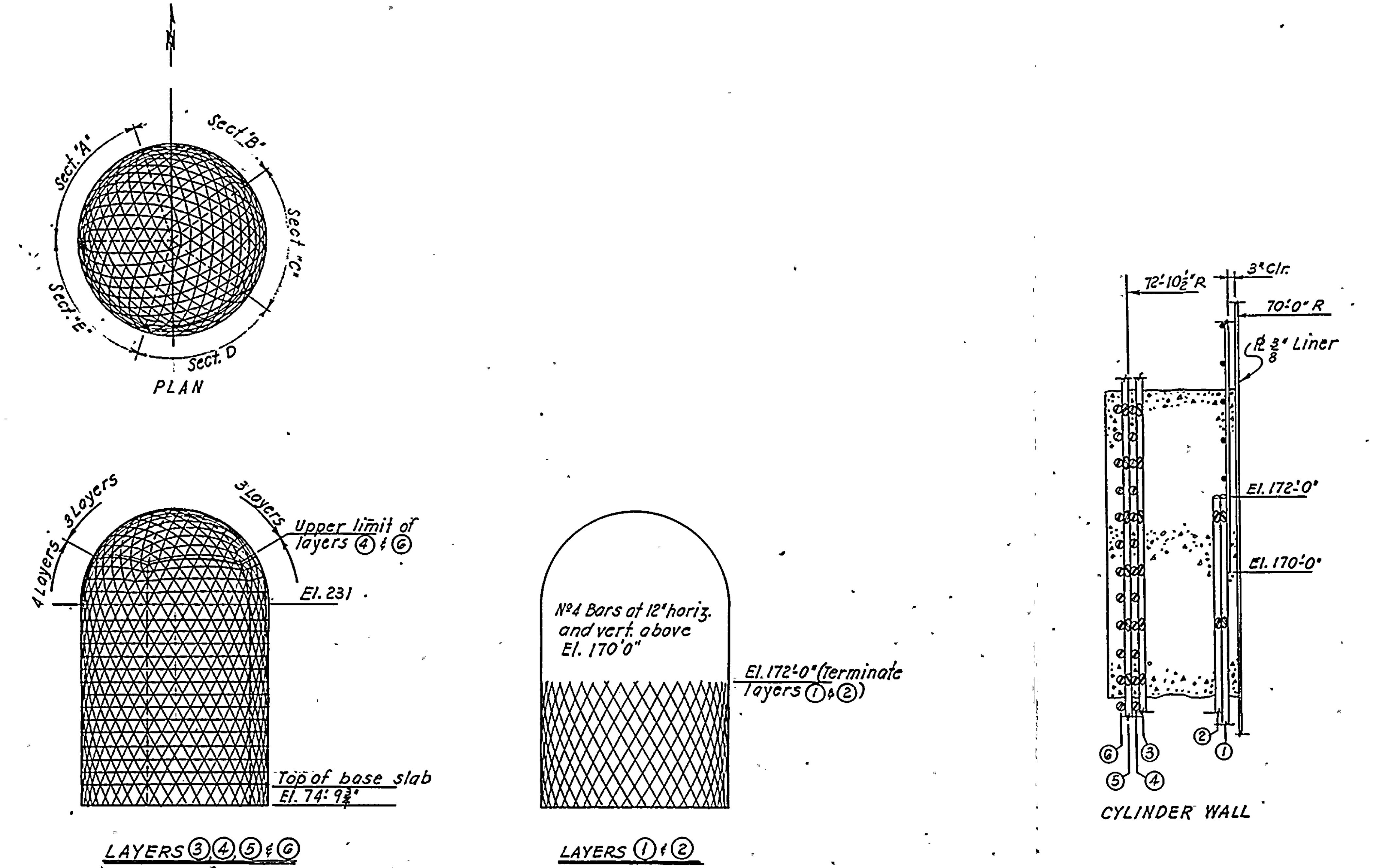
Amendment 3 February 1974



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-0R

COMPUTER PROGRAM VERIFICATION
SHEARWALL 4

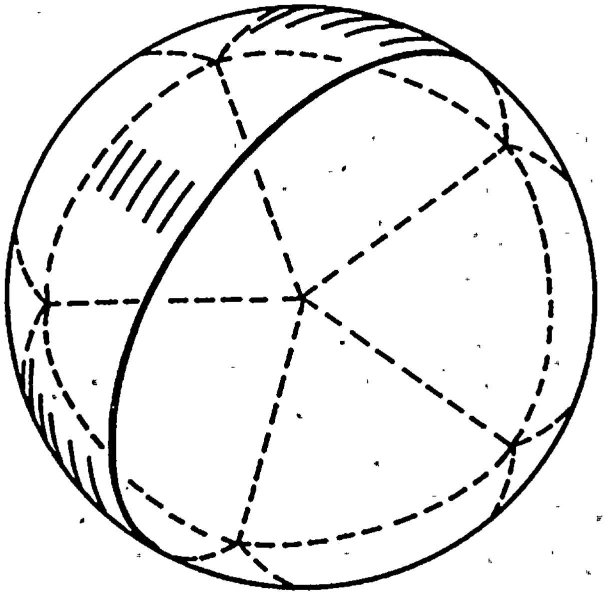


UNITS 1 AND 2
DIABLO CANYON SITE

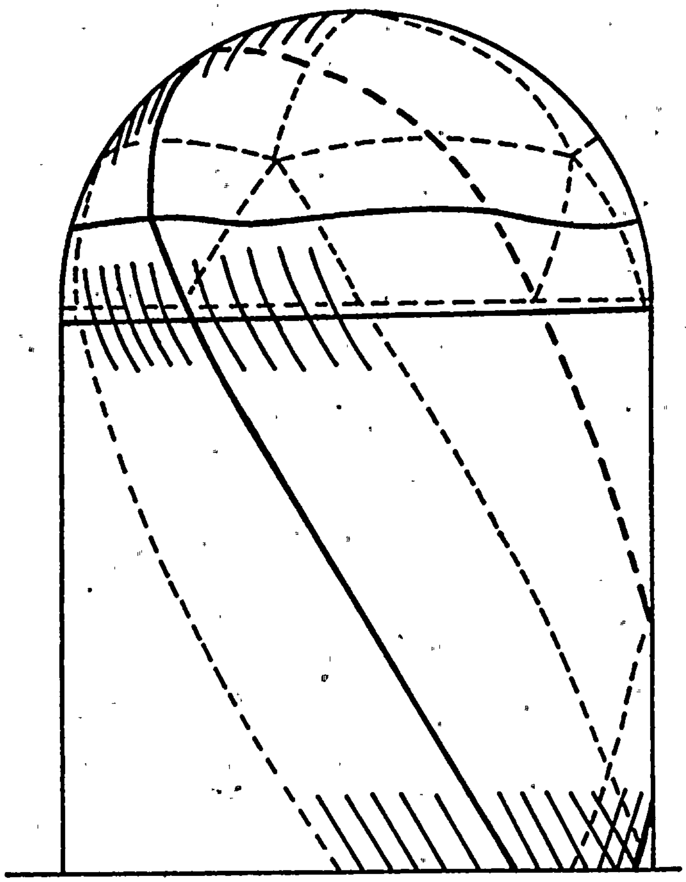
FIGURE 3.8-1

CONTAINMENT STRUCTURE

REINFORCING STEEL ARRANGEMENT



PLAN



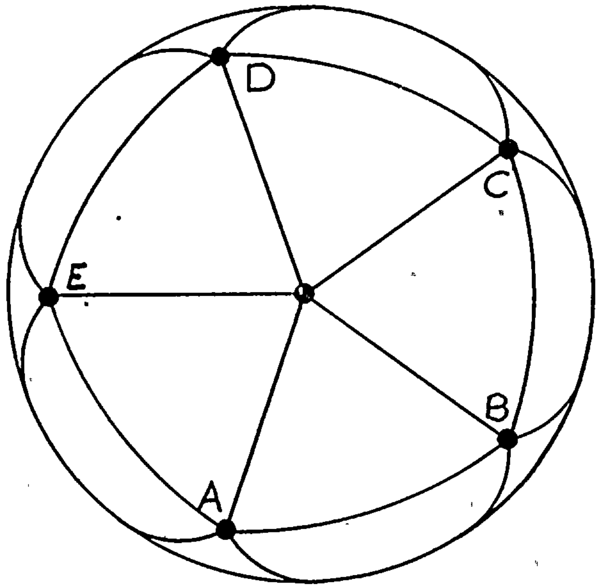
ELEVATION

UNITS 1 AND 2
DIABLO CANYON SITE

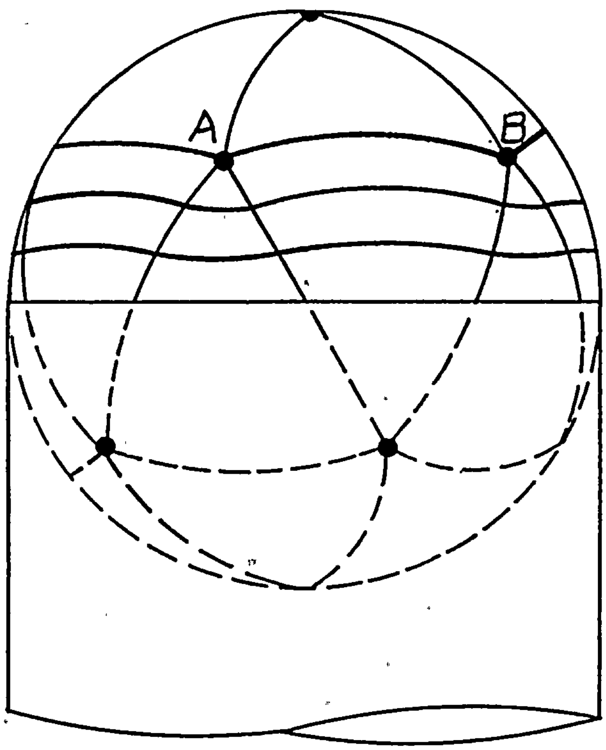
FIGURE 3.8-2

CONTAINMENT STRUCTURE

TYPICAL REINFORCING LOOP

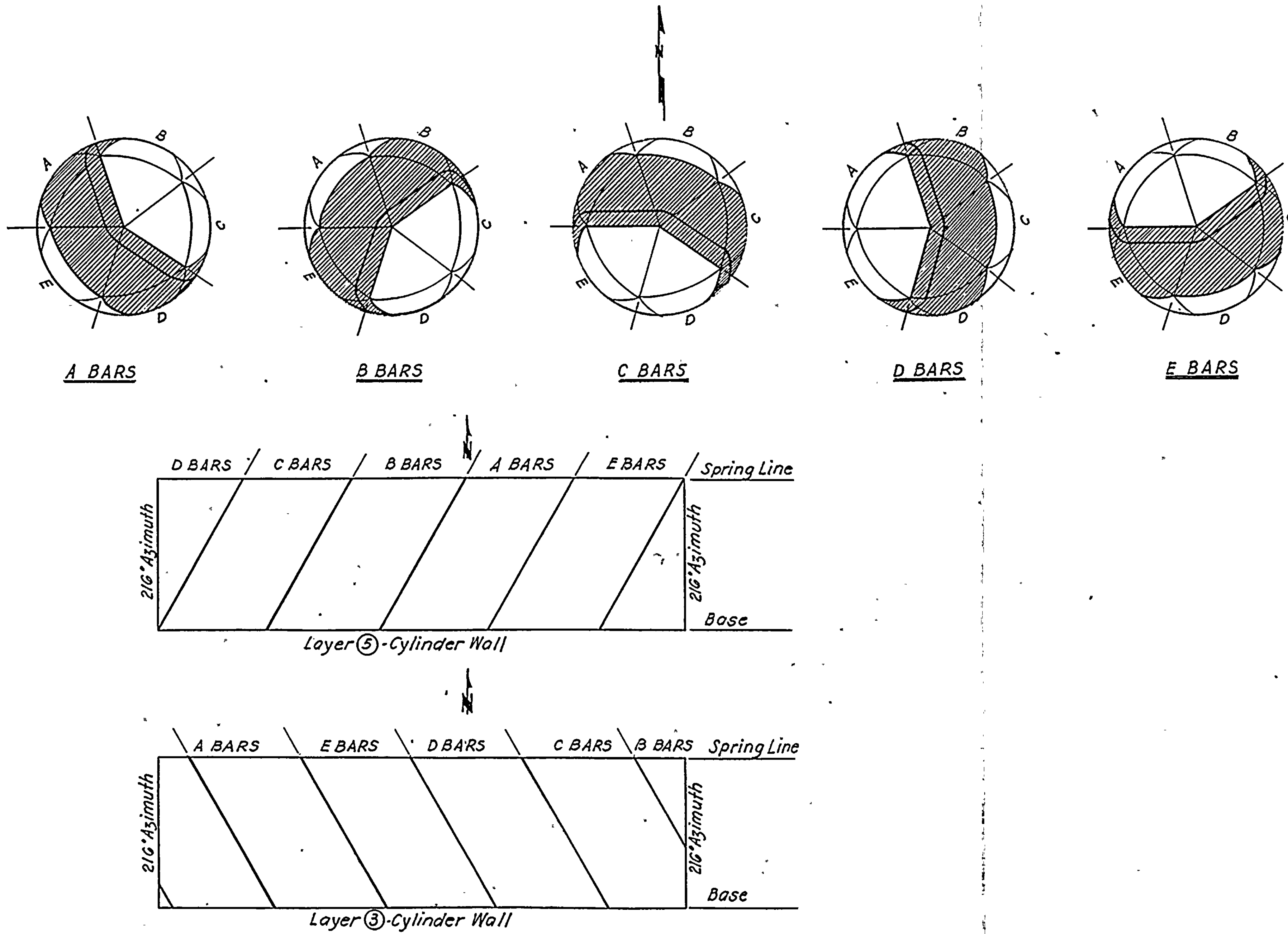


SKETCH 1



SKETCH 2

UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 3.8-3 CONTAINMENT STRUCTURE DOME SPHERICAL TRIANGLES

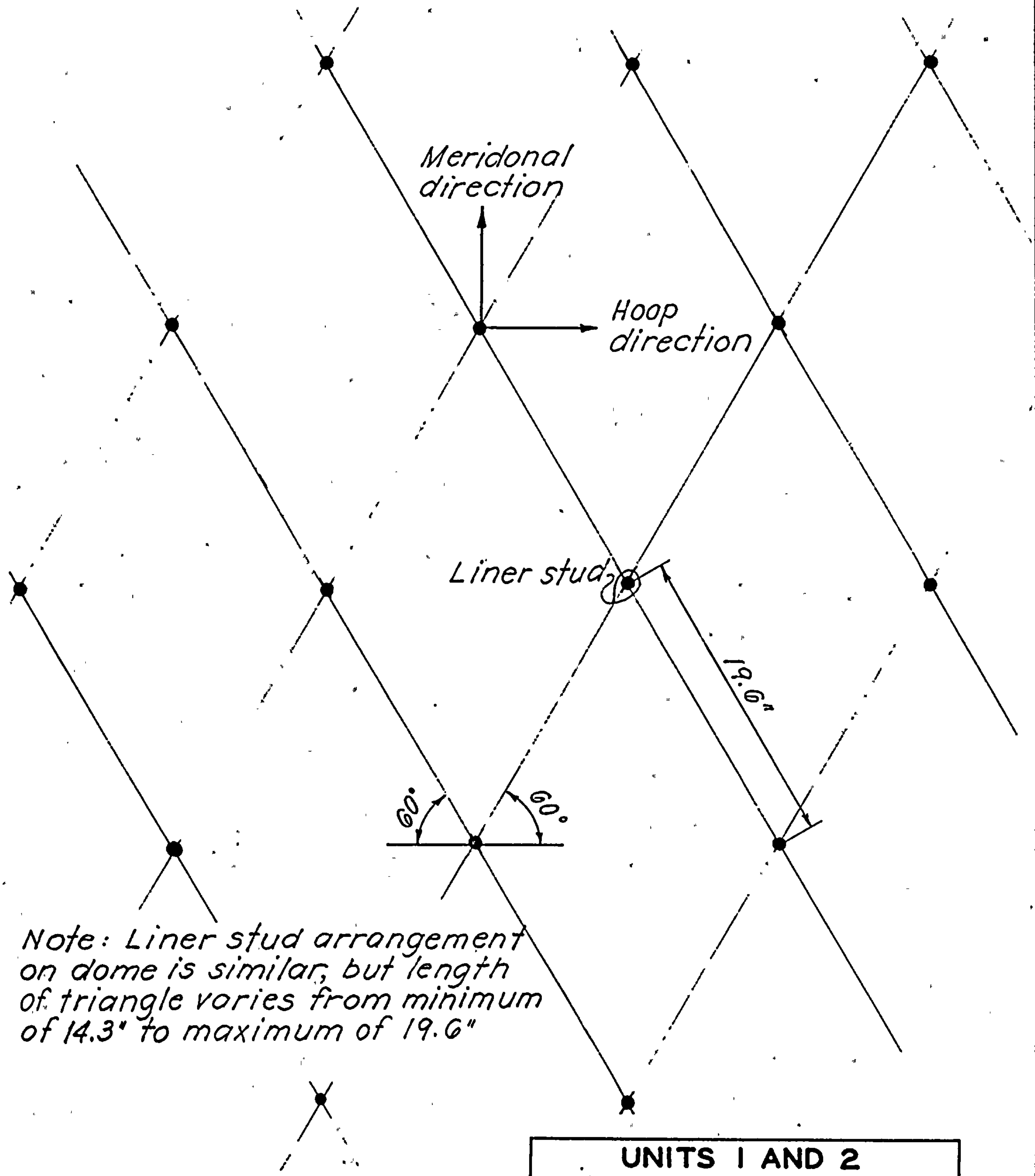


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-4

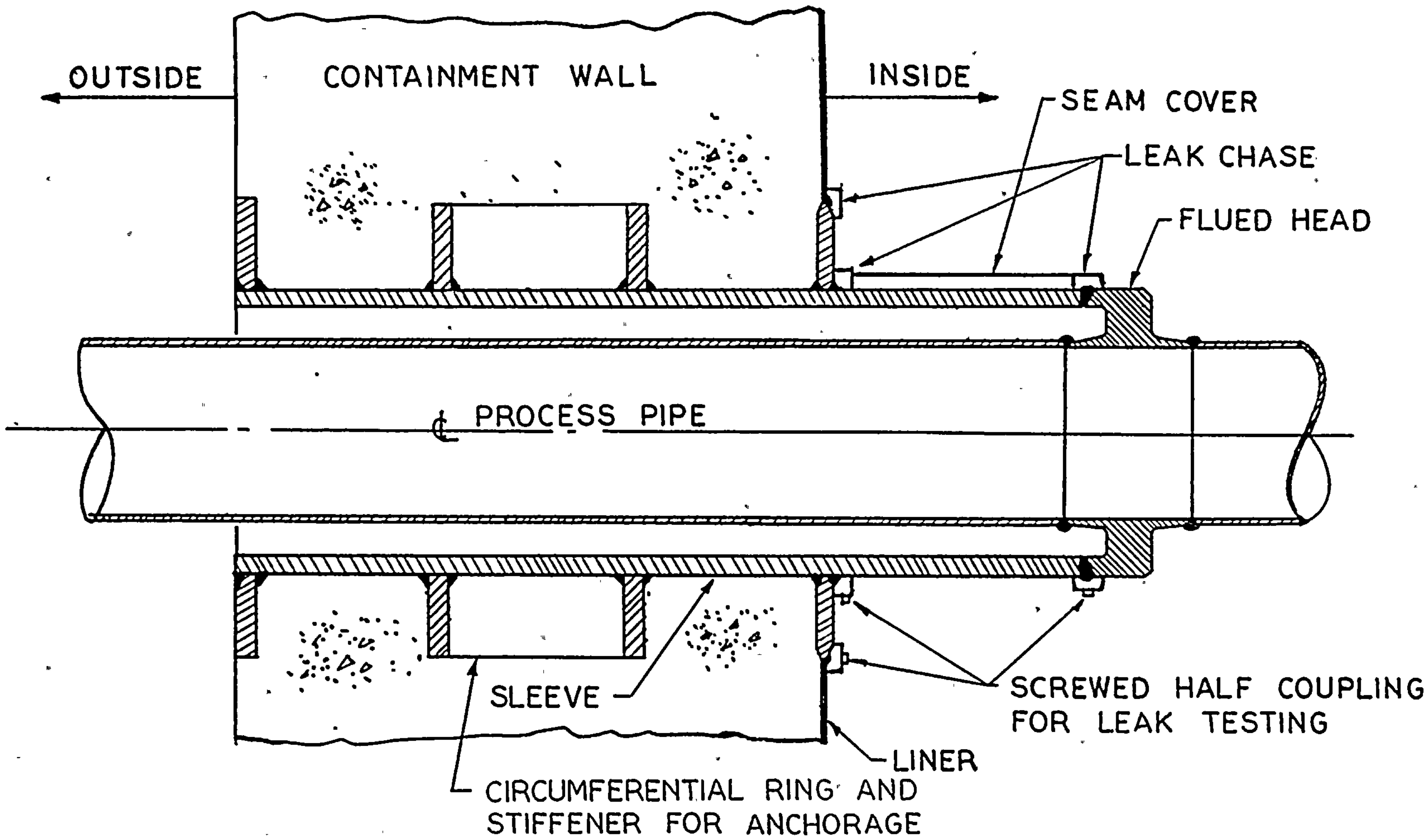
CONTAINMENT STRUCTURE

DOME AND CYLINDER BARS



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-5
CONTAINMENT STRUCTURE
LINER STUD ARRANGEMENT

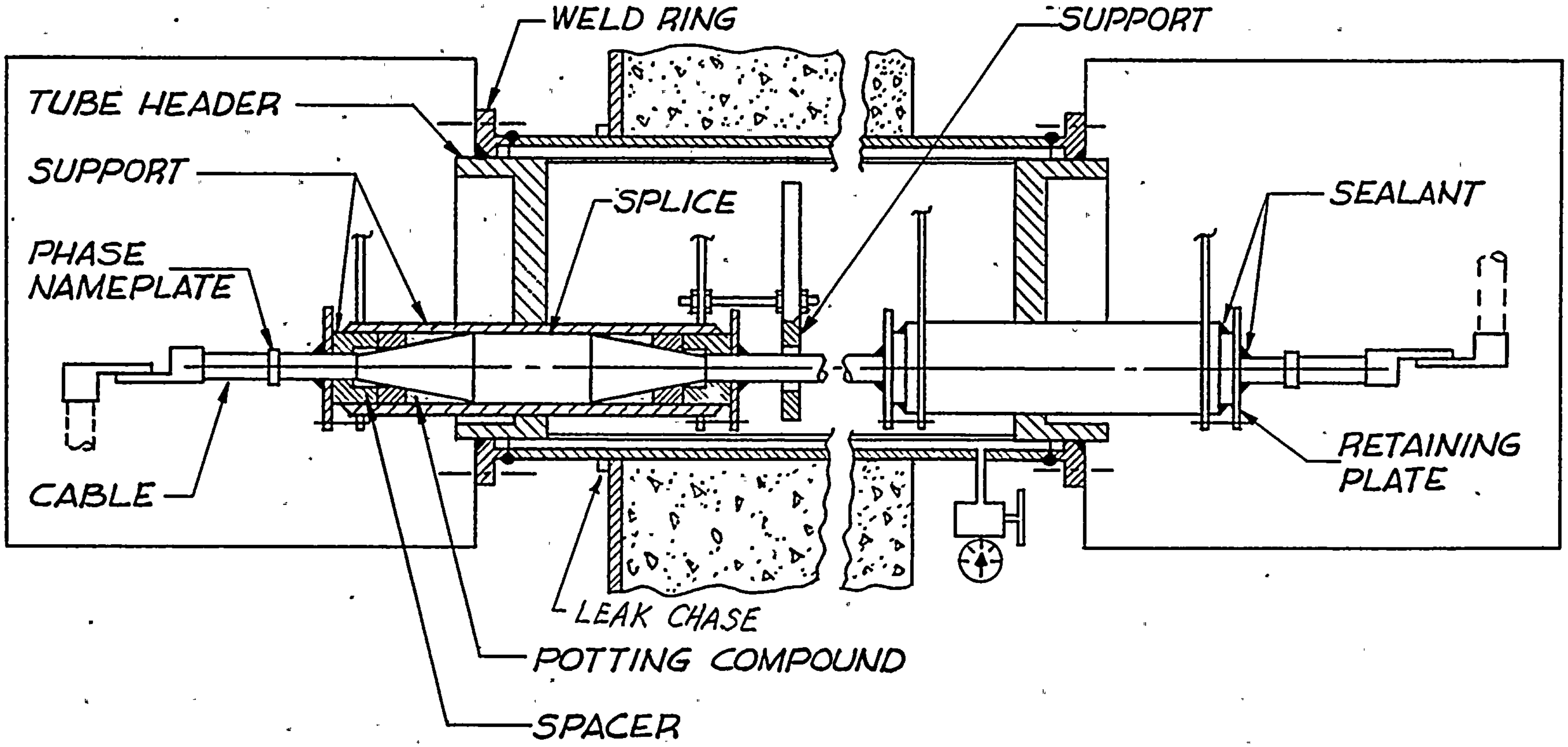


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-6

CONTAINMENT STRUCTURE

TYPICAL PIPING PENETRATION

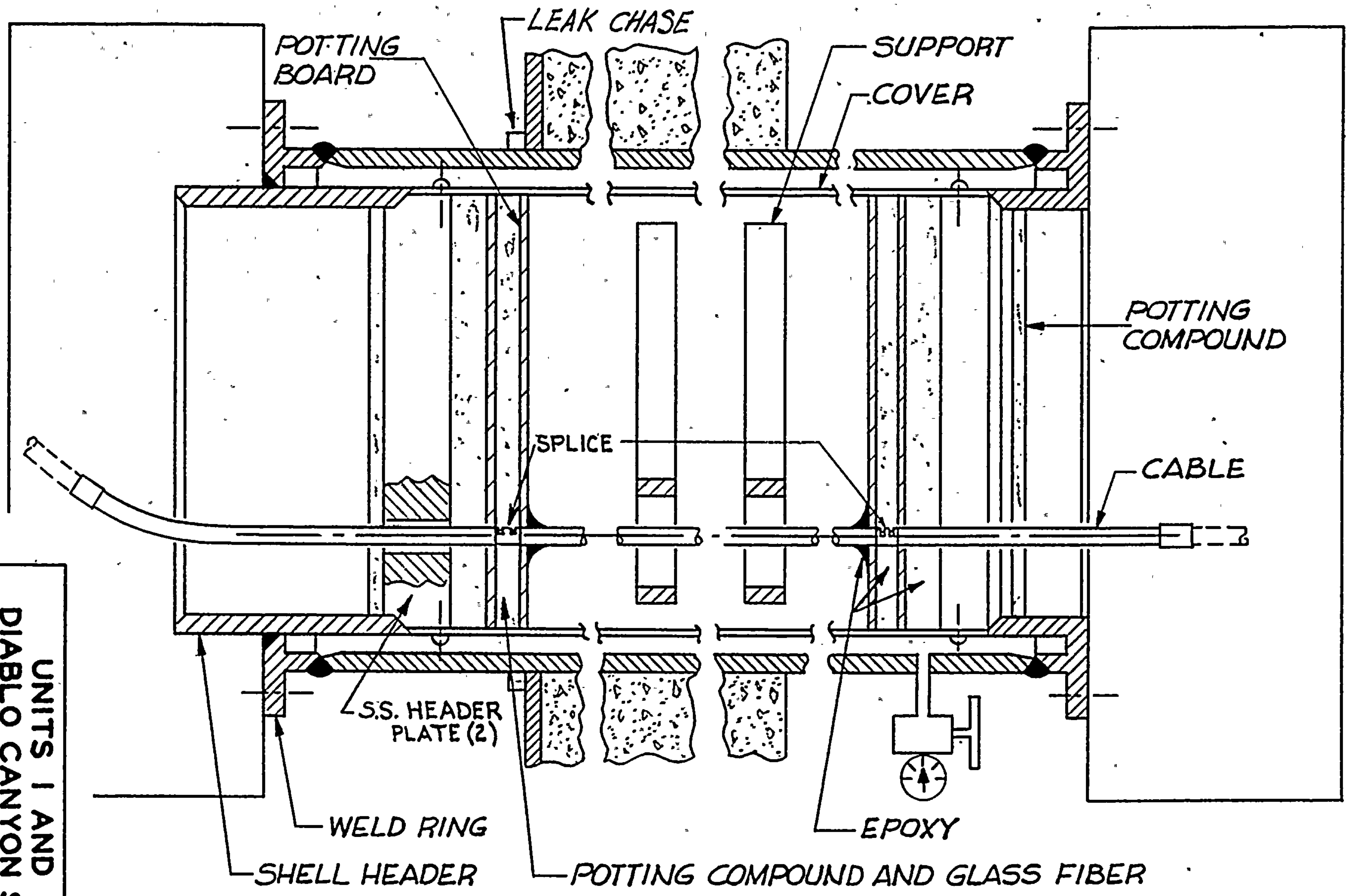


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-8

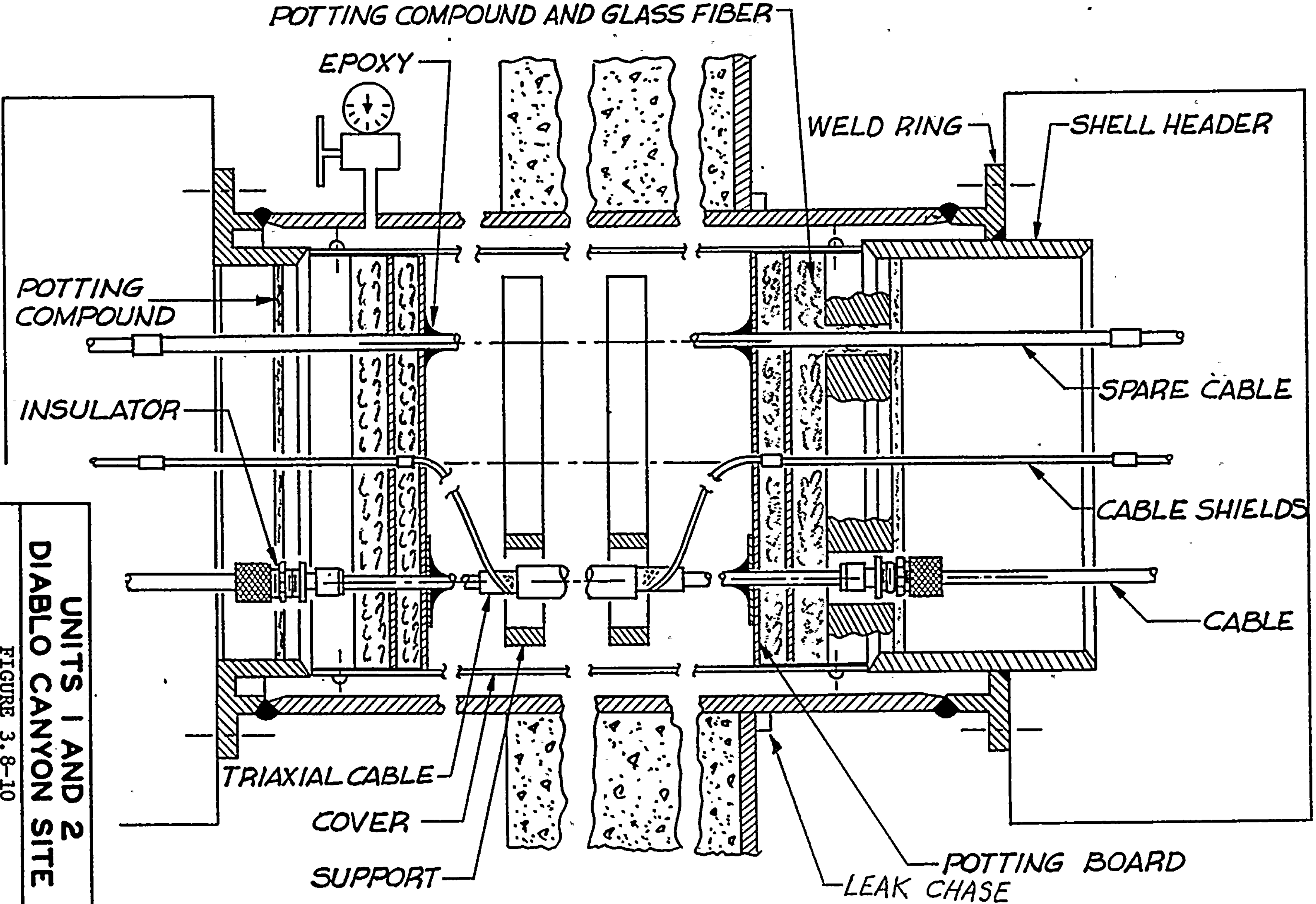
CONTAINMENT STRUCTURE

TYPICAL 12 KV POWER PENETRATION

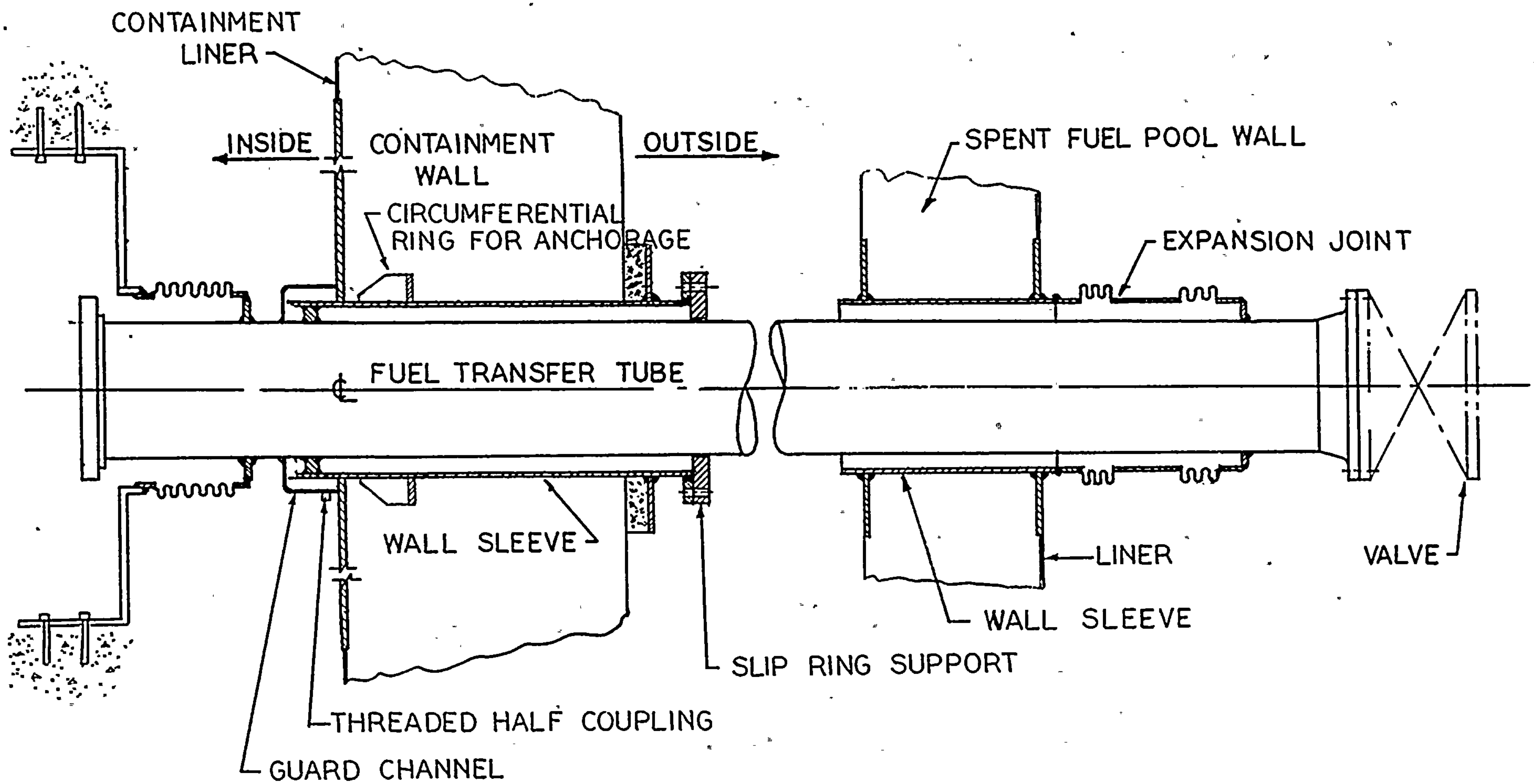


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-9
CONTAINMENT STRUCTURE
TYPICAL LOW VOLTAGE PENETRATION



UNITS 1 AND 2
DIABLO CANYON SITE
FIGURE 3.8-10
CONTAINMENT STRUCTURE
TYPICAL INSTRUMENTATION PENETRATION

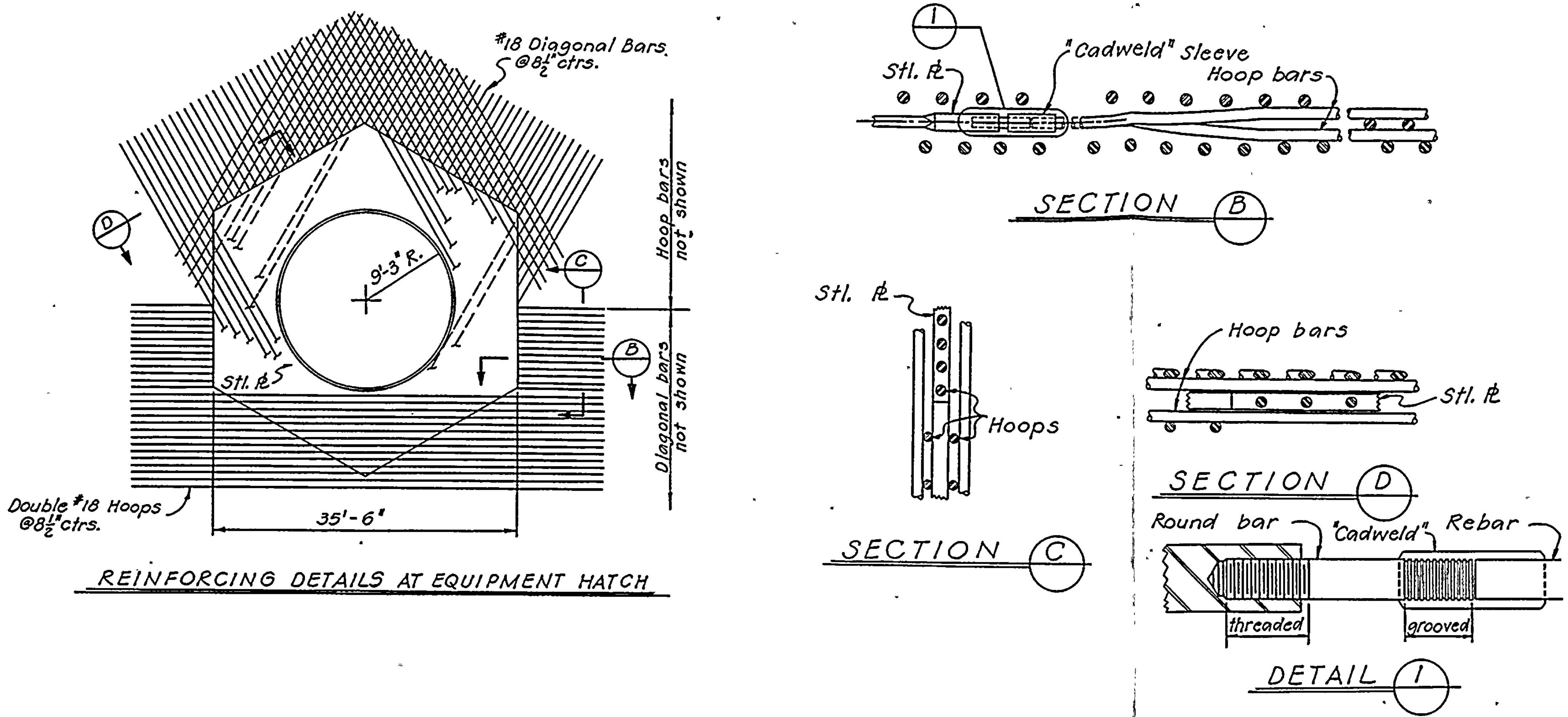


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-11

CONTAINMENT STRUCTURE

FUEL TRANSFER TUBE PENETRATION

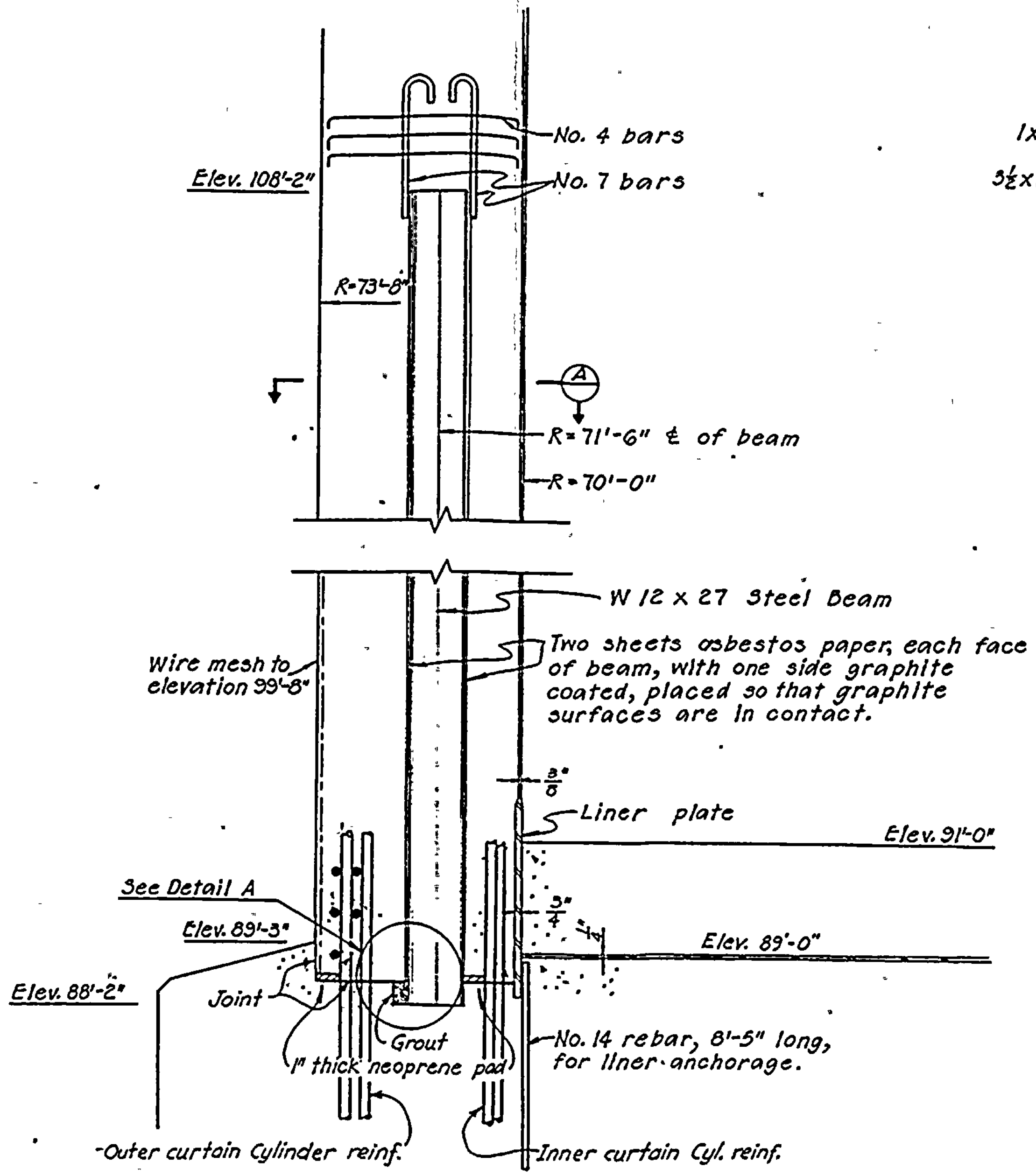


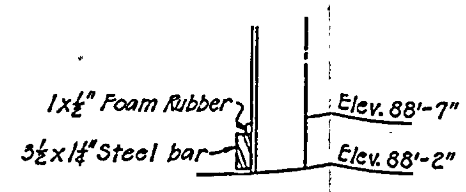
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-12

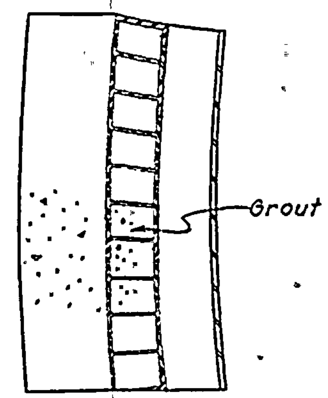
CONTAINMENT STRUCTURE

HEXAGONAL COLLARS





DETAIL A



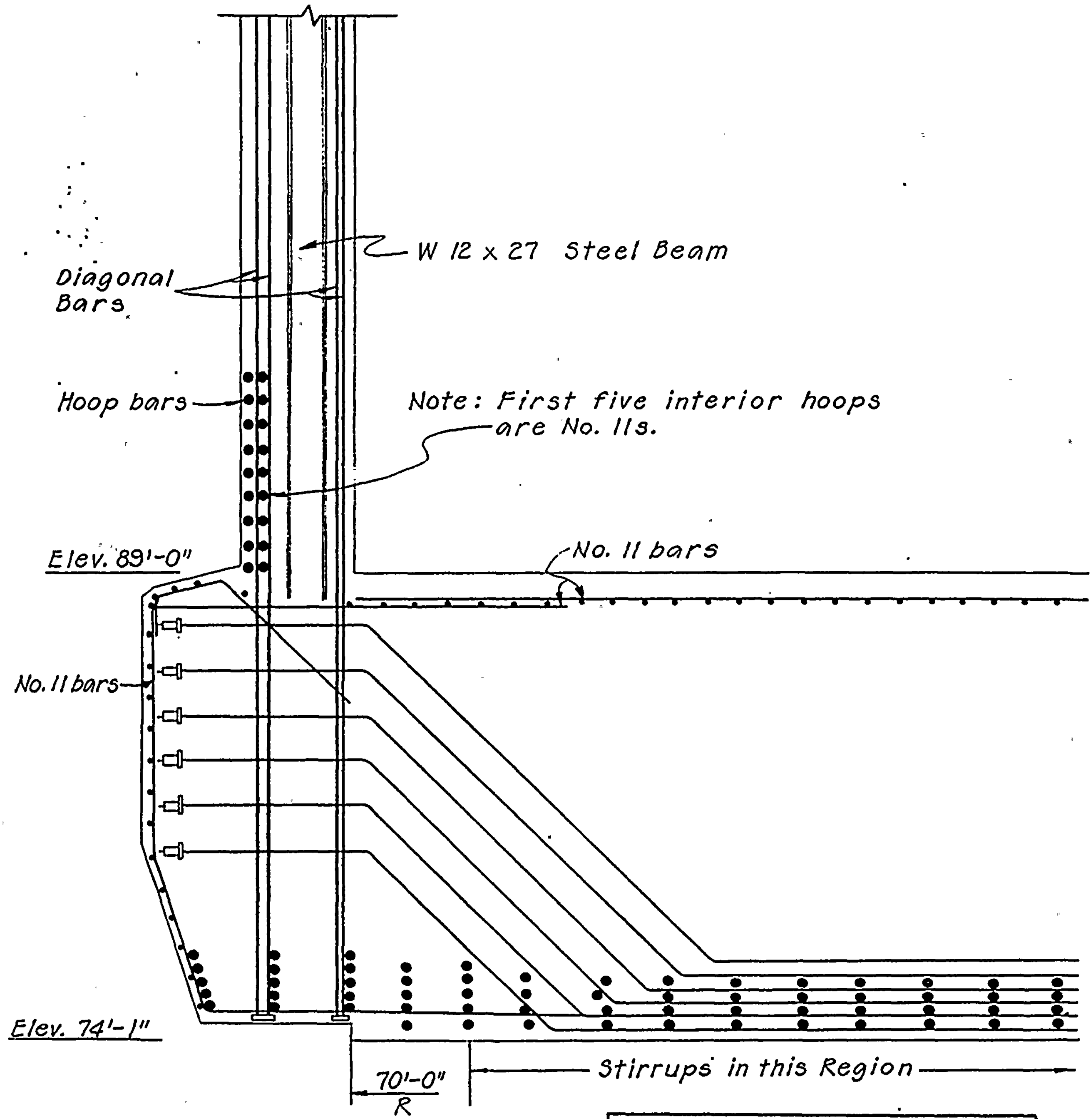
SECTION (A)

NOTES:

SEE FIGURE 3.8-15 FOR REINFORCING STEEL IN THIS REGION.

UNITS 1 AND 2
DIABLO CANYON SITE

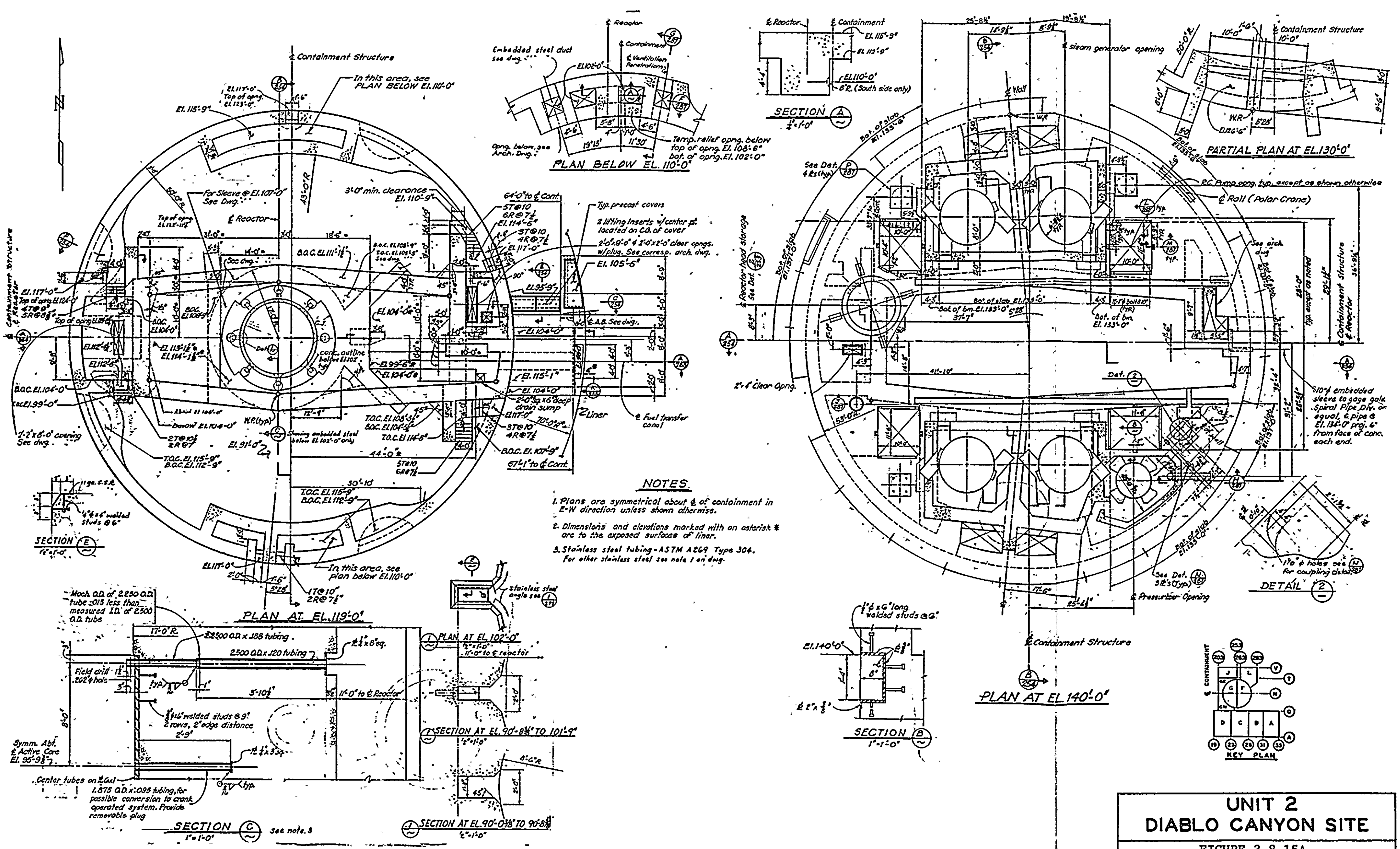
FIGURE 3.8-14
CONTAINMENT STRUCTURE
EMBEDDED BEAMS
CYLINDER-BASE SLAB JUNCTURE



Bars are No. 18 unless noted.

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-15
CONTAINMENT STRUCTURE
REINFORCING STEEL
BASE SLAB AND WALL

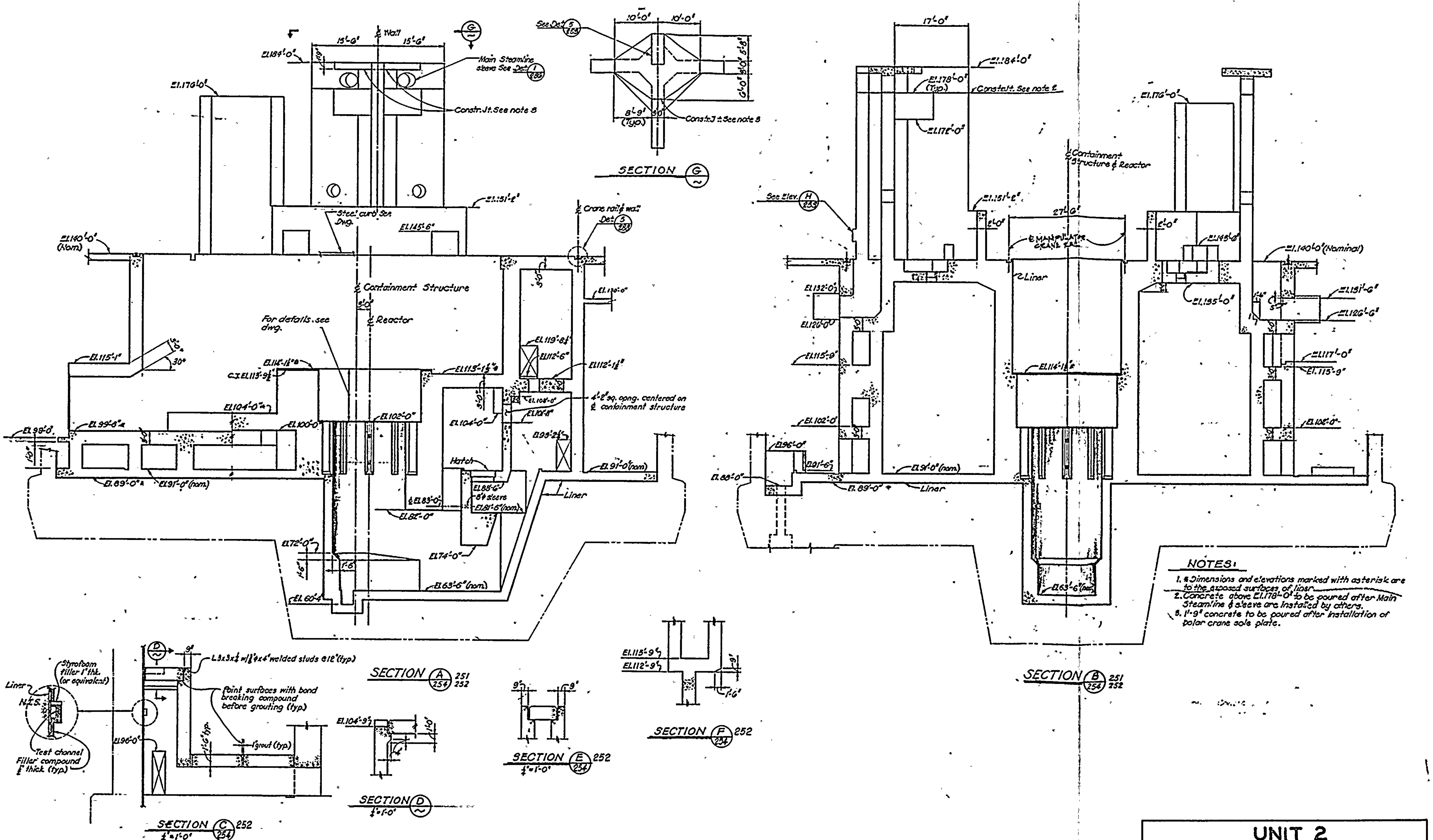


NOTES

1. Plans are symmetrical about ℄ of containment in E-W direction unless shown otherwise.
2. Dimensions and elevations marked with an asterisk * are to the exposed surfaces of liner.
3. Stainless steel tubing-ASTM A269 Type 304. For other stainless steel see note 1 on dwg.

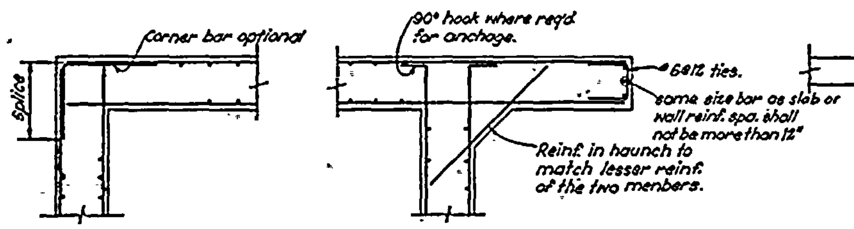
UNIT 2
DIABLO CANYON SITE

FIGURE 3.8-15A
INTERNAL STRUCTURE
CONCRETE OUTLINE
ELEVATION 119 AND 140

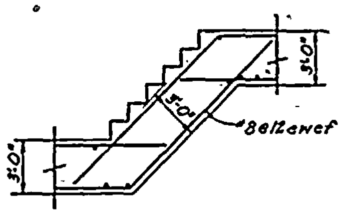


UNIT 2
DIABLO CANYON SITE

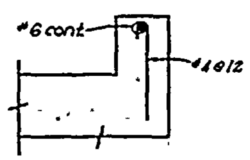
FIGURE 3.8-15B
INTERNAL STRUCTURE
CONCRETE OUTLINE
SECTIONS



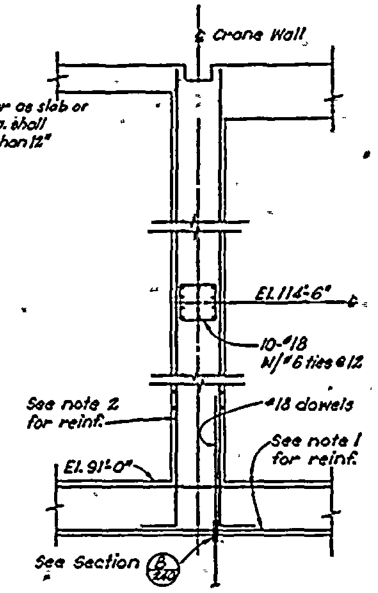
WALL AND SLAB DETAIL



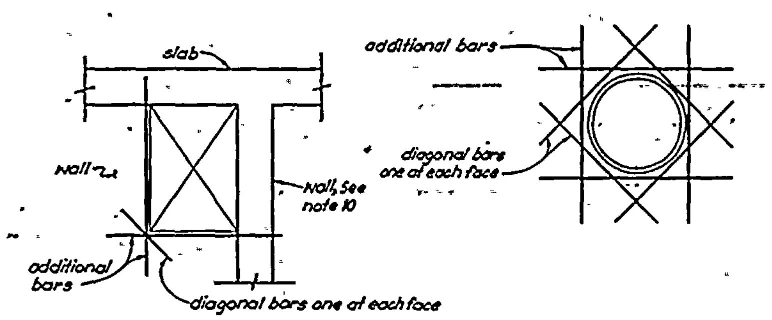
STAIR DETAIL



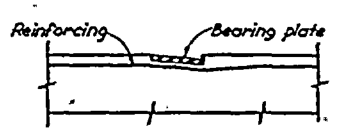
CURB



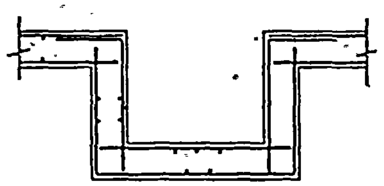
CRANE WALL REINFORCING



REINFORCING AT WALL OPENING
See Notes 8 & 9



DETAIL OF REINFORCING BENT AROUND EQUIPMENT LATERAL SUPPORT BEARING ℙ



SUMP DETAIL

TYPICAL REINFORCING
Where Not Shown Otherwise

1. Concrete slabs and walls below EL 91'-0" but above liner to be #6@12 e.w.e.f.
2. Circular crane wall to be #11@12 e.f. vertically and horizontally, in addition stirrups #6@12 between azimuths 25° to 65°, 120° to 155°, 200° to 250° and 290° to 340° shall be placed from EL 91'-0" to EL 106'-0".
3. All #11 bars to have anchorage and splice lengths as required for Ultimate Strength Design unless otherwise shown.
4. Other Walls.

Thickness	Reinforcing
3'-0" and more	#11@12 e.f. Vert. and Hor.
more than 2'-0" but less than 3'-0"	#8@12 e.f. " " "
1'-0" to 2'-0" inclusive	#6@12 e.f. " " "
less than 1'-0"	#6@12 Vert. and Hor. in the Ctr.

5. Slab at El. 140'-0", #11@12 e.w.t. & b. unless otherwise shown.
6. Other Slabs.

Thickness	Reinforcing
less than 2'-0"	#6@12 e.w.e.f.
2'-0" to 3'-0"	#8@12 e.w.e.f.
more than 3'-0"	#11@9 e.w.e.f.

7. Irregular concrete blocks and shapes.

the least Dimension	Reinforcing
less than 2'-0"	#6@12 e.w.e.f.
2'-0" to 3'-0"	#8@12 e.w.e.f.
more than 3'-0"	#11@12 e.w.e.f.

8. Bars interrupted by an opening shall be replaced by bars of same size and same amount.
Size of diagonal bars shall be the same as wall reinf.

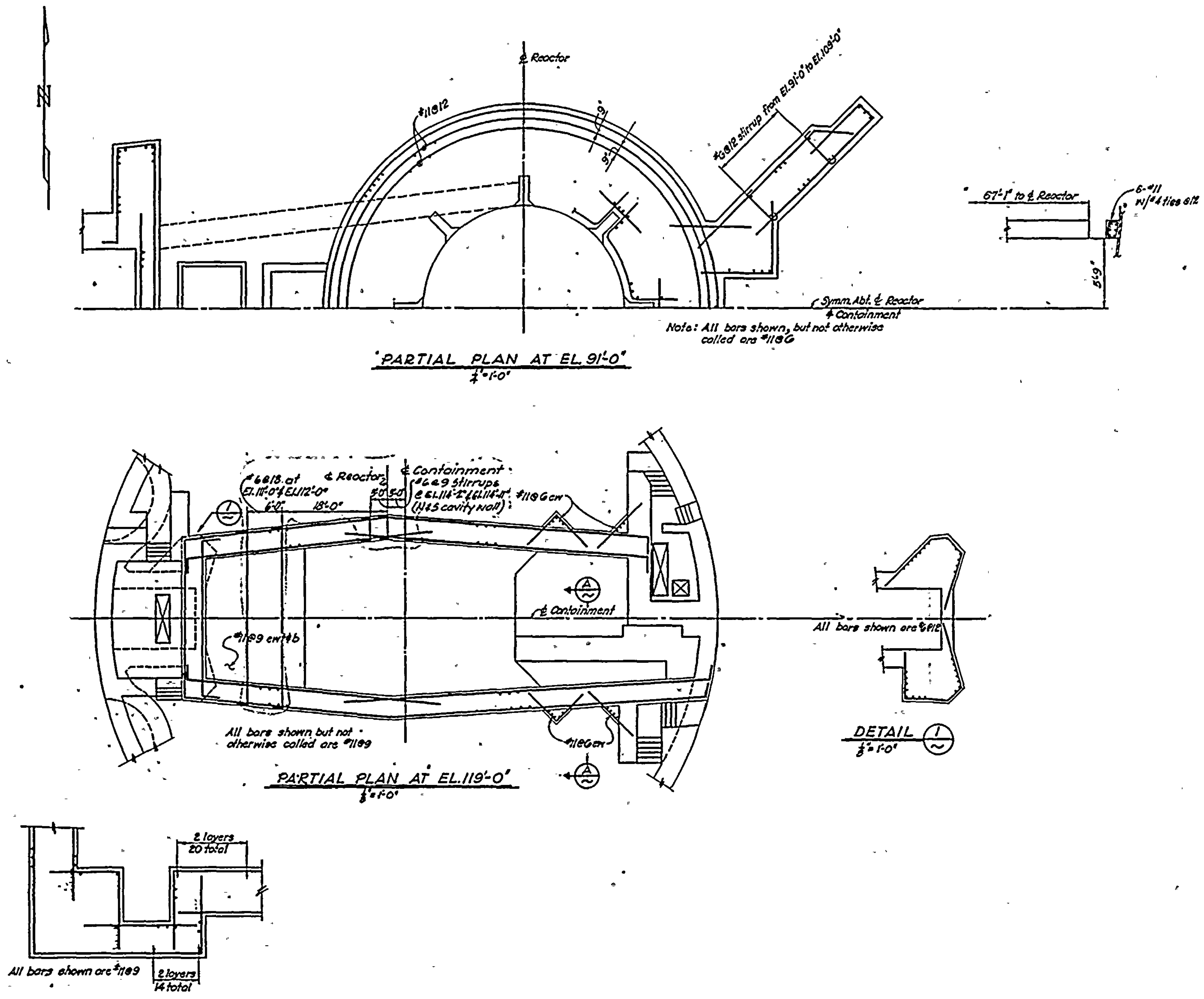
Length of additional bar = opening size + 8'-6"
" " diagonal bar for rectangular opening = 8'-6"
" " " bar for circular opening = opening size + 8'-6".

9. Additional and diagonal bars are required only for the opening larger than 12"φ.
10. Additional bars and diagonal bars are not needed when the side of an opening is formed by a wall or slab.
11. Design requirements while same as for Unit No. 1 have been presented in Unit No. 2 dwgs. in more general and concise manner in an effort to improve clarity and to allow the contractor more freedom in placement of reinforcing.
In a few cases the requirements as presented here may differ slightly from those for Unit No. 1 as a result of standardization. In all these cases the contractor may elect to build in accordance with Unit No. 1 requirements.

UNIT 2
DIABLO CANYON SITE

FIGURE 3.8-15C

INTERNAL STRUCTURE
CONCRETE DETAILS



PARTIAL PLAN AT EL 91'-0"
1/4" = 1'-0"

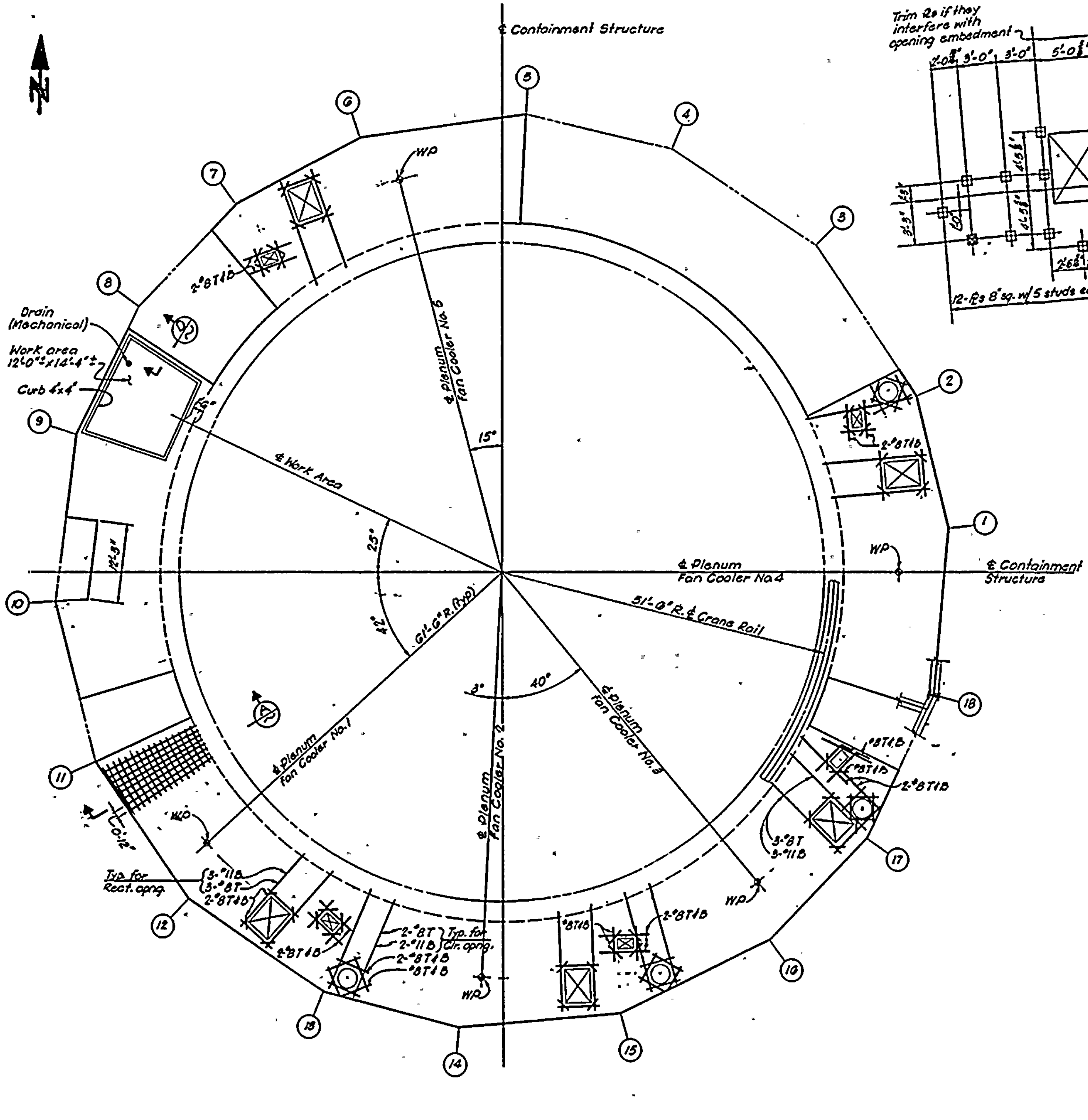
PARTIAL PLAN AT EL. 119'-0"
1/8" = 1'-0"

DETAIL 1
1/8" = 1'-0"

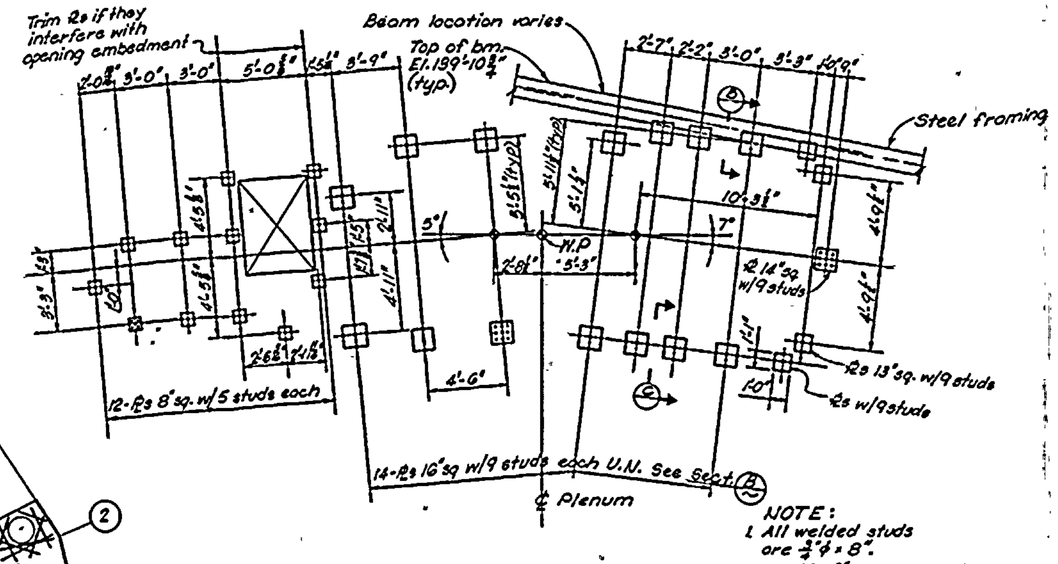
SECTION A
1/4" = 1'-0"

UNIT 2
DIABLO CANYON SITE

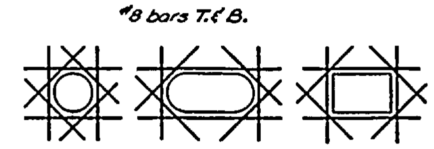
FIGURE 3.8-15D
INTERNAL STRUCTURE
CONCRETE SECTIONS AND DETAILS



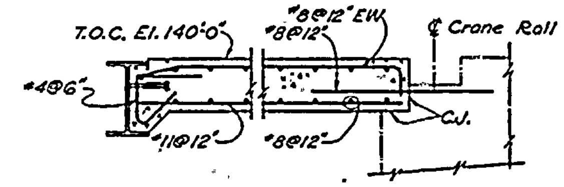
PLAN AT EL 140'-0"
1/8"=1'-0"



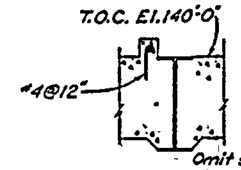
TYP. FAN COOLER MOUNTING PLATE ARRANGEMENT
1/4"=1'-0"



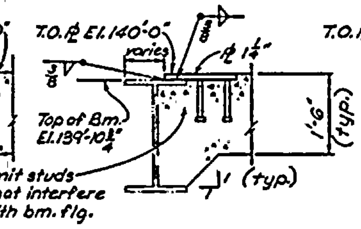
TYP. SLAB PENETRATIONS
(Except those shown on plan & noted otherwise.)



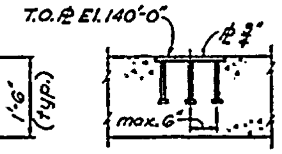
SECTION A Typ.
1/2"=1'-0"



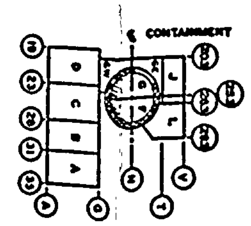
SECTION D
1/2"=1'-0"



SECTION B
1/2"=1'-0"

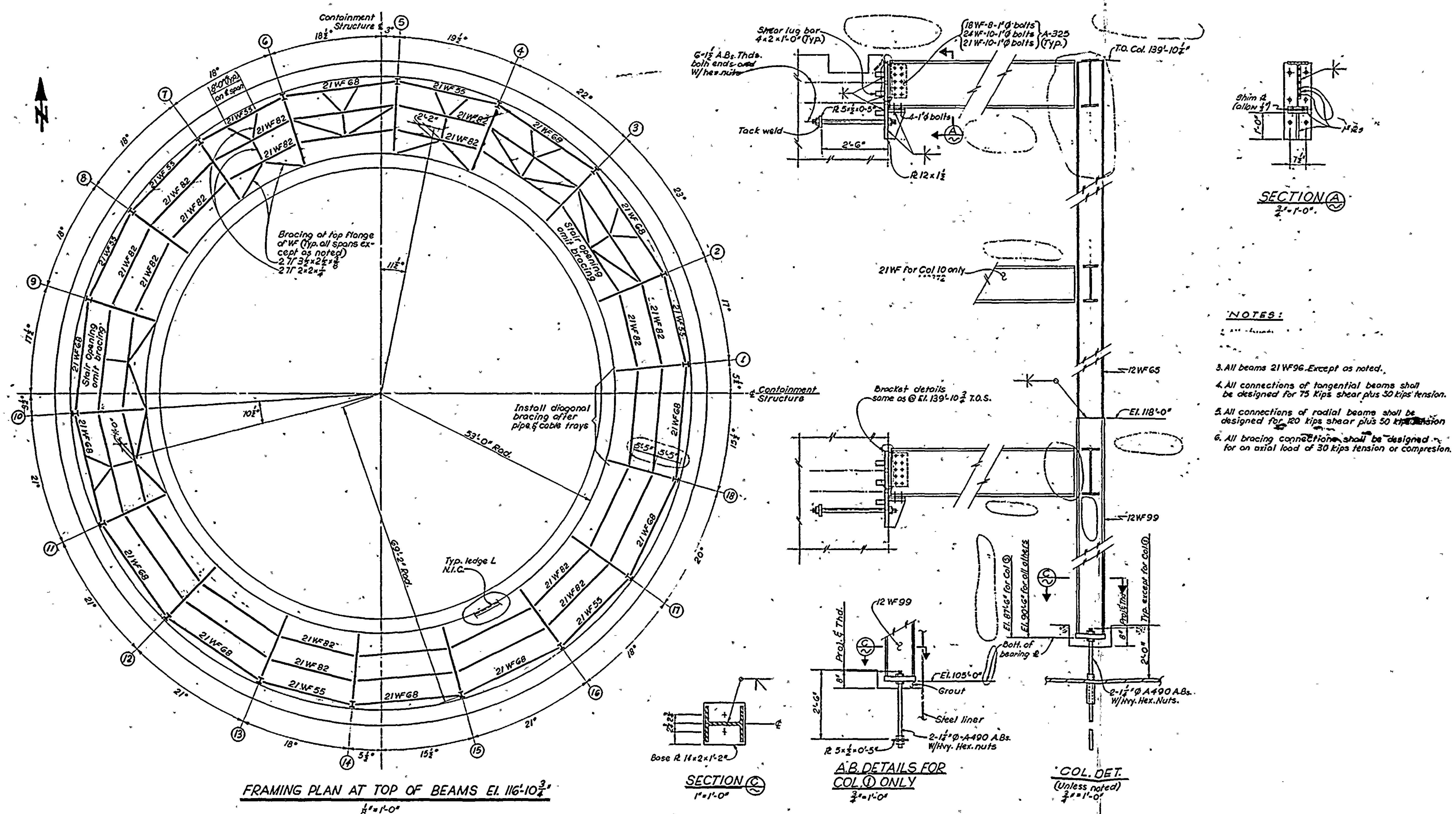


SECTION C
1/2"=1'-0"



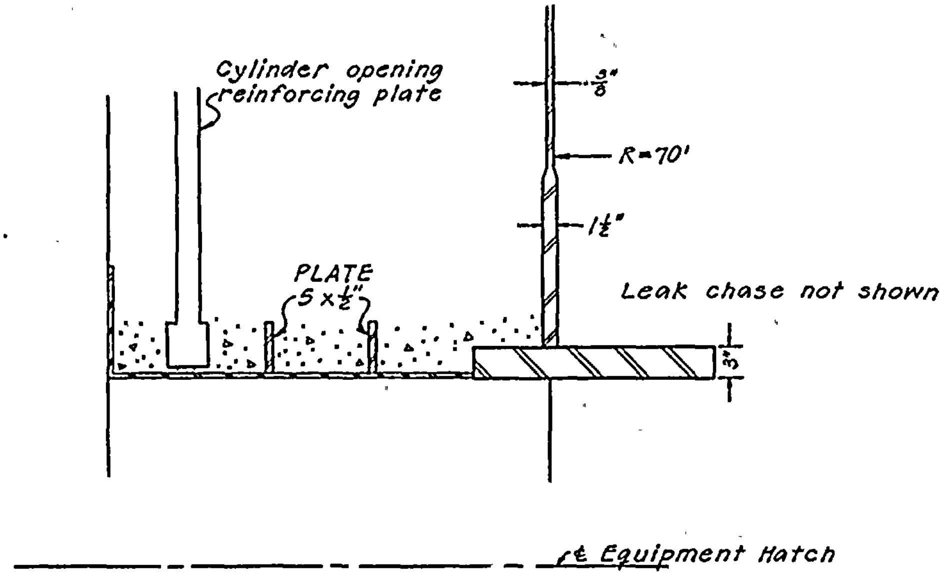
UNIT 2
DIABLO CANYON SITE

FIGURE 3.8-15E
INTERNAL STRUCTURE
ANNULUS PLATFORM
ELEVATION 140

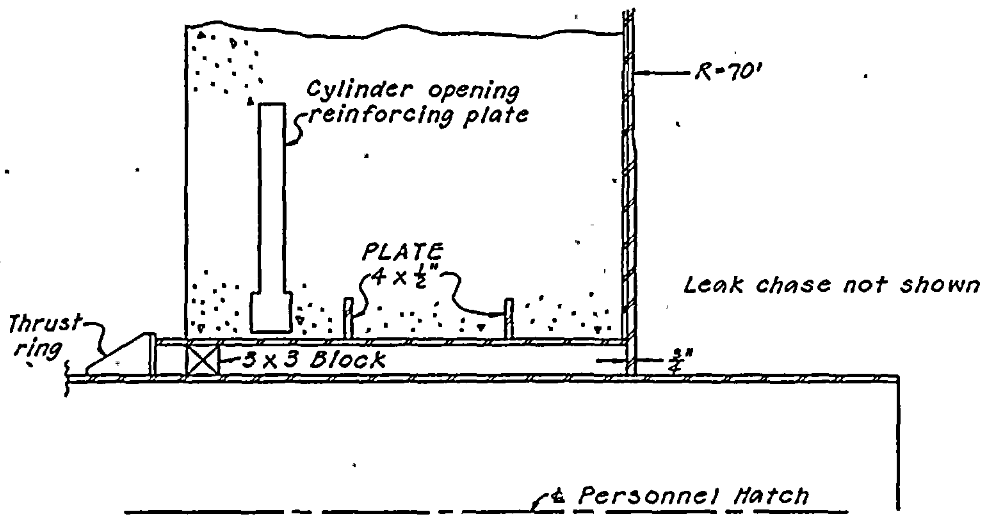


UNIT 2
DIABLO CANYON SITE

FIGURE 3.8-15F
INTERNAL STRUCTURE
ANNULUS PLATFORM
ELEVATION 117



EQUIPMENT HATCH



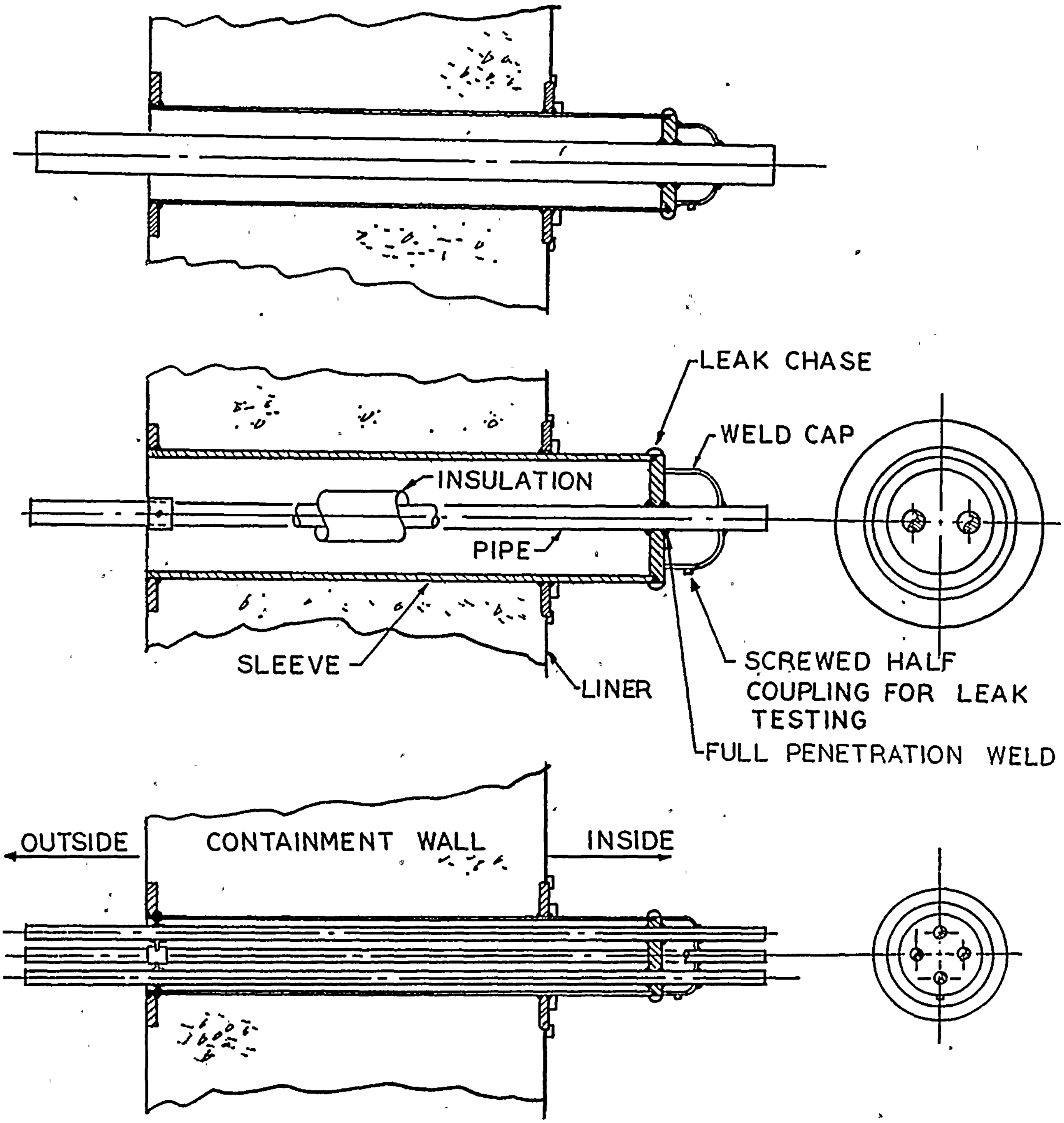
Unit I PERSONNEL HATCH

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-13

CONTAINMENT STRUCTURE

ACCESS HATCH SLEEVES

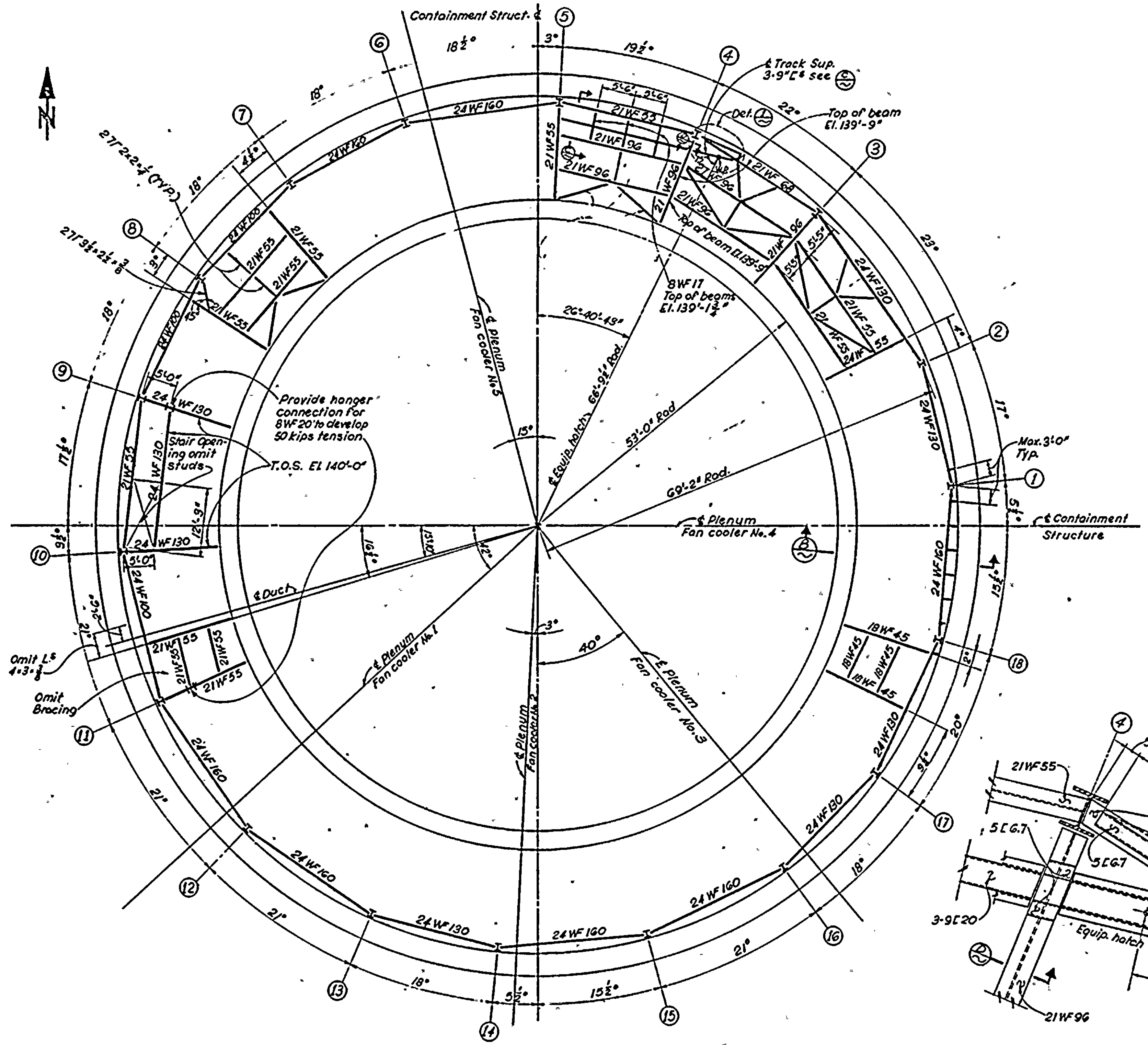


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-7

CONTAINMENT STRUCTURE

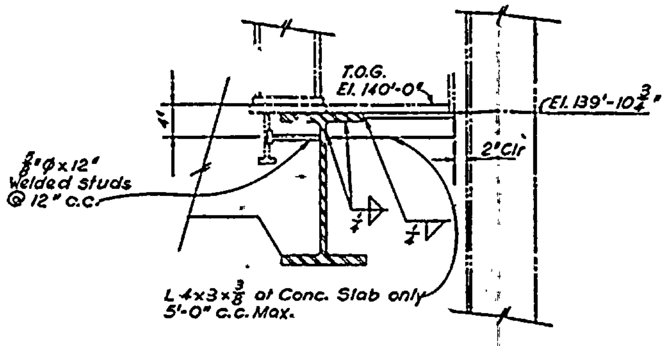
TYPICAL PIPING PENETRATION



FRAMING PLAN AT TOP OF BEAMS

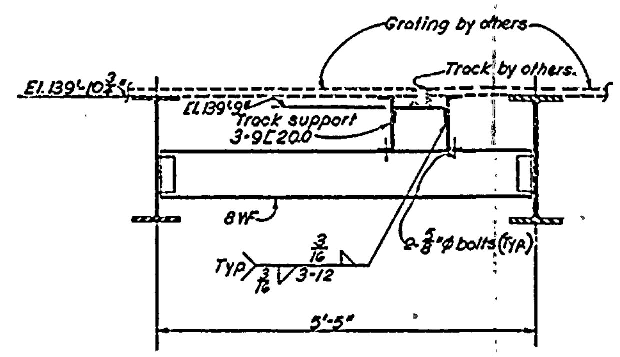
Top of Beams @ El. 139'-10 3/4" except as noted

1/8" = 1'-0"



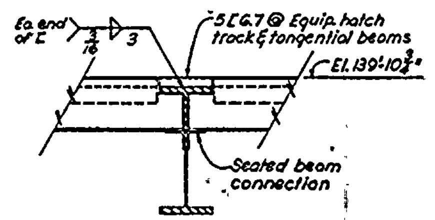
SECTION B

1" = 1'-0"



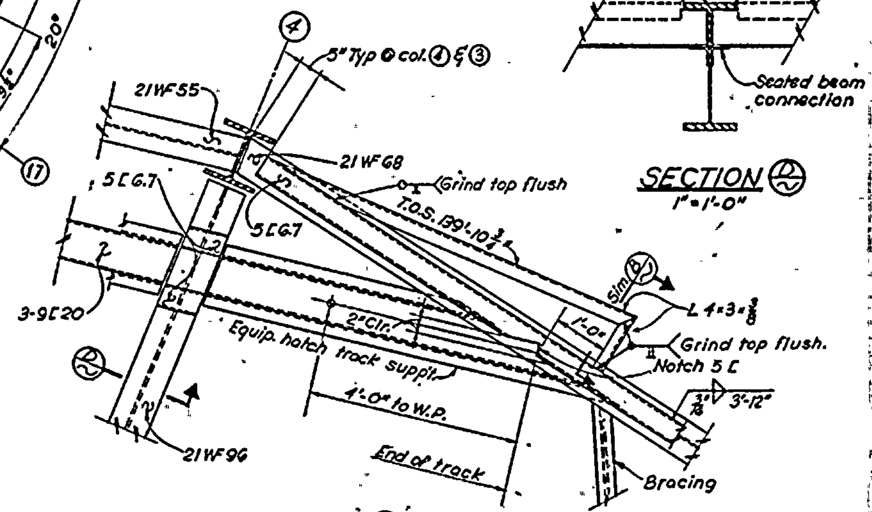
SECTION C

1" = 1'-0"



SECTION D

1" = 1'-0"



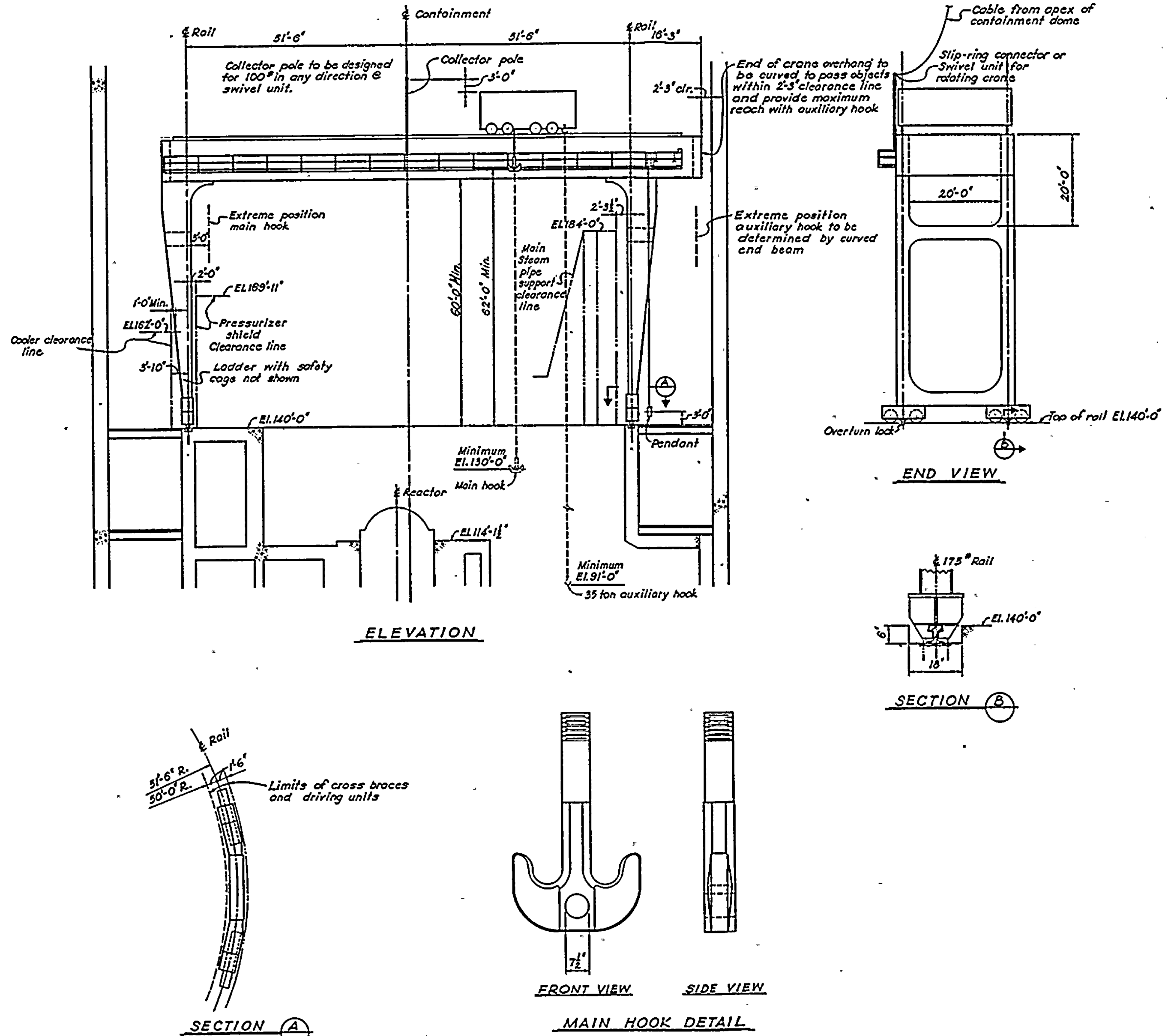
DETAIL A

3/4" = 1'-0"

NOTES:

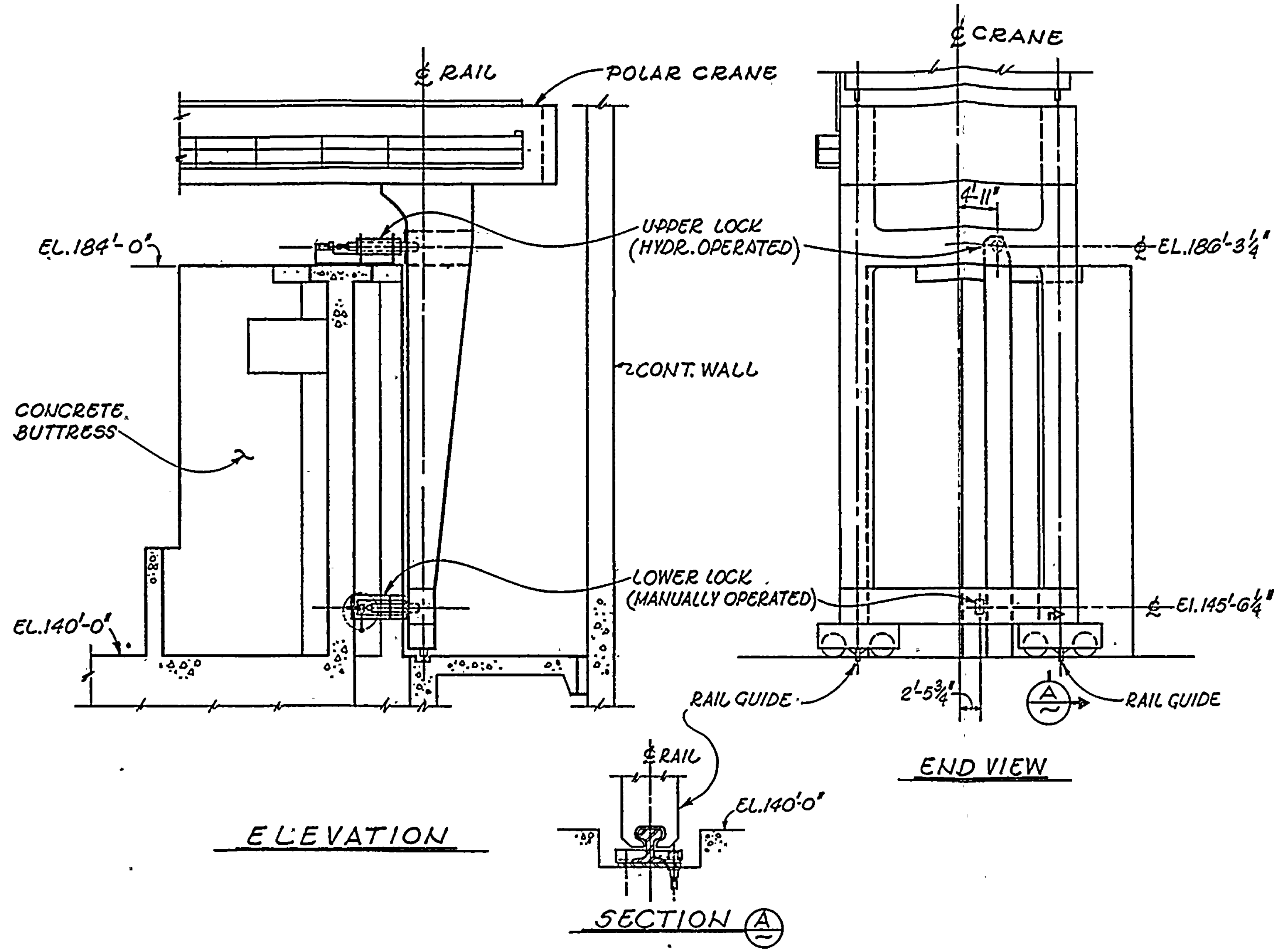
3. beam connections shall be designed for the following shear forces:
24 WF 160 - 140 Kips.
24 WF 130 - 120 Kips.
24 WF 100 - 100 Kips.
21 WF 96 - 75 Kips.
21 WF 68 & 55 - 70 Kips.
18 WF 45 - 60 Kips.

UNIT 2 DIABLO CANYON SITE
FIGURE 3.8-15G INTERNAL STRUCTURE ANNULUS FRAMING ELEVATION 140



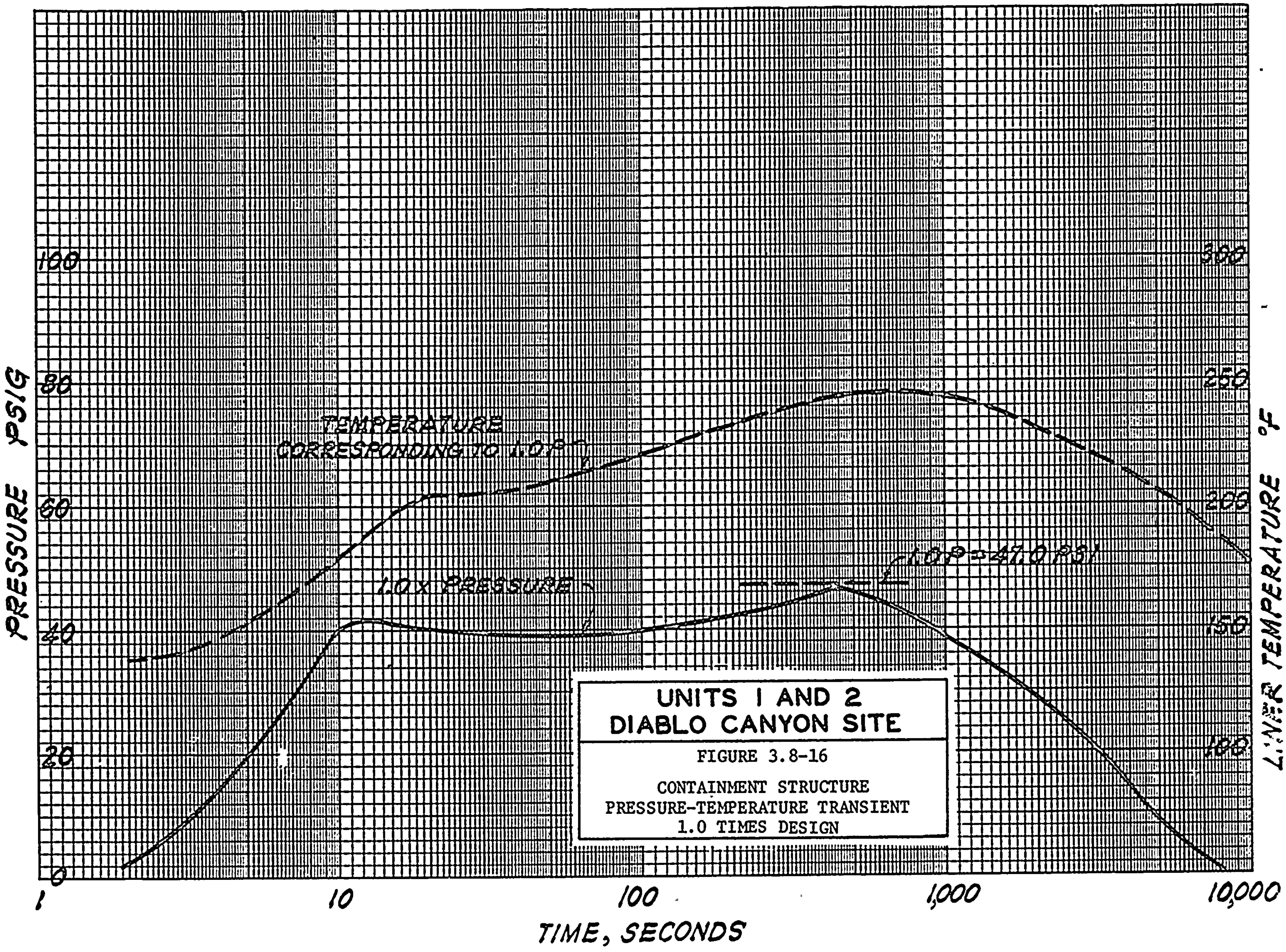
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-15H
CONTAINMENT STRUCTURE
POLAR CRANE



UNITS 1 AND 2
DIABLO CANYON SITE

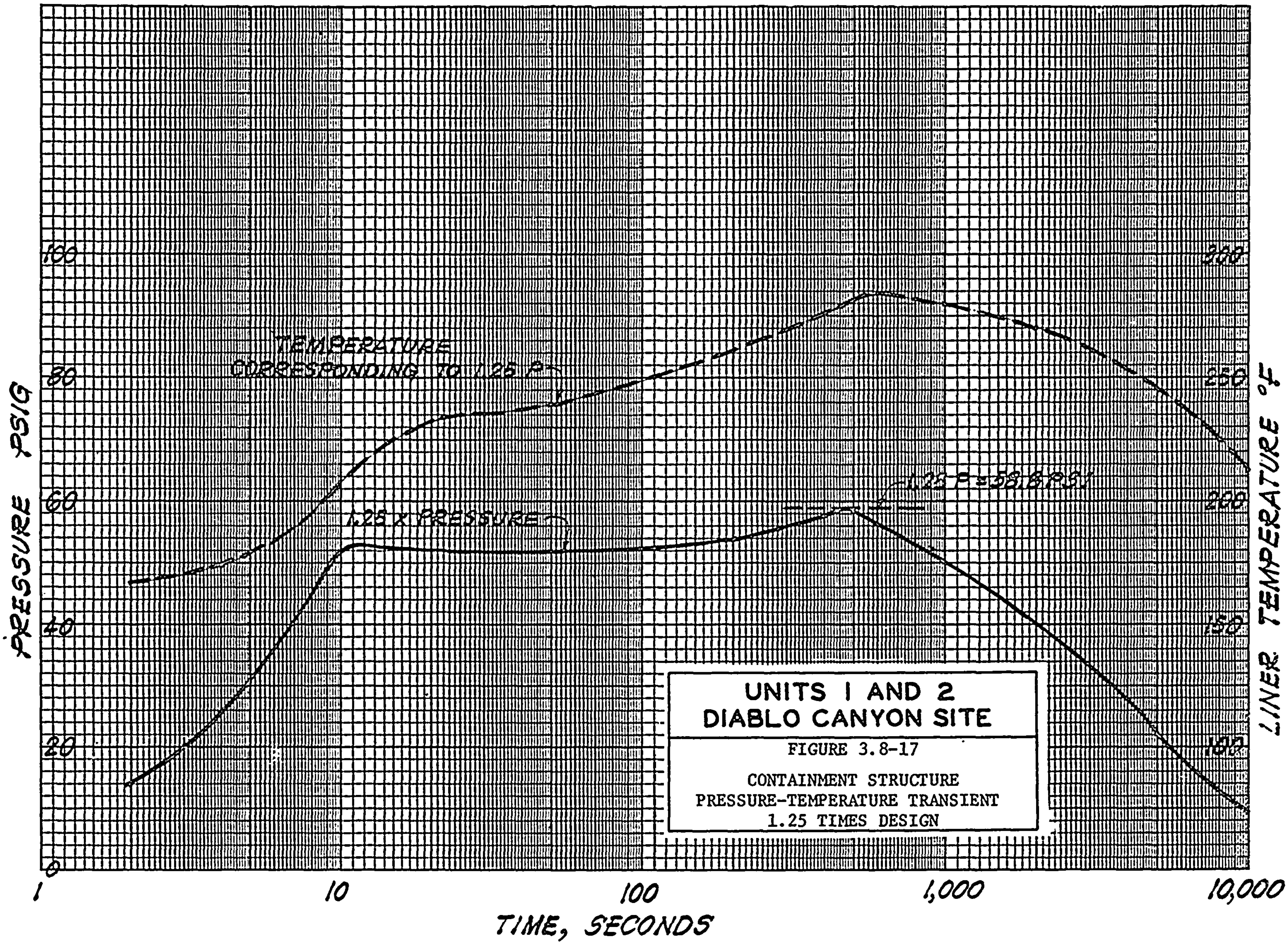
FIGURE 3.8-151
POLAR CRANE
SEISMIC LOCKS

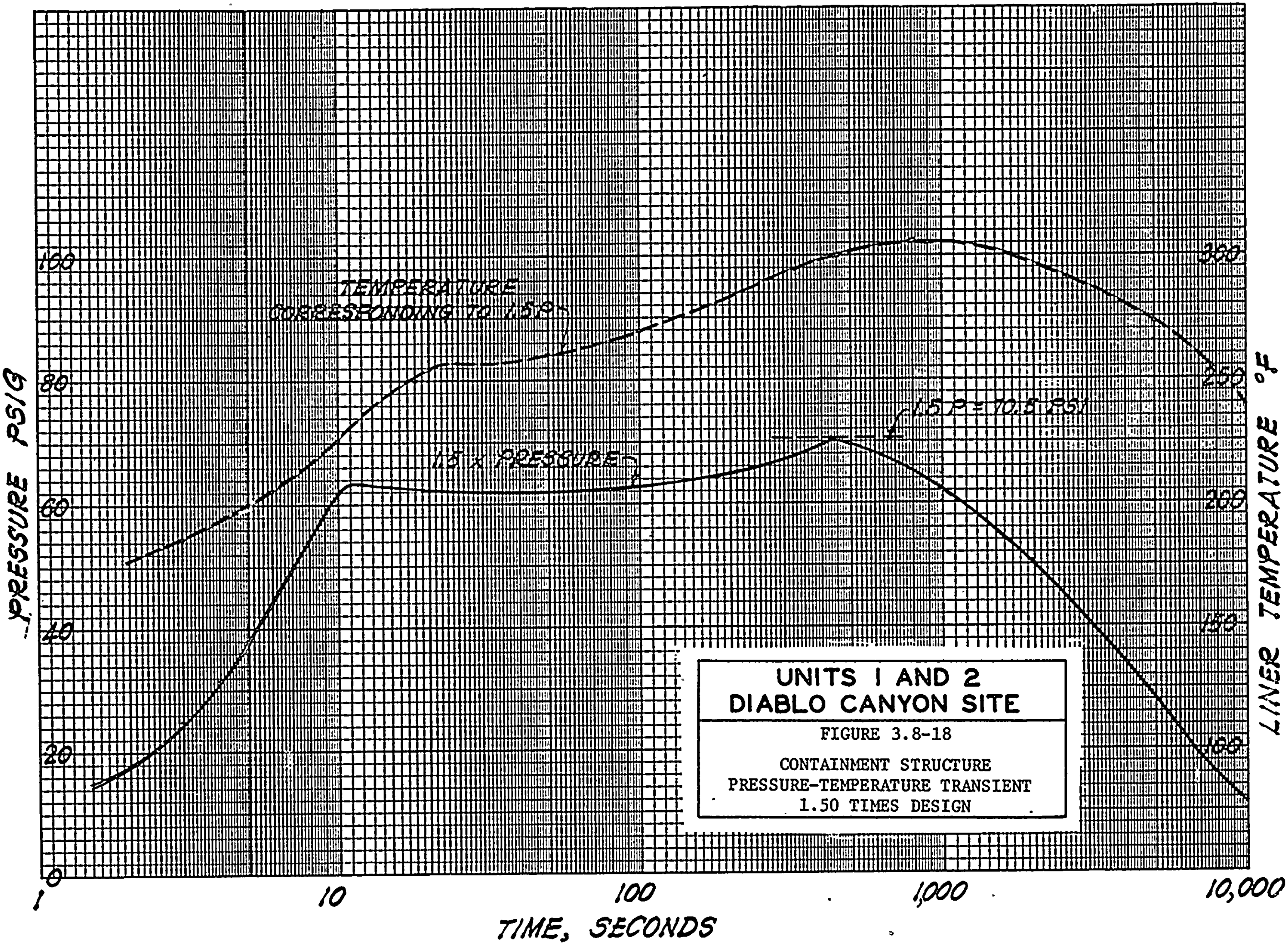


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-16

CONTAINMENT STRUCTURE
PRESSURE-TEMPERATURE TRANSIENT
1.0 TIMES DESIGN

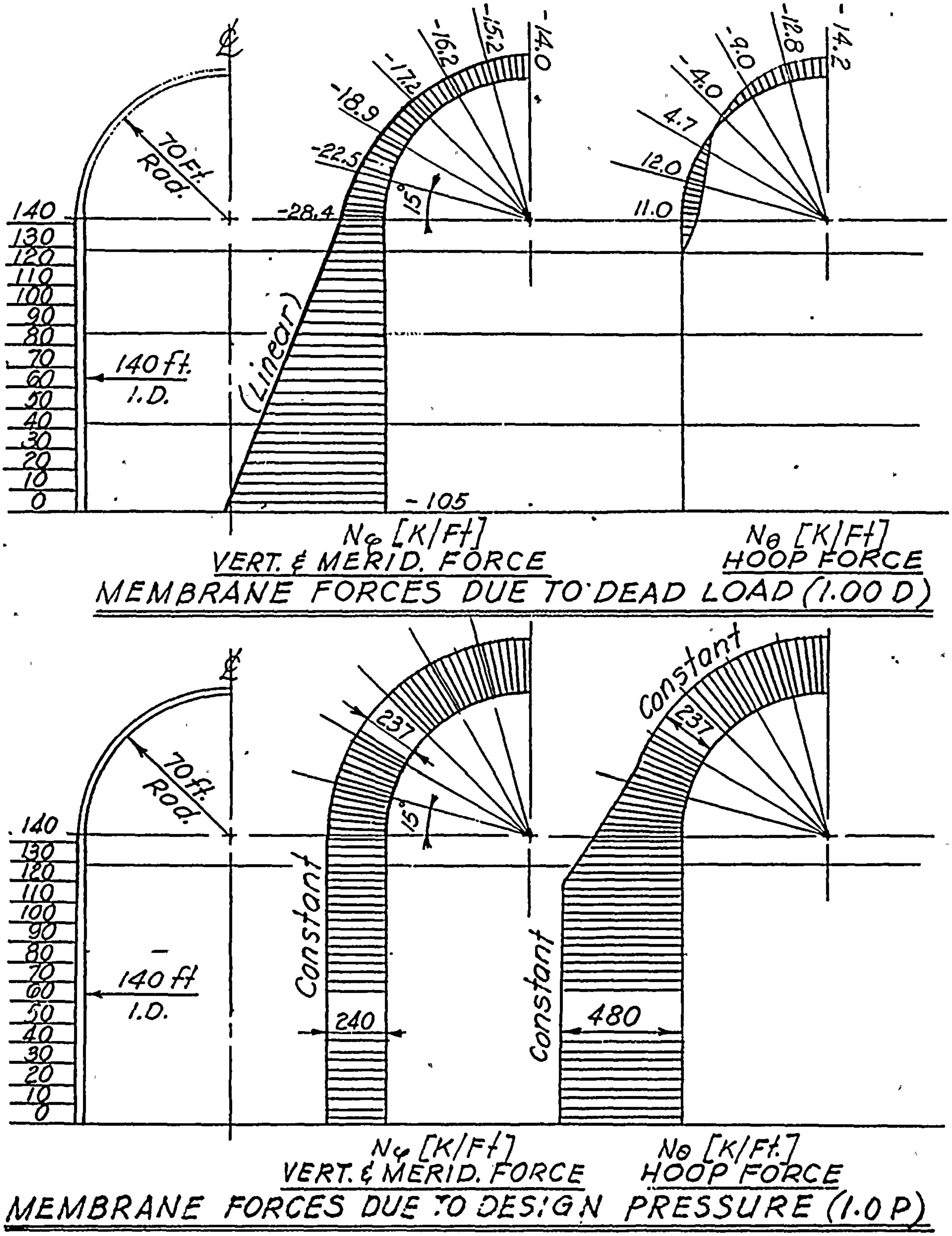

PRESSURE PSIG
LINER TEMPERATURE °F
TIME, SECONDS
1
10
100
1,000
10,000
0
20
40
60
80
100
150
200
250
300
TEMPERATURE CORRESPONDING TO 1.25 P
1.25 X PRESSURE
1.25 P = 58.8 PSI
UNITS 1 AND 2
DIABLO CANYON SITE
FIGURE 3.8-17
CONTAINMENT STRUCTURE
PRESSURE-TEMPERATURE TRANSIENT
1.25 TIMES DESIGN



UNITS 1 AND 2
DIABLO CANYON SITE

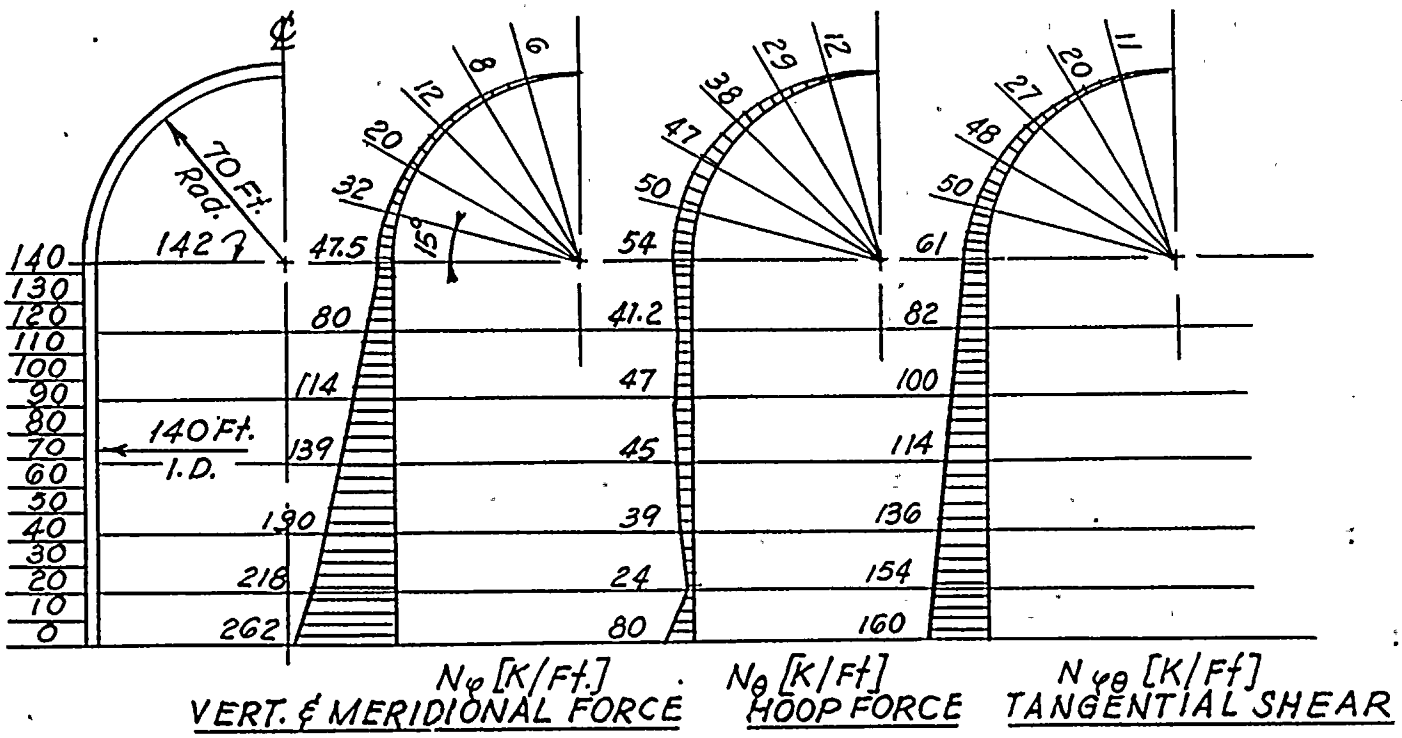
FIGURE 3.8-18

CONTAINMENT STRUCTURE
PRESSURE-TEMPERATURE TRANSIENT
1.50 TIMES DESIGN

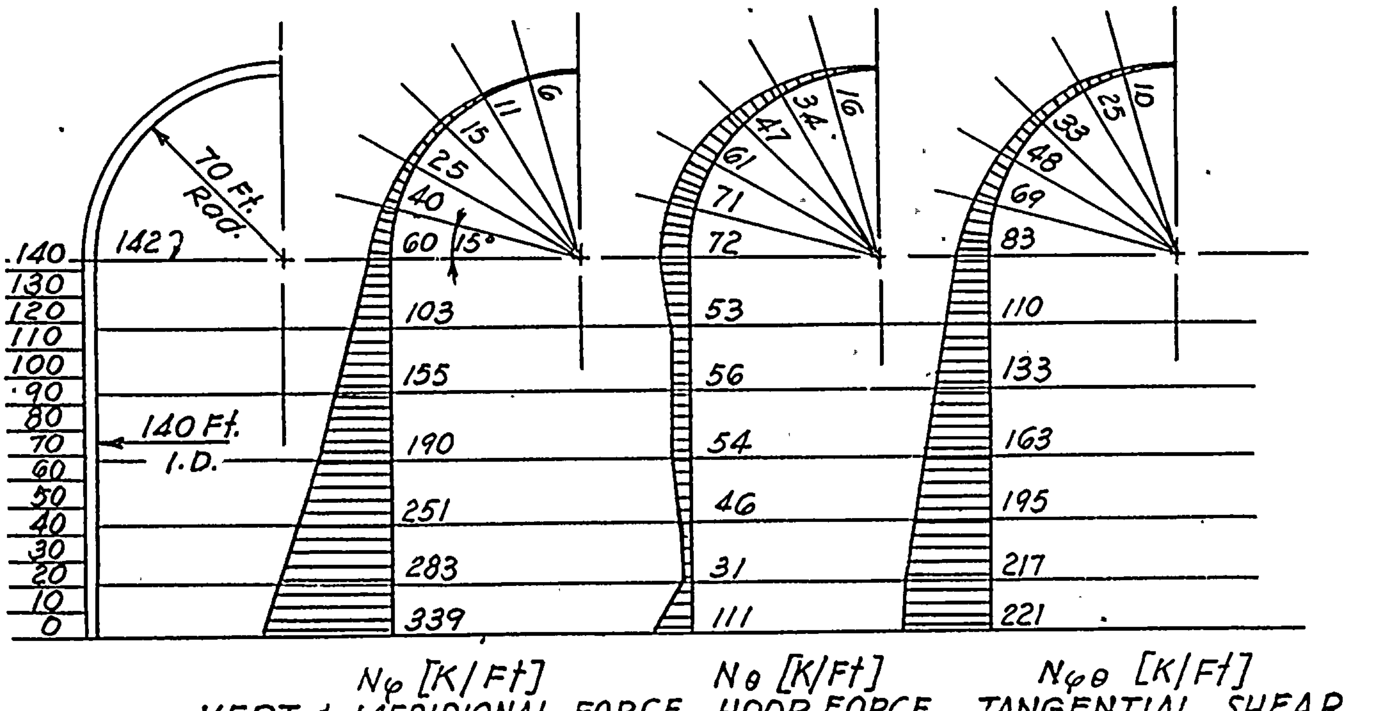


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-19
CONTAINMENT STRUCTURE
MEMBRANE FORCES
DEAD LOAD AND PRESSURE



MEMBRANE FORCES DUE TO 1.25x DESIGN EARTHQUAKE (1.25 DE)

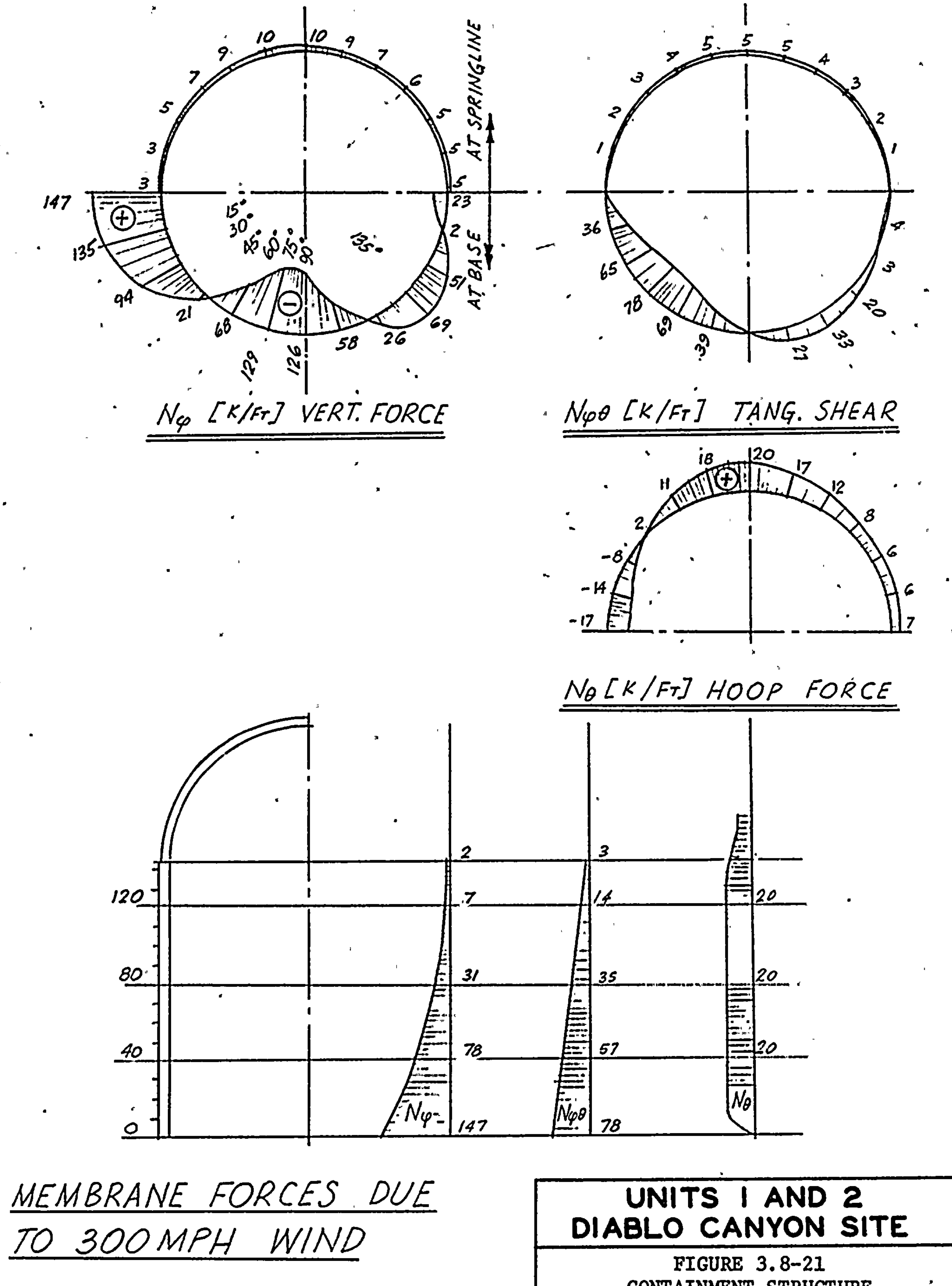


MEMBRANE FORCES DUE TO DOUBLE DESIGN EARTHQUAKE (DDE)

UNITS 1 AND 2
DIABLO CANYON SITE

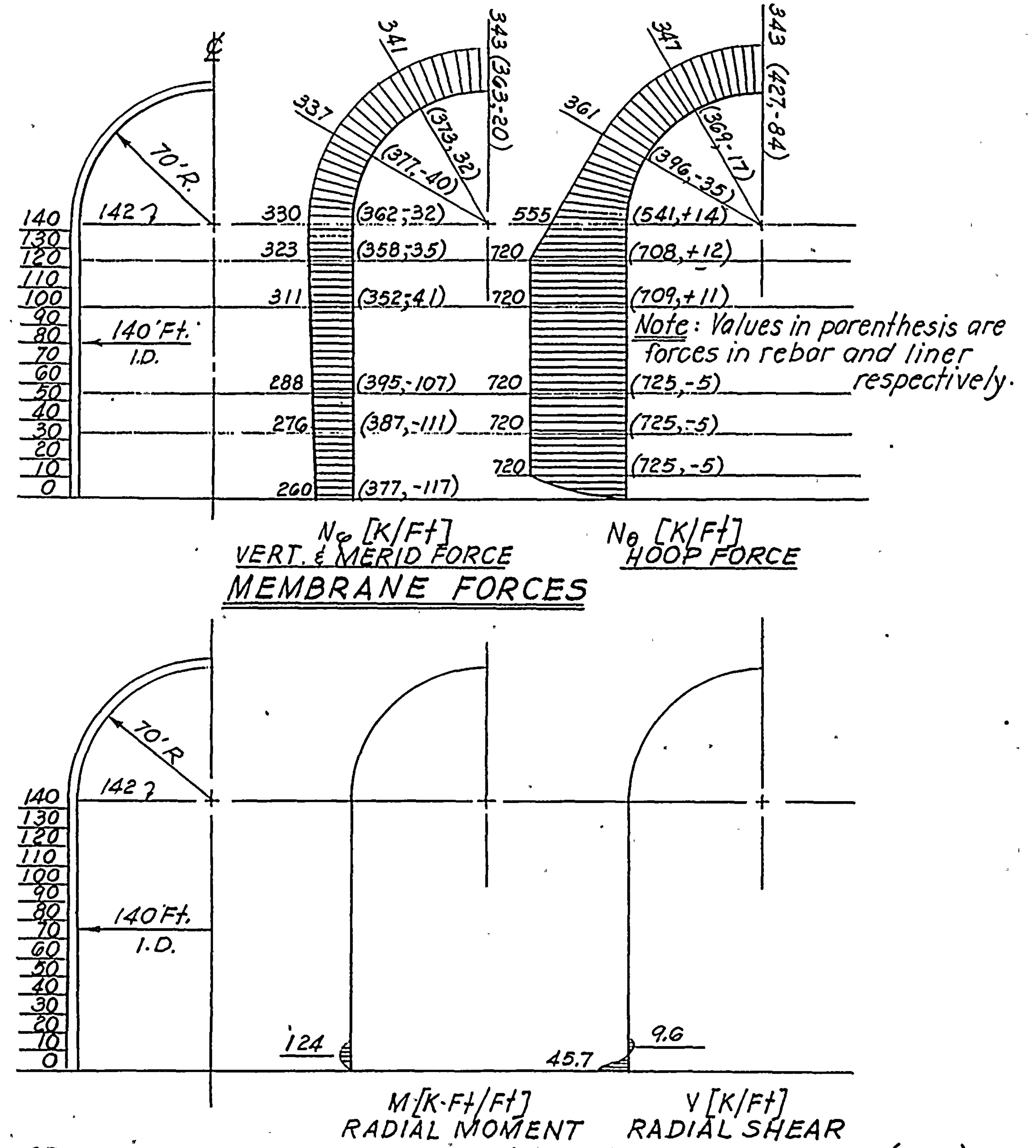
FIGURE 3.8-20
CONTAINMENT STRUCTURE
MEMBRANE FORCES
EARTHQUAKE

Amendment 2 January 1974



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-21
CONTAINMENT STRUCTURE
MEMBRANE FORCES
WIND LOAD

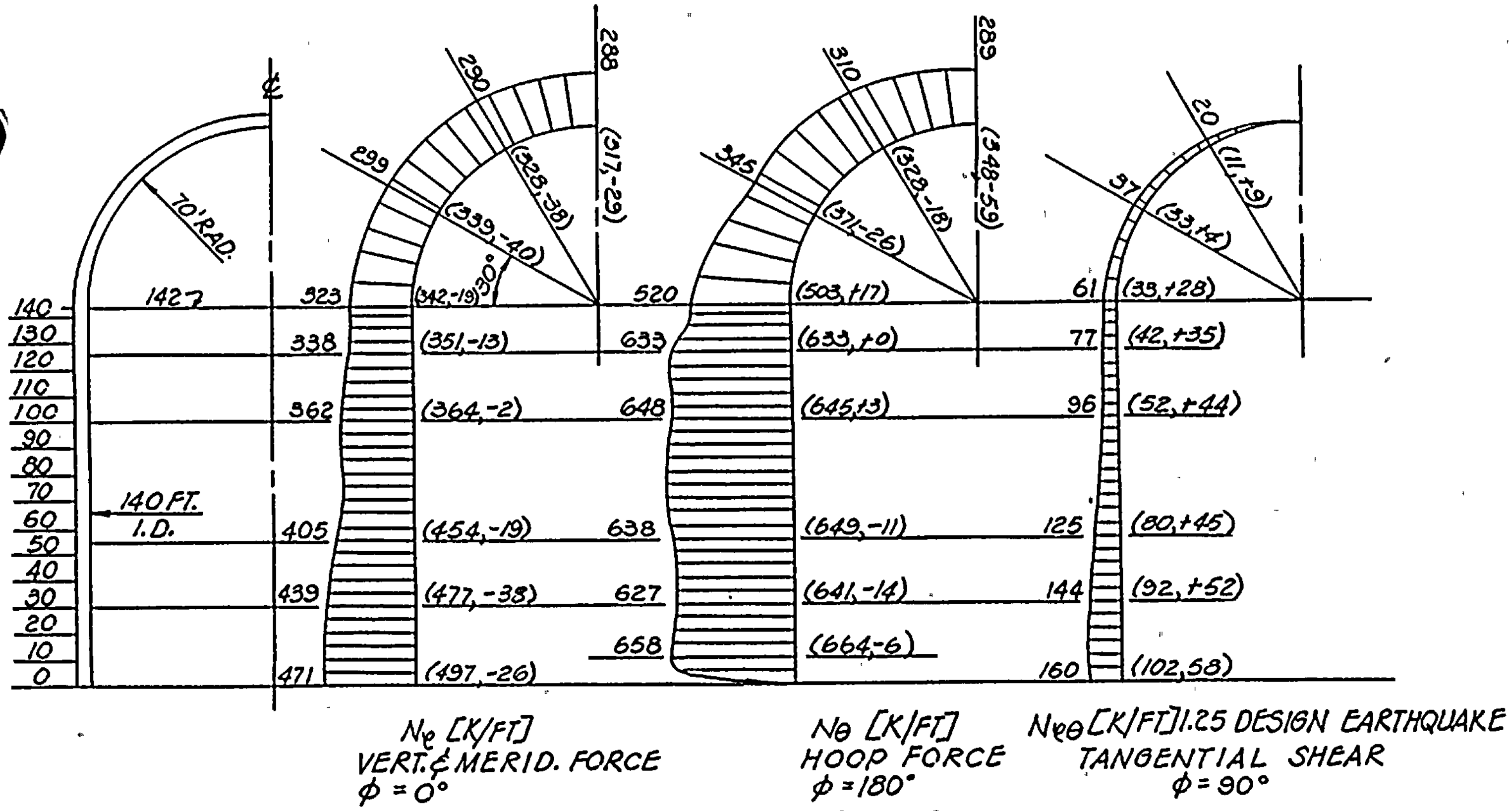


FORCES DUE TO LOAD CONDITION (a): C = 1.0D ± .05D + 1.5P + 1.0 (T)

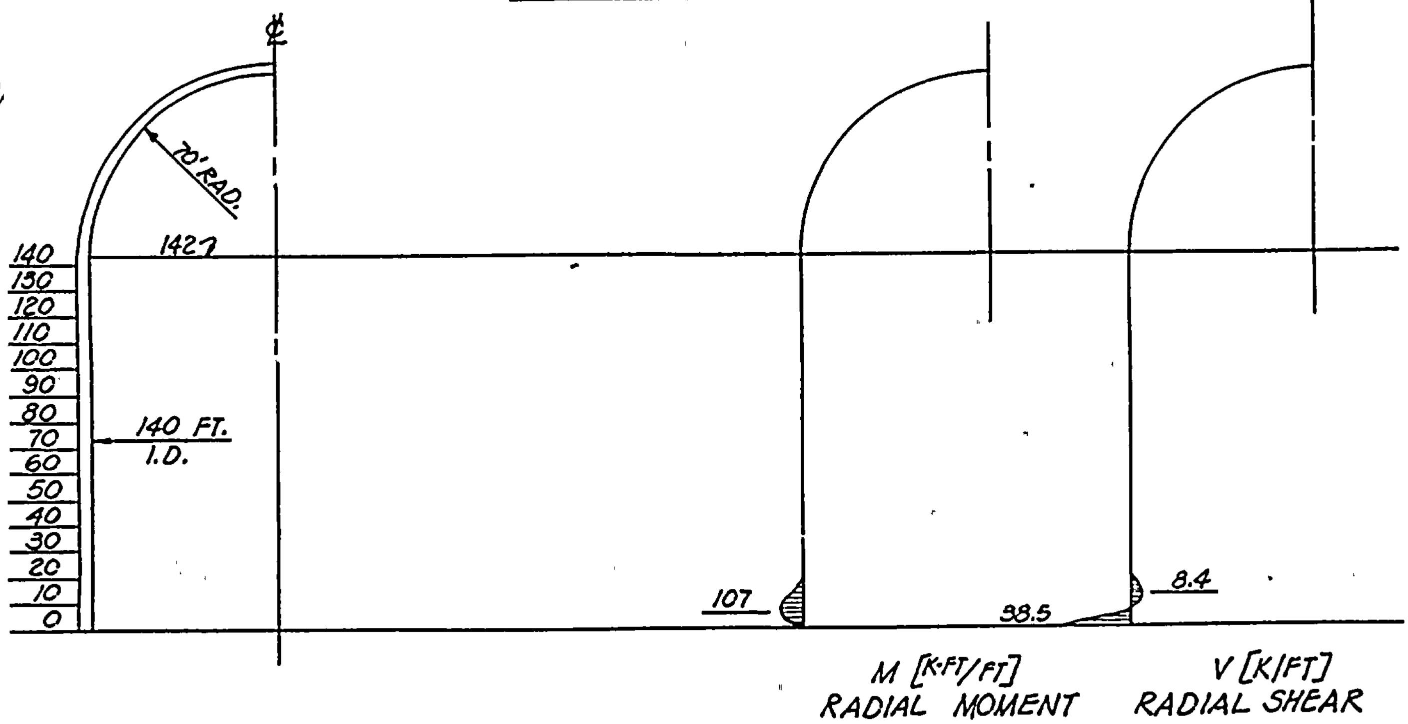
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-23
CONTAINMENT STRUCTURE
MEMBRANE FORCES
ACCIDENT CONDITION

Amendment 2 January 1974



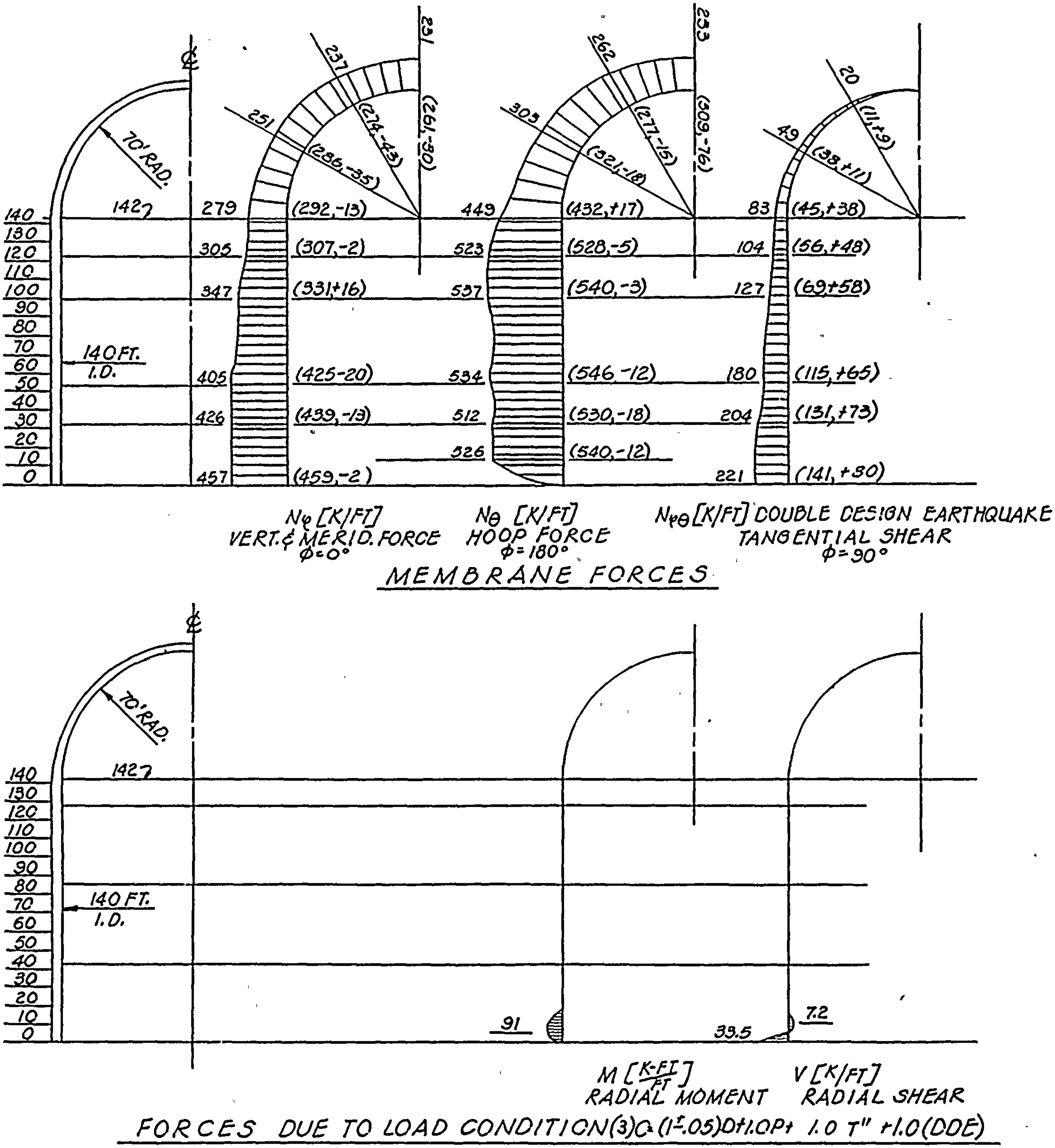
MEMBRANE FORCES



FORCES DUE TO LOAD CONDITION (2): C=1.00±.05D+1.25P+1.0T'+1.25(DE)

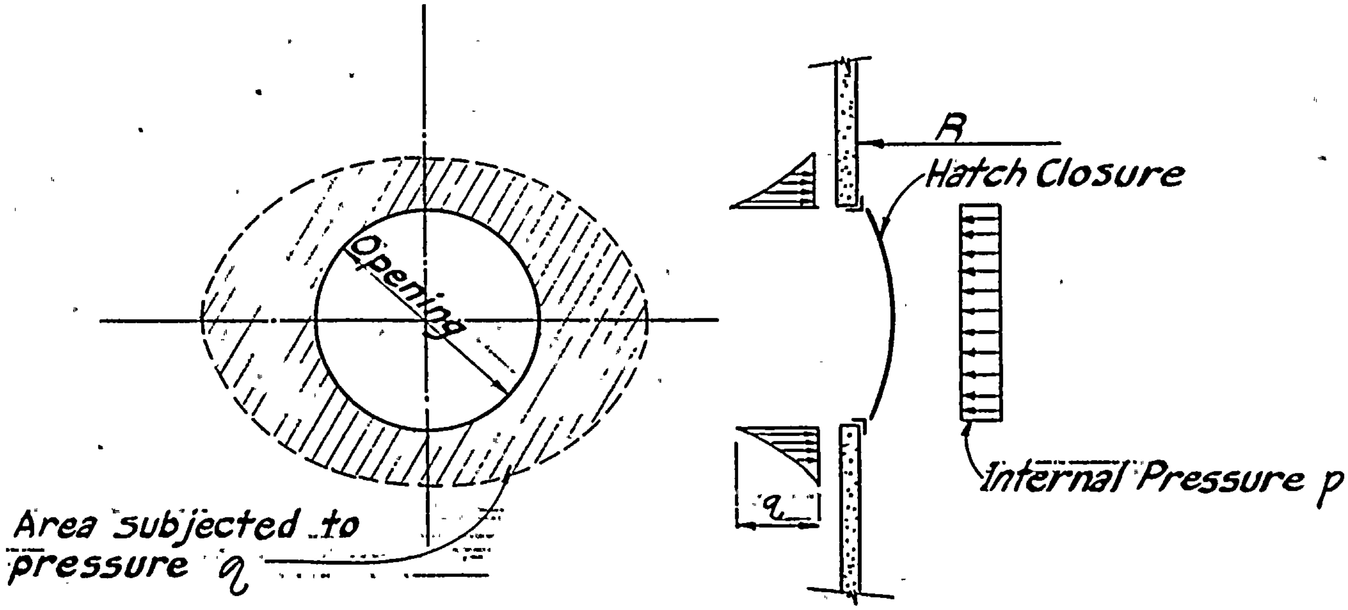
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-24
CONTAINMENT STRUCTURE
MEMBRANE FORCES
ACCIDENT CONDITION



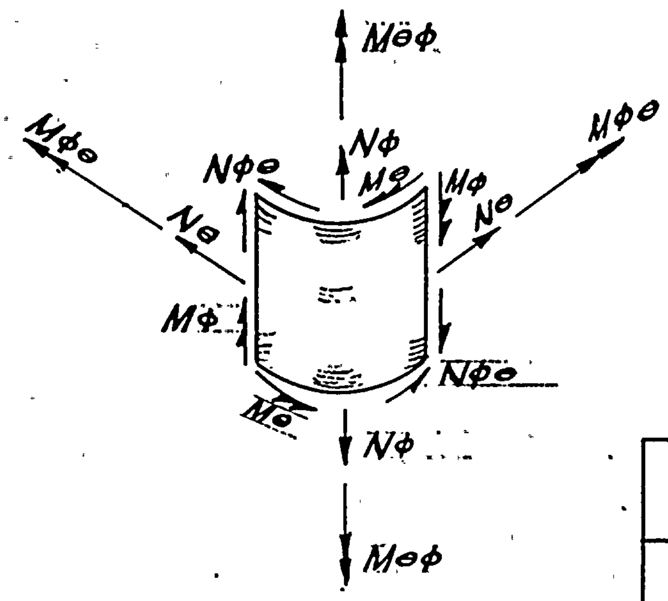
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-25
CONTAINMENT STRUCTURE
MEMBRANE FORCES
ACCIDENT CONDITION



$$q = -\frac{N_\theta}{R} + P$$

STATICAL MODEL FOR CALCULATING BENDING MOMENT AROUND LARGE PENETRATIONS

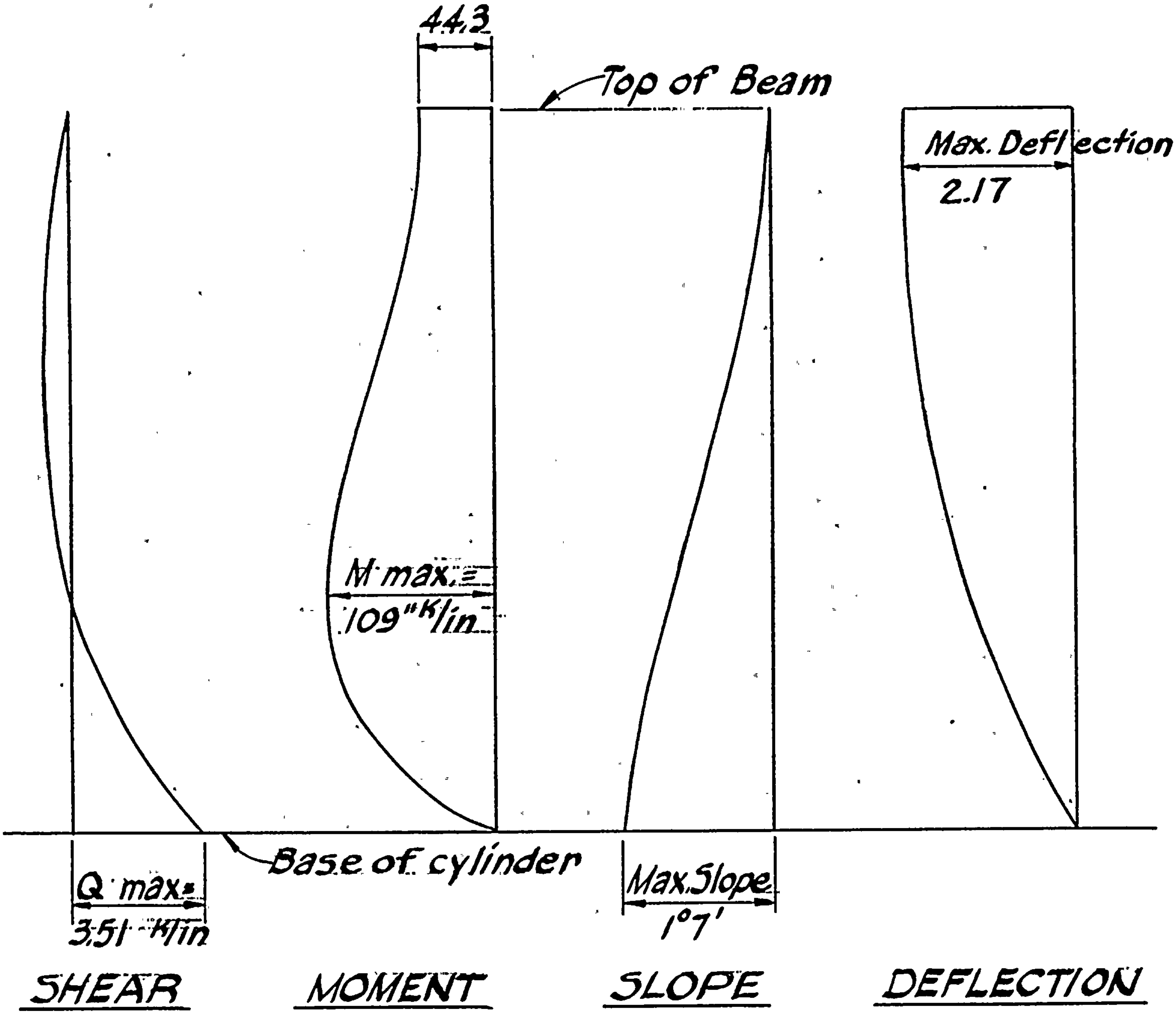


UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-26

CONTAINMENT STRUCTURE

ANALYSIS OF LARGE PENETRATIONS

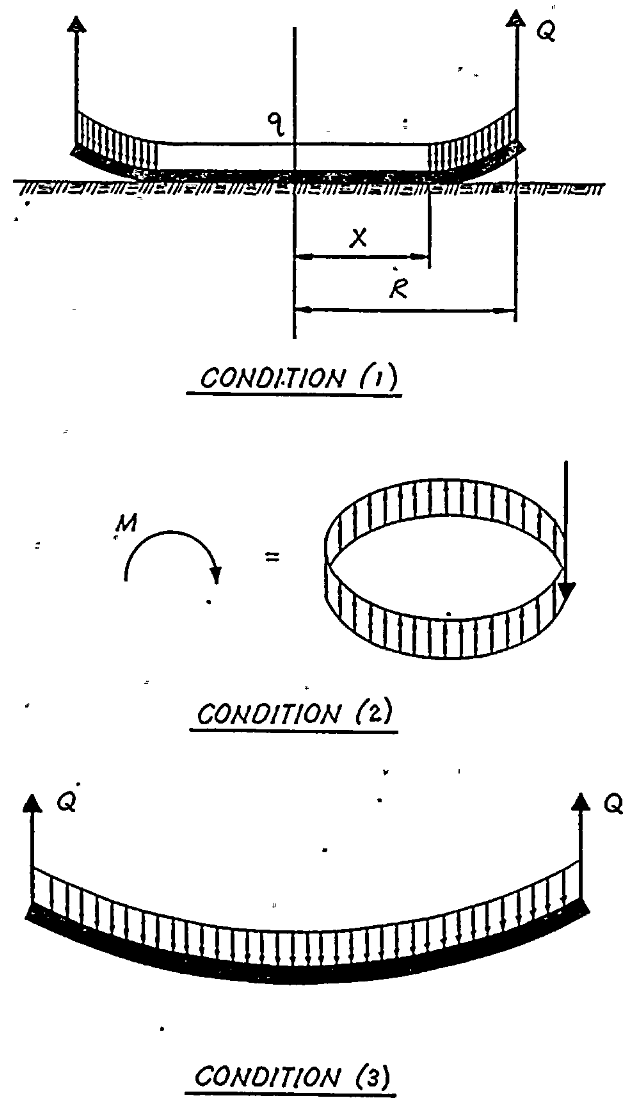


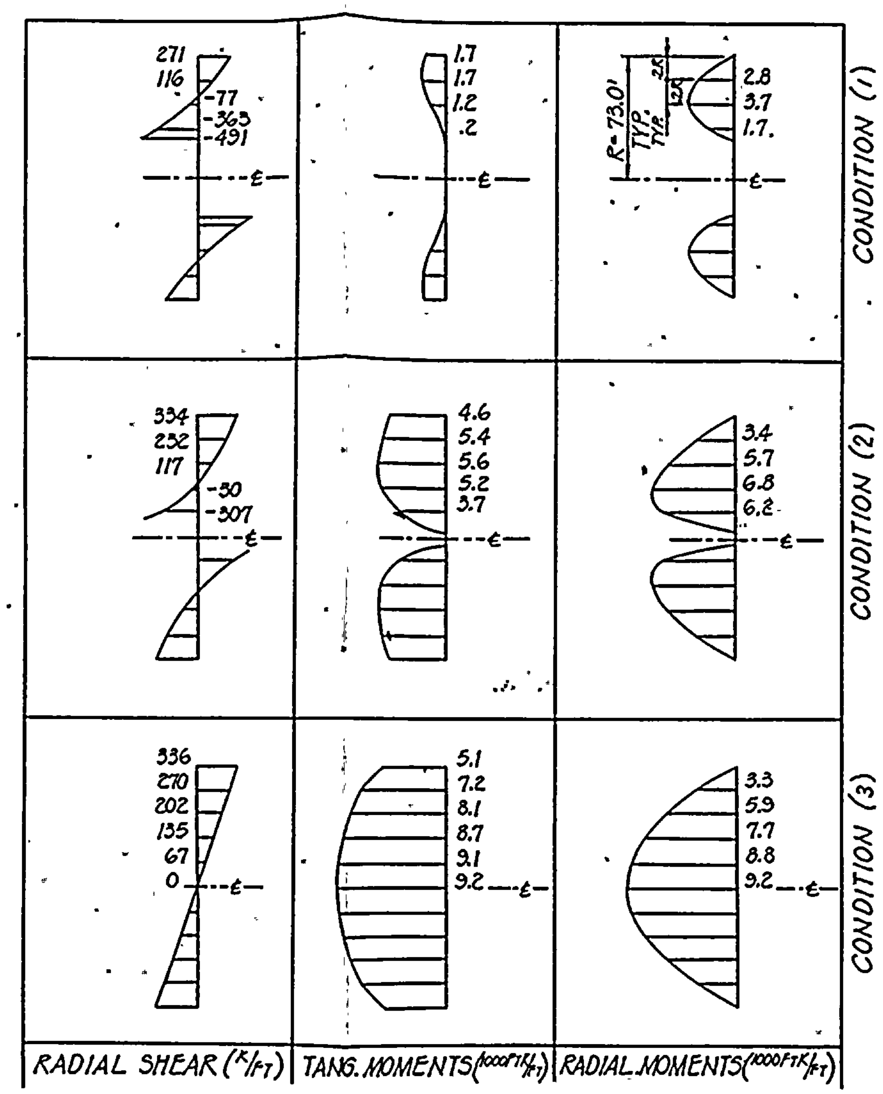
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-27

CONTAINMENT STRUCTURE

EMBEDDED BEAM ANALYSIS





UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28

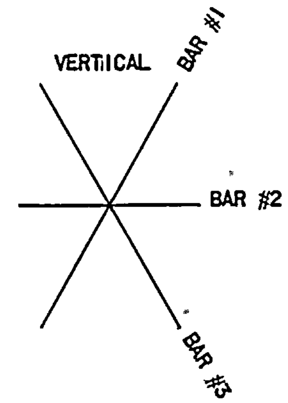
CONTAINMENT STRUCTURE

BASE SLAB ANALYSIS

CONDITION (a) C = 1.0D ± 0.05D + 1.5P + 1.0 T"

CRACKED SECTION

		LINER NOT CONSIDERED			LINER CONSIDERED					
		REBAR STRESSES			REBAR STRESSES			LINER STRESS		
		1	2	3	1	2	3	σ_y	σ_x	τ_{xy}
ALLOWABLE STRESS		57.0	57.0	57.0	57.0	57.0	57.0			
ELEVATION (ABOVE BASE SLAB)										
DOME	213	39.4	20.6	39.4	41.7	30.0	41.7	-4.5	-18.7	0.0
DOME	211	38.0	30.6	41.1	41.5	37.0	43.7	-8.5	-9.4	3.2
DOME	204	36.8	42.5	42.4	41.4	44.5	44.4	-7.2	-3.7	0.0
DOME	193	38.5	42.2	36.3	42.2	44.3	40.8	-7.6	-6.1	1.4
DOME	178	39.4	41.3	30.9	42.7	43.4	37.3	-9.1	-7.8	3.2
DOME	160	38.6	34.3	38.6	42.1	38.9	42.1	-6.9	-10.5	0.0
	142	38.9	55.7	38.9	42.8	52.6	42.8	-7.5	3.1	0.0
CYLINDER	132	38.6	46.8	38.6	42.4	46.9	42.4	-7.5	-2.6	0.0
CYLINDER	123	38.1	54.2	38.1	42.3	52.1	42.3	-8.2	2.6	0.0
CYLINDER	100	36.7	54.5	36.7	41.5	52.3	41.5	-9.5	2.5	0.0
CYLINDER	77	35.4	54.9	35.4	40.7	52.6	40.7	-10.6	2.4	0.0
CYLINDER	54	22.7	55.2	22.7	31.1	52.5	31.1	-24.4	-1.2	0.0
CYLINDER	32	21.7	55.6	21.7	30.5	52.8	30.5	-25.4	-1.2	0.0
CYLINDER	12	21.1	55.8	21.1	30.1	52.9	30.1	-26.1	-1.2	0.0



ALL STRESSES IN KIPS/SQUARE INCH

MINUS SIGN INDICATES COMPRESSION

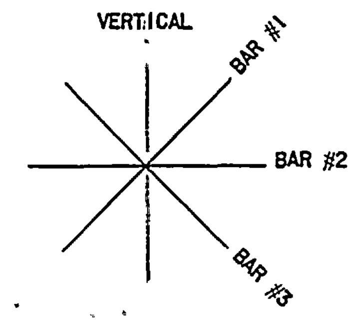
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28A
CONTAINMENT STRUCTURE
EXTERIOR SHELL STRESSES
ACCIDENT CONDITION 1

CONDITION (b) $C = 1.0D \pm 0.05D + 1.25P + 1.0\ T' + 1.25DE$

CRACKED SECTION - LINER CONSIDERED

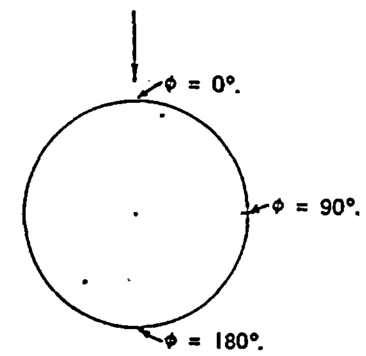
	ELEVATION (ABOVE BASE SLAB)	φ = 0° REBAR STRESSES			φ = 0° LINER STRESSES			φ = 90° REBAR STRESSES			φ = 90° LINER STRESSES			φ = 180° REBAR STRESSES			φ = 180° LINER STRESSES		
		1	2	3	σ_y	σ_x	τ_{xy}	1	2	3	σ_y	σ_x	τ_{xy}	1	2	3	σ_y	σ_x	τ_{xy}
	ALLOWABLE STRESS	57.0	57.0	57.0				57.0	57.0	57.0				57.0	57.0	57.0			
DOME	213	36.4	25.5	36.4	-7.0	-20.1	0.0	35.8	26.0	36.8	-6.2	-27.4	0.4	36.1	26.5	36.1	-7.8	-19.3	0.0
DOME	211	36.8	31.6	37.3	-10.5	-12.6	2.7	36.3	31.5	38.3	-10.2	-21.0	3.2	35.8	32.4	38.3	-11.2	-11.2	2.7
DOME	204	37.5	37.1	37.0	-9.0	-9.2	0.0	36.0	36.7	40.8	-9.7	-6.5	1.8	34.6	40.1	40.1	-10.6	-4.3	0.0
DOME	193	38.6	35.4	34.2	-9.1	-11.9	-1.3	35.5	36.3	38.7	-10.4	-8.6	1.5	34.9	41.4	36.7	-11.3	-5.3	1.2
DOME	178	39.5	33.3	31.8	-9.6	-14.0	-2.6	34.9	35.7	36.8	-11.6	-10.4	0.9	35.0	42.1	33.1	-13.3	-5.9	3.0
DOME	160	38.6	29.1	38.6	-6.2	-16.3	0.0	31.6	33.8	41.8	-9.7	-12.5	4.4	35.0	39.0	35.0	-13.1	-8.2	0.0
	142	40.4	39.9	40.4	-5.5	-5.1	0.0	30.5	45.4	44.0	-10.3	-0.9	6.2	34.2	50.9	34.2	-15.2	3.3	0.0
CYLINDER	132	40.7	33.0	40.7	-4.6	-11.6	0.0	29.3	40.1	44.8	-10.5	-6.1	7.1	33.1	46.1	33.1	-16.5	-1.6	0.0
CYLINDER	123	41.4	40.1	41.4	-4.5	-4.2	0.0	28.5	44.5	45.5	-11.1	-1.6	7.8	32.3	47.9	32.3	-18.1	0.0	0.0
CYLINDER	100	43.0	39.9	43.0	-2.6	-3.9	0.0	25.6	44.7	46.9	-12.6	-1.7	9.8	29.5	49.7	29.5	-22.7	0.6	0.0
CYLINDER	77	44.2	39.0	44.2	-1.4	-4.4	0.0	23.7	44.8	47.7	-13.9	-1.8	11.0	26.9	50.3	26.9	-26.9	0.3	0.0
CYLINDER	54	35.7	37.4	35.7	-13.2	-8.9	0.0	16.4	44.8	38.2	-25.8	-4.8	10.0	19.2	50.2	19.2	-37.7	-2.5	0.0
CYLINDER	32	37.5	39.4	37.5	-11.2	-6.4	0.0	14.2	45.0	39.3	-27.0	-4.8	11.5	15.9	50.8	15.9	-43.0	-3.2	0.0
CYLINDER	12	38.4	37.0	38.4	-9.8	-8.4	0.0	13.1	45.0	40.5	-28.0	-4.8	12.6	13.9	53.5	13.9	-46.2	-1.3	0.0



ALL STRESSES IN KIPS/SQUARE INCH

MINUS INDICATES COMPRESSION

EARTHQUAKE DIRECTION



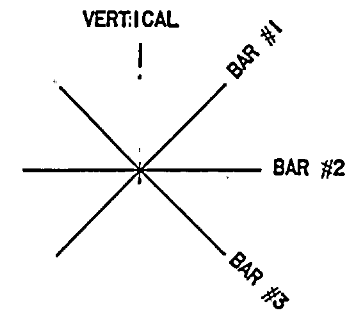
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28B
CONTAINMENT STRUCTURE, EXTERIOR SHELL STRESSES - ACCIDENT CONDITION 2 - LINER CONSIDERED

CONDITION (C) C = 1.0D ± 0.05D + 1.0P + 1.0 T + 1.0 DDE

CRACKED SECTION - LINER CONSIDERED

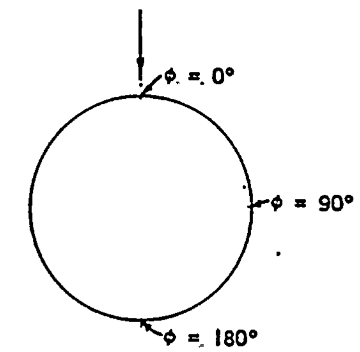
		φ = 0°						φ = 90°						φ = 180°					
		REBAR STRESSES			LINER STRESSES			REBAR STRESSES			LINER STRESSES			REBAR STRESSES			LINER STRESSES		
		1	2	3	σ_y	σ_x	τ_{xy}	1	2	3	σ_y	σ_x	τ_{xy}	1	2	3	σ_y	σ_x	τ_{xy}
	ALLOWABLE STRESS	57.0	57.0	57.0				57.0	57.0	57.0				57.0	57.0	57.0			
	ELEVATION (ABOVE BASE SLAB)																		
DOME	213	30.0	20.8	30.0	-7.6	-18.5	0.0	29.2	21.4	30.4	-8.1	-18.0	.0.5	29.6	22.0	29.6	-8.6	-17.6	0.0
DOME	211	30.4	26.0	30.6	-10.5	-12.3	2.2	29.8	25.7	31.7	-10.8	-11.5	2.8	29.1	26.8	31.8	-11.3	-10.7	2.2
DOME	204	31.6	30.0	29.9	-8.8	-10.2	0.0	29.6	29.7	34.1	-10.0	-7.1	1.9	27.8	33.9	33.9	-10.9	-3.9	0.0
DOME	193	33.0	27.5	27.6	-8.6	-13.0	1.0	28.7	29.2	33.1	-10.6	-8.6	2.2	27.6	35.7	30.7	-12.0	-4.2	1.0
DOME	178	33.8	25.3	26.0	-8.8	-15.1	2.0	27.4	28.4	32.4	-11.7	-10.4	2.5	27.4	37.2	27.5	-14.1	-4.1	2.6
DOME	160	32.6	21.2	32.6	-5.6	-17.4	0.0	23.0	28.0	37.2	-10.4	-11.9	6.2	27.8	35.3	27.8	-14.9	-5.7	0.0
	142	34.5	30.2	34.5	-4.7	-8.1	0.0	21.4	37.4	39.8	-10.8	-2.5	8.4	26.8	44.6	26.8	-16.8	3.1	0.0
CYLINDER	132	35.2	27.6	35.2	-3.8	-10.2	0.0	20.0	33.4	40.8	-11.2	-6.3	9.6	25.5	38.4	25.5	-18.6	-3.2	0.0
CYLINDER	123	36.2	30.2	36.2	-2.9	-7.1	0.0	18.7	36.2	41.7	-12.0	-3.4	10.6	24.4	40.6	24.4	-20.6	-1.2	0.0
CYLINDER	100	39.1	30.2	39.1	0.5	-6.0	0.0	15.6	36.4	43.8	-13.4	-3.5	12.9	20.1	42.7	20.1	-27.6	-0.8	0.0
CYLINDER	77	40.5	29.6	40.5	1.9	-6.1	0.0	12.3	36.5	45.8	-15.0	-3.6	15.3	17.8	43.3	17.8	-31.7	-1.2	0.0
CYLINDER	54	33.4	28.7	33.4	-8.0	-9.4	0.0	6.8	36.4	38.2	-24.8	-5.9	14.4	10.5	44.3	10.5	-41.9	-2.6	0.0
CYLINDER	32	34.5	29.5	34.5	-6.4	-8.2	0.0	6.1	35.9	41.7	-25.9	-5.9	16.3	6.3	44.5	6.3	-46.2	-4.0	0.0
CYLINDER	12	35.6	28.6	35.6	-6.9	-8.7	0.0	12.4	32.9	51.1	-26.9	-5.9	17.7	-0.3	47.9	-0.3	-54.1	-3.1	0.0



ALL STRESSES IN KIPS/SQUARE INCH

MINUS SIGN INDICATES COMPRESSION

EARTHQUAKE DIRECTION



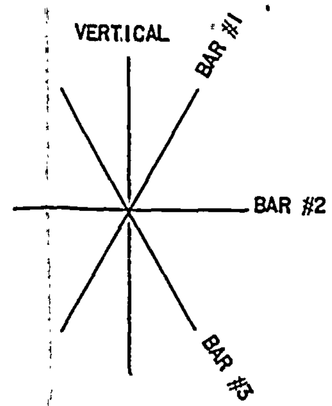
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28C
CONTAINMENT STRUCTURE, EXTERIOR SHELL STRESSES - ACCIDENT CONDITION 3 - LINER CONSIDERED

CONDITION (c) C = 1.0D ± 0.05D + 1.0P + 1.0 T + 1.0DDE

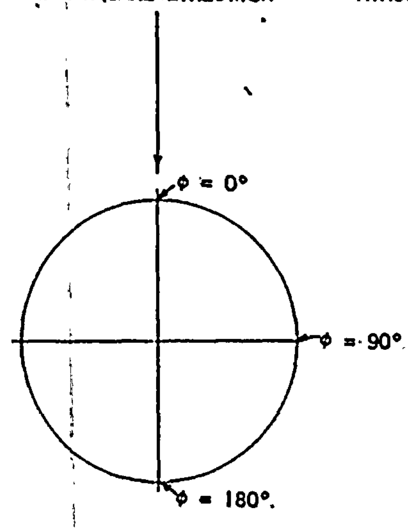
CRACKED SECTION - LINER NOT CONSIDERED

	ELEVATION (ABOVE BASE SLAB)	ϕ = 0° REBAR STRESSES 1	2	3	ϕ = 90° REBAR STRESSES 1	2	3	ϕ = 180° REBAR STRESSES 1	2	3
ALLOWABLE STRESS		57.0	57.0	57.0	57.0	57.0	57.0	57.0	57.0	57.0
DOME	213	26.5	12.6	26.5	25.1	13.7	27.0	25.6	14.7	25.6
DOME	211	26.5	19.4	26.1	25.6	19.0	28.1	24.2	20.9	28.1
DOME	204	28.3	24.6	24.6	24.3	24.4	32.2	20.8	32.1	32.0
DOME	193	30.7	20.7	21.3	22.7	23.4	31.2	20.8	35.0	26.8
DOME	178	31.6	17.5	19.9	20.3	22.1	30.6	20.8	38.3	21.4
DOME	160	30.4	12.1	30.4	13.3	23.4	37.6	20.8	35.5	20.8
CYLINDER	142	32.9	26.5	32.9	8.9	38.0	42.8	18.8	49.5	18.8
CYLINDER	132	34.3	24.0	34.3	6.4	32.3	44.8	16.8	39.5	16.8
CYLINDER	123	36.0	27.5	36.0	3.9	36.2	46.4	14.7	42.6	14.7
CYLINDER	100	40.9	27.4	40.9	-1.8	36.4	50.1	7.1	45.8	7.1
CYLINDER	77	43.5	26.5	43.5	-7.7	36.7	54.0	2.9	46.6	3.0
CYLINDER	54	31.9	25.6	31.9	-9.3	36.8	39.7	-3.1	48.4	-3.1
CYLINDER	32	33.5	26.6	33.5	-10.5	36.0	45.1	-9.4	48.8	-9.4
CYLINDER	12	35.2	25.3	35.2	-1.1	31.5	59.4	-19.4	53.8	-19.4



ALL STRESSES IN KIPS/SQUARE INCH

MINUS SIGN INDICATES COMPRESSION



UNITS 1 AND 2
DIABLO CANYON SITE

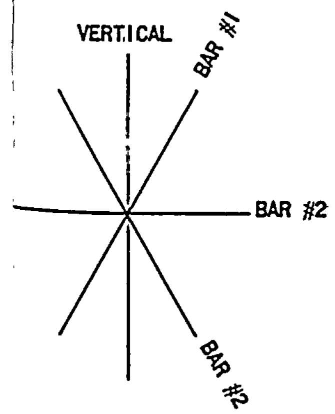
FIGURE 3.8-28E
CONTAINMENT STRUCTURE, EXTERIOR SHELL STRESSES - ACCIDENT CONDITION 3 - LINER NOT CONSIDERED

Amendment 2 January 1974

CONDITION (b) C = 1.0D ± 0.05D + 1.25P + 1.0 T' + 1.25DE

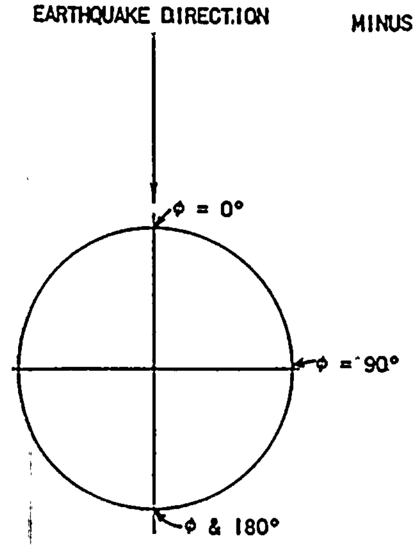
CRACKED SECTION - LINER NOT CONSIDERED

	φ = 0°			φ = 90°			φ = 180°		
	REBAR STRESSES			REBAR STRESSES			REBAR STRESSES		
	1	2	3	1	2	3	1	2	3
ALLOWABLE STRESS	57.0	57.0	57.0	57.0	57.0	57.0	57.0	57.0	57.0
ELEVATION (ABOVE BASE SLAB)									
DOME									
213	33.1	16.2	33.1	32.0	17.0	33.5	32.4	17.9	32.4
211	32.8	24.5	32.9	32.1	24.3	34.7	31.0	25.8	34.7
204	33.5	32.2	32.1	30.7	31.7	39.1	27.8	38.0	37.9
193	35.5	29.5	27.9	29.6	31.0	36.0	28.7	40.2	32.2
178	36.7	26.7	25.1	28.6	30.2	33.3	29.1	42.0	26.4
160	35.9	20.9	35.9	23.4	28.8	40.9	28.7	37.3	28.7
142	38.1	38.1	38.1	20.0	46.8	44.9	26.8	55.6	26.8
CYLINDER									
132	39.1	28.9	39.1	17.9	38.9	46.5	25.1	47.6	25.1
123	39.9	38.9	39.9	16.2	45.1	47.7	23.6	50.1	23.6
100	42.7	38.4	42.7	11.1	45.4	50.3	18.5	52.7	18.5
77	44.8	37.1	44.8	7.7	45.7	51.8	14.0	53.7	14.0
54	31.9	35.3	31.9	2.0	46.0	36.1	6.8	53.9	6.8
32	34.5	37.9	34.5	-1.4	46.3	37.8	1.7	54.8	1.7
12	35.9	34.6	35.9	-3.1	46.3	39.6	-1.4	58.7	-1.4



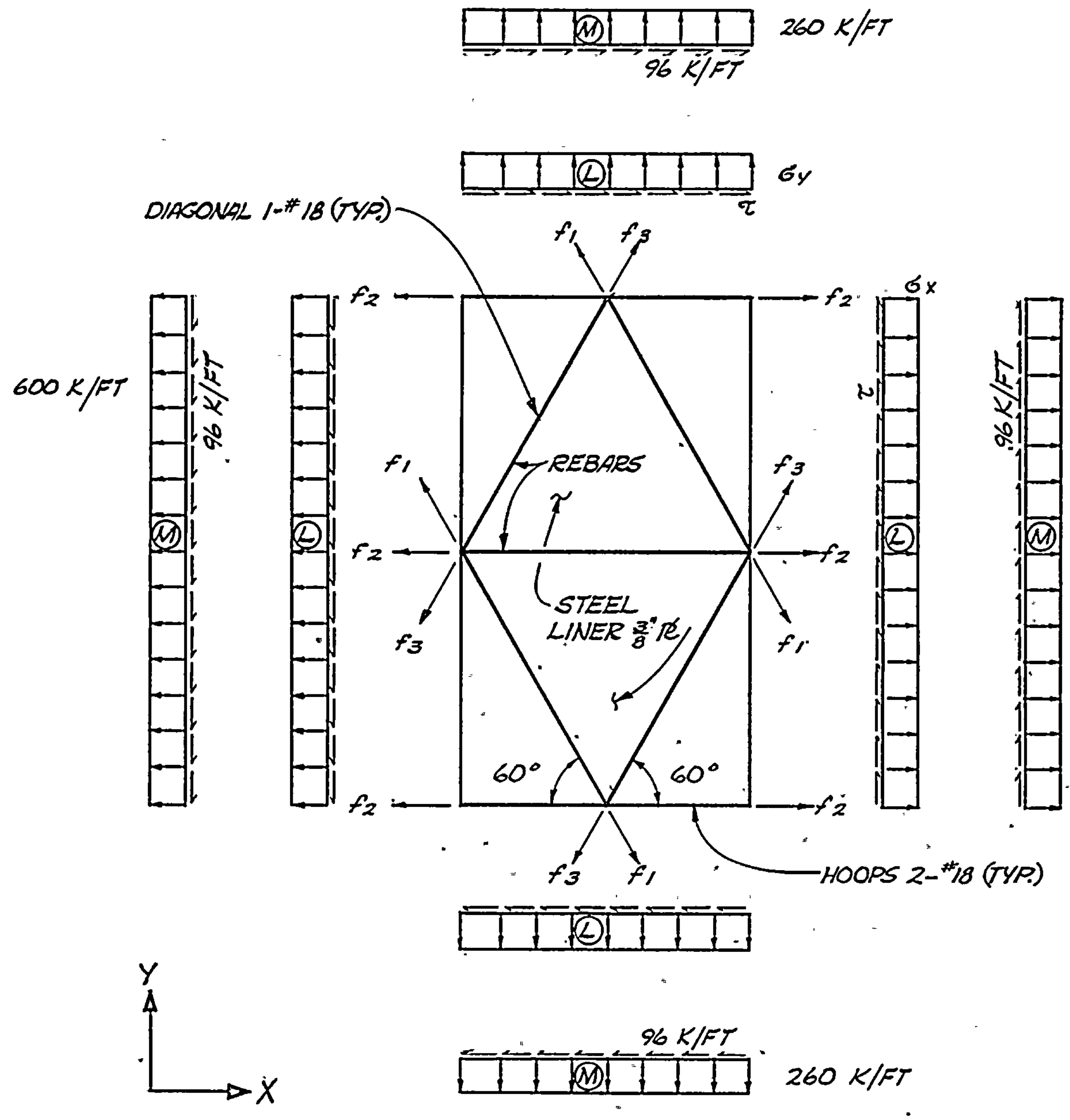
ALL STRESSES IN KIPS/SQUARE INCH

MINUS SIGN INDICATES COMPRESSION



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28D
CONTAINMENT STRUCTURE, EXTERIOR SHELL STRESSES - ACCIDENT CONDITION 2 - LINER NOT CONSIDERED

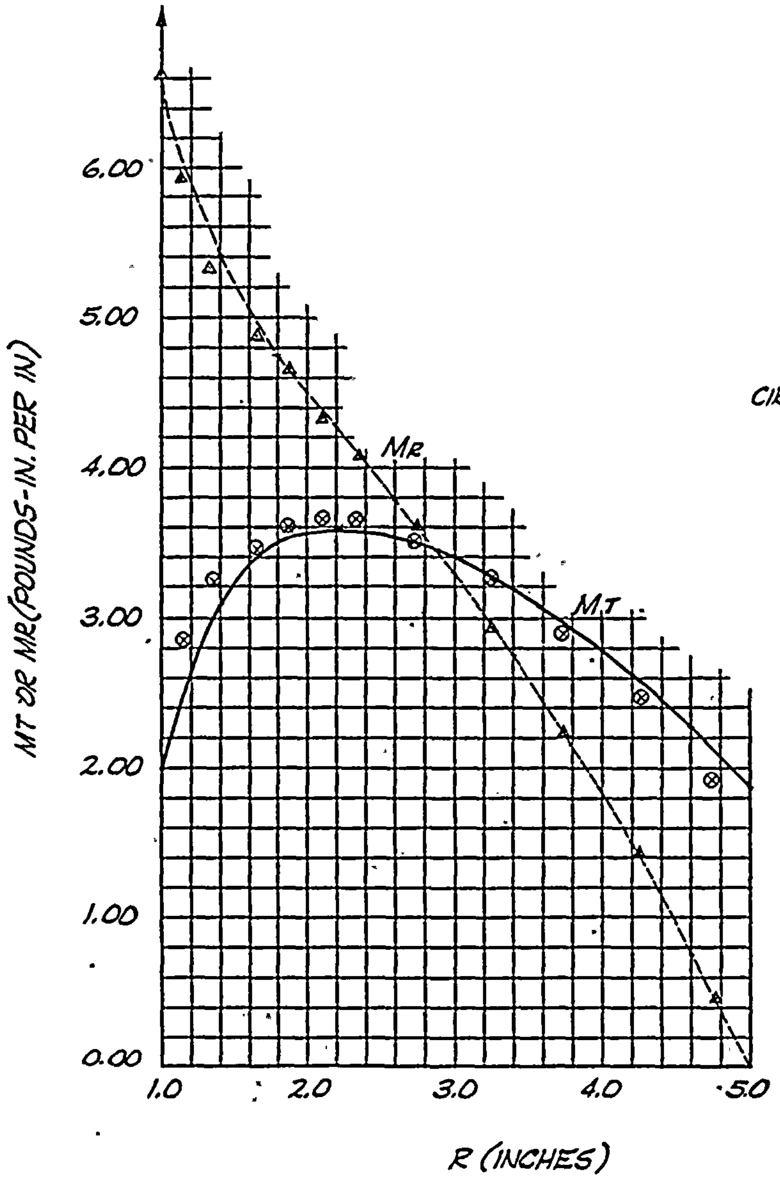


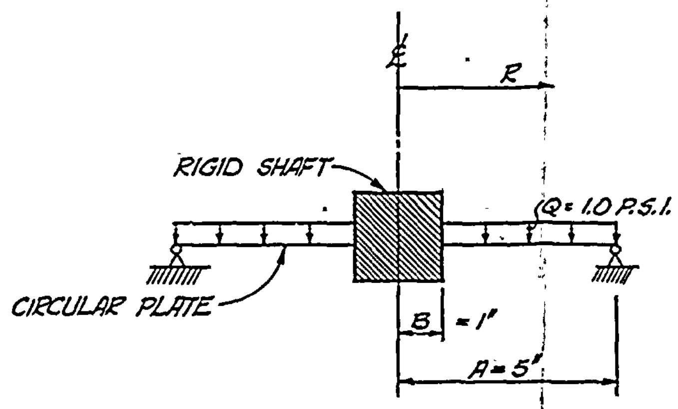
LEGEND:
(L) - LINER STRESSES
(M) - MEMBRANE FROCES

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28F
COMPUTER PROGRAM VERIFICATION
STRAN AND FINITE ELEMENT PROGRAM
TEST PROBLEM MODEL

Amendment 3 February 1974





MODEL

LEGEND:

EXACT SOLUTIONS:

--- MR: RADIAL MOMENT

—— MT: TANGENTIAL MOMENT

APPROXIMATE SOLUTIONS (FINITE ELEMENT PROGRAM):

△ MR

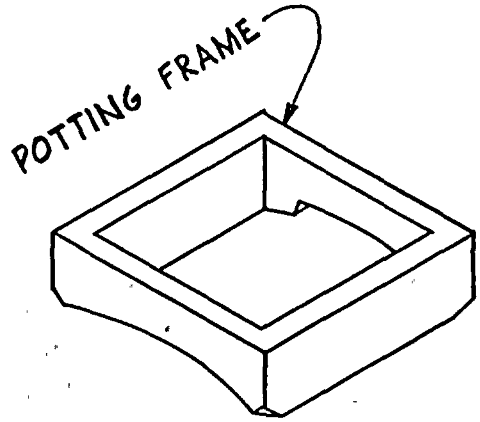
⊗ MT

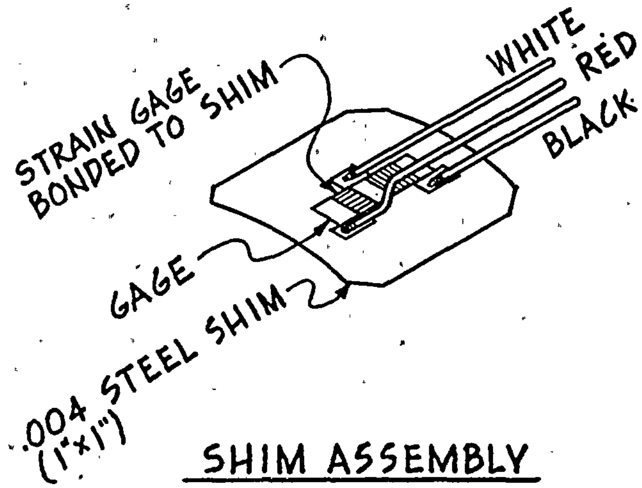
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.8-28G
COMPUTER PROGRAM VERIFICATION
PLH800 PROGRAM
TEST PROBLEM MODEL AND RESULTS

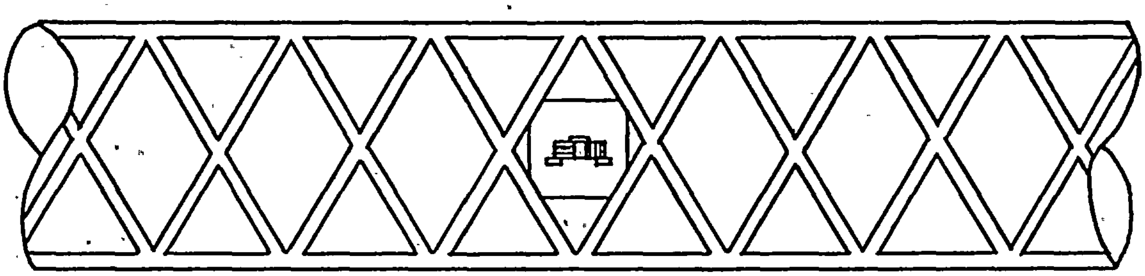
NOTES FOR FIELD INSTALLATION

1. REMOVE MILL SCALE AND POLISH REBAR SURFACE INSIDE OF DIAMOND AREA
2. BOND SHIM ASSEMBLY TO POLISHED REBAR
3. PLACE SEALANT AND PROTECTIVE COVER OVER SHIM ASSEMBLY.





SHIM ASSEMBLY



UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 3.8-29
CONTAINMENT STRUCTURE
TYPICAL REBAR STRAIN GAUGE