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Dalwyn R. Davidson
VICE PRESIDENT
SYSTEM ENGINEERING AND CONSTRUCTION

April 26, 1982

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Mr. A. Schwencer
Chief, Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Draft SER
Containment Systems Branch

Dear Mr. Schwencer:

This letter and its attachment is submitted to provide revised responses to the concerns identified in the Draft SER for Containment Systems.

It is our intention to incorporate these responses in a subsequent amendment to our Final Safety Analysis Report.

Very Truly Yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD: mlb

cc: Jay Silberg
John Stefano
Max Gildner

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480.41 The response to question 480.4 and 480.31 was inadequate. Provide the following information regarding subcompartment analysis, blowout panels and air seals.

a) Both the blowout panels and air seals are hinged to swing open to prevent them from becoming missiles. Provide and discuss initial performance testing and subsequent periodic functional testing of the blowout panels and air seals that will verify that they will function as designed.

b) Provide an analysis that show there will be no missiles generated by the blowout panels or air seals.

c) Factory Mutual Research Corporation (FMRC) report that provides the experimental data to support the assumptions used in the subcompartment analysis was deficient. Discuss the number and location of the fasteners used in the static test, small scale dynamic tests, and large scale tests and how they relate to those used in the actual design of Perry.

d) Appendix 3 of FMRC report indicates that the force per bolt to shear the Group V bolt is in the range of 270 to 290 pounds. However, in the large scale tests, the force per bolts to shear bolts was less than 0.1 pounds. Explain this discrepancy.

e) In the two large scale dynamic tests of FMRC's report 12 inch wide strips of galvanized steel were used in the blowout panel. Discuss how these strips were held together and provide drawings of the test blowout panel and the blowout panel used in Perry.

f) Discuss how the blowout functions during the large scale test in FMRC report, i.e., did all the vent areas become available immediately following the release pressure, or did only part of the vent area become available. Include in your discussion whether or not the panel remains intact after it was released.

g) Compare the pressure transient in the test chambers described in FMRC report to those predicted in Perry subcompartments. If different, provide assurance that the panel will remain intact.

RESPONSE

An analysis will be performed to show that the blowout panels will not act as missiles and that they will remain intact during the transient.

480.42 Response to question 480.5 and 480.32 was incomplete. Provide the following information.

a) Transient loading as a function of time on the major component and structure in the reactor annulus region that was used to establish the adequacy of the design. This should include both load forcing functions (e.g., f(t), fx(t), fy(t)), and transient moments (e.g., M(t), Mx(t), My(t), as resolved about a specific, identified coordinate system

b) Provide plan and elevation drawings of the biological shield wall annulus region in enough detail to verify the nodalization model used in the analysis.

RESPONSE

Additional information was provided as a revised response to question 480.5

480.5 For each subcompartment analyzed, provide the following information:

a) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variation circumferentially, axially, and radially within the compartment. Describe and justify the nodalization sensitivity study performed for the major component supports evaluation, where transient forces and moments acting on the components are of concern.

b) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

c) Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

d) Provide the peak and transient loadings on the major components used to establish the adequacy of the supports design. This should include the load forcing functions (e.g., $f(t)$, $f_x(t)$, $f_y(t)$) and transient moments (e.g., $M(t)$, $M_x(t)$, $M_y(t)$) as resolved about a specific, identified coordinate system.

480.05

Response

The response to items a) and b) is provided in revised Section 6.2.1.2.

c. The contribution to the total design force or moment loading from any individual node may be evaluated using the dimensions and coordinate system provided in Part d), plus the design nodal pressure information provided in Tables 6.2-12, 6.2-13, and Figures 6.2-29 and 6.2-30.

d. Force and moment components on the biological shield wall were evaluated based on an inside diameter of 13'-9½". The coordinate system is as shown in Figures B and C. Peak resultant shears and moments for the design loading time steps are given in Figure D. Transient force and moment components about the X1 and X3 axes are shown in Figures E.1 through E.8.

   Note that since the recirculation line breaks occur on a major axis of the coordinate system [recirc. suction at 0^{o} (X1) and recirc. discharge at 90^{o} (X3)] net transient force and moments about the second axis will be zero. Resultant shear and moment for the feedwater break can be obtained by using the SRSS method of the individual components.

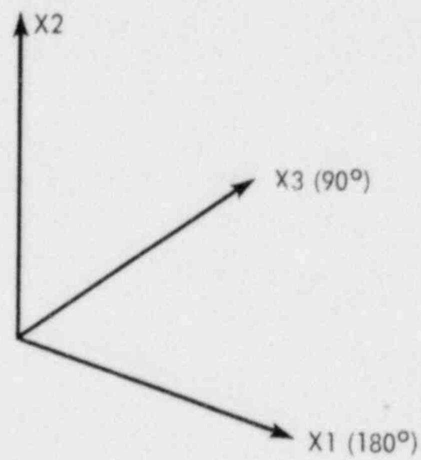


FIGURE A
RIGHT HAND CO-ORDINATE SYSTEM
UTILIZED FOR BIO WALL
ANNULUS PRESSURIZATION LOADINGS

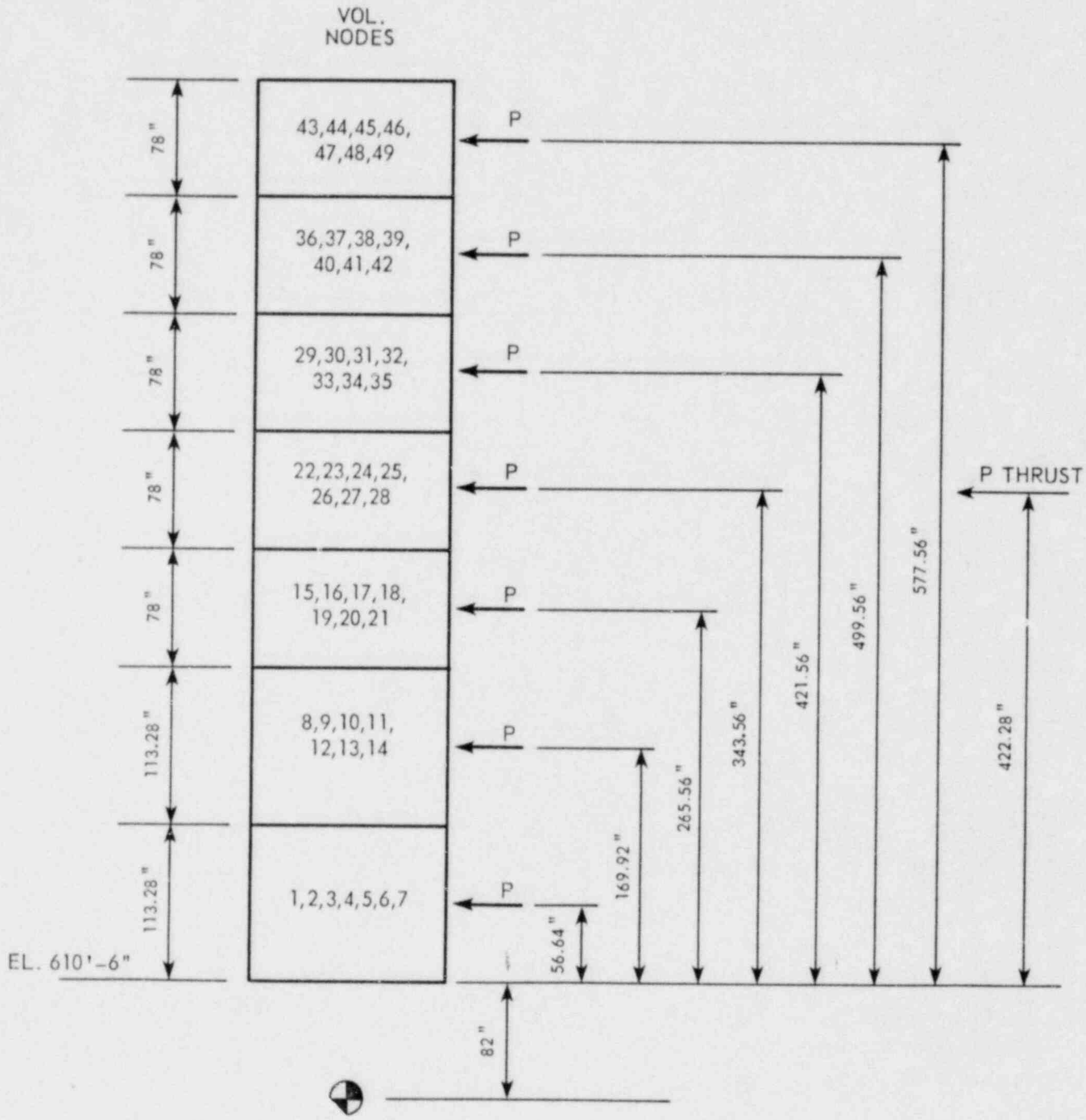


FIGURE B
FORCE MOMENT ARMS ON BIO WALL
FOR ANNULUS PRESSURIZATION DUE
TO FEEDWATER LINE BREAK

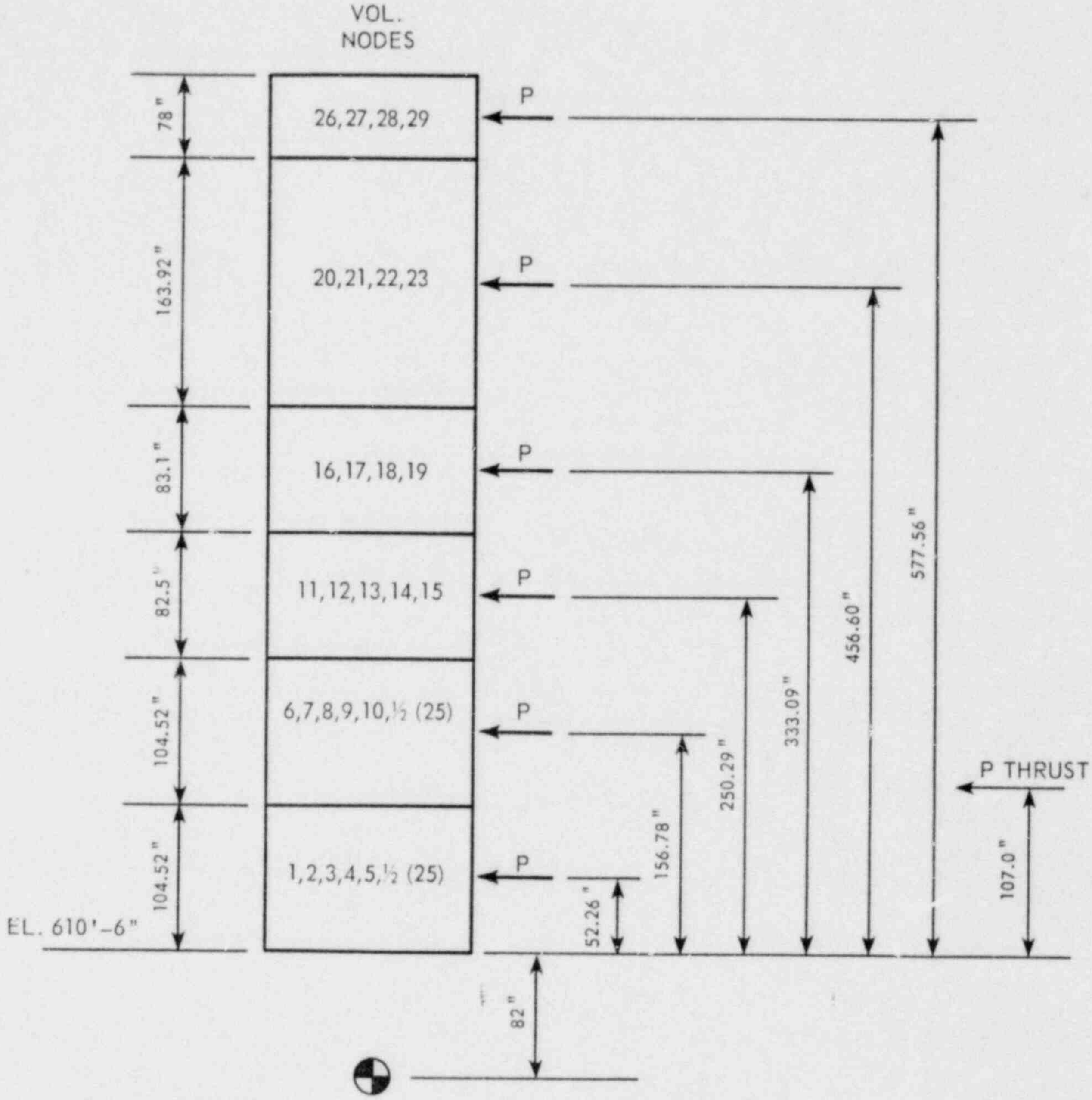
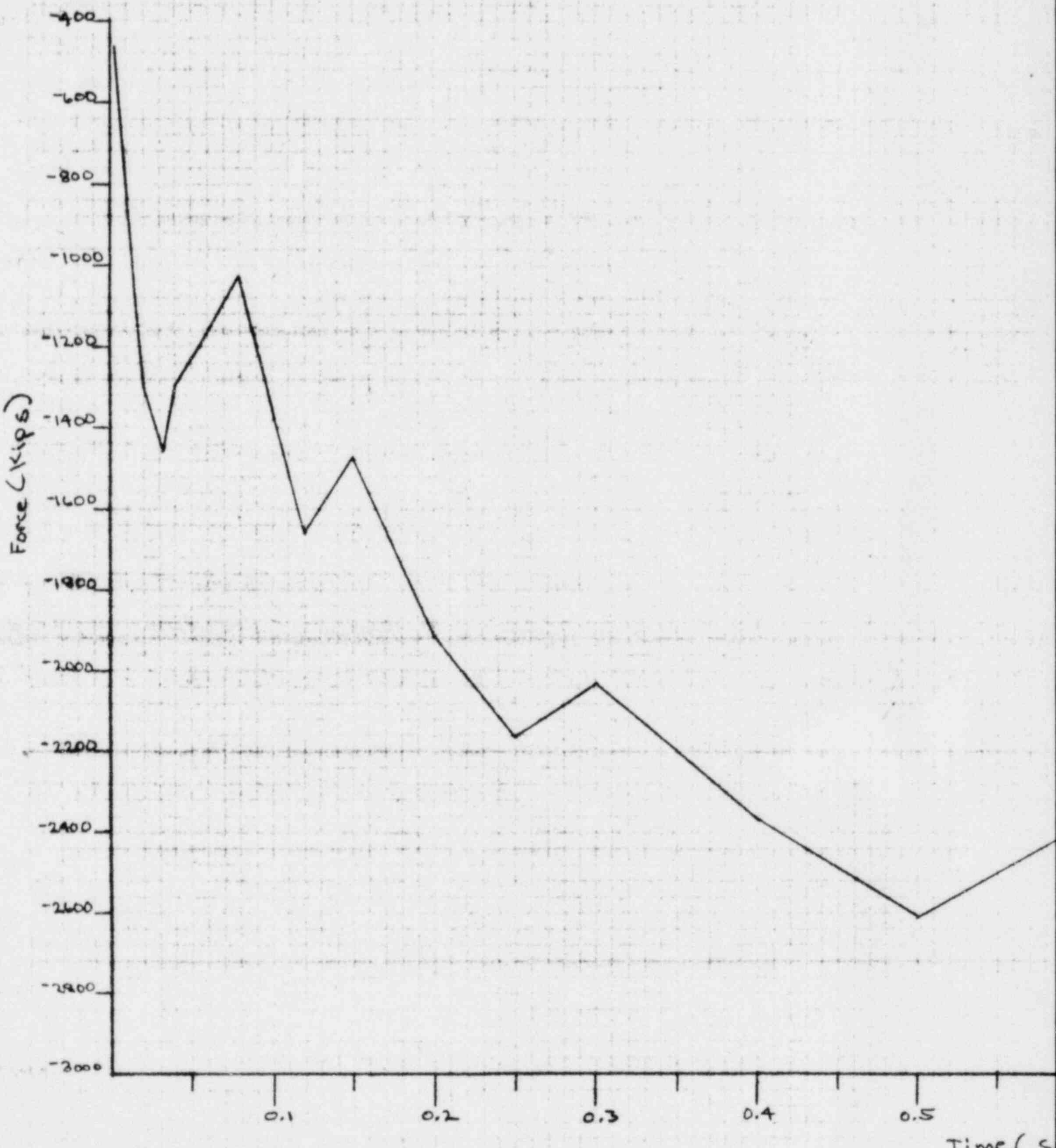


FIGURE C
FORCE MOMENT ARMS ON BIO WALL FOR ANNULUS PRESSURIZATION DUE TO RECIRC.DISCHARGE AND SUCTION LINE BREAKS

BREAK	SHEAR (KIPS)	MOMENT (IN-KIP)	TIME STEP (SEC)
FEEDWATER	2713.5	1,123,300.0	0.500
RECIRC. SUCTION	1380.5	286,137.0	0.024
RECIRC. DISCHARGE	1865.4	385,443.0	0.024
RECIRC. DISCHARGE	1652.0	464,821.0	0.400

FIGURE D
RESULTANT BIO WALL FORCES AND
MOMENTS DUE TO ANNULUS PRESSURIZATION

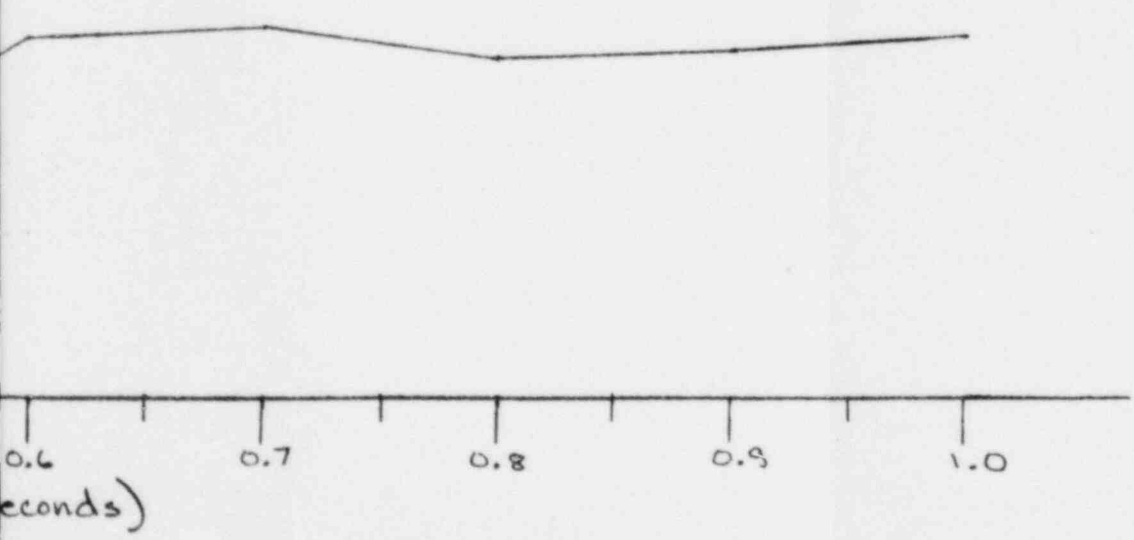


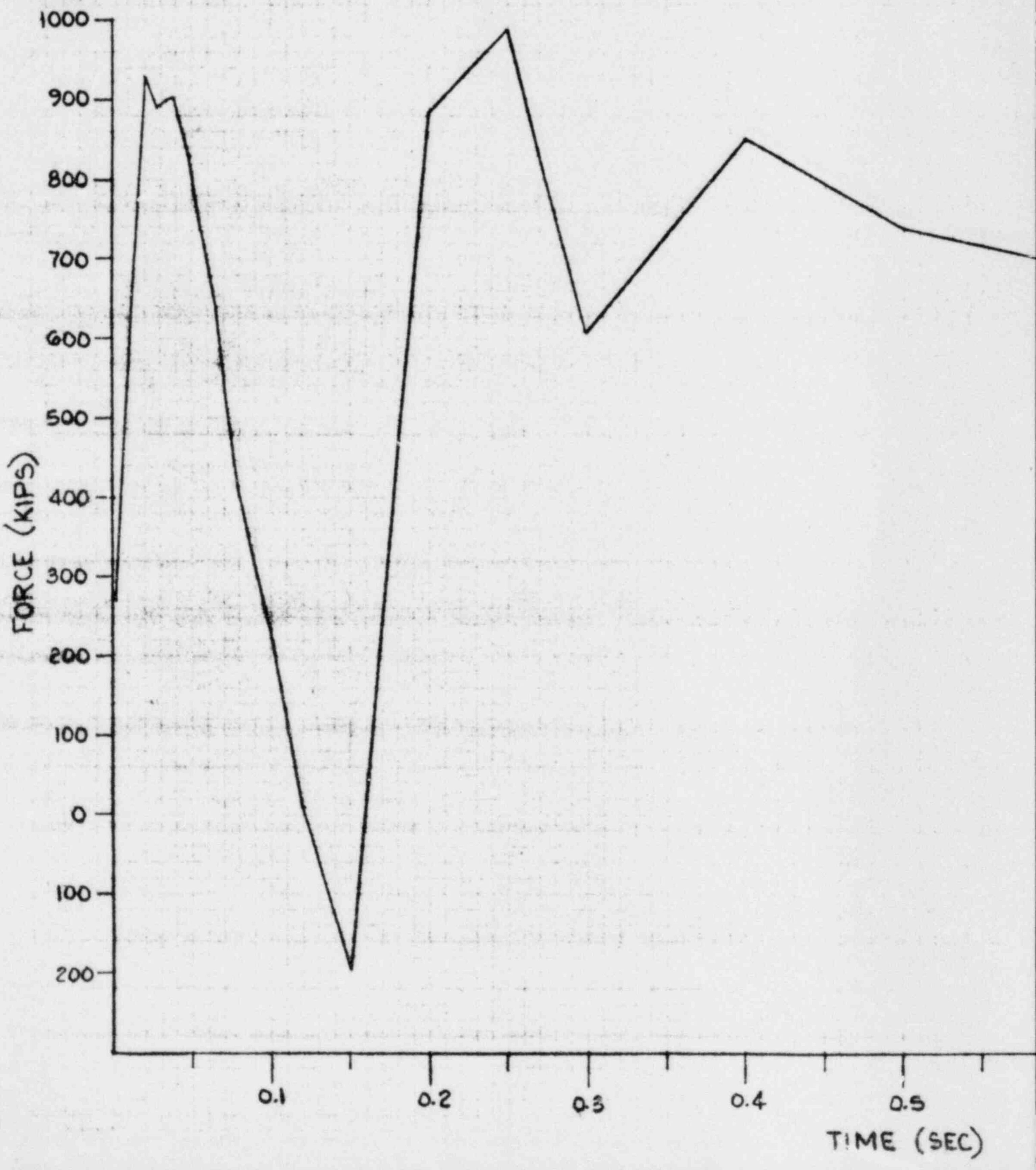
-400
-600
-800
-1000
-1200
-1400
-1600
-1800
-2000
-2200
-2400
-2600
-2800
-3000
Force (Kips)
0.1
0.2
0.3
0.4
0.5
Time(S

	MADE	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.
	CHK'D	
	SQ. CF.	
	C. DFN.	
	ENG.	WORK ORDER / SIZE / DRAWING / REV
	REV. CH. APP. DATE	

Perry Nuclear Power Plant
Feedwater Line Break
Force X1
Bio-Shield Wall

fig. E.1





1000
900
800
700
600
500
400
300
200
100
0
100
200
FORCE (KIPS)
0.1
0.2
0.3
0.4
0.5
TIME (SEC)

	MADE	GILBERT ASSOCIATES, INC.
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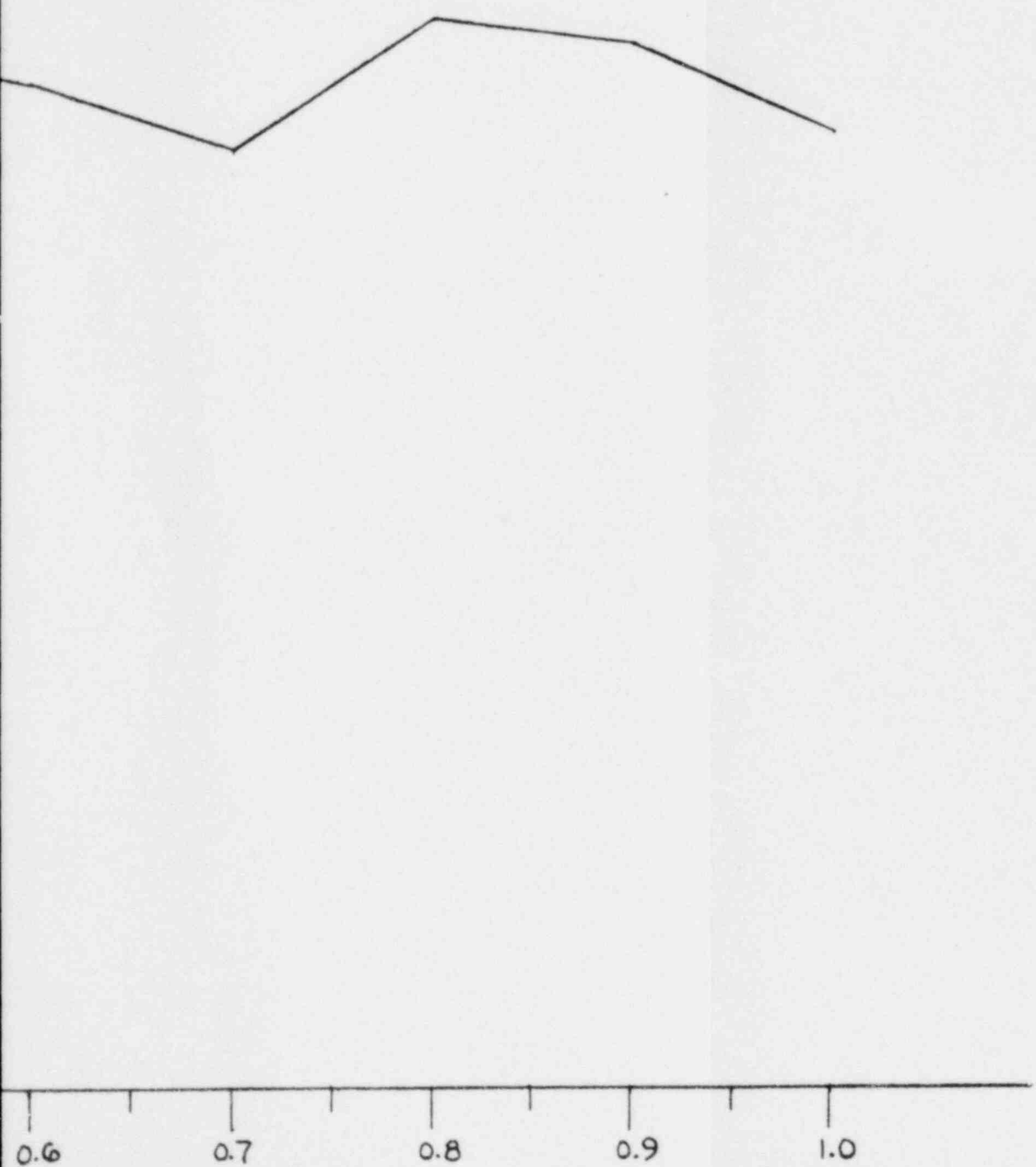
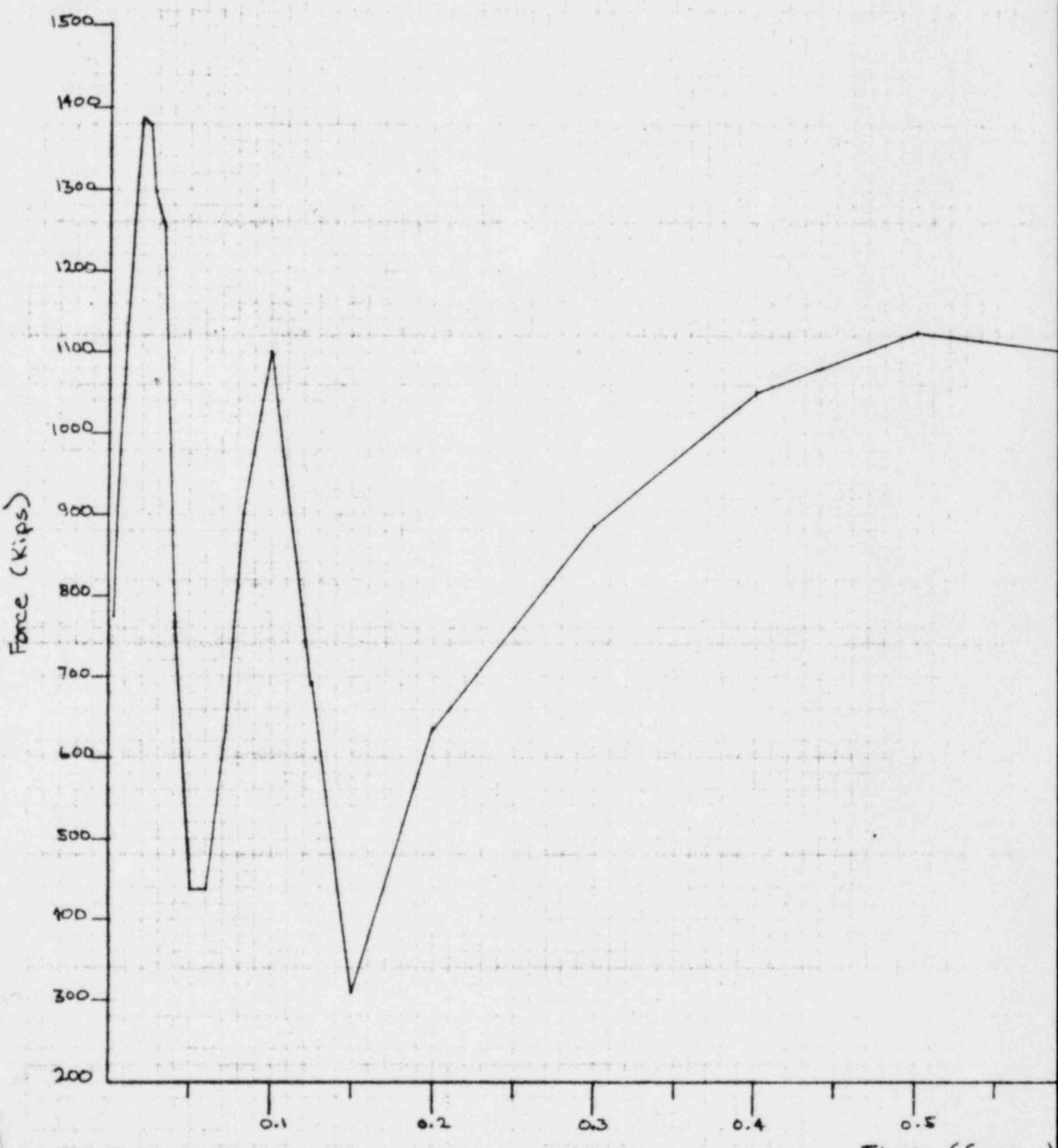
Perry Nuclear Power Plant
Feedwater Line Break
Force X3
Bio-Shield Wall

fig E.2



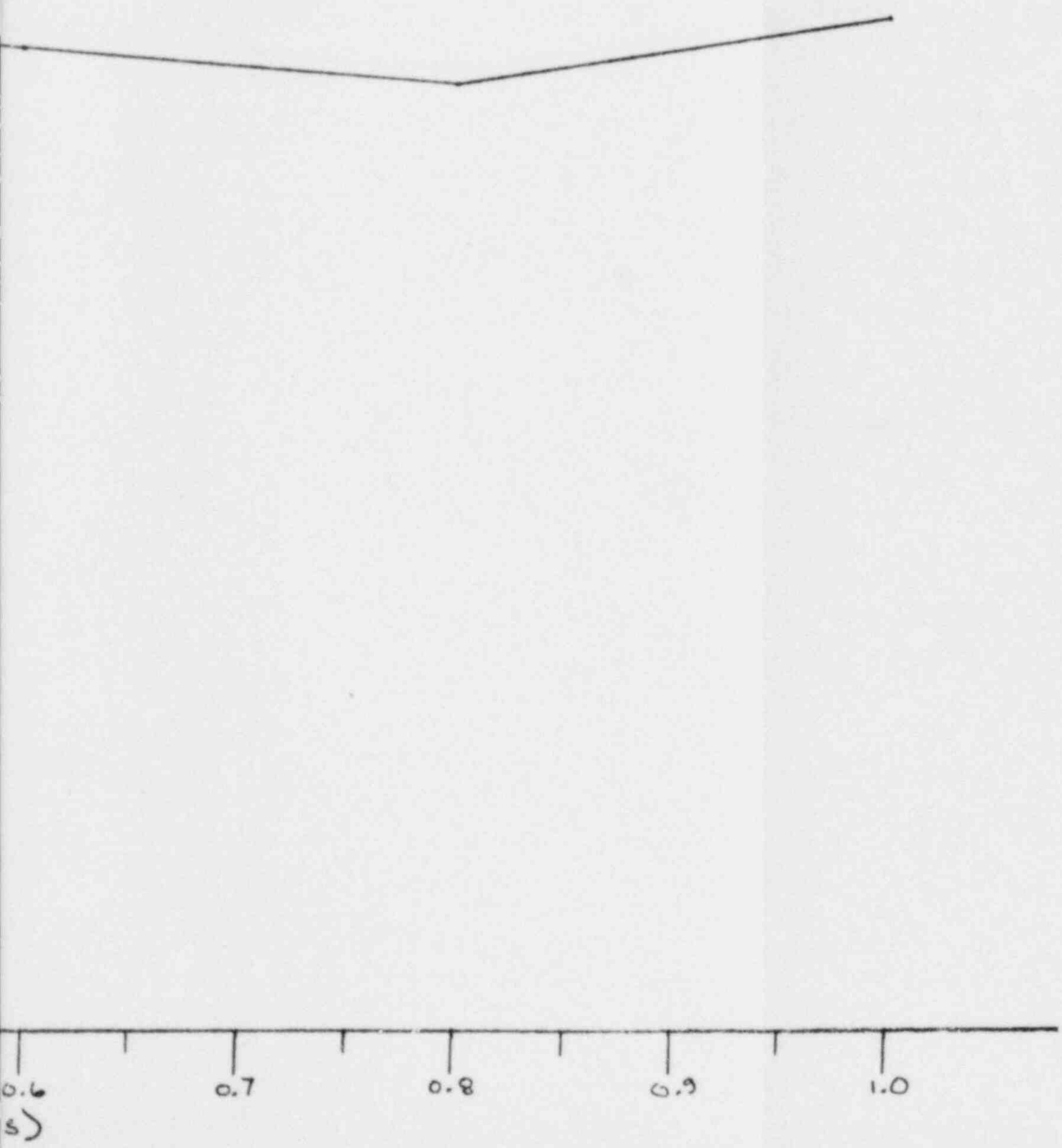


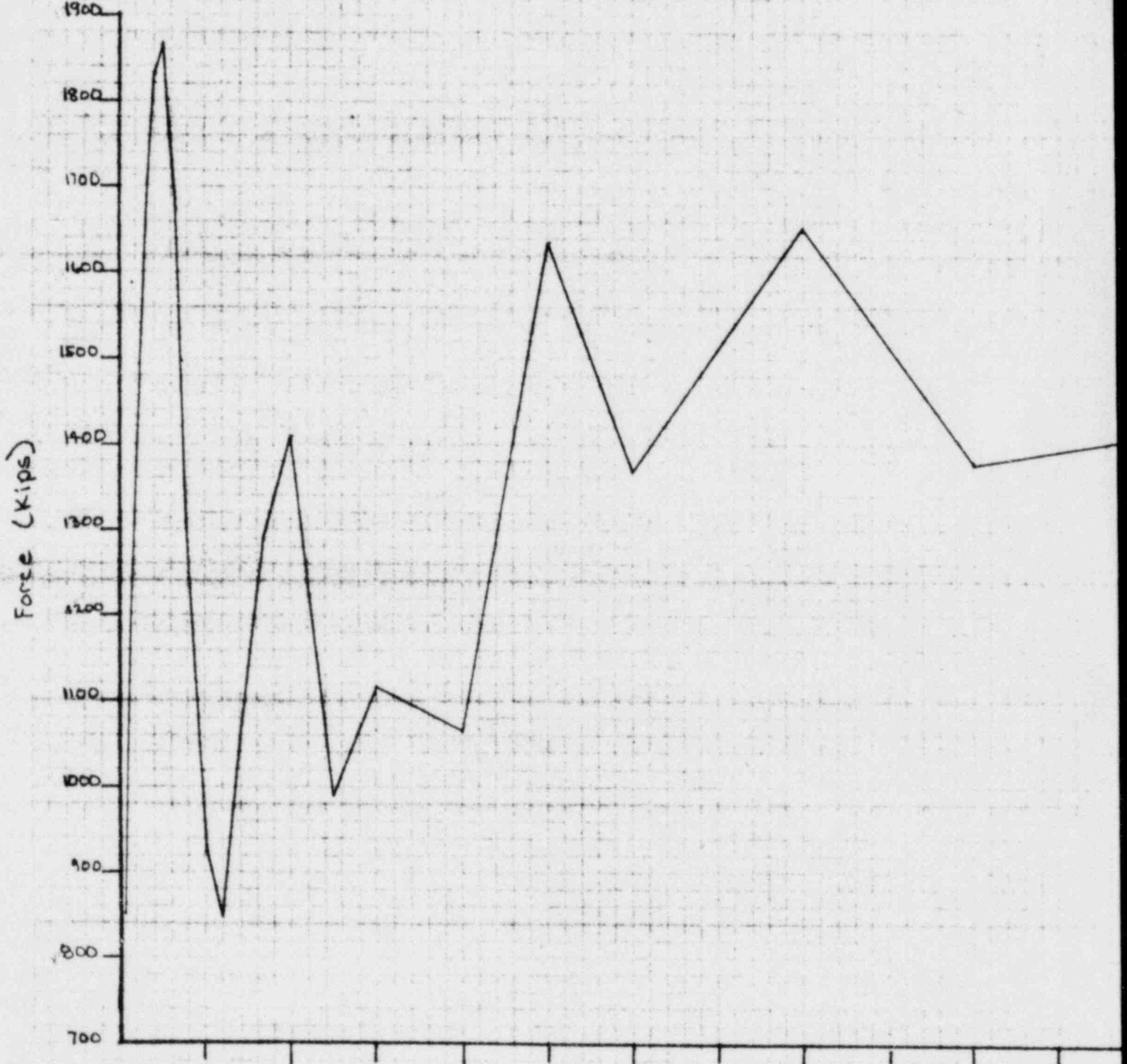
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
Force (Kips)
0.1
0.2
0.3
0.4
0.5
Time (Second

	MADE	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.
	CHK'D.	
	SQ. CF.	
	CF. DFM.	
	ENG.	WORK ORDER / SIZE / DRAWING / REV.
	REV. CH. APP. DATE	

Perry Nuclear Power Plant
Recirculation Suction Line Break
Force X1
Bio-Shield Wall

fig 5.3



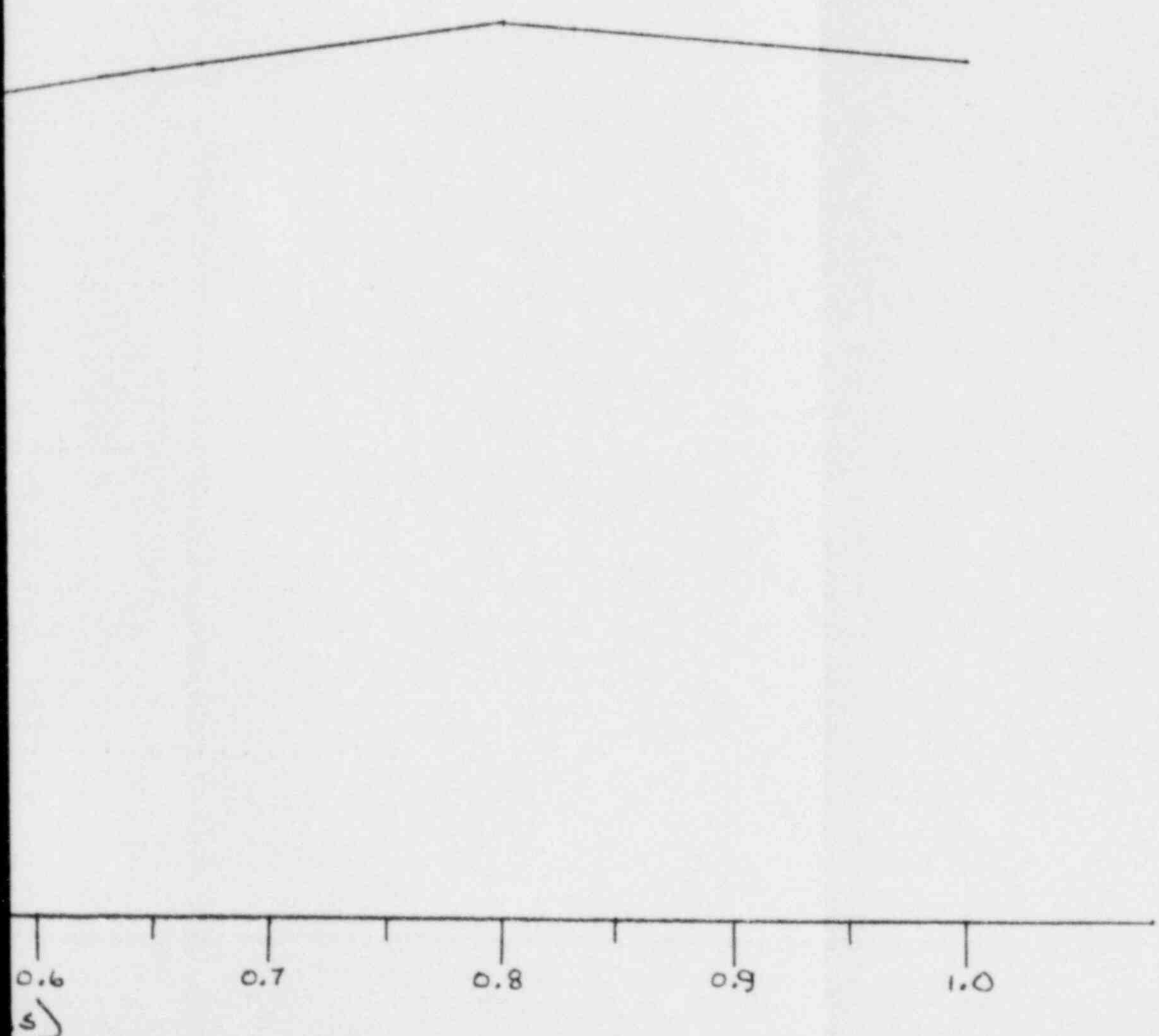


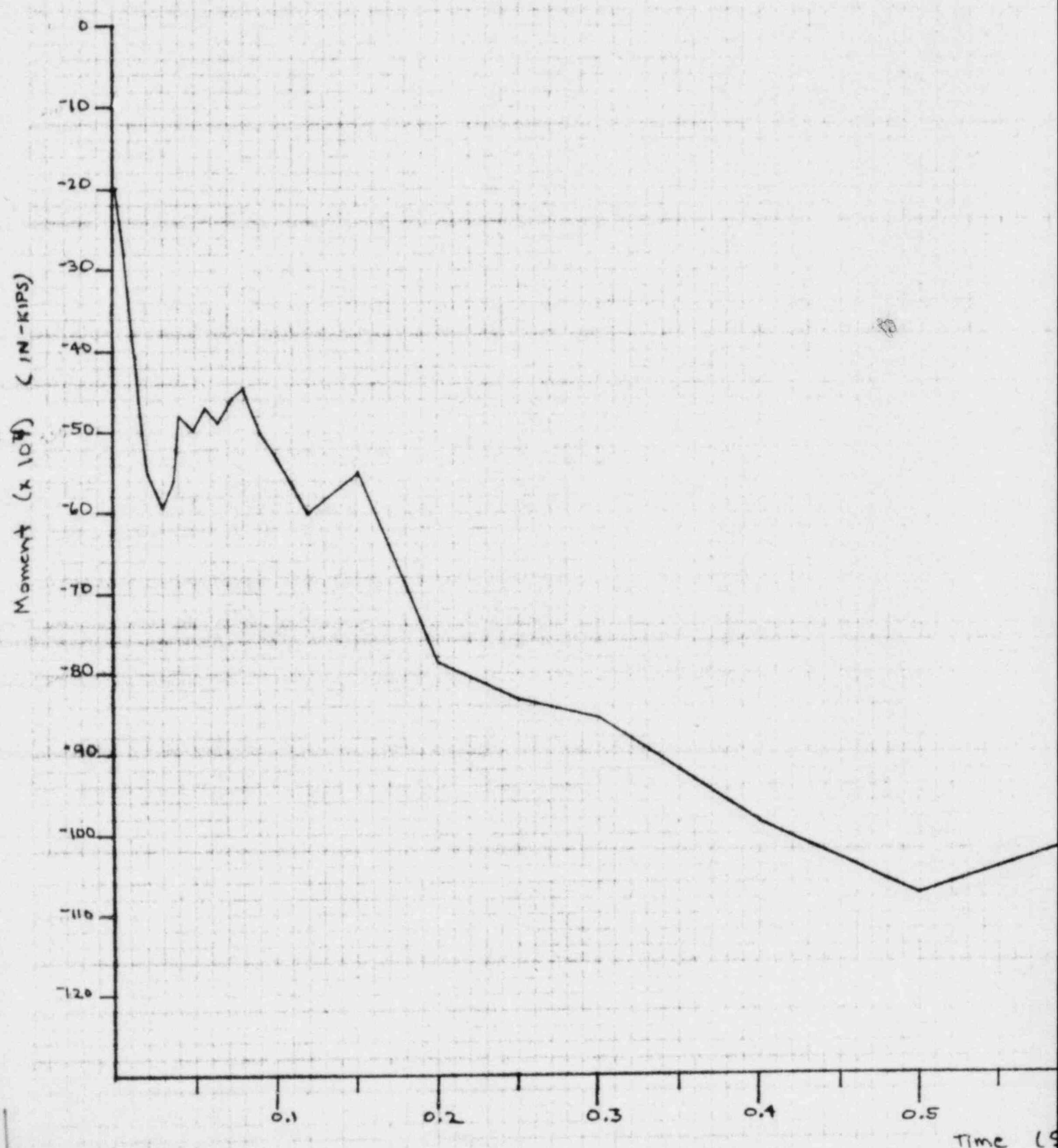
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
700
Force (Kips)
0.1
0.2
0.3
0.4
0.5
Time (Second

	MADE	**GILBERT ASSOCIATES, INC.** ENGINEERS AND CONSULTANTS READING, PENNA.
	CHK'D.	
	SQ CP.	
	CP. DPN.	
	ENG	WORK ORDER / SIZE / DRAWING / REV.
	REV. CH. APP. DATE	

Perry Nuclear Power Plant
Recirculation Discharge Line Break
Force X.3
Bio-Shield Wall

fig 6.4



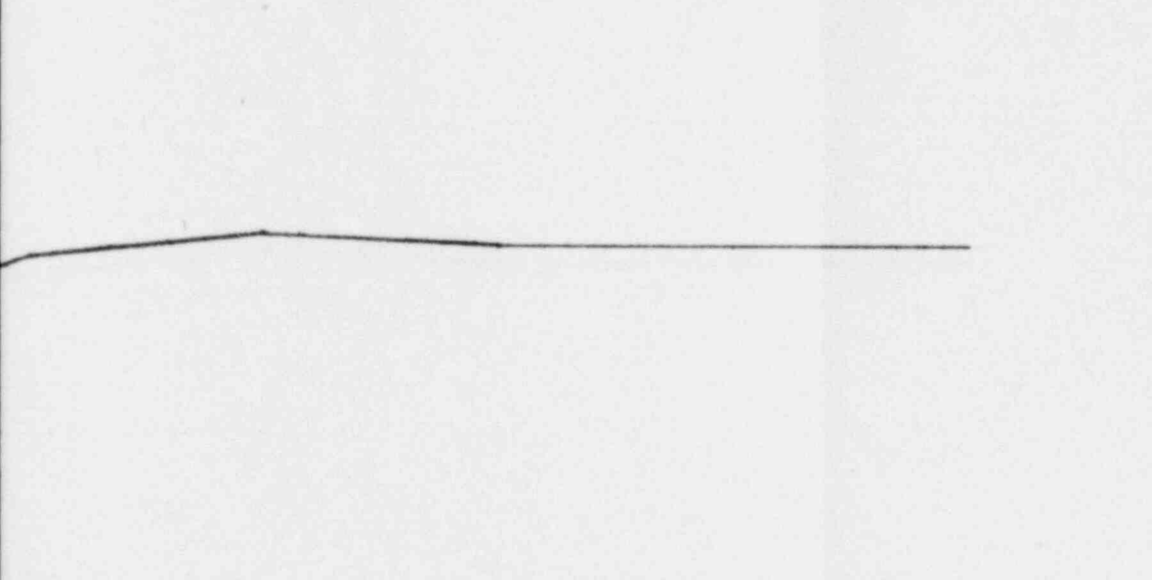


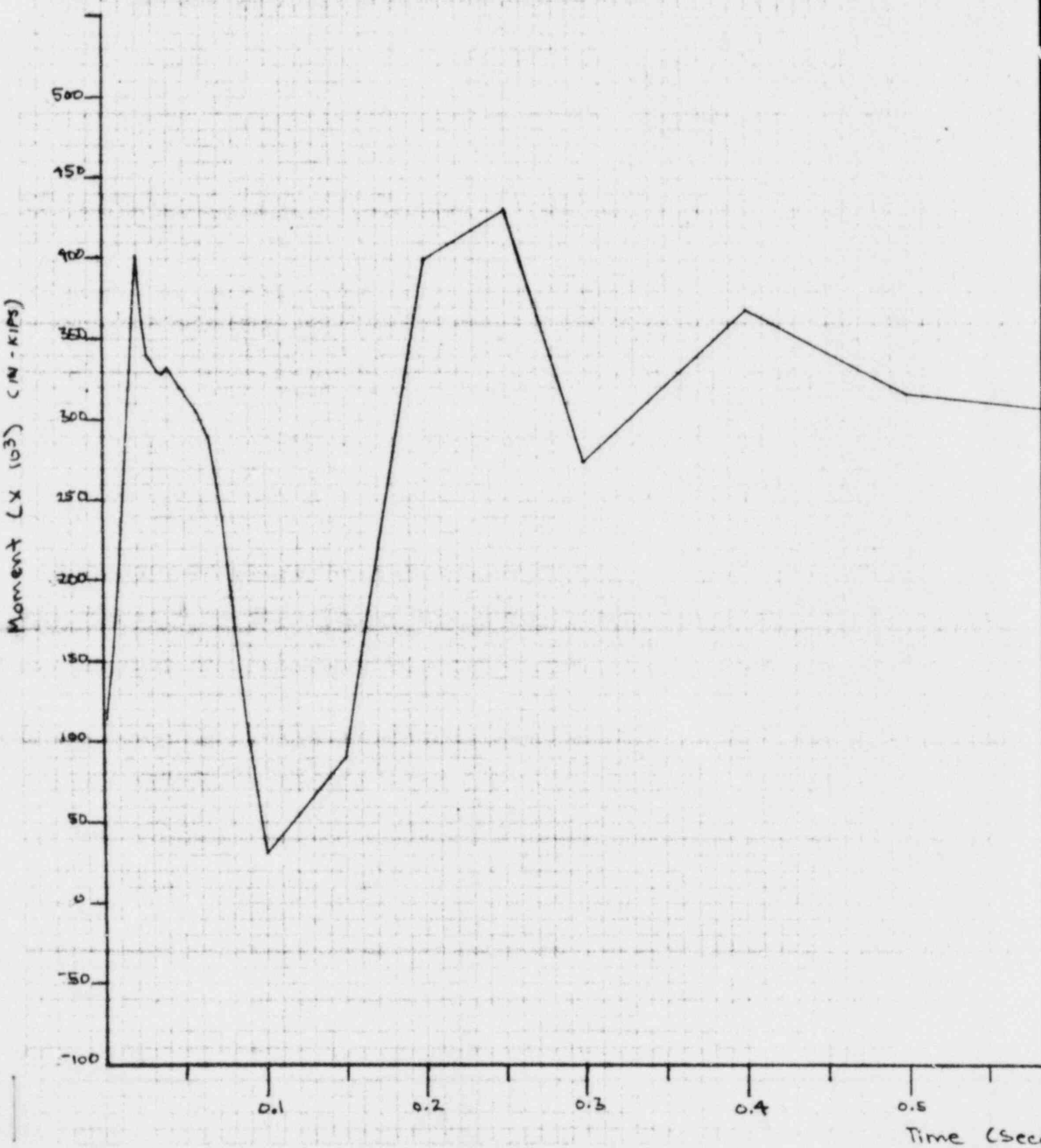
Moment (x 10⁴) (IN-KIPS)
0
-10
-20
-30
-40
-50
-60
-70
-80
-90
-100
-110
-120
0.1
0.2
0.3
0.4
0.5
Time

	MADE	**GILBERT ASSOCIATES, INC.** **ENGINEERS AND CONSULTANTS** READING, PENNA.
	CHK'D.	
	SQ. CP.	
	CP. DFN.	
	ENG.	WORK ORDER | SIZE | DRAWING | REV.
	REV. CH. APP. DATE	

Perry Nuclear Power Plant
Feedwater Line Break
Moment X1
Bio-Shield Wall

fig E.5





500
450
400
350
300
250
200
150
100
50
0
-50
-100
Moment (X 10³) (IN - KIPS)
0.1
0.2
0.3
0.4
0.5
Time (Seco

	MADE	GILBERT ASSOCIATES, INC. ENGINEERS AND CONSULTANTS READING, PENNA.
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	CF. DFM.	
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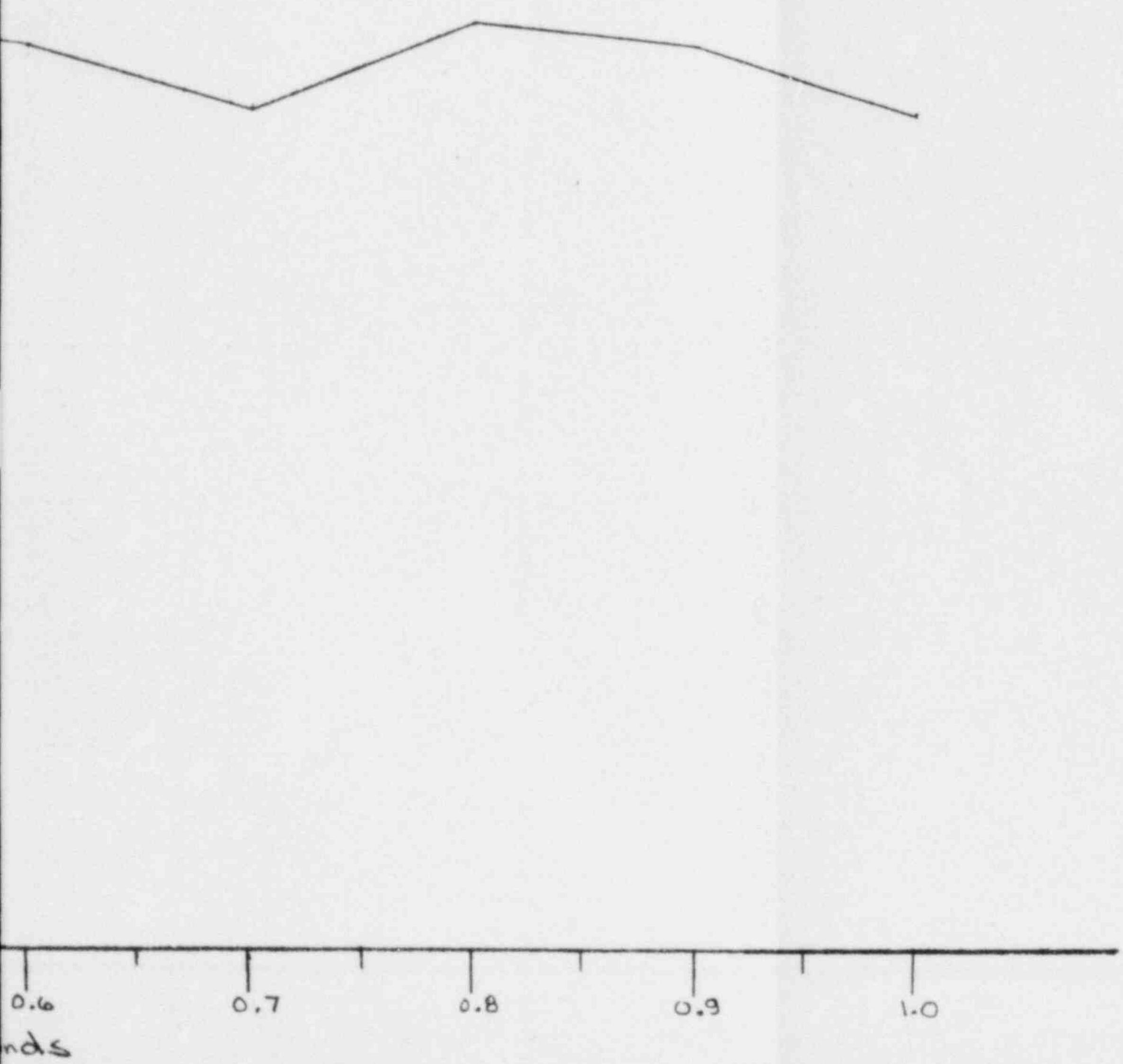
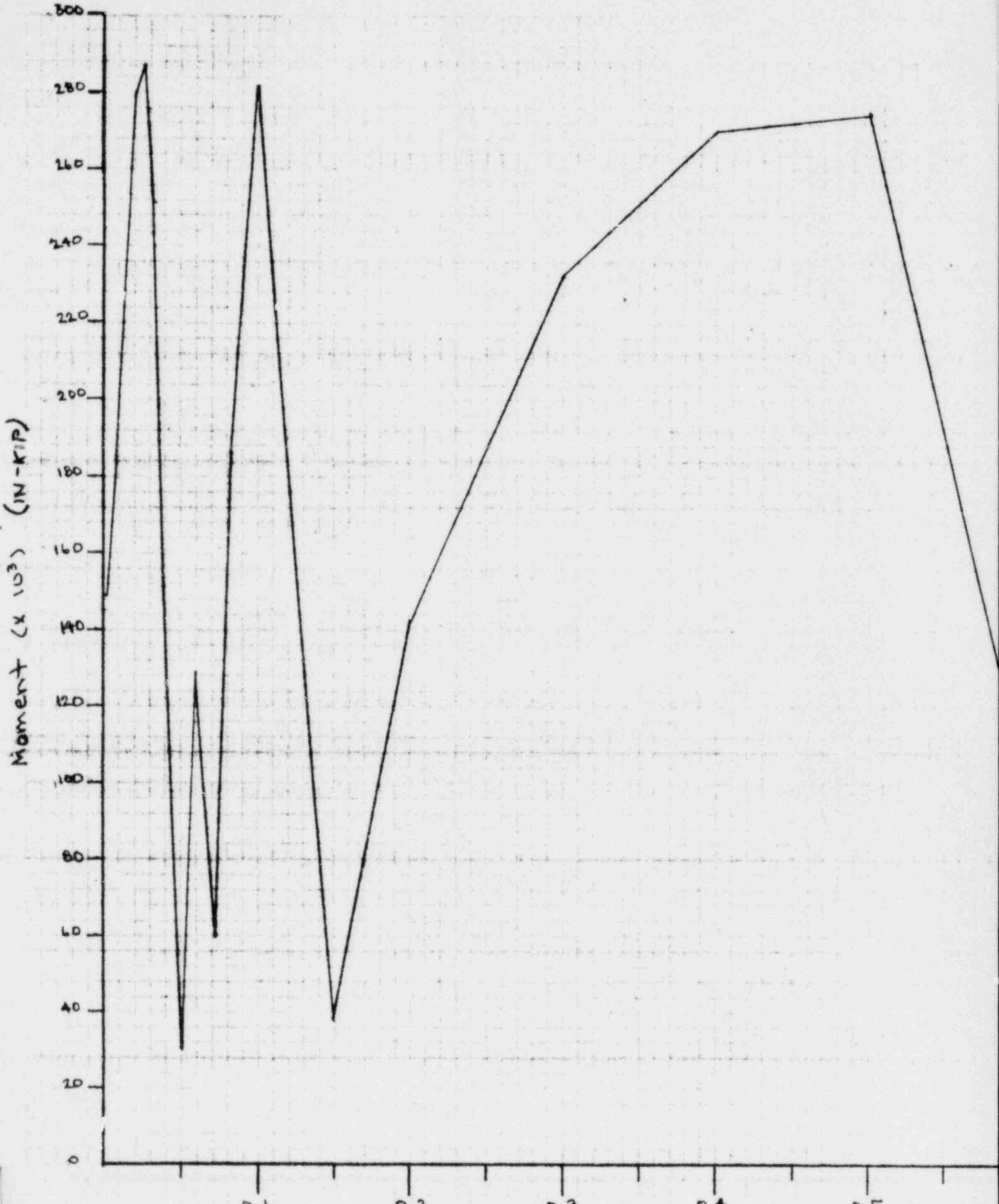
	REV. CH. APP. DATE	

Perry Nuclear Power Plant
Feedwater Line Break
Moment X3
Bio-Shield Wall

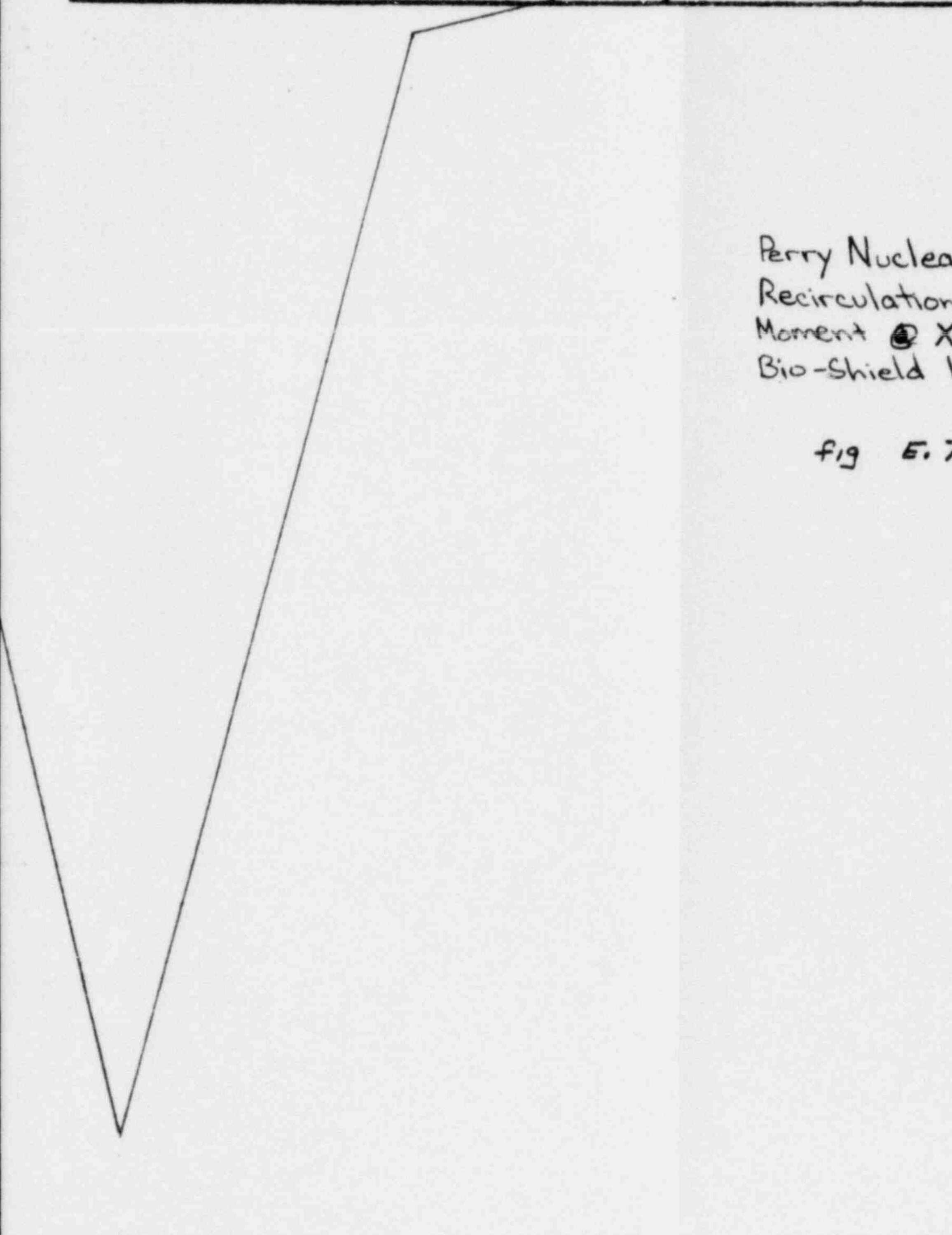
fig E.7





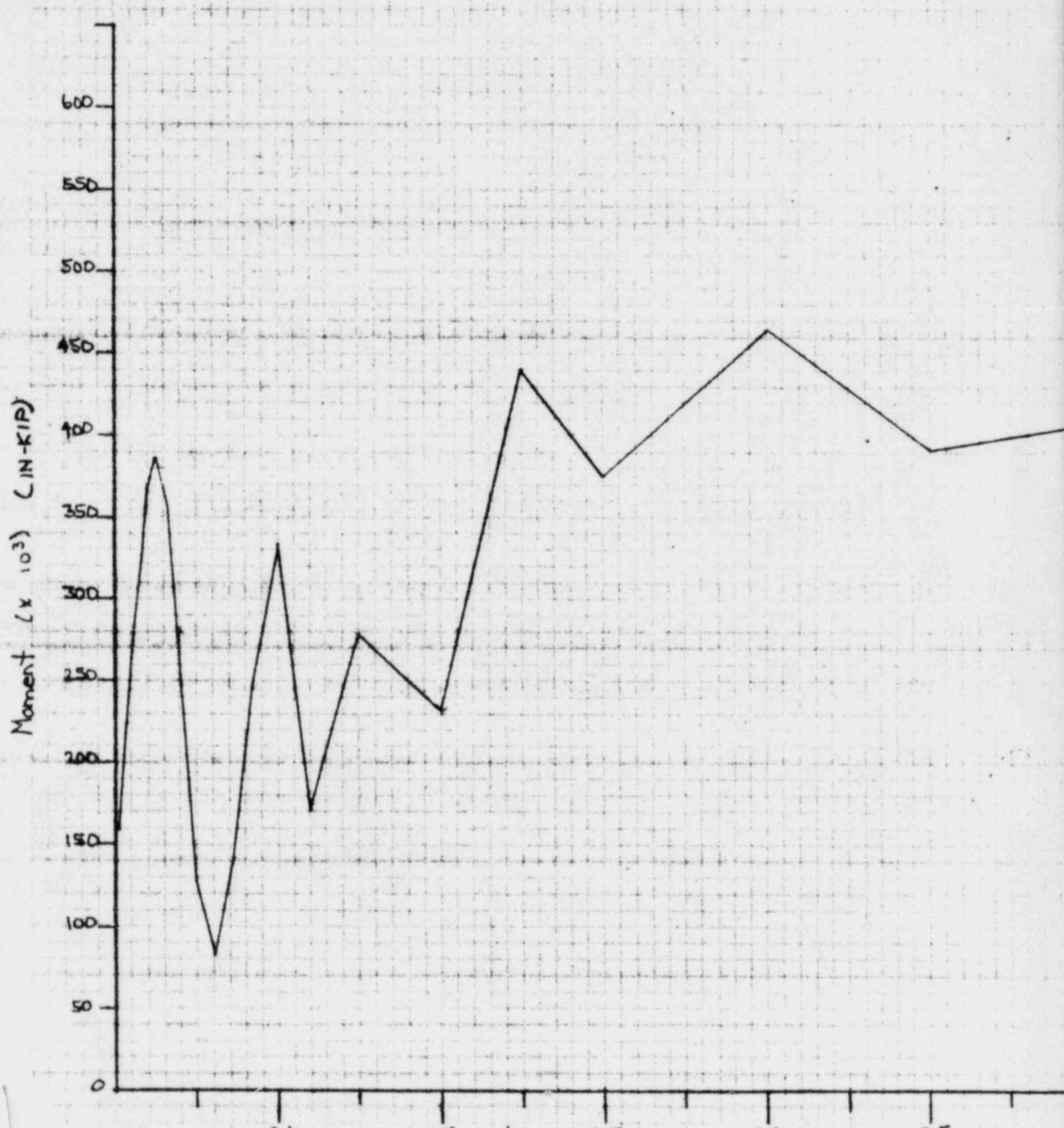
300
280
260
240
220
200
180
160
140
120
100
80
60
40
20
0
Moment (x 10^3) (IN-KIP)
0.1
0.2
0.3
0.4
0.5
Time (Seconds)

	MADE	GILBERT ASSOCIATES, INC.
	CHK'D.	ENGINEERS AND CONSULTANTS
	SQ. CP.	READING, PENNA.
	CP. DPN.	
	ENG.	WORK ORDER / SIZE / DRAWING / REV.
	REV. CH. APP. DATE	



Perry Nuclear Power Plant
Recirculation Suction Line Break
Moment @ X3
Bio-Shield Wall

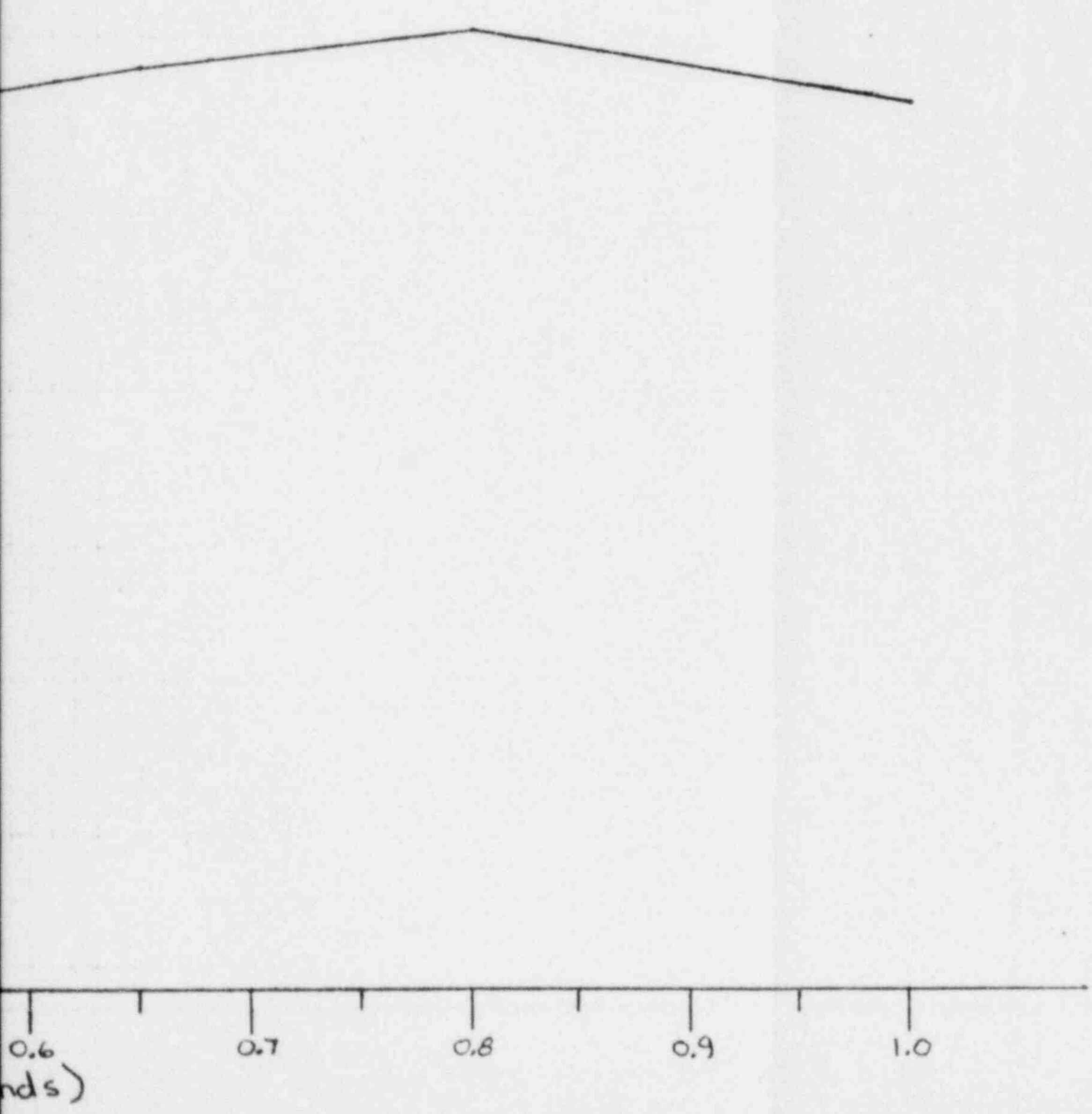
fig E.7


Moment (x 10^3) (IN-KIP)
600
550
500
450
400
350
300
250
200
150
100
50
0
0.1
0.2
0.3
0.4
0.5
Time (Seco

	MADE	GILBERT ASSOCIATES, INC.
	CHK'D.	ENGINEERS AND CONSULTANTS
	SQ. CP.	READING, PENNA.
	CP. OPN.	
	ENG.	WORK ORDER / SIZE / DRAWING / REV.
	REV. CH. APP. DATE	

Perry Nuclear Power Plant
Recirculation Discharge Line Break
Moment @ X1
Bio-Shield Wall

fig E.9



480.43 During a transient that results in overpressurization of the primary system, the safety/relief valves (SRV) will open and steam will be released to the suppression pool. The steam is then condensed in the suppression pool. That condensation process could heat the pool to a temperature when the condensation process becomes unstable. The staff has recently developed criteria which specify the plant-unique information needed to determine if these temperature limits are acceptable. Below is an abstraction from the draft NUREG-0783 which summarizes the various requirements. State how Perry meets the resloution of this issue as specified below.

RESPONSE

A suppression pool temperature monitoring system is utilized in PNPP Unit 1 & 2 Mark III Containment design to provide continuous detection, measurement, and indication of suppression pool water temperature during normal operation & during and after postulated accidents.

Sixteen local temperature sensors are located & installed approx. 20 degrees apart around the periphery of the suppression pool inner liner wall near each SRV quencher. Signals from each temperature sensor transmitter will be inputed into PNPP's "Emergency Response Information System" for determination of Suppression Pool Bulk average temperature. Control Room personnel will also utilize procedures for determination of Bulk average temperature for back-up considerations.

Eight temperature sensors are installed at the 588'6" elevation & eight at the 592'2" elevation. Low Leval alarm of suppression pool water level is set at the 592'10" elevation Low-Low leval alarm of suppression pool water level is at the 591'3" elevation.

All 16 suppression pool water temperature sensors are provided with readout indication and recording in the control room. All channels can also be monitored on the "ERIS" control room CRT. Bulk average temperature of the suppression pool as stated previously will be provided by ERIS with back-up support from CR operating procedures.

The range of the suppression pool temperature instrumentation is 50^{o}F - 258F

The following temperature set points (bulk temperature) require specific operator action using plant operating procedures:

1) 95^{o}F - Suppression pool temperature must be lowered within 24 hours or the reactor plant must be shutdown
2) 110^{o}F - Reactor plant must be shutdown immediately
3) 120^{o}F - The reactor vessel must be depressurized within 12 hours

The suppression pool temperature monitoring system is designed to Seismic Category I, Quality Group B requirements. This system conforms to the guidelines set forth in Reg Guide 1.97 Rev. 2 as clarified by NUREG-0737. Divisional Class IE power is utilized with onsite auxiliary back-up power available to each monitor.

Analyses will be provided to justify the Perry Technical Specification limit on suppression pool temperature in accordance with the criteria defined in draft NUREG-0783, "Suppression Pool Temperature Limits for BWR Containment.

480.44 The results of the annulus pressure response in the secondary containment that are presented in Figures 6.2.58 show that the pressure never goes positive. However, Figure 6.2.57 shows that expansion of the containment steel shell will about one inch. This will cause about a two percent increase in the annulus pressure or an increase of about 6.5 inches of water. Therefore, justify and provide a detailed discussion on the temperature profile within the containment steel shell that was used in the annulus pressure analysis. The discussion should include how the temperature of the containment atmosphere and suppression pool was used and what heat transfer coefficients were used.

RESPONSE

The one inch dimension provided on Figure 6.2-57 is the thickness of the steel shell used in the analysis.

The containment LOCA temperature profile was used as input to the analysis. The containment atmosphere above the suppression pool is conservatively assumed to be the suppression pool temperature. This is conservative since the primary means of atmosphere temperature increase is natural convection. As an example during the first 15 seconds of the LOCA transient approximately $6\,10^{4}$ BTU will be transferred to the containment atmosphere from the pool.

The steel shell temperatures were calculated internally by Contempt and the results show negligible increase in the time period of interest.

The heat transfer coefficients are conservatively assumed to be 2 BTU/hr.-ft^2 - ^{o}F on both sides of the containment steel shell. This is a conservative value based on natural convection and radiation. See FSAR Table 6.2-29 and 6.2-31. The suppression pool is effectively insulated from the annulus by concrete.

480.45 Table 6.2-33 lists the secondary containment penetrations that are considered potential paths through which radioactivity in the primary containment could bypass the leakage collection and filtration system associated with the secondary containment. However, this table does not include all systems that penetrate either the primary or secondary containment boundaries, or both. For those lines that are not considered bypass leakage paths, provide justification for their elimination.

Response

The following notes will be added to Table 6.2-33 to address the containment penetrations not considered potential bypass leakage paths.

7. Closed systems outside containment

   A piping system which penetrates containment and is a closed system outside containment is not a potential bypass leakage path. The redundant containment isolation provisions for each penetration consists of an isolation valve and a closed system outside containment and is in compliance with 10 CFR 50, Appendix A, Criteria 54. The closed system is missile protected, Seismic Category 1, Safety Class 2 and has a temperature and pressure rating in excess of that for the containment. The following penetrations are excluded from consideration as potential bypass leakage paths since they are classified as closed systems outside containment.

   Unit 1:

   P101, P102, P103, P104, P105, P106, P107, P112, P113, P115, P118, P123, P131, P132, P401, P402, P403, P407, P408, P409, P410, P411, P412, P419, P421, P425, P429, P431, P433, P434.

   Unit 2:

   P101, P102, P103, P107, P106, P108, P109, P113, P114, P111, P133, P117, P132, P408, P401, P402, P403, P404, P405, P409, P411, P417, P418, P432, P406, P219, P419, P421, P220, P221.

8. Closed System Inside Containment

A piping system which penetrates containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere is not a potential bypass leakage path. The redundant containment isolation provisions for each penetration consists of an outboard containment isolation valve and a closed system inside containment and is in compliance with 10 CFR 50, Appendix A, Criteria 54. The closed system is Seismic Category 1, Safety Class 2 and has a pressure and temperature rating in excess of that for the containment. The following penetrations are excluded from consideration as potential bypass leakage paths since they are classified as closed systems inside containment.

Unit 1:

P109, P116, P119, P120, P304, P312, P318, P319, P320

Unit 2:

P428, P125, P110, P427, P303, P215, P422, P423

9. Water Seals

A piping system which penetrates containment and is sealed with a sealing system is not a potential bypass leakage path. The sealing system meets the requirements of 10CFR50 Appendix J in that the seal water system inventory is sufficient to assure the sealing function for at least 30 days at a pressure of 1.10 Pa. In addition, the system is missile protected, Seismic Category 1, Safety Class 2 and has a temperature and pressure rating in excess of that for the containment. The following penetrations are excluded from consideration as potential bypass leakage paths since the containment isolation valves are maintained sealed by a sealing system:

Unit 1: P205

Unit 2: P304

10. Leakage Control Systems

The Feedwater and Main Steam Systems have been excluded as a potential bypass leakage path. The main steam system has a dedicated leakage control system (refer to FSAR Section for 6.7) which controls leakage from the isolation valves. The feedwater system has a dedicated leakage control system (refer to new FSAR Section 6.9) which pressurizes the piping between the inboard check valve and outboard gate valve. The leakage control systems meet single failure criteria, are missile protected, Seismic Category 1, Safety Class 2 and has a temperature and pressure rating in excess of that for the containment. The following penetrations fall into the above criteria:

Unit 1: P121, P122, P124, P414, P415, P416

Unit 2: P112, P115, P116, P410, P415, P414

480.46 Section 6.2.4.2.2.1 provides the justification for those lines which penetrate containment and that do not explicitly meet the requirements of GDC 55, 56 and 57. However, it is not possible to relate the justification given in section 6.2.4.2.2.1 to those systems listed in Table 6.2-32 and Figure 6.2.60. Provide a justification and penetration number for each line that does not meet the explicit requirements of GDC 55, 56 and 57.

Response

FSAR Section 6.2.4.2.2 has been revised to note penetration numbers(Unit 1/ Unit 2) and N/A where no containment penetration occurs.

6.2.4.2.2 Justification of Differences from General Design Criteria

The GDCs were not established specifically for BWR plants; rather, these criteria are intended to guide the design of all water cooled nuclear power plants. As a result, the GDCs are generic in nature and subject to a variety of interpretations. For this reason some cases exist where there is no "one-to-one" correspondence between the applicability of an individual GDC and plant design. In such cases, GE has developed a design that meets the intent of the criteria.

The isolation criteria within the GDCs contain clauses, such as, "unless it can be demonstrated ... on some other defined bases", which allows for an alternate design with reliability and performance capabilities that reflect the importance to safety of isolating the piping systems.

Such alternates are described in Sections 6.2.4.2.2.1 through 6.2.4.2.2.3. The final measure by which GE is assured that the BWR design is in agreement with the GDCs is receipt of the Advisory Committee on Reactor Safeguards (ACRS) letters permitting construction and operation of previous plants with comparable valving arrangements.

6.2.4.2.2.1 Justification with Respect to General Design Criterion 55

The reactor coolant pressure boundary, as defined in 10 CFR 50, Section 50.2 (v), consists of the following: reactor pressure vessel; pressure retaining appurtenances attached to the vessel; valves, and pipes which extend from the reactor pressure vessel to, and including, the outermost isolation valve. The lines of the reactor coolant pressure boundary which penetrate containment are capable of isolating the containment, thereby precluding any significant release of radioactivity. Similarly, for lines which do not penetrate containment, but which do comprise a portion of the reactor coolant pressure boundary, the design ensures that isolation of the reactor coolant pressure boundary can be achieved. Items a, b, and c, below, address influent lines, effluent lines, and conclusions with respect to GDC 55, respectively.

a. Influent Lines

Influent lines which penetrate containment and the drywell directly to the reactor coolant pressure boundary are equipped with at least two isolation valves. One valve is inside the drywell; the second, as close as possible to the external side of containment. These isolation valves protect the environment. Where needed, protection of the containment in the event of pipe rupture outside the drywell but within containment is further insured by extension of the drywell by use of guard pipes. These guard pipes, together with the isolation valves, assure protection in the event of an active failure between drywell and containment. Table 6.2-34 lists those influent lines that comprise part of the reactor coolant pressure boundary and penetrate containment. The purpose of this table is to summarize the design of each line with respect to the requirements of GDC 55. Items 1 through 8, below, demonstrate that, although a word for word comparison with GDC 55 is not always practical, it is possible to demonstrate adequate isolation provisions on some other defined basis.

1. Feedwater Lines (P121/P112 & P414/P410)

Feedwater lines are part of the reactor coolant pressure boundary since they penetrate both the containment and drywell and connect to the reactor pressure vessel. Each line includes three isolation valves and is enclosed in a guard pipe.

The isolation valve inside the drywell is a check valve. The other two isolation valves are outside containment. An air operated check valve is located as close as possible to the outside containment wall. The outermost valve is a motor operated gate valve.

Extension of the drywell by means of the guard pipe protects the containment from overpressurization in the event of a feedwater line break between the drywell and containment walls. The internal design temperature and pressure for the guard pipes which enclose the feedwater lines are the same as the design valves specified for the enclosed feedwater lines.

Should a break occur in a feedwater line, the inside check valve prevents significant loss of reactor coolant inventory and provides immediate isolation. The outermost motor operated valve does not close automatically upon occurrence of a protection system signal since, during a LOCA accident, maintenance of reactor coolant makeup from all sources is desirable. This valve, however, can be remotely closed from the control room to provide long term leakage protection when, in the judgement of the operator, continued makeup from the feedwater system is no longer necessary.

2. High Pressure Core Spray Line (P410/P411)

The high pressure core spray line penetrates both the containment and the drywell and connects to the reactor pressure vessel. Isolation is provided by an air testable check valve inside the drywell and a motor operated block valve as close as possible to the outside of the containment wall. This block valve maintains long term leakage control. Position indication for the air testable check valve is provided in the control room. The block valve is automatically and remote manually operated. A guard pipe is not necessary since influent high pressure core spray fluid is at such a low energy level during system operation that containment overpressurization cannot result should the line break between the containment and the drywell.

3. Low Pressure Core Spray and Low Pressure Coolant Injection Lines (P112/P113, P113/P114, P412/P418, & P411/P417)

Isolation of the low pressure core spray and low pressure coolant injection system lines is accomplished by use of an air testable check valve and a motor operated block valve. The check valve is located as close as possible to the reactor vessel and is normally closed. This valve protects against containment overpressurization in the event of a line break between the check valve and the containment wall by preventing high energy reactor coolant from entering containment. The block valve located outside containment is automatically and remote manually operated and is also normally closed. The block valve is

automatically opened at the appropriate time to assure that acceptable fuel design limits are not exceeded during LOCA. A guard pipe is not necessary since system fluid energy during operation is sufficiently low to preclude the possibility of containment overpressurization should a break occur.

4. Control Rod Drive System Lines (204/302)

The control rod drive system, located between the reactor pressure vessel and containment, includes two types of influent lines: the supply line that penetrates containment; and the insert and withdraw lines that penetrate the drywell.

Isolation of the supply line is accomplished by a check valve inside containment and a remote manually actuated motor operated block valve as close as possible to the outside of the containment wall.

The insert and withdraw lines are not part of the reactor coolant pressure boundary since these lines do not communicate directly to reactor coolant. The basis upon which these lines are designed is commensurate with the importance to safety of maintaining the pressure integrity of these lines. The classification of these lines is Quality Group B and they are designed in accordance with the ASME Code, Section III, Class 2.

In the design of the control rod drive system, it has been accepted practice to omit automatic valves for isolation purposes since inclusion of such a valve would introduce a possible failure mechanism into the shutdown (scram) function. Manual shutoff valves are provided for isolation. In the event of a break in these lines, the manual valves provide isolation capability. In addition, a ball check valve in the control rod drive flange housing automatically seals the insert line in the event of a break. Containment overpressurization will not result from a line break in contaiment since these lines contain small volumes of fluids, resulting in relatively small blowdown masses.

As shown in Figure 6.2-60, arrangement nos. 3 and 4, the recirculation pump seal water supply line is connected to the control rod drive system downstream of the inboard containment isolation valve.

5. Residual Heat Removal Head Spray and Reactor Core Isolation Cooling Lines (P123/P117)

The residual heat removal head spray and reactor core isolation cooling lines join outside containment to form a common line which penetrates both the containment and the drywell and connects to the reactor pressure vessel. An air testable check valve is provided inside the drywell as close as possible to the reactor pressure vessel. Two valves, a check valve and a remote manually actuated, motor operated block valve, are located outside containment in each line. The line is also enclosed in a guard pipe.

The air testable check valve inside the drywell is normally closed. The check valves outside containment assure immediate containment isolation in the event of a line break. The block valve in each line is manually actuated to provide long term leakage control.

The guard pipe provides protection against containment overpressurization in the event of a line break between the drywell and containment walls. Should the check valve inside the drywell fail coincident with a line break, the guard pipe would direct the released fluid into the drywell.

Position indication lights are provided in the control room for each of the air testable check valves inside the drywell.

6. Standby Liquid Control Line (N/A)

The standby liquid control system is located between the containment and the drywell. The standby liquid control line penetrates the drywell and connects to the reactor pressure vessel. Isolation is provided by a check valve inside the drywell and a check valve and explosive valve outside the drywell. The explosive valve provides an absolute seal for long term leakage control, as well as preventing leakage of sodium pentaborate into the reactor pressure vessel during normal reactor operation. Since the standby liquid control line is

normally an isolated, nonflowing line, rupture is extremely improbable. However, should a break occur subsequent to actuation of the explosive valve, the check valves ensure isolation.

7. Residual Heat Removal Shutdown Cooling Return Lines (P121/P112 & P414/P410)

The residual heat removal shutdown cooling return lines discharge into the feedwater line between the air operated check valve and the motor operated block valve outside of containment. A check valve and a normally closed, motor operated, remote manually actuated gate valve provide for isolation of the residual heat removal shutdown cooling return lines.

8. Reactor Water Cleanup System Line (P419/P432)

The discharge line from the reactor water cleanup pumps penetrates containment and serves the reactor water cleanup regenerative heat exchangers inside containment. Automatically actuated motor operated block valves, one inside, one outside containment, provide for isolation.

b. Effluent Lines

Effluent lines that form part of the reactor coolant pressure boundary and penetrate containment and/or the drywell are equipped with at least two isolation valves. One valve is inside the drywell, the other outside, but as close as possible to, the containment. Where needed, the containment is protected, in the event of a pipe rupture outside of the drywell but inside containment, by guard pipes which enclose the process lines, forming an extension of the drywell. This combination of isolation valves and guard pipes assures protection in the event of a failure between drywell and containment walls.

Table 6.2-35 lists those effluent lines that comprise part of the reactor coolant pressure boundary and that penetrate containment and/or the drywell. Items 1 through 4, below, address specifics of these lines.

1. Steam Lines (P124/P116, P416/P414, P122/P115, and P415/P415)

Steam lines include main steam, main steam drain, residual heat removal, and reactor core isolation cooling steam lines.

The main steam lines from the reactor pressure vessel to the turbine penetrate both drywell and containment. Main steam line drains (one for each main steam line) in the drywell are headered together to form one line which penetrates both drywell and containment. Isolation for the main steam lines and main steam drain line is provided by automatically actuated block valves, one inside the drywell and one outside containment.

The residual heat removal steam supply and reactor core isolation cooling turbine steam line branches from the main steam line inside the drywell. Isolation for this line is provided by normally open, remote, manually actuated, motor operated block valves, one inside the drywell, one outside containment.

Use of guard pipes to enclose these steam lines prevents containment overpressurization in the event of line break between the drywell and containment walls. The internal design temperature and pressure for the guard pipes which enclose these steam lines are the same as the design values specified for the enclosed lines.

2. Reactor Water Cleanup Lines (P131/P132)

The reactor water cleanup pumps are located outside containment; the heat exchangers and filter demineralizers, inside containment, but outside the drywell. The reactor water cleanup pump suction line from the reactor recirculation system lines and the reactor bottom head penetrates the drywell and containment. Two automatically actuated, motor operated valves provide for isolation of this line. One valve is just inside the drywell; the other, outside containment. A guard pipe encloses the line between the drywell and containment walls.

The reactor water cleanup pump discharge line to the heat exchangers and filter demineralizers penetrates containment. Two automatically actuated, motor operated valves (one inside and one outside containment) provide for isolation of this line.

A blowdown line from the filter demineralizers penetrates containment and divides to form separate lines to the condenser and radwaste system. Automatically actuated, motor operated block valves, one inside and one outside containment, provide for isolation of this line.

The return line from the filter demineralizers penetrates containment and connects to the feedwater line between the containment wall and the air operated (feedwater) check valve. Two automatically, actuated, motor operated block valves provide for isolation of this line. One valve is inside, the other outside of containment.

3. Residual Heat Removal Shutdown Cooling Line (P421/P406)

The residual heat removal shutdown cooling line branches from the B reactor recirculation loop and penetrates both the drywell and containment. Normally closed, remote manually actuated, motor operated valves, one inside the drywell, one outside containment, provide for isolation of this line. A guard pipe encloses this line from the drywell wall to the containment wall to protect against containment overpressurization in the event of a line break.

4. Recirculation System Sample Line (N/A)

A sample line from the recirculation system penetrates the drywell. This line is 3/4 inches in diameter and is designed in accordance with the requirements of the ASME Code, Section III, Class 2. A sample probe with a 1/8 inch diameter hole is located inside one recirculation discharge line within the drywell. In the event of a line break, this probe acts as a restricting orifice and limits escaping fluid flow. Two air operated valves which fail closed are provided for isolation of this line. One valve is inside drywell ; the other, outside.

c. Conclusions Concerning General Design Criterion 55

To assure protection against the consequences of accidents involving the release of radioactive material, piping which forms portions of the reactor coolant pressure boundary has been shown to provide adequate isolation capability on a case by case basis. In all cases, a minimum of two barriers is shown to protect against release of radioactive materials. Where necessary to protect the containment against overpressure, guard pipes are provided which enclose the process pipes between the drywell and containment walls.

In addition to satisfying the requirements of GDC 55, the pressure retaining components which comprise the reactor coolant pressure boundary are designed to satisfy other appropriate requirements which minimize the probability or consequences of an accident rupture. Quality requirements for these components ensure that they are designed, fabricated, and tested to the highest reactor plant component standards. The classification of components which comprise the reactor coolant pressure boundary is Quality Group A and such components are designed in accordance with the ASME Code, Section III, Class 1. Additional information concerning classification is presented by Table 3.2-1. The containment and reactor vessel isolation control system is addressed in Section 7.3.

6.2.4.2.2.2 Justification with Respect to General Design Criterion 56

GDC 56 requires that lines that penetrate containment and communicate with the containment interior must have two isolation valves, one valve inside containment, the other outside, unless it can be demonstrated that the containment isolation provisions for a specific class of lines are acceptable on some other basis.

Table 6.2-36 lists those lines that penetrate primary containment and connect to the drywell and suppression chamber. The purpose of this table is to summarize the design of each listed line with respect to the requirements of GDC 56. Although a word for word comparison with GDC 56 is, in some cases, not practical,

it is possible to demonstrate adequate isolation provisions on some other defined basis. It should be noted that this criterion does not reflect consideration of the BWR suppression pool design, in that those lines which connect to the suppression pool would require placement of inside containment isolation valve underwater. All of the lines which connect to the suppression pool are to or from the individual watertight ECCS pump rooms. Items a, b, c, and d, below, address influent lines to the suppression pool, effluent lines from the suppression pool, influent and effluent lines from the drywell and suppression pool free volume, and conclusions with respect to GDC 56, respectively.

a. Influent Lines to the Suppression Pool (P105/P106 and P409/P409 and P407/P404 and P408/P405)

1. Low Pressure Core Spray, High Pressure Core Spray, and Residual Heat Removal Test and Pump Minimum Flow Bypass Lines, and Residual Heat Removal Steam Condensing Mode Bypass Line

The low pressure core spray, high pressure core spray and residual heat removal test lines have isolation capability commensurate with the importance to safety of isolating these lines. Each line has a normally closed, motor operated valve located outside containment. Containment isolation requirements are satisfied on the basis that the test lines are normally closed, low pressure lines, constructed to the same quality standards as the containment. Furthermore, the consequences of a break in one of these lines result in no significant effect on safety. All of these lines terminate below the minimum suppression pool drawdown level.

The test return lines are also used for suppression pool return flow during other modes of operation. This reduces the number of penetrations, minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the test return lines downstream of the test return isolation valve. The bypass lines are isolated by motor operated valves and a restricting orifice is provided downstream of the valves.

2. Reactor Core Isolation Cooling Pump Minimum Flow Bypass, Turbine Exhaust, and Turbine Exhaust Vacuum Relief (P106/P108 & P123/P117)

The reactor core isolation cooling pump minimum flow bypass, turbine exhaust, and turbine exhaust vacuum relief lines penetrate containment and discharge to the suppression pool. The minimum flow bypass and turbine exhaust lines are each equipped with a motor operated, remote manually actuated gate valve outside and as close to containment as possible. A check valve upstream of the gate valve provides for immediate isolation in the event of a break upstream of the check valve. The turbine exhaust vacuum relief line has two automatic and remote manually actuated isolation valves located outside containment. The motor operated gate valve in the minimum flow bypass line is normally closed. The turbine exhaust line motor operated gate valve is designed to be locked open in the control room and is interlocked to preclude opening of the reactor core isolation cooling pump turbine steam inlet valve if the turbine exhaust valve is not fully open. The turbine exhaust vacuum relief line valves are normally open.

3. Residual Heat Removal Heat Exchanger Vent and Relief Valve Discharge Lines (P107/P109 & P118/P133 & P429/P419 & P431/P421)

Residual heat removal heat exchanger vent lines discharge to the suppression chamber. Two normally closed, remote manually actuated, motor operated valves outside containment and a check valve, located between containment and the drywell, provide isolation.

Relief valve discharge lines from the residual heat removal heat exchangers and various emergency core cooling system suction and discharge lines discharge to the suppression pool. These vent lines are isolated by the relief valves. The addition of block valves would defeat the purpose of the relief valves. The relief valves set pressure is greater than 1.5 times containment design pressure.

b. Effluent Lines from the Suppression Pool (P103/P103 & P401/P401 & P101/P101 & P102/P102 & P402/P402 & P403/P403)

The low pressure core spray, high pressure core spray, reactor core isolation cooling, and residual heat removal suction lines are equipped with remote manually actuated, motor operated gate valves outside containment. These valves provide the ability to isolate in the event of a line break and also provide long term leakage control. The high pressure core spray and reactor core isolation cooling pump suction lines also include check valves.

In addition, suction piping from the suppression pool is considered an extension of containment since this piping must be available for long term use following a design basis LOCA. Therefore, this piping is designed to the same quality standards as the containment. Thus, the need for isolation is obviated to some degree by providing a high-quality system and by the fact that the piping runs to the water-tight ECCS pump rooms. Also, the emergency core cooling system discharge line fill system (emergency core cooling system waterleg pumps) takes suction from the respective emergency core cooling system pump effluent line from the suppression pool downstream of the isolation valve. The emergency core cooling system discharge line fill system suction line includes a manual valve, provided for operational purposes. This system is isolated from containment by the respective emergency core cooling system pump suction valve from the suppression pool (see Table 6.2-32).

Also, each ECCS pump room is provided with leak detection capabilities as discussed in Section 9.3.3. If leakage from a seal or gasket is detected in one of the pump rooms during normal plant conditions, the remotely operated valve installed in the pump suction line would be closed, thereby isolating the leaking component from the suppression pool water. No seals or gaskets are installed between the containment penetration and the isolation valve. The only potential path for leakage of suppression pool water into the ECCS pump rooms is through the pumps' suction lines since these are the only lines penetrating the containment at an elevation below the suppression pool water level.

Therefore, the need to size the ECCS pump rooms so that the volume of suppression pool water needed to fill the ECCS pump room would not reduce the suppression pool level below the minimum drawline is not required due to the leak detection and isolation capabilities incorporated in the design. The potential reduction in suppression pool water inventory before detection and isolation of a leaking seal or gasket in the pump room would be insignificant. Suppression pool makeup water during normal plant conditions is from the condensate water storage tank.

c. Influent and Effluent Lines from Drywell and Suppression Pool Free Volume

1. Combustible Gas Control and Post LOCA Atmosphere Sampling Lines (P302/P211 & P318/P422, P423 & P425/P219)

The combustible gas control system backup purge line which penetrates containment includes two normally closed, remote manually actuated valves, one inside and one outside containment. The post LOCA sampling system lines which penetrate containment and connect to the drywell and suppression chamber air volume are equipped with two normally closed, solenoid operated isolation valves in series. These valves are located outside containment and provide assurance of isolation of these lines.

The piping outside containment is a closed loop is considered an extension of containment since it must be available for long term use following a design basis LOCA. Therefore, it is designed to the same quality standards as containment.

2. Containment Purge and Exhaust Lines (V313/V216 & V314/V214)

The containment purge and exhaust lines are equipped with three automatically actuated isolation valves. One valve is outside containment and one valve is in each of the two branch lines inside containment.

d. Conclusions Concerning General Design Criterion 56

To assure protection against the consequences of accidents involving the release of significant amounts of radioactive material, piping that penetrates containment has been shown to provide adequate isolation capability on a case by case basis in accordance with GDC 56.

In addition to satisfying the isolation requirements specified by GDC 56, the pressure retaining components of these systems are designed, fabricated, and tested in accordance with the requirements of the ASME Code, Section III. In some cases, provision of a high quality system obviates the need for isolation valves due to the diminished probability of a rupture in such a system. Additional information concerning classification is presented by Table 3.2-1. The containment and reactor vessel isolation control system is addressed in Section 7.3.

6.2.4.2.2.3 Justification with Respect to General Design Criterion 57

Lines that penetrate containment and for which neither GDC 55 nor GDC 56 are governing comprise the closed system isolation valve group. Influent lines are equipped with a motor operated valve outside containment and a check valve inside containment. Effluent lines are equipped with motor operated valves both outside

and inside containment. In all cases, the isolation valves are located as close to containment wall as possible. System lines within this group include the following:

a. Nuclear closed cooling water supply and return. (P310/P201 & P311/P202)
b. Fuel pool cooling influent and effluent.(P203/P301 & P301/P222)
c. Condensate makeup influent and effluent. (P108/P424 & P111/P426)
d. Containment chilled water supply and return. (P404/P104 & P405/P105)
e. Plant service air. (P308/P203)
f. Instrument air. (P305/P205 & P306/P204 & P312/P215)
g. Demineralized water. (P309/P207)
h. Equipment drain sump to radwaste. (P417/P128)
i. Fire protection carbon dioxide. (P206/P206)
j. Floor drain effluent to radwaste. (P418/P127)

k. Backwash receiving tank to radwaste. (P420/P412)
l. Nitrogen supply. (P117/P123)
m. Safety related instrument air. (P116/P125 & P304/P303)

6.2.4.2.3 Consideration of NRC Branch Technical Position CSB 6-4, "Containment Purging during Normal Operation"

The containment purge system is designed to achieve the objectives stated in Branch Technical Position CSB 6-4. Purge system containment isolation valves are capable of isolating containment within 5 seconds. The containment purge system is described in Section 6.5.1.

Radiological consequences due to the occurrence of a postulated LOCA when the containment is being purged during normal operation have been examined to determine compliance with the dose criteria set forth in Branch Technical Position CSB 6-4. The calculated site boundary doses are 0.45 Rem to the thyroid and 81 mRem whole body. These doses are a small fraction of the 10 CFR 100 guideline values.

480.48 With regard to your containment leakage testing program, we will require that:

a) all isolation valves listed in Table 6.2-40 be Type C tested;

b) the feedwater lines (items 9 and 10) be vented and drained for Type C test, tested with air and leakage included in 0.60 La; and

c) HPCS pump discharge to RPV (item 32), and LPCS pump discharge to RPV (item 35) be tested with air and leakage included in 0.60 La.

Response

a) All containment isolation valves listed in Table 6.2-40 will be type C tested, except the isolation valves for instrument lines which penetrate the containment and conform to the requirements at Regulatory Guide 1.11. Isolation valves, pressurized by a water seal system, will be Type C tested with water and the leakage excluded in combined leakage rate, consistent with the Type C test acceptance criteria in 10CFR50, Appendix J.

Examples of such lines are discussed in b) and c) below and lines that terminate below the water level of the suppression pool. Sufficient pool inventory is available to maintain a 30 day pressure at 1.10Pa. The piping up to the isolation valve is seismic Category I, safety class 2. Missile and pipe whip are not concerns for this piping.

b) The feedwater lines are Type C tested with water and the leakage is not included in the 0.60 La. This is consistent with 10CFR50 Appendix J acceptance criteria since a dedicated Feedwater leakage control system is provided. Refer to response to 480.45 (NOTE 10 to Table 6.2-33) and new FSAR Section 6.9

c) HPCS and LPCS pump discharge lines to the reactor vessel are Type C tested with water and the leakage is not included in 0.60La, consistent with 10CFR50, Appendix J acceptance criteria. The system lines are maintained filled with water by a discharge line fill system as well as being a closed system outside containment. Refer to response to 480.45 (note 7 to table 6.2-33 and FSAR Section 6.3.2.2.5). The fill pumps and all associated equipment are Seismic Category I and Quality Group B. Post-LOCA, the Systems' main pumps will be operating to maintain a water seal . Following manual termination of the main pump the discharge line fill pumps will maintain a water seal in the lines.