SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO. 1

SUPPLEMENT NO. 1
PRELIMINARY SAFETY ANALYSIS REPORT

0230 DOCKET 50-312

~~00146~~ DECEMBER 1968

8004090 703 B

BEFORE THE UNITED STATES ATOMIC ENERGY COMMISSION

~~Regulatory Formal File Cy.~~

In the Matter of	)	Docket No. 50-312
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	
(Rancho Seco)	)	Supplement No. 1

Now comes SACRAMENTO MUNICIPAL UTILITY DISTRICT (SMUD) and amends its above-numbered application by submitting herewith Supplement No. 1 This supplement reflects an increase in Reactor Building diameter, revised Reactor Building design criteria, new fuel handling method and revised structural design bases.

Subscribed in Sacramento, California, this 15th day of December 1968.

DOCKETED USAEC DEC 18 1968 REGULATORY MAIL SECTION DOCKET CLERK

Respectfully submitted,

SACRAMENTO MUNICIPAL UTILITY DISTRICT

By Paul E. Shaad
Paul E. Shaad
General Manager and Chief Engineer

E. K. DAVIS
DAVID S. KAPLAN
Attorneys for Sacramento
Municipal Utility District

By David S. Kaplan
David S. Kaplan

Subscribed and sworn to before me this 15th day of December 1968.

Betty Mattier, Notary Public in and for the County of Sacramento, State of California.

BETTY MATTIER, NOTARY PUBLIC SACRAMENTO CO., CALIF. MY COMMISSION EXPIRES JAN. 12, 1972

My Commission expires January 12, 1972.

DEC 19 1968 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

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~~00147~~

Docket No. 50-312
December 15, 1968

SUPPLEMENT NO. 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

Supplement No. 1 to Sacramento Municipal Utility District's Preliminary Safety Analysis Report reflects an increase in Reactor Building diameter from 116 feet to 130 feet, revised Reactor Building design criteria, new fuel handling methods and structural design bases. The Reactor Building pressure transient analysis will remain approximately the same since the Reactor Building height is reduced to 185 feet thus maintaining essentially the same net volume as reported in the PSAR.

For clarity and completeness supplement No. 1 includes complete sections of those areas modified as presented in the PSAR.

~~00149~~

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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This Preliminary Safety Analysis Report is submitted in support of Sacramento Municipal Utility District's (hereinafter referred to as SMUD or the District) application for a construction permit and facility license for one nuclear unit (designated Unit 1) at its Rancho Seco site in the Southeast portion of Sacramento County, California. The District's Service Area is shown on Figure 1.1-1. The plant location is shown on Figure 1.1-2.

The generating unit will operate initially at core power levels up to 2452 Mwt. All physics and core thermal hydraulics information in this report are based on the reference core design of 2452 Mwt. It is expected that the nuclear supply system will be capable of an ultimate output of 2584 Mwt, (including 16 Mwt contribution from reactor coolant pumps). All power conversion systems will be designed to accommodate the ultimate unit output. Site parameters, principal structures, engineered safeguards, and certain hypothetical accidents are evaluated for the expected ultimate core output of 2568 Mwt.

The nuclear steam supply system is a pressurized water reactor type, similar to systems now operating or under construction. It uses chemical shim and control rods for reactivity control and generates steam with a small amount of superheat in once-through steam generators. The nuclear steam supply system and fabrication of the first two cores will be supplied by the Babcock & Wilcox Company.

Construction is scheduled for completion in time for fuel loading to begin by December 1, 1972, and for commercial operation by May 1, 1973. To meet this schedule, construction is to begin by January 1, 1969.

The station plot plan and general arrangement of major equipment and structures, including the reactor building, auxiliary building, and turbine structure is shown on Figures 1.1-3 through 1.1-8.

The organization of this report follows as closely as possible the AEC's guide announced in the Federal Register on August 16, 1966; and the technical contents are organized in conformance with the AEC 70 General Design Criteria announced in the Federal Register dated July 11, 1967. Every attempt has been made in this report to be completely responsive to the guide and to all known pertinent questions asked of other applicants up until the time of this writing.

As the plant design progresses from conceptual design to final detailed design, the plant description and analyses will be subject to change and refinement. This report presents descriptive material and analyses of a "reference-design." Any significant changes to the criteria or designs

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which affect safety will be promptly brought to the attention of the AEC by revised insert pages, and additional information will be submitted by suitable supplements.

SMUD is fully responsible for the complete safety and adequacy of the plant. Aid in the design, construction, management, testing, and start-up of the unit will be supplied principally by Bechtel Corporation and the Babcock & Wilcox Company (B&W). Assistance will also be rendered by other consultants and suppliers as may be required.

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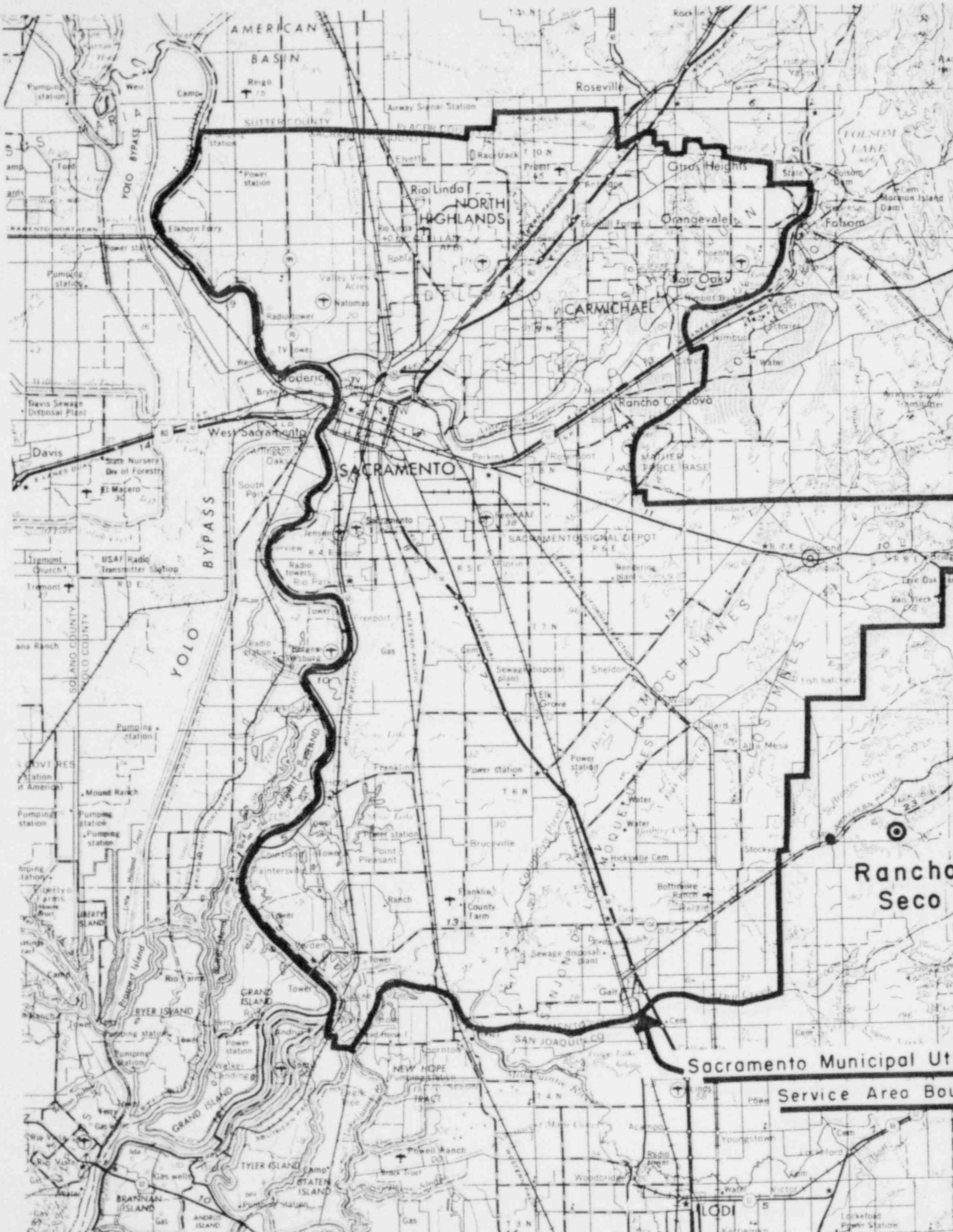

Sacramento Municipal Uti
Service Area Bou
Rancho
Seco
SACRAMENTO
AMERICAN
BASIN
Roseville
Citrus Heights
Rio Linda
NORTH
HIGHLANDS
Orangevale
Folsom
FOLSOM
LAKE
Fair Oaks
CARMICHAEL
Rancho Cordova
MATHER
FORCE BASE
West Sacramento
Davis
YOLO BYPASS
YOLO
BYPASS
SACRAMENTO SIGNAL DEPOT
Florin
Freeport
Elk
Grove
Sheldon
Franklin
Bruceville
Point
Pleasant
Galt
SAN JOAQUIN CO
Thornton
NEW HOPE
TRACT
GRAND
ISLAND
RYER ISLAND
TYLER ISLAND
STATEN
ISLAND
BRANNAN
ISLAND
Rio Vista
Woodbridge
LODI
Lockeford
Clay
Herald
Victor
Acampo
Youngstown
Courtland
Walnut Grove
Isleton
Broderick
Bryte
Natomas
Elverta
Antelope
Roseville
Tremont
El Macero
Sloughhouse
Clay
Dillard
Alta Mesa
Hicksville Cem
Sewage disposal plant
COSUMNES
MOKELUMNE

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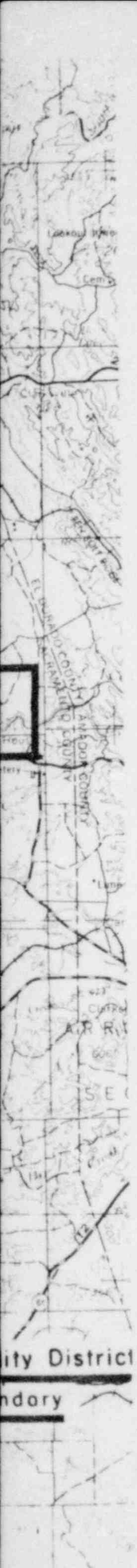

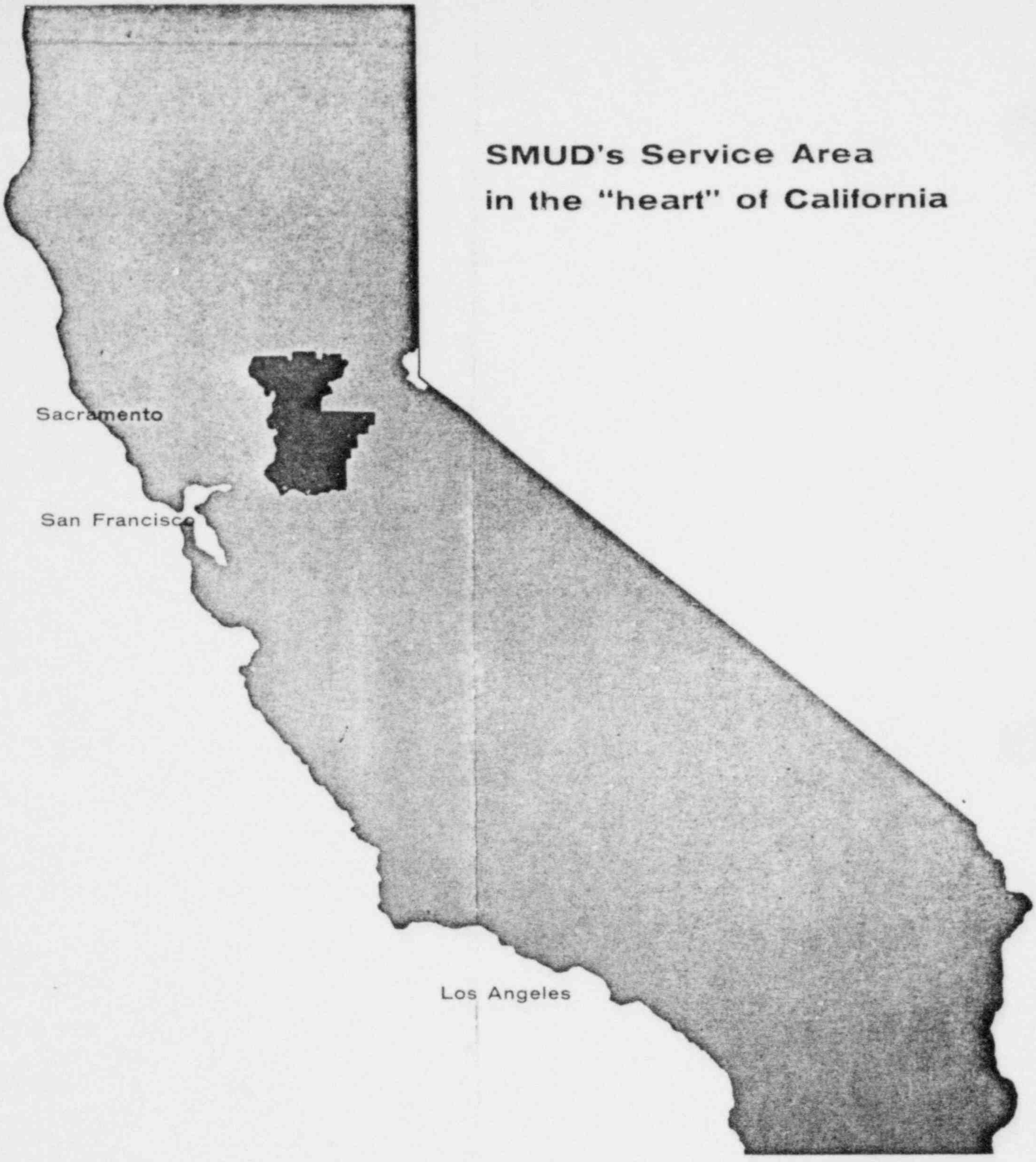

FIGURE 1.1-1
SACRAMENTO MUNICIPAL UTILITY DISTRICT
SERVICE AREA BOUNDARY

SMUD ~~00152~~
SACRAMENTO MUNICIPAL UTILITY DISTRICT

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WAREHOUSE & SHOPS
INTAKE
REACTOR
SWITCHYARD
UNIT 1
FUTURE UNIT 2
ADMIN
NUCLEAR SERVICE SPRAY POND
NUCLEAR SERVICE SPRAY POND
FENCE
GRADE EL. 164.0'
SERVICE ROAD
INTAKE CANAL

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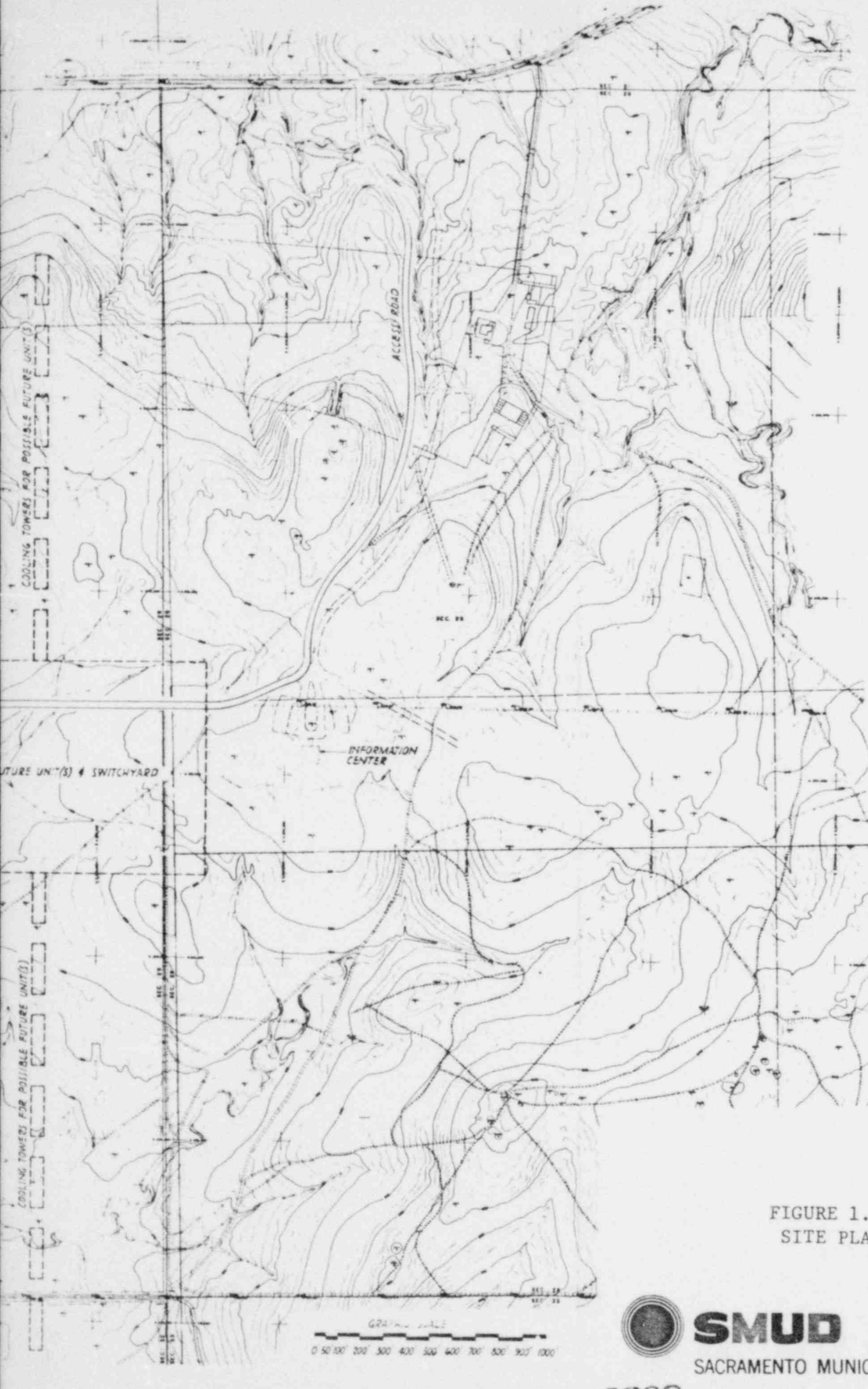

FIGURE 1.1-2
SITE PLAN

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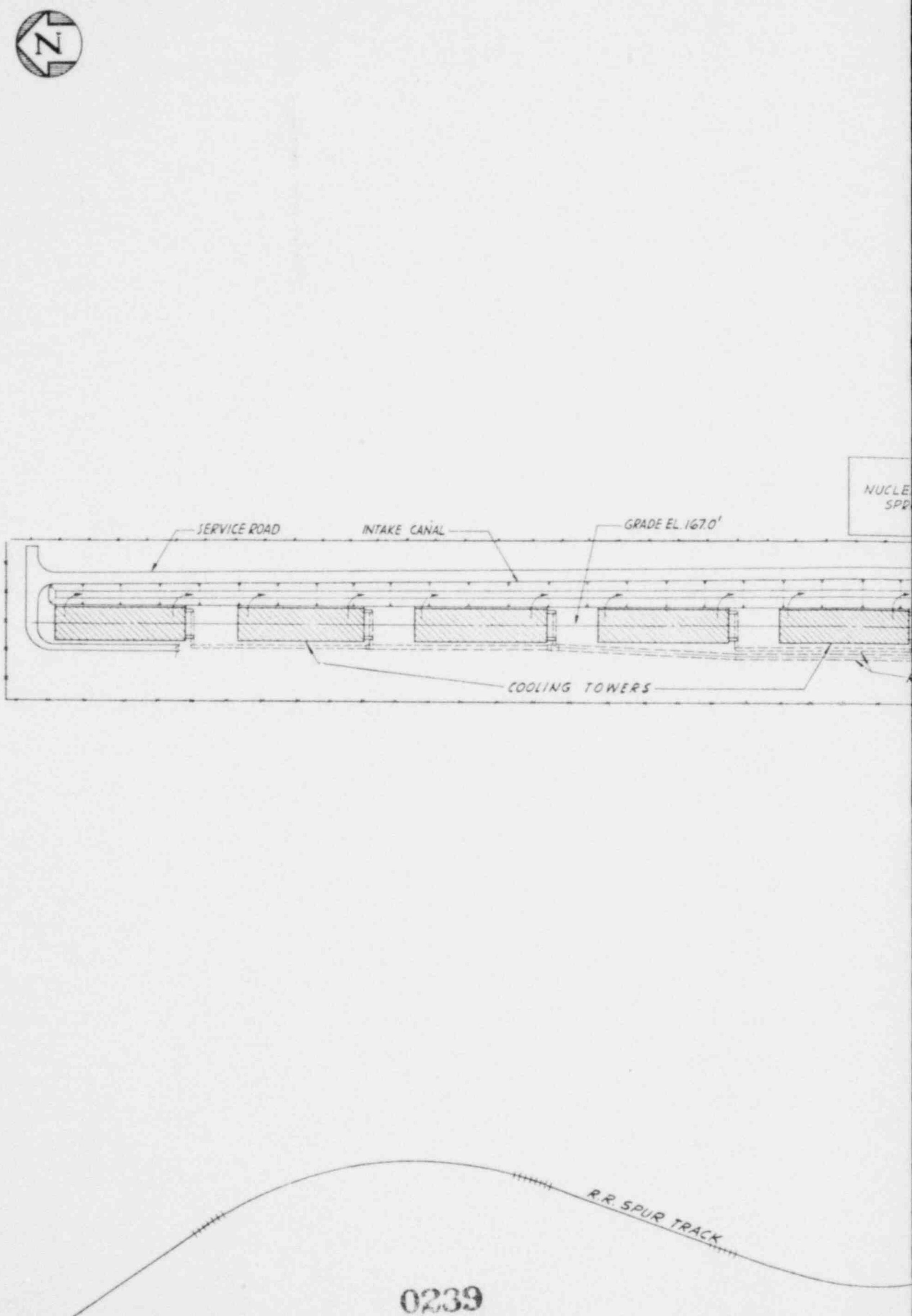
N
NUCLE
SPR
SERVICE ROAD
INTAKE CANAL
GRADE EL. 167.0'
COOLING TOWERS
R.R. SPUR TRACK

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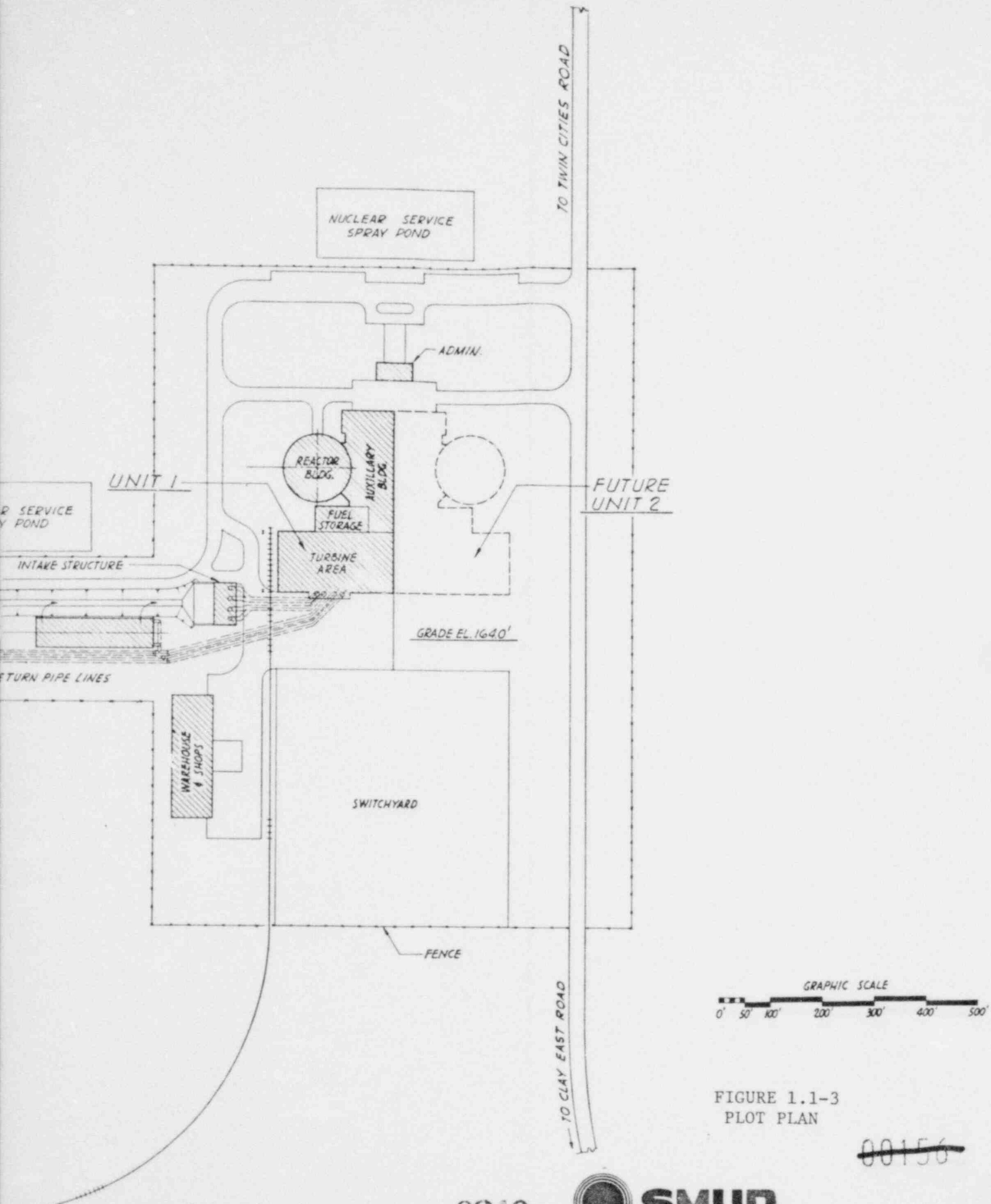

FIGURE 1.1-3
PLOT PLAN

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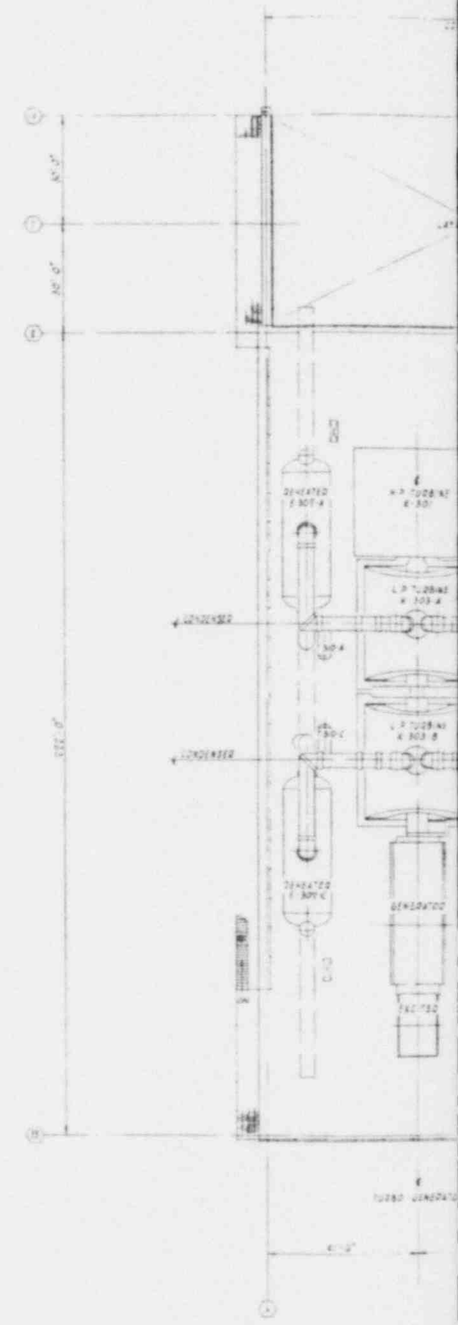

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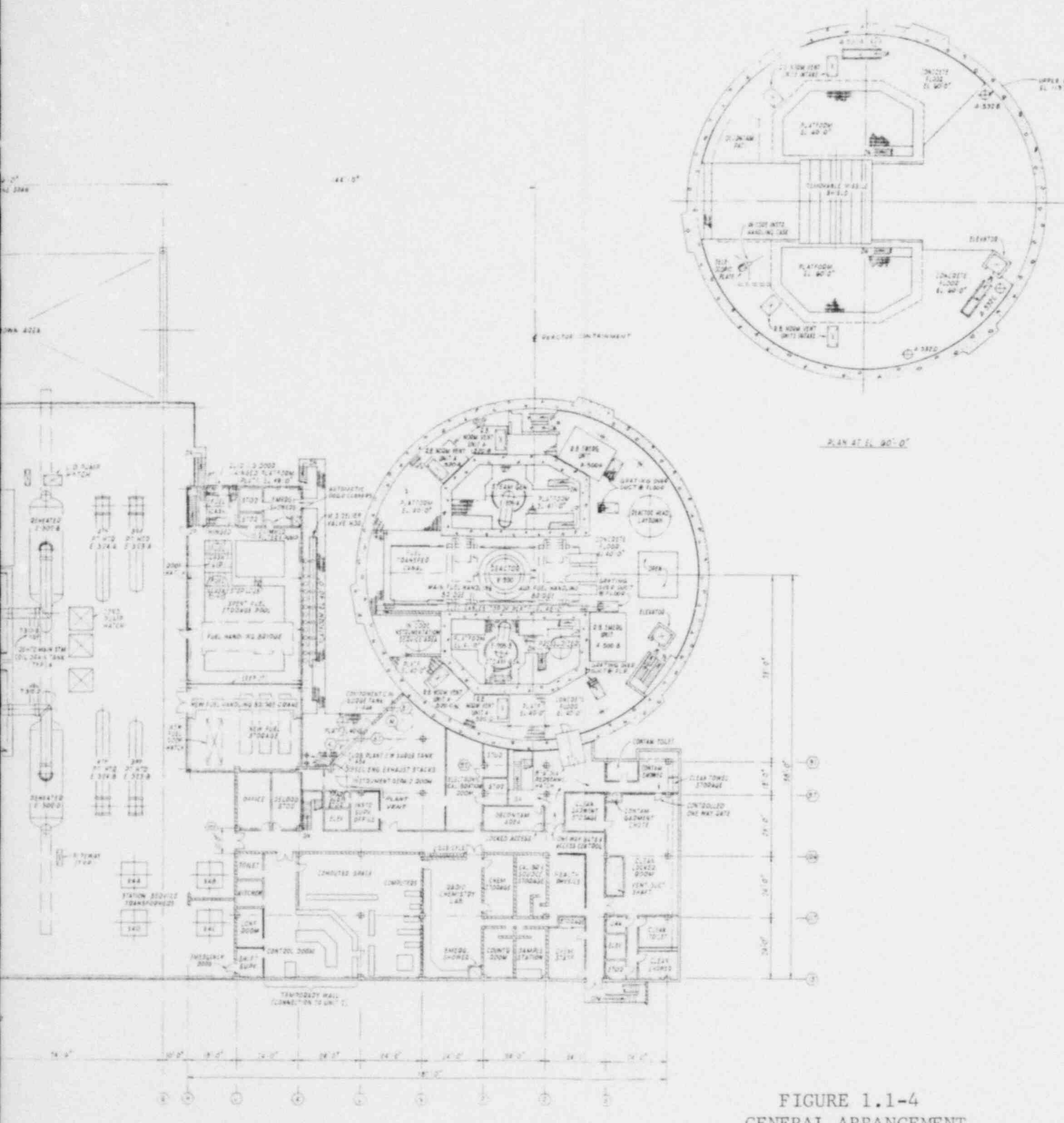

FIGURE 1.1-4
GENERAL ARRANGEMENT
PLAN AT TURBINE DECK

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

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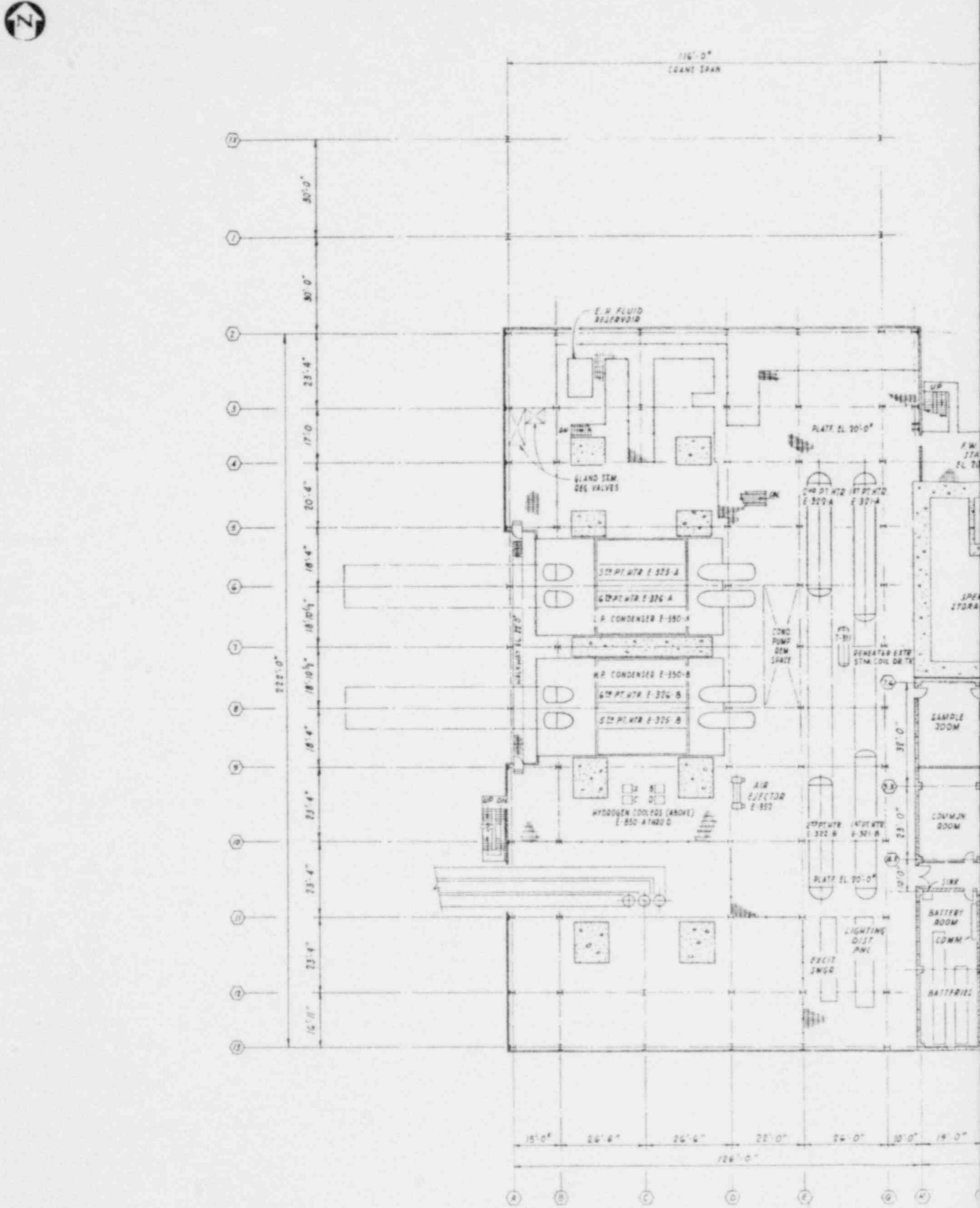

116'-0"
CRANE SPAN
E.H. FLUID RESERVOIR
GLAND STM. REG. VALVES
PLATF. EL. 20'-0"
2ND PT. HTR E-322-A
1ST PT. HTR E-321-A
5TH PT. HTR E-325-A
6TH PT. HTR E-326-A
L.P. CONDENSER E-350-A
H.P. CONDENSER E-350-B
6TH PT. HTR E-326-B
5TH PT. HTR E-325-B
COND. PUMP REM. SPACE
REHEATER EXTR. STM. COIL DR. TK
SAMPLE ROOM
COMMUN. ROOM
BATTERY ROOM
COMM.
BATTERIES
SINK
AIR EJECTOR E-352
HYDROGEN COOLERS (ABOVE) E-350 A THRU D
2ND PT. HTR E-322-B
1ST PT. HTR E-321-B
PLATF. EL. 20'-0"
LIGHTING DIST. PNL
EXCIT. SWGR.
WALKWAY EL. 22'-0"
UP
DN
222'-0"
30'-0"
30'-0"
23'-4"
17'-0"
20'-4"
18'-4"
18'-10½"
18'-10½"
18'-4"
23'-4"
23'-4"
23'-4"
16'-11"
38'-0"
23'-0"
10'-0"
15'-0"
26'-6"
26'-6"
22'-0"
24'-0"
10'-0"
19'-0"
126'-0"

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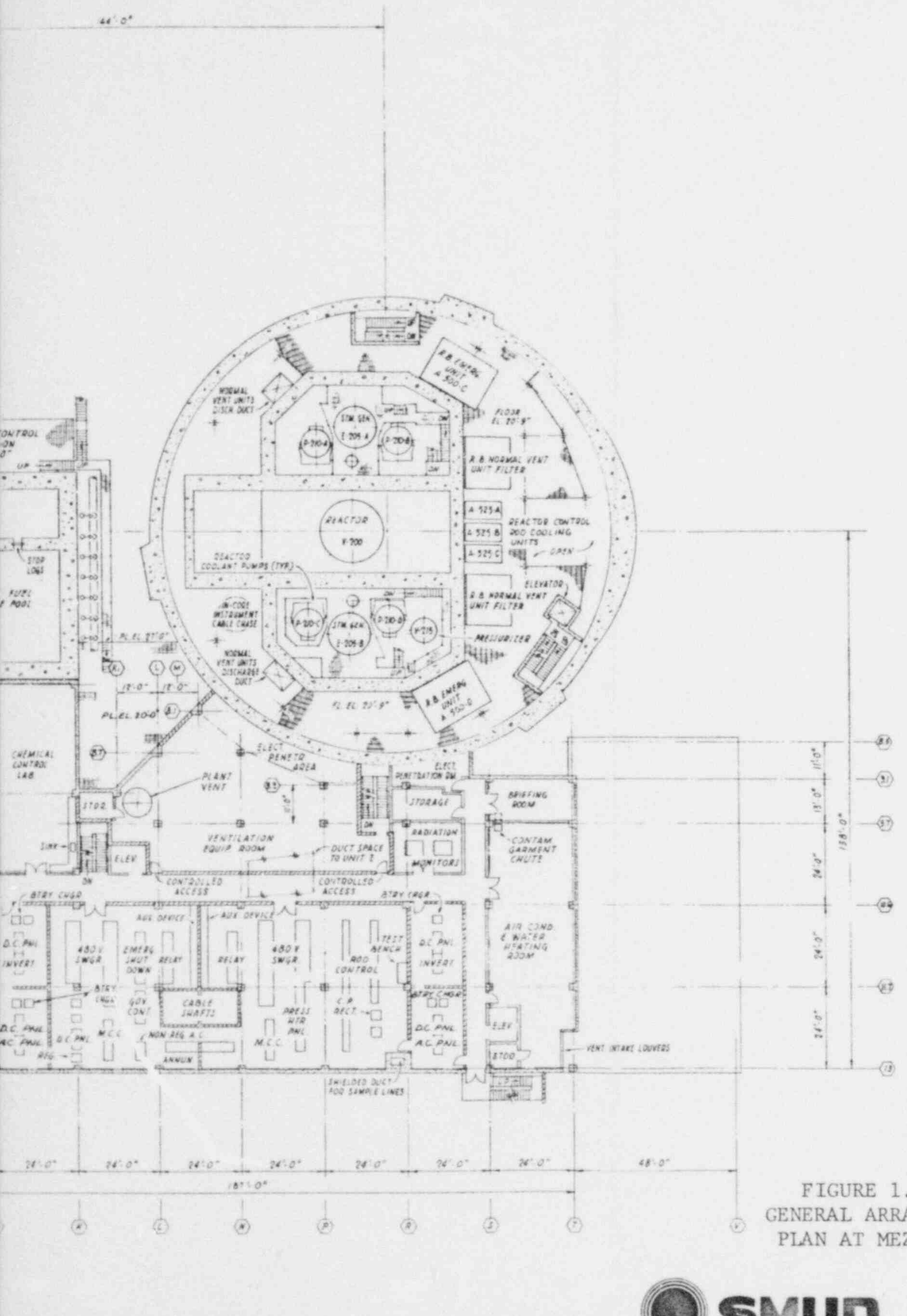

FIGURE 1.1-5
GENERAL ARRANGEMENT
PLAN AT MEZZANINE

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SACRAMENTO MUNICIPAL UTILITY DISTRICT

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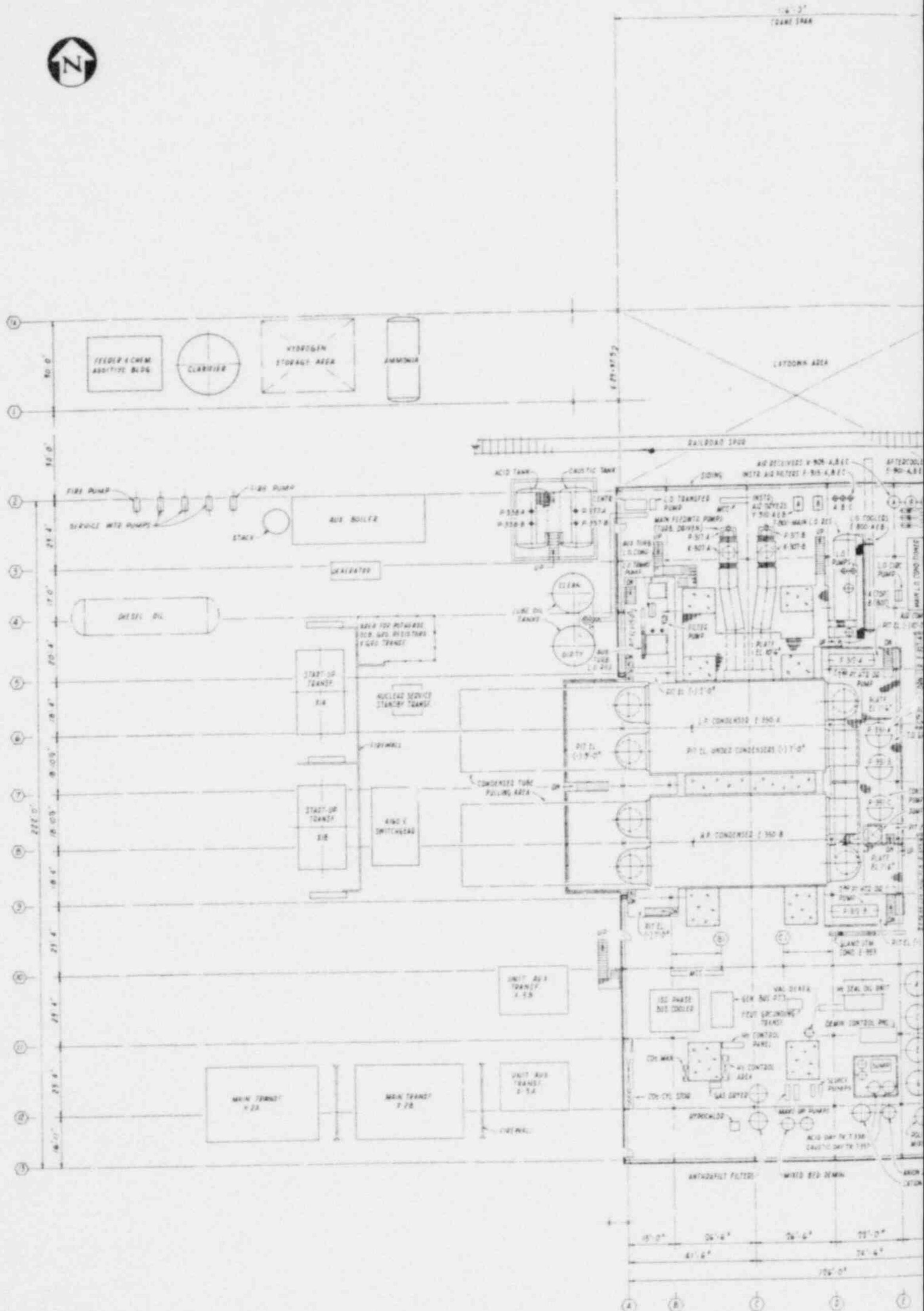
Supplement 1



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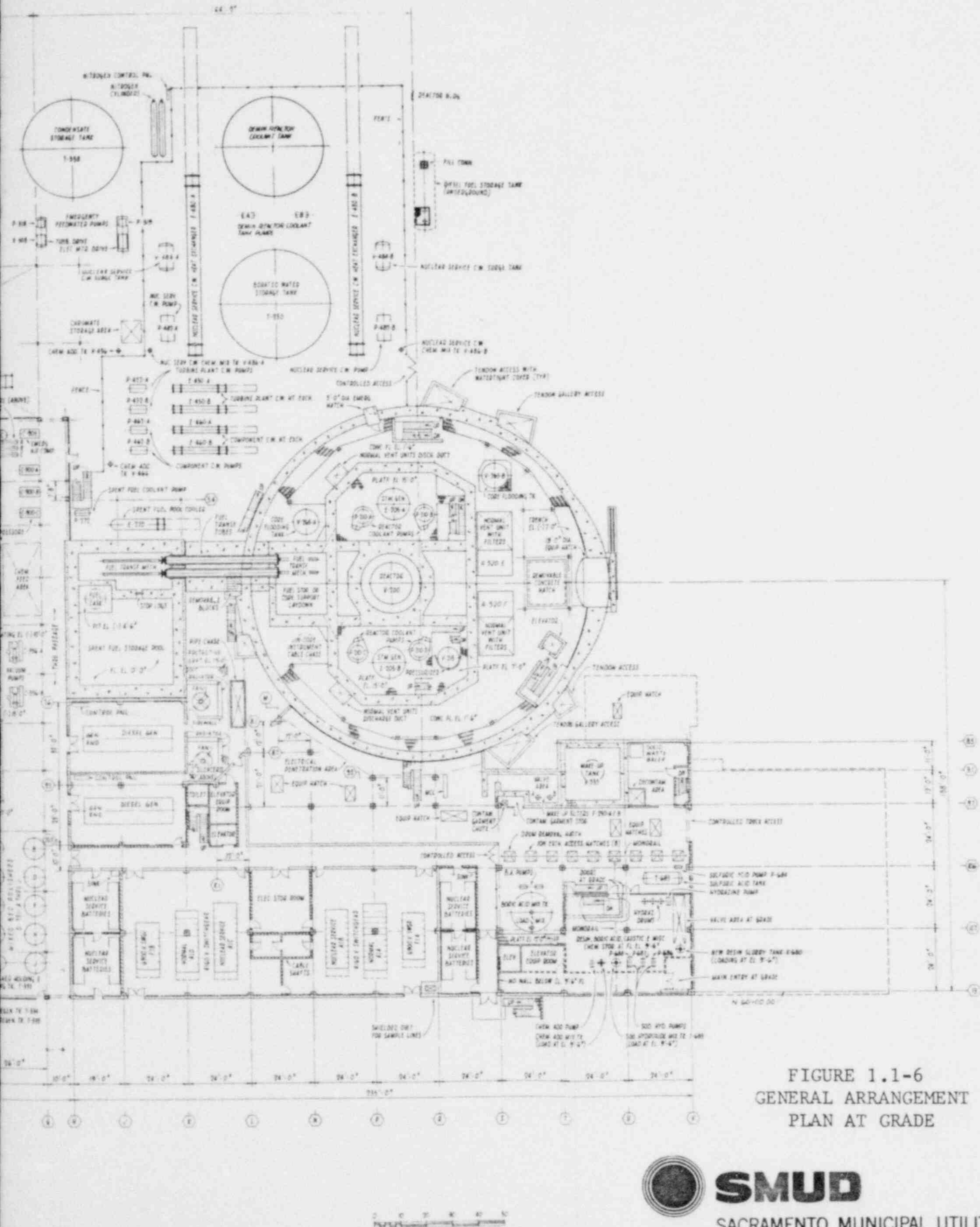

FIGURE 1.1-6
GENERAL ARRANGEMENT
PLAN AT GRADE

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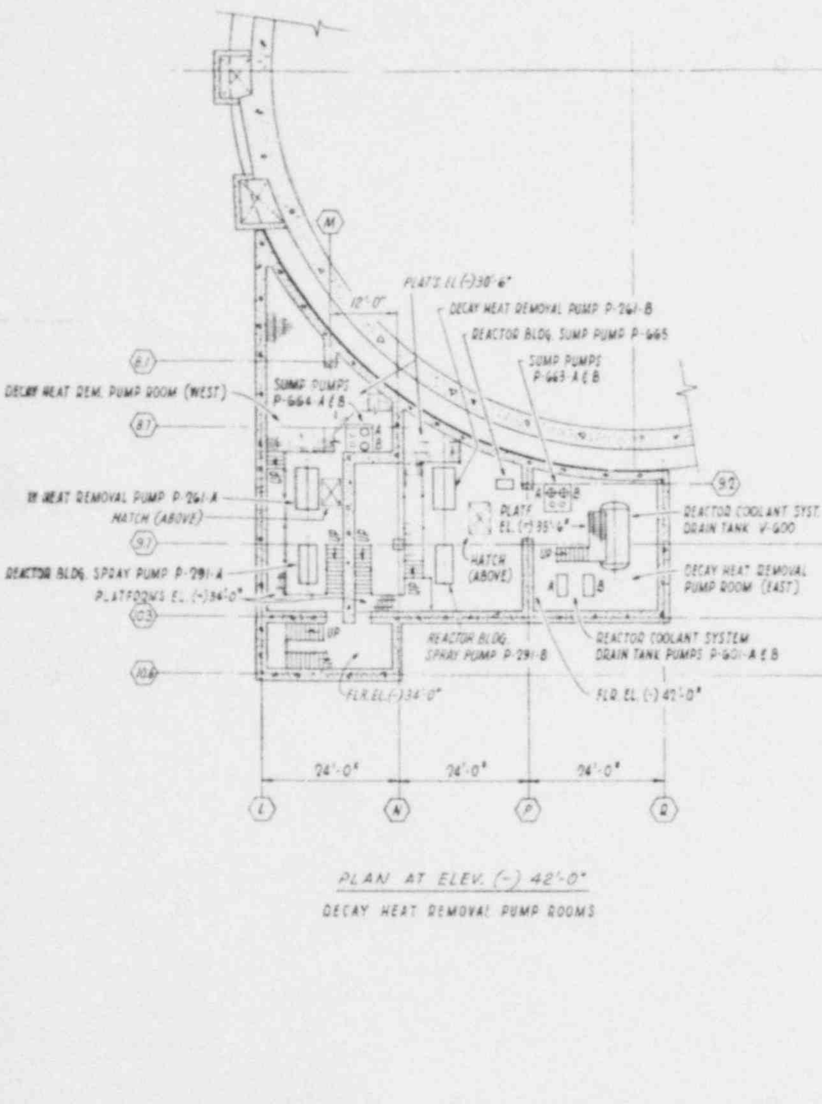
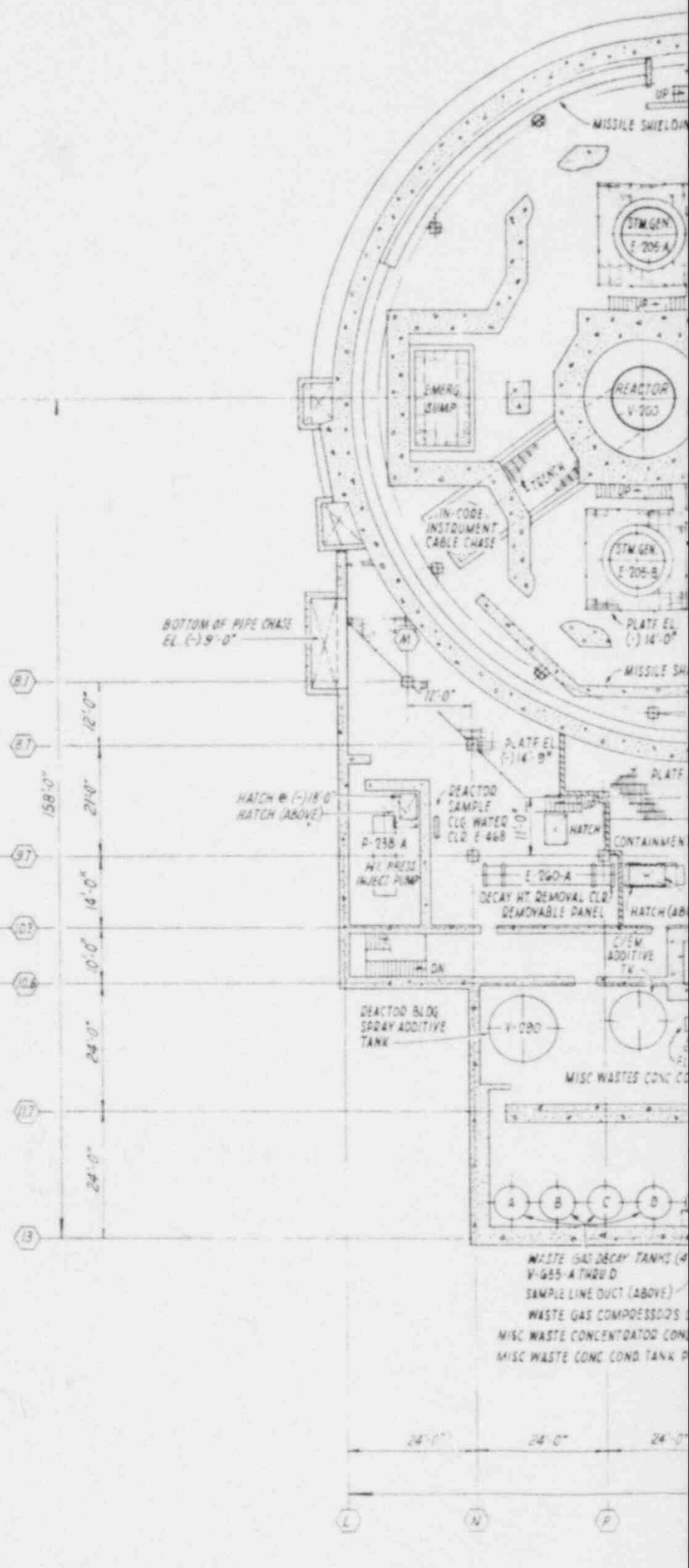


PLAN AT ELEV. (-) 42'-0"

DECAY HEAT REMOVAL PUMP ROOMS



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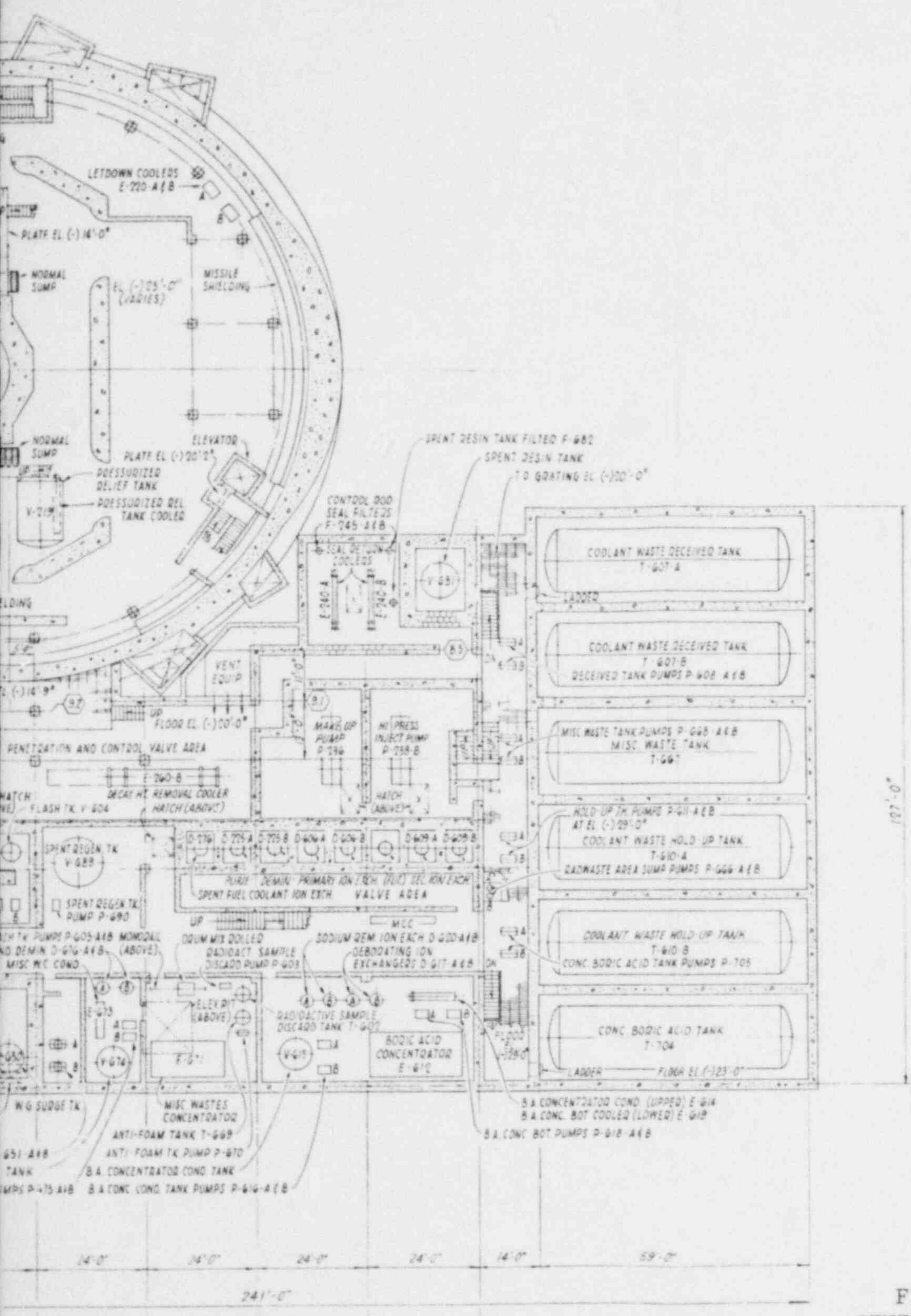

FIGURE 1.1-7
GENERAL ARRANGEMENT
PLAN BELOW GRADE

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Supplement 1

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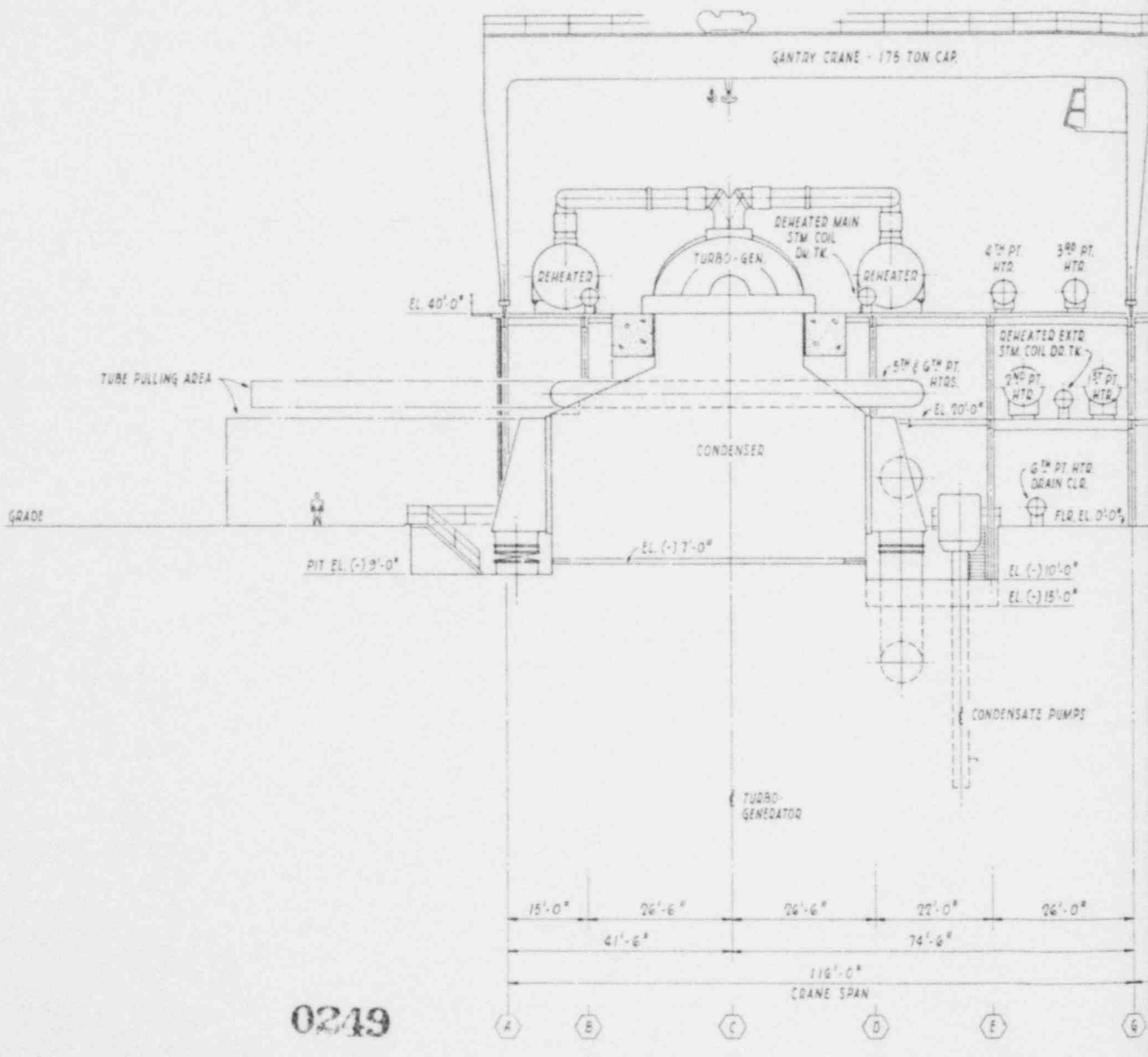
GANTRY CRANE - 175 TON CAP.
REHEATED MAIN STM. COIL DR. TK.
TURBO-GEN.
REHEATER
REHEATER
4TH PT. HTR.
3RD PT. HTR.
EL. 40'-0"
REHEATER EXTR. STM. COIL DR. TK.
5TH & 6TH PT. HTRS.
2ND PT. HTR.
1ST PT. HTR.
TUBE PULLING AREA
EL. 20'-0"
CONDENSER
6TH PT. HTR. DRAIN CLR.
GRADE
FLR. EL. 0'-0"
PIT EL. (-) 9'-0"
EL. (-) 7'-0"
EL. (-) 10'-0"
EL. (-) 15'-0"
CONDENSATE PUMPS
TURBO-GENERATOR
15'-0"
26'-6"
26'-6"
22'-0"
26'-0"
41'-6"
74'-6"
116'-0"
CRANE SPAN
A
B
C
D
E
G

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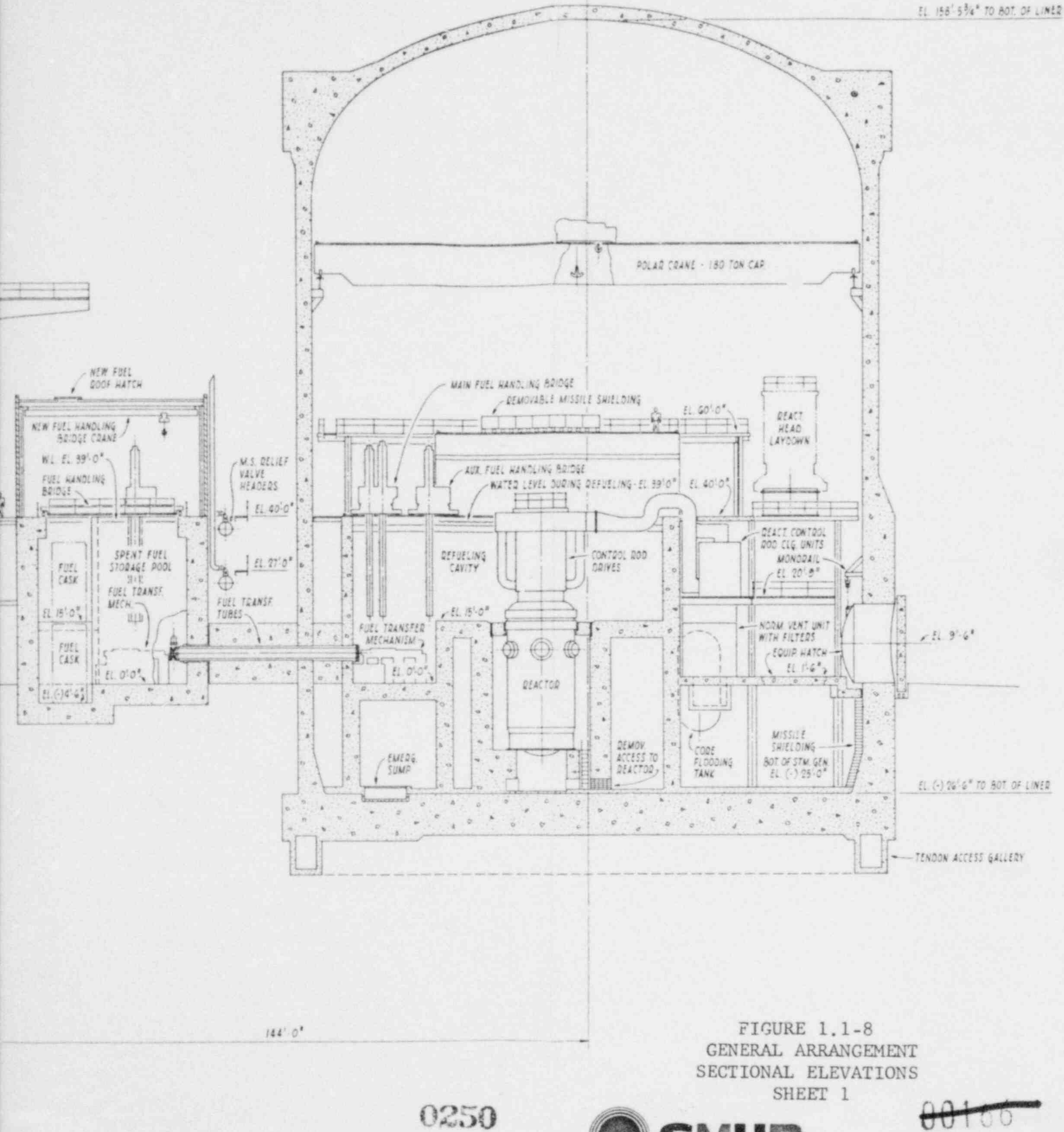

FIGURE 1.1-8
GENERAL ARRANGEMENT
SECTIONAL ELEVATIONS
SHEET 1

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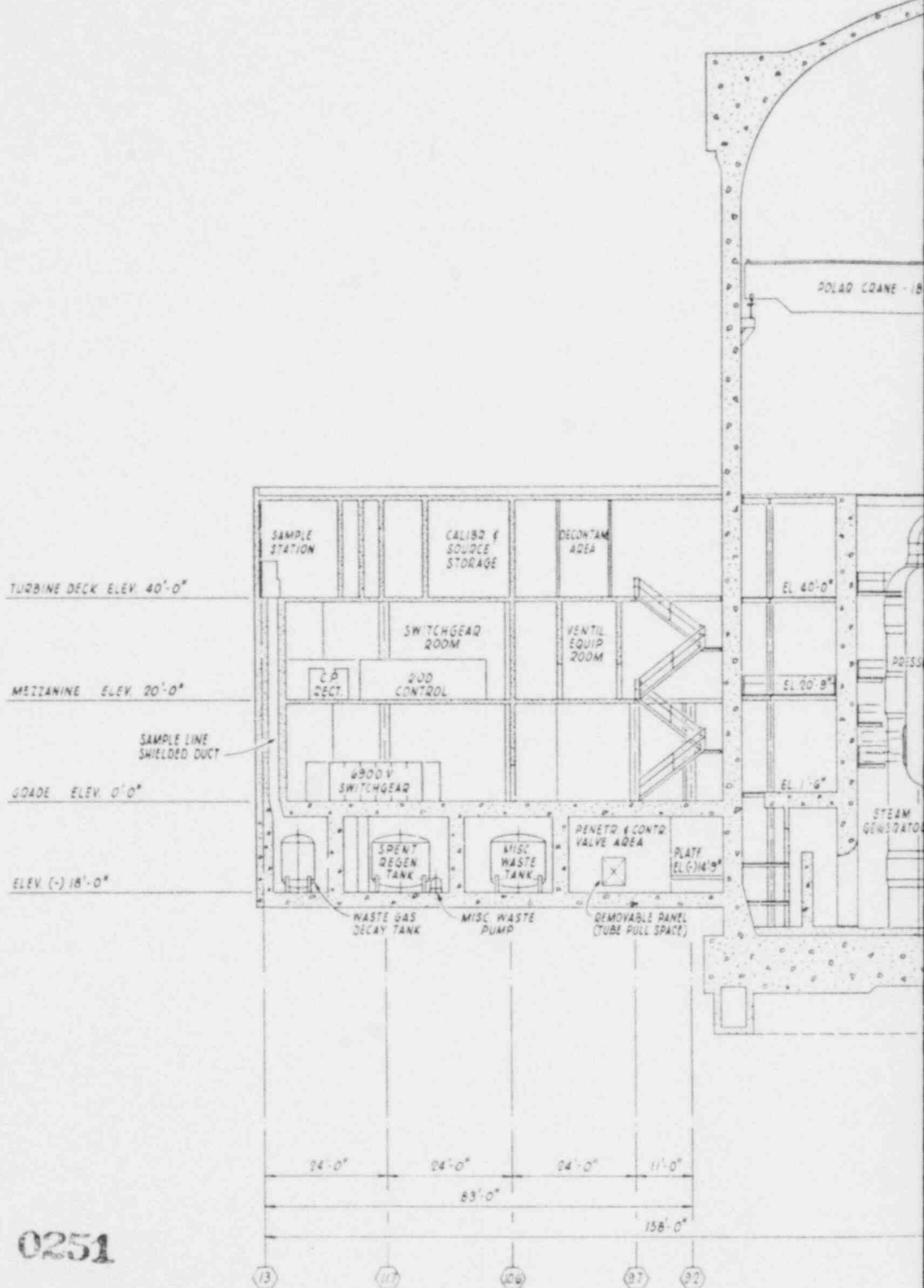
SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Supplement 1


POLAR CRANE
SAMPLE STATION
CALIBR & SOURCE STORAGE
DECONTAM AREA
TURBINE DECK ELEV. 40'-0"
EL 40'-0"
SWITCHGEAR ROOM
VENTIL EQUIP ROOM
C.P. RECT.
ROD CONTROL
MEZZANINE ELEV. 20'-0"
EL 20'-9"
SAMPLE LINE SHIELDED DUCT
4900 V SWITCHGEAR
GRADE ELEV. 0'-0"
EL 1'-6"
STEAM
SPENT REGEN TANK
MISC WASTE TANK
PENETR & CONTR VALVE AREA
PLATF EL(-)14'-9"
ELEV. (-) 18'-0"
WASTE GAS DECAY TANK
MISC WASTE PUMP
REMOVABLE PANEL (TUBE PULL SPACE)
24'-0"
24'-0"
24'-0"
11'-0"
83'-0"
158'-0"
13
117
106
87
82

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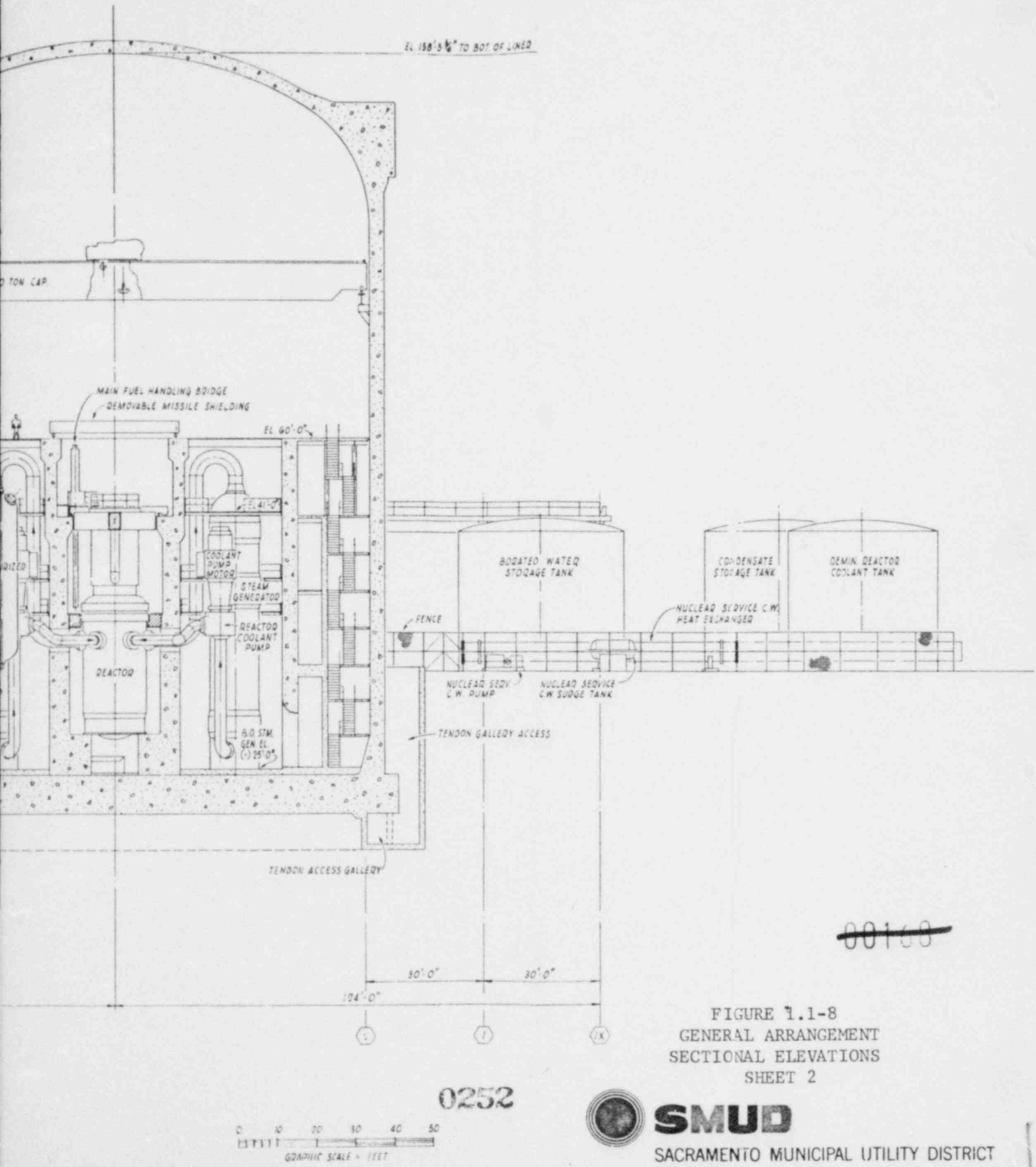

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FIGURE 1.1-8
GENERAL ARRANGEMENT
SECTIONAL ELEVATIONS
SHEET 2

0252

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SACRAMENTO MUNICIPAL UTILITY DISTRICT

Supplement 1

1.3 TABULAR CHARACTERISTICS

Table 1.3-1 is a comparative list of important design and operating characteristics of the Rancho Seco Nuclear Generating Station, Crystal River Nuclear Generating Plant Units 3 and 4 (Florida Power Corporation), Oconee Nuclear Station Units 1, 2, and 3 (Duke Power Company), and Turkey Point Units 3 and 4 (Florida Power and Light Company). The design and operating parameters of the Crystal River, Oconee, and Turkey Point stations are close to those of the Rancho Seco facility. The Crystal River and Oconee units each have the same rated core power as the Rancho Seco facility and are near-duplicates in other respects. The data in Table 1.2-1 represent information presented in available station descriptions, and Safety Analysis Reports submitted for licensing.

The design of each of these stations is based on information developed from operation of commercial and prototype pressurized water reactors over a number of years. The Rancho Seco design is based on this existing power reactor technology and has not been extended beyond the boundaries of known information or operating experience.

The similarities and differences of the features of the reactor stations listed in Table 1.2-1 are discussed in the following paragraphs. In each case, the item number used refers to the item numbers used in the table.

1.3.1 ITEM 1 - HYDRAULIC AND THERMAL DESIGN PARAMETERS

The parameters listed in this section are the same for the first three columns and are similar to the Turkey Point units. The differences in power level are reflected chiefly in the total heat output, core size (fuel loading), coolant flow rate, and total heat transfer surface. They amount only to a scaling down of the parameters above for a decrease in the thermal reactor power level, and do not alter the safety-related characteristics of the reactors. The departure from nucleate boiling ratio (DNBR) and the maximum ratio of peak-to-average total heat input per fuel rod ($F_{\Delta h}$ nuc.) are representative of a more conservative design for the B&W reactors than for the other reactor presented. These comparisons are discussed in detail in 3.2.3.2.

1.3.2 ITEM 2 - CORE MECHANICAL DESIGN PARAMETERS

The dimensions, materials, and technology for each of these reactors are similar. (Note the same design basis for Rancho Seco, Crystal River, and the Oconee units.) This uniformity is again due to optimization of the operating parameters for this type of reactor, and differences are related to the power levels.

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Item		Rancho
1	Hydraulic and Thermal Design Parameters	
	Rated Heat Output (core), Mwt	2,452
	Rated Heat Output (core), Btu/hr	8,369 x
	Maximum Overpower, %	14
	System Pressure (nominal), psia	2,200
	System Pressure (minimum steady state), psia	2,150
	Power Distribution Factors	
	Heat Generated in Fuel and Cladding, %	97.3
	$F_{\Delta h}$ (nuclear)	1.85
	F_q (nuclear)	3.15
	Hot Channel Factors	
	F_q (nuc. and mech.)	3.24
	DNB Ratio at Rated Conditions	2.27 (W 1.60 (B
	Minimum DNB Ratio at Design Overpower	1.73 (W 1.38 (B
	Coolant Flow	
	Total Flow Rate, lb/hr	131.3 x
	Effective Flow Rate for Heat Transfer, lb/hr	120.9 x
	Effective Flow Area for Heat Transfer, ft^2	47.75
	Average Velocity Along Fuel Rods, ft/sec	15.70
	Average Mass Velocity, lb/hr-ft^2	2.53 x
	Coolant Temperature, F	
	Nominal Inlet	555
	Maximum Inlet due to Instrumentation Error and Deadband	557
	Average Rise in Vessel	47.8
	Average Rise in Core	49.3
	Average in Core	579.7
	Average in Vessel	578.9
	Nominal Outlet of Hot Channel	644.4
	Average Film Coefficient, Btu/hr-ft^2-F	5,000
	Average Film Temperature Difference, F	31
	Heat Transfer at 100% Power	
	Active Heat Transfer Surface Area, ft^2	48,578
	Average Heat Flux, Btu/hr-ft^2	167,62
	Maximum Heat Flux, Btu/hr-ft^2	543,00
	Average Thermal Output, kw/ft	5.4
	Maximum Thermal Output, kw/ft	17.5
	Maximum Clad Surface Temperature at Nominal Pressure, F	654
	Fuel Central Temperature, F	
	Maximum at 100% Power	4,160
	Maximum at 114% Overpower	4,400
	Thermal Output, kw/ft at Maximum Overpower	
2	Core Mechanical Design Parameters	
	Fuel Assemblies	
	Design	CRA ca
	Rod Pitch, in.	0.558

TABLE 1.3-1
COMPARISON OF DESIGN PARAMETERS
per station unit basis unless noted)

Seco Nuclear Unit No. 1	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point No. 3 or 4
	2,452	2,452	2,097
10^6	$8,369 \times 10^6$	$8,369 \times 10^6$	$7,157 \times 10^6$
	14	14	12
	2,200	2,200	2,250
	2,150	2,150	2,220
	97.3	97.3	97.4
	1.85	1.85	1.75
	3.15	3.15	3.12
	3.24	3.24	3.25
3)	2.27 (W-3)	2.27 (W-3)	1.85 (W-3)
W-168)	1.60 (BAW-168)	1.60 (BAW-168)	
3)	1.73 (W-3)	1.73 (W-3)	1.30 (W-3)
W-168)	1.38 (BAW-168)	1.38 (BAW-168)	
10^6	131.3×10^6	131.3×10^6	100.6×10^6
10^6	120.9×10^6	120.9×10^6	91.5×10^6
	47.75	47.75	39.0
	15.70	15.70	13.9
10^6	2.53×10^6	2.53×10^6	2.35×10^6
	555	555	546.5
	557	557	550.5
	47.8	47.8	54
	49.3	49.3	59
	579.7	579.7	577
	578.9	578.9	574
	644.4	644.4	647
	5,000	5,000	5,500
	31	31	30
	48,578	48,578	42,460
	167,620	167,620	164,200
	543,000	543,000	533,600
	5.4	5.4	5.3
	17.5	17.5	17.3
	654	654	657
	4,160	4,160	4,070
	4,400	4,400	4,270
	19.9	19.9	19.4
n	CRA can	CRA can	RCC canless
	0.558	0.558	0.563

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TABLE 1.3-1(Continued)

Item		Rancho Seco Unit N
	Overall Dimensions, in.	8.522 x 8.522
	Fuel Weight (as UO_2), lb	201,520
	Total Weight, lb	283,200
	Number of Grids per Assembly	8
	Fuel Rods	
	Number	36,816
	Outside Diameter, in.	0.420
	Diametral Gap, in.	0.006
	Clad Thickness, in.	0.026
	Clad Material	Zircaloy-4
	Fuel Pellets	
	Material	UO_2 sintered
	Density, % of theoretical	95
	Diameter, in.	0.362
	Length, in.	0.8
	Control Rod Assemblies (CRA)	
	Neutron Absorber	5% Cd-15% In-
	Cladding Material	304 SS - cold
	Clad Thickness, in.	0.018
	Number of Assemblies	69
	Number of Control Rods per Assembly	16
	Core Structure	
	Core Barrel ID/OD, in.	147/150
	Thermal Shield ID/OD, in.	155/159
3	Preliminary Nuclear Design Data	
	Structural Characteristics	
	Fuel Weight (as UO_2), lb	201,520
	Clad Weight, lb	43,000
	Core Diameter, in. (equivalent)	128.9
	Core Height, in. (active fuel)	144
	Reflector Thickness and Composition	
	Top (water plus steel), in.	12
	Bottom (water plus steel), in.	12
	Side (water plus steel), in.	18
	H_2O/U (unit cell - cold)	2.97
	Number of Fuel Assemblies	177
	Fuel Rods/Fuel Assembly	208
	Performance Characteristics	
	Loading Technique	3 region
	Fuel Discharge Burnup, Mwd/Mtu	
	Average First Cycle	12,460
	Equilibrium Core Average	28,200
	Feed Enrichments, w/o U-235	
	Region 1	2.29
	Region 2	2.64
	Region 3	2.90
	Equilibrium	2.94
	Control Characteristics	
	Effective Multiplication (beginning of life)	
	Cold, No Power, Clean	1.302
	Hot, No Power, Clean	1.247
	Hot, Rated Power, Xe and Sm Equilibrium	1.158

Nuclear o. 1	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3		Turkey Point No. 3 or 4
	8.522 x 8.522	8.522 x 8.522		8.426 x 8.426
	201,520	201,520		179,000
	283,200	283,200		226,200
	8	8		8
	36,816	36,816		32,028
	0.420	0.420		0.422
	0.006	0.006		0.0065
	0.026	0.026		0.0243
	Zircaloy-4	Zircaloy-4		Zircaloy
	UO_2 sintered	UO_2 sintered		UO_2 sintered
	95	95		94-93
	0.362	0.362		0.3669
	0.8	0.8		0.600
80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag		5% Cd-15% In-80% Ag
worked	304 SS - cold worked	304 SS - cold worked		304 SS - cold worked
	0.018	0.018		0.019
	69	69		41
	16	16		20
	147/150	147/150		133.5/137.25
	155/159	155/159		141.0/147.5
	201,520	201,520		179,000
	43,000	43,000		35,600
	128.9	128.9		119.5
	144	144		144
	12	12		10
	12	12		10
	18	18		15
	2.97	2.97		3.48
	177	177		157
	208	208		204
	3 region	3 region		3 region
	12,460	8,260		14,000
	28,200	28,200		27,000
		Nos. 1 and 3		
	2.29	2.24		2.28
	2.64	2.47		2.43
	2.90	2.77		2.73
	2.94	3.09		--
	Nos. 3 and 4	Nos. 1 and 3	No. 2	
	1.302	1.312	1.255	1.275
	1.247	1.258	1.201	1.225
	1.158	1.167	1.119	1.170 ~~00173~~

TABLE 1.3-1 (Continued)

Item		Rancho Seco Nu Unit No
	Control Rod Assemblies	
	Material	5% Cd-15% In-80%
	Number of Assemblies	69
	Number of Absorber Rods per CRA	16
	Total Rod Worth $\left(\frac{\Delta k}{k}\right)$, %	10.0
	Boron Concentrations	
	To Shut Reactor Down With Rods Inserted (clean), cold/hot ppm	1290/1080
	Boron Worth (hot), $\%\left(\frac{\Delta k}{k}\right)$/ppm	1/100
	Boron Worth (cold), $\%\left(\frac{\Delta k}{k}\right)$/ppm	1/75
	Kinetic Characteristics	
	Moderator Temperature Coefficient, $\left(\frac{\Delta k}{k}\right)$/F	$+1.0 \times 10^{-4}$ to -
	Moderator Pressure Coefficient, $\left(\frac{\Delta k}{k}\right)$/psi	-1.0×10^{-6} to +
	Moderator Void Coefficient, $\left(\frac{\Delta k}{k}\right)$/% void	$+1.0 \times 10^{-4}$ to -
	Doppler Coefficient, $\left(\frac{\Delta k}{k}\right)$/F	-1.1×10^{-5} to -
4	Principal Design Parameters of the Reactor Coolant System	
	System Heat Output, MWt	2,468
	System Heat Output, Btu/hr	$8,423 \times 10^6$
	Operating Pressure, psig	2,185
	Reactor Inlet Temperature, F	555
	Reactor Outlet Temperature, F	603
	Number of Loops	2
	Design Pressure, psig	2,500
	Design Temperature, F	650
	Hydrostatic Test Pressure (cold), psig	3,125
	Coolant Volume, including pressurizer, ft^3	11,800
	Total Reactor Flow, gpm	352,000
5	Reactor Coolant System Code Requirements	
	Reactor Vessel	ASME III, Class
	Steam Generator	
	Tube Side	ASME III, Class
	Shell Side	ASME III, Class
	Pressurizer	ASME III, Class
	Pressurizer Relief Tank	ASME III, Class
	Pressurizer Safety Valves	ASME III
	Reactor Coolant Piping	USASI B31.1
	Reactor Coolant Pump Casing	ASME III, Class
6	Principal Design Parameters of the Reactor Vessel	
	Material	SA-533 Grade B 18-8 stainl

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clear 1	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point No. 3 or 4
Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag
	69	69	41
	16	16	20
	10.0	10.0	7.0
	1290/1080	1290/1150	2300/2500
	1/100	1/100	1/130
	1/75	1/75	1/100
$.0 \times 10^{-4}$	$+1.0 \times 10^{-4}$ to -3.0×10^{-4}	$+1.0 \times 10^{-4}$ to -3.0×10^{-4}	$+1.0 \times 10^{-4}$ to -3.0×10^{-4}
$.0 \times 10^{-6}$	-1.0×10^{-6} to $+3.0 \times 10^{-6}$	-1.0×10^{-6} to $+3.0 \times 10^{-6}$	-1.0×10^{-6} to $+3.0 \times 10^{-6}$
$.0 \times 10^{-3}$	$+1.0 \times 10^{-4}$ to -3.0×10^{-3}	$+1.0 \times 10^{-4}$ to -3.0×10^{-3}	$+0.5 \times 10^{-3}$ to -2.0×10^{-3}
$.7 \times 10^{-5}$	-1.1×10^{-5} to -1.7×10^{-5}	-1.1×10^{-5} to -1.7×10^{-5}	-1.0×10^{-5} to -2.0×10^{-5}
	2,468	2,468	2,097
	$8,423 \times 10^6$	$8,423 \times 10^6$	$7,156 \times 10^6$
	2,185	2,185	2,235
	555	555	546.5
	603	603	600.8
	2	2	3
	2,500	2,500	2,485
	650	650	650
	3,125	3,125	3,110
	11,800	11,800	9,800
	352,000	352,000	266,400
A	ASME III, Class A	ASME III, Class A	ASME III, Class A
A	ASME III, Class A	ASME III, Class A	ASME III, Class A
A	ASME III, Class A	ASME III, Class A	ASME III, Class C
A	ASME III, Class A	ASME III, Class A	ASME III, Class A
C	ASME III, Class C	ASME III, Class C	ASME III, Class C
	ASME III	ASME III	ASME III
	USASI B31.1	USASI B31.1	USASI B31.1
A	ASME III, Class A	ASME III, Class A	
clad with ess steel	SA-533 Grade B, clad with 18-8 stainless steel	SA-533 Grade B, clad with 18-8 stainless steel	SA-302, Grade B, clad with Type 304 austenitic SS

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TABLE 1.3-1(Continued)

Item		Rancho Seco Nu Unit No
	Design Pressure, psig	2,500
	Design Temperature, F	650
	Operating Pressure, psig	2,185
	Inside Diameter of Shell, in.	171
	Outside Diameter Across Nozzles, in.	249
	Overall Height of Vessel and Closure Head (over CRD nozzles), ft-in.	42-0
	Minimum Clad Thickness, in.	1/8
7	Principal Design Parameters of the Steam Generators	
	Number of Units	2
	Type	Vertical, once-th with integral s heater.
	Tube Material	Inconel
	Shell Material	Carbon steel
	Tube Side Design Pressure, psig	2,500
	Tube Side Design Temperature, F	650
	Tube Side Design Flow, lb/hr	65.66×10^6
	Shell Side Design Pressure, psig	1,050
	Shell Side Design Temperature, F	600
	Operating Pressure, Tube Side, Nominal, psig	2,185
	Operating Pressure, Shell Side, Maximum, psig	910
	Maximum Moisture at Outlet at Rated Load, %	35 F superheat
	Hydrostatic Test Pressure (tube side-cold), psig	3,125
8	Principal Design Parameters of the Reactor Coolant Pumps	
	Number of Units	4
	Type	Vertical, single
	Design Pressure, psig	2,500
	Design Temperature, F	650
	Operating Pressure, Nominal, psig	2,185
	Suction Temperature, F	555
	Design Capacity, gpm	88,000
	Design Total Developed Head, ft	370
	Hydrostatic Test Pressure (cold), psig	3,125
	Motor Type	A-C Induction, s speed
	Motor Rating (nominal), hp	9,000
9	Principal Design Parameters of the Reactor Coolant Piping	
	Material	Carbon steel cl
	Hot Leg (ID), in.	36
	Cold Leg (ID), in.	28

lear 1	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point No. 3 or 4
	2,500	2,500	2,485
	650	650	650
	2,185	2,185	2,235
	171	171	155.5
	249	249	240/235-3/8
	42-0	42-0	41-0
	1/8	1/8	5/32
	2	2	3
ough per-	Vertical, once-through with integral superheater.	Vertical, once-through with integral superheater.	Vertical, U-tube with integral moisture separator.
	Inconel	Inconel	Inconel
	Carbon steel	Carbon steel	Carbon steel
	2,500	2,500	2,485
	650	650	650
	65.66×10^6	65.66×10^6	33.53×10^6
	1,050	1,050	1,085
	600	600	600
	2,185	2,185	2,235
	910	910	1,005
	35 F superheat	35 F superheat	1/4
	3,125	3,125	3,110
	4	4	3
stage	Vertical, single stage	Vertical, single stage	Vertical, single stage. Radial flow with bottom suction and horizontal discharge.
	2,500	2,500	2,485
	650	650	650
	2,185	2,185	2,235
	555	555	546.5
	88,000	88,000	88,800
	370	370	256
	3,125	3,125	3,110
ngle	A-C Induction, single speed	A-C Induction, single speed	A-C Induction, single speed
	9,000	9,000	5,500
with SS	Carbon steel clad with SS	Carbon steel clad with SS	Austenitic SS
	36	36	29
	28	28	27-1/2

TABLE 1.3-1 (Continued)

Item		Rancho Seco Nuclea Unit No.1
	Between Pump and Steam Generator (ID), in.	28
10	Reactor Building System Parameters	
	Type	Steel-lined, prestress post-tensioned concr vertical cylinder wi flat bottom and shal domed roof.
	Design Parameters	
	Inside Diameter, ft	130
	Height, ft	185
	Free Volume, ft^3	2,005,000
	Reference Incident Pressure, psig	59
	Reference Incident Energy (E_1), Btu	306,700,000
	Energy Required to Produce Incident Pressure (E_2), Btu	341,806,000
	Ratio: E_1/E_2	0.897
	Ratio: $(E_2 - E_1)/E_1$	0.115
	Concrete Thickness, ft	
	Vertical Wall	3-3/4
	Dome	3-1/4
	Reactor Building Leak Prevention and Mitigation	Leak-tight penetration and continuous steel liner. Automatic is tion where required.
	Gaseous Effluent Purge	Discharge vent above t of Reactor Building (~200 ft above grad
11	Engineered Safeguards	
	Safety Injection System	
	No. of High Head Pumps	3
	No. of Low Head Pumps	2
	Reactor Building Emergency Coolers	
	No. of Units	4
	Air Flow Cap'y. Each, at Accident Condition, cfm	40,000
	Core Flooding System	
	No. of Tanks	2
	Total Volume, ft^3	2,820
	Postaccident Filters	
	No. of Units	2
	Air Flow Cap'y. Each, at Post Accident Conditions, cfm	54,500
	Type	HE & Charcoal

	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point No. 3 or 4	
	28	28	31	
d, te, h ow	Steel-lined, prestressed, post-tensioned concrete, vertical cylinder with flat bottom and shallow domed roof.	Steel-lined, prestressed, post-tensioned concrete, vertical cylinder with flat bottom and shallow domed roof.	Steel-lined, prestressed, post-tensioned concrete, vertical cylinder with flat bottom and shallow domed roof.	
	130	116	116	
	187	206	177	S1
	2,000,000	1,900,000	1,550,000	
	55	59	59	
	306,700,000	306,700,000	272,000,000	
	335,200,000	341,806,000	300,000,000	
	0.915	0.897	0.907	
	0.093	0.115	0.103	
	3-1/2	3-3/4	3-3/4	
	3	3-1/4	3-1/4	
la-	Leak-tight penetrations and continuous steel liner. Automatic isolation where required.	Leak-tight penetrations and continuous steel liner. Automatic isolation where required.	Leak-tight penetrations and continuous steel liner. Automatic isolation where required.	
p)	Discharge vent above top of Reactor Building (~200 ft above grade)	Discharge vent above top of Reactor Building (~200 ft above grade)	Through particulate filters and monitors. Part of the main exhaust system.	
	3	3	3	
	2	3	2	2
	3	3	3	
	54,000	54,000	80,000	2
	2	2	3	
	2,820	2,820	3,600	
	None	None inside Reactor Building. Leakage from penetrations collected, filtered, and discharged through station vent.	None	2

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TABLE 1.3-1 (Continued)

Item		Rancho Se[co] Un[it]
	(Deleted)	
	Reactor Building Spray	
	No. of Pumps	2
	Including Spray Additive Injection	Yes
	Emergency Power	
	Generator Units, No.	2
	Type	Diesel
	Engineered Safeguards Operable From Emergency Power Source (minimum)	All enginee[red] equipment being ope[rable] site emer[gency]

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...o Nuclear ...t No. 1	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point No. 3 or 4
	2	2	2
	Yes	No	No
	3	2	2 for both Units
	Diesel	Not comparable*	Diesel
...red safeguards ... is capable of ...rated from on- ...gency power.	All engineered safeguards equipment is capable of being operated from on-site emergency power.	All engineered safeguards equipment is capable of being operated from on-site emergency power. * Two 70 KW hydroelectric units with one overhead and one underground feeder. Also, one of three 44 MVA gas turbine units located 30 miles distant dedicated soley for back up emergency power.	1 High head Safety Injection (SI) pump 1 Low head SI pump 3 Containment air recirculation units 1 Containment spray pump 1 Service water pump ~~00181~~

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There are also small differences in the mechanical assembly of the fuel rods and the number of control rods used between the first three columns (B&W units) and the Turkey Point units. The increased number of control rods in the B&W reactors provides for maneuverability and flexibility of operation. The B&W reactors utilize a canned fuel assembly which provides structural integrity and protection of the fuel rods against damage during fuel handling operations.

1.3.3 ITEM 3 - PRELIMINARY NUCLEAR DESIGN DATA

Since these reactors have essentially the same core geometrical configuration, the fuel loading differs by an amount proportional to the physical size of the reactor core. Note that the design data for the Rancho Seco, Crystal River, and the Oconee units are the same except for the first-cycle burnup and feed enrichments. These differences reflect the greater first-cycle burnup of Rancho Seco and Crystal River over Oconee. Oconee Unit 1 has a first-cycle fuel sharing program with Unit 2, which requires a lower first-cycle enrichment for Unit 2.

The basis of the 2.97 (H_2O/U) for the B&W units is the ratio of water molecules to the atoms of U metal in the fuel assembly envelope. This value of the water-to-U-metal volume ratio, consistent with the entry for Turkey Point Units 3 or 4, is 3.53. Each core has a three-region fuel loading, but differs in the fuel burnup ratio that is to be used.

The B&W reactor designs offer 3.0 percent greater reactivity control in the control rods over Turkey Point. This is reflected in the lesser concentration of boron that is required to control the reactivity over the lifetime of the reactor core. Some slight differences are noted in reactivity coefficients.

1.3.4 ITEM 4 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM

Most of the features in this section are directly related to material properties and the amount of heat produced in the reactor core. Note that the B&W units are identical. The parameters are scaled in proportion to the power of the reactor. The major difference is the number of coolant loops required to remove the heat produced.

For the B&W units, only two loops are required since once-through steam generators are used instead of the U-tubes-in-shell design. The greater cooling capacity of these steam generators permits a reduction in the number of cooling loops for an equivalent amount of heat removed.

1.3.5 ITEM 5 - REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

The B&W units are identical. Code requirements for the shell side of the steam generator conform to the ASME III Class A specification. This is

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considered to be a contribution to the safety of the vessel since it enhances the integrity because of the more stringent ASME III Class A design, material, and quality control requirements.

1.3.6 ITEM 6 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL

The B&W units are identical. These vessel designs are characterized by a thinner thermal shield and a relatively larger diameter. The larger diameter provides for additional water between the edge of the core and the vessel, which leads to additional neutron attenuation.

1.3.7 ITEM 7 - PRINCIPAL DESIGN FEATURES OF THE STEAM GENERATORS

The steam generators in the B&W units are the same. They are basically different from the Turkey Point units since they are once-through design and incorporate an integral superheat section.

1.3.8 ITEM 8 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS

The B&W designs are the same. In each specific tabular parameter the relative number or size is in proportion to the total amount of heat removed from the core. The B&W reactor pumps have higher head and horsepower requirements than the Turkey Point units have for approximately the same flow because of the increased flow losses of the once-through steam generators and the use of only two reactor coolant loops.

1.3.9 ITEM 9 - PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING

The B&W unit piping designs are the same. They utilize carbon steel clad with stainless steel.

1.3.10 ITEM 10 - REACTOR BUILDING PARAMETERS

All reactor buildings are basically of the same design and construction. The differences are physical dimensions, amount of concrete shielding needed, and design incident pressures, which are a direct result of station layout, engineered safeguards, system capacities, and site location. The reactor building design and shielding offer satisfactory protection to the surrounding population in case of an accident and during normal operation of the generating units.

1.3.11 ITEM 11 - ENGINEERED SAFEGUARDS

Rancho Seco engineered safeguards are basically the same as Crystal River in that they consist of two completely independent 100% systems. Oconee differs in that it includes a penetration room ventilation system for each unit. Rancho Seco will use a spray additive injection in the reactor building spray for iodine removal because of differences in designs and site characteristics.

5. CONTAINMENT SYSTEM

5.1 STRUCTURAL DESIGN

5.1.1 GENERAL DESCRIPTION OF CONTAINMENT STRUCTURE

The reactor containment is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. Hoop tendons are placed in 240 degree systems using three buttresses as anchorages, with the tendons staggered so that half of the tendons at each buttress terminate at that buttress. The foundation slab is reinforced with conventional reinforcing steel. A welded steel liner is attached to the inside face of the concrete shell to insure a high degree of leaktightness. The base liner is installed on top of the structural slab and will be covered with concrete. The structure will provide biological shielding for both normal and accident situations. S1

The reactor containment will completely enclose the entire reactor and reactor coolant system and ensure that an acceptable upper limit for leakage of radioactive materials to the environment would not be exceeded even if gross failure of the reactor coolant system were to occur. The approximate dimensions of the reactor containment are: inside diameter, 130 feet; inside height, 185 feet; vertical wall thickness, 3-3/4 feet; dome thickness, 3-1/4 feet; and the foundation slab, 8 feet. The building encloses the pressurized water reactor, steam generators, reactor coolant loops and portions of the auxiliary and engineered safeguards systems. The internal net free volume is 2,005,000 cubic feet. S1 S1

Full advantage is being taken in the design of this reactor building of the experience gained in the review of similar designs with the AEC for the Florida Power and Light Company's Turkey Point Plant, Consumers Power Company's Palisades Plant, Wisconsin-Michigan Power Company's Point Beach Plant and Duke Power Company's Oconee Nuclear Station, as well as containment designs by others which meet the same functional requirements.

Representative details of the construction that will be used are shown in Figures 5.1-1, 5.1-2, 5.1-3 and Table 5.1-1. 1

5.1.2 BASIS FOR DESIGN LOADS

The reactor containment will be designed for all credible conditions of loading, including normal loads, loads during maximum credible accident, test load, and loads due to adverse environmental conditions. The following loadings will be considered:

a. The loading caused by the pressure and temperature transients of the design base accident.

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b. Structure dead load

c. Live loads

d. Earthquake load

e. Wind loads

f. External pressure load

The two critical loading conditions are those caused by the design base accident resulting from failure of the reactor coolant system and those caused by an earthquake.

5.1.2.1 Maximum Credible Accident Load

The minimum design pressure and temperature of the containment will be equal to the peak pressure and temperature occurring as a result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical double-ended severance of a 36-inch ID reactor coolant pipe.

The supports for the reactor coolant system will be designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible.

Transients resulting from the design base accident and other, lesser, accidents are presented in Section 14 and serve as the basis for a containment design pressure of 59 psig.

The design pressure will not be exceeded during any subsequent long-term pressure transient caused by the combined effects of such heat sources as residual heat and metal-water reactions. These effects will be overcome by the combination of emergency-powered engineered safeguards and structural heat sinks.

5.1.2.2 Structure Dead Load

Dead load will consist of the weight of the concrete wall, dome, base slab, and any internal concrete. Weights used for dead load calculations will be as follows:

a. Concrete	148 lb/ft^3
b. Steel reinforcing	489 lb/ft^3 using nominal cross-sectional areas of reinforcing as defined in ASTM for bar sizes and nominal cross-sectional areas of prestressing
c. Steel lining	489 lb/ft^3, using nominal cross-sectional area of lining

5.1.2.3 Live Loads

live loads will include snow and ice loads on the roof of the containment dome. The roof load will be 20 pounds per horizontal square foot.

Equipment loads will be those specified on the drawings supplied by the manufacturers of the various pieces of equipment.

Live loads will be assumed for the design of internal slabs consistent with the intended use of the slabs.

5.1.2.4 Earthquake Loads

Earthquake loading is predicated upon a design earthquake at the site having a horizontal ground acceleration of 0.13g. In addition, a maximum hypothetical earthquake having a ground acceleration of 0.25g will be used to check the design to ensure no loss of function. 3

Seismic response spectrum curves are given in Appendix 5A for both horizontal ground motion and vertical ground motion. A dynamic analysis will be used to arrive at equivalent static loads for design. Seismic loads will be combined as outlined in Appendix 5A. S1

5.1.2.5 Wind Loads 1

Wind loading for the containment structure is based on Figure 1 (b) of ASCE Paper 3269, "Wind Forces on Structures," using the fastest wind speed for a 100-year recurrence period. ASCE Paper 3629 will also be used to determine shape factors, gust factors, and variation of wind velocity with height. Based upon the site location and inland classification the design wind velocity is 90 MPH at a reference 30 feet above ground level. 1

5.1.2.6 External Pressure Load

External pressure loading with a differential of approximately two pounds per square inch from outside to inside will be considered.

The external design pressure is equivalent to have a barometric pressure rise to 31 inches of mercury after the containment was sealed at 29 inches of mercury. Therefore, operation of purge valves will not be required due to barometric changes during normal operation.

The external design pressure is also adequate to permit the containment to be cooled to 60 F from an initial maximum operating condition of 120 F. Therefore, operation of purge valves will not be necessary during this condition. Vacuum breakers are not required.

5.1.3 CONSTRUCTION MATERIALS

Basically four materials will be used for the foundation and the containment structure. These are:

a. Concrete

b. Reinforcing steel

c. Steel prestressing tendons

d. Steel liner plate

Detailed specifications and working drawings for these materials and their installation will be of such scope as to assure that the quality of work will be commensurate with the necessary integrity of the containment structure.

Basic specifications for these materials include the following.

5.1.3.1 Concrete

All concrete work will be in accordance with ACI 318-63 "Building Code Requirements for Reinforced Concrete" and to ACI-301 "Specifications for Structural Concrete for Buildings." Concrete will be a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, and water. Admixtures will be added to improve the quality and workability of the fluid concrete during placement and to retard the set of the concrete. Maximum practical size aggregate, water reducing additives, and a low slump of two to three inches will be used to minimize shrinkage and creep. Aggregates will conform to "Standard Specifications for Concrete Aggregate" ASTM Designation C33. Fine aggregate will consist of sharp, hard, strong, and durable sand, free from adherent coatings, clay loam, alkali, organic material, or other deleterious substances.

Acceptability of aggregates will be based on the ASTM Tests as stated in Section 5.4.3.1.

Cement will be Type II as specified in "Standard Specifications for Portland Cement" ASTM Designation C150 and will be tested to comply with ASTM C-114.

Water for mixing concrete will be clean and free from any deleterious amounts of acid, alkali, salts, oil, sediment, or organic matter.

The water will be potable and will not contain impurities in amounts that will cause a change of more than 25 percent in setting time for the Portland Cement, nor a reduction in the compressive strength of mortar of more than 5 percent as compared with results obtained using distilled water.

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A water-reducing agent will be employed to reduce shrinkage and creep of concrete. Admixtures containing chlorides will not be used. The following types of agent will be tested with the concrete materials selected for the containment structure.

a. Pozzolith No. 8

b. Pozzolith 100 R

c. Plastiment

d. Placewell LS

The agent selected will be the one providing the smallest shrinkage as determined by ASTM C-494, "Specifications for Chemical Admixtures for Concrete."

Concrete mixes will be designed in accordance with ACI 613 using materials qualified and accepted for this work. Only mixes meeting the design requirements specified for containment structure concrete will be used.

Trial mixes will be tested in accordance with applicable ASTM Codes as indicated below:

Test	ASTM
Making and Curing Cylinder in Laboratory	C-192
Air Content	C-231
Slump	C-143
Bleeding	C-232
Compressive Strength Tests	C-39

Eight cylinders will be cast from each design mix for two tests on each of the following days: 3, 7, 28, and 90. The concrete will have a design compressive strength of 5000 psi at 28 days for the containment wall and dome and 4000 psi at 28 days for the containment base slab.

Test cylinders will be cast from the mix proportions selected for construction and the following concrete properties will be determined:

a. Uniaxial creep

b. Modulus of elasticity and Poisson's ratio

c. Autogenous shrinkage

d. Thermal diffusivity

e. Thermal coefficient of expansion

f. Compressive strength

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Concrete samples will be taken from the mix according to ASTM C-172, "Sampling Fresh Concrete." From these samples, cylinders for compression testing will be made. They will be stripped within 24 hours after casting and marked and stored in the curing room. These cylinders will be made in accordance with ASTM C-31, "Tentative Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field."

Slump, air content, and temperature measurements will be taken when cylinders are cast. Slump tests will be performed in accordance with ASTM C-143, "Standard Method of Test for Slump of Portland Cement Concrete." Air content tests will be performed in accordance with ASTM C-231, "Standard Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method." Compressive strength tests will be made in accordance with ASTM C-39, "Method of Test for Compressive Strength of Molded Concrete Cylinders."

Evaluation of compression tests will be in accordance with ACI 214-65.

A full-time inspector, who has had experience in concrete work, will continuously check the concrete batching and placing operations.

5.1.3.2 Reinforcing Steel

Reinforcing steel in the base slab of the containment structure and around penetrations in the cylinder will be deformed billet steel bars conforming to ASTM Designation A432-65. This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7 percent in an 8 inch specimen. Deformed billet steel bars conforming to ASTM A15 or A408 - Intermediate Grade will be used in the cylinder wall and the domed roof to control shrinkage and tensile cracks. This steel has a minimum yield strength of 40,000 psi and a minimum tensile strength of 70,000 psi. The A15 steel has a minimum elongation of 12 percent in 8 inch specimen, while A408 has a minimum elongation of 10 percent in an 8 inch specimen.

Mill test results will be obtained from the reinforcing steel supplier for each heat of steel to show proof that the reinforcing steel has the specified composition, strength, and ductility. Splices in reinforcing bar sizes No. 11 and smaller will be lapped in accordance with ACI 318-63, and for bars larger than No. 11, Cadweld splices will be used.

Welding of reinforcing steel, if required, will be performed by qualified welders in accordance with AWS D12.1, "Recommended Practice for Welding Reinforcing Steel, Metal Inserts, and Connections in Reinforced Concrete Construction," but tack welding will not be permitted.

5.1.3.3 Steel Prestressing Tendons

There are a number of post tensioning systems suited to containment vessels available in this country. The ultimate capacity of the wire or strand systems presently considered suitable for this containment vary
S1 from 494 kips to 2000 kips. Listed below are pertinent features of some of these post tensioning systems:

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Manufacturer	Freyssinet	BBRV	Stress Steel (SEEE)
Designation	Wire strand	Wires	Wire strand
Ultimate Capacity	494 Kips	2000 Kips	792 Kips
Design Capacity	296 Kips	1200 Kips	468 Kips
End Anchorage	Open end, male and female cone	Buttonhead	Swaged, threaded collar and nut

S1

The above systems have been used in foreign and domestic prestressed concrete reactor vessels, containment structures and domestic conventional structures. The design will permit the use of any system that meets the specified requirements. S1

The 12-0.5" Freyssinet system was used in the French containment vessel at Brennilis. A larger Freyssinet system of 12-0.6" seven-wire strands is being used for the prestressed reactor vessel at Wylfa, England. The 90 wire BBRV tendons are presently being used for the secondary containment structures at the Turkey Point, Palisades, Point Beach and the Oconnee plants. The 170 wire BBRV tendons are currently being used for the Fort St. Vrain reactor vessel. BBRV has the following US Licensees: S1

a. American Stress Wire Corp.

b. Prescon Corp.

c. Joseph T. Ryerson & Sons, Inc.

d. Western Concrete Structures

e. Prestressing Industries S1

Stress Steel is the US licensee for the SEEE 19-0.5" strand system. The 0.5" strands are the same as used in the Freyssinet systems. The SEEE strand was used as the prestressing for the French EDF-3 and EDF-4 reactor vessels. SEEE has recently developed a 28-0.5" strand tendon based on the same principles as the 19-0.5" strand tendon. The steel strand will conform to "Specifications for Uncoated Seven-Wire Stress-Relieved Strand for Prestressed Concrete," ASTM A-416 with a minimum ultimate strength of 250,000 psi or seven-wire 270 K strands with minimum ultimate strength of 270,000 psi. The minimum yield strength for all strands will not be less than 85 percent of the specified minimum breaking strength.

The BBRV system is presented as the basis for licensing of the Rancho Seco plant. If in the final design, a change in system is desired, SMUD will apply for an amendment. The prestressing wire used in the tendons will conform to "Specifications for Uncoated Stress-Relieved Wire for Prestressed Concrete," ASTM A-421. Anchorages will develop 100 percent of guaranteed minimum ultimate strength of tendons. The tendons will be housed in ungalvanized, spiral-wrapped, semi-rigid, corrugated sheath and will be greased but not grouted. The prestressing wire will be protected against atmospheric corrosion during its shipment and installation, and during the life of the structure. Prior to shipment the wire will 4

S1 be coated with a thin film of petrolatum containing rust inhibitors, such as Dearborn Chemical Company 500 R. The interior surface of the sheathing is also coated with the same material during manufacture to protect against rusting during shipment, storage, and prior to filling with a grease-like material. The sheathing filler material used for permanent corrosion protection is a modified, thixiotropic, refined petroleum oil base product such as Dearborn Chemical Company NO-Ox-Id CM Casing Filler. It has a proven history of serving as a casing filler for cased pipelines under railroads and highways. The material will be introduced into the sheathing after stressing by pumping at ambient temperature.

The tendon anchorages will develop the minimum guaranteed ultimate tensile strength of the prestressing steel without permanent deformation and without excessive slip. The unit compressive stress developed by the bearing plate under the anchorage will be in conformance with ACI Code 318-63.

Dynamic earthquake loading acceptance standards of the tendons and anchorages are based on 500 cycles of rapid loading from a stress level of 0.70 f's to a stress level of 0.75 f's and back again in cycles of 0.1 second.

The number of cycles where the peaks exceed one half of the maximum value falls in the range between 20 and 30 cycles. A highly conservative factor was applied to the number of observed cycles to provide a margin of safety comparable to the reliance placed on the anchorage system in developing the tendon.

Earthquake, wind, and accident loadings will not generate more than 100 cycles of maximum stress variations during the life of the plant. In addition, the stress level due to these loadings will be between 0.60 f's to 0.64 f's, which is on the safe side. Of this range of 0.04 f's only 10 percent is actually due to seismic loading.

Anchorage performance requirements are now established by the Seismic Committee of the Prestressed Concrete Institute and published in their journal of June, 1966. These requirements are as follows and will be met by the tendon system:

"All anchors of unbonded tendons should develop at least 100 percent of the guaranteed ultimate strength of the tendon. The anchorage gripping shall function in such a way that no harmful notching effect would occur on the tendon. Any such anchorage system used in earthquake areas must be capable of maintaining the prestressing force under sustained and fluctuating load and under the effect of shock. Anchors should also possess adequate reserve strength to withstand any overstress to which they may be subjected during the most severe probable earthquake. Particular attention should be directed to accurate positioning and alignment of end anchorages."

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The bearing plate is included as a part of the prestressing system and at its interface with the concrete there is one of the greater interactions with the structures. The average contact pressure between bearing plate and concrete is limited to those permissible by ACI 318-63. The maximum contact pressure exceeds the average at locations nearest where the end anchor contacts the bearing plate. This results from a bending of the bearing plate. It is possible that the bending stress near the end anchor will reach yield since concrete differential creep will increase the bending of the plate from its initially loaded condition, and the largest bending stress in the bearing plate results from a yield moment for the plate. Although a difficult analytical problem, because of the inelastic nature of the materials and difficulty in defining boundary conditions, long experience and testing has shown that concrete reinforcing as used under column base plates and prestressing bearing plates, coupled with the bearing and base plate design approach, result in bearing and base plates that have not failed even though exposed to sub-zero temperatures in heavy industrial structures such as outdoor steel mill crane columns, power plant coal crusher structures and trestle columns. Also, communications with the chief engineer of an established prestressing company indicates that they have not experienced bearing plate failures when stressing tendons to $0.8f'_s$, the highest that had been planned to be imposed on the bearing plates, even while stresssing in sub-zero weather. Since the bearing plates are designed and tested to proven standards; are not subjected to large cycling loads or repeated impact loads; and are used in a climate less extreme than those for which the standards have been proven, it appears that they can satisfy the performance requirements. 4

In determining the appropriateness of applying NDT requirements, consideration has been given to:

(1.) The proven history, as cited above, for the use of both structural steel and post-tensioning systems under similar or worse environments to that of this application.

(2.) The A-36 or similar steel used for the bearing plates is in general a better quality steel than the A-7 that has the proven history cited above.

(3.) The fact that neither the AISC Structural Steel Code nor the ACI 318-63 has ever seen fit to incorporate such requirements in those code.

The conclusion that is drawn from the above discussion is that brittle fracture is not a problem for post-tensioning bearing plates.

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Certified test results of engineering data for the selected post-tensioning system will be furnished upon selection of a vendor. These results will include tests of ultimate strength, yield strength, wire area reduction, jacking stress, initial prestress, effective prestress, stress-strain relations, and elongation at rupture, and dynamic fatigue characteristics of the anchorages.

In addition, a number of assemblies will be made up with tendons and anchorages as they would be for final installation. These sample tendons will be used in static tests to failure, as appropriate.

5.1.3.4 Steel Liner Plate

The containment structure will be lined with welded steel plate conforming to ASTM-A442, Grade 60, to ensure low leakage. This steel has a minimum yield strength of 30,000 psi and a minimum elongation in an 8 inch specimen of 22 percent.

The design, construction, inspection and testing of the liner plate, which acts as a leak tight membrane and is not a pressure vessel, is not covered by any recognized code or specification. However, the liner plate and structural shapes will be supplied to the requirements of ASTM A-442 "Carbon Steel Plates with Improved Transition Properties" and ASTM A-20, "Specification for General Requirements for Delivery of Rolled Steel Plates of Flange and Firebox Qualities."

All components of the liner which must resist the full design pressure, such as penetrations, are selected to meet the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, of the ASME Code. ASTM A-516 Grade 60 or 70 made to ASTM A-300 is typical of a steel which meets these requirements and will be used as a plate material for penetrations. This material has excellent weldability characteristics and as much ductility as is obtainable in any commercially available pressure vessel quality steel.

In accordance with ASME Code Case 1347, allowable stresses for A-516 Grade 60 and 70 are the same as those permitted for A-201 Grade B and A-212 Grade B, respectively.

The A-442 material was chosen on the basis that it has sufficient strength as well as ductility to resist the expected stresses from design criteria loading and at the same time preserve the required leak tightness of the containment.

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The liner plate is designed to function only as a leak tight membrane. It is not designed to resist the tension stresses from internal applied pressure which may result from any credible accident conditions. The structural integrity of the containment is maintained by the prestressed, post-tensioned concrete. Since the principal applied stress to the liner plate membrane, from shrinkage and creep of the concrete, will be in compression and no significant applied tension stresses are expected from internal pressure loading, there is no need to apply special nil ductility transition temperature criteria to the liner plate material. On the other hand, all material for containment parts which must resist applied internal pressure stresses, such as penetrations, shall be impact tested in accordance with the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, of the ASME Code.

A-442 steel is readily weldable by all of the commercially available arc and gas welding processes.

A fundamental requirement for fabrication and erection of the liner plate is that all welding procedures and welding operators be qualified by tests as specified in Section IX of the ASME Code. This Code requires testing of welded transverse root and face bend samples in order to verify adequate weld metal ductility. Specifically, Section IX of the Code requires that transverse root and face bend samples be capable of being bent cold 180 degrees to an inside radius equal to twice the thickness of the test sample. Satisfactory completion of these bend tests is accepted as adequate evidence of required weld metal and plate material compatibility.

Mill test results will be obtained for the liner plate material. The plate will be visually checked for thickness, possible laminations and pitting.

The surfaces of the liner plate not to be embedded in concrete will be protected by an initial surface cleaning and prime coat of paint applied at the fabrication plant to protect it until installation. A suitable finish paint will be applied to the exposed surface after the plate is installed. 1

5.1.4 CONTAINMENT STRUCTURE DESIGN CRITERIA

Safety of the structure under extraordinary circumstances and performance of the containment structure at various loading stages are the main considerations in establishing the structural design criteria.

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The two basic criteria are:

a. The integrity of the liner plate shall be guaranteed under all loading conditions.

b. The structure shall have a low-strain elastic response such that its behavior will be predictable under all design loadings.

The strength of the containment structure at working stress and over-all yielding will be compared to various loading combinations to ensure safety. The containment structure will be examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion to ensure proper performance. The structure will be designed to meet the performance and strength requirements under the following conditions:

a. Prior to prestressing

b. At transfer of prestress

c. Under sustained prestress

d. At design loads

e. At factored loads

S1 Deviations in allowable stresses for the design loading conditions in the working stress method will be permitted if the factored load capacity criteria are fully satisfied. All design will be in accordance with the ACI Code 318-63 unless otherwise stated herein.

No special design bases are required for the design and checking of the base slab. It will act primarily in bending rather than membrane stress. This condition is covered by the ACI Code 318-63. The loads and stresses in the cylinder and dome will be determined as described below.

5.1.4.1 Design Method

The structure will be analyzed using a finite element computer program for individual and various combinations of loading cases of dead load, live load, prestress, temperature and pressure. The computer output will include direct stresses, shear stresses, principal stresses, and displacements of
4 each nodal point.

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Stress plots which show the total stresses from appropriate combinations of
loading cases will be made and areas of high stress will be identified. The
modulus of elasticity will be corrected to account for the nonliner stress-
strain relationship at high compression, if necessary. Stresses then will 4
be recomputed if these are sufficient areas which require attention.

In order to consider creep deformations, the modulus of elasticity of con-
crete under sustained loads such as dead load and prestress will be differ- S1
entiated from the modulus of elasticity of concrete under instantaneous
loads such as internal pressure and earthquake loads.

The forces and shears will be added over the cross-section and the total 4
moment, axial force, and shear will be determined. From these values, the
straight-line elastic stresses will be computed and compared to the allowable
values. The ACI 318-63 design methods and allowable stresses will be used
for concrete and prestressed and non-prestressed reinforcing steel except as
noted in these criteria.

5.1.4.2 Loads Prior to Prestressing

Under this condition the structure will be designed as a conventionally reinforced concrete structure. It will be designed for dead load, live loads (including construction loads), and a reduced wind load. Allowable stresses will be according to ACI 318-63 Code.

5.1.4.3 Loads at Transfer of Prestress

The containment structure will be checked for prestress loads and the
stresses compared with those allowed by the ACI 318-63 Code with the follow-
ing exceptions: ACI 318-63, Section 26, allows concrete stress of 0.60 f'_{ci}
at initial transfer. In order to limit creep deformations, the membrane
compression stress will be limited to 0.30 f'_{ci} whereas in combination with 4
flexural compression the maximum allowable stress will be limited to 0.60 f'_{ci}
per the ACI Code.

For local stress concentrations with non linear stress distribution as
predicted by the finite element analysis, 0.75 f'_{ci} will be permitted when 4
local reinforcing is included to distribute and control these localized
strains. These high local stresses are present in every structure but they
are seldom identified because of simplifications made in design analysis.
These high stresses are allowed because they occur in a very small percentage
of the cross-section, are confined by material at lower stress and would
have to be considerably greater than the values allowed before significant
local plastic yielding would result. Bonded reinforcing will be added to 4
distribute and control these local strains.

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S1 Membrane tension and flexural tension will be permitted provided they do not jeopardize the integrity of liner plate. Membrane tension will be permitted to occur during the post-tensioning sequence but will be limited to 1.0 $\sqrt{f'_c}$. The stress in the liner plate due to combined membrane tension and flexural tension will be limited to 0.5 fy. When there is flexural tension, but no membrane tension, the section will be designed in accordance with section 2605 (a) of the ACI Code.

Shear criteria will be in accordance with the ACI 318-63 Code, Chapter 26 as modified by the equations shown in paragraph 5.1.4.6, using a load factor of 1.5 for shear loads.

5.1.4.4 Loads Under Sustained Prestress

4 The conditions for design and the allowable stresses for this case will be the same as above except that the allowable tensile stress in non-prestressed reinforcing will be limited to 0.5 fy. ACI 318-63 limits the concrete compression to 0.45 f'_c for sustained prestress load. Values of 0.30 f'_c and 0.60 f'_c will be used as described above, which bracket the ACI allowable value. However, with these same limits for concrete stress at transfer of prestress, the stresses under sustained load will be reduced due to creep.

5.1.4.5 At Design Loads

This loading case is the basic "working stress" design. The containment structure will be designed for the following loading cases:

(a) $D+F+L+T_O$

(b) $D+F+L+P+T_A$

4 Where:

D = Dead Load

L = Appropriate Live Load

F = Appropriate Prestressing Load

P = Pressure Load (varies with time from design pressure to no pressure)

S1 T_O= Thermal Loads due to Operating Temperature

T_A= Thermal Loads Based on a Temperature Corresponding to a Pressure P

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Sufficient prestressing will be provided in the cylindrical and dome portions of the vessel to eliminate membrane tensile stress (tensile stress across the entire wall thickness) under design loads. Flexural tensile cracking will be permitted but will be controlled by bonded unprestressed reinforcing steel. The control of cracking under operating condition denoted by equation (a) above will be in accordance with the provision of ACI 318-63.

Under the design loads the same performance limits stated in 5.1.4.3 will apply with the following exception: S1

(a) If the net membrane compression is below 100 psi it will be neglected and a cracked section will be assumed in the computation of flexural bonded reinforcing steel. The allowable tensile stresses in bonded reinforcing will be 0.5 fy.

(b) When the maximum flexural stress does not exceed $6\sqrt{f'_c}$ and the extent of the tension zone is no more than 1/3 the depth of the section, bonded reinforcing steel will be provided to carry the entire tension in the tension block. Otherwise, the bonded reinforcing steel will be designed assuming a cracked section. When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the bonded reinforcing steel will be 0.5 fy minus the stress in bonded reinforcing due to the thermal gradient as determined in accordance with the method of ACI-505.

(c) The problem of shear and diagonal tension in a prestressed concrete structure should be considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is used to eliminate membrane tensile stress, membrane principal tension is not critical at design loads. Membrane principal tension due to combined membrane tension and membrane shear is considered under 5.1.4.6.

Flexural principal tension is the tension associated with bending in planes perpendicular to the surface of the shell and shear stress normal to the shell (radial shear stress). S1

The present ACI 318-63 provisions of chapter 26 for shear are adequate for the design purposes with proper modifications as discussed under 5.1.4.6, using a load factor of 1.5 for shear loads. S1

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Crack control in the concrete will be accomplished by adhering to the ACI-ASCE Code Committee standards for the use of reinforcing steel. These criteria are based upon a recommendation of the Prestressed Concrete Institute, and are as follows:

0.25 percent reinforcing shall be provided at the tension face for small members

4 0.20 percent for medium size members

0.15 percent for large members

S1 A minimum of 0.25 percent bonded steel reinforcing will be provided in two perpendicular directions on the exterior faces of the wall and dome for proper crack control.

4 The liner plate is attached on the inside faces of the wall and dome. Since, in general, there is no tensile stress due to temperature on the inside faces, bonded reinforcing steel is not necessary at the inside faces.

S1 5.1.4.6 Factored Loads

4 The structure will be checked for the factored loads and load combinations given below.

The load factors are the ratio by which loads will be multiplied for design purposes to assure that the load/deformation behavior of the structure is one of elastic, low strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. It also places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads.

4 The final design of the containment structure will satisfy the following load combinations and factors:

(a) $C = 1/\phi \ (1.05D+1.5P+1.0T_A+1.0F)$

S1 (b) $C = 1/\phi \ (1.05D+1.25P+1.0T_A+1.25H+1.25E+1.0F)$

4 (c) $C = 1/\phi \ (1.05D+1.25H+1.0R+1.0F+1.25E+1.0T_o)$

(d) $C = 1/\phi \ (1.0D+1.0P+1.0T_A+1.0H+1.0E'+1.0F)$

S1 (e) $C = 1/\phi \ (1.0D+1.0H+1.0R+1.0E'+1.0F+1.0T_o)$

4 (Wind, W, replaces earthquake, E, where wind stresses control) ~~00199~~

0283

Where: C = required capacity of the structure to resist factored loads. S1

Ø = capacity reduction factor (defined in Section 5.1.4.7)

D = dead loads of structures and equipment plus any other permanent loading contributing stress, such as hydrostatic or soil. In addition, a portion of live load is added when it includes items such as piping, cable and trays suspended from floors. An allowance is made for future additional permanent loads.

P = design accident pressure load

F = effective prestress loads

R = force or pressure on structure due to rupture of any one pipe

H = force on structure due to thermal expansion of pipes due to design conditions

T_O= thermal loads due to the temperature gradient through wall during operating conditions

T_A= thermal loads due to the temperature gradient through the wall and expansion of the liner. It is based on a temperature corresponding to the factored design accident pressure.

E = design earthquake load

E'= maximum earthquake load

W = wind load

Equation (a) assures that the containment will have the capacity to withstand pressure loadings at least 50 percent greater than those calculated for the postulated loss-of-coolant accident alone.

Equation (b) assures that the containment will have the capacity to withstand loadings at least 25 percent greater than those calculated for the postulated loss-of-coolant accident with a coincident design earthquake or wind.

Equation (c) assures that the containment will have the capacity to withstand earthquake loadings 25 percent greater than those calculated for the design earthquake coincident with rupture of any attached piping. 4

Equation (d) and (e) assure that the containment will have the capacity to withstand either the postulated loss-of-coolant accident or the rupture of any attached piping coincident with the maximum hypothetical earthquake.

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The stress in prestressing steel and bonded reinforcing steel will be limited to fy, where fy is the guaranteed minimum yield stress given in the appropriate ASTM specifications. The membrane compressive stress in concrete will
S1 be limited to 0.85 f'_c. The flexural compressive stress in concrete will be allowed to go up to f'_c (28 day ultimate compressive stress). The ultimate strength assumptions of the ACI code for concrete beams in flexure will be allowed. The peak strain in the concrete due to secondary moments, membrane loads and thermal loads will be limited to 0.003 inch/inch.

S1 The peak strain in the liner plate considering peak strain in the concrete and flexural strain in the liner will be limited to 0.005 inch/inch.

The following criteria will be used for the design of membrane shear.

Principal membrane tension in the concrete due to combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, will be calculated.

When the value of principal membrane tension exceeds $3\sqrt{f'_c}$, the combination of reinforcing steel and prestressing steel will resist the calculated value
S1 of principal membrane tension without exceeding the above mentioned stress limitation.

When the value of principal membrane tension does not exceed $3\sqrt{f'_c}$ and when the principal concrete tension due to combined membrane tension, membrane shear and flexural tension due to bending moments or thermal gradients will exceed $6\sqrt{f'_c}$, bonded reinforcing steel will be provided in the following manner:

(a.) Thermal flexural tension - Bonded reinforcing steel will be provided in accordance with the methods of ACI-505. The
S1 minimum area of steel provided will be 0.25 percent in each direction.

(b.) Bending moment Tension - Sufficient bonded reinforcingsteel will be provided to resist the moment on the basis of cracked section theory using the yield stresses stated above with the following exception:

When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the reinforcing steel will be fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the methods of ACI-505.

~~00201~~

0285

Shear stress limits and shear reinforcing for radial shear will be in accordance with chapter 26 of ACI 318-63 with the following exceptions: 4

a. Formula 26-12 of the code shall be replaced by

$$V_{ci} = K\ b'd\ \sqrt{f'_c} + M_{cr}\left(\frac{V}{M'}\right) + V_i \qquad (1)$$

3

where

$$K = \left[1.75 - \frac{0.036}{np'} + 4.0\ np'\right]$$

but not less than 0.6 for $p' \geqq 0.003$. 4

For $p' < 0.003$, the value of K shall be zero.

$$M_{cr} = \frac{I}{Y}\left[6\sqrt{f'_c} + f_{pe} + f_n + f_i\right]$$

f_{pe} = Compressive stress in concrete due to prestress applied normal to the cross-section after all losses, (including the stress due to any secondary moment) at the extreme fibre of the section at which tension stresses are caused by live loads.

f_n = Stress due to axial applied loads, (f_n shall be negative for tension stress and positive for compression stress).

f_i = Stress due to initial loads, at the extreme fibre of a section at which tension stresses are caused by applied loads, (including the stress due to any secondary moment. f_i shall be negative for tension stress and positive for compression stress.) 3

$$n = \frac{505}{\sqrt{f_c'}}$$

$$p' = \frac{As'}{bd}$$

V = Shear at the section under consideration due to the applied loads.

0286

M' = Moment at a distance d/2 from the section under consideration, measured in the direction of decreasing moment, due to applied loads.

V_i = Shear due to initial loads (positive when initial shear is in the same direction as the shear due to applied loads.)

Lower limit placed by ACI-318-63 on V_{ci} as 1.7 b'd $\sqrt{fc'}$ will not be applied.

b. Formula 26-13 of the code shall be replaced by

3 $$V_{cw} = 3.5\ b'd\ \sqrt{fc'} \left(\sqrt{1 + \frac{f_{pc} + f_n}{3.5\ \sqrt{f'_c}}} \right) \qquad (2)$$

The term f_n is as defined above. All other notations are in accordance with chapter 26, ACI-318-63.

(1) This formula is based on the recent tests and work done by Dr. A. H. Mattock of the University of Washington.

(2) This formula is based on the commentary for Proposed Redraft of section 2610-ACI-318 by Dr. A. H. Mattock, dated December, 1962.

When the above mentioned equations show that allowable shear in concrete
4 is zero, radial horizontal shear ties will be provided to resist all the calculated shear.

5.1.4.7 Yield Capacity Reduction Factors

The yield capacity of all load carrying structural elements will be reduced by a yield capacity reduction factor (Ø) as given below. The justification for these numerical values is given in Appendix 5-E. This factor will provide for "the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision while individually within required tolerance and the limits of good practice, occasionally may combine to result in undercapacity" (refer to footnote on page 66 of ACI 318-63 Code).

Yield Capacity Reduction Factors:

Ø = 0.90 for concrete in flexure

Ø = 0.85 for tension shear bond and anchorage in concrete

Ø = 0.75 for spirally reinforced concrete compression members

Ø = 0.70 for tied compression members

0287 ~~00283~~

Ø = 0.90 for fabricated structural steel

Ø = 0.90 for reinforcing steel in direct tension

Ø = 0.90 for welded or mechanical splices of reinforcing steel 4

Ø = 0.85 for lap splices of reinforcing steel

Ø = 0.95 for prestressed tendons in direct tension

5.1.4.8 Prestress Losses

In accordance with the ACI Code 318-63, the design will provide for prestress losses caused by the following effects:

a. Seating of anchorage

b. Elastic shortening of concrete

c. Creep of concrete

d. Shrinkage of concrete

e. Relaxation of prestressing steel stress 4

f. Frictional loss due to intended or unintended curvature in the tendons.

All of the above losses can be predicted with a reasonable degree of accuracy.

The environment of the prestress system and concrete is not appreciably different, in this case, from that found in numerous bridge and building applications. Considerable research has been done to evaluate the above items and is available to designers in assigning the allowances. Building code authorities consider it acceptable practice to develop permanent designs based on these allowances.

5.1.4.9 Liner Plate Criteria

The design criteria which will be applied to the containment liner to meet the specified leak rate under accident conditions are as follows:

a. That the liner be protected against damage by missiles. (See 5.1.4.10)

b. That the liner plate strains be limited to allowable values that have been shown to result in leak tight vessels or pressure piping.

c. That the liner plate be prevented from developing significant distortion.

d. That all discontinuities and openings be well anchored to accommodate the forces exerted by the restrained liner plate, and that careful attention be paid to details of corners and connections to minimize the effects of discontinuities.

Pressure vessels, pressure piping, high pressure hydraulic tubing, and similar containers are made by cold forming, drawing, and dishing operations where strains may approach the elongation capacity of the material. (For
S1 mild steel at failure, this elongation varies from 15 percent to 30 percent.) These forming operations result in high strains both in tension and compression. Vessels and piping components manufactured by these methods have a history of high leak tight integrity proving that subjecting the steel material to high strain does not affect its leak tight integrity.

The best basis for establishing allowable liner plate strains is considered to be that portion of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically, the following sections have been adopted as guides in establishing allowable strain limits:

a. Para. N-412 (m)	Thermal Stress (2)
b. Para. N-414.5	Peak Stress Intensity Table N-413 Fig. N-414, N-415 (A)
c. Para. N-412 (n)	
d. Para. N-415.1	

Implementation of the ASME design criteria requires that the liner material be prevented from experiencing significant distortion due to the thermal load and that the stresses be considered from a fatigue standpoint. (Para. N-412 (m) (2).)

The following fatigue loads will be considered in the design of the liner plate.

a. Thermal cycling due to annual outdoor temperature variations. Daily temperature variations will not enetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the liner plate. The number of cycles for this loading will be 40 cycles for the plant life of 40 years.

b. Thermal cycling due to containment interior temperature varying during the startup and shutdown of the reactor system. The number of cycles for this loading will be assumed to be 500 cycles.

c. Thermal cycling due to the MCA will be assumed to be one cycle. Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the liner plate. The attachment sleeve will be designed in accordance with ASME Section III fatigue considerations. All penetrations will be reviewed for a conservative number of cycles to be expected during the plant life.

The thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, Nuclear Vessels of the ASME Boiler and Pressure Vessel Code. The allowable stress in Figure N-415 (A) are for alternating stress intensity for carbon steels and temperatures not exceeding 700 F. In addition, the ASME Code further requires that significant distortion of the material be prevented.

In accordance with ASME Code Paragraph 412 (M) 2, the liner plate is retrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700 F and also satisfies the criteria for limiting strains on the basis of fatigue consideration.

Paragraph 412 (N) Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels. Because of the conservative factors it contains on both stress intensity and stress cycles, and its being a part of a recognized design code, Figure N-415 (A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below 10 cycles, 10 cycles is being used for an MCA instead of one cycle.

Establishing an allowable strain based on the one significant thermal cycle of the accident condition would permit an allowable strain (from Fig. N-415A) of approximately 2 percent. The strain in the liner plate at proportional limit will be approximately 0.1 percent. The liner plate will be allowed to go beyond proportional limit strains during the accident condition. Maximum allowable tensile or compressive strain has been conservatively set at 0.5 percent (compared to 2 percent shown above). The maximum predicted membrane strain in the liner plate during accident conditions has been found to be 0.25 percent. The maximum combined membrane and flexural strain is predicted to be 0.40 percent. S1

At the design accident pressure condition, there will be no tensile stress anywhere in the liner plate membrane. This is true both at the time of initial pressure release and under any later pressure temperature condition. The purpose of specifying an NDT temperature requirement is to provide protection against a brittle fracture or cleavage mode of failure. However, this type of failure is precluded by the absence of tensile stresses.

No allowable compressive strain value has been set for the test condition because the value will be less than that experienced under the accident conditions. The maximum predicted compressive membrane strain will be approximately 0.07 percent. S1

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The maximum allowable tensile strain will be 0.2 percent under test conditions. The predicted value will be very nearly zero.

The stability of the liner plate will be ensured by the stiffening and anchoring of the plate to the prestressed concrete structure.

3 (Deleted)

S1 The maximum compressive membrane strains are caused by accident pressure, thermal loading, prestress, shrinkage and creep. The maximum strains will not exceed .0025 in./in. and the liner plate will always remain in a stable condition.

The conservative design approach of the stiffening system used in the liner plate to prevent significant distortions at accident conditions, and stringent welding and weld inspection requirements will ensure that the leak tightness of the liner plate at accident conditions will not change from that at the test conditions.

In isolated areas the liner plate may have initial inward curvature due to construction. The anchors will be designed to resist the forces and moments induced when a section of the liner plate between anchors has initial inward curvature.

S1 The anchor and welds will be designed to accommodate the differential force or displacements caused by the partially restrained panel. The liner plate will be anchored at all discontinuities to eliminate excessive strains at the discontinuities. The forces in the liner plate at the discontinuities will be evaluated by use of the finite element computer program and the anchors will be designed to resist these forces.

At all penetrations the liner plate will be thickened to reduce stress concentrations in accordance with the ASME Boiler and Pressure Vessel Code 1965, Section III, Nuclear Vessels. The thickened portion of the liner plate will then be anchored to the concrete by use of anchor studs completely around the penetrations. For details of the penetrations see Figure 5.1-2. The sleeves, pipe cap, and all welds associated with the penetrations will be designed to resist all loads previously mentioned and also the prestress forces and internal design pressure.

S1 (Deleted)

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5.1.4.10 Missile Protection Criteria

High pressure reactor coolant system equipment which could be the source of missiles is suitably screened either by the concrete shield wall enclosing the reactor coolant loops, by the concrete operating floor or by special missile shields to block any passage of missiles to the containment walls. Potential missile sources are oriented so that the missile will be intercepted by the shields and structures provided. A structure is provided over the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

Missile protection will be provided to comply with the following criteria.

a. The containment and liner will be protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.

b. The engineered safeguards systems and components required to maintain containment integrity will be protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria will be developed and implemented using the following methods:

a. Components of the reactor coolant system will be examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.

b. Missile velocities will be calculated considering both fluid and mechanical driving forces which can act during missile generation.

c. The reactor coolant system will be surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe and designed to stop the missiles.

d. The structural design of the missile shielding will take into account both static and impact loads and will be based upon the state of the art of missile penetration data.

The types of missiles for which missile protection will be provided are:

a. Valve stems

b. Valve bonnets ~~00288~~

c. Instrument thimbles

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d. Various types and sizes of nuts and bolts

e. External missiles originating from loading conditions outlined in 5.1.2.5.

S1 (Deleted)

Protection is not provided for certain types of missiles for which postulated accidents are considered incredible because of the material characteristics, inspections, quality control during fabrication, and conservative design as applied to the particular component. Included in this category are missiles caused by massive, rapid failure of the reactor vessel, steam generator, pressurizer and main coolant pump casings.

5.1.4.11 Main Steam Turbine Missiles

The turbine-generator supplier* has made a study of potential missiles resulting from the failure of turbine-generator rotating elements operating at design (115% of normal operating) speed or, due to failure of components that control admission of steam to the turbine, operating at excessive overspeed. Historical reliability plus continued improvements in design indicate that excessive overspeed is the only envisaged cause of turbine-generator failure. An overspeed condition is very unlikely because of the redundancy and reliability of the turbine control and protection system and of the steam system. Nevertheless, the consequences of a turbine-generator runaway caused by all the steam admission valves stuck fully open upon a full load rejection have been analyzed.

On the low pressure turbine, an analysis was made of the bursting speed of each disc based upon an ultimate tensile strength 20 percent greater than the minimum guaranteed tensile strength. Each alloy steel disc is accurately machined and shrunk on to an alloy steel rotor. After shrinking on all the discs, the blading is installed, and the finished assembly is balanced to close tolerances. The completed rotor is tested at operating temperature and overspeed. Based upon experience and tests, it is concluded that the mode of failure of a disc, should it occur, is a rupture in two or four parts. The maximum speed at which the unit might run with no disc failure is 175 percent of nominal speed. At this speed the first discs will rupture. Immediately following the rupture of the first discs, the steam flow between the blades of the remaining discs would be significantly reduced, the turbine-generator would slow down, and further disc failures would not be anticipated. A disc rupture in the low pressure turbine would result in a missile which would strike and deeply deform the inner cylinder 1 causing some deformation of the inner cylinder 2 and of the outer cylinder but will be contained within the unit. Therefore, no outside missile is anticipated to be generated from the low pressure turbine.

*Westinghouse Electric Corporation

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On the high pressure turbine the maximum theoretical speed at which the unit may run, based on the admission steam thermodynamic properties and blade geometry, is 205 percent of normal speed. As shown here before, the maximum speed at which the unit may run, based upon the stress analysis of the low pressure unit, is 175 percent of nominal speed. The minimum bursting speed of the spindle, based on the minimum guaranteed mechanical properties of the spindle material, is 270 percent of nominal speed. Therefore, it is concluded that no missiles should be generated by the high pressure turbine due to a unit runaway.

A more complete description of the results of the analysis of potential turbine missiles is contained in a report titled Consideration of the Consequences of a Turbine Generator Failure in Which Missiles are Generated, included as Appendix 5-D.

5.1.5 STRUCTURAL DESIGN ANALYSIS

The containment structure will be analyzed by a finite element computer program for individual loading cases of dead load, live load, temperature, and pressure as described in 5.1.4.1.

The ACI-318-63 code design methods and allowable stresses will be used for concrete and prestressed and non-prestressed reinforcing steel except as noted herein.

5.1.5.1. Critical Design Areas

Based on a recent design study of prestressed concrete containment structures, it has been substantiated that the main areas for design analysis are:

a. The restraints at the top and bottom of the cylinder

b. The restraints at the edge of the spherical sector dome

c. The stresses around the large penetrations

d. The behavior of the base slab relative to an elastic foundation

e. The stresses due to transient temperature gradients in the liner plate and concrete

f. Stresses within the ring girder

g. Penetrations and concentrated loads

h. Seismic or wind loads

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5.1.5.2 Analytical Techniques

The containment structure analysis will be performed by the finite element method developed by E. L. Wilson, under sponsorship of National Science Foundation Research Grant G18986. This program has been further developed to apply to axisymmetric structures. Such a method of analysis is normally used only for thick-walled structures where conventional shell analysis yields inaccurate results. Good correlation has been demonstrated between the finite element analysis method and the test results for thick wall model vessels.

The design analysis for items a - f is done using the finite computer program because all of the conditions are axisymmetric. Items g and h are non-axisymmetric and are handled by techniques described in 5.1.5.5, 5.1.5.6 and 5.1.5.7. Effect of the non-axisymmetric loads will be combined with those obtained from the finite element technique.

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Conventional analysis of frames and trusses can be considered to be examples of the finite element method.

In the application of the method to an axisymmetric solid (e.g.,) a concrete containment structure, the continuous structure is replaced by a system of rings of triangular cross-section which are interconnected along circumferential joints. Based on energy principles, work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are unknowns of the system. A solution of this set of equations is inherent in the solution of the finite element system.

The finite element grid of the structure base slab will be extended down into the foundation material to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab.

The use of a finite element analysis will permit an accurate determination of the stress pattern at any location on the structure. The analysis method has been demonstrated on the following types of structures:

a. Arch dams (including a portion of the foundation)

b. Thick-walled prestressed concrete vessels

c. Spacecraft heat shields

d. Rocket nozzles

The computer program used in the analysis will handle the following inputs:

a. Seven different materials

b. Non-linear stress-strain curves for each material

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c. Any shape transient temperature curves

d. Any shape axisymmetric loading

The program outputs will be:

a. The direct stress and shear stress for each element

b. The principal stresses and their directions for each element

c. The deflections for each nodal point

An auxiliary computer program will plot stress curves based on the above analysis program outputs.

Additional information regarding this technique, the computer program employed, and a comparison of the results with other analytical methods is contained in Appendix 5-G.

5.1.5.3 Thermal Loads

The thermal loads are a result of the temperature differential within the structure. The design temperature gradients for this structure are shown on Figure 5.1-4. The finite element analysis was prepared so that when temperatures are given at every nodal point, stresses are calculated at the center of each element. This way the liner plate was handled as an integral part of the structure, having different material properties, and not as a mechanism which would act as an outside source to produce loading on the concrete portion of the structure.

The liner plate is designed to have plastic deformation as a result of prestressing and high thermal stresses.

The finite element method includes this analysis too, by successive approximations, changing the modulus of elasticity of those elements which are subject to stresses higher than the proportional limit.

The output of the computer analysis shows the effect of the thermal loads on liner plate and concrete. The liner plate and the inside of the concrete are subject to compressive stress and the outside of the concrete section is subject to tension. These tension stresses balance the compressive stresses so that, except close to any discontinuity, there is no resultant membrane force. That is, all the compressive forces in the liner plate are carried by the prestressed concrete and reinforcement near the outside surface of the structure.

The compressive stresses in the liner plate exceed the proportional limit in the case of the design basis accident. An increased temperature would keep the liner plate in plastic condition, but only a negligible additional stress could develop, and thermal stresses would stay unchanged.

5.1.5.4 Tendon Failure Analysis

S1 There will be approximately 156 vertical tendons and 201 hoop tendons. The hoop tendons will be placed in three 240° sections around the cylinder using three buttresses as anchorages. Therefore, failure of a hoop tendon or a series of adjoining tendons or spaced hoop tendons is limited between 240° segments of the containment vessel.

It is well known that all prestressed tendons are subject to the most critical stress during initial tensioning. There will be a loss of prestress on the order of 15 percent due to elastic and plastic losses, which will reduce the stress level. Even at the factored yield loads, the stress in the tendons will not be as high as during initial tensioning. Each of the tendons has been pre-tested at the time of initial jacking and the stress in the tendons under accident loading is only 80 percent of this jacking stress. This means that the possibility of tendon failure under design accident loading is quite remote.

Although it is felt that there is ample reserve capacity in the tendon and structure, the complex nature of the structural behavior makes it difficult to predict the effect of a hypothetical series of tendon failures until the final design is complete.

It is estimated that if two or three of the tendons fail during accident conditions, and if they are side by side or close together, it will not affect the integrity of the structure or the liner because the thick concrete walls will be sufficient to transmit the force from the adjoining tendons without resulting in any serious local stresses.

5.1.5.5 Stresses Near Equipment Openings

Analytical solutions for the determination of state of stress in the vicinity of equipment openings are obtained from reference to the following articles.

a. "State of Stress in a Circular Cylindrical Shell with a Circular Hole," by A. C. Eringen, A. K. Naghdi, and C. C. Thiel - Welding Research Council Bulletin No. 102, January, 1965.

b. Samuel Levy, A. E. McPherson and F. C. Smith, "Reinforcement of a Small Circular Hole in a Plane Sheet Under Tension," Journal of Applied Mechanics, June, 1948.

The analysis of the containment structure as a whole is first carried out without considering the openings in it. This analysis has been done by using the finite element program.

The containment structure with the opening in it is then analyzed in the following steps.

a. Formulation of differential equations for the shell in complex variable form with the center of the hole as the origin (See reference a above)

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b. Solution of the differential equations (See reference a above)

c. Evaluation of parameters in the solution (See reference a above)

d. Formulation of the boundary conditions based on the stresses obtained from the vessel analysis above without the hole

e. Calculation of membrane forces, moments, and shears around and at the edge of the opening

f. The wall thickness around the opening will then be increased and reinforced to carry the higher forces, moment, and shears. The affect of the thickening on the stress concentration factors will be considered using reference b above.

g. Evaluation of some of the effects of prestressing that are not handled in reference a above

h. Finally, the design will be checked to ensure that the strength of the reinforcement provided replaces the strength removed by the opening. This check is to maintain a good degree of compatibility between the general vessel shell and the area around the opening.

Details of the reinforcing and deflected strand pattern around the equipment hatch opening are shown on Figure 5.1-3.

The pattern of membrane stresses at design accident loading is not expected to be significantly different from the pattern of membrane stresses during the proof test, since the membrane stresses due to pressure are far more significant than those due to dead load or seismic load.

The deflection of the tendons does not significantly affect the stress concentrations. This is a plane stress analysis, and did not include the effect of the curvature of the shell; however, it gives an assurance of the correctness of the assumed membrane stress pattern caused by the prestressing around the opening.

The seismic load creates vertical membrane stress in the structure based on a cantilevered circular beam subjected to base accelerations. The membrane stresses at the opening are modified by appropriate stress concentration factors.

The temperature variation through the concrete wall creates a stress condition like one caused by a moment, constant in all directions on the continuous cylindrical or spherical surface. However, at any discontinuity, such as an opening, stress concentrations occur.

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Using the center of the opening as the reference point to relate the directions of moments, the radial moment is zero at the edge of the opening, there being no resistance against radial rotation. The hoop moment is highly increased, the outside fiber being forced to take the shape of a larger circle, while the inside fiber takes the shape of a smaller circle.

Away from the edge of the opening both moments gradually reach the constant value on the undisturbed portion of the cylinder.

In the case of 1.5P (prestress fully neutralized) + 1.0T (accident temperature) the cracked concrete with highly strained tension reinforcement constitutes a shell with stiffness decreased but still constant in all directions. In order to control the increased hoop moment around the opening, the hoop reinforcement should produce strength about twice that of the radial one.

In the case of accident temperature combined with low internal pressure, very small or no tension develops on the outside, so the thermal strains will be built up without the relieving effect of the cracks. However, as has already been stated elsewhere, the liner plate will reach its yield stress, and so will the concrete at the inside corner of the penetration, thereby relieving once again the very high stresses, but still carrying the high moment in the state of redistributed stresses.

For the analysis of the thermal stresses around the opening the same method was used as for the other loadings.

At the edge of the opening a uniformly distributed moment equal but opposite to the moment existing on the rest of the shell was applied, and evaluated using the methods of the preceding reference and the effects were superimposed on the stresses calculated by the computer using the finite element method for axisymmetric solids.

5.1.5.6 Seismic Analysis

The loads on the containment structure caused by earthquake will be determined as a result of a dynamic analysis of the structure. The dynamic analysis will be made on an idealized structure of lumped masses and weightless elastic columns acting as spring restraints. The analysis will be performed in two stages; the determination of the natural frequencies of the structure and its mode shapes, and the modal response of these modes to the earthquake by the spectrum response method. Appendix 5-A contains more details on the seismic design basis for this plant.

The natural frequencies and mode shapes are computed from the equations of motion of the lumped masses established in a stiffness of displacement method and are solved by iteration techniques by a fully tested digital computer program. The form of the equations is:

$$(K) \quad (\Delta) \quad = \quad \omega^2 \quad (M) \quad (\Delta)$$

(K) = Matrix of stiffness coefficients including the combined effects of shear and flexure

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(M) = Matrix of concentrated masses

(Δ) = Matrix of mode shape

ω = Angular frequency of vibration

The results of this computation are the several values of ω_n and mode shapes $(\Delta)_n$ for n = 1, 2, 3 --- m where m is the number of degrees of freedom (i.e., lumped masses) assumed in the idealized structure.

The response of each mode of vibration to the design earthquake is then computed by the response spectrum technique, as follows:

a. The base shear contribution of the n^{th} mode

$$V_n = W_n \; S_{an} \; (W_n \gamma)$$

Where:

W_n = Effective weight of the structure in the n^{th} mode computed from:

$$Wn = \frac{(\Sigma_x \; \Delta_{xn} \; W_x)^2}{\Sigma_x \; (\Delta_{xn})^2 \; W_x}$$ where the

subscript x refers to levels throughout the height of the structure.

ω_n = Angular frequency of the n^{th} mode

$S_{an} \; (\omega_n, \gamma)$ = Spectral acceleration of a single degree of freedom system with a damping coefficient of γ, obtained from the response spectrum.

b. The horizontal load distribution for the n^{th} mode is then computed as:

$$F_x = V_n \left(\frac{\Delta_{xn} \; W_x}{\Sigma_x \Delta_{xn} \; W_x} \right)$$

The several mode contributions are then combined to give the final response of the structure to the design earthquake.

c. Additional Considerations:

The number of modes to be considered in the analysis will be determined at the time of final design to adequately represent the structure being analyzed. Since the spectral response technique yields the maximum value of response for each mode, and these maxima do not occur at the same time, the response of the modes of vibration will be combined on a root-mean-square basis to obtain the most probable value of maximum response.

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The mathemetical model to be used in the earthquake analysis of the containment structure will be established with due consideration given to the rotational and translational stiffness of the surrounding soil.

The following values of damping and ground acceleration will be used in the analysis together with the natural periods to obtain spectral accelerations:

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Type of Motion	Design Earthquake % Damping	Maximum Hypothetical Earthquake % Damping
Structural	2%	5%
Translation	30%	30%
Rocking	5%	9%

5.1.5.7 Wind Analysis

The loads caused by the incident design wind on the containment structure are a function of the kinetic energy per unit volume of the moving air mass. The product of one-half of the air density and the square of the resultant design velocity results in the dynamic pressure of the design wind.

The dynamic pressure (PSF) for standard air at 0.07651 pcf corresponding to 15°C at 760 mm of mercury in terms of the velocity at the appropriate height zone is given by:

$$q = 0.002558\ V^2$$

Similarly, the design pressure to be mutliplied by the projected elevation
1 area to obtain the total force on the structure includes the effect of the
shape coefficient (C_d) and is given by:

$$p = q \times C_d = 0.002558V^2C_d$$

The shape coefficient includes the effect of drag, lift, shape, aspect ratio, and surface smoothness. The containment structure has an aspect ratio (h/d) of 1.48 and surface smoothness (t/d) of 1.57 percent. The shape coefficient is found to be:

$$C_d = 0.70$$

Where

h = Containment height above ground
t = Projection of buttresses
d = Maximum outside diameter of the containment structure

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The final design pressure including the variation of design wind velocity with height are as follows:

Height Above Ground (H)	Fastest Wind (V)	Design Pressure ($p = 0.002558V^2C_d$)
0-50 ft.	90 MPH	14.5 PSF
50-150 ft.	105 MPH	19.8 PSF
150-400 ft.	125 MPH	28.0 PSF

The variation of design pressure in the horizontal direction follows the ASCE recommendations of Table 4(f). Maximum positive and negative pressures in terms of the incident pressure at the appropriate height level are 1.0 p and -1.7 p respectively.

A gust factor of 1.1 will be taken in accordance with the report recommendations.

The final wind pressures will be applied to the structure as equivalent static loads.

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TABLE 5.1-1

CONCRETE PROTECTION FOR REINFORCEMENT AND PRESTRESSING TENDONS

LOCATION	TYPE OF STEEL	COVERAGE TO BE USED	ACI 318-63 INTERIOR	ACI 318-63 EXTERIOR
Dome	Reinforcing	18 & 14S 2-1/4 IN. Others 2-1/4 IN.	2-1/4 & 1-3/4 IN. 1-1/2 IN.	2-1/4 & 2 IN. 2 IN.
	Prestressing	6 IN.	1-1/2 IN.	2 IN.
Cylinder	Reinforcing	18 & 14S 2-1/4 IN. Others 2-1/4 IN.	2-1/4 & 1-3/4 IN. 1-1/2 IN.	2-1/4 & 2 IN. 2 IN.
	Prestressing	6 IN.	1-1/2 IN.	2 IN.
Base Mat	Bottom Reinforcing & Unformed Surfaces	18 & 14S 3 IN. Others 3 IN.		3 IN. 3 IN.
	Top Reinforcing	18 & 14S-2-1/4 IN. Others -2-1/4 IN.	2-1/4 & 1/3-4 IN. 1-1/2 IN.	2-1/4 & 2 IN. 2 IN.

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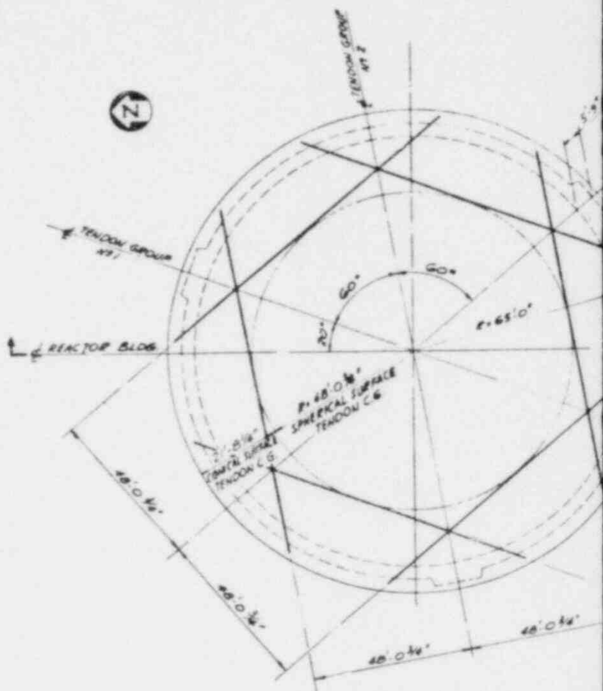
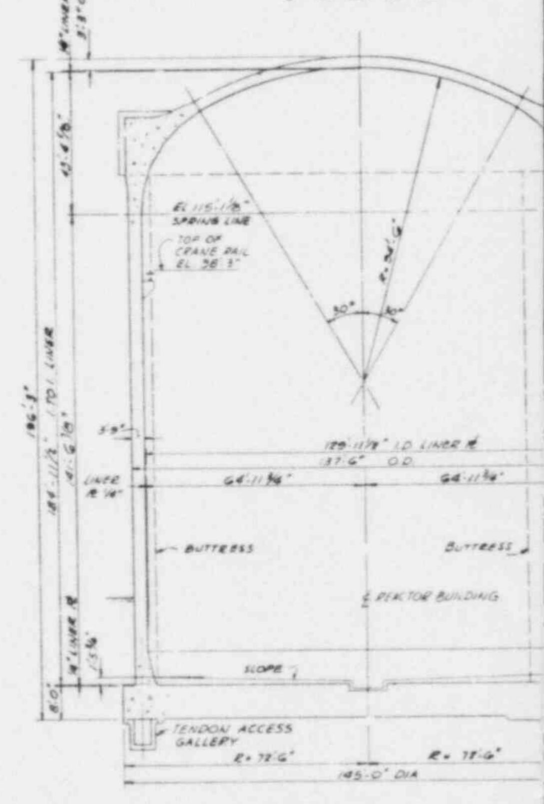


ROOF PLAN

0 10 20 30 40'



SECTION A

0 10 20 30 40'

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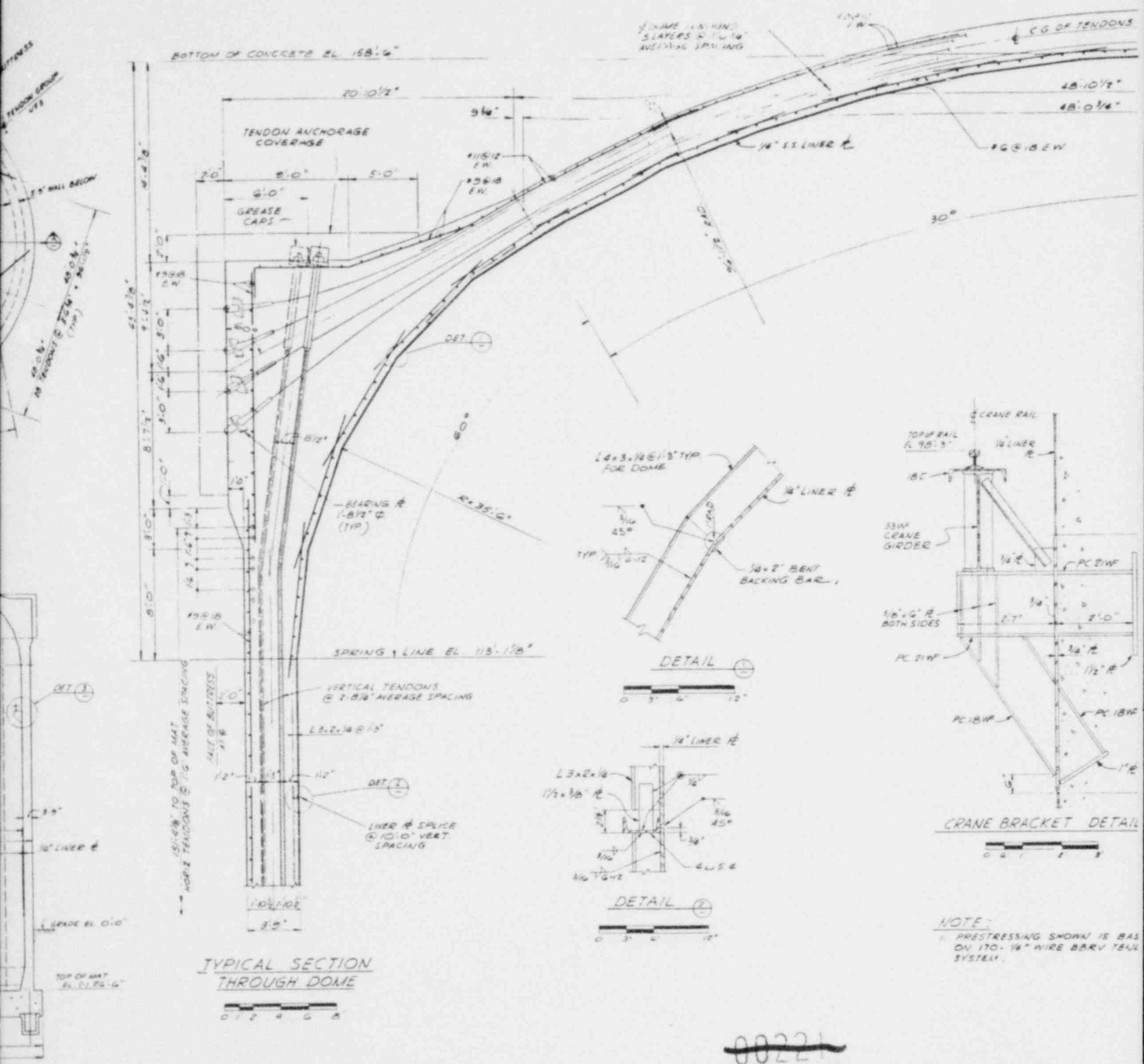

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FIGURE 5.1-1
REACTOR BUILDING
TYPICAL DETAILS
SHEET 1

SMUD
SACRAMENTO MUNICIPAL UTILITY DIS

Suppleme

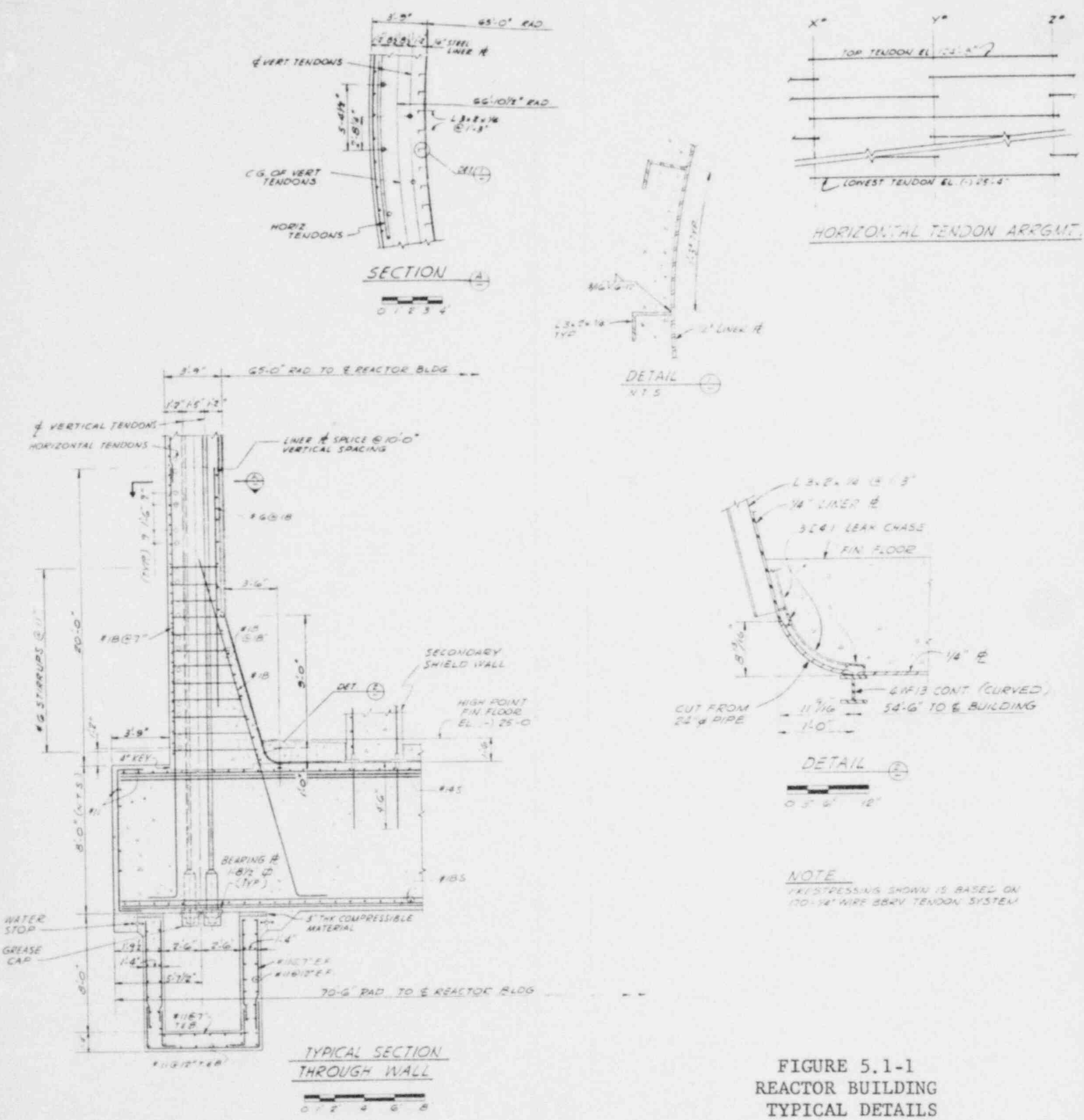

FIGURE 5.1-1
REACTOR BUILDING
TYPICAL DETAILS
SHEET 2

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SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

Supplement 1

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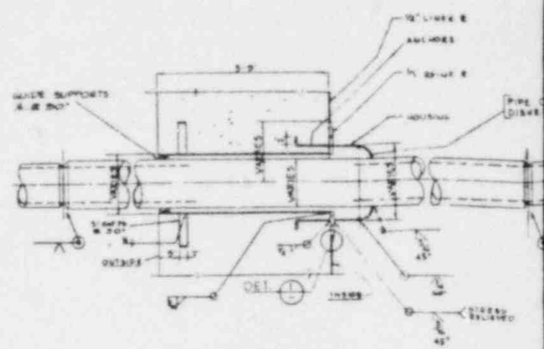
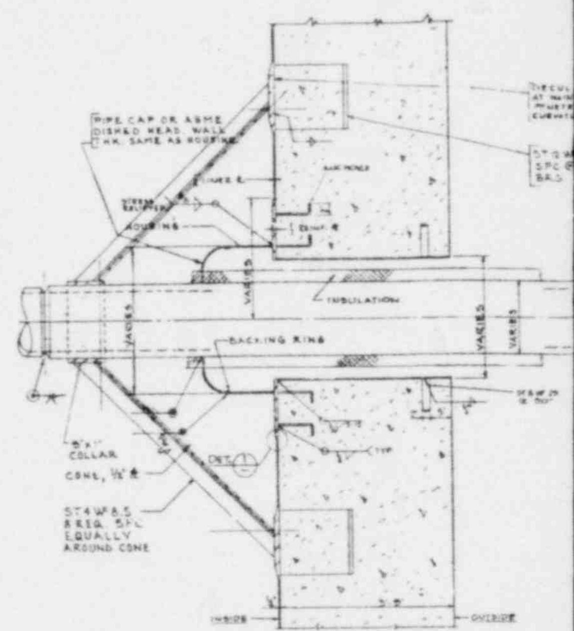
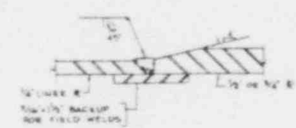


TYPE - I
TYP. COLD PIPE PENETRATION
LESS THAN 150° F.
NO SCALE



TYPE - II
TYP. HOT PIPE PENETRATION
GREATER THAN 150° F.
NO SCALE



DET. 1

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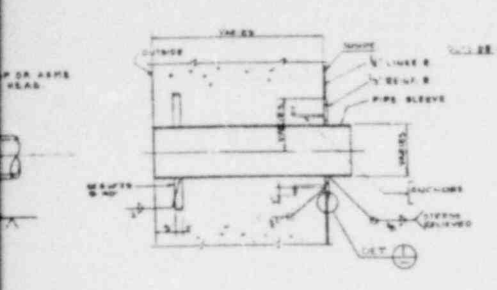
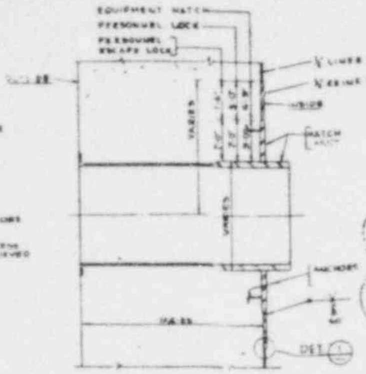
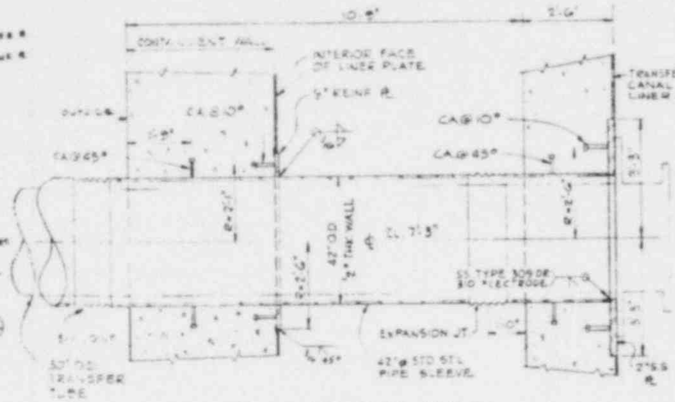


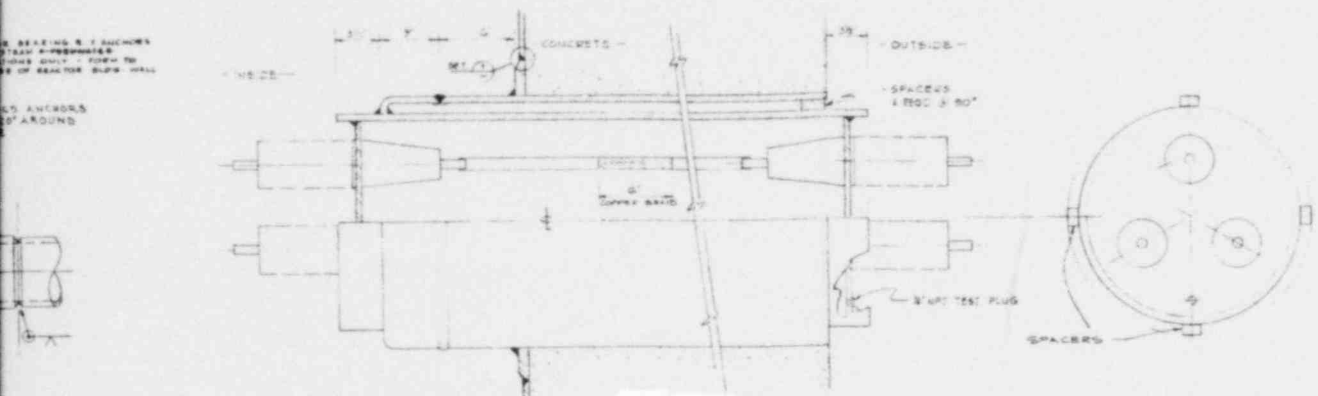
TYPE III
TYP ELECTRICAL PENETRATION
NO SCALE



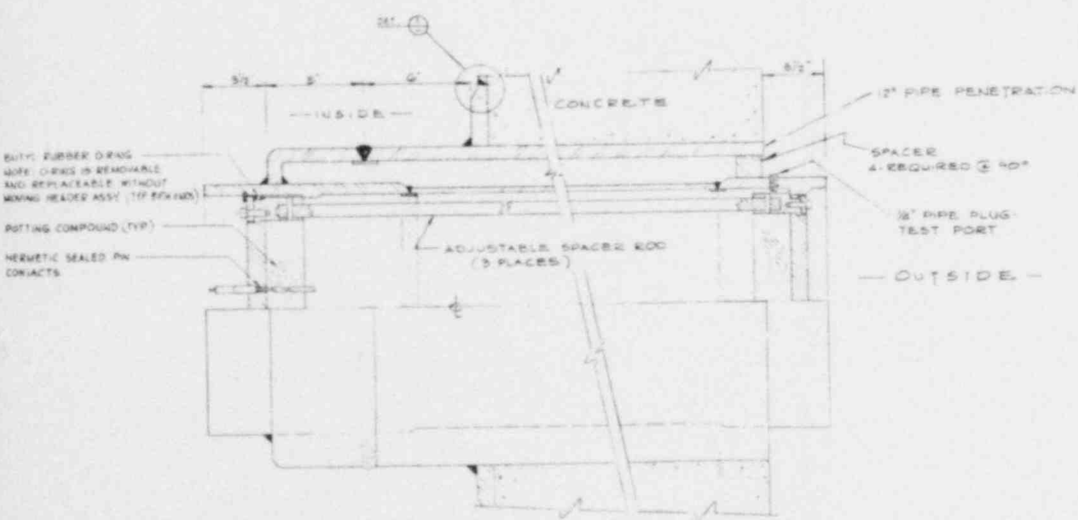
TYPE IV
HATCH PENETRATION
NO SCALE



FUEL TRANSFER TUBE PENETRATION



TYPICAL 6.16 [illegible] RICAL PENETRATION



TYPICAL LOW VOLTAGE POWER, CONTROL & INSTRUMENTATION PENETRATION
NO SCALE

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FIGURE 5.1-2
TYPICAL ELECTRICAL AND
PIPING PENETRATIONS

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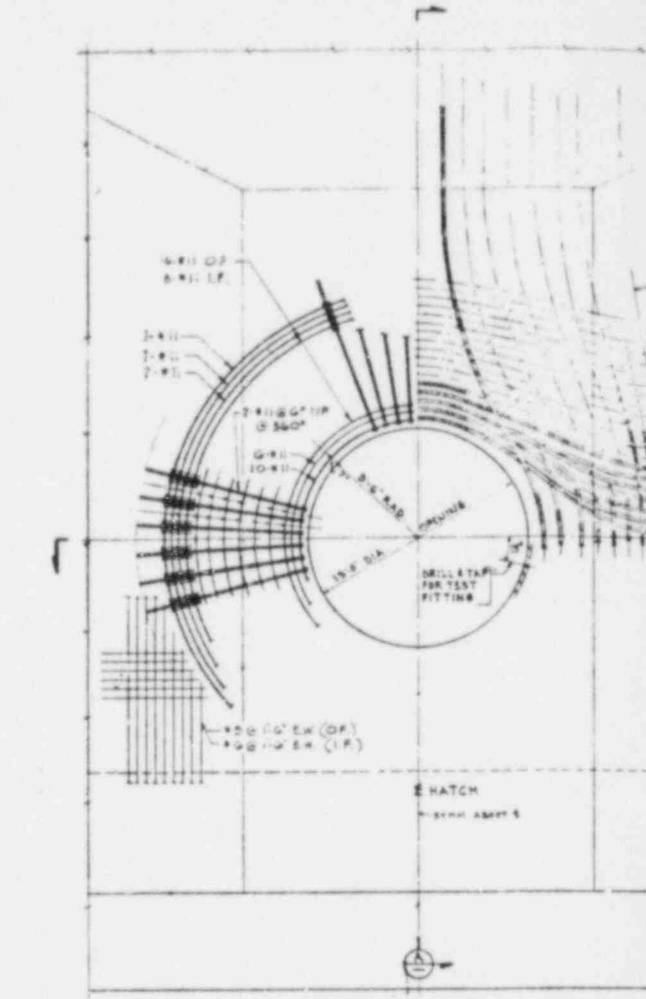
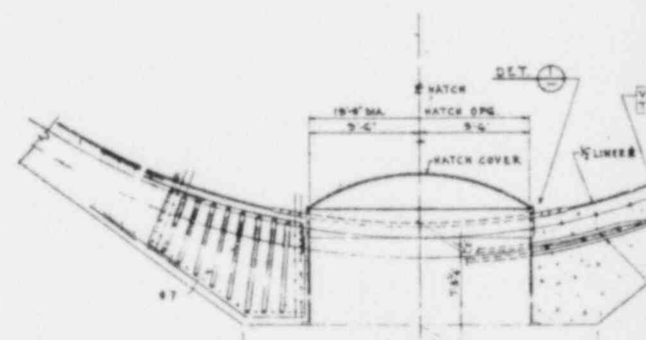
SACRAMENTO MUNICIPAL UTILITY DISTRICT



ELEVATION - EQUIPMENT HATCH



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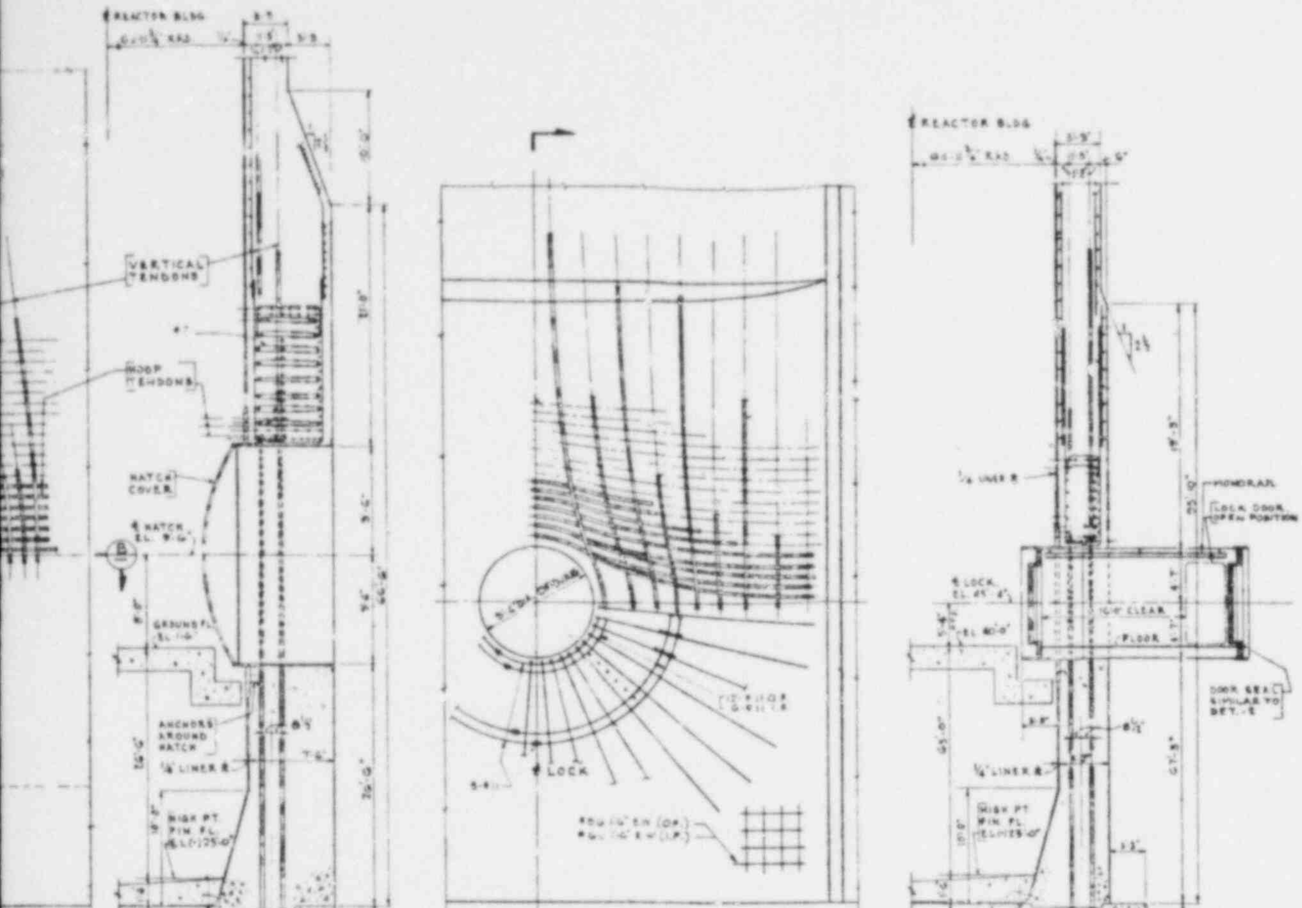

FIGURE 5.1-3
TYPICAL EQUIPMENT AND
PERSONNEL OPENINGS

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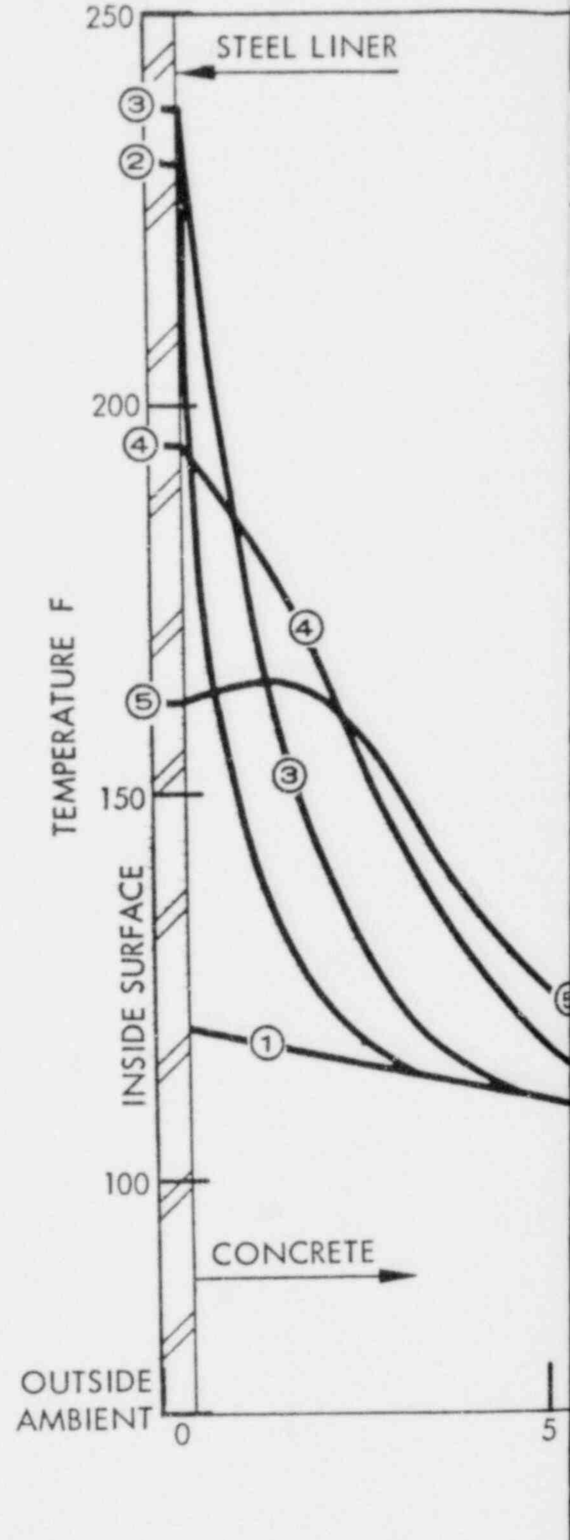
SACRAMENTO MUNICIPAL UTILITY DISTRICT


250
STEEL LINER
③
②
200
④
TEMPERATURE F
INSIDE SURFACE
⑤
150
①
100
CONCRETE
OUTSIDE AMBIENT
0
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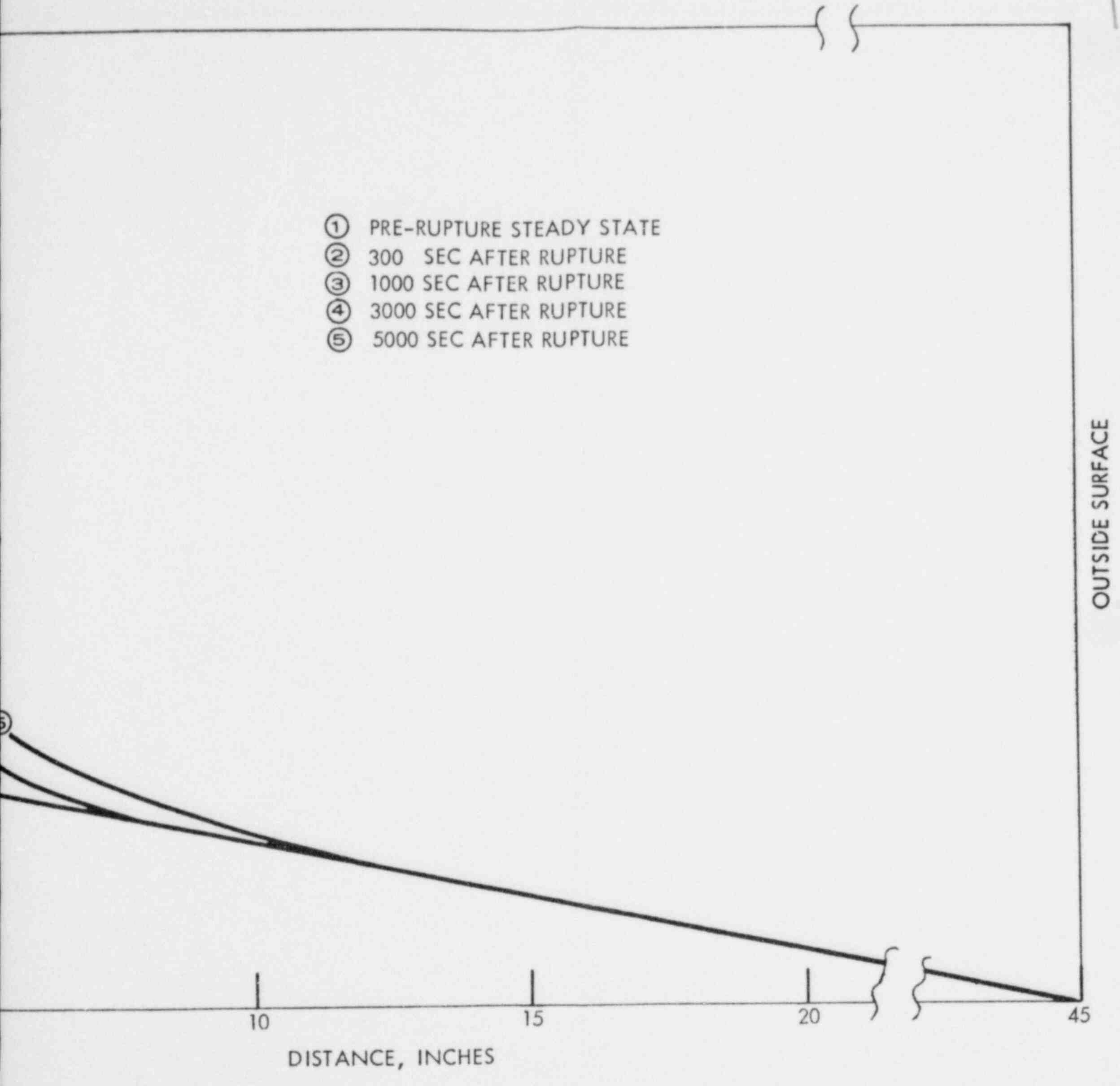

FIGURE 5.1-4
THERMAL GRADIENT ACROSS
CONTAINMENT WALL

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5.4 CONSTRUCTION PRACTICES AND QUALITY ASSURANCE

5.4.1 ORGANIZATION OF QUALITY ASSURANCE PROGRAM

Quality of both materials and construction of the containment structure will be assured by a continuous program of quality control and inspection by Bechtel Corporation, the Engineer-Construction Manager, and by the Applicant.

Qualified field supervisory personnel and inspectors will be assigned to the project to carry out the work in accordance with the specifications and drawings. Project design personnel will make frequent visits to the job site to coordinate the construction with the design. Inspectors will be experienced and thoroughly familiar with the type of work to be inspected, particularly in the field of prestressed concrete. The inspector will be given complete access to the work and will perform such examinations as are necessary to satisfy himself that the standards set forth in the applicable codes and specifications have been met.

Where material does not satisfy the standards, he will have the authority to stop work until the necessary alterations are made.

5.4.2 APPLICABLE CONSTRUCTION CODES

The following codes of practice will be used to establish standards of construction procedure.

a. ACI 301 - Specification for Structural Concrete for Buildings (proposed)

b. ACI 318 - Building Code Requirements for Reinforced Concrete

c. ACI 347 - Recommended Practice for Concrete Formwork 1

d. ACI 605 - Recommended Practice for Hot Weather Concreting

e. ACI 613 - Recommended Practice for Selecting Proportions for Concrete

f. ACI 614 - Recommended Practice for Measuring, Mixing and Placing Concrete

g. ACI 315 - Manual of Standard Practice for Detailing Reinforced Concrete Structures

h. Part UW - Requirements for Unfired Pressure Vessels Fabricated by Welding of Section VIII of the ASME Boiler and Pressure Vessel Code

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i. AISC - Steel Manual, Code of Standard Practice

j. PCI - Inspection Manual

k. AWS - Code for Welding in Building Construction (D 1.0-66)

l. AWS - Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction (D 12.1-61)

In every instance the construction procedure for the containment will equal or exceed the recommendations set forth in the foregoing publications. The extent to which each detailed process will exceed standard requirements cannot be described without incorporating all applicable job specifications and anticipating hypothetical construction problems and conditions. In general, however, wherever the applicable codes specify minimum and ideal criteria, the ideal will be incorporated into the specifications.

5.4.3 CONSTRUCTION MATERIALS INSPECTION AND INSTALLATION

Materials to be used in the containment structure include concrete materials, reinforcing steel, prestressing system materials, and steel liner plate. The user inspection and testing of each material will be as follows:

5.4.3.1 Concrete Materials

a. Cement

In addition to the tests required by the cement manufacturers, the following user tests will be performed:

(1) ASTM C 114 - Chemical Analysis

(2) ASTM C 115 - Fineness of Portland Cement

(3) ASTM C 151 - Autoclave Expansion

(4) ASTM C 191 - Time of Set

(5) ASTM C 109 - Compressive Strength

(6) ASTM C 190 - Tensile Strength

The purpose of the above tests is to ascertain conformance with ASTM Specification C 150. In addition, tests ASTM C 191 and ASTM C 109 will be repeated periodically during construction to check storage environmental effects on cement characteristics. The tests will supplement visual inspection of material storage procedures.

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b. Water Reducing Agents

A concrete testing laboratory will perform the necessary strength and shrinkage tests on various water reducing agents to establish the particular additive with the most desirable characteristics for this application.

c. Aggregates

User tests of concrete aggregate include the following:

Test		Basis For	Results to be Achieved
ASTM No.	Name		
C 131	Los Angeles Abrasion	ASTM Spec C-33	To conform with specification
C 142	Clay Lumps	ASTM Spec C-33	To conform with specification
C 117	Material Finer than #200 Sieve	ASTM Spec C-33	To conform with specification
C 87	Mortar Making Properties	ASTM Spec C-33	To conform with specification
C 40	Organic Impurities	ASTM Spec C-33	To conform with specification
C 289	Potential Reactivity (Chemical)	ASTM Spec C-33	To conform with specification
C 136	Sieve Analysis	ASTM Spec C-33	To conform with specification
C 88	Soundness	ASTM Spec C-33	To conform with specification
C 127	Specific Gravity and Absorption	ASTM Spec C-33	Mix Design Calculations
C 128	Specific Gravity and Absorption	ASTM Spec C-33	Mix Design Calculations
C 295	Petrographic	ASTM Spec C-33	To conform with specification

In addition to the foregoing initial user tests, a daily inspection control program will be carried on during construction to ascertain consistency in potentially variable characteristics such as gradation and organic content.

d. Water

Water to be used in concrete mixing will be sampled and analyzed by a qualified testing laboratory to assure conformance with specifications.

5.4.3.2 Concrete

a. Design Mix

Design mixes and the associated tests will be provided by a qualified concrete testing laboratory. The design of mixes will be in accordance with ACI 613 to obtain material proportions for the specified concrete.

During construction the field inspection personnel will make any minor modifications that may be necessitated by variations in aggregate gradation or moisture content.

b. Compressive Strength

Concrete strength, slump, and temperature inspections will be performed. The purpose of the tests and inspection is to ascertain conformance to specifications. The basis for the proposed inspection procedure is ACI Manual of Concrete Inspection with upgraded modifications to meet the more stringent requirements of this application.

5.4.3.3 Reinforcing Steel

a. Material

All reinforcing steel will be user-tested in accordance with ASTM specifications. Tests will include one tension and one bend test per heat or per mill shipment, whichever is less, for each diameter bar. Test samples will be obtained at the fabrication plant. High strength bars will be clearly identified prior to shipment to prevent any possibility of mix-up with lower strength reinforcing bars.

b. Mechanical Splices

The "Cadweld" inspection program is detailed in Appendix 5-C. Ordinary welded splices will not be used for main bars in the
4| containment structure, as stated in 5.1.3.2.

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c. Fabrication

Visual inspection of fabricated reinforcement will be performed to ascertain dimensional conformance with specifications and drawings.

d. Placement

Visual inspection of in-place reinforcement will be performed by the placing inspector to assure dimensional and location conformance with drawings and specifications.

5.4.3.4 Prestress System

a. Wires

Sampling and testing of the tendon material used in construction will conform to ASTM Standard A-421 or ASTM A-416. The following procedure will be used:

(1) Buttonhead rupture tests from each reel of wire will be made.

(2) Each size of wire from each mill heat and all strands from each manufactured reel to be shipped to the site shall be assigned an individual lot number and tagged in such a manner that each such lot can be accurately identified at the job site. Anchorage assemblies shall likewise be identified. All unidentified prestressing steel or anchorage assemblies received at the job site will be subject to rejection.

(3) Random samples as specified in the ASTM Standards stated above will be taken from each lot of prestressing steel to be used in the work. With each sample of prestressing steel wire or strand that is tested, there shall be submitted a certificate stating the manufacturer's minimum guaranteed ultimate tensile strength of the sample to be tested. Stress-strain curves will be plotted and the yield and tensile strength verified. For the prefabricated tendons, one completely fabricated prestressing test specimen 5 feet in length including anchorage assemblage, will be tested for each size of tendon contained in an individual manufacturing run. The anchorages will develop the minimum guaranteed ultimate strength of the tendon and the minimum elongation of the tendon material as required by the applicable ASTM specification.

Field inspection will ensure that there are no visible mechanical or metallurgical notches or pits in the tendon material.

b. Installation

All prestressing installation work shall be continuously inspected by a qualified inspector. All measuring equipment used for installation will be calibrated and certified by an approved independent testing laboratory. During stressing operations, records will be kept by the engineer for use in comparing force measurements with elongation for all tendons. The resultant cross-reference will provide a final check on measurement accuracy. Measurement accuracy and rejection allowances will be in accordance with ACI-318, Chapter 26.

c. Grease

Grease will be sampled after delivery and submitted to a qualified testing laboratory for chemical analysis to establish conformance with specifications.

5.4.3.5 Containment Liner

a. Steel Plate

Steel plate will be tested at the mill in full conformance to the applicable ASTM specifications. Certified mill test reports will be supplied for review and approval by the engineer.

There will be no impact testing done on the liner plate material. The purpose of impact testing is to provide protection against brittle failure. The possibility of a brittle fracture of the liner plate is precluded because at the design accident pressure condition there will not be any tensile stress anywhere in the liner plate whether there is instantaneous release of pressure or there is some time lag in temperature load application. Therefore, the NDT temperature of the liner plate loses significance.

b. Fabrication and Installation

Welding inspection will conform to the quality control inspection procedure described in detail by Appendix 5-H.

Dimensional tolerances will be checked by an installation inspector to prevent unanticipated installation deformations. The radial tolerance will limit variation of the design
1 radius to $\pm$ 1-1/2 in. In addition, no more than 3/4 in. deviation in radius will be allowed on a single plate section using a 15 ft template. The maximum inward deflection of the liner plate between the stiffeners (spaced at 15 in.) will be 1/8 in. relative to a 15 in. straight edge.

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5.4.4 SPECIFIC CONSTRUCTION TOPICS

5.4.4.1 Bonding of Concrete Between Lifts

Horizontal construction joints will be prepared for receiving the next lift by wet sandblasting, by cutting with an air-water jet, or by bush hammering. Surface set retardant compounds will not be used.

When wet sandblasting is to be employed, it will be continued until all laitance, coating, stains, debris, and other foreign materials are removed. The surface of the concrete will be washed thoroughly to remove all loose material.

When air-water cutting is to be used, it will be performed after initial set has taken place but before the concrete has taken its final set. The surface will be cut with a high pressure air-water jet to remove all laitance and to expose clean, sound aggregate, but not so as to undercut the edges of the larger particles of aggregate. After cutting, the surface will be washed and rinsed as long as there is any trace of cloudiness of the wash water. Where necessary to remove accumulated laitance, coatings, stains, debris, and other foreign material, wet sandblasting will be used before placing the next lift, to supplement air-water cutting.

Horizontal surfaces will be wetted and covered with one-quarter inch to one-half inch of mortar of the same cement-sand ratio as used in the concrete, immediately before the concrete is placed.

Vertical joints will also be sandblasted or bush hammered, cleaned, and wetted before placing concrete.

5.4.4.2 Prestressing Sequence

The detailed stressing sequence will be based on the following general requirements to minimize unbalanced loads and differential stresses in the structure.

a. Every second hoop tendon will be tensioned at the three buttresses within a strip extending from 20 feet above the construction opening to the bottom of the ring girder.

b. (Deleted) S1

c. The alternate hoop tendons will be tensioned within a strip extending from 30 feet above the construction opening to one 50 feet below the bottom of the ring girder.

d. The dome tendons will be fully tensioned using a balanced approach.

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3 e. The construction opening will be closed.

f. The remaining tendons will be installed including buttonheads.

S1 g. Every second hoop tendon in the lower region will be stressed following the pattern used on the upper hoop tendons.

3 h. The vertical tendons will be tensioned.

S1 i. All remaining tendons will be tensioned

The procedure for prestressing will be carefully worked out with the post-tensioning vendor.

All procedures will be subject to the approval of the Engineer.

The Engineer will provide to the vendor prestressing forces for each tendon, the anticipated concrete elastic, shrinkage, and creep prestress losses, and the maximum prestress forces for each stage of prestressing. The vendor will incorporate all this information along with any steel relaxation, friction, and anchorage losses to establish the initial jacking force for each sequential operation.

Force and stress measurements will be made by measuring the elongation of the prestressing steel and comparing it with the force indicated by the jack-dynamometer or pressure gage. The gage will indicate the pressure in the jack within plus or minus two percent. Force-jack pressure gage or dynamometer combinations will be calibrated against known precise standards just before application of prestressing forces begins and all calibrations will be so certified prior to use. Pressure gages and jacks so calibrated will always be used together.

During stressing, records will be kept of elongations as well as pressures obtained. Lift-off stress reading will be taken at the end of each stressing operation to check the actual stress in the tendon. Jack-dynamometer or gage combinations will be checked against elongation of the tendons and the cause of any discrepancy exceeding plus or minus five percent of that predicted by calculations (using average load elongation curves) will be corrected, and if caused by differences in load-elongation from averages will be so documented. Calibration of the jack-dynamometer or pressure gage combinations will be maintained accurately within above limits.

5.4.4.3 Liner Plate Sequence

The construction sequence of the containment liner plate is as follows:

1 a. Inserts for the erection of the floor and the first wall section will accompany the placing of the base slab. The installation and testing of the floor liner plate can be done at anytime prior to the placing of the liner plate protective slab.

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b. The first wall liner plate will be prefabricated, welded, tested, and then installed after erection of the buttresses in segments of approximately 120 degrees (center to center of buttresses) in lifts of 10 ft. This is followed by the welding and testing of the remaining horizontal and vertical seams. The installation of wall tendon tubes and reinforcing steel is followed by the pouring of the walls in alternating 120 degree segments. 1 S1 1 S1

c. The procedure is repeated for the remaining lifts of the wall in similar fashion. Liner plate erection is permitted to reach two courses (approximately 10 ft. each) above the concrete pours provided adequate bracing is included.

d. Upon completion of the wall the erection trusses for the dome are installed. The erection of the dome liner plate is completed including all welding and testing. This is followed by a 7 in. concrete pour upon the entire dome area to support the remaining dome concrete. 1

The above procedure is a general outline. The exact sequence of construction will be determined by the contractor and approved by the engineer to ensure the quality of work.

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5.5 CONTAINMENT SYSTEM INSPECTION, TESTING, AND SURVEILLANCE

5.5.1 TESTS TO ENSURE LINER INTEGRITY

a. Construction Tests: These take place during the erection of the containment building liner.

b. Pre-operational Tests: These are performed after the erection of the structure is complete but before reactor operation.

5.5.1.1 Tests on Liner During Construction

Inspection procedures to be employed during construction for the liner seam welds, liner fastening, and around penetrations will consist of visual inspection, vacuum box soap bubble testing, radiography, and dye penetrant testing.

a. Visual Inspection of Welds

All of the welding will be visually examined by a technician responsible for welding quality control. The criteria for workmanship and visual quality of welds will be as follows.

(1) Each weld will be uniform in width and size throughout its full length. Each layer of welding shall be smooth and free of slag, cracks, pinholes, and undercut, and shall be completely fused to the adjacent weld beads and base metal. In addition, the cover pass shall be free of coarse ripples, irregular surface, nonuniform bead pattern, high crown, and deep ridges or valleys between beads. Peening of welds will not be permitted.

(2) Butt welds shall be of multipass construction, slightly convex, of uniform height, and have full penetration.

(3) Fillet welds shall be of the specified size, with full throat and legs of uniform length.

b. Soap Bubble Tests

All of the welding which will be covered by concrete or otherwise inaccessible after construction will be vacuum box soap bubble tested. In this test a vacuum box containing a window is placed over the area to be tested and is evacuated to produce at least 5 psi pressure differential. Before the vacuum box is placed over the test area, a soap solution is applied to the weld and any leaks will be indicated by bubbles observed through the window in the box. The soap solution consists of equal parts of corn

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syrup, liquid detergent, and glycerin. The solution shall be prepared not more than 24 hours preceding the test and its bubble formation properties shall be checked with a sample leak every half hour during the test.

c. Radiography

Radiography will be used as an aid to quality control. The primary purpose of the liner plate and the welds therein is to provide leak tightness integrity to the post-tensioned concrete containment vessel. Structural integrity of the containment is provided by the post-tensioned concrete and not by the liner plate.

Radiography is not recognized as an effective method for examining welds to assure leak tightness. Therefore, the only benefit which can be expected from radiography in connection with obtaining leak tight welds is an aid to quality control. Random radiography of each welder's work will provide structural verification that the welding is or is not under control and being done in accordance with the previously established and qualified procedures. Additionally, employing random radiography to inspect each welder's work has been proved by past experience to have a positive psychological effect on improving overall welding workmanship.

The criterion for radiographic techniques shall be in accordance with Paragraph UW-51 of Section VIII of the ASME Code. At least one spot radiograph shall be taken in the first 10 feet of welding completed in the flat, vertical, horizontal, and overhead positions by each welder. Thereafter, approximately 2 percent of the welding will be spot examined on a random basis, in such manner that an approximately equal number of spot radiographs will be taken from the work of each welder.

d. Dye Penetrant

Dye penetrant inspection will also be used as an aid to quality control. The field welding inspectors will use dye penetrant inspection to closely examine welds judged to be of questionable quality on the basis of the initial visual inspection. Also, dye penetrant inspection will be used to confirm the complete removal of all defects from areas which have been prepared for repair welding. Dye penetrant inspection of liner plate welds will be in accordance with Appendix VIII, "Methods for Liquid Penetrant Examination," of Section VIII of the ASME Boiler and Pressure Vessel Code.

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e. Initial Leak Test for Base Slab Liner Plate Welds

The welds for each section of base slab liner plate will be vacuum box soap bubble tested immediately upon installation. After successfully passing this leakage test, they will be covered with test channels and the particular welds associated with that section of liner plates will be pressure tested using a Freon-Air mixture or soap bubble test at design pressure for a period of at least two hours with no drop in pressure. Any repairs will be carried out utilizing the same high standards and control exercised in the initial construction.

5.5.1.2 Pre-operational Integrated Leak Test

The design leak rate will not be more than 0.1 percent by volume of the contained atmosphere in 24 hours at 59 psig. It has been demonstrated that, with good quality control during erection, this is a reasonable requirement.

The basis of the leak rate test is the reference volume method. Every effort is made to demonstrate the leak tightness of the reference volume system. The entire reference volume system is pressurized to a minimum of 100 psi gage with air containing 20 weight percent Freon. All reference volume joints are bagged with plastic and the system held at this pressure for 48 hours. The reference volume system, especially the joints, is checked with a halogen leak detector to demonstrate integrity.

In addition to the usual calculation of leak rate as a function of pressure differential, air is returned to the reactor containment at the conclusion of the test through a precision gas meter until the differential pressure is returned to its original condition. This provides a check on the calculated leak rate. Reactor containment ambient temperature and humidity are also measured during the course of the test to provide further backup information.

The initial leak rate test consists of establishing a leak rate at design pressure. Because the containment is a thick-walled concrete structure, short-term temperature or meteorological variations should not have any appreciable effect on the containment ambient temperature and pressure. It should, therefore, be possible to establish meaningful leak rates in a shorter term test than might be required on a bare steel vessel. The containment will be held at each test pressure for a minimum of 24 hours.

5.5.2 STRENGTH TEST

A pressure test will be made on the completed building using air at 1.15 times the design pressure. This pressure will be maintained on the building for a period of one hour. During this test, measurements and observations will be made to verify the adequacy of the structure design.

5.5.3 IN-SERVICE TENDON SURVEILLANCE PROGRAM

The in-service surveillance program for the containment structure consists of evaluating the tendon system performance and the corrosion protection system performance. Further, the containment structure is a passive type system where mechanical operational failures are non-existent, thus only requiring that the system remain in status-quo and available to perform
3 its function in the unlikely event that it will be required. It is the intent of the surveillance programs to provide sufficient in-service historical evidence necessary to maintain confidence that the integrity of the containment structure is being preserved.

To accomplish the surveillance program, the following quantity of tendons will be made available for inspection and lift-off readings.

S1 Horizontal - Three 240° tendons comprising one complete hoop system.

Vertical - Three tendons spaced approximately 120° apart.

Dome - Three tendons spaced approximately 120° apart.

The surveillance program for structural integrity and corrosion protection will consist of the following operations being performed during each inspection.

a. Lift-off readings will be taken for all nine tendons.

b. One tendon of each directional group will be relaxed and three wires or one strand removed as samples for inspection.

c. After the inspection the tendons will be retensioned to the stress level measured at the lift-off reading, and then checked by a final lift-off reading.

3 d. Should the inspection reveal any significant corrosion (pitting, or loss of area), further inspection of the other two sets will be made to determine the extent of the corrosion and its significance to the load carrying capacity of the structure. Samples of corroded wire or strand will be tested to failure to evaluate the effects of any corrosion.

Inspection requirements for containment, as well as those for other systems, will be a part of the technical specifications for the plant and therefore, will be included in the operating license application. Changes in these inspection requirements or their elimination at any time during plant life will be subject to AEC regulations governing technical specifications and will require review and approval by the AEC after justification by the applicant.

Conservative testing requirements will be established at the operating license stage and nothing in the design of the containment will preclude the testing of tendons or pneumatic testing of the containment during its life.

It is expected that experience gained in surveillance testing of the containment structure as well as from other sources during the life of the plant will show that testing frequency can be relaxed subject to AEC review and approval. 3

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9.6 FUEL HANDLING SYSTEM

9.6.1 DESIGN BASES

9.6.1.1 General System Function

The fuel handling system (Figure 9.6-1) is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after postirradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release.

The reactor is refueled with equipment designed to handle the spent fuel assemblies under water from the time they leave the reactor vessel until they are placed in a cask for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, economic, transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Borated water insures subcritical conditions during refueling.

9.6.1.2 New Fuel Storage Area

The new fuel storage area is a separate and protected area for the dry storage of new fuel assemblies. The new fuel storage area is sized to accommodate the maximum number of new fuel assemblies required for refueling of the reactor as dictated by the fuel management program. The new fuel assemblies are stored in racks in parallel rows having a center-to-center distance of 21 in. in both directions. This spacing is sufficient to maintain a k_{eff} of less than 0.9 when wet.

9.6.1.3 Spent Fuel Storage Pool

The spent fuel storage pool is a reinforced concrete pool lined with stainless steel; it is located in the fuel storage building. The pool is sized to accommodate 255 spent fuel assemblies which allows for a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of spent fuel assemblies from the reactor as established by the fuel management program. The spent fuel assemblies are stored in racks in parallel rows having a center-to-center distance of 21 in. in both directions. Control rod assemblies requiring removal from the reactors are stored in the spent fuel assemblies. 2

9.6.1.4 Spent Fuel Transfer Tube

Two horizontal tubes are provided to convey fuel between the reactor building and the spent fuel storage pool. These tubes contain tracks for the fuel transfer carriages, gate valves on the spent fuel storage pool S1

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side, and a means for flanged closure on the reactor building side. The
S1 fuel transfer tubes penetrate into the fuel transfer canal at the lower depth, where space is provided for the rotation of the fuel transfer carriage baskets containing fuel assemblies.

9.6.1.5 Fuel Transfer Canal

The fuel transfer canal is a passageway in the reactor building extending from the reactor vessel to the reactor building wall. It is formed by an upward extension of the primary shield walls. The enclosure is a reinforced concrete structure lined with stainless steel; it forms a canal above the reactor vessel, which is filled with borated water for refueling.

Space is available in the fuel transfer canal for underwater storage of the reactor vessel internals upper plenum assembly.

The deeper fuel transfer station portion of the fuel transfer canal, can be used for storage of the reactor vessel internals core barrel and thermal
S1 shield assemblies by temporarily removing the new fuel handling racks.

9.6.1.6 Miscellaneous Fuel Handling Equipment

This equipment consists of fuel handling bridges, fuel handling tools, new fuel storage racks, spent fuel storage racks, new fuel handling racks, fuel transfer containers, control rod handling tools, viewing equipment, fuel transfer mechanisms, and shipping casks. In addition to the equipment directly associated with the handling of fuel, equipment is provided for handling the reactor closure head and the upper plenum assembly to expose the core for refueling.

9.6.2 SYSTEM DESCRIPTION AND EVALUATION

9.6.2.1 Receiving and Storing Fuel

New fuel assemblies are received in shipping containers and stored dry in racks having a center-to-center distance of at least 21 in. They are subsequently moved into the reactor building in one of the following ways.

S1 a. After reactor shutdown, new fuel assemblies can be transferred from new fuel storage area into the reactor building through the equipment hatch and stored directly in the new fuel handling racks in the transfer canal adjacent to the transfer tubes.

S1 b. After reactor shutdown, new fuel assemblies can be transferred from the new fuel storage area into the reactor building transfer canal by way of the spent fuel storage pool with the use of the fuel transfer carriage and the fuel transfer tubes.

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9.6.2.2 Loading and Removing Fuel

Following the reactor shutdown and reactor building entry, the refueling procedure is begun by removing the reactor closure head and control rod drives assembly. Head removal and replacement time is minimized by the use of two stud tensioners. The stud tensioner is a hydraulically operated device that permits preloading and unloading of the reactor closure studs at cold shutdown conditions. The studs are tensioned to their operational load in two steps in a predetermined sequence. Required stud elongation after tensioning is verified by micrometer measurements.

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts are supported in the closure head bolt holes with specially designed spacers. Removal of the studs with the reactor closure head minimizes handling time and reduces the chance of thread damage.

The reactor closure head assembly is handled by a lifting fixture supported from the reactor building crane. It is lifted out of the canal onto a head storage stand located on the operating floor. The stand is designed to protect the gasket surface of the closure head. The lift is guided by two closure head alignment pins installed in two of the stud holes. These pins also provide proper alignment of the reactor closure head with the reactor vessel and internals when the closure head is replaced after refueling. The studs and nuts can be removed from the reactor closure head at the storage location for inspection and cleaning using special stud and nut handling fixtures. A stud and alignment pin storage rack is provided.

The annular space between the reactor vessel flange and the bottom of the fuel transfer canal is sealed off, before the canal is filled, by a seal clamped to the canal shield plate flange and the reactor vessel flange. The fuel transfer canal is then filled with borated water.

The upper plenum assembly is removed from the reactor by the reactor building crane and stored under water on a stand on the fuel transfer canal floor using a lifting device with special adapters.

Refueling operations are carried out using the fuel handling bridge cranes which span the fuel transfer canal. One bridge is used to shuttle spent fuel assemblies from the core to the transfer station and new fuel assemblies from the new fuel handling racks to the core. During this operation the second bridge relocates partially spent fuel assemblies in the core as specified by the fuel management program. S1

Fuel assemblies are handled by a pneumatically operated fuel handling tool attached to a telescoping and rotating mast which moves laterally on the bridge. Control rod assemblies are handled by a control rod handling tool attached to a second mast located on the bridges in the reactor building.

The fuel handling bridge moves a spent fuel assembly from the core under water to the transfer station where the fuel assembly is lowered into the

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fuel transfer carriage fuel basket. The control rod handling tool attached to the second mast is used to transfer a control rod assembly to a new fuel assembly in the adjacent new fuel handling racks. This new fuel assembly with control rod assembly is carried to the reactor by the fuel handling tool and located in the core.

Spent fuel assemblies removed from the reactor are transported to the spent fuel storage pool from the reactor building via the fuel transfer tubes by means of motor driven fuel transfer carriages. The spent fuel assemblies are removed from the fuel transfer carriage basket using a pneumatically-operated fuel handling tool attached to a movable mast located on the fuel handling bridges. These motor-driven bridge span the spent fuel storage pool and permit the refueling crew to store or remove new fuel assemblies in any one of the many vertical storage rack positions.

S1

The fuel transfer mechanisms are underwater motor-driven carriages that run on tracks extending from the spent fuel storage pool through the transfer tube and into the reactor building. Rotating fuel baskets are mounted on one end of the fuel transfer carriages to receive fuel assemblies in a vertical position. The hydraulically operated fuel baskets on the end of the carriages being used for refueling are rotated to a horizontal position for passage through the transfer tubes, and then rotated back to a vertical position in the spent fuel storage pool for vertical removal of the fuel assemblies.

Once refueling is completed, the fuel transfer canal water is drained by suction through a pipe located in the deep transfer station area. The canal water is pumped to the borated water storage tank to be available for the next refueling or for emergency cooling following a loss-of-coolant accident.

During operation of the reactor, the carriage is stored in the spent fuel storage pool, thus permitting the gate valve on the spent fuel storage pool side of the transfer tube to be closed and the blind flanges to be installed on the reactor building side of the tube.

The spent fuel storage pool has space for a spent fuel shipping cask, as well as for required fuel storage. Following a sufficient decay period, the spent fuel assemblies are removed from storage and loaded into the spent fuel shipping cask under water for removal from the site. Casks up to 100 tons in weight can be handled.

A decontamination area is located in the building adjacent to the spent fuel storage pool; in this area the outside surfaces of the casks can be decontaminated before shipment by using steam, water, or detergent solutions, and manual scrubbing to the extent required.

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9.6.2.3 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations.

All fuel assembly storage facilities, new and spent, maintain an eversafe geometric spacing of 21 in. between assemblies. The new and spent fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby ensuring the necessary spacing between assemblies. Although new fuel assemblies are stored dry, the 21 x 21 in. spacing ensures an eversafe geometric array in unborated water. Under these conditions, a criticality accident during refueling or storage is not considered credible.

All fuel handling and transfer containers are also designed to maintain an eversafe geometric array. Mechanical damage to the fuel assemblies during transfer operations is possible, although remote. Since the fission product release would occur under water, the amount of activity reaching the environment will present no appreciable hazard. A fuel handling accident analysis is included in Section 14.

All spent fuel assembly transfer operations are conducted under water. The water level in the fuel transfer canal provides a minimum of 10 ft of water over the active fuel line of the spent fuel assemblies during movement from the core into storage; this limits radiation at the surface of the water to less than 10 mrem/hr. The spent fuel storage racks are located to provide a minimum of 13 ft of water shielding over stored assemblies to limit radiation at the surface of the water to no more than 2.5 mrem/hr during the storage period. The depth of the water over the fuel assemblies, as well as the thickness of the concrete walls of the transfer canal, is sufficient to limit the maximum continuous radiation levels in the working area to 2.5 mrem/hr.

Water in the reactor vessel is cooled during shutdown and refueling by the decay heat removal system described in 9.5. In case of a power failure, this system will be operated by the auxiliary power supply. The spent fuel storage pool water is cooled by the spent fuel cooling system as described in 9.4. A power failure during the refueling cycle will create no immediate hazardous condition owing to the large water volume in both the fuel transfer canal and spent fuel storage pool. With a normal quantity of spent fuel assemblies in the storage pool and no cooling available, the water temperature in the spent fuel storage pool would increase as discussed in 9.4.2.3.

During the refueling period the water level in both the fuel transfer canal and the spent fuel storage pool is the same, and the fuel transfer tube valves are continuously open. This eliminates the necessity for interlocks between the fuel transfer carriage and fuel transfer tube valve operations. S1

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S1 The simplified movement of a transfer carriage through the horizontal fuel transfer tubes minimizes the danger of jamming or derailing. To cope with such an eventuality, the open tube design provides access to the entire length of the fuel transfer carriage travel from the fuel transfer canal. All operating mechanisms of the system are located in the fuel storage building for ease of maintenance and accessibility for inspection before the start of refueling operations.

S1 During reactor operation bolted and gasketed closure plate located on the reactor building flange of the fuel transfer tubes, prevent leakage of water from the spent fuel storage pool into the transfer canal in the event of a leak through the fuel transfer tube valve. Both the spent fuel storage pool and the fuel transfer canal are completely lined with stainless steel for leak-tightness and ease of decontamination. The fuel transfer tube will be appropriately attached to the liner to maintain leak integrity. The spent fuel storage pool cannot be accidentally drained since water must be pumped out through a suction pipe. The fuel transfer mechanisms are designed to permit initiation of the carriage travel and the carriage fuel basket rotation from the building in which the carriage fuel basket is being loaded or unloaded.

All electrical gear is located above water for greater integrity and ease of maintenance. The hydraulic systems that actuate the rotating fuel baskets use storage pool water for operation to eliminate contamination.

The fuel transfer canal and storage pool water will have a boron concentration of 2,270 ppm. Although this concentration is sufficient to maintain core shutdown if all of the control rod assemblies were removed from the core, only a few control rods will be removed at any one time during the fuel shuffling and replacement. Although not required for safe storage of spent fuel assemblies, the spent fuel storage pool water will also be borated so that the transfer canal water will not be diluted during fuel transfer operations.

The fuel handling bridge mast travel is designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

Relief valves are provided on each stud tensioner to prevent overtensioning of the studs due to excessive pressure.

Gross failures of fuel are prevented by safety margins in the design and control of the core. The fuel assembly utilizes a free-standing Zircaloy fuel rod of sufficient length to accommodate the expected fission gas release from the fuel.

Any leaking fuel assemblies will be removed from the core for verification of leakage and placed in a failed fuel container. This operation is done in the fuel transfer canal and completely seals off the leaking fuel assembly before the fuel transfer mechanism transfers it out of the fuel transfer canal into the spent fuel storage pool. The design of the failed fuel containers will comply with 10 CFR 71 so that a defective fuel assembly
S1 can be safely stored and shipped while sealed in the failed fuel container.

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9.6.2.4 Operational Limits

Certain manipulations of the fuel assemblies and reactor internals during refueling may result in short-term exposures with radiation levels greater than 2.5 mrem/hr. The exposure time will be limited so that the integrated doses to operating personnel do not exceed the limits of 10 CFR 20.

The fuel handling bridge is limited to the handling of fuel and control rod assemblies and reactor closure head studs only. All lifts for handling the reactor closure head and reactor internals will use the reactor building crane.

Travel speeds for the fuel handling bridge, masts, and fuel transfer carriage will be controlled to ensure safe handling conditions.

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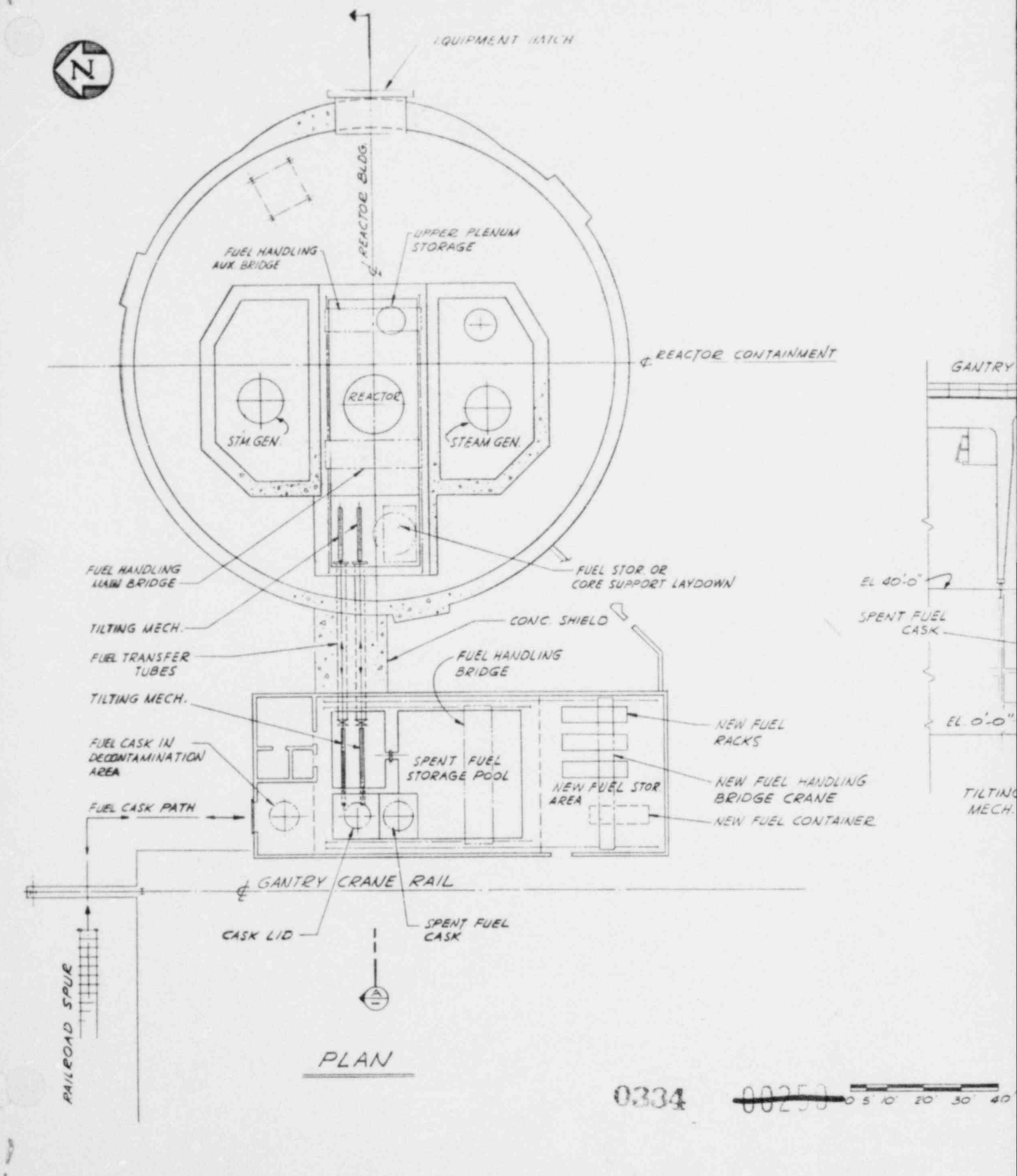
N
EQUIPMENT HATCH
REACTOR BLDG.
UPPER PLENUM STORAGE
FUEL HANDLING AUX. BRIDGE
REACTOR CONTAINMENT
REACTOR
STM. GEN.
STEAM GEN.
FUEL HANDLING MAIN BRIDGE
FUEL STOR. OR CORE SUPPORT LAYDOWN
TILTING MECH.
CONC. SHIELD
FUEL TRANSFER TUBES
FUEL HANDLING BRIDGE
TILTING MECH.
NEW FUEL RACKS
FUEL CASK IN DECONTAMINATION AREA
SPENT FUEL STORAGE POOL
NEW FUEL STOR. AREA
NEW FUEL HANDLING BRIDGE CRANE
FUEL CASK PATH
NEW FUEL CONTAINER
GANTRY CRANE RAIL
CASK LID
SPENT FUEL CASK
RAILROAD SPUR
GANTRY
EL 40'-0"
SPENT FUEL CASK
EL. 0'-0"
TILTING MECH.
0 5' 10' 20' 30' 40'

PLAN

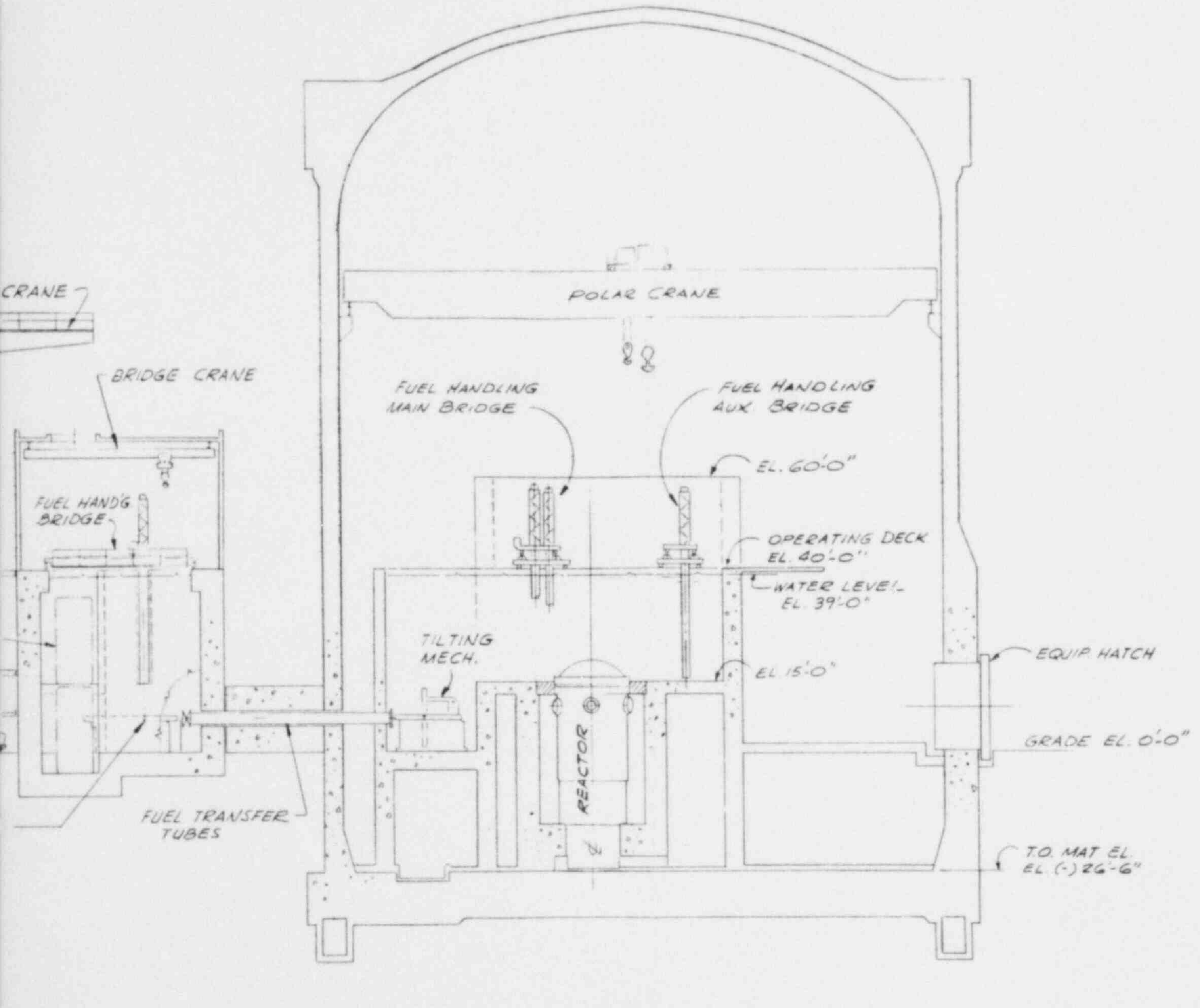

FIGURE 9.6-1
FUEL HANDLING SYSTEM

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APPENDIX 5A

DESIGN BASES FOR STRUCTURES, SYSTEMS AND EQUIPMENT

1.0 GENERAL

The design bases for structures for normal operating conditions are governed by the applicable building design codes. The design bases for specific systems and equipment are stated in the appropriate PSAR Section. The basic design criterion for the maximum hypothetical accident and seismic conditions is that there be no loss of function if that function is related to public safety.

2.0 CLASSES OF STRUCTURES, SYSTEMS AND EQUIPMENT

2.1 CLASS 1

Class 1 structures, systems and equipment are those whose failure could cause uncontrolled release of radioactivity or those essential for immediate and long-term operation following a loss-of-coolant accident. They are designed to withstand the appropriate seismic loads and other applicable loads without loss of function. When a system as a whole is referred to as Class 1, portions of the system not associated with the loss of function criteria, may be designated as Class 2. Class 1 structures will be sufficiently isolated or protected from Class 2 structures to ensure that their integrity is maintained at all times. 2

The following are Class 1 structures. 2

- Containment structure shell
- Nuclear service spray ponds and pipe lines 2
- The auxiliary building that houses engineered safeguards systems, control room, and radioactive materials
- Fuel storage facilities 2
- Supports for Class 1 system components
- Storage Reservoir 3

Class 1 equipment and systems are as follows 2

- Reactor vessel and internals including control rods and control rod drives
- Primary coolant system components (steam generators, pressurizer, pumps, etc.) and piping, including vent and drain piping inside the containment

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- Containment penetrations up to and including the first isolation valve outside containment

3 - Main steam and main feedwater piping up to the code limit of ASME Boiler and Pressure Vessel Code Section I. (Stop Valves)

- Atmospheric dump and main steam safety valves and associated piping from main steam headers

- New and spent fuel storage racks and fuel handling equipment

- Motor-driven and steam-driven auxiliary feedwater pump, condensate storage tank and associated piping

2 - Emergency diesel generators including fuel supply

- Containment building crane (unloaded condition)

2 - Control boards, switchgear, load centers, batteries, transformers and cable runs serving Class 1 equipment

3 - Nuclear service raw water system

- Nuclear service cooling water system

- Containment spray system

- Containment air recirculation

2 - Low pressure injection and decay heat removal

- Core flooding injection tanks and piping

2 - High pressure injection, make-up and purification

3 - Borated water storage tank

2 - Radioactive waste treatment

- Spent fuel cooling and clean-up

1 - Plant Vent

2.2 CLASS 2

Class 2 structures, systems, and equipment are those whose failure would not result in the release of radioactivity and would not prevent reactor shutdown.

3.0 DESIGN BASES

3.1 CLASS 1 STRUCTURES DESIGN

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3.1.1 Containment Structure

The design of the containment structure for all credible conditions of loading including normal loads, accident loads, thermal loads and environmental loads (including seismic) is found in Section 5, Containment System.

3.1.2 Storage Reservoir

The design of the storage reservoir is found in Appendix 2G, Storage Reservoir Criteria. 1

3.1.3 Structures Other Than Containment and Reservoir

a. Normal Operations

Loads encountered during normal plant operation including design earthquake loads are resisted by the structure using design methods of appropriate standards and codes insofar as they are applicable. Normal allowable stresses are used for this load condition.

b. Accident, Wind and/or Seismic Conditions

Class 1 structures are, in general, proportioned to maintain elastic behavior when subjected to various combinations of dead loads, accident loads, thermal loads, wind loads and/or earthquake loads. Concrete structures are designed for ductile behavior wherever possible; that is, with steel stress controlling the design.

The final design of Class 1 structures other than the containment structure satisfies the following load combinations and factors.

$$A = 1.0D + 1.0L + 1.0H$$
$$A = 1.0D + 1.0H + 1.0E/W$$
$$Y = 1/\emptyset\ (1.0D + 1.0H + 1.0E')$$
$$Y = 1/\emptyset\ (1.0D + 1.0R)$$
$$Y = 1.0D + 1.0R + 1.0E'$$ 3

In areas where dead load subtracts from critical stress 0.90 D will be used.

A = capacity of the structure based upon allowable code stresses with no stress increase.
Y = required yield capacity of the structures.
D = dead load of structure and equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads. In addition, a portion of "live load" is added when such load is expected to be present when the plant is operating. An allowance is also made for future permanent loads.
L = appropriate live load
R = load on structure from reaction or pressure due to rupture of any one pipe. 2
H = load on structure due to thermal expansion of pipes under operating conditions.
E = "design earthquake" load resulting from (Figure 5A-1) ground surface acceleration of 0.13g. 3
E' = "maximum hypothetical earthquake" load resulting from (Figure 5A-2) seismic ground surface acceleration of 0.25g. 3
W = wind load

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Ø = 0.90 for reinforced concrete in flexure.
Ø = 0.85 for tension, shear, bond, and anchorage in reinforced concrete.
Ø = 0.75 for spirally-reinforced concrete compression members.
Ø = 0.70 for tied compression members.
Ø = 0.90 for fabricated structural steel.
Ø = 0.90 for mild reinforcing steel (not prestressed) in direct tension.
Ø = 0.95 for prestressed tendons in direct tension.

The containment structure and engineered safeguards systems components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding
1 or erosion of barriers is permissible due to jet or missile impact, provided there is no general failure. Missile barriers are designed on the basis of absorbing energy by plastic yielding.

The final design of the missile barrier and equipment support structures inside the containment will be reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe reactions and earthquake loads without loss of function. The deflections or deformations of structures and supports will be checked to assure that the functions of the containment and engineered safeguards equipment are not impaired.

2 The maximum displacement of structures, systems and equipment will be considered so the design and major components will have structural separation (with the exception of the fuel transfer
1 tube) which will allow for differential movement. The foundation material is uniform and of such a nature that no permanent settlement or tilting will result from seismic loads.

3.2 CLASS 1 SYSTEMS AND EQUIPMENT DESIGN

Components and systems classified as Class 1 will be designed in accordance with the following criteria

a. For Case I stress intensities shall be maintained within the allowable working stress limits accepted as good practice, and where applicable, as set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code and USASI B31.1 Code for Pressure Piping.

3 b. For Cases II, III, and IV stress intensities and the corresponding strains shall be limited so that the function of the component or system shall not be impaired as to prevent a safe and orderly shutdown of the plant.

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Case	Loading Combination	Stress Limits
I	Design loads + design earthquake loads	$P_m \leq 1.0\ S_m$
		$P_\ell + P_b \leq 1.5\ S_m$
II	Design loads + maximum hypothetical earthquake loads	$P_m \leq 1.2\ S_m$
		$P_\ell + P_b \leq 1.2\ (1.5\ S_m)$
III	Design loads + pipe rupture loads	$P_m \leq 1.2\ S_m$
		$P_\ell + P_b \leq 1.2\ (1.5\ S_m)$
IV	Design loads + maximum hypothetical earthquake loads + pipe rupture loads	$P_m \leq 2/3\ S_u$
		$P_\ell + P_b \leq 2/3\ S_u$

3

* where P_m = Primary general membrane stress intensity

S_m = Allowable membrane stress intensity

P_ℓ = Primary local membrane stress intensity

P_b = Primary bending stress intensity

S_u = Ultimate stress for unirradiated material at operating temperature

* (1) All symbols have the same definition or connotation as those in ASME B&PV Code Section III, Nuclear Vessels.

* (2) Portions of systems or components not covered by ASME B&PV Code Section III will be designed in accordance with applicable codes using the stress limits stated above. 2

* (3) The limits given above are for primary stresses; general membrane, local membrane, and bending. Based on available stress-strain curves, the corresponding strains will be about 20% of the uniform strain or less. This applies to all pertinent material, in the unirradiated condition.

* (4) Maximum local strains, due to $P_\ell + P_b$ will be in the order of 3% for carbon steel and 10% for type 18-8 stainless steel for cases I and II in unirradiated material. For Cases III and IV the piping will be restrained to prevent secondary damage. 4

All components will be designed to insure against structural instabilities regardless of stress levels. 3

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2 Adequate flexibility will be provided in piping and other interconnecting elements to allow for differential movement during an earthquake.

3.3 CLASS 2 STRUCTURES, SYSTEMS AND EQUIPMENT DESIGN

All Class 2 structures, systems and equipment are designed in accordance with the applicant's standard practice. However, in no case will the design criteria in this classification be less restrictive than that required by standard applicable codes and standards of the requirements of the Uniform Building Code.

4.0 EARTHQUAKE LOADS FOR CLASS 1 STRUCTURES, SYSTEMS AND EQUIPMENT

Seismic Forces (E and E') -- AEC publication TID 7024, "Nuclear Reactors and Earthquakes," is used as the basic design guide for seismic analysis.

The "design earthquake," to be used for this plant, has a maximum ground acceleration of 0.13g horizontally. The "maximum hypothetical earthquake" has a maximum ground acceleration of 0.25g horizontally.

Seismic loads on structures, systems and equipment are determined by
realistic evaluation of dynamic properties and the acceleration from
S1 response spectrum. The response spectra for the horizontal component are
given in Figures 5A-1(a) and 5A-2(a), and the response spectra for the
vertical component are given in Figures 5A-1(b) and 5A-2(b).

The structures, systems and equipment will be designed for the horizontal component, the vertical component or a combination of the horizontal and vertical components acting simultaneously.

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5.0 DAMPING * 3

Stress Level	% Critical Damping	
	Type and Condition of Structure	Percentage of Critical Damping
1. Low, well below proportional limit, stresses below 1/4 yield point	a. Vital piping	0.5
	b. Steel, reinforced or prestressed concrete, no cracking, no joint slip	0.5 to 1.0
2. Working stress, no more than about 1/2 yield point	a. Vital piping	0.5 to 1.0
	b. Welded steel, prestressed concrete, well reinforced concrete (only slight cracking)	2
	c. Reinforced concrete with considerable cracking	3 to 5
	d. Bolted and/or riveted steel	5 to 7
3. At or just below yield point	a. Vital piping	2
	b. Welded steel, prestressed concrete (without complete loss in prestress)	5
	c. Prestressed concrete with no prestress left	7
	d. Reinforced concrete	7 to 10
	e. Bolted and/or riveted steel	10 to 15
4. All ranges	Rocking of Entire Structure	5 to 9 **
	Translation of Entire Structure	30% **

3

* Reference 8

** Reference 5, 6, 7

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6.0 LOADINGS COMMON TO ALL STRUCTURES

Ice or Snow Loading - A uniform distributed live load of 20 pounds per square foot on all roofs provides for any anticipated snow and/or ice loading.

7.0 REFERENCES

1. "Nuclear Reactors and Earthquakes" AEC Publication TID-7024.

2. Housner, G.W., Design of Nuclear Power Reactors Against Earthquakes, Proceedings of the Second World Conference on Earthquake Engineering, Page 133, Japan (1960).

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4. Task Committee on Wind Forces, "Wind Forces on Structures," ASCE Paper No. 3269.

5. Arnold, R. N., Bycroft, G. N., Warburton, G. B., Forced Vibrations of a Body on an Infinite Elastic Solid, Journal of Applied Mechanics, American Society of Mechanical Engineers, Vol. 22, No. 3 (1955).

6. Whitman, R. V., Richart, F. E., Design Procedures for Dynamically Loaded Foundations, Journal of the Soil Mechanics and Foundations Division, American Society of Civil Engineers, SM6 (Nov. 1967).

3 7. Parmelee, R. A., Building-Foundation Interaction Effects, Journal of the Engineering Mechanics Division, American Society of Civil Engineers, EM2 (April 1967).

8. Newmark, N. M., Design Criteria for Nuclear Reactors Subjected to Earthquake Motions, Presented at the International Atomic Energy Agency Meeting in Tokyo, (June 1967).

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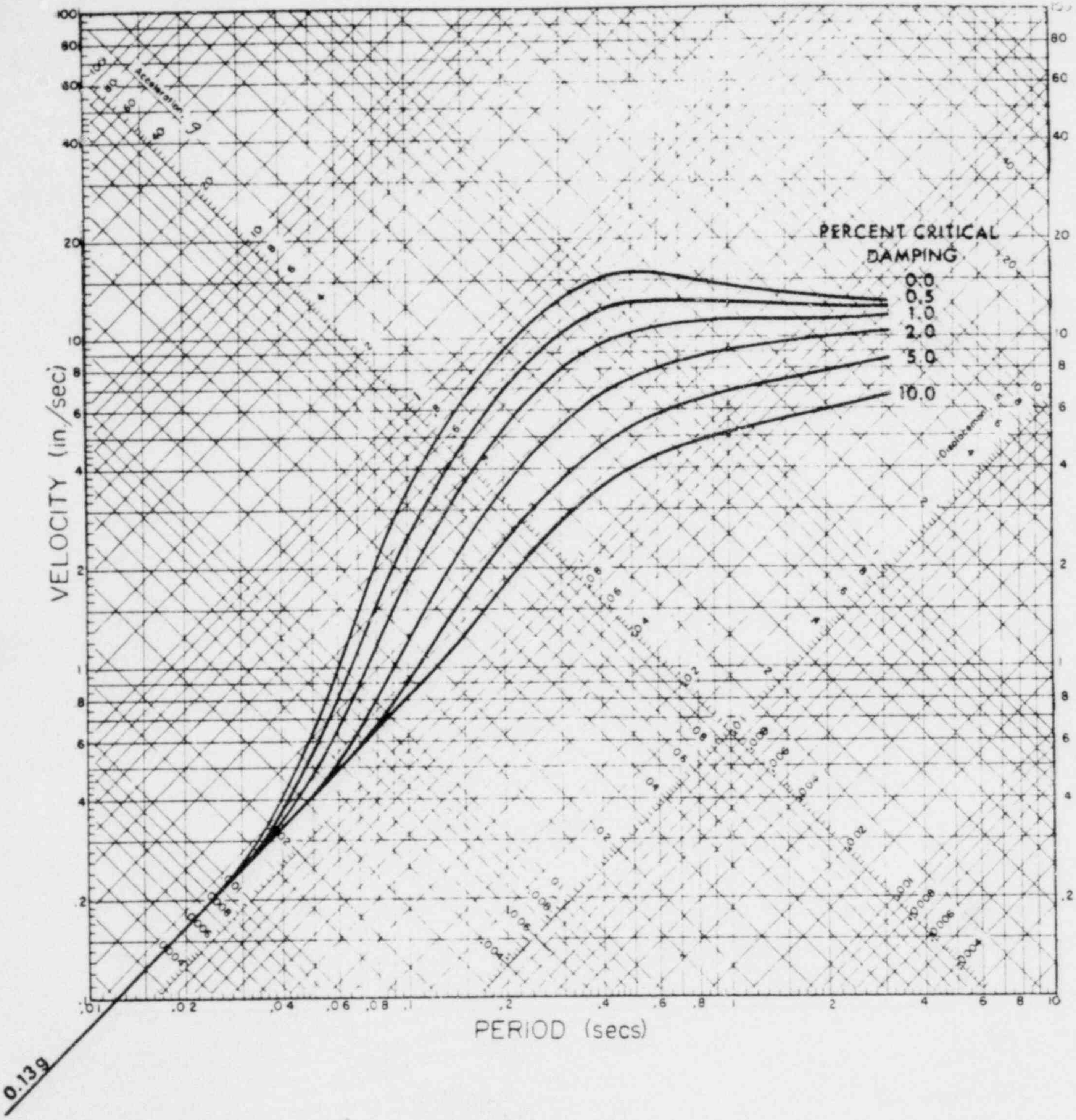

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FIGURE 5A-1 (a)
DESIGN EARTHQUAKE
HORIZONTAL RESPONSE SPECTRUM

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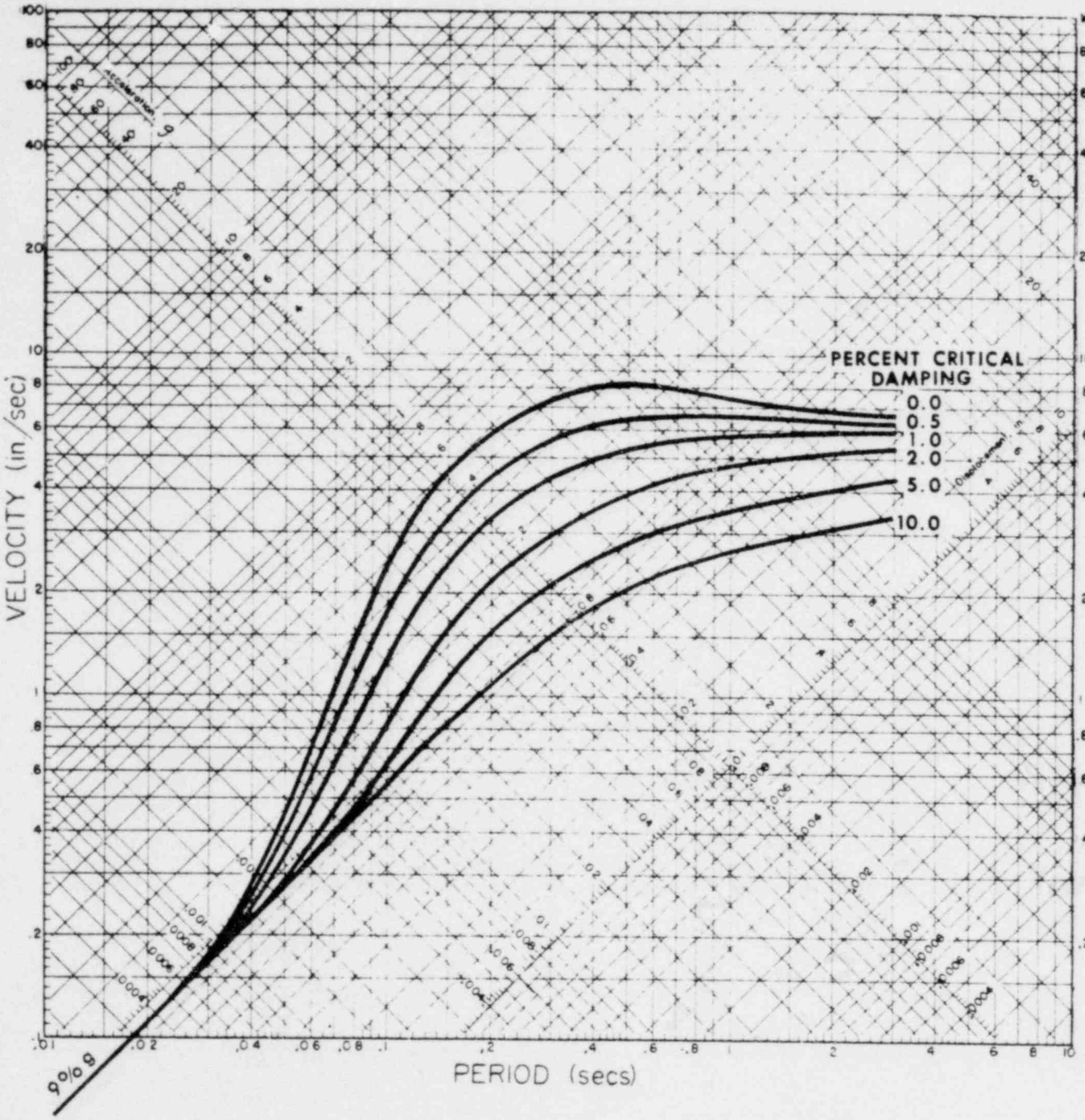

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FIGURE 5A-1 (b)
DESIGN EARTHQUAKE
VERTICAL RESPONSE SPECTRUM

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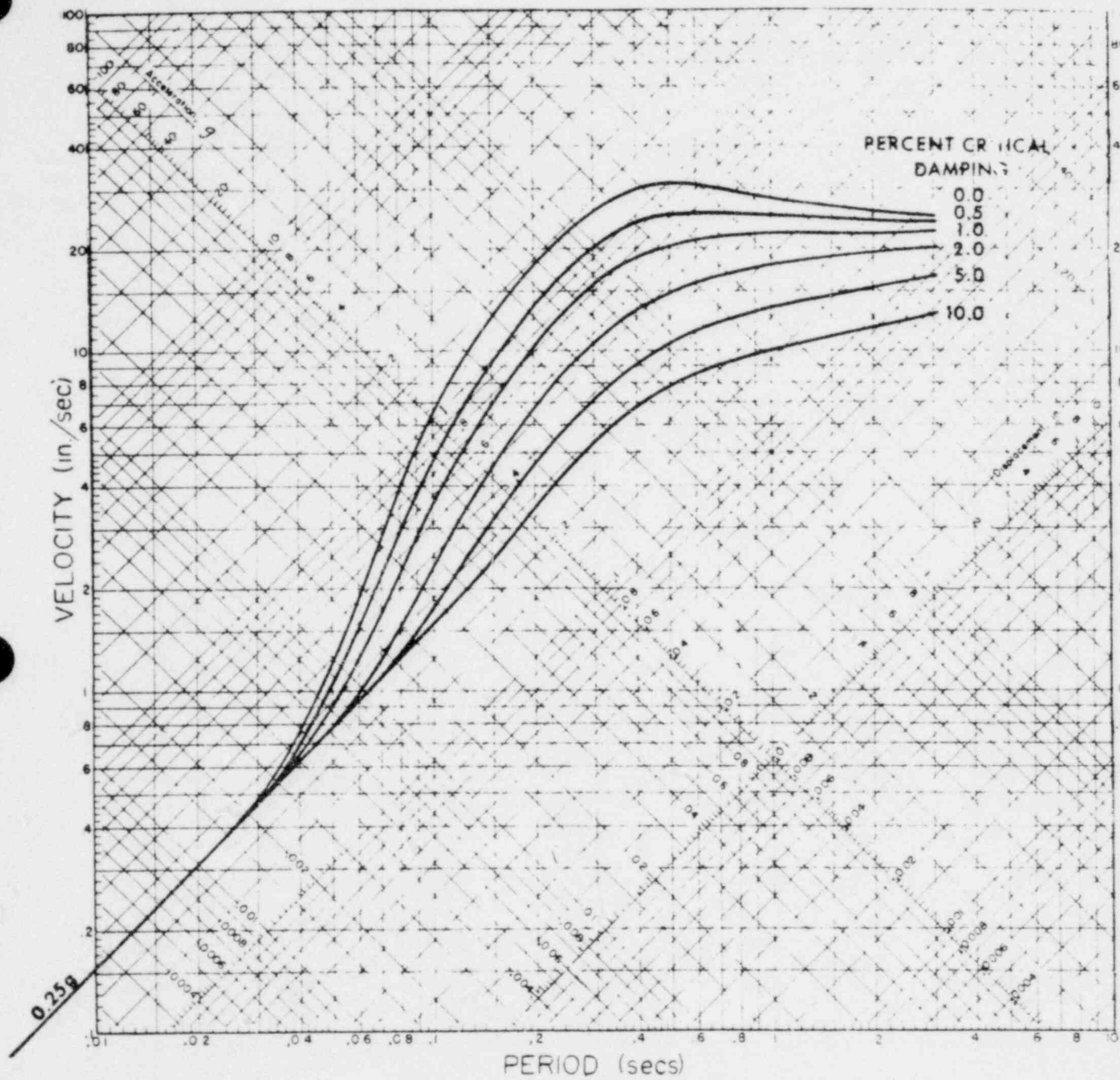

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FIGURE 5A-2 (a)
MAXIMUM HYPOTHETICAL EARTHQUAKE
HORIZONTAL RESPONSE SPECTRUM

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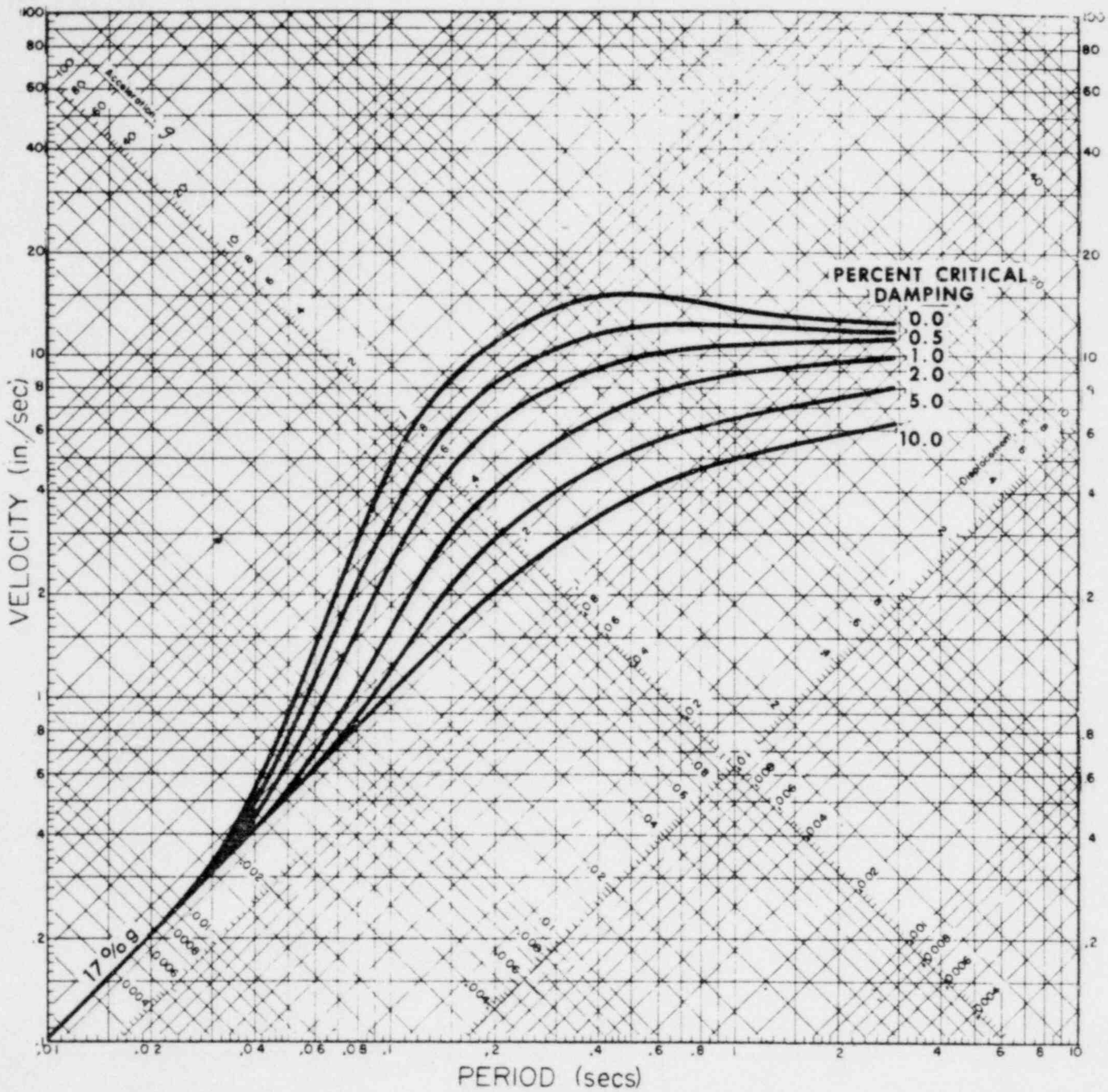

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FIGURE 5A-2 (b)
MAXIMUM HYPOTHETICAL EARTHQUAKE
VERTICAL RESPONSE SPECTRUM

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QUESTION 5J.7 (DRL 5.2)

CONTAINMENT STRUCTURAL DESIGN

5J.7.1 (DRL 5.2.1)

In certain circumstances the containment structure base may be located below water level. It appears that no layer of porous concrete and no membrane water-proofing exists between the soil and the containment. Consider the possibility of cracking of the concrete in the base mat, in the cylindrical wall and in the prestressing gallery. Ground water may reach the liner and the prestressing tendon anchors. The effect on the stability of the liner and possible corrosion of liner and tendons should be investigated. Explain drainage provisions at the containment lower section.

ANSWER

The normal groundwater is approximately 95 feet below the bottom of the containment and under no circumstances is it expected to rise to the elevation of the base slab. Figure 2.4-2 shows the relationship of the site to the surrounding area. The containment will be completely surrounded by paved areas or other buildings. The paving slopes away from the containment and will be detailed to prevent water seepage down the exterior walls.

5J.7.2 (DRL 5.2.2)

Provide the following:

(a) A preliminary design drawing of the containment, presenting details of the base slab, dome ring beam, cylinder-slab juncture, vertical buttresses and inspection gallery; showing reinforcing, prestressing, and liner features, including liner anchors;

(b) Scaled load plots for moment, shear, deflection, longitudinal force, and hoop tension, in order that an appraisal can be made of the significance of the various loadings which influence the containment design; Provide these plots as a function of containment height for prestress, dead, pressure, design earthquake wind, liner thermal (normal and accident) and concrete thermal (normal and accident) loading; 4

(c) The normal operating transient and steady state thermal gradients to be used in the design of the containment for typical winter and typical summer day;

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(d) The transient and steady state thermal gradients through the containment envelope during the design basis accident for typical winter and typical summer day.

ANSWER

(a) The preliminary design details of the containment structure are shown in Figure 5.1-1, 5.1-2, and 5.1-3 of Section 5 in the PSAR. These figures have been modified to include the requested preliminary information which is available.

(b) Preliminary scaled load plots for moment, shears, forces, and deflections are included in Figures 5J.7.2-1 through 5J.7.2-6 for all loading conditions. A plot of the thermal loading of the liner plate has been excluded due to the nature of its membrane action under prestress and thermal conditions. However, a description of this action is discussed in Questions 5J.7.4, 5J.7.18, 5J.7.19 and 5J.7.20 as well as Sections 5.1.3.4, 5.1.4.6 and 5.1.4.9 of the PSAR.

(c) (d) The steady state and transient thermal gradients through the containment walls are shown in Figure 5.1-4 as submitted in Amendment 1. To provide futher clarification this figure has been modified to demonstrate the effects of the maximum range of temperature fluction which are possible at the site. This modification is shown in the Figures 5J.7.2-7 and 5J.7.2-8. In addition to the gradients shown thermal stress differences due to varying concrete thicknesses (i.e., through the ring girder) will be considered for all conditions including startup.

Two items require consideration:

(1) The temperature range indicated in these figures represents extreme conditions. Normal fluctuations are anticipated to be substantially less severe. Refer to Section 2.3.3.6 and Appendix 2A.

(2) It should be noted that these fluctuations even under extreme conditions have little influence on the thermal gradients generated by postulated accident conditions.

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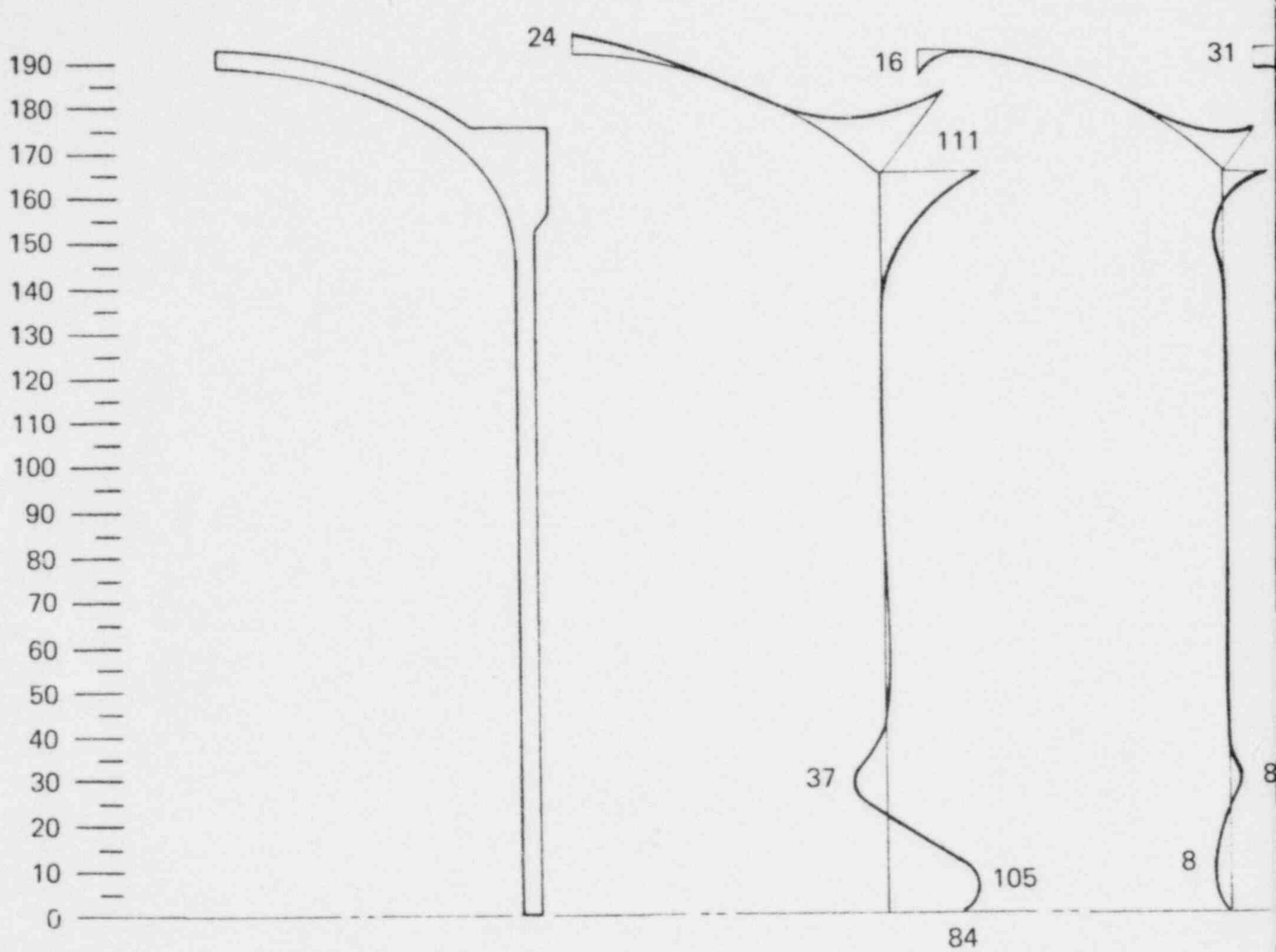

MERIDIONAL

MOMENT $\left(\frac{Ft\ K}{Ft}\right)$

HOOP

MOMENT $\left(\frac{Ft\ K}{Ft}\right)$

NOTE:
MOMENTS SHOWN ON TENSION SIDE.

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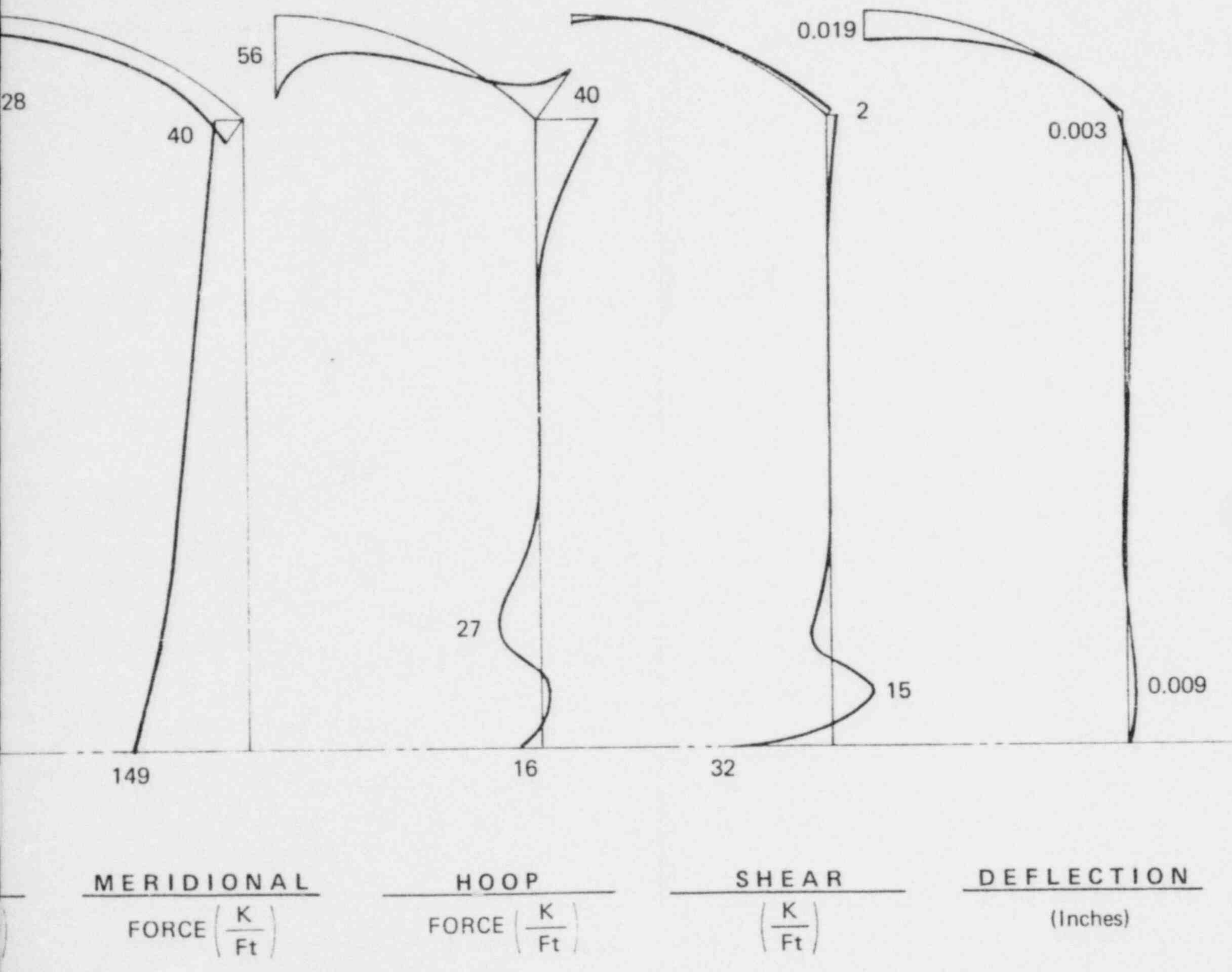

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FIGURE 5J.7.2-1
CONTAINMENT STRUCTURE STRESS ANALYSIS
PRELIMINARY MANUAL CALCULATIONS
DEAD LOAD (D)

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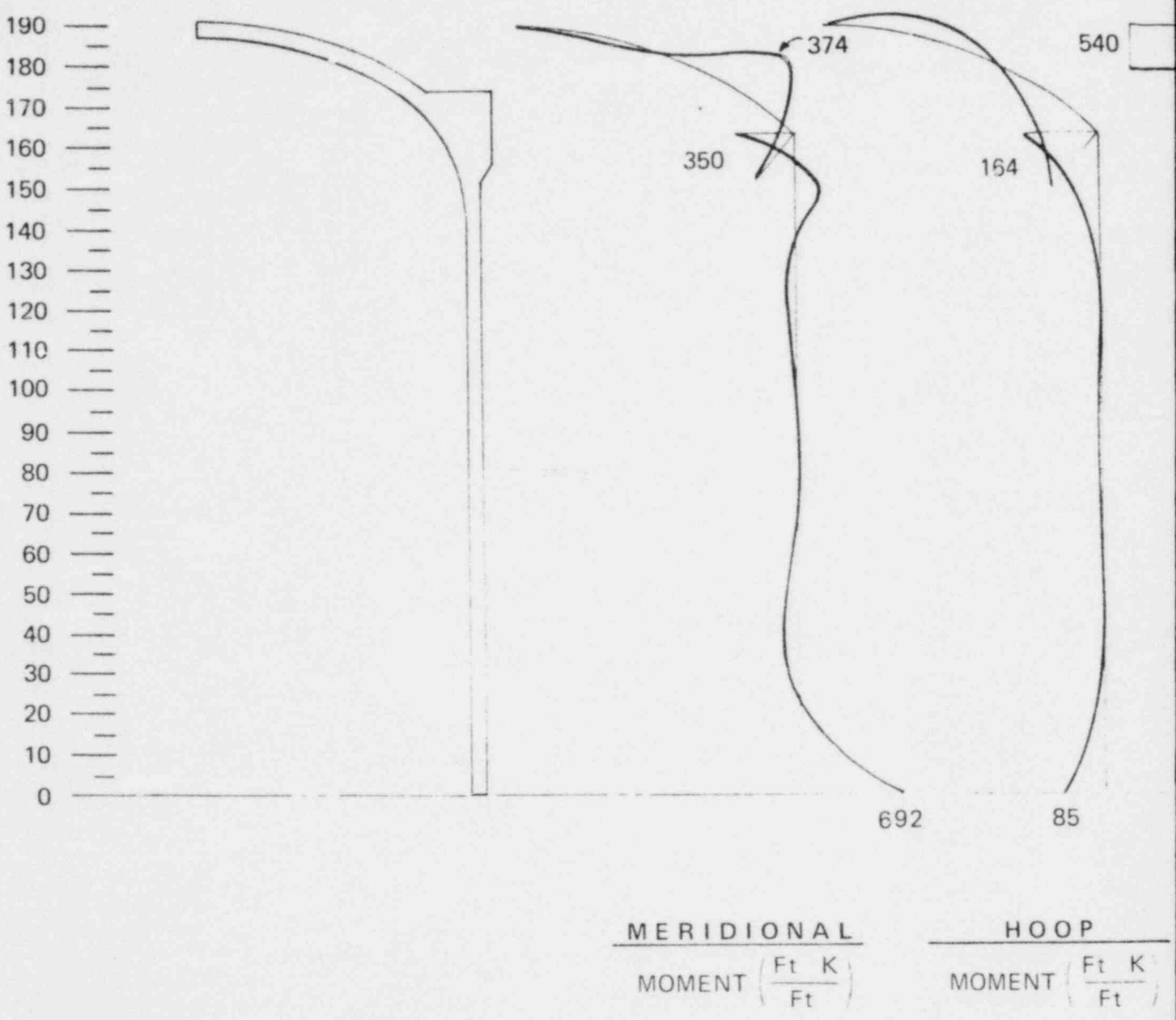

NOTE:
MOMENTS SHOWN ON TENSION SIDE.

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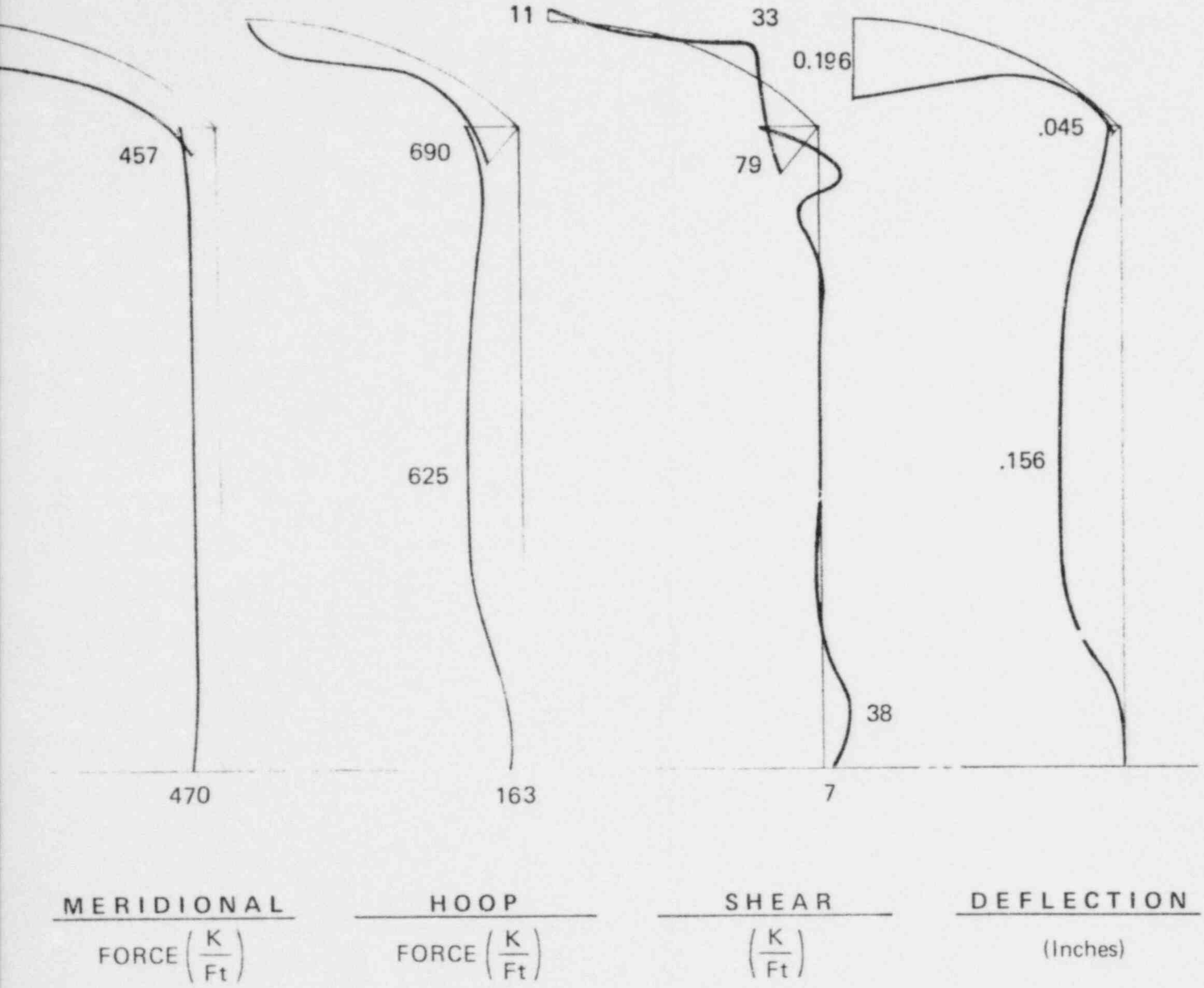

FIGURE 5J.7.2-2
CONTAINMENT STRUCTURE STRESS ANALYSIS
PRELIMINARY MANUAL CALCULATIONS
PRESTRESS LOAD (F)

0353

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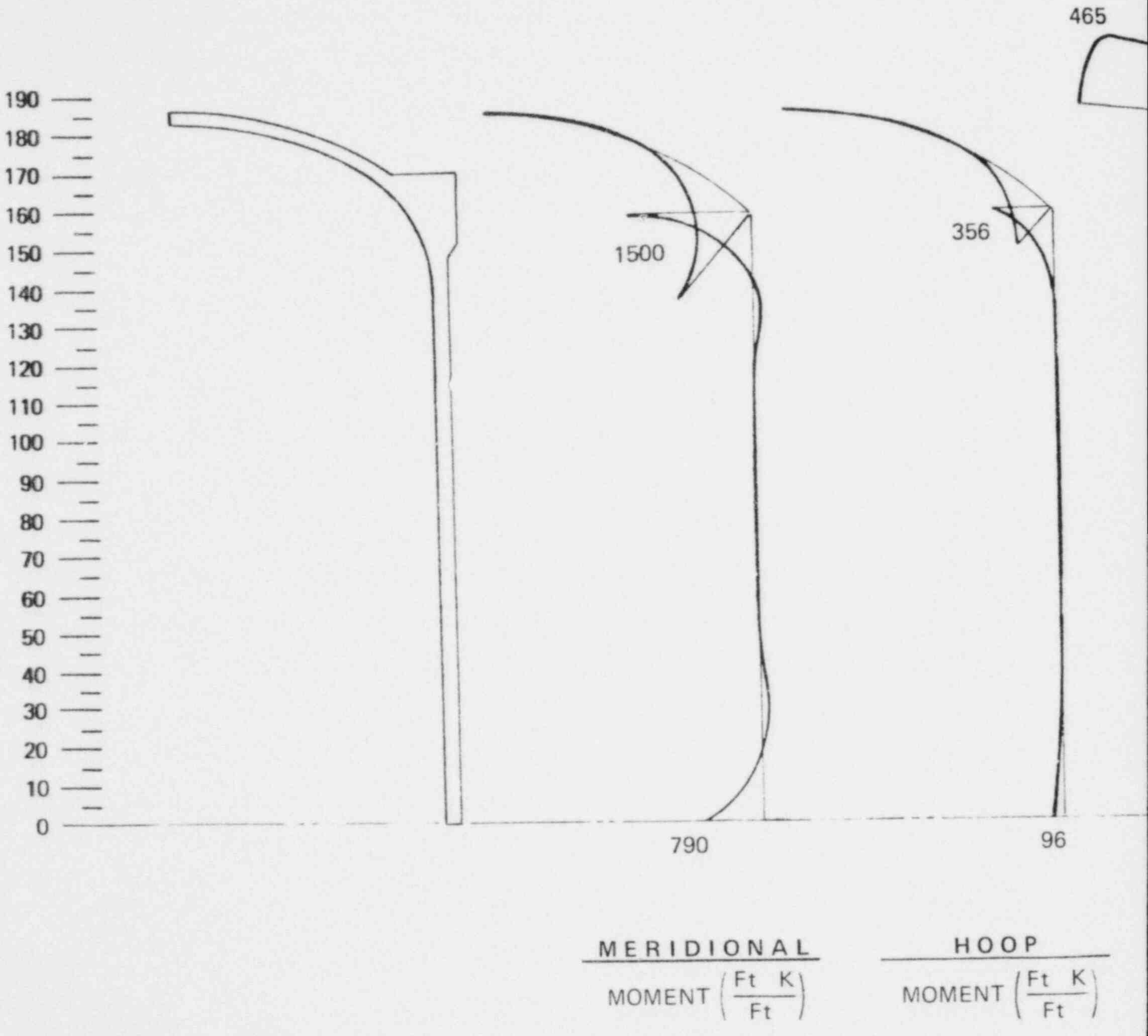
Supplement 1



NOTE:
MOMENTS SHOWN ON TENSION SIDE.

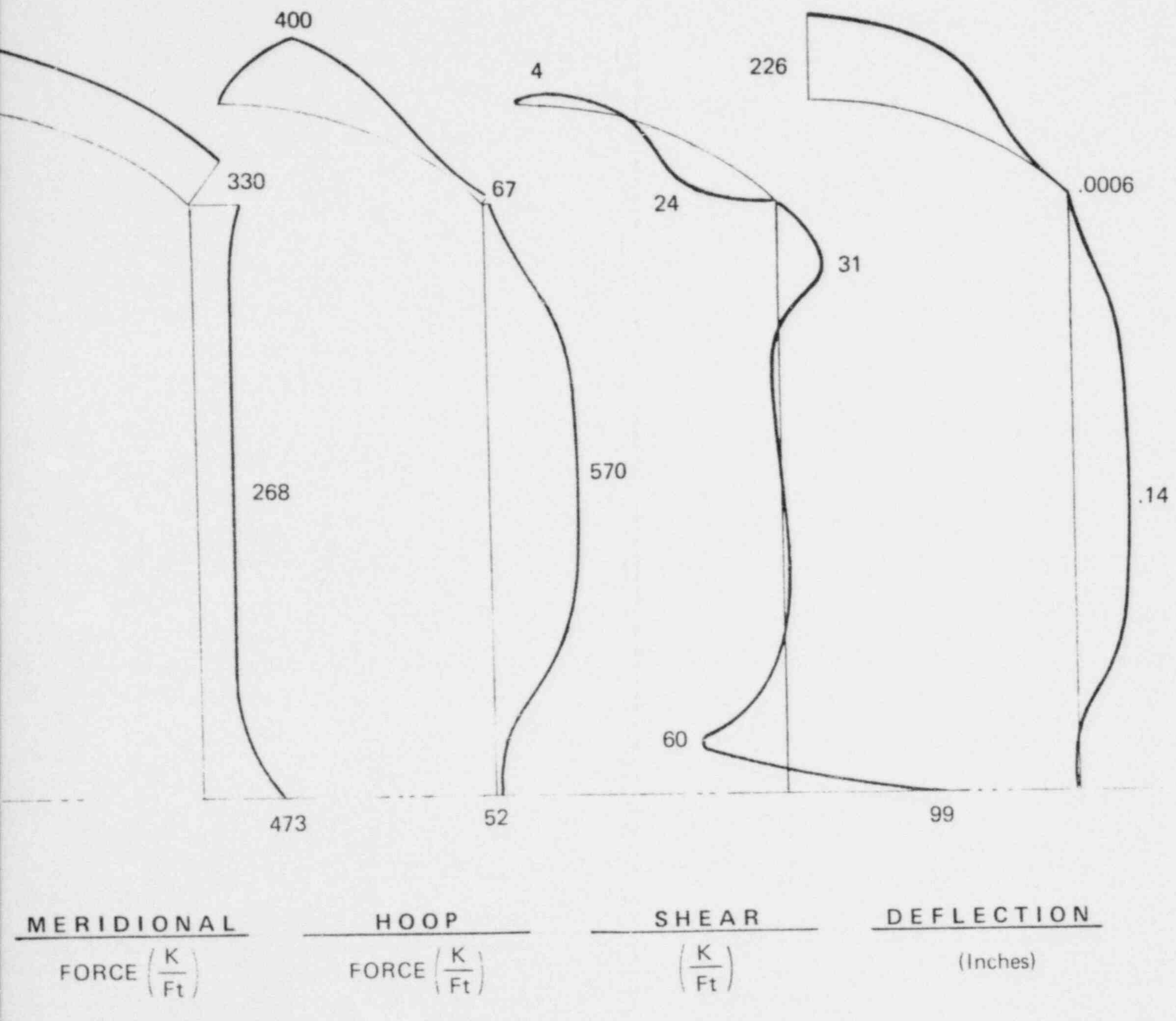
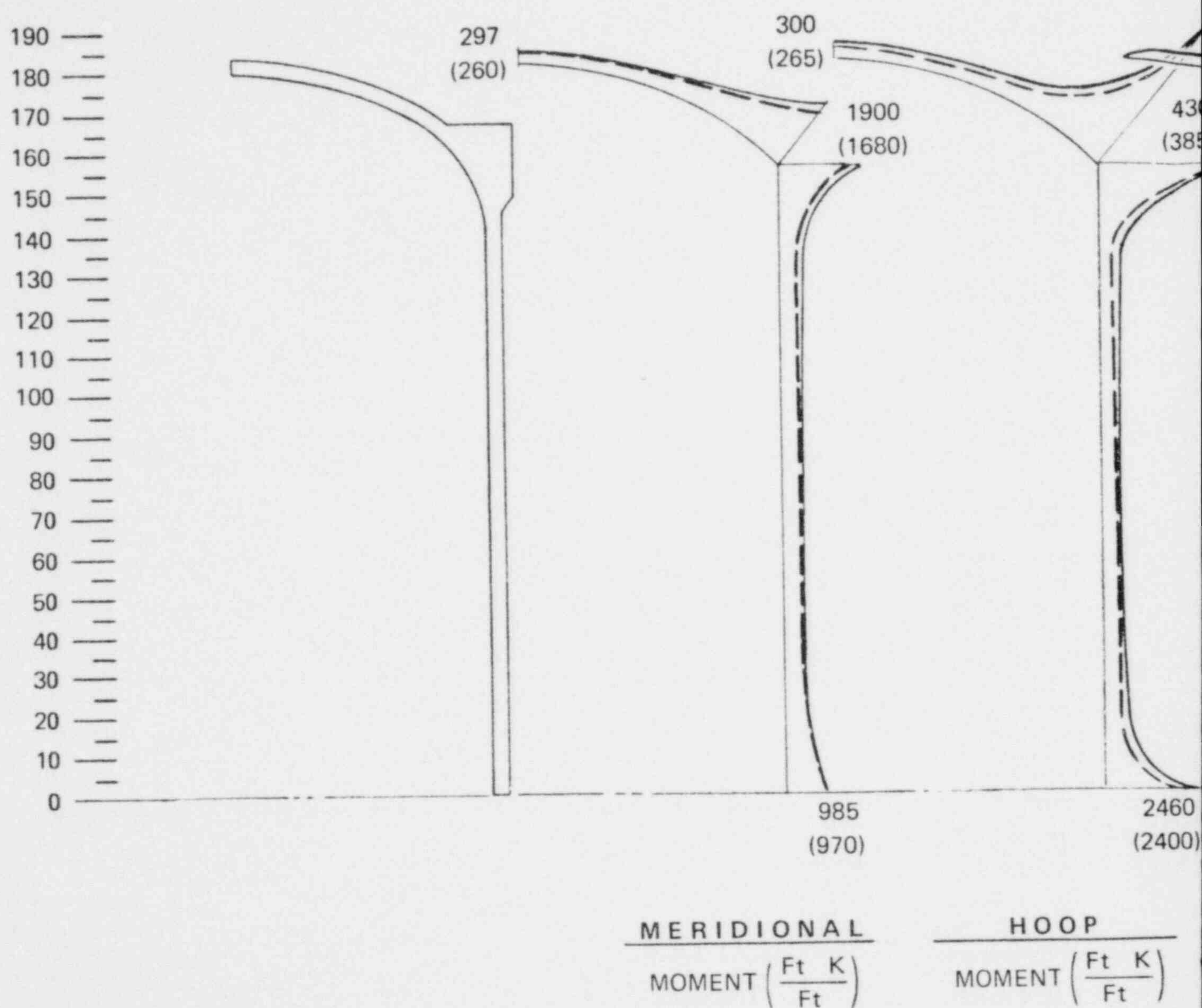


FIGURE 5J.7.2-3
CONTAINMENT STRUCTURE STRESS ANALYSIS
PRELIMINARY MANUAL CALCULATIONS
PRESSURE LOAD (P)

0355

SMUD ~~00271~~
SACRAMENTO MUNICIPAL UTILITY DISTRICT



NOTE:
MOMENTS SHOWN ON TENSION SIDE.

~~00272~~

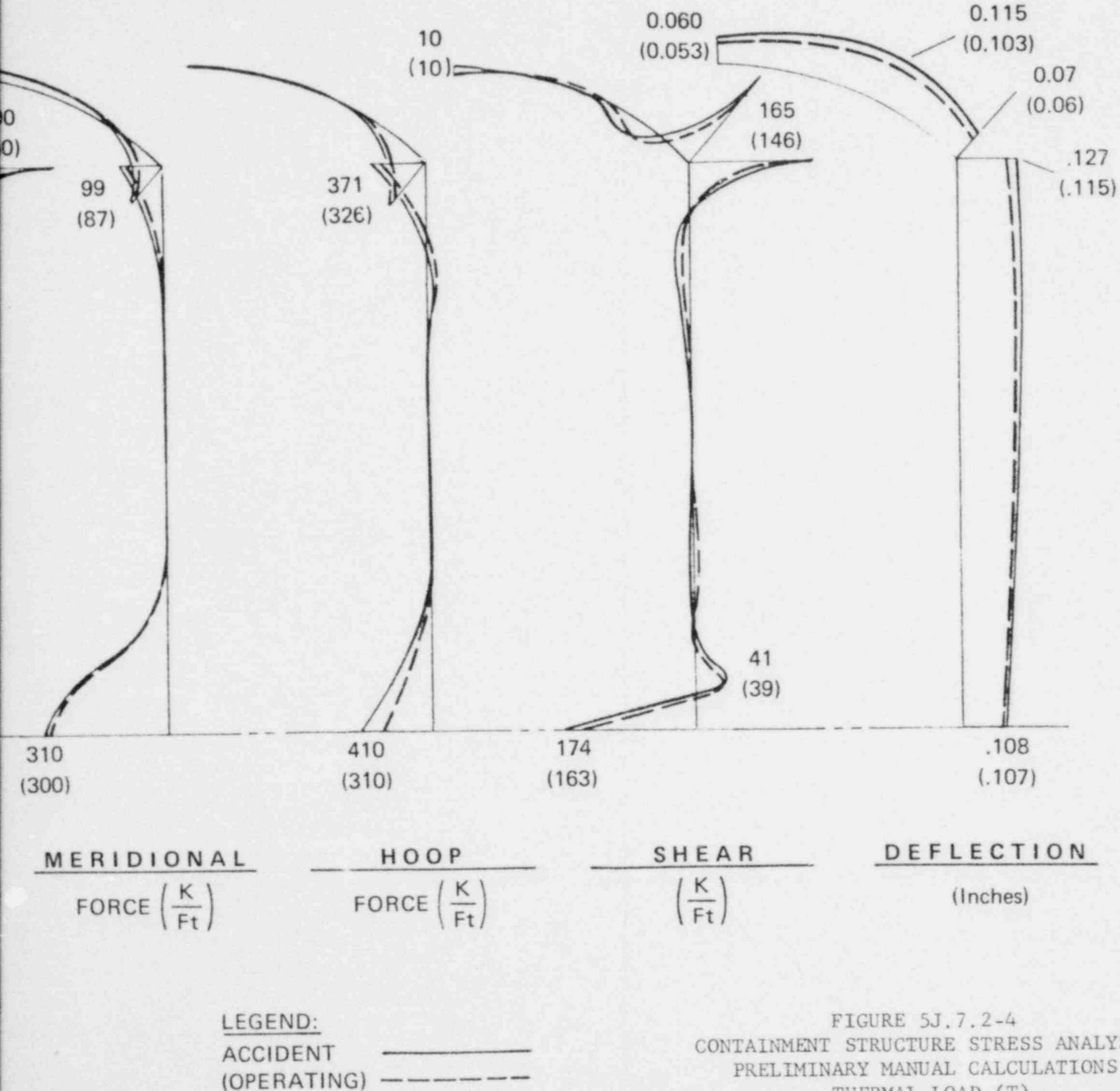


FIGURE 5J.7.2-4
CONTAINMENT STRUCTURE STRESS ANALYSIS
PRELIMINARY MANUAL CALCULATIONS
THERMAL LOAD (T)

0357

SMUD ~~00273~~

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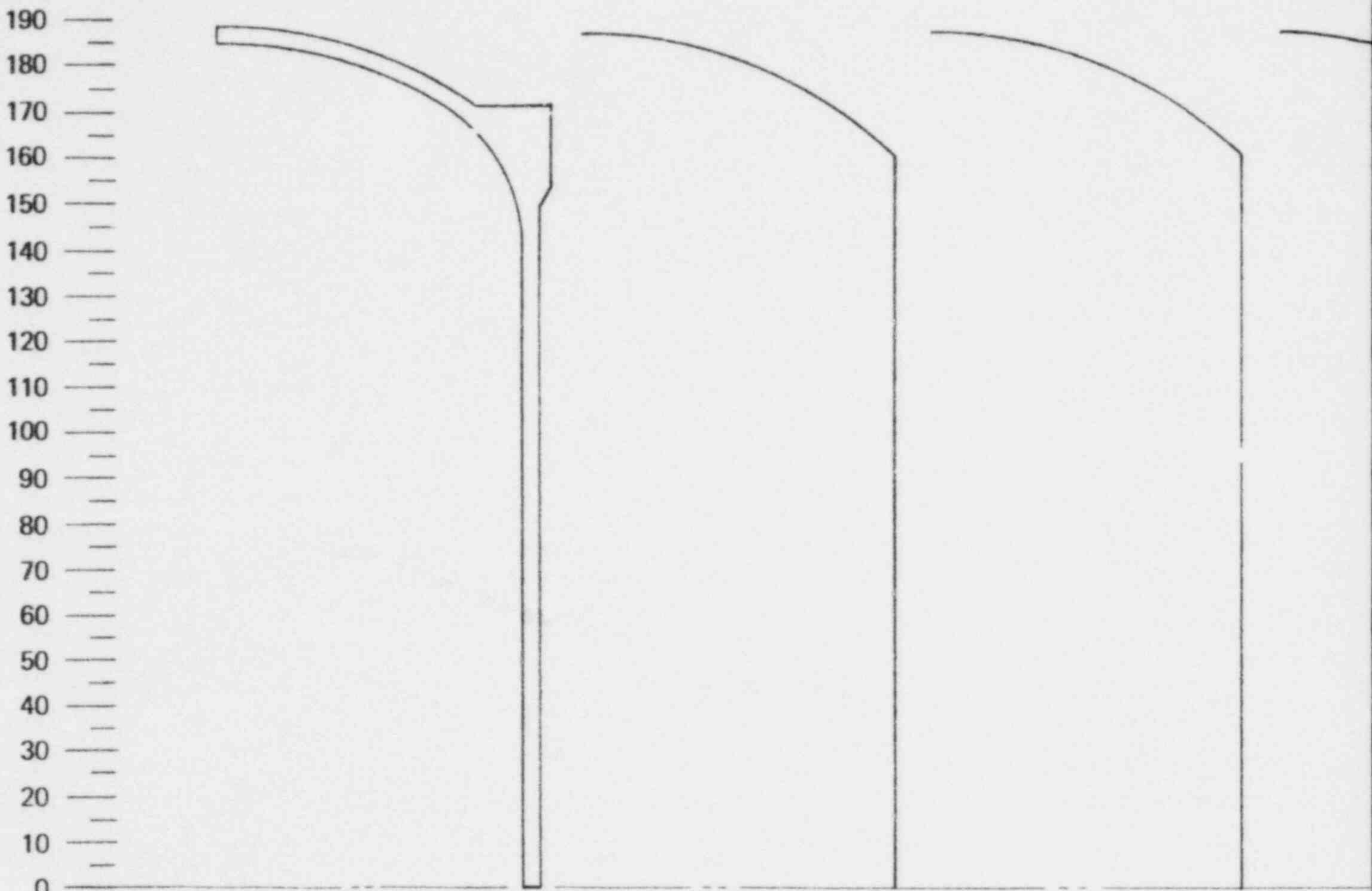

MERIDIONAL

MOMENT $\left(\frac{Ft\ K}{Ft}\right)$

HOOP

MOMENT $\left(\frac{Ft\ K}{Ft}\right)$

NOTE:
MOMENTS SHOWN ON TENSION SIDE.

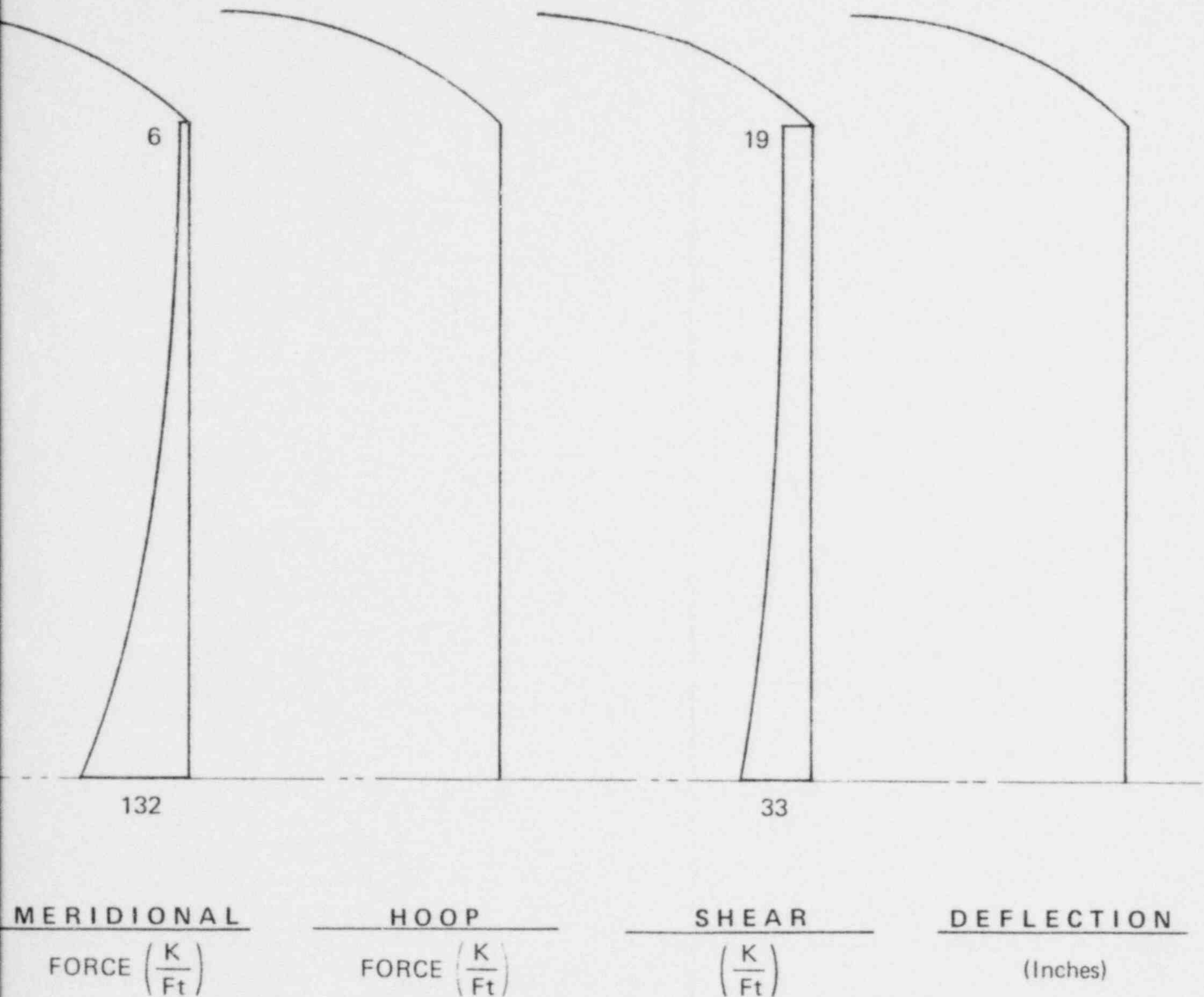

FIGURE 5J.7.2-5
CONTAINMENT STRUCTURE STRESS ANALYSIS
PRELIMINARY MANUAL CALCULATIONS
EARTHQUAKE LOAD (E)

0359

~~00275~~

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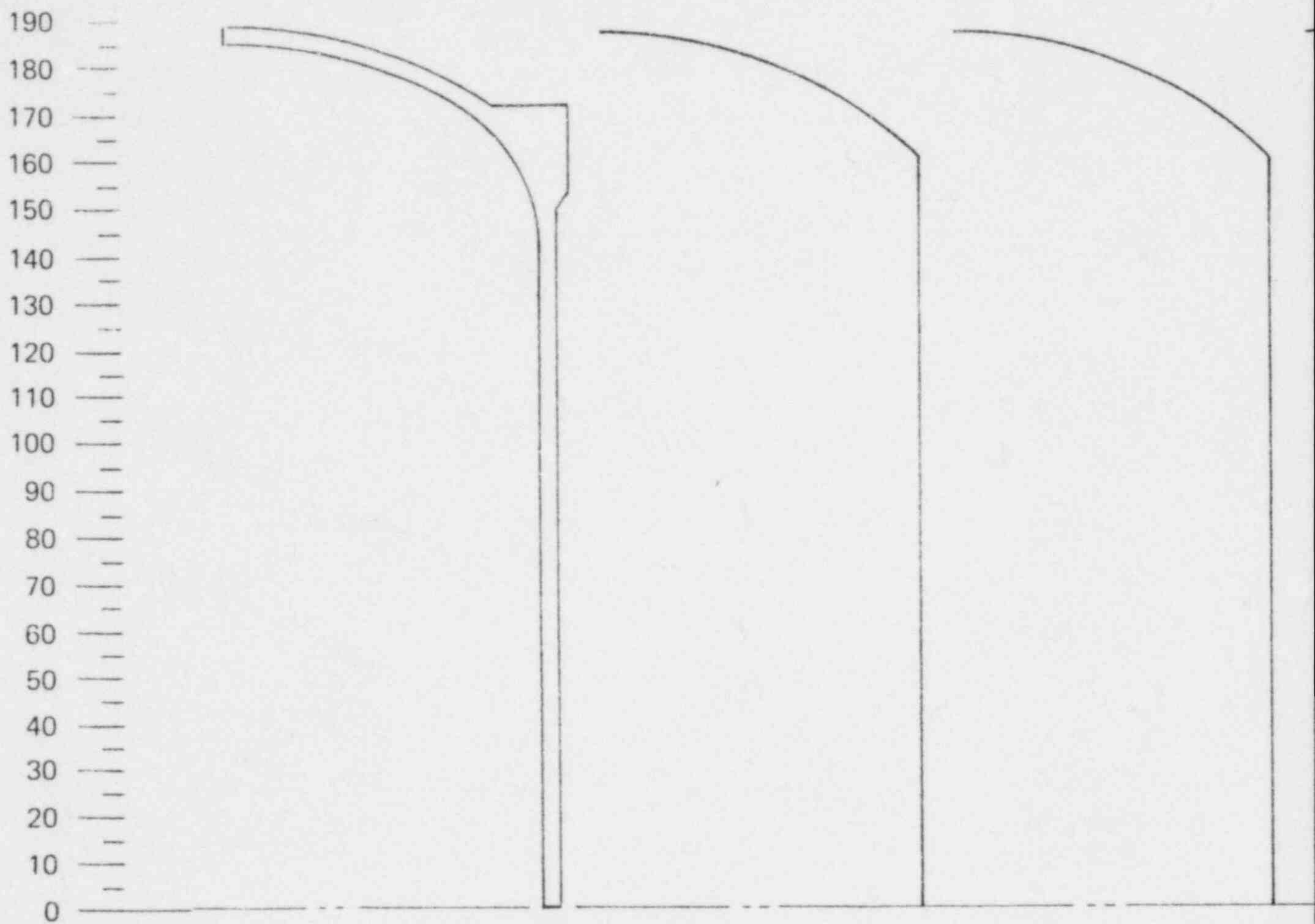

MERIDIONAL

MOMENT $\left(\frac{Ft \ K}{Ft}\right)$

HOOP

MOMENT $\left(\frac{Ft \ K}{Ft}\right)$

NOTE:
MOMENTS SHOWN ON TENSION SIDE.

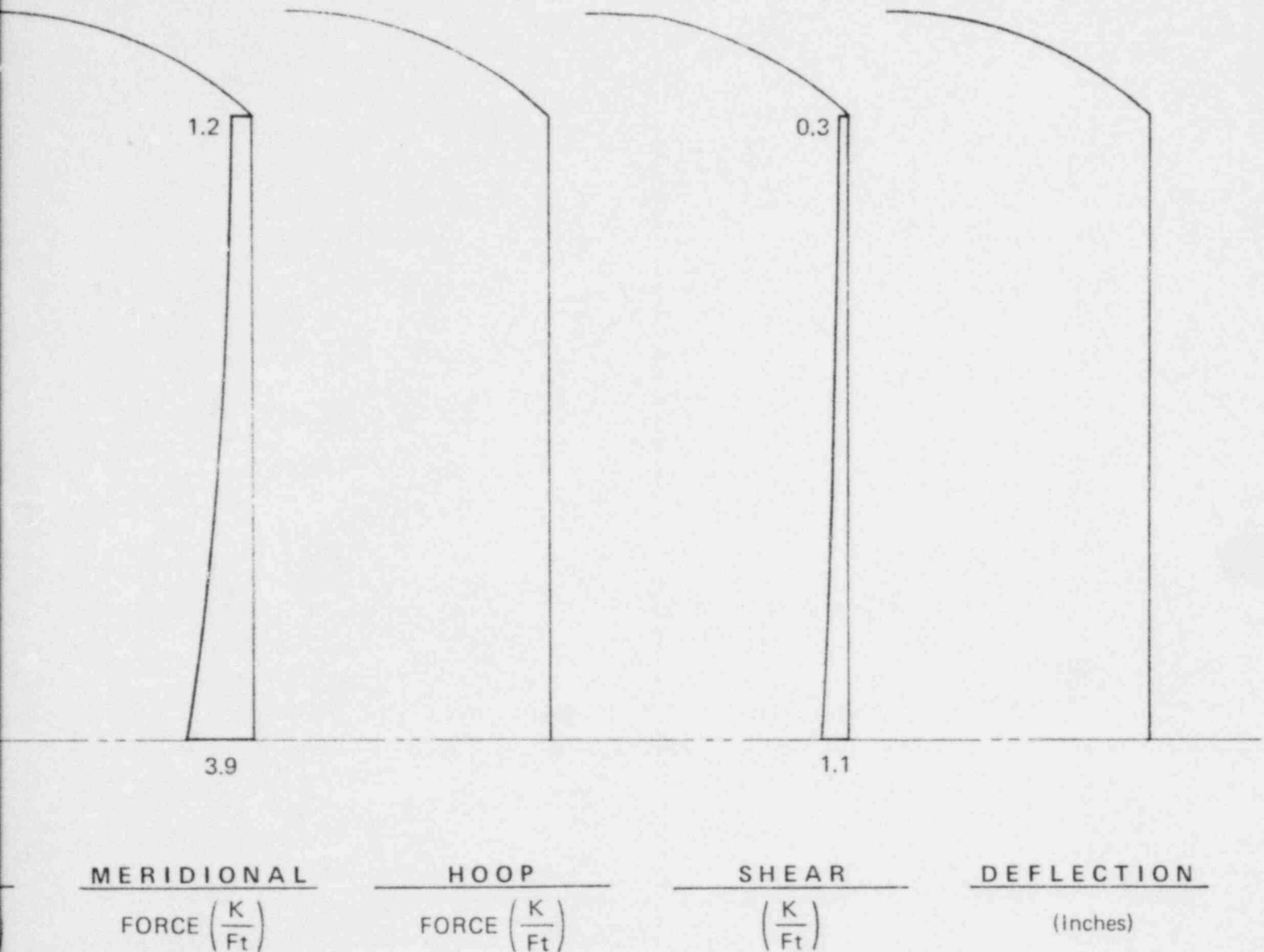

FIGURE 5J.7.2-6
CONTAINMENT STRUCTURE STRESS ANALYSIS
PRELIMINARY MANUAL CALCULATIONS
WIND LOAD (W)

0361

SMUD ~~00277~~

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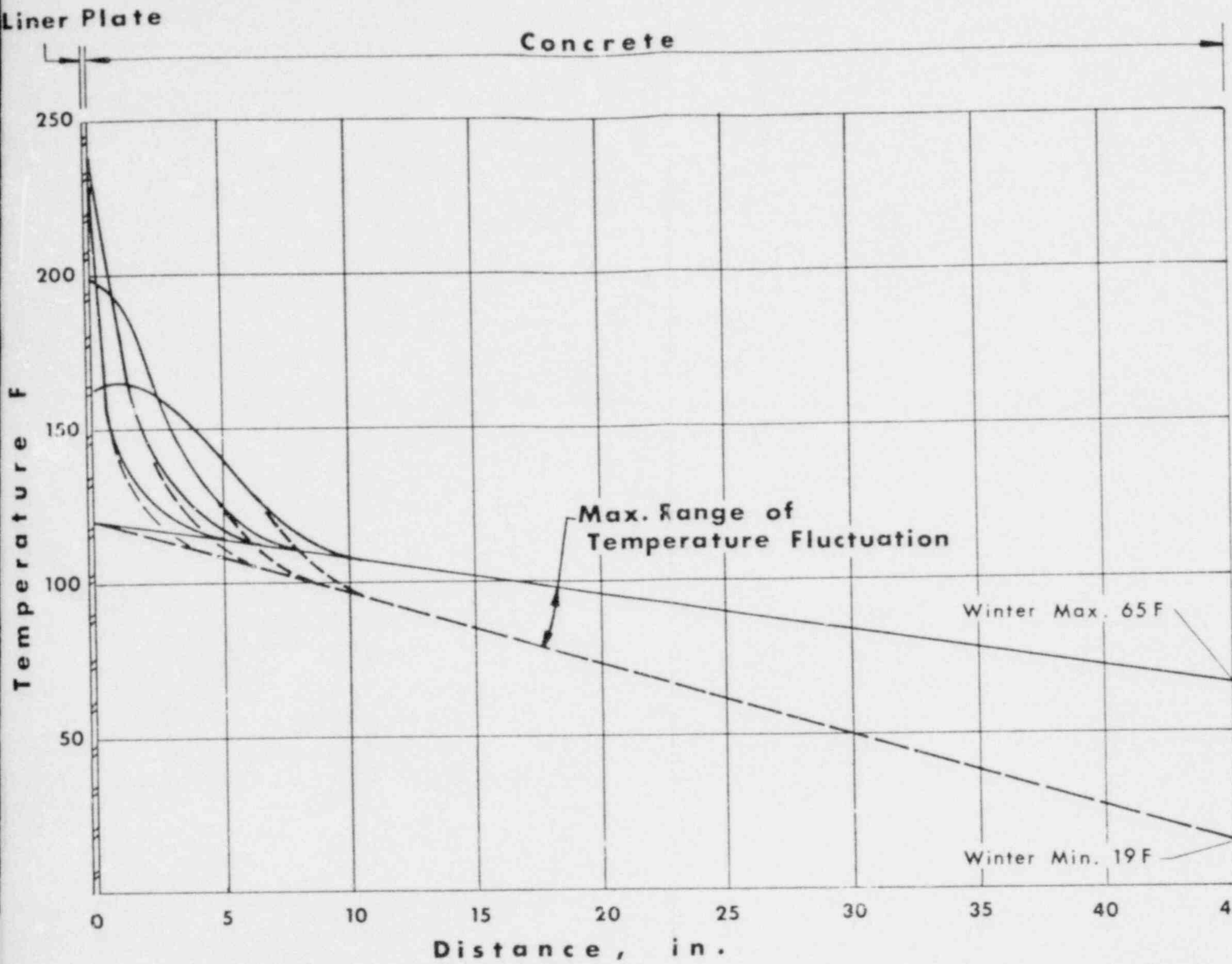


0362

FIGURE 5J.7.2-7
WINTER THERMAL GRADIENT
ACROSS CONTAINMENT WALL

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

~~00270~~

IMAGE EVALUATION
TEST TARGET (MT-3)

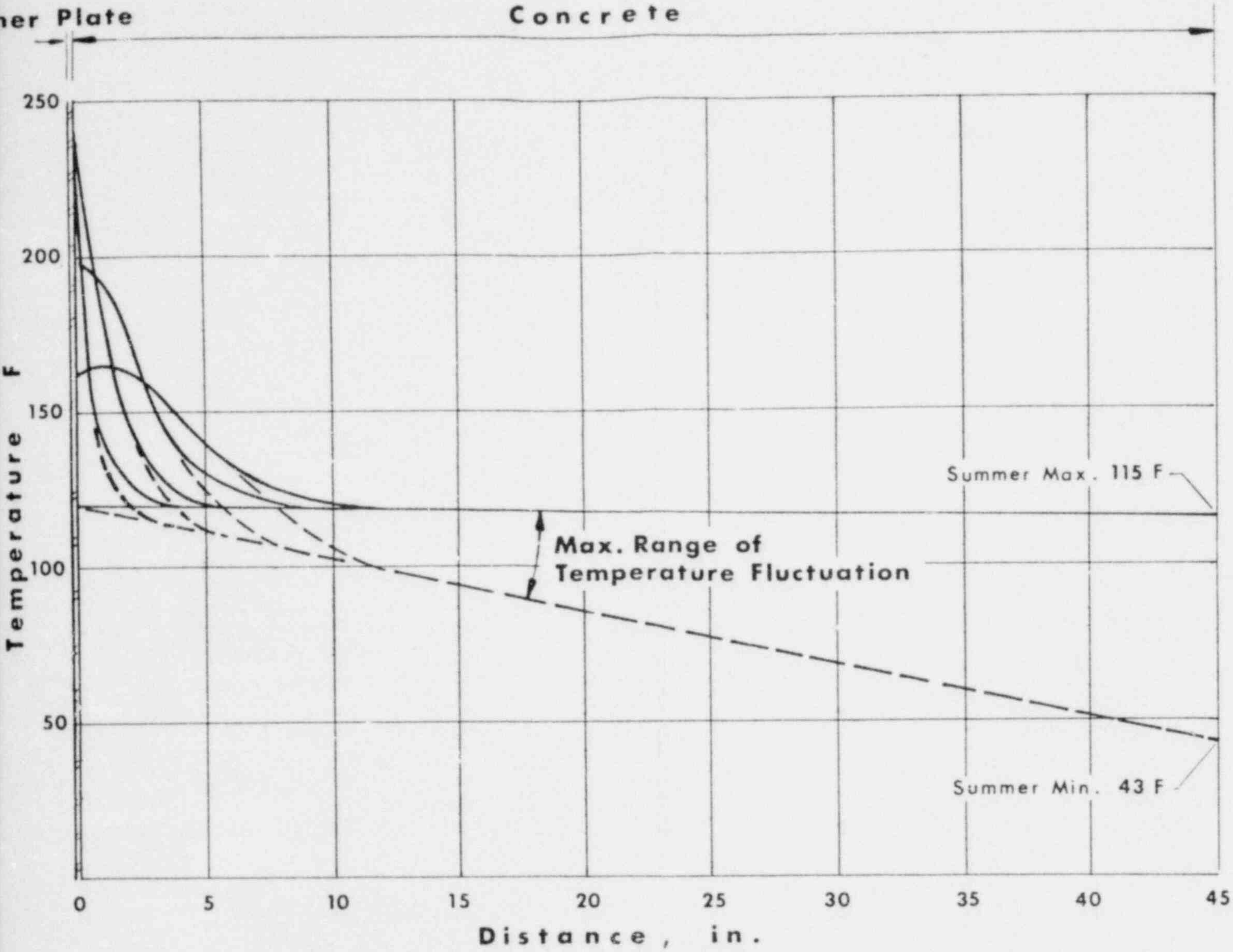
6"

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART



0001

FIGURE 5J.7.2-8
SUMMER THERMAL GRADIENT
ACROSS CONTAINMENT WALL

SMUD

~~00279~~

SACRAMENTO MUNICIPAL UTILITY DISTRICT