SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO. 1

PRELIMINARY SAFETY ANALYSIS REPORT

Volume I

NOVEMBER 1967

I. INTRODUCTION AND SUMMARY

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1. INTRODUCTION AND SUMMARY

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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 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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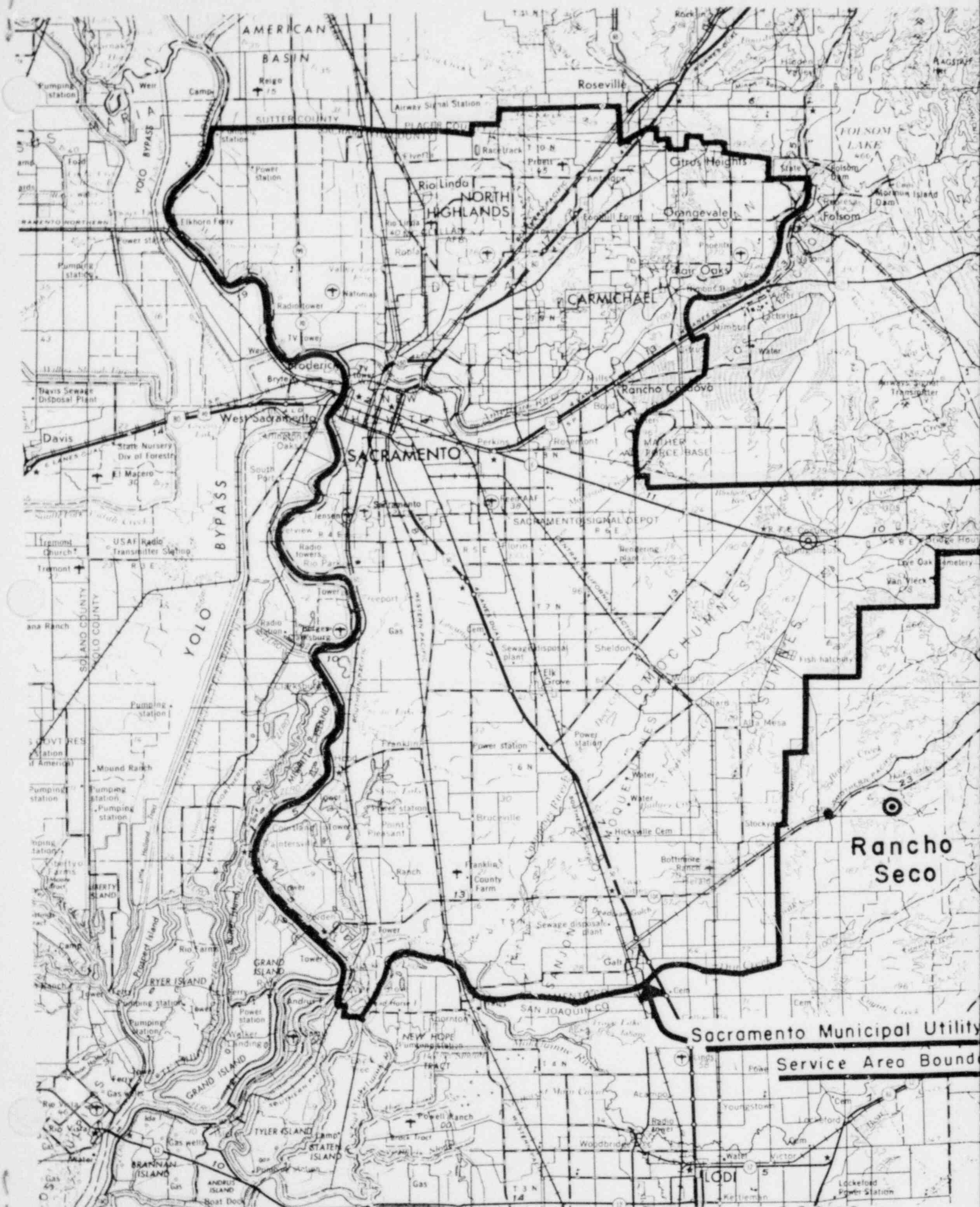

Sacramento Municipal Utility
Service Area Bound
Rancho
Seco
SACRAMENTO
NORTH HIGHLANDS
CARMICHAEL
Citrus Heights
Orangevale
Fair Oaks
Rio Linda
Roseville
Folsom
FOLSOM LAKE
Rancho Cordova
MATHER FORCE BASE
West Sacramento
Davis
AMERICAN BASIN
SUTTER COUNTY
PLACER COUNTY
YOLO BYPASS
YOLO
SACRAMENTO SIGNAL DEPOT
Elk Grove
Galt
LODI
GRAND ISLAND
RYER ISLAND
TYLER ISLAND
STATEN ISLAND
BRANNAN ISLAND
SAN JOAQUIN CO
SOLANO COUNTY
YOLO COUNTY
COSUMNES
MOKELUMNE

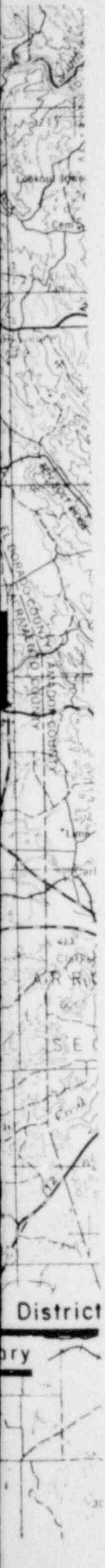

FIGURE 1.1-1
SACRAMENTO MUNICIPAL UTILITY DISTRICT
SERVICE AREA BOUNDARY

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SACRAMENTO MUNICIPAL UTILITY DISTRICT


SWITCHYARD
REACTOR
UNIT 1
FUTURE UNIT 2
NUCLEAR SERVICE SPRAY POND
NUCLEAR SERVICE SPRAY POND
WAREHOUSE & SHOPS
INTAKE
FENCE
GRADE EL. 764.0'
COOLING TOWERS FOR UNIT 1
COOLING TOWERS FOR FUT UNIT 2

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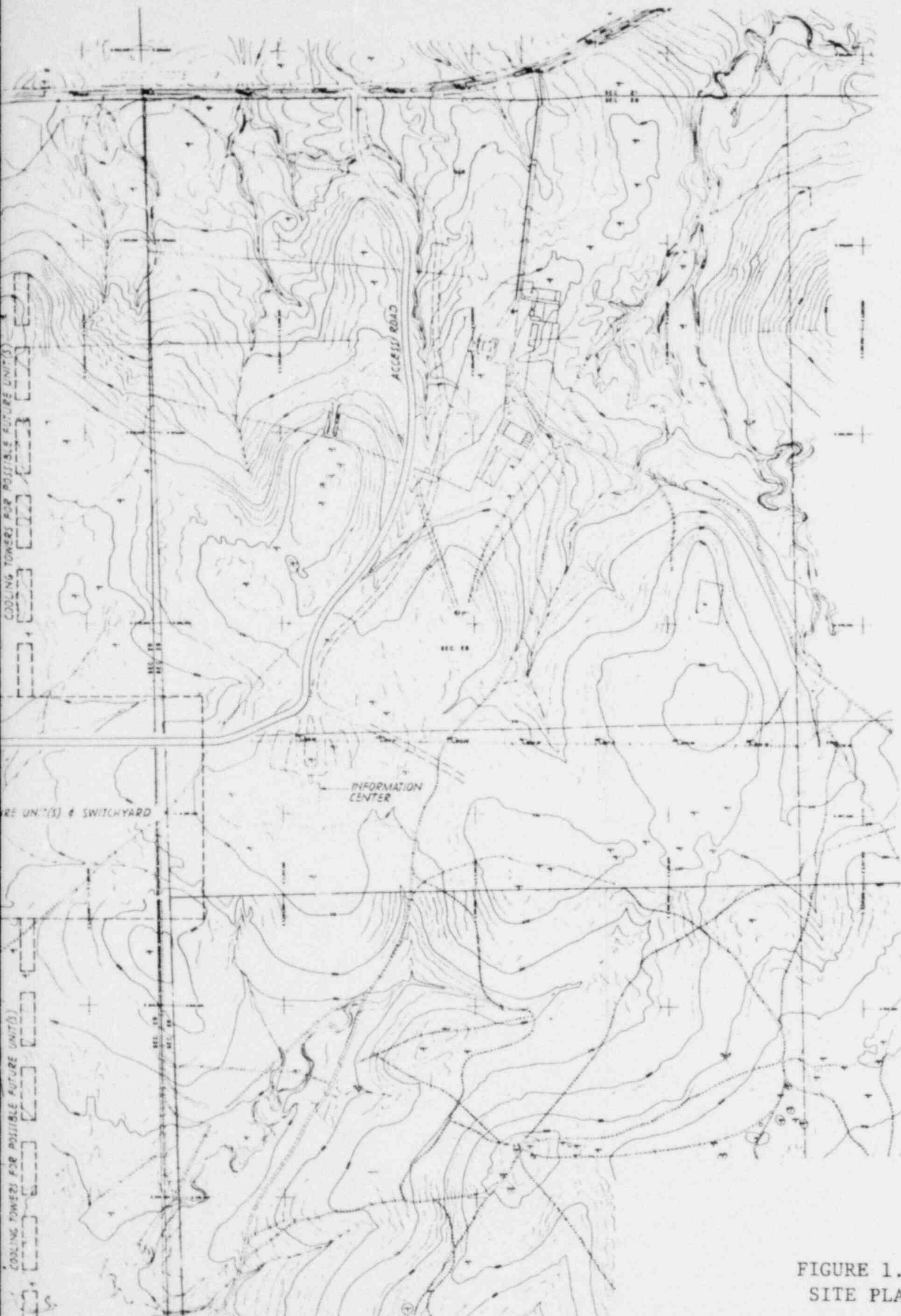

FIGURE 1.1-2
SITE PLAN

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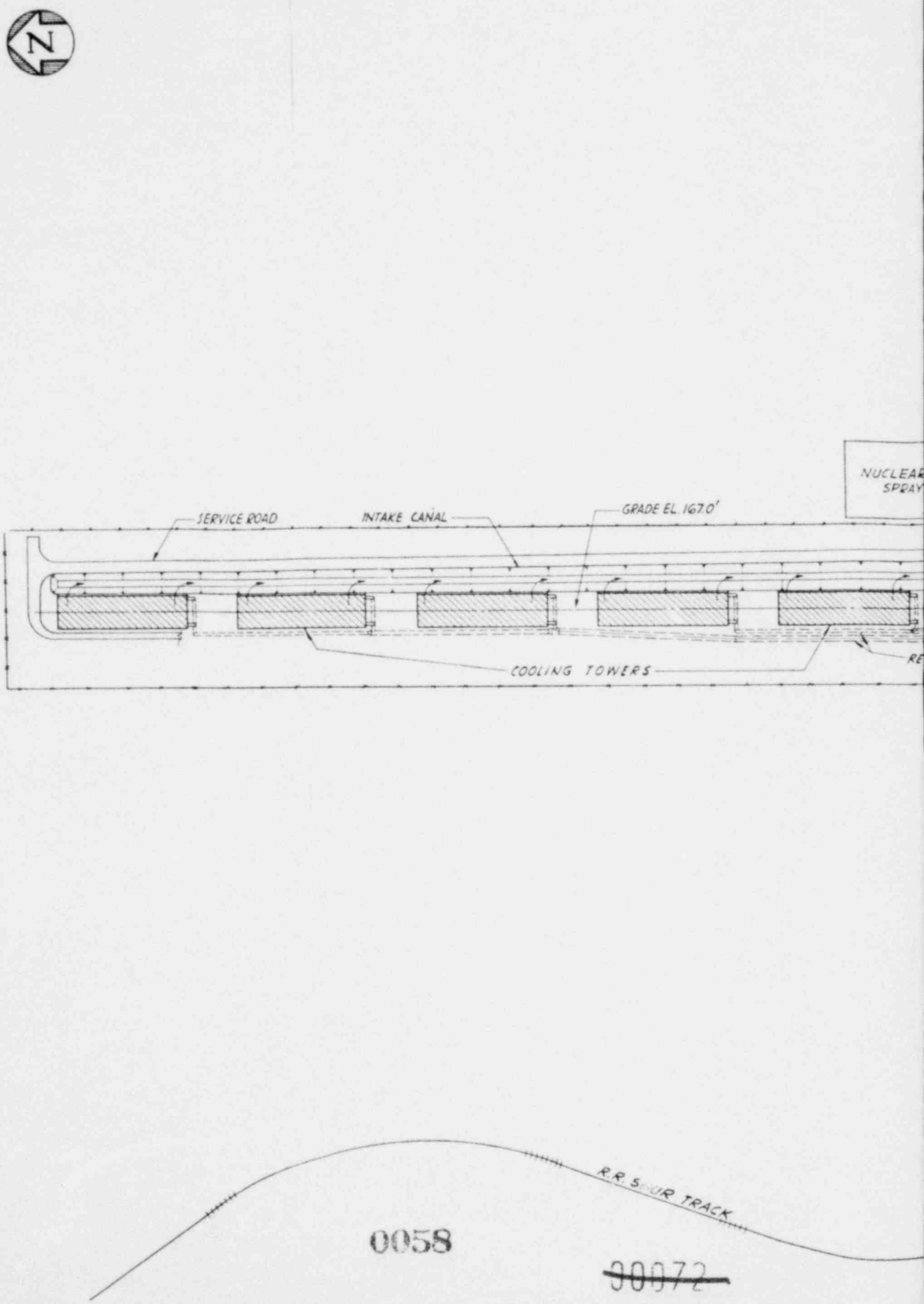
SACRAMENTO MUNICIPAL UTILITY DISTRICT


N
NUCLEAR
SPRAY
SERVICE ROAD
INTAKE CANAL
GRADE EL. 167.0'
COOLING TOWERS
R.R. SPUR TRACK

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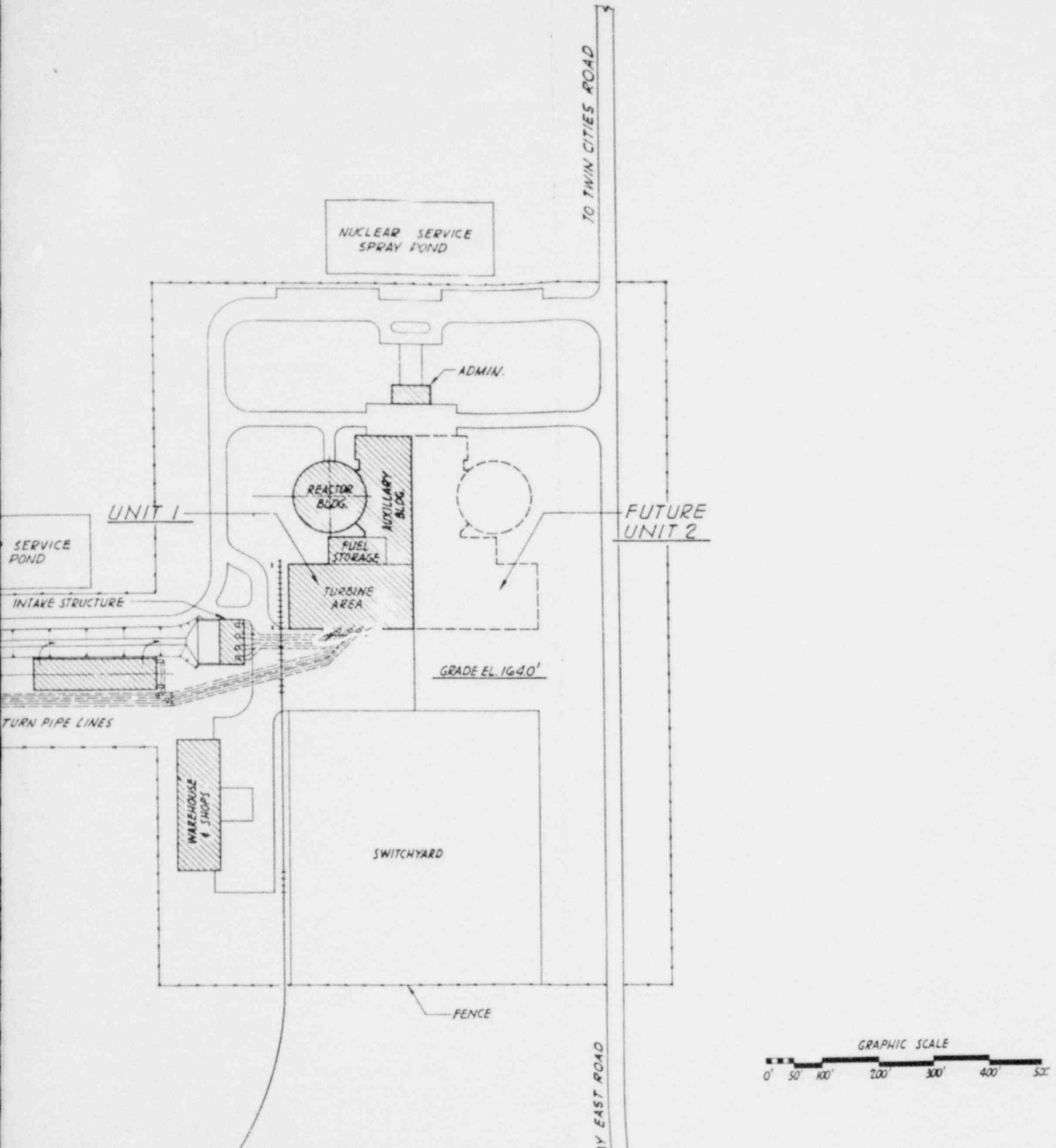

FIGURE 1.1-3
PLOT PLAN

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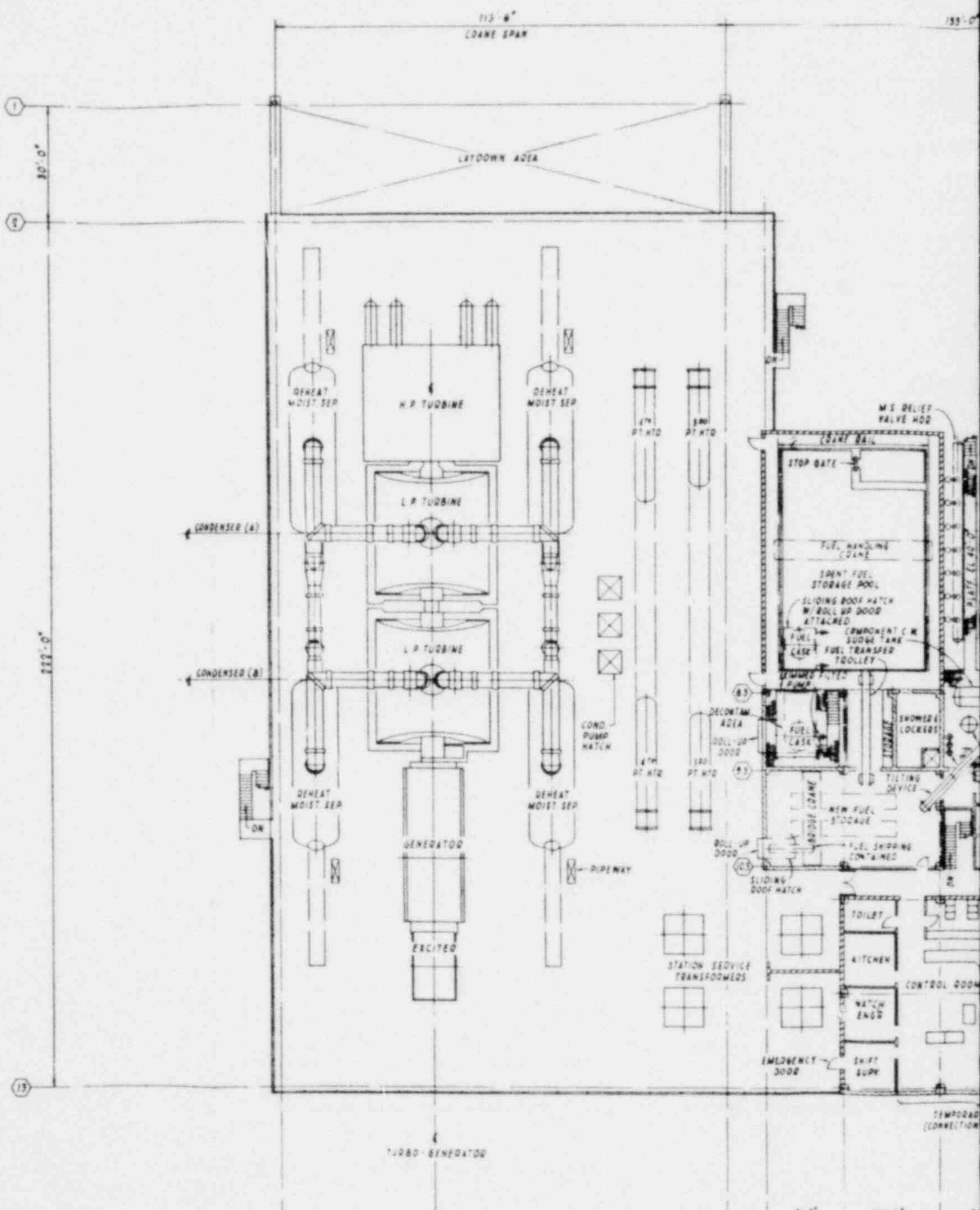
Amendment 2



115'-6"
CRANE SPAN
LAYDOWN AREA
30'-0"
REHEAT MOIST. SEP
H.P. TURBINE
L.P. TURBINE
CONDENSER (A)
CONDENSER (B)
GENERATOR
EXCITER
PIPEWAY
COND. PUMP HATCH
M.S. RELIEF VALVE HDR
STOP GATE
FUEL HANDLING CRANE
SPENT FUEL STORAGE POOL
FUEL TRANSFER TROLLEY
DECONTAM AREA
FUEL CASK
TILTING DEVICE
NEW FUEL STORAGE
SLIDING ROOF HATCH
STATION SERVICE TRANSFORMERS
TOILET
KITCHEN
CONTROL ROOM
SHIFT SUPV
EMERGENCY DOOR
TURBO-GENERATOR
35'-0"
74'-6"
10'-0"
19'-0"
24'-0"

0060

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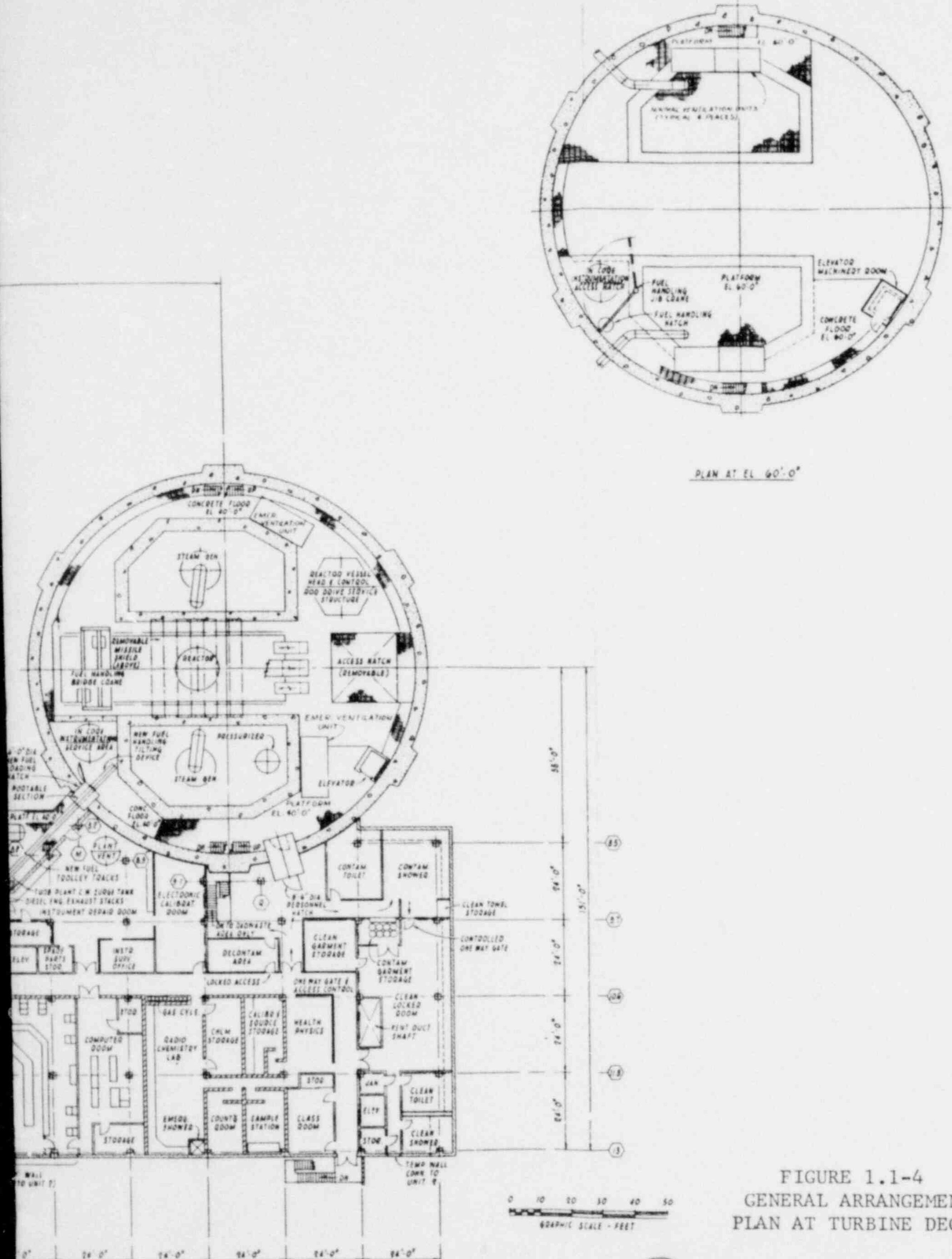

FIGURE 1.1-4
GENERAL ARRANGEMENT
PLAN AT TURBINE DECK

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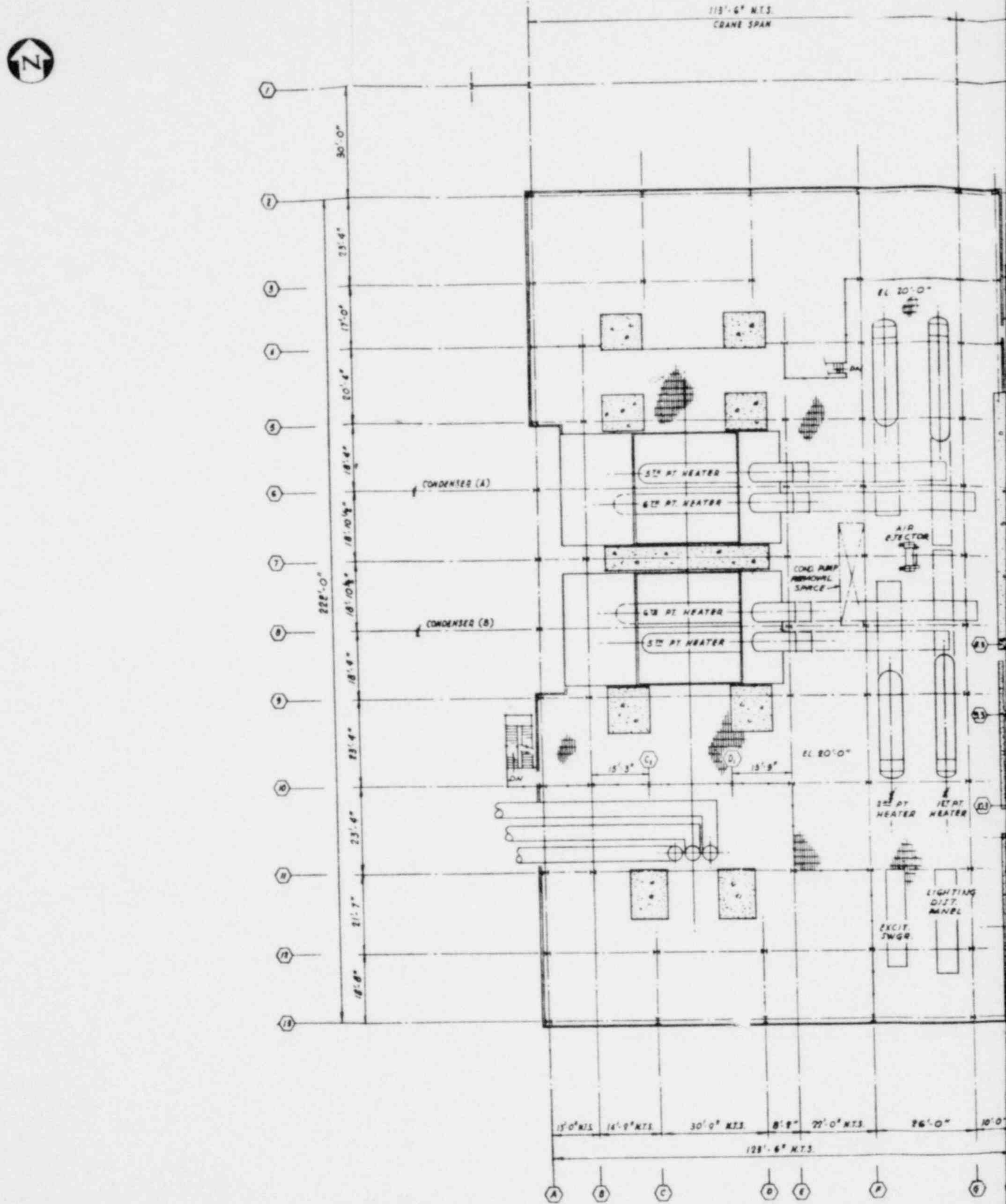
Amendment 2



113'-6" N.T.S.
CRANE SPAN
CONDENSER (A)
CONDENSER (B)
5TH PT HEATER
6TH PT HEATER
4TH PT HEATER
3RD PT HEATER
EL 20'-0"
AIR EJECTOR
COND. PUMP REMOVAL SPACE
2ND PT HEATER
1ST PT HEATER
LIGHTING DIST. PANEL
EXCIT. SWGR.
222'-0"
128'-6" N.T.S.

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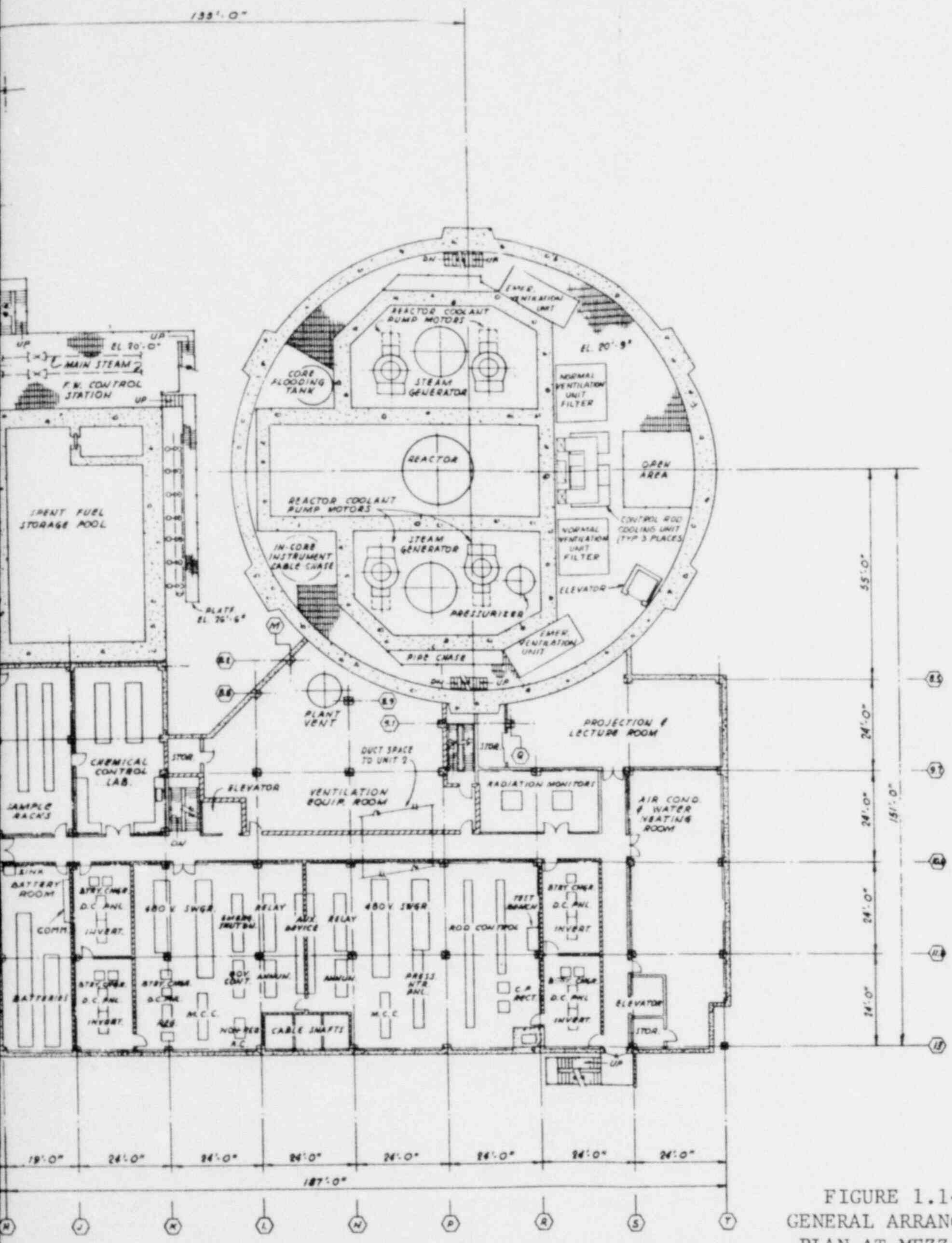

FIGURE 1.1-5
GENERAL ARRANGEMENT
PLAN AT MEZZANINE

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SACRAMENTO MUNICIPAL UTILITY DISTRICT

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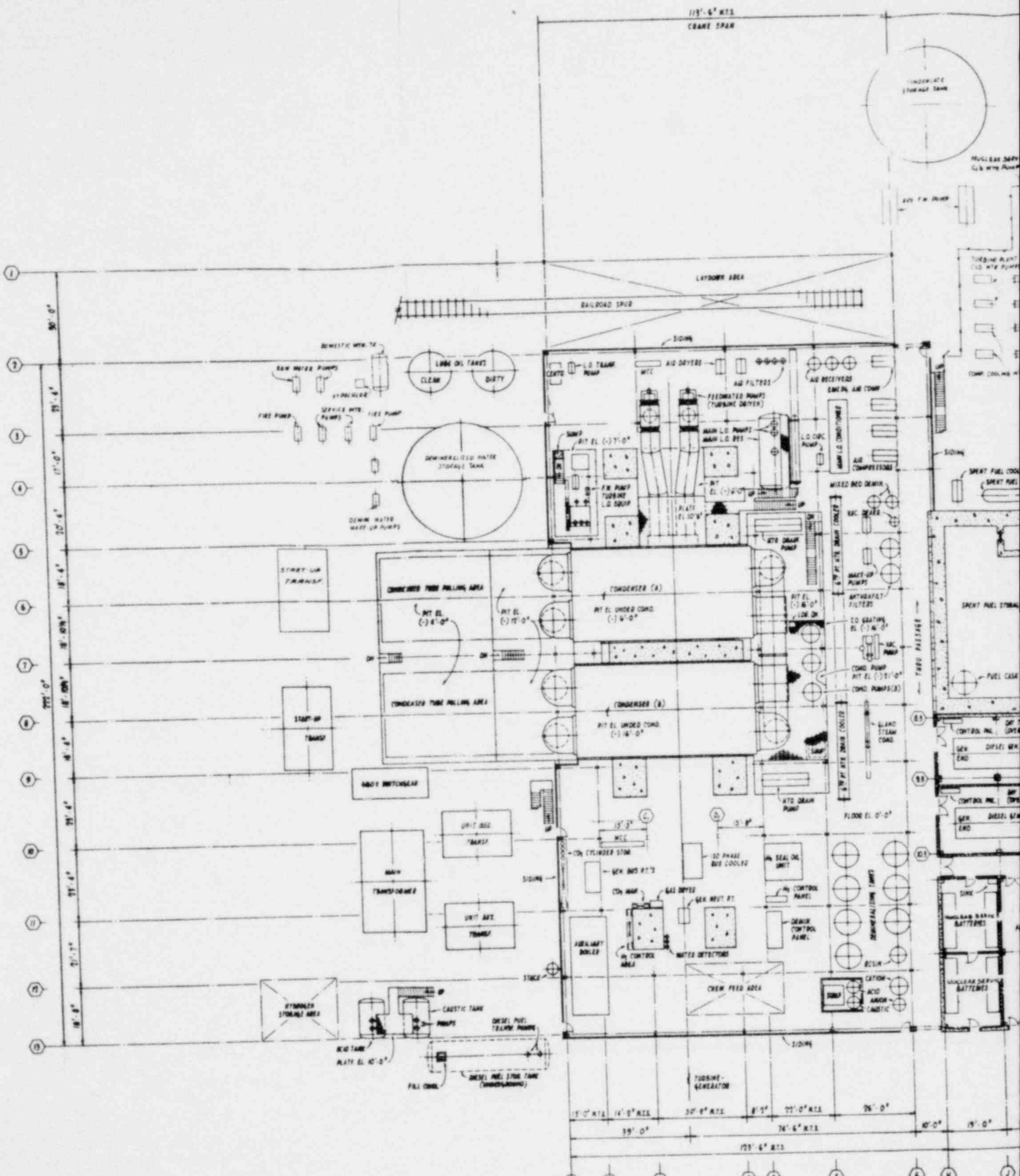

119'-6" MTS
CRANE SPAN
CONDENSATE STORAGE TANK
LAYDOWN AREA
RAILROAD SPUR
SIDING
LUBE OIL TANKS
CLEAN
DIRTY
DEMINERALIZED WATER STORAGE TANK
CONDENSER (A)
CONDENSER (B)
START-UP TRANSF.
MAIN TRANSFORMER
UNIT AUX. TRANSF.
AUXILIARY BOILER
CHEM FEED AREA
TURBINE-GENERATOR
THRU PASSAGE
SPENT FUEL STORAGE

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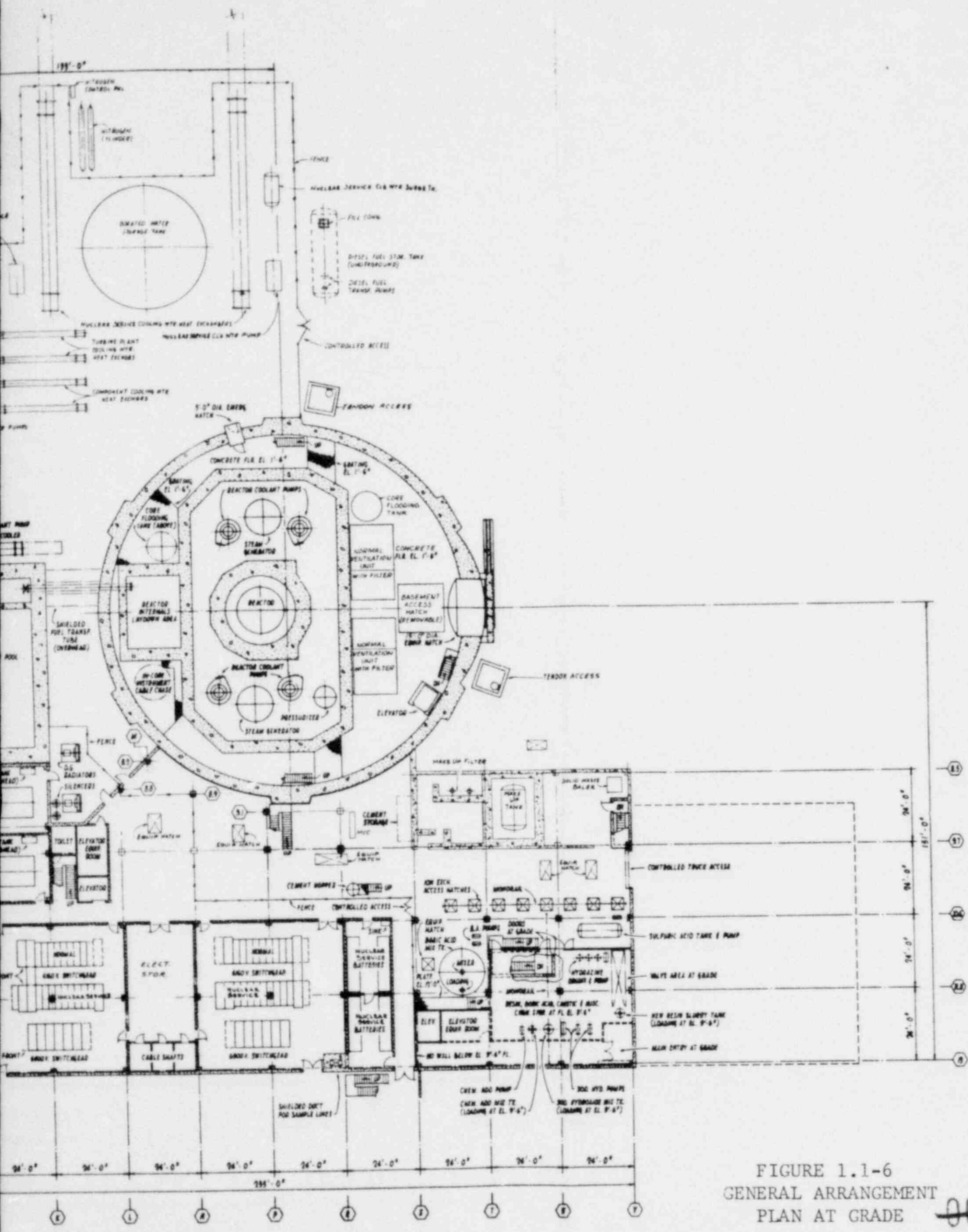

FIGURE 1.1-6
GENERAL ARRANGEMENT
PLAN AT GRADE

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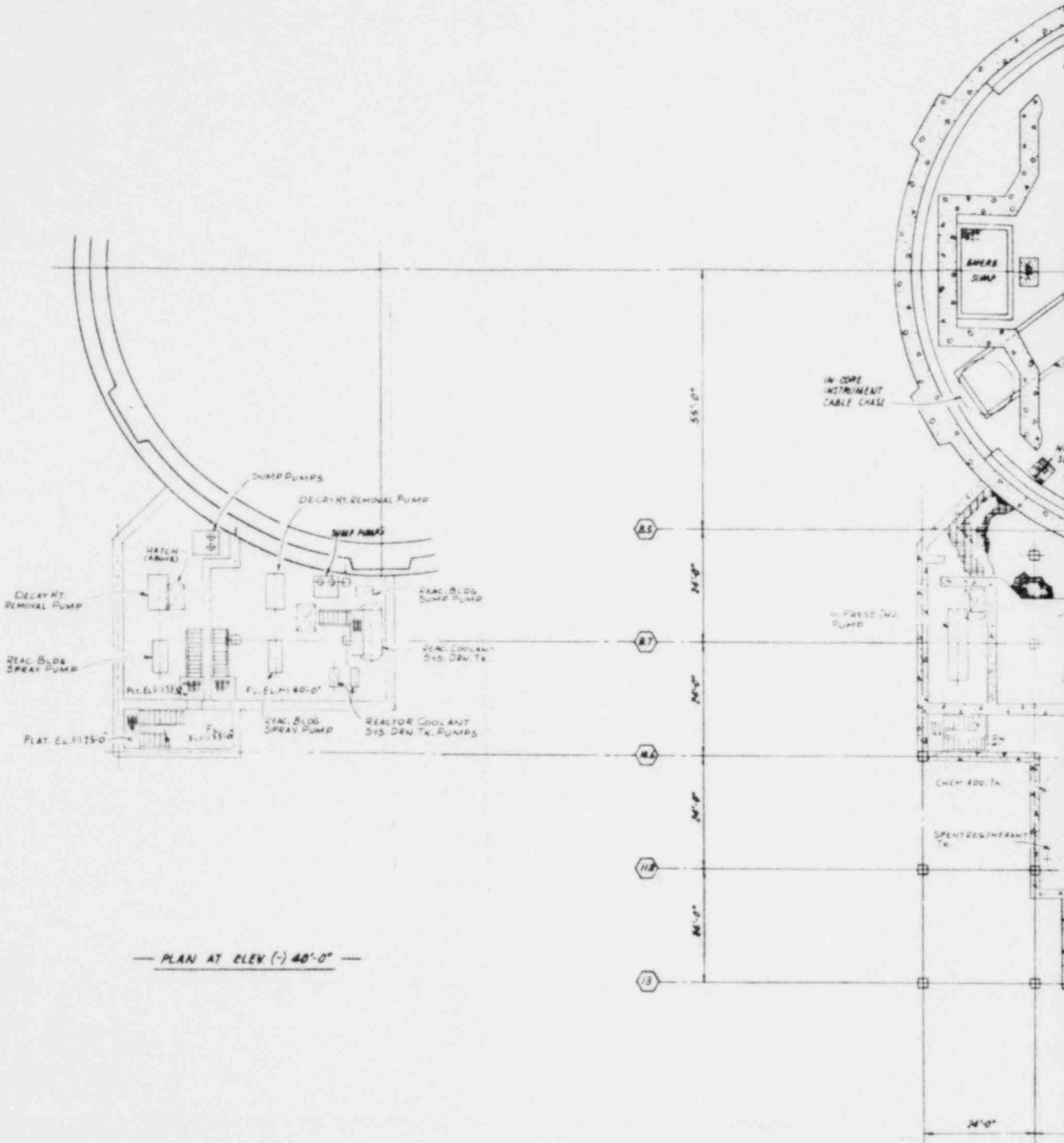

— PLAN AT ELEV (-) 40'-0" —

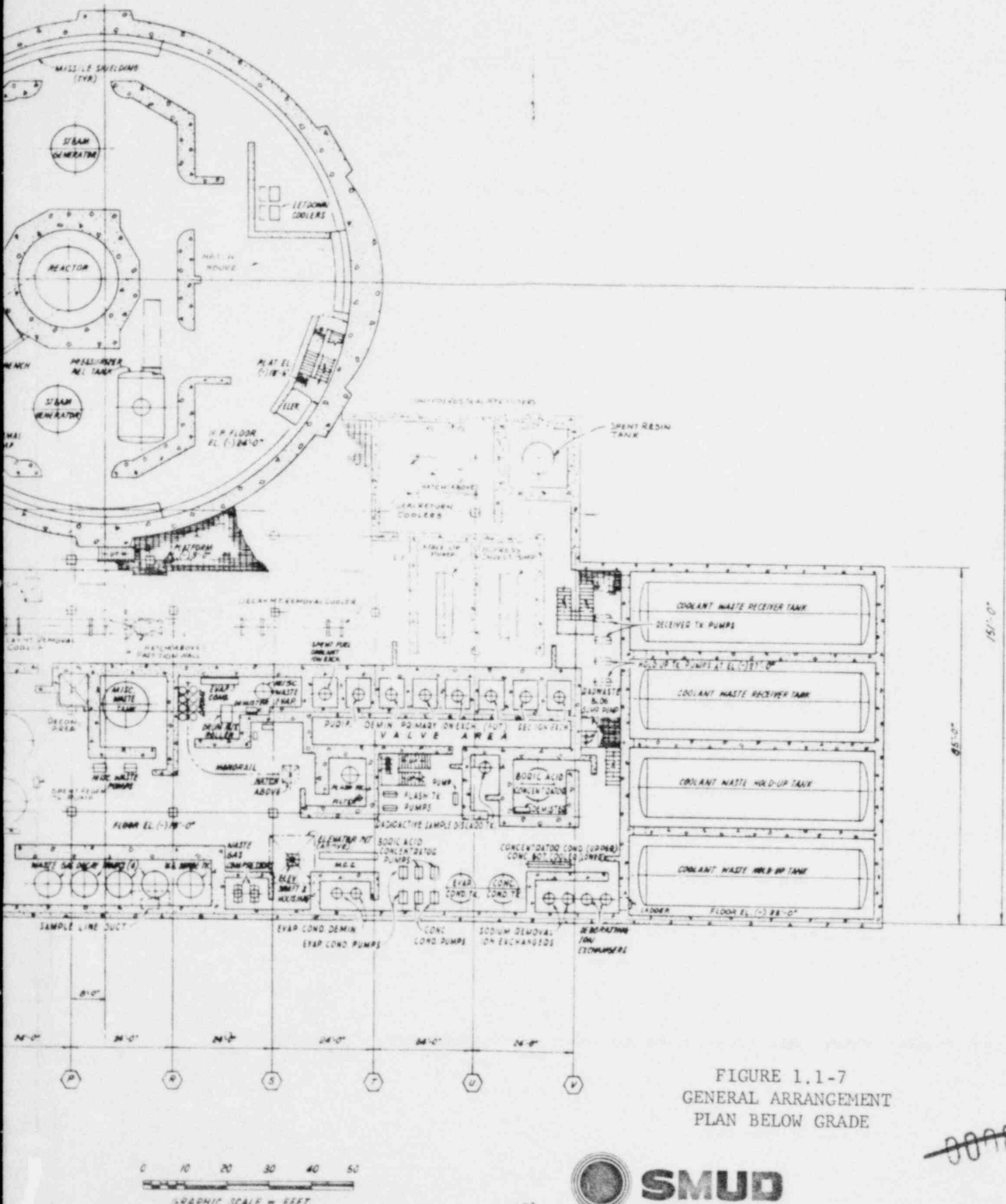

FIGURE 1.1-7
GENERAL ARRANGEMENT
PLAN BELOW GRADE

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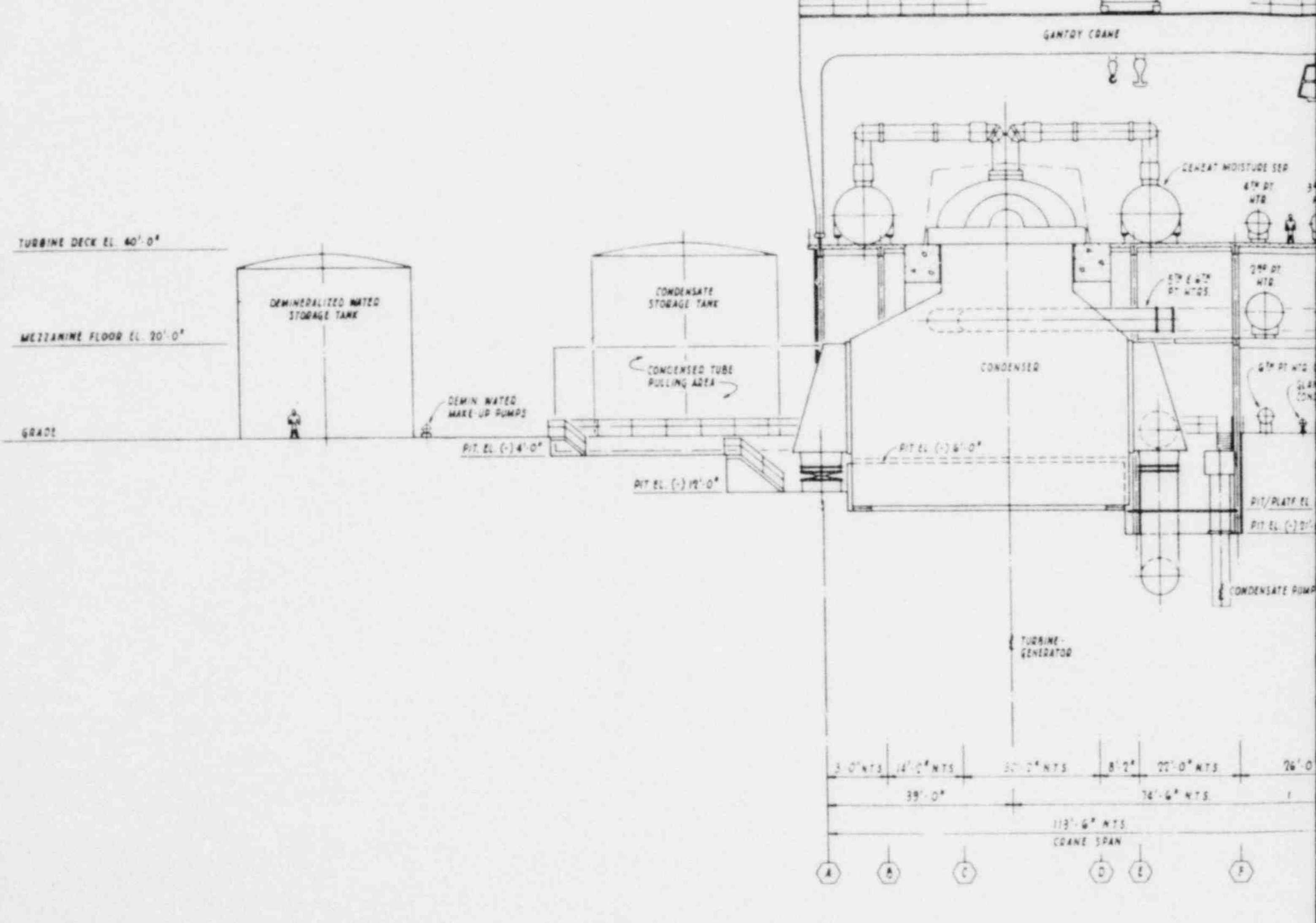
Amendment 2


TURBINE DECK EL. 60'-0"
MEZZANINE FLOOR EL. 20'-0"
GRADE
DEMINERALIZED WATER STORAGE TANK
DEMIN WATER MAKE-UP PUMPS
CONDENSATE STORAGE TANK
CONDENSER TUBE PULLING AREA
PIT EL. (-) 4'-0"
PIT EL. (-) 19'-0"
GANTRY CRANE
REHEAT MOISTURE SEP
4TH PT HTR
5TH & 6TH PT HTRS
2ND PT HTR
CONDENSER
PIT EL. (-) 6'-0"
PIT/PLATF EL
CONDENSATE PUMP
TURBINE-GENERATOR
3'-0" NTS
14'-0" NTS
8'-2"
22'-0" NTS
39'-0"
74'-6" NTS
113'-6" NTS
CRANE SPAN
A
B
C
D
E
F

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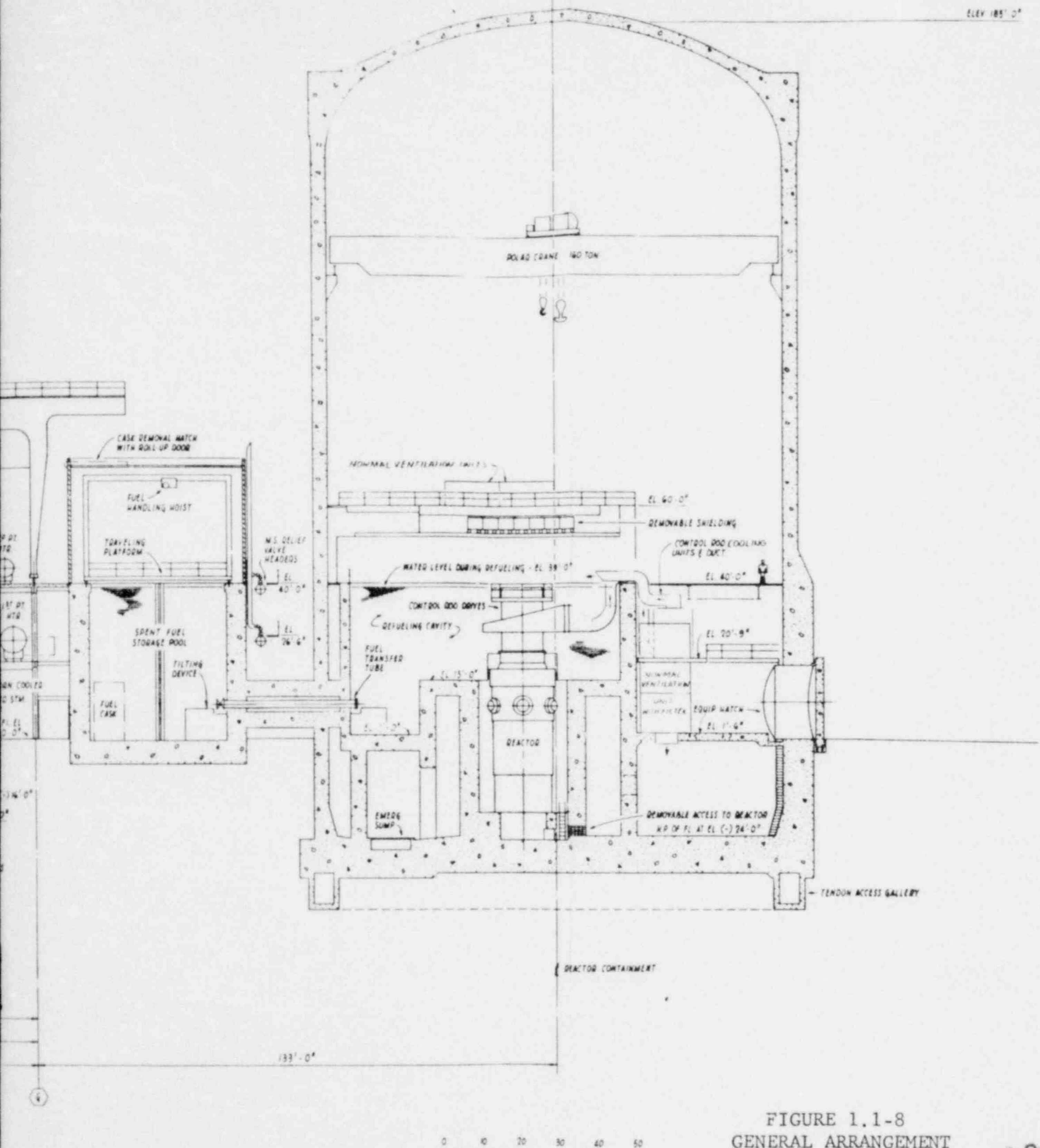

FIGURE 1.1-8
GENERAL ARRANGEMENT
SECTIONAL ELEVATIONS
SHEET 1

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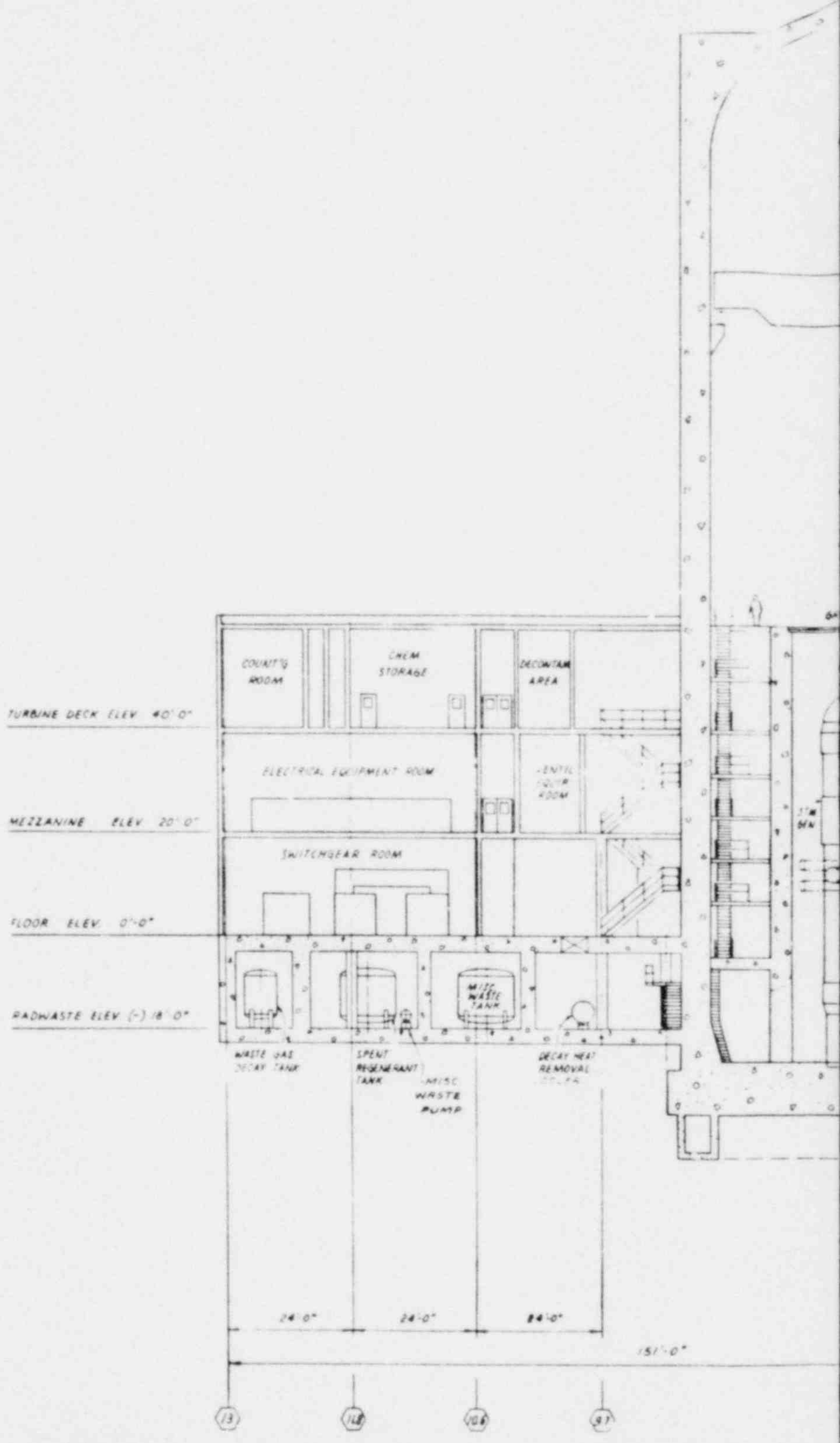

TURBINE DECK ELEV. 40'-0"
MEZZANINE ELEV. 20'-0"
FLOOR ELEV. 0'-0"
RADWASTE ELEV. (-) 18'-0"
COUNT'G ROOM
CHEM STORAGE
DECONTAM AREA
ELECTRICAL EQUIPMENT ROOM
SWITCHGEAR ROOM
MISC WASTE TANK
WASTE GAS DECAY TANK
SPENT REGENERANT TANK
MISC WASTE PUMP
DECAY HEAT REMOVAL
24'-0"
24'-0"
24'-0"
151'-0"
13
10

0070 ~~00084~~

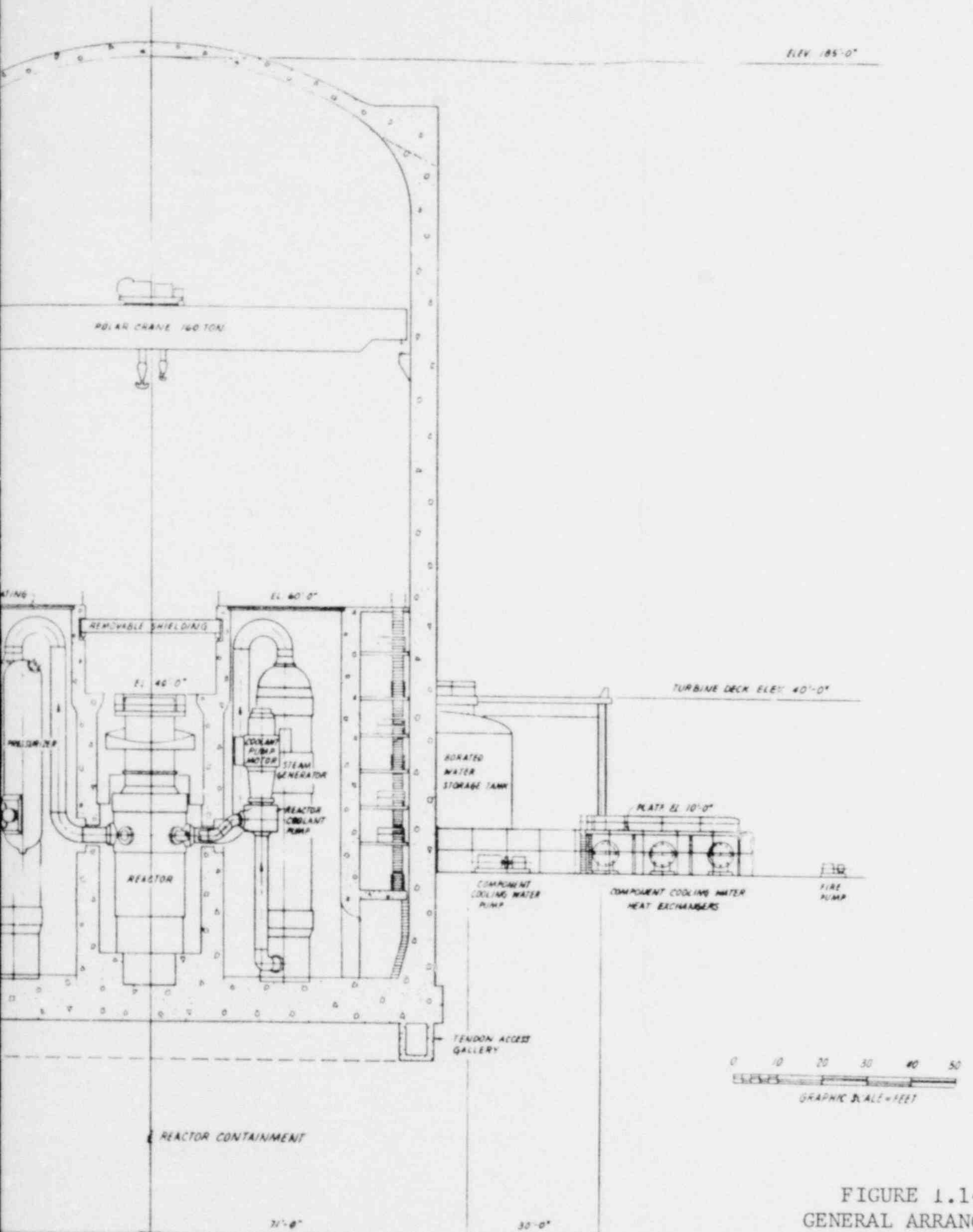

FIGURE 1.1-8
GENERAL ARRANGEMENT
SECTIONAL ELEVATIONS
SHEET 2

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1.2 DESIGN HIGHLIGHTS

1.2.1 SITE CHARACTERISTICS

The 2,480 acre site is characterized by a 2,100 foot minimum exclusion radius; isolation from population centers; sound foundation for structures; an abundant supply of makeup water for cooling towers; an ample supply of emergency power; and favorable conditions of meteorology, geology, seismology, and hydrology.

1.2.2 POWER LEVEL

Initially licensed power for the reactor core is proposed at 2,452 Mwt, and core performance analyses in this report are based on this initial power level. Operating confirmation of reactor core parameters is expected to support an ultimate core power level of 2,568 Mwt, and the unit will be designed to operate at this output. The analyses of accidents that could release fission products to the environment have been evaluated on the basis of 2,568 Mwt. An additional 16 Mwt will be available to the cycle from the contribution of the reactor coolant pumps, resulting in a gross electrical output of about 850 Mwe nominal.

1.2.3 PEAK SPECIFIC POWER LEVEL

The peak specific power level in the fuel for initial operation at 2,452 Mwt results in a maximum thermal output of 17.5 kw per ft of fuel rod. This value is comparable with other reactors of this size and therefore does not represent an extrapolation of technology. This comparison may be seen in the information presented in Table 1.3-1 which is described in detail in 1.3.

1.2.4 REACTOR BUILDING

The leak-tight structure required to contain the design base accident (DBA) defined in Section 14 is the reactor building.

The prestressed, post-tensioned concrete reactor building is of essentially the same design as the containment buildings for the Turkey Point Plant (Docket Nos. 50-250 and 251), the Oconee Station (Docket Nos. 50-269, 270, and 287) and the Palisades Station (Docket No. 50-255). The Rancho Seco engineered safeguards are similar to those for these plants and present neither uncommon solutions to engineering problems nor significant extrapolations in technology.

1.2.5 ENGINEERED SAFEGUARDS

Engineered safeguards are employed to reduce the potential radiation dose to the general public from the maximum hypothetical accident to less than the guideline values of 10 CFR 100. This is accomplished by automatic isolation of all reactor building fluid penetrations that are not required for limiting the consequences of the accident, thus eliminating potential leakage paths. Long-term potential releases following the accident are reduced by rapidly decreasing the reactor building pressure to near atmospheric within 24 hours, thereby reducing the driving potential for fission product escape.

In addition, the engineered safeguards will prevent core meltdown should the worst postulated loss-of-coolant accident occur. This is accomplished by injection core flooding systems of large capacity, parts of which are continuously operated for normal purposes and are therefore immediately available for emergency duty. These systems, coupled with the thermal, hydraulic, and blowdown characteristics of these reactors, will reliably prevent metal-water reactions, and any core melting (or disfiguration of the core into a geometry to prevent further cooling).

The engineered safeguards equipment of the nuclear unit, along with the normal operating modes, are as follows:

3 a. High pressure injection - normally on standby. Makeup pump which operates as part of the make-up and purification system can also provide high pressure injection.

b. Core flooding system.

c. Low-pressure injection - normally operates for shutdown cooling as part of the decay heat removal system.

d. Reactor building spray cooling and iodine removal system - normally shut down.

2 e. Reactor building emergency cooling system - normally on standby.

f. Reactor building isolation system - operates on test or accident signal.

2 g. Two sets of high efficiency and charcoal filters with 54,500 cfm total capacity each are provided for normal containment air cleanup and can be used post-accident for removal of fission products in the containment volume.

Table 1.2-1 lists equipment supplied for the engineered safeguards.

1.2.6 ELECTRICAL SYSTEMS AND EMERGENCY POWER

Rancho Seco Unit 1 will have the following sources of electric power.

a. The generator will feed its own auxiliaries. Upon a trip separating the switchyard from the transmission system, the turbine-generator is designed to stay in operation upon a load dump from full load down to auxiliary load.

b. Three, 230-kv transmission lines, one each to Hurley, Hedge and Tesla. 2

c. Two, 230-kv transmission lines to Bellota switchyard.

d. Two quick-starting emergency diesel-generator units connected to supply the nuclear service buses which are presently estimated to be rated at 2850 kw each. 2

e. Batteries for control power only.

The plant electrical systems will consist of multiple redundant buses and bus ties supplying all power, instrumentation, and controls. The engineered safeguards are supplied from separate nuclear service power buses, each of 3
which can be supplied from the 230-kv switchyard (start-up transformer) or the emergency diesel-generators.

The station electrical systems will be designed to provide reliable power for plant and personnel safety under all modes of operation and shutdown.

TABLE 1.2-1
ENGINEERED SAFEGUARDS

Function	Total Equipment Installed	
High Pressure Injection	3 pumps (1 normal makeup) 1 storage tank	2
Core Flooding System	2 tanks	
Low Pressure Injection	2 pumps (decay heat removal) 2 heat exchangers	2
Reactor Building Spray System	2 pumps 1 tank	
Reactor Building Emergency Cooling System	2 pumps 4 emergency cooling units	2
Reactor Building Filters	2 high efficiency } no credit taken 2 charcoal	2

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1.2.7 ONCE-THROUGH STEAM GENERATORS

The steam generators are of a design based on extensive research, development, and experimental work on boiling heat transfer performed by B&W over the past 11 years. Each generator is a vertical shell-and-tube, counterflow heat exchanger with reactor coolant on the tube side and steam on the shell side. Feedwater is pumped into the generator, heated to saturation by direct mixing with steam, converted to steam and superheated in a single pass through the generator. The basic design parameters, such as feedwater heating, boiling length, superheat length, and performance characteristics, have been confirmed by testing of a full-length 7-tube unit and a 37-tube unit. Tests are continuing to provide additional data in these design
2 areas for the 37-tube test unit. In addition, testing will continue with one, 19 tube full-length unit.

With the once-through design, natural circulation flow is adequate to remove full decay heat without the use of reactor coolant pumps. Thus, with total loss of pumps, the fuel will not reach departure from nucleate boiling.

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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		R
1	Hydraulic and Thermal Design Parameters	
	Rated Heat Output (core), Mwt	2,4
	Rated Heat Output (core), Btu/hr	8,3
	Maximum Overpower, %	14
	System Pressure (nominal), psia	2,2
	System Pressure (minimum steady state), psia	2,1
	Power Distribution Factors	
	Heat Generated in Fuel and Cladding, %	97.
	$F_{\Delta h}$ (nuclear)	1.8
	F_q (nuclear)	3.1
	Hot Channel Factors	3.2
	F_q (nuc. and mech.)	2.2
	DNB Ratio at Rated Conditions	1.
		1.
	Minimum DNB Ratio at Design Overpower	1.
	Coolant Flow	
	Total Flow Rate, lb/hr	13
	Effective Flow Rate for Heat Transfer, lb/hr	12
	Effective Flow Area for Heat Transfer, ft^2	47
	Average Velocity Along Fuel Rods, ft/sec	15
	Average Mass Velocity, lb/hr-ft^2	2.
	Coolant Temperature, F	
	Nominal Inlet	55
	Maximum Inlet due to Instrumentation Error and Deadband	55
	Average Rise in Vessel	47
	Average Rise in Core	49
	Average in Core	57
	Average in Vessel	57
	Nominal Outlet of Hot Channel	6
	Average Film Coefficient, Btu/hr-ft^2-F	5
	Average Film Temperature Difference, F	3
	Heat Transfer at 100% Power	
	Active Heat Transfer Surface Area, ft^2	4
	Average Heat Flux, Btu/hr-ft^2	1
	Maximum Heat Flux, Btu/hr-ft^2	5
	Average Thermal Output, kw/ft	5
	Maximum Thermal Output, kw/ft	1
	Maximum Clad Surface Temperature at Nominal Pressure, F	6
	Fuel Central Temperature, F	
	Maximum at 100% Power	
	Maximum at 114% Overpower	
	Thermal Output, kw/ft at Maximum Overpower	
2	Core Mechanical Design Parameters	
	Fuel Assemblies	
	Design	
	Rod Pitch, in.	

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

ncho 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
52	2,452	2,452	2,097
69 x 10^6	8,369 x 10^6	8,369 x 10^6	7,157 x 10^6
	14	14	12
00	2,200	2,200	2,250
50	2,150	2,150	2,220
3	97.3	97.3	97.4
5	1.85	1.85	1.75
5	3.15	3.15	3.12
4	3.24	3.24	3.25
7 (W-3)	2.27 (W-3)	2.27 (W-3)	1.85 (W-3)
0 (BAW-168)	1.60 (BAW-168)	1.60 (BAW-168)	
73 (W-3)	1.73 (W-3)	1.73 (W-3)	1.30 (W-3)
38 (BAW-168)	1.38 (BAW-168)	1.38 (BAW-168)	
1.3 x 10^6	131.3 x 10^6	131.3 x 10^6	100.6 x 10^6
0.9 x 10^6	120.9 x 10^6	120.9 x 10^6	91.5 x 10^6
.75	47.75	47.75	39.0
.70	15.70	15.70	13.9
53 x 10^6	2.53 x 10^6	2.53 x 10^6	2.35 x 10^6
5	555	555	546.5
7	557	557	550.5
.8	47.8	47.8	54
.3	49.3	49.3	59
79.7	579.7	579.7	577
78.9	578.9	578.9	574
44.4	644.4	644.4	647
,000	5,000	5,000	5,500
1	31	31	30
8,578	48,578	48,578	42,460
67,620	167,620	167,620	164,200
43,000	543,000	543,000	533,600
.4	5.4	5.4	5.3
7.5	17.5	17.5	17.3
54	654	654	657
,160	4,160	4,160	4,070
,400	4,400	4,400	4,270
19.9	19.9	19.9	19.4
CRA can	CRA can	CRA can	RCC canless
0.558	0.558	0.558	0.563

TABLE 1.3-1(Continued)

Item		Rancho Seco Nuclear Unit No. 1
	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-80% Ag
	Cladding Material	304 SS - cold worked
	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

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
5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag	5% Cd-15% In-80% Ag
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

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TABLE 1.3-1 (Continued)

Item		Rancho Seco Nuclear Unit No. 1
	Control Rod Assemblies	
	Material	5% Cd-15% In-80% Ag
	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 -3.0
	Moderator Pressure Coefficient, $\left(\frac{\Delta k}{k}\right)$/psi	-1.0×10^{-6} to +3.0
	Moderator Void Coefficient, $\left(\frac{\Delta k}{k}\right)$/% void	$+1.0 \times 10^{-4}$ to -3.0
	Doppler Coefficient, $\left(\frac{\Delta k}{k}\right)$/F	-1.1×10^{-5} to -1.7
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 A
	Steam Generator	
	Tube Side	ASME III, Class A
	Shell Side	ASME III, Class A
	Pressurizer	ASME III, Class A
	Pressurizer Relief Tank	ASME III, Class C
	Pressurizer Safety Valves	ASME III
	Reactor Coolant Piping	USASI B31.1
	Reactor Coolant Pump Casing	ASME III, Class A
6	Principal Design Parameters of the Reactor Vessel	
	Material	SA-533 Grade B, cl 18-8 stainless

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	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point Nos. 3 or 4
	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
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}
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 $+3.0 \times 10^{-6}$
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}
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	600.6
	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
	ASME III, Class A	ASME III, Class A	ASME III, Class A
	ASME III, Class A	ASME III, Class A	ASME III, Class A
	ASME III, Class A	ASME III, Class A	ASME III, Class C
	ASME III, Class A	ASME III, Class A	ASME III, Class A
	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
	ASME III, Class A	ASME III, Class A	
ad with 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 Nuclear Unit No. 1
	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-through with integral superheater.
	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 stage
	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, single speed
	Motor Rating (nominal), hp	9,000
9	Principal Design Parameters of the Reactor Coolant Piping	
	Material	Carbon steel clad with SS
	Hot Leg (ID), in.	36
	Cold Leg (ID), in.	28

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
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
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
A-C Induction, single speed	A-C Induction, single speed	A-C Induction, single speed
9,000	9,000	5,500
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 Nuclear Unit No.1
	Between Pump and Steam Generator (ID), in.	28
10	Reactor Building System Parameters	
	Type	Steel-lined, prestressed post-tensioned concret vertical cylinder with flat bottom and shallo domed roof.
	Design Parameters	
	Inside Diameter, ft	116
	Height, ft	206
	Free Volume, ft^3	1,900,000
	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 penetrations and continuous steel liner. Automatic isol tion where required.
	Gaseous Effluent Purge	Discharge vent above to of Reactor Building (~200 ft above grade
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
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
2,000,000	1,900,000	1,550,000
55	59	56
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	3-1/2
3	3-1/4	3
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.
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
3	3	3
54,000	54,000	80,000
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

TABLE 1.3-1(Continued)

Item		Rancho Seco Nuclear Unit No. 1
	(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 engineered safeguar equipment is capable being operated from c site emergency power.

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	Crystal River Plant Unit 3 or 4	Oconee Nuclear Station Unit 1, 2, or 3	Turkey Point No. 3 or 4	
				2
	2	2	2	
	Yes	No	No	2
ds of n-	3 Diesel All engineered safeguards equipment is capable of being operated from on-site emergency power.	2 Not comparable* 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.	2 for both Units Diesel 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	

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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. 2
Rancho Seco will use a spray additive injection in the reactor building spray for iodine removal because of differences in designs and site characteristics.

1.4 PRINCIPAL DESIGN CRITERIA

Presented in this section are criteria reflecting the design intent for Rancho Seco in consideration of and addressing to the 70 General Design Criteria for Nuclear Power Plant Construction Permits[1] proposed by the Atomic Energy Commission.

It is recognized and should be noted that at the time of preparation of this report, these 70 criteria have been submitted to the public for comments and have yet to be adopted in the form presented herein. No attempt has been made in this report to comment on the language or intent of the criteria.

The principal safety features that meet each criterion are summarized herein. In the discussion of each criterion, reference is made to sections of this report where more detailed information is presented.

1.4.1 CRITERION 1 - QUALITY STANDARDS (CATEGORY A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Answer:

a. Essential Systems and Components

The integrity of systems, structures, and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. These systems, structures, and components are:

(1) Fuel assemblies
(2) Reactor vessel internals
(3) Reactor coolant system
(4) Reactor instrumentation, controls and protection systems
(5) Engineered safeguards
(6) Radioactive materials handling systems
(7) Reactor building
(8) Electric power sources

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b. Codes and Standards

The Table 1.4-1 references applicable codes and standards for the nuclear unit as included in the PSAR.

c. Quality Assurance Operations

1 Appendix 1B includes a description of the Rancho Seco project quality assurance operations.

TABLE 1.4-1
APPLICABLE CODES AND STANDARDS

Item	Codes or Design Bases	Quality Assurance	Test Procedures	Acceptance Procedures
A-1.	3.1.2.4.2	*	3.3.3	13
2 A-2.	3.1.2.4.1 3.2.4.1	3.3.4	3.3.4	13
A-3.	4.1.4 4.1.5	4.1.4.4 4.3.1.1.2	4.4	13
A-4.	3.1.2.4.3 3.1.2.4.4 7.1.1.2	3.2.4.3.2 3.2.4.3.4 7.1.1.2	3.3.3 7.1.1.2 7.1.1.2.7 7.1.2.3.6 7.1.3.3 7.1.3.5 7.3.1.1.1	13
A-5.	9	9	9	13
A-6.	9	9	9	13
A-7.	5.1.2 App. 5-A, E, F, G	5.4 App. 5-H, I	5.5 App. 5-B, H, I	5.4, 5.5 App. 5-B, H, I
A-8.	8.1		8.4	13

2 *Fuel assembly production quality control and process procedures are being developed by B&W and vendor organizations.

1.4.2 CRITERION 2 - PERFORMANCE STANDARDS (CATEGORY A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Answer:

a. Essential Systems and Components

The integrity of systems, structures, and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. These systems, structures, and components are:

(1) Fuel assemblies
(2) Reactor vessel internals
(3) Reactor coolant system
(4) Reactor instrumentation, controls and protection systems
(5) Engineered safeguards
(6) Radioactive materials handling systems
(7) Reactor building
(8) Electric power sources

b. Performance Standards

These essential systems and components have been designed to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena, recorded for the vicinity of the site, with an appropriate margin to account for uncertainties in the historical data.

These natural phenomena are listed below with PSAR references.

Phenomenon	Reference Section
1. Earthquake	5.1.4, 5.1.5, App. 2D, App. 5A
2. Flood	2.4
3. Wind	5.1.2, 5.1.5.7
4. Snow and ice	5.1.2

3

1.4.3 CRITERION 3 - FIRE PROTECTION (CATEGORY A)

The reactor facility shall be designed (a) to minimize the probability of events such as fires and explosions and (b) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room and components of engineered safety features.

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Answer:

The reactor facility will be designed to minimize the probability of fires and explosions and to minimize the potential effects of such events to safety.

The potential magnitude of a fire in the control room will be limited by the following factors:

a. Materials used in control room construction will be non-flammable.

b. Control cables and switchboard wiring will be constructed of materials that have passed the flame tests described in Insulated Power Cable Engineers Association Publication S-61-502 and National Electrical Manufacturers Association Publication WC 5-1961.

c. Furniture in the control room will be of metal construction.

d. Combustible supplies such as log books, records, procedures, manuals, etc., will be limited to the amounts required for plant operation.

e. All areas of the control room will be readily accessible for fire extinguishing.

f. Adequate fire extinguishers will be provided.

g. The control room will be occupied at all times by a qualified person who has been trained in fire extinguishing techniques.

The only flammable materials inside the control room will be:

a. Paper in the form of logs, records, procedures, manuals, diagrams, etc.

b. Small amounts of combustible materials used in the manufacture of various electronic equipment.

The above list indicates that the flammable materials will be distributed to the extent that a fire would be unlikely to spread. Therefore, a fire, if started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the control room ventilation system.

Adequate fire extinguishers will also be provided in the reactor building. The safety of the nuclear unit would not be affected by fires due to these materials. The operation of the engineered safety features will not be impaired due to the effects of fires.

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1.4.4 CRITERION 4 - SHARING OF SYSTEMS (CATEGORY A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Answer:

Not applicable to the single facility Rancho Seco Nuclear Generating Station.

1.4.5 CRITERION 5 - RECORDS REQUIREMENTS (CATEGORY A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or be under his control throughout the life of the reactor.

Answer:

The District will maintain or have under its control records of the design, fabrication, and construction of essential components of the plant throughout the life of the reactor.

1.4.6 CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A)

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Answer:

The reactor is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady state operation including the transients given in the criterion. Fuel clad integrity is ensured under all modes of anticipated operation by avoiding clad overstressing and overheating. The evaluation of clad stresses includes the effects of internal and external pressures, temperature gradients and changes, clad-fuel interactions, vibrations, and earthquake effects. The free-standing clad design prevents collapse at the end volume region of the fuel rod and provides sufficient radial and end void volume to accommodate clad-fuel interactions and internal gas pressures (3.2.4.2).

Clad overheating is prevented by satisfying the following core thermal and hydraulic criteria (3.2.3.1.1):

a. At the design overpower no fuel melting will occur.

b. A 99 percent confidence exists that at least 99.5 percent of the fuel rods in the core will be in no jeopardy of experiencing a DNB during continuous operation at the design overpower of 114 percent.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine power mismatch. The reactor is operated at a constant average coolant temperature above 15 percent power and has a negative power coefficient to dampen the effects of power transients. The reactor control system will maintain the reactor operating parameters within preset limits, and the reactor protection system will shut down the reactor if normal operating limits are exceeded by preset amounts (Sections 7.1 and 14).

Reactor decay heat will be removed through the steam generators until the reactor coolant system is cooled to 250 F. Steam generated by decay heat will supply the steam-driven feedwater pump turbine and can also be vented to atmosphere and/or bypassed to the condenser. The steam generators are supplied feedwater from either the main steam-driven feedwater pumps, which can be operated at a reduced flow rate for decay heat removal, or from either a steam- or motor-driven emergency feed pump, each of which is sized at 5 percent of full feedwater flow.

The main feedwater pumps supply water contained in the feedwater train and the condensate storage tank to the steam generators. The emergency feed pump takes suction from the condensate train or the condensate storage tank. These sources provide at least 250,000 gallons of water storage which is sufficient for decay heat removal for about 24 hours after reactor shutdown with the primary heat sink (condenser) isolated. The condenser is normally available so that water inventory is not depleted. (Section 10)

The reactor coolant pumps are provided with sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated. (14.1.2.6 and Figure 9.5-2)

1.4.7 CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (CATEGORY B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

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Answer:

Power oscillations resulting from variation of coolant temperature are minimized by constant average coolant temperature above 15 percent power. Power oscillations from spatial xenon effects are minimized by the large negative power coefficient. Reactor trip prevents fuel clad damage resulting from DNB.

The ability of the reactor control and protection system to control the oscillations resulting from variation of coolant temperature within the control system dead band and from spatial xenon oscillations has been analyzed. Variations in average coolant temperature provide negative feedback and enhance reactor stability during that portion of core life in which the moderator temperature coefficient is negative. When this coefficient is positive, rod motion will compensate for the positive feedback. The maximum power change rate resulting from temperature oscillations within the control system dead band has been calculated to be less than 1 percent/minute. Since the unit has been designed to follow ramp load changes of 10 percent/minute, this is well within the capability of the control system.

Control flexibility with respect to xenon transients is provided by the combination of control rods and in-core instrumentation. Axial, radial, or azimuthal neutron flux changes will be detected by the in-core instrumentation. Individual or groups of control rods can be positioned to suppress and/or correct flux changes (3.2.2.2.3). 2

1.4.8 CRITERION 8 - OVERALL POWER COEFFICIENT (CATEGORY B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Answer:

The overall power coefficient is negative in the operating range (Table 3.2-9).

1.4.9 CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (CATEGORY A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Answer:

The reactor coolant pressure boundary meets this criterion by the following:

a. Material selection, design, fabrication, inspection, testing, and certification in keeping with the ASME (Section III) and USASI (B31.1) Codes.

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b. Quality manufacture including weld qualification test plates, permanent identification of materials, welder qualification tests, and extensive production nondestructive testing.

c. Service life based on instability of the material, effects of mechanical shock or vibratory loading, and radiation effects with special consideration to increase in the brittle fracture transition temperature due to neutron irradiation (Section 4.1).

1.4.10 CRITERION 10 - CONTAINMENT (CATEGORY A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Answer:

The containment system, which consists of the reactor building and the engineered safeguards systems, is designed to provide protection for the public from the consequences of an unlikely event of loss-of-coolant accident, which is based on a sudden break in the reactor coolant system.

The reactor building is designed to safely sustain all internal and external loading conditions that may reasonably be expected to occur during the life of the station, including an accident causing an internal pressure of 57 psig
2 with a coincident temperature of 286 F combined with the maximum hypothetical earthquake. Due consideration has been given to all site factors and local environment as they relate to the public's health and safety.

See Sections 5 and 14 for details.

1.4.11 CRITERION 11 - CONTROL ROOM (CATEGORY B)

The facility shall be provided with a control room from which actions to maintain safe-operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Answer:

Considering the detailed station design provisions to ensure continuous
2 control room access, it is unlikely that the necessity could arise for
evacuation of the control room. Section 7.4.8.2 enumerates the reasons

for considering the control room permanently available. Nevertheless, provisions will be made to maintain the reactor in a safe-hot shutdown condition if access to the control room is lost.

Should the control room become uninhabitable, the reactor would be manually tripped, and neutron level and control rod position would be verified before evacuation takes place. Equipment will be provided external to the control room for the following functions:

a. Operation of the main and auxiliary feed pumps with controls available at the pump (main feed pumps continue to operate with offsite power available; otherwise, auxiliary feed pumps start automatically, upon loss of main feed pumps).

b. Operation of the feedwater control valves by use of handjacks (the feedwater control valves will automatically control the feedwater to each steam generator to maintain the required level for shutdown operation, and the handjack would be required only on malfunction of the automatic feedwater control system). 2

c. Operation of the makeup pump to maintain pressurizer level. (The pump would be lined up to the borated water storage tank for this operation.)

d. Operation of the makeup and letdown valves by use of handjacks at the valves. (Manual operation of the makeup valve would be required only if automatic pressurizer level control fails.)

e. Manual operation of main turbine steam bypass valve and/or main steam relief valves. (Handjacks required only if automatic functions fail.)

f. Observing reactor coolant pressure and temperature, pressurizer level, main steam pressure, steam generator level, and feedwater pressure.

g. Station communications equipment will be provided for the Senior Control Operator at a location such as one of the switchgear rooms located at the mezzanine level, below the control room. As a minimum, the capability would be provided for station-wide paging and telephone communications with such stations as the feedwater control station, coolant makeup and boration equipment areas, and other key plant locations. 3

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Should the reactor be tripped and the control room evacuated, reactor decay heat would be removed via the steam generators, with steam exhausting through the main turbine steam bypass valve and/or atmospheric vent valve. Either the main or the auxiliary feed pumps would continue to supply feedwater to the steam generators.

Under these conditions it is expected that a balance will be maintained between heat removal and decay heat generation, the reactor coolant system maintained at normal average temperature, and no significant makeup should be required for several hours. Any makeup can be supplied by operation of the makeup pump taking suction from the borated water tank and discharging through the normal makeup system lineup. These makeup operations could be conducted locally, and the controls and instrumentation summarized in paragraphs a. through g. above are adequate for maintaining the plant in a
3 safe - hot condition during the period of control room inaccessibility.

Inasmuch as the plant can be maintained in a safe hot shutdown condition from outside the control room until access to the control room is regained, the need for taking the plant to a cold shutdown condition from outside the control room is not anticipated. However, the ability to bring the plant to a cold shutdown condition from outside the control room exists with the present plant design. Through local controls, local or remote readout, and temporary connection of electrical equipment after restoring offsite power, all necessary functions could be performed outside the control room, and thus with proper manpower and coordination the plant could be cooled down over an extended period of time. Such an action would include the formulation at that time of an emergency plan and procedure based on an assessment of the situation.

1.4.12 CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (CATEGORY B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Answer:

1.4.12.1 Protection Systems

The protection systems, which consist of the reactor protection system and the safeguards actuation system, perform the most important control and safety functions. The protection systems extend from the sensing instruments to the final actuating devices, such as trip circuit breakers and pump or valve motor contactors.

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The reactor protection system monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage caused by departure from nucleate boiling (DNB), and to protect against reactor coolant system damage caused by high system pressure. The safeguards actuation system monitors parameters to detect failure of the reactor coolant system, and initiates reactor building isolation and engineered safeguards operation to contain radio-active fission products in the reactor building.

1.4.12.2 Regulating Systems

The reactivity control system monitors power output and regulates output by means of the control rods and the soluble boron.

The integrated control system maintains constant average reactor coolant temperature and constant steam pressure at the turbine during steady state and transient operation between 15 and 100 percent full power.

1.4.12.3 Instrumentation Systems

Instrumentation systems include the nuclear instrumentation system, which monitors the neutron flux level from source to 125 percent of rated power; the non-nuclear process instrumentation which measures temperatures, pressures, flows and levels in the reactor coolant system, steam system, and reactor auxiliary systems.

Detailed information is in Section 7.

1.4.13 CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (CATEGORY B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Answer:

This criterion is met by reactivity control means and control room display. Reactivity control is by movable control rods and by chemical neutron absorber (boric acid) dissolved in the reactor coolant. The position of each control rod will be displayed in the control room. The reactivity status of soluble boron will be indicated by the position of the control rods. Periodic boron concentration in the reactor coolant is determined by the sampling system and is reported to the reactor operator (Section 7.2).

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1.4.14 CRITERION 14 - CORE PROTECTION SYSTEMS (CATEGORY B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Answer:

The reactor design meets this criterion by reactor trip provisions and engineered safeguards. The reactor protection system is designed to limit reactor power which might result from unexpected reactivity changes and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits. In a loss-of-coolant accident, the engineered safeguards protection system automatically actuates the high pressure and low pressure injection systems. The core flooding tanks are self-actuating. Certain long-term operations in the emergency core cooling systems which do not require immediate actuation are performed manually by the operator, such as remote switching of the low pressure injection pumps to the recirculation mode and sampling of the recirculated coolant. These operations are performed manually to improve the overall reliability of the system (Section 7.1).

1.4.15 CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (CATEGORY B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Answer:

The reactor protection system senses abnormal reactor power level, reactor outlet temperature, reactor coolant pressure, and reactor startup rate, and trips the reactor for each abnormal condition. The engineered safeguards protection system senses reactor coolant pressure which initiates core coolant injection, and senses reactor building pressure which initiates
3 coolant injection and emergency building cooling. The engineered safeguards protection system includes provisions for the removal of radioactive iodine during the period following the design basis accident. The isolation system also is actuated as part of the safeguards system and protects against the release of radioactive iodine.

1.4.16 CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (CATEGORY B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Answer:

Instrumentation is provided to meet this criterion by measuring fluid volume changes (pressurizer and reactor building sumps) and radioactivity levels in the reactor building. An increase in net makeup to the combined reactor coolant system and connected high pressure injection and purification system will indicate leakage (Section 4.2.7).

1.4.17 CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (CATEGORY B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Answer:

The fixed radiation monitoring system will be designed to indicate and alarm high radiation levels throughout the plant. Visual presentation of readings, recorded presentation, and an audible/visible alarm at both the detector location and the control room will be provided. All instrumentation for the radiation monitoring system will obtain its voltage supply from the 120-volt essential ac service buses and each detector will have a loss-of-power alarm. The normal high radiation alarm setpoint will be 10 percent above the normal operational reading of the detector. A maximum alarm point will be set to correspond to the MPC values specified in 10 CFR 20. The maximum alarm point set at 10 CFR 20 values could be either an actual value or a calculated number corresponding to 10 CFR 20 limits. A description of the detectors appears in 11.2.2.2.

Determination of detector ranges will depend upon the normal background at the detector locations and the calculated levels for abnormal conditions. Radioactive test sources will be available to allow overall system performance to be verified at regular intervals.

The radiation monitoring system shall be checked and calibrated at an appropriate frequency. When any portion of the radiation monitoring system requires maintenance, that unit shall be completely checked and calibrated immediately after completion of maintenance. Radiation monitoring of plant effluent will include alarms and indications designed to provide early warning of possible equipment malfunctions or potential biological hazards. Plant gaseous effluent will be monitored to ensure that prescribed limits of radiation release are not exceeded. There will be no direct release of liquid effluent to the environment. The release of gaseous effluent will be controlled within the limits of 10 CFR 20.

All gaseous radioactivity will be discharged to the atmosphere through the plant vent. The source of this activity is the gaseous waste disposal system. After a delay in the waste gas decay tanks, the gas will be analyzed prior to release. Flow in the discharge line from the waste disposal system and flow in the plant vent will be monitored to ensure that the system is operating correctly and that the releases are within the limits of 10 CFR 20.

A monitor in the gaseous waste discharge line to the plant vent will be equipped with an indicator and an alarm to annunciate a high activity level.

A high-level interlock will stop the discharge of gaseous effluents from the waste disposal system and direct the effluent to the waste gas decay tank.

The condenser air ejector discharge will be monitored for gaseous activity. Should leakage of reactor coolant to the steam occur and result in high activity levels, the monitor on the air ejector will initiate an alarm in the control room.

All liquid waste will be collected, stored, and analyzed for radioactive concentration prior to removal from the site by an AEC-licensed disposal firm. Activity concentration will be determined by sampling the stored liquid waste prior to disposal.

The component cooling water system that removes heat from potentially radioactive sources is monitored to detect accidental releases.

3 It is planned to monitor the activity in the reactor coolant letdown flow. Gross fuel failure will be detected by regular laboratory analysis of reactor coolant samples.

These radiation monitors are commercially-available equipment. The required characteristics will be established during detailed station design. The minimum sensitivity of detectors, when combined with appropriate dilution factors, will ensure safe limits of release.

The environmental monitoring program is designed to establish environmental radiation levels and detect any changes which may occur. Sampling points will be chosen, both on-site and off-site, generally in prevailing wind directions and near population centers.

The samples collected will include the following:

a. Water (streams, wells, lakes and rain)

b. Airborne particulate material (suspended and settlement or fallout)

c. Soil and silt

d. Vegetation

e. Milk

f. Fish and animals

The gross alpha and gross beta activity of the samples will generally be measured and specific radionuclides will be identified when appropriate. Special analyses, such as determination of Sr^{90} in milk, will be performed by an outside laboratory.

Since the program has not been completely developed, the frequency of observations and sampling have not been finally determined, but it is planned to determine integrated radiation doses at monthly intervals. Water and airborne particulate samples will be collected weekly and their radioactivity determined. Other samples will be collected and analyzed on a frequency to be determined.

1.4.18 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (CATEGORY B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Answer:

Heat generated in waste storage is small enough that it can all be considered to radiate away. Thus there are no heat removal systems that can fail. Continuous monitoring of spent fuel pool water temperature for failure of the cooling system is not required because water temperature would increase very slowly due to such a failure.

Monitoring and alarm instrumentation is provided for high radiation levels in the auxiliary building, due to waste storage failure, and in fuel handling areas.

1.4.19 CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (CATEGORY B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Answer:

The protection systems design meets this criterion by specific location, ample design capacity, component redundancy, and in-service testing. The major design criteria stated below have been applied to the design of the instrumentation.

a. No single component failure shall prevent the protection systems from fulfilling their protective function when action is required.

b. No single component failure shall initiate unnecessary protection system action, provided implementation does not conflict with the criterion above.

Manual testing facilities are built into the protection systems to provide for:

a. Pre-operational testing to give assurance that the protection systems can fulfill their required functions.

b. On-line testing to provide operability and to demonstrate reliability (7.1.1).

1.4.20 CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (CATEGORY B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Answer:

Reactor protection is by four channels with 2/4 coincidence, and engineered safeguards is by three channels with 2/3 coincidence. All protection system functions are implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels. Redundant protection channels and their associated elements are electrically independent and packaged to provide physical separation. The reactor protection system initiates a trip of the channel involved when modules, equipment, or subassemblies are removed (7.1.1).

1.4.21 CRITERION 21 - SINGLE FAILURE DEFINITION (CATEGORY B)

Multiple failures resulting from a single event shall be treated as a single failure.

Answer:

The protection systems meet this criterion in that the instrumentation is designed so that a single event cannot result in multiple failures that would prevent the required protective action (7.1.2.3.7 and 7.1.3).

1.4.22 CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (CATEGORY B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Answer:

2 The protection systems instrument strings are electrically and physically independent from each other and isolated from control instrumentation systems. Shared instrumentation for protection and control functions satisfies the single failure criteria by the employment of isolation techniques to the multiple outputs of various instrument strings (7.1.1.2.3 and 7.3.1.2.2).

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1.4.23 CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (CATEGORY B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Answer:

The protection systems are designed to extreme ambient conditions and to redundancy. Protection systems instrumentation will operate from 40 - 140 F and sustain (except for neutron detectors) the loss-of-coolant conditions of 59 psig and 286 F and still be operable. Out-of-core neutron detectors are designed for 175 F and 150 psig. Protection system instrumentation will be subject to environmental (qualification) test as required by the proposed IEEE Standard for Nuclear Power Plant Protection Systems. Protective equipment outside of the reactor building, control room, and relay room is designed for continuous operation in an ambient temperature of 120 F and 90 percent relative humidity (7.1.1).

1.4.24 CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (CATEGORY B)

In the event of loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Answer:

In the event of loss of all off-site power, a drop in load to auxiliary load on the nuclear power system is accomplished by the step load reduction detailed in Section 10.1.1. In addition, emergency power sources provide a dependable supply of power for the critical services in the unlikely event of simultaneous of loss of normal and standby power (Section 8.2.3).

See Section 8 for details.

1.4.25 CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (CATEGORY B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Answer:

The protection systems will meet the testing criterion. The test circuits will take advantage of the systems' redundance, independence, and coincidence features which make it possible to initiate trip signals manually in any part of one protective channel without affecting the other channels. This test feature will allow the operator to interrogate the systems from the input of any bistable up to the final actuating device at any time during reactor operation without disconnecting permanently-installed equipment.

The power breakers in the reactor trip circuit may also be manually tested during operation. The only limitation is that not more than one power supply may be interrupted at a time without causing a reactor trip.

See Section 7 for details.

1.4.26 CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (CATEGORY B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Answer:

The reactor protection system will trip the reactor on loss of power. The engineered safeguards protection system is supplied with multiple sources of electric power for control and valve action. A total loss of electrical power to the engineered safeguards protections system will cause its instrumentation to assume a tripped position except for the circuit which actuates the building spray valves and with the exception of the final control relays. These relays require power to trip. However, since the engineered safeguards equipment also requires power to operate, this relay need not assume the tripped position upon a total loss of power. The multiple power supplies for the control relays are also battery-backed and therefore more reliable than the power supply for the equipment.

The system is designed for continuous operation under adverse environments. The reactor protection system instrumentation within the reactor building is designed for continuous operation in an environment of 140 F, 60 psig, and 100 percent relative humidity. The neutron detectors are designed for continuous operation in an environment of 175 F, 90 percent relative humidity, and 150 psig. Engineered safeguards equipment and vital instrumentation inside the reactor building are designed for conditions (59 psig, 286 F, and 100% RH) which are in excess of the requirements of the loss-of-coolant accident (7.1.1 and 7.1.2).

Redundant instrument channels are provided for the reactor protection and safeguards actuation systems. Loss of power to each individual reactor protection channel will trip that individual channel. Loss of all instrument power will trip the reactor protection system thereby releasing the control rods and will activate the safeguards actuation system controls (with the exception of the reactor building spray valves).

Manual reactor trip is designed so that failure of the automatic reactor trip circuitry will not prohibit or negate the manual trip. The same is true with respect to manual operation of the engineered safeguards equipment.

1.4.27 CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (CATEGORY A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

Answer:

This criterion is met by control rods and soluble boron addition to, or removal from, the reactor coolant (7.2.2.1).

1.4.28 CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (CATEGORY A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Answer:

One highly redundant reactivity control system consisting of 69 control rods is provided to rapidly make the core subcritical upon a trip signal and to protect the core from damage due to the effects of any operating transient. The soluble absorber reactivity control system can make the reactor subcritical even from ultimate power. However, its action is slow, and the ability to protect the core from damage which might result from rapid load changes such as an ultimate load turbine trip is not a design criteria for this system. The high degree of redundancy in the control rod system is considered sufficient to meet the intent of this criterion (3.2.2.1).

1.4.29 CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (CATEGORY A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Answer:

The reactor design meets this criterion under normal operating conditions, and under the accident conditions set forth in 14.1. The reactor is designed with the capability of providing a shutdown margin of at least 1% $\Delta k/k$ with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero power condition. The minimum hot shutdown margin of 2.1% $\Delta k/k$ occurs at the end of life (3.2.2.1).

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1.4.30 CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (CATEGORY B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Answer:

The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and accident conditions as set forth in Section 14. Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the maximum allowable reactor cooldown rate of 100 F/hour. Thus subcriticality is assured during cooldown with the most reactive control rod totally unavailable (3.2.2.1).

1.4.31 CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (CATEGORY B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Answer:

The reactor design meets this criterion. A reactor trip will protect against continuous withdrawal of any one rod.

1.4.32 CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (CATEGORY A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Answer:

The reactor design meets this criterion by safety features which limit the maximum reactivity insertion rate. These include rod group withdrawal interlocks, soluble boron concentration reduction interlock, maximum rate of dilution water addition, and dilution time cutoff (14.1.2.4). In addition, the rod drives and their controls have an inherent feature to limit overspeed in the event of malfunctions (3.2.4.3). Ejection of the maximum worth control rod will not lead to coolant boundary rupture or internals damage which would interfere with emergency core cooling (14.2.2.2).

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1.4.33 CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (CATEGORY A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Answer:

The reactor design meets this criterion. There are no credible mechanisms whereby damaging energy releases are liberated to the reactor coolant. Ejection of the maximum worth control rod will not lead to coolant boundary rupture (14.2.2.2).

1.4.34 CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (CATEGORY A) |1

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type of failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Answer:

The reactor coolant pressure boundary design meets this criterion by the following:

a. Selection of reactor vessel plate material opposite the core to a specified Charpy-V-notch test result of 30 ft-lb or greater at a corresponding NDTT of 10 F or less.

b. Determination of the fatigue usage factor resulting from expected states and transient loading during detailed design and stress analysis.

c. Quality control procedures including permanent identification of materials and nondestructive testing for flaw identification.

d. Operating restrictions to prevent failure resulting from increase in brittle fracture transition temperature due to neutron irradiation, including a material irradiation surveillance program (4.1.4).

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1.4.35 CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (CATEGORY A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120 F above the nil-ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60 F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Answer:

The reactor vessel is the only reactor coolant system component exposed to a significant level of neutron irradiation, and is therefore the only component subject to material irradiation damage. The end-of-unit-life NDTT value of the vessel opposite the core will be not more than 260 F. Unit operating procedures will be established to limit the operating pressure to 20 percent of the design pressure when the reactor coolant system temperature is below NDTT +60 F throughout unit life (4.1.4).

1.4.36 CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (CATEGORY A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

Answer:

The reactor coolant pressure boundary components meet this criterion in the sense that space is provided for nondestructive testing methods during plant shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 will be established (4.4.3).

1.4.37 CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (CATEGORY A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

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Answer:

The reactor design meets this criterion. The emergency core cooling systems can protect the reactor for any size leak up to and including the circumferential rupture of the largest reactor coolant pipe (14.2.2.3).

1.4.38 CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (CATEGORY A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Answer:

All engineered safeguards systems are designed so that a single failure of an active component will not prevent operation of that system or reduce the capacity below that required to maintain a safe condition. Two independent reactor building cooling systems, each having full heat removal capacity, are used to prevent overpressurization.

The high pressure injection, core flooding and low pressure injection systems have redundance of equipment to ensure availability of capacity.

Some engineered safeguards systems have both a normal and an emergency function, thereby providing nearly continuous testing of operability. During normal operation, the make-up system supplies reactor coolant make-up. One specific pump will be used continuously for make-up so that the major wear is only on one pump. The two remaining high pressure injection pumps will be used as stand-bys for the make-up pump when it is down for maintenance. 2

Engineered safeguards equipment piping, which is not fully protected against missile damage, utilizes dual lines to preclude loss of the protective function as a result of the secondary failure.

Testing and inspection of the engineered safeguards systems is covered in detail for each system in the criteria where such information is specifically asked for.

See Section 6 for details.

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1.4.39 CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (CATEGORY A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Answer:

The design of the electrical system for the plant will provide the following sources of electric power:

a. Power to the plant from its own generator

b. 230-kv transmission system

c. Automatic, fast start-up diesel engine generators

d. Batteries for control power only

The onsite power system is capable of providing the necessary power for the safety features, assuming the failure of a single active component. The offsite power system also has this capability, except in the extreme case of a regional blackout.

See Section 8 for details.

1.4.40 CRITERION 40 - MISSILE PROTECTION (CATEGORY A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Answer:

Protective walls and slabs, local missile shielding or restraining devices will be provided to protect the reactor building liner plate and engineered safeguard systems within the reactor building against damage from missiles generated by equipment failures. The concrete surrounding the reactor coolant system serves as radiation shielding and is an effective barrier against missiles. Local missile barriers will be provided for control rod drive mechanisms.

For those parts of the safeguards systems susceptible to missile damage, redundant equipment is provided to assure required operation.

The portion of the high pressure and low pressure injection systems within the reactor building consists of two injection lines. The high pressure

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injection lines are connected to the reactor coolant inlet piping on opposite sides of the reactor vessel.

For most of the routing, these lines will be outside the reactor and steam genera. r shielding and hence protected from missiles originating within these areas. The portions of the injection lines located between the primary reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of missiles in this area.

The reactor building spray system spray headers are located outside and above the reactor and steam generator concrete shield. During operation, a movable shield also provides missile protection for the area immediately above the reactor vessel. The spray headers are therefore protected from missiles originating within the shield.

With regard to the reactor building emergency cooling units, all equipment, piping, valves, and instrumentation in the reactor building are located to minimize the possibility of missile damage. The emergency cooling units and associated piping are located outside the secondary concrete shielding.

1.4.41 CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (CATEGORY A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single act e component.

Answer:

All engineered safeguards systems are designed so that a single failure of an active component will not prevent ope ation of that system or reduce the capacity below that required to maintain a safe condition.

The core flooding tanks contain check valves which operate to permit flow of emergency coolant from the tanks to the reactor vessel. These valves are self-actuating and need no external signal or external-supplied energy to make them operate. Accordingly, it is not considered credible that they would fail to operate when needed.

1.4.42 CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (CATEGORY A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

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Answer:

The engineered safety features are designed to function in the unlikely event of loss-of-coolant accident with no impairment of capability due to the effects of the accident.

See Section 6 for details.

1.4.43 CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (CATEGORY A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Answer:

The engineered safety features are designed to meet this criterion. The water injected to ensure core cooling is sufficiently borated to ensure core subcriticality. Nonessential sources of water inside the reactor building are automatically isolated to prevent dilution of the borated coolant. Essential sources of postaccident cooling water are monitored to detect leakage which may lead to dilution of boron content. An analysis has been made which demonstrates that the injection of cold water on the hot reactor coolant system surfaces will not lead to further failure. The design of the equipment and its actuating system insures that water injection will occur in a sufficiently short time period to preclude significant metal-water reactions and subsequent energy releases to the reactor building (14.2).

1.4.44 CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (CATEGORY A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and

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is not lost during the entire period this function is required following the accident.

Answer:

Emergency core cooling is provided by pumped injection, and the core flooding tanks. This equipment prevents clad melting for the entire spectrum of reactor coolant system failures ranging from the smallest leak to the complete severance of the largest reactor coolant pipe. Pumped injection is subdivided in such a way that there are two separate and independent strings, each including both high pressure and low pressure coolant injection and each capable of providing 100% of the necessary core injection with the core flooding tanks. The core flooding tanks are passive components which are needed for only a short period of time after the accident, thereby assuring 100% availability when needed. 2

1.4.45 CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Answer:

Reactor vessel internals and water injection nozzles can be inspected by remote visual means during refueling. Other critical parts can be inspected periodically.

1.4.46 CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (CATEGORY A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability, and required functional performance.

Answer:

The emergency core cooling systems design meets this criterion by periodic tests performed on components not normally in service (6.1.4). 2

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1.4.47 CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Answer:

The high pressure and low pressure injection systems are included as part of normal service systems. Consequently, the active components can be tested periodically for delivery capability. The core flooding system delivery capability can be tested during shutdown or refueling. In addition, all valves will be periodically cycled to ensure operability. With these provisions, the delivery capability of the emergency core cooling systems can be periodically demonstrated (6.1.4).

1.4.48 CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Answer:

The operational sequence that would bring the emergency core cooling systems into action, including transfer to alternate power sources, can be tested in parts.

See Sections 6 and 7 for details.

1.4.49 CRITERION 49 - CONTAINMENT DESIGN BASIS (CATEGORY A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Answer:

The Reactor Building, including access openings and penetrations, has a design pressure of 59 psig at 286°F. The greatest transient peak pressure, associated with a hypothetical rupture of the piping in the reactor coolant system and the effects of a credible metal-water reaction, will not exceed these values.

The reactor building and engineered safeguards systems have been evaluated for various combinations of credible energy releases. The analysis accounts for system energy, decay heat, metal-water reactions, and the burning of the resultant hydrogen. The cooling capacity of either reactor building cooling system is adequate to prevent over-pressurization of the structure, and to return the reactor building to near atmospheric pressure within 24 hours.

The use of injection systems for core flooding will limit the reactor building pressure to less than the design pressure. If a metal-water reaction is uninhibited by the active quenching systems the resultant peak reactor building pressure is less than the design pressure.

The high pressure injection and low pressure injection systems have redundancy of equipment to ensure availability of capacity.

Electric motors, valves, and damper operators, which must function within the reactor building during accident conditions, will operate in a steam-air atmosphere at 286 F and 59 psig.

1.4.50 CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (CATEGORY A)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30° F above nil-ductility transition (NDT) temperature.

Answer:

The reactor building steel liner plate will be maintained at temperatures above the 30 F nil-ductility temperature plus 30 F, or above 60 F. The average normal operating temperature for the atmosphere inside the reactor building will be 120 F. The liner plate is completely enclosed by thick concrete walls (slab and roof of the reactor building) and will thus not be subject to sudden variations due to changes in external temperatures. In addition, the bottom liner plate is protected by a minimum thickness of 6 inches of cover concrete. Nil-ductility is not a consideration at the higher temperatures associated with accidental conditions. |4 |3

See Section 5 for details.

1.4.51 CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (CATEGORY A)

If part of the reactor coolant pressure boundary is outside the containment appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Answer:

No lines which contain high-temperature, high-pressure reactor coolant penetrate the reactor building except the sampling lines. These small sampling lines are normally isolated by two valves in series. Therefore, it is only during a sampling operation that a line failure would require operator action to prevent escape of coolant external to the reactor building. This is a procedure that the operator would normally perform.

The make-up and purification system diverts a small amount of reactor coolant outside the reactor building. This high pressure and high temperature coolant is cooled before it leaves the reactor building. Lines serving this function contain isolation valves that can be closed to prevent uncontrolled release of reactor coolant if a line fails external to the reactor building. The letdown coolers are supplied with water from the component cooling system. Any leakage of reactor coolant through the letdown coolers will be into this system rather than to the environment. The component cooling system is monitored to detect leakage of reactor coolant.

Leakage of contaminated coolant from engineered safeguards equipment located external to the reactor building has been evaluated, and the resultant environmental consequences are well below 10 CFR 100 limits at the site boundary, and have been included in the total accidental dose calculations.

See Section 14 for details.

1.4.52 CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (CATEGORY A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Answer:

Emergency building atmosphere cooling is provided to limit post-accident building pressures to design values and is accomplished by reactor building air recirculation and cooling units backed up by the reactor building spray system.

2 Emergency reactor building atmosphere cooling is performed with four units. Each of the four units contains a cooling coil and a direct-driven fan. For emergency cooling, all units will operate under post-accident conditions with the heat being rejected to the nuclear service

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cooling water system. Each of these units can remove 60×10^6 Btu/hr under peak reactor building temperature conditions.

Simultaneously with the air recirculation cooling, reactor building sprays are supplied with water by two pumps which take suction on the borated water storage tank until this coolant source is exhausted or in emergency from the top 12 feet of the spent fuel pit which contains borated water. After the supply from the borated water storage tank is exhausted, the spray pumps take suction from the reactor building sump recirculation line. This continued spraying serves to reduce the reactor building atmosphere to the temperature of the reactor building sump.

The function of cooling the reactor building atmosphere is fulfilled by either of the two methods described above, and redundancy and separation of equipment within each system will provide for protection of building integrity. For details see Section 9. 2

1.4.53 CRITERION 53 - CONTAINMENT ISOLATION VALVES (CATEGORY A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Answer:

The general design basis governing isolation valve requirements is: Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building, and various types of isolation valves.

1.4.54 CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (CATEGORY A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Answer:

The purpose of the initial integrated leakage rate test is to measure the percentage by weight of air which can leak out of the reactor building per day at the design pressure. The specified design leakage of 0.10 percent or less per day will be measured by using the reference method. During the test, air pressure, water vapor pressure, and air temperature will be measured by using high accuracy instruments. The reactor building ventilation system will be used continuously throughout the test to achieve complete

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air mixing and control of air temperature. Duration of the test will be a minimum of 24 hours. A similar test will be performed at a reduced pressure and all important parameters recorded.

1.4.55 CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (CATEGORY A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design during plant lifetime.

Answer:

Leakage rate testing will be done periodically at the reduced pressure referred to in Criterion 54. The acceptable leakage rate at this pressure will be determined from results of the initial leakage rate test. Frequency of the tests will be dependent upon the available margin between test values of leakage and the acceptable limit.

1.4.56 CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (CATEGORY A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

Answer:

Monitoring of the penetration cannisters will be accomplished by using a modified reference method and a quality control chart technique.

See Section 5.8 for details.

1.4.57 CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATIONS VALVES (CATEGORY A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Answer:

Each isolation valve will be tested periodically during normal operation or during shutdown conditions to ensure its operability when needed and to ensure that valve leakage does not exceed acceptable limits.

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1.4.58 CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (CATEGORY A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Answer:

The containment pressure-reducing systems are the reactor building spray system and the reactor building cooling units.

Performance testing of all active components of the reactor building spray system will be accomplished as described in Criterion 59. During these tests the equipment will be visually inspected for leaks. Valves and pumps will be operated and inspected after any maintenance to ensure proper operation.

The equipment, piping, valves and instrumentation of the reactor building cooling units are arranged so that they can be visually inspected. The emergency cooling units and associated piping are located outside the secondary concrete shield around the reactor coolant system loops. Personnel can enter the reactor building periodically to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial start up.

1.4.59 CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (CATEGORY A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Answer:

The active components of the reactor building spray system will be tested on a regular schedule as follows:

Component	Test
Reactor Building Spray Pumps	These pumps will be tested singly by opening the valves in the test line (reactor building spray valves closed) and the borated water storage tank outlet valves. Each pump in turn will be started by plant operator action and checked for flow establishment to each of the spray headers. Flow will also be tested through each of the borated water storage tank outlet valves by operating these valves.

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Component	Test
Borated Water Storage Tank Outlet Valves	These valves will be tested in performing the pump test listed above.
Reactor Building Spray Injection Valves	With the pumps shut down and the borated water storage tank outlet valves closed, these valves will each be opened and closed by operator action.
Reactor Building Spray Nozzles	Under the conditions specified for the previous test and with the reactor building spray valves alternately open, smoke will be blown through the test connections.
2 Reactor Building Emergency Cooling Units	With the nuclear service cooling water system operating, the cooling coil valves will be opened and the units started. The cooling water flow and fan operation will be monitored.

1.4.60 CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (CATEGORY A)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzle as is practical.

Answer:

The delivery capability of the reactor building spray system will be tested up to the last valve before the spray nozzles, on a regular schedule.

1.4.61 CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (CATEGORY A)

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Answer:

2 The reactor building emergency cooling units can be tested at full operation at any time. Transfer to alternate power sources can also be tested. The operational sequence that would bring the reactor building spray system

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into action, including the transfer to alternate power sources, can be tested in parts.

1.4.62 CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (CATEGORY A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers.

Answer:

Air cleanup is accomplished by the reactor building ventilation system which:

a. Filters all air during normal operation at a minimum rate of two air changes per hour. 2

b. Filters contaminated air through high efficiency and charcoal filters before discharging it to atmosphere.

c. Purges the reactor building with outside air whenever desired.

All critical parts can be physically inspected.

1.4.63 CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (CATEGORY A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Answer:

Active components of the cooling part of the ventilation system are normally in operation at all times. Thus, their operability and performance can be determined at any time.

Active components of the purge part of the ventilation system can be tested each time the purge system is put into operation.

1.4.64 CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (CATEGORY A)

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

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Answer:

A capability for periodic testing and surveillance of the air cleanup system to ensure that (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits is provided.

Filters in the cooling system will be tested and inspected periodically. Filters in the purge system are outside the reactor building and available for testing and inspection at any time.

1.4.65 CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (CATEGORY A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

Answer:

The full operational sequence that would bring the air cleanup system into action, including the transfer to alternate power sources and the design air flow delivery capability, can be tested at any time.

1.4.66 CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (CATEGORY B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Answer:

Both new and spent fuel assemblies are stored in racks in parallel rows having a center-to-center distance of 21 in. in both directions. New fuel is stored in air. Spent fuel is stored in borated water. The spacing is sufficient to maintain a k_{eff} of less than 0.9 for the new fuel assemblies when in unborated water.

1.4.67 CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (CATEGORY B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

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Answer:

The spent fuel cooling systems cools and purifies spent fuel pool water during normal plant operation and transfers borated fluid between the refueling canal and storage during refueling. The system is designed to remove the decay heat from the fuel assemblies located in the spent fuel storage pool. It is designed to handle a maximum of 1-1/3 cores.

See Section 9.4 for details.

1.4.68 CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (CATEGORY B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

Answer:

The spent fuel storage racks are located to provide a minimum of 13 feet of water shielding over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mrem/hr during the storage period.

The exposure time during refueling will be limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The waste disposal system is designed to provide for controlled handling and disposal of liquid, gaseous, and solid wastes which will be generated during plant operation. The principal design criterion is to ensure that station personnel and the general public are protected against excessive exposure to radiation from wastes in accordance with limits defined in 10 CFR 20.

1.4.69 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (CATEGORY B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Answer:

The spent fuel storage pool is a separate building. The liquid waste holdup tanks and the gaseous waste storage and disposal equipment are located within a separate area of the auxiliary building. Both of these areas provide confinement capability in the event of an accidental release of radioactive materials, and both are ventilated with discharges to the plant vent. Analysis has demonstrated that the accidental release of the maximum activity content of the waste gas decay tanks will not cause doses in excess of the limits set forth in 10 CFR 100.

Radioactive liquid effluent leakage into the component cooling water system will be determined by monitors on the component cooling water pump suction header. Any accidental leakage from liquid waste storage tanks will be collected in the auxiliary building sump and transferred to other tanks to prevent releases to the environment.

A small purification loop is provided for removing fission products and other contaminants in the spent fuel storage pool water.

1.4.70 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (CATEGORY B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence, except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Answer:

The radioactive waste system will collect, segregate, process, and dispose of radioactive solids, liquids, and gases in such a manner as to ensure compliance with 10 CFR 20.

Liquid and solid wastes will be processed in a batch manner for off-site disposal. Gaseous wastes released to the environment will be monitored and discharged with suitable dilution to assure tolerable activity levels on the site and at the site boundary.

The gaseous waste system will store accumulated gas that is released during operation. The contents of the delay tanks will be sampled, and a release rate established consistent with the prevailing environmental conditions. In-line monitoring will provide a continuous check on the release of activity.

Permanently-installed area detectors and the plant vent detectors are used to monitor the discharge levels to the environment. In addition, portable monitors are available on site for supplemental surveys, if necessary.

Radioactive liquid effluent leakage into the component cooling water systems will be determined by monitors. These monitors are used for normal operational protection as well as for accident conditions. Radiation detectors monitor the gaseous residual activity prior to discharge to the plant vent and environment. The limits of 10 CFR 100 will not be exceeded under any conditions.

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1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

The research and development programs that have been initiated to establish final design or to demonstrate the capability of the design for future operation at a higher power level are summarized as follows:

1.5.1 XENON OSCILLATIONS

An analysis to evaluate the possibility of xenon oscillations throughout core life is underway. A modal analysis to determine critical parameters has been completed, and the detailed spatial calculations are in progress. If it is determined that such oscillations may occur, appropriate design changes to eliminate or control the oscillations will be incorporated.

See also Section 3.2.2.2.3, and Sections 3A.8, 3A.9, 3A.10, 3A.11, 3A.12, and 3A.16 of Appendix 3A.

1.5.2 THERMAL AND HYDRAULIC PROGRAMS

B&W is conducting a continuous research and development program for heat transfer and fluid flow investigations applicable to the design of the Rancho Seco Station. Two important aspects of this program are:

a. Reactor Vessel Flow Distribution and Pressure Drop Tests 3

A 1/6-scale model of the vessel and internals is under test to measure the flow distribution to the core, fluid mixing in the vessel and core, and the distribution of pressure drop within the reactor vessel.

b. Fuel Assembly Heat Transfer and Fluid Flow Test

Critical heat flux data have been obtained on single channel tubular and annular test sections with uniform and nonuniform heat fluxes, and on the multiple rod fuel assemblies with uniform heat fluxes. These data have been obtained for a range of pressure, temperature, and mass velocities encompassing the reactor design conditions. This work is being extended to include multiple rod fuel assemblies with nonuniform axial heat generation. Additional mixing, flow distribution, and pressure drop data will be taken on models of various reactor flow cells and on partial full-scale fuel assemblies.

See also Section 3.3.2, and Sections 1A.2-1 and 1A.5 d. of Appendix 1A.

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1.5.3 FUEL ROD CLAD FAILURE

A study of clad failure mechanisms associated with a loss-of-coolant accident is presently underway. This study has included identification of the potential failure mechanisms, a search of the literature to obtain applicable data, evaluation and application of existing data, and scoping tests to obtain data on potential failure mechanisms. The initial results of this study include the identification of the failure mechanisms, an evaluation of the information available in the literature concerning these mechanisms, and an evaluation of the effects of these mechanisms on the reactor system design.

The objective of the study is to ensure that there are no potential failure mechanisms that might interfere with the ability of the emergency core cooling systems to terminate the core temperature transient and remove decay heat in the event of a loss-of-coolant accident. These potential failure mechanisms include clad melting, zirconium-water reaction, eutectic formation between the Zircaloy-clad and the stainless-steel spacer grids, the possibility of clad embrittlement as a result of the quenching during core flooding, and clad perforation or deformation accompanying its failure. In the case of clad melting and zirconium-water reaction, our present design limit for peak clad temperature precludes these as possible failure modes. Information available in the literature, along with experimental evidence from tests conducted by B&W, show that brittle fracture of the cladding will not occur as a result of quenching following a loss-of-coolant accident, 3 and that eutectic formation between dissimilar core materials will not interfere with the flow of emergency core coolant after the accident.

B&W has undertaken a program to evaluate the effects of perforation and deformation of fuel rods during the temperature transient following the loss-of-coolant accident. Preliminary tests have been run on nine samples of Zircaloy-4 cladding filled with ceramic pellets, and additional experiments are planned to gain a clearer understanding of the effects of temperature excursions on Zircaloy-clad fuel elements. Current plans include performance of a three-phase program. In the first two phases which are experimental, single-rod excursions will be performed to better establish temperature-pressure relationships at the time of clad perforation. The single-rod tests of the first phase will also investigate the extent of deformation to be expected under the varying conditions associated with simulated in-reactor temperature excursions. These will include the effects of hydrogen concentration and oxide films. The second phase of the program will consist principally of multirod tests to explore the effect of the restraining action of spacer grids and adjacent fuel rods and to determine the randomization of the localized deformation in an assembly of fuel rods. In the third phase of the program, the data obtained from the two experimental phases will be applied to the analysis of the effects in a loss-of-coolant accident.

See also Section 14A.30 of Appendix 14A.

1.5.4 HIGH BURNUP FUEL TESTS

B&W is conducting a program to obtain a better understanding of fuel growth rates and irradiation effects on cladding, the influence of hydrogen on cladding, and fission-gas release at high burnup for the specific design burnup projected for peak-power regions in the reactor.

The fuels irradiation program will test fuel specimens at design temperatures and at exposures in excess of those obtained in the fuel rod. The specimens irradiated to the design burnup are scheduled to be completed in mid-1969. The program will provide information on the swelling rate of UO_2 as a function of burnup, density, heat rate, and cladding restraint. Fuel specimens will be operated at heat rates up to 21.5 kw/foot, which is in excess of the peak specific power in the core. The burnup will range up to 75,000 MWD MTU. The fuel rods will operate with a cladding surface temperature of 650 F.

A detailed report of sources of information for the irradiation of clad and fuel has been presented in Section 3.2.4.2 plus references. In addition to the PSAR references, irradiation of fuel assemblies or partial fuel assemblies with Zircaloy-clad UO_2 is in progress in the Saxton and Big Rock Point reactors. These data will demonstrate the behavior of fuel assemblies under the combined effects of irradiation, pressure cycles, thermal gradients, reactor coolant environment, and fuel-clad restraints.

A program has been carried out to determine the effects of irradiation on the mechanical properties of Zircaloy-4. Tests were conducted to temperatures as high as 775 F. 3

See also Sections 1A2-5 and 1A.7 of Appendix 1A.

1.5.5 INTERNALS VENT VALVES

The internals vent valves will be designed to relieve the pressure generated by steaming in the core following the LOCA so that the core will remain sufficiently covered. The valves will also be designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe.

Testing of the valves will consist of the following:

a. A full-sized valve assembly (seat, locking mechanism, and socket) will be hydrostatically pressure-tested at static conditions to the maximum pressure expected to result during the blowdown.

b. Sufficient tests will be conducted at zero pressure to determine the frictional loads in the hinge assembly, the inertia of the valve cover, and the cover rebound resulting from impact of the cover on the seat so that the valve response to cyclic blowdown forces may be determined analytically.

c. The valve assembly will be pressurized to determine what pressure differential is required to cause the valve to begin to open. A determination of the pressure differential required to open the valve to its maximum open position will be simulated by mechanical means.

d. A valve assembly will be installed and removed remotely in a test stand to judge the adequacy of handling equipment.

e. A valve assembly will be prototype tested over an appropriate range of vibration frequencies and amplitudes to verify the analytical results showing that the valve will not unseat because of vibration during normal operation.

Since the temperature differential existing across the valve assembly during normal operation in the reactor is only approximately 55 F, and since the same material is used for the valve seat, socket, and cover, there is no need to conduct tests at elevated temperatures.

See also Section 3.3.4, and Sections 1A.2-4, 1A.3, 1A.6, and 1A.11 of Appendix 1A.

3 1.5.6 CONTROL ROD DRIVE LINE TEST

The test assembly for this program is a full-sized fuel assembly with associated control rod and control rod guide, adjacent internals, and control rod drive. The unit is being tested under conditions of temperature, pressure, flow, and water chemistry specified for the full-sized reactor installation. This program will embrace a prototype phase in which the unit will be subjected to misalignment, varying flow, and temperature. The second phase of this program is one of life-testing where the unit will be continuously cycled to cover the number of feet of travel and the number of trips anticipated for its life in the reactor. Both phases of the program will confirm the operability of the drive line in normal and misaligned conditions, confirm the rod drop times and load-carrying characteristics of the actuator, indicate vibration and fretting wear characteristics of the control rods and fuel assemblies, and determine the wear characteristics of all the drive line components. Also, a component test program is being conducted using autoclave testing of selected components at reactor pressure and temperature. The purpose of this program is to seek out potential material and/or design problems prior to production unit testing.

1.5.7 ONCE-THROUGH STEAM GENERATOR TEST

Testing necessary to prove the adequacy of the once-through steam generator design for service at the initial power level and to confirm the size and configuration of the unit has been completed.

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Steady state and load-changing operations using once-through steam generator models were performed to demonstrate the ability of the unit to follow the transients and to demonstrate the interaction of the control system with the water level, steam pressure, and flows. The test equipment consisted of one 37-tube, full-length unit; one 19-tube, full-length unit; and a full-length, 7-tube unit. The tubes were fabricated in accordance with the production techniques anticipated for the full-sized unit. 3

The latter portion of the program included tests to determine the natural frequency of the tubes in the steam generator by subjecting them to artificially-induced vibrations from an external source. The buckling and vibration characteristics verify the structural integrity of the tube design.

Primary and secondary blowdown tests on the models have demonstrated integrity of the unit under conditions of rapid depressurization and large tube-to-shell temperature differentials. The results of these tests are being used in the development and verification of analytical models for steam system blowdown analyses. Verification of these models is scheduled for completion during the first quarter of 1969. 4

See also Sections 1A.2-3 and 1A.5 A. in Appendix 1A.

1.5.8 SELF-POWERED DETECTOR TESTS 3

The test units for this program are the self-powered detectors described in 7.3.3. These units have been tested in the B&W Test Reactor at conditions of temperature and neutron flux anticipated in a central station reactor. These units are currently being tested in the Big Rock Point Nuclear Power Plant where they are exposed to temperature, neutron flux, and flow for conditions approximating those in the Rancho Seco Station. The results of these programs will provide a detector system with predictable characteristics of performance and longevity under incore conditions.

1.5.9 BLOWDOWN FORCES ON INTERNALS

B&W has developed an analog computer model to obtain detailed information on the forces imposed on the reactor vessel internals during the subcooled portion of blowdown following a reactor coolant system rupture. The model can be used to simulate leaks at any location in the hot or cold leg piping. Leak sizes up to a complete shear of a main coolant pipe can be simulated. Resistance to flow between all regions is simulated, as well as the inertia of the fluid in the connecting flow paths. Pressure in each region is calculated by using the equations for conservation of mass and momentum and assuming an isentropic expansion of water in each region. The minimum pressure in each region is restricted to the saturation pressure corresponding to the temperature in that region.

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Test results have been obtained by the Phillips Petroleum Company for the blowdown of a vessel with and without simulated reactor vessel internals. Additional blowdown testing has been conducted and is still underway using
3 the 1/4-scale LOFT vessel. Tests have been conducted with and without internals in the vessel.

The tests that have been completed, together with those that are underway, will provide an adequate amount of test data to verify the B&W analytical
4 model. Verification of the model is scheduled for completion during the fourth quarter of 1968.

1.5.10 RADIO IODINE SPRAY REMOVAL SYSTEM

4 SMUD agrees to participate in one of the two R & D programs on radio iodine spray removal system additives presently reviewed by AEC.

SMUD's selection will be made by August 1, 1968.

1.6 SMUD'S COMPETENCE TO BUILD AND OPERATE NUCLEAR PLANT

SMUD is a public agency organized pursuant to laws of the State of California. It has sole responsibility for supplying electric energy to consumers within the capital city of Sacramento and surrounding area, totaling 650 square miles. Its operation commenced in January 1947. That year an average of 68,000 consumers created a peak demand of 77 megawatts and consumed some 380 million kilowatt-hours. Today, the corresponding figures are 205,000 consumers, 750 megawatts and 3000 million kilowatt-hours. This growth has been met effectively with a high level of service and at low rates.

The increased energy required during these two decades of rapid growth has been obtained through purchases from the US Bureau of Reclamation, through purchases and interchange with the Pacific Gas and Electric Company, and through construction by the District of high-head hydro plants. Construction of the latter has been underway since 1957. Today, 260 megawatts of this hydro are in operation. By 1971, this resource will have been fully developed with 600 megawatts of generation in operation.

These sources, together with additional power and energy available for a limited period of time from the Northwest and delivered via the Pacific Northwest-Pacific Southwest EHV Intertie, should satisfy SMUD's energy requirements through the year 1972 and into the year 1973. Appropriate margins are considered in this estimate.

SMUD's first attention to nuclear power was in 1956 when the present general manager (then manager of engineering) obtained clearance for access to nuclear power information, some of which at that time was still classified. This was followed by visits to Shippingport during and subsequent to completion of that nuclear plant. Later, the progress at the small nuclear plants being constructed at Hallam, Nebraska and Piqua, Ohio was followed through first-hand reports. A more close observation, now with some staff involved, was made of developmental work at the Vallecito nuclear test facilities and of construction of the nuclear power plant at Humboldt Bay. Finally, following publication of Jersey Central's Oyster Creek cost studies in February 1964 and the AEC seminar at Germantown for power company executives in March 1964, SMUD determined to plan nuclear development for the next required large block of energy.

The schedule for this nuclear development program was spread over a period of nine years. The program was to locate an acceptable area for a nuclear plant by 1965, to purchase a site in this area by 1966, to interview and to select architect-engineers for the project by 1967, to contract for the major equipment required for the plant by 1968, and to have the plant completed five years later--by 1973. To date, this schedule has been kept. Awards for the turbine-generator and nuclear steam supply system were made by mid-1967--slightly ahead of schedule.

Pursuit of this schedule has involved a major effort by the general manager, a member of his staff, and seven qualified members of his engineering, operating and construction departments. The past three-year period of

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activity has included visits by various members of this group to major nuclear plants now under construction in this country, to the offices of architect-engineers and utilities designing and building these projects, and to the facilities of three of the four domestic manufacturers of large nuclear steam supply systems. Also during this three-year period, six of these employees (including the Manager of Operations) each acquired six units of post-graduate credit in nuclear engineering. For one of these employees, this was a repeat of a six-week intensive summer course at the University of Michigan which also included laboratory work.

For large hydro plant construction it has been a policy of SMUD not to use "turnkey" contracts, but to retain a competent engineering firm to design the plant and to manage the various contractors who construct the plant. Equipment for the plant is purchased by SMUD and construction contracts for the plant are let by SMUD. Experience has shown that to accomplish project construction satisfactorily in this manner, SMUD must have within its own organization an overall engineering competence in the project field.

The design and construction of Rancho Seco nuclear plant will be handled in this same manner. Thus, following the interview and appraisal of each of the engineering firms in this country who by the end of 1966 had achieved experience in nuclear power plant work, the Vernon, California, office of the Bechtel Corporation was selected to perform the design of Rancho Seco and to manage the contractors who will construct the plant. Bechtel inspection forces also will follow the manufacture of critical pieces of equipment which have been or will be purchased directly by SMUD.

SMUD's assistant chief engineer, Mr. John Mattimoe, who is experienced in both design and construction of large power projects and who supervises the District's major project design and construction activities, has been assigned the responsibility for this project within SMUD's organization. Under Mr. Mattimoe, a nuclear plant group is being established to encompass the necessary competence in nuclear engineering, mechanical engineering, and electrical engineering in addition to civil engineering,
2 in the accomplishment of this assignment.* This will be a permanent group which will be expanded as may be required to perform the same work on future plants. The structure of the internal organization and the outside consultants and contractors reporting to the assistant chief engineer is shown on Figure 1.6-1.

A properly trained and properly licensed force with experienced supervisory personnel to operate Rancho Seco is to be formed in the five years between now and 1973 when the plant is to be completed. Top responsibility for this important job is being undertaken directly by Mr. Herbert Hunt, who is SMUD's operating department manager. The planned structure for this organization is as shown in Figure 12.2-1. The employees required will be recruited and trained well in advance of the scheduled plant start-up with key personnel at the plant site during most of the construction period. Large plant operating experience for some of these employees is to be available at plants owned and operated by the Pacific Gas and Electric

2 * Appendix 1C provides a detailed description of the acquisition, training and use of these personnel.

Company. The schedule of the build-up and training of this operating group is shown in Figure 12.3-1. As a direct and constant check on the nuclear operation, with particular emphasis on safety, a Nuclear Operations Safety Review Committee will be established. It will be entirely separate from the operating department. It will be composed primarily of individuals from nuclear design and construction who will report to the assistant chief engineer. 2

In addition to the organization described, SMUD plans to employ outside consultants who, on a periodic basis, will conduct safety audits of the plant and review other aspects of the operations. These consultants will report directly to the general manager and chief engineer.

The relationship of the operating department manager and the assistant chief engineer to the remainder of the SMUD organization is shown on Figure 1.6-2.

We believe the foregoing demonstrates that:

a. SMUD's top management currently has a good understanding of the magnitude of its nuclear undertaking and of the responsibility this undertaking entails.

b. The firm of architect-engineers retained by SMUD to perform the detailed design of the plant and supervision of construction is responsible to SMUD alone, and is barred by contract from participating in supply or construction contracts, eliminating the possibility of any conflict of interest. 2

c. SMUD now has, and is further developing, a competent group of engineers having the required disciplines to independently appraise and direct the work performed for SMUD by the architect-engineers, the nuclear steam suppliers, and others.

d. The operating group for the plant will be recruited and trained specifically for this plant well in advance of the plant's operation and with exposure for principal members of the group to the plant as it is being constructed.

e. Operation of the plant, especially its safety aspects, will be subjected to a rigorous system of scrutiny totally outside the operating department.

f. The SMUD organization--current and planned--has ample depth and breadth to handle the construction and operation of a nuclear power plant.

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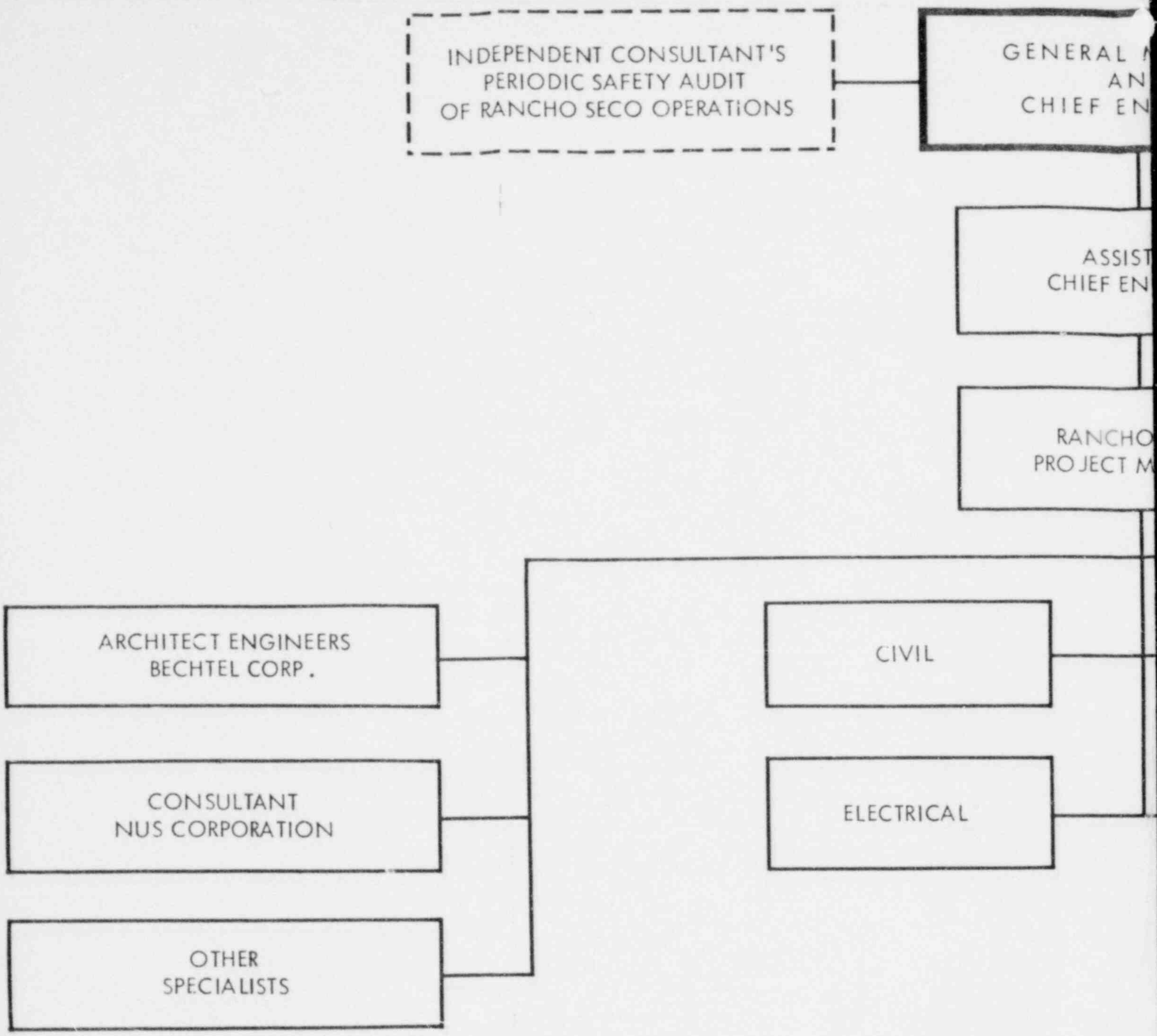
INDEPENDENT CONSULTANT'S PERIODIC SAFETY AUDIT OF RANCHO SECO OPERATIONS
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CONSULTANT NUS CORPORATION
OTHER SPECIALISTS
CIVIL
ELECTRICAL

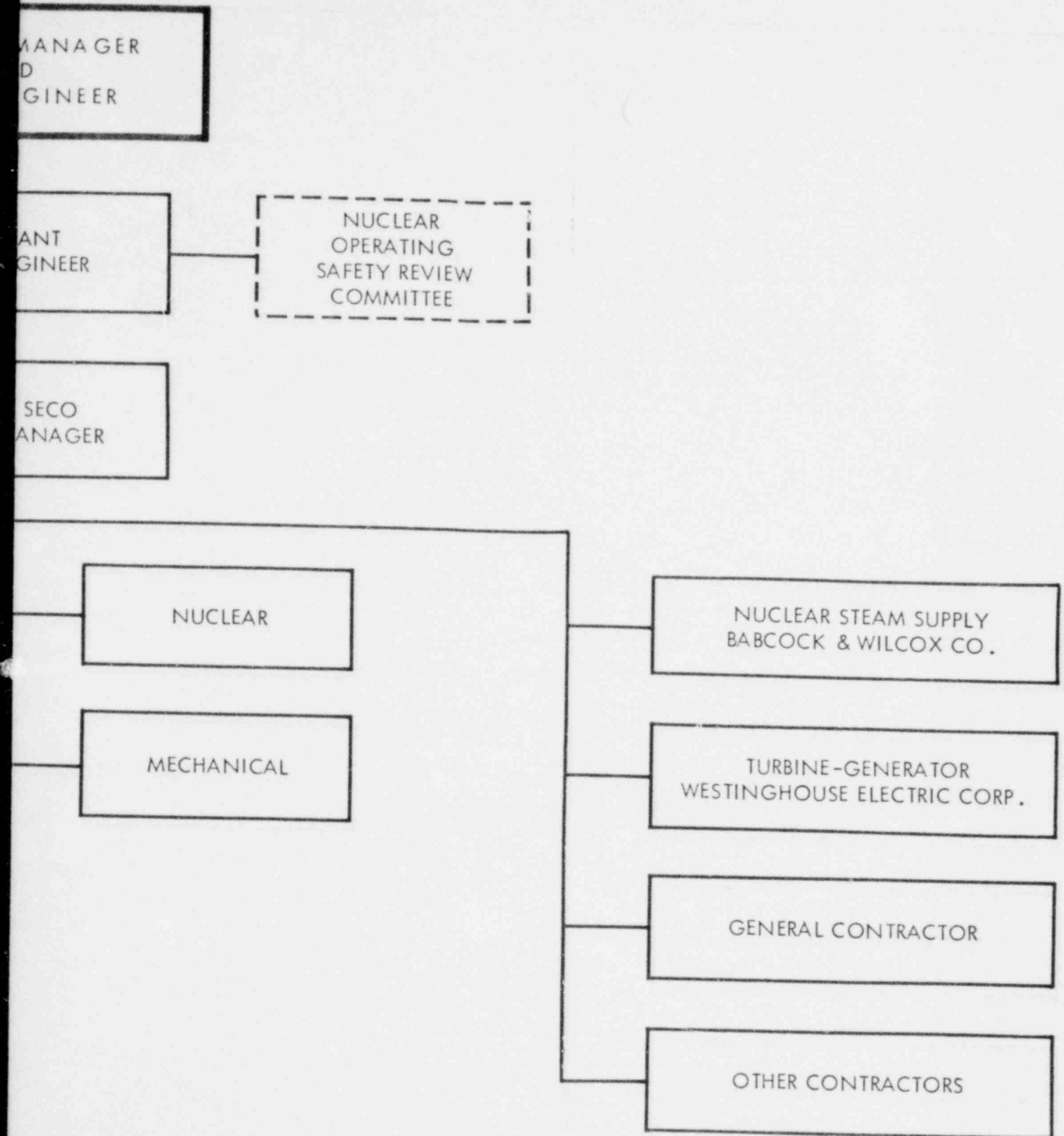
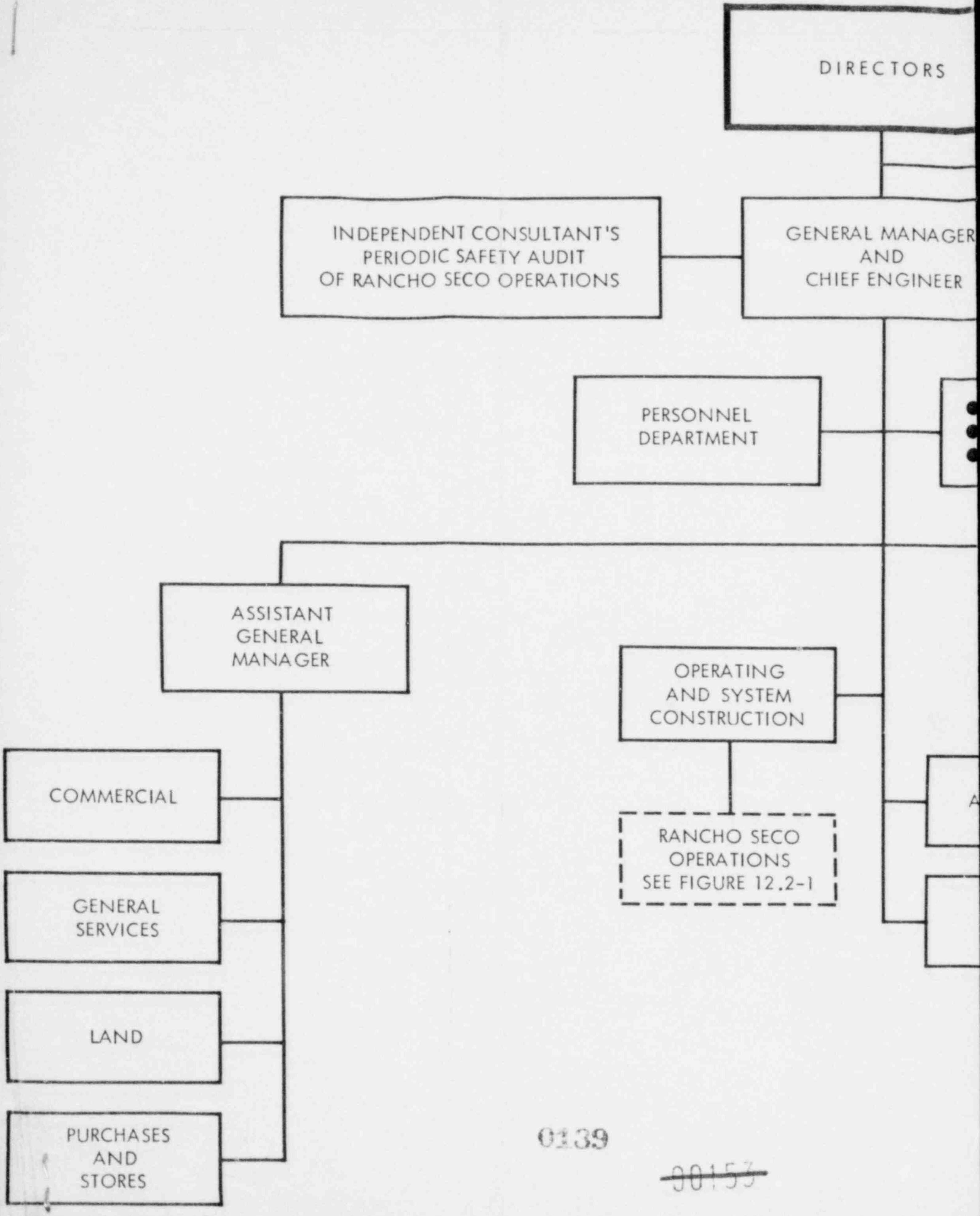


FIGURE 1.6-1
NUCLEAR
PROJECT ORGANIZATION



DIRECTORS
INDEPENDENT CONSULTANT'S PERIODIC SAFETY AUDIT OF RANCHO SECO OPERATIONS
GENERAL MANAGER AND CHIEF ENGINEER
PERSONNEL DEPARTMENT
ASSISTANT GENERAL MANAGER
OPERATING AND SYSTEM CONSTRUCTION
COMMERCIAL
RANCHO SECO OPERATIONS SEE FIGURE 12.2-1
GENERAL SERVICES
LAND
PURCHASES AND STORES

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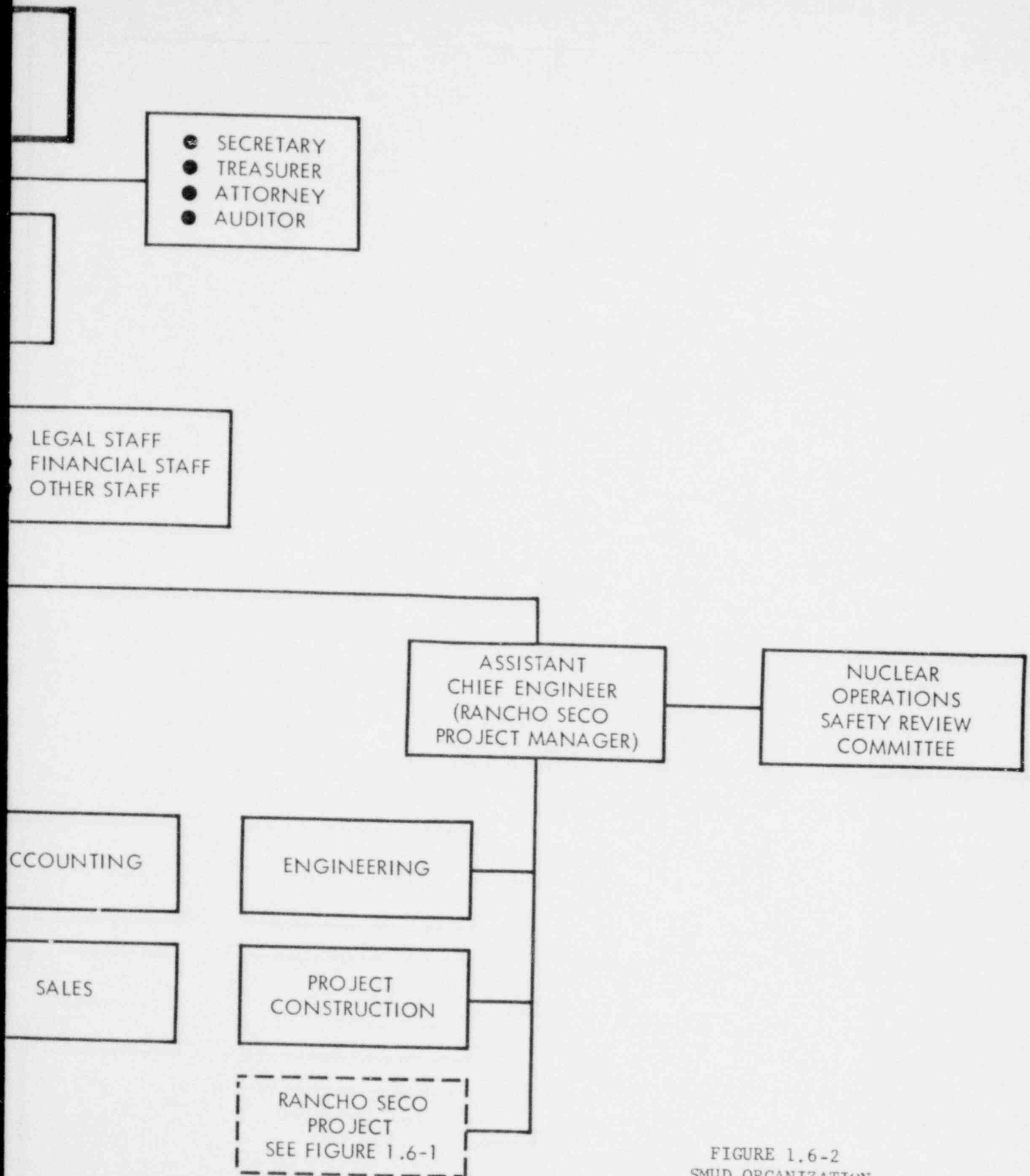

FIGURE 1.6-2
SMUD ORGANIZATION

1.7 IDENTIFICATION OF CONTRACTORS AND AGENTS

SMUD will be responsible for the design, purchasing, construction and operation of Rancho Seco Unit No. 1. Bechtel Corporation has been retained to act as architect-engineer and to supervise construction of the plant. This practice has been successfully followed for all of SMUD's generating facilities now in service or planned.

A contract has been entered into with the Babcock & Wilcox Company to design, manufacture, and deliver to the site the complete nuclear steam supply system. In addition, Babcock & Wilcox will supply competent technical consultation during erection, initial fuel loading, testing, and initial start-up of the complete nuclear steam supply system. Also, B&W will cooperate in the training and licensing of operating personnel prior to and during the start-up and initial operating period.

A contract was awarded to Westinghouse Corporation for a turbine-generator. Other supply and construction contracts will be awarded as required to meet completion dates.

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1.8 CONCLUSIONS

SMUD believes the personnel available to it to design and construct the Rancho Seco Nuclear Generating Station are well qualified. A fully competent operating staff will be employed and trained well in advance of start-up. The District is utilizing a conservative design which will produce electric power safely and economically.

Toward this end--

a. The site has been examined and found to be suitable for the nuclear plant. The plant at this site is compatible with surrounding population and land uses, present and expected. Site characteristics of meteorology, hydrology, geology, and seismology are favorable.

b. The reactor system chosen is a practical design of proven type. Its expected performance will not require fuel exposures or energy-release rates higher than those presently proved achievable by using materials now available. Its shutdown margin and performance characteristics are comparable to those used in existing reactors. Before it commences commercial operation, the reactor system will be thoroughly tested to confirm that the desirable features were designed into it, and that it will perform as expected with full safety margins.

c. The reactor will be installed in an enclosure that is both modern and conservative in design, which will be able to contain and control all materials, vapors, or energies which could credibly be released as a result of an accident under any coincident condition. Supplementing the enclosure capability will be engineered safeguards that will reduce to a very minimum the consequences of any accident and ensure that the dynamic conditions existing after an accident are kept well within safeguards design parameters.

d. The plant waste and emergency systems will be designed to release only gaseous effluents permitted by the AEC Regulations. No liquid waste effluents will be directly released to the environment.

e. SMUD is in the process of recruiting additional personnel to round out its engineering staff with employees having special competence in the field of nuclear generation.

f. A training program is planned that will adequately prepare operating personnel so that they will be qualified to test, start-up and operate the nuclear unit.

Accordingly, it is SMUD's conclusion that the proposed Rancho Seco Nuclear Generating Unit No. 1 can be designed, constructed, and operated in a safe manner; that the proposed design will provide adequate protection to the public from any sequence of events resulting in disablement of equipment from causes, natural or mechanical; and that the District and their consultants are qualified to design, construct, start, operate and maintain these proposed nuclear generating units in accordance with all applicable laws and regulations and in a manner satisfactory to the Atomic Energy Commission, to the public interests, and to itself.

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1.9 REFERENCES

1. "General Design Criteria," Federal Register AEC 70 (July 11, 1967).

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