Westinghouse Technology Manual

Chapter 17.0

Plant Operations

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17.0 PLANT OPERATIONS

Learning Objectives:

1. Arrange the following evolutions in the proper order for a plant startup from cold shutdown:

   a. Start all reactor coolant pumps,
   b. Place all engineered safety systems in an operable mode,
   c. Establish no-load T_{avg},
   d. Take the reactor critical,
   e. Start a main feedwater pump,
   f. Load main generator to the grid, and
   g. Place steam generator level control system in automatic.

17.1 Introduction

This chapter will briefly discuss the basic procedures for startup, power operation, and shutdown of the pressurized water reactor described in this manual. The discussion will be general in nature and is designed to show how the systems previously discussed are utilized during plant operations.

17.2 Plant Heatup

17.2.1 Initial Conditions

The nuclear steam supply system (NSSS) is in the "cold shutdown" mode (T_{avg} = 120°F, pressurizer pressure = 50 - 100 psig, boron concentration sufficient to yield 10% shutdown margin, pressurizer solid, reactor coolant pumps off). Decay heat is being removed by the residual heat removal system (RHR) with letdown from RHR established for reactor coolant system cleanup. Pressure in the solid system (Figure 17-1) is being maintained by adjusting charging and letdown flow. The steam generators are in the "wet layup" condition (filled to the 100% level with water) and all secondary systems are secured with the exception of one circulating water pump. The main and feedwater pump turbines are on the turning gear. All pre-startup checklists have been completed.

17.2.2 Operations

A pressurized water reactor may have a positive moderator temperature coefficient at low temperatures due to the soluble poison in the moderator. To minimize the magnitude of the positive moderator temperature coefficient or make it negative, the plant is brought to near operating temperatures with reactor coolant pump heat before the reactor is made critical. To operate the reactor coolant pumps, reactor coolant system pressure must be increased to approximately 400 psig to satisfy net positive suction head requirements. (Pressure must be maintained below 425 psig while RHR is aligned to the reactor coolant system.) When operating the reactor coolant pumps at low pressures, the reactor coolant pump number one seal bypass valve must be open to ensure adequate flow to cool and lubricate the pump radial bearing.

Pressure is increased by maintaining charging flow greater than letdown flow. When pressure is stable between 400 and 425 psig, the reactor coolant pumps are started to begin reactor coolant system heatup. Pressurizer heaters are energized to begin pressurizer heatup. Residual heat removal flow is diverted through the bypass line to bypass the heat exchanger and allow heatup.

This RHR system alignment is maintained to provide adequate letdown for pressure control and to remove the excess coolant volume produced by expansion due to heatup. During the entire heatup and pressurizer draining process, approximately one-third of the reactor coolant system volume (30,000 gallons of water) will be diverted to the holdup tanks through the chemical and volume control system.

As the reactor coolant system temperature approaches 200°F, steam generator draining is commenced through the normal blowdown system. If reactor coolant system oxygen concentration is high, hydrazine is added through the chemical and volume control system for oxygen scavenging. Oxygen must be in specification before exceeding 250°F.

After oxygen is within specification, a hydrogen blanket is established in the volume control tank. This is accomplished by securing the nitrogen regulator, opening the vent from the volume control tank to the waste gas header, and raising the volume control tank level to force the nitrogen to the waste gas system. After the volume control tank level has raised to approximately 95%, the hydrogen regulator is placed in service and the last of the nitrogen is purged to the waste gas system. Volume control tank level is allowed to return to normal with the hydrogen regulator maintaining an overpressure of approximately 15 - 20 psig.

When pressurizer temperature reaches saturation temperature for the pressure being maintained (450°F for 400 psig), a pressurizer bubble is established. Reactor coolant system temperature is approximately 250 - 300°F. The bubble is established by maximizing letdown and minimizing charging flow. This will cause the pressurizer level to decrease. System pressure will be maintained at 400 psig as the saturated pressurizer water flashes to steam. Pressure control can now be accomplished only by heater and spray operation. Residual heat removal is maintained in service to provide an additional letdown path to minimize the time necessary to "draw a bubble" in the pressurizer.

The main and auxiliary steam lines are warmed as steam is available during the plant heatup. Main steam isolation valves are opened initially as heatup begins.

As reactor coolant pressure continues to increase, letdown flow will also increase. The low pressure letdown valve is adjusted (closed) until the normal letdown pressure (340 psig) is achieved and then orifice isolation valves are shut as necessary to maintain letdown flow below the maximum.

Before reactor coolant system temperature reaches 350°F, the residual heat removal system is isolated from the reactor coolant system and is aligned for at-power operation (emergency core cooling system lineup). All reactor coolant system letdown is now through the normal letdown orifice path to the chemical and volume control system.

After the residual heat removal system is isolated from the reactor coolant system, system pressure is allowed to increase as the pressurizer temperature increases.

When pressurizer level, as read on the hot calibrated channels, indicates the no-load programmed setpoint, charging flow is placed in automatic. As system heatup continues, pressurizer level will try to increase due to coolant expansion. Pressurizer level control will compensate by reducing charging flow.

When reactor coolant system pressure reaches 1,000 psig, the emergency core cooling system accumulator discharge valves are opened and all emergency core cooling system equipment is checked for proper alignment.

After reactor coolant pump number one seal leakoff has increased to at least one gallon per minute on all reactor coolant pumps, the number one seal bypass valve is closed.

As pressure increases above P-11, the low pressurizer pressure engineered safety features actuation signal is automatically unblocked. Pressurizer heaters and spray valves are placed in automatic control when pressure reaches the normal operating value of 2235 psig.

When steam pressure is at or above 125 psig, main and feed pump turbine gland seals are established, and a condenser vacuum is drawn. Condenser vacuum is established by mechanical vacuum pumps and/or steam jet air ejectors.

As reactor coolant system heatup continues, the high steam flow engineered safety features actuation signal will be automatically unblocked when T_{avg} increases above 540°F. The steam dump system, operating in pressure control mode, will dump steam to the main condenser when steam pressure reaches a predetermined setpoint (normally 1,005 psig which is saturation pressure for the 547°F no-load reactor coolant system temperature). The steam dump system will dissipate the excess decay and reactor coolant pump heat and maintain T_{avg} approximately equal to 547°F. The startup feedwater system is used to feed the steam generators to maintain level at the no-load value.

Plant conditions are now as follows: normal operating temperature and pressure, reactor shutdown, normal condenser vacuum, steam dump to the condenser in the steam pressure mode, main and feedwater pump turbines on the turning gear, and all electrical power supplied from off-site.

The next step in the startup of the plant is to take the reactor critical.

17.3 Reactor Startup to Minimum Load

Reactor startups are normally performed at no-load temperature where the moderator temperature coefficient is at a low or negative value.

If necessary, the reactor coolant boron concentration is adjusted to the required value prior to startup. The required value is calculated by performing a reactivity balance (estimated critical condition calculation). For a pressurized water reactor, a specific critical rod height is chosen and boron concentration is adjusted to a value which will produce criticality at the desired rod height. Control rods must always be withdrawn above the rod insertion limit prior to criticality to ensure adequate "cocked" reactivity to satisfy shutdown margin requirements.

Immediately prior to reactor startup, functional checks are performed to ensure proper operation of the source and intermediate range nuclear instrumentation channels. A source and intermediate range channel are recorded and the "source range high flux at shutdown" alarm is blocked.

The shutdown rod banks (if not already withdrawn) are withdrawn in sequence, and then, the control banks are withdrawn in manual to achieve criticality. After criticality is achieved, a positive startup rate is established, and power level in increased. When power exceeds the source range permissive (P-6) setpoint, the source range trip is blocked and source range high voltage deenergized.

Power is then increased to 10^{-8} amps in the intermediate range where neutron flux is stabilized (leveled out) and critical data are taken. After critical data are taken, the reactor power increase is continued until the "point of adding heat" is reached. This is the power level (about 1% power) where the reactor is producing sensible heat.

The reactor operator hold 1% power while the turbine-driven main feedwater pump is warmed and placed in service. Feedwater supply is switched from the auxiliary feedwater system to the main feedwater pump.

Reactor power is increased to about 5% power in preparation for rolling the main turbine. Increasing reactor power will cause the steam dump valves to open further to dissipate the excess heat. Steam generator feedwater is controlled manually through the small (4 -6 inch) bypass valves to maintain level at the program setpoint. Providing excess reactor power yields a constant steam load as the turbine is rolled. As the turbine takes more steam, the steam dump valves will modulate closed. This makes control of the reactor and steam generator levels much easier. A heater drain pump is energized at this time.

The turbine acceleration rate is chosen, and the turbine is accelerated to synchronous speed. With the turbine at synchronous speed, reactor power is increased to six percent so that reactor power is greater than the initial turbine load.

The turbine is synchronized with the utilities electrical grid, and the generator output breaker is closed. The electrohydraulic control system automatically assumes five percent of full rated load. After turbine operation and other applicable instrumentation is checked, a turbine loading rate is chosen, and the turbine load is increased toward 15%.

As turbine load is increased, the reactor operator withdraws control rods to maintain $T_{avg} = T_{ref}$. During the load increase, the steam dump valves will shut as steam pressure decreases. When the valves are shut, steam dump control is shifted to T_{avg} control to be ready for a possible load rejection or reactor trip. Steam generator level continues to be controlled by manual operation of the main feedwater regulating bypass valves.

When power level exceeds the setpoint of the nuclear at-power permissive (P-10), the intermediate range rod withdrawal stop and the intermediate and power range (low setpoint) trips are manually blocked.

At or above fifteen percent power, the rod control system and steam generator level control system are placed in the automatic mode.

17.4 Power Operations

Power level is increased by selecting a desired load and load rate with the turbine electrohydraulic control system and allowing the reactor to follow the turbine load change. As turbine load increases, T_{avg} will tend to decrease. The automatic rod control system will sense this and withdraw control rods to increase reactor power.

As load is increased to 30% power, a second condensate/booster pump is started, and main generator hydrogen pressure is increased to its maximum value (75 psig).

As load increases between thirty and fifty percent, additional circulating water, feedwater, and heater drain pumps are started. At approximately 35% load, reheating steam is cut into the moisture separator-reheaters.

The single loop loss of flow permissive (P-8) enables the single loop loss of flow reactor trip when reactor power exceeds 35%.

At approximately 50% load, the third condensate/booster pump is started, and a calorimetric (heat balance) calibration of the power range nuclear instruments is performed.

Further calorimetrics are performed at 70% and 100% power to ensure proper calibration of the power range nuclear instrumentation.

Negative reactivity added by the power defect during the power increase is counteracted by automatic withdrawal of the control rods while the negative reactivity due to xenon and samarium production is counteracted by dilution of soluble poison from the coolant.

At all time, when the reactor is critical, the control rod banks must be maintained withdrawn above their respective insertion limits (Figure 17-2). All shutdown banks and control banks "A" and "B" must be fully withdrawn, and control banks "C" and "D" must be withdrawn at least as specified in Figure 17-2. Maintaining the rods above the rod insertion limit ensures sufficient available negative reactivity to achieve required shutdown margin in the event of a reactor trip.

17.5 Plant Shutdown

Plant shutdown is accomplished by essentially reversing the steps described in plant startup.

APPENDIX 17-1

PLANT STARTUP FROM COLD SHUTDOWN

I. INITIAL CONDITIONS

A. Cold shutdown - Mode 5
$K_{eff} < 0.99$
0% rated thermal power
$T_{avg} < 200°F$

B. Pressurizer

1. Temperature approximately 320°F, with a steam bubble established.

2. Level approximately 25% with level control in automatic.

C. RCS temperature 150 - 160°F

Note: Temperature may be less than 150°F depending on decay heat load from the core.

D. RCS pressure 100 psig

1. Charging and RHR letdown established

2. RCS pressure maintained by pressurizer temperature @ 320°F

3. RHR system in operation

E. Steam generators filled to wet-layup (100% level indication)

F. Secondary systems shutdown. Main turbine and main feedwater pump turbines on their turning gear

G. Pre-startup checklists completed

II. Instructions

A. Heatup from cold shutdown to hot shutdown (Mode 5 to Mode 4)

1. Permission received from operation supervisor for startup

2. Verify shutdown rods withdrawn or verify sufficient shutdown margin availability

3. Verify or establish RCP seal injection flow

4. Begin pressurizer heatup to increase RCS pressure.

CAUTION: Do not exceed a heatup rate of 100°F/hr on the pressurizer, 100°F/hr on the RCS, or 320°F T between pressurizer and spray temperature. Use auxiliary sprays for pressurizer-RCS mixing.

5. Maintain the RCS temperature < 160°F by adjusting flow through the RHR heat exchangers

6. Startup checklist for Technical Specification requirements completed

7. Begin establishing steam generator water levels to 50% on narrow range indication (steam generator blowdown system).

8. Open main steam line isolation valves

9. If required, commence condensate cleanup

10. Establish condenser vacuum

11. Continue pressurizer heatup to 430°F (RCS pressure 325 psig). Use the low pressure letdown control valve to maintain letdown flow. RCS pressure control is via heater and spray actuation.

12. Start the reactor coolant pumps. After five minutes running, sample the RCS for chemistry specifications. Partially open pressurizer sprays for mixing.

13. Stop residual heat removal system pumps

14. Allow RCS temperature to increase to 200°F

15. When RCS temperature reaches 200°F, determine that primary system water chemistry is within specifications

16. When condensate chemistry is within specifications as determined by chemical lab, align condensate and feedwater system to normal configuration.

17. Verify control rod drive cooling fans on before RCS temperature reaches 160°F

18. Terminate residual heat removal letdown to chemical and volume control prior to exceeding 350°F and 425 psig.

B. Heatup from Hot Shutdown to Hot Standby (Mode 4 to Mode 5)

1. Startup checklist for Technical Specification requirements completed

2. Complete emergency core cooling system master checklist

3. As the RCS pressure increases, maintain letdown flow 120 gpm by increasing the setting of the low pressure letdown control valve, and by closing the letdown orifice isolation valves as necessary.

4. Prior to reaching 1,000 psig in the RCS, open each of the cold leg accumulator isolation valves. Remove each valve's power supply.

5. When RCP no. 1 seal leakoff is > 1 gpm, or RCS pressure > 1,500 psig, close RCP seal bypass return valve. Verify no. 1 seal leakoff remains > 1 gpm.

6. When RCS pressure reaches 1,970 psig, verify pressurizer low pressure safety injection logic auto reset.

7. When T_{avg} exceeds 540°F, verify steam line safety injection logic auto reset.

8. The steam dump control system is in pressure control mode (set at 1,005 psig) to maintain RCS temperature at 547°F.

9. Place RCS pressure control in automatic to maintain 2235 psig.

10. Establish hot standby conditions of 540 - 547°F T_{avg}.

C. Heatup from Hot Standby to Power Operations (Mode 3 to Mode 1)

1. Administrative permission to take the reactor critical has been obtained.

2. Notify system dispatcher of unit startup and approximate time the generator will be tied on to the system.

3. Notify onsite personnel of reactor startup over P/A system.

4. If shutdown banks have not been withdrawn, complete a shutdown margin calculation (assuming SD banks out) and if desired SD margin will exist, withdraw the shutdown banks to the fully withdrawn position.

Note: Nuclear instrumentation shall be monitored very closely in anticipation of unplanned reactivity rate of change.

5. Calculate the estimated critical boron concentration for the desired critical control bank rod position (normally 150 steps on Bank D).

6. If necessary, conduct a boron concentration change to the estimated critical boron concentration. Equalize boron concentration between the reactor coolant loops and the pressurizer by turning on pressurizer backup heaters.

Note: Nuclear instrumentation shall be monitored very closely in anticipation of unplanned reactivity rate of change.

Note: Block the source range high flux level at shutdown alarm at both source range panels.

7. Withdraw the control bank rods in manual and take the reactor critical.

   a. Block source range trip at P-6

   b. Record critical data at 10^{-8} amps

8. If the control bank height at criticality is below the minimum insertion limits for the 0 percent power conditions.

   a. Re-insert all control bank rods to the bottom of the core.

   b. Recalculate the estimated critical boron concentration

   c. Borate to the new estimated critical boron concentration

   d. Withdraw the control bank rods in manual and take the reactor critical

9. Withdraw rods to bring reactor power to approximately 1% on power range indicators and select the highest power range channel to be recorded on NR-45.

10. Start a main feedwater pump at 1% power and maintain steam generator levels at 50 percent narrow range level indication during secondary plant startup by throttling the feedwater bypass regulating valves and operating the master feedwater pump speed controller and the individual steam generator feedwater pump control station in auto.

CAUTION: Coordinate all steam generator steam removal and significant feedwater changes with the reactor panel operator while rod control is in manual.

11. Turbine has been on turning gear at least one hour

12. Increase reactor power by manual adjustment of the control bank until the steam dump is bypassing steam flow equivalent to 8 percent nuclear power.

13. Verify the unit auxiliary and startup transformer cooling systems are aligned for automatic operation.

14. Start the turbine, bring it up to speed, and connect the generator to the grid. Transfer station power from the startup transformer to the unit auxiliary transformer.

15. Increase generator load at the desired rate, while maintaining T_{avg} by manual rod control.

16. Transfer feedwater flow from bypass valves to the main feed regulating valves. Maintain programmed level during this process.

17. When reactor power increases above 10 percent, ensure the nuclear at-power permissive (P-10) light comes on and the turbine at-power permissive (P-13) and at-power permissive (P-7) lights clear.

18. Manually block the intermediate range reactor trip and the power range low setpoint reactor trip after P-10 has been actuated.

19. When turbine power has increased above 15 percent, and T_{avg} equals T_{ref}, transfer reactor control system to automatic.

20. After rod control is placed in automatic, check steam pressure less than steam dump set point and steam dump valves full closed, then transfer steam dump to T_{avg} mode.

21. Above 15 percent power, transfer steam generator feedwater regulating valve control to auto when level is at setpoint and steam flow equals feed flow.

22. Continue turbine load increase to 100%

    a. Start secondary system components as required during power escalation. Additional components would include items such as condensate pumps, heater drain pumps, feedwater pumps, and condenser circulating water pumps.

    b. Maintain rate of load increase within plant design limits. These limits would include the loading limits imposed upon the main turbine and the limits imposed by boron dilution rates.

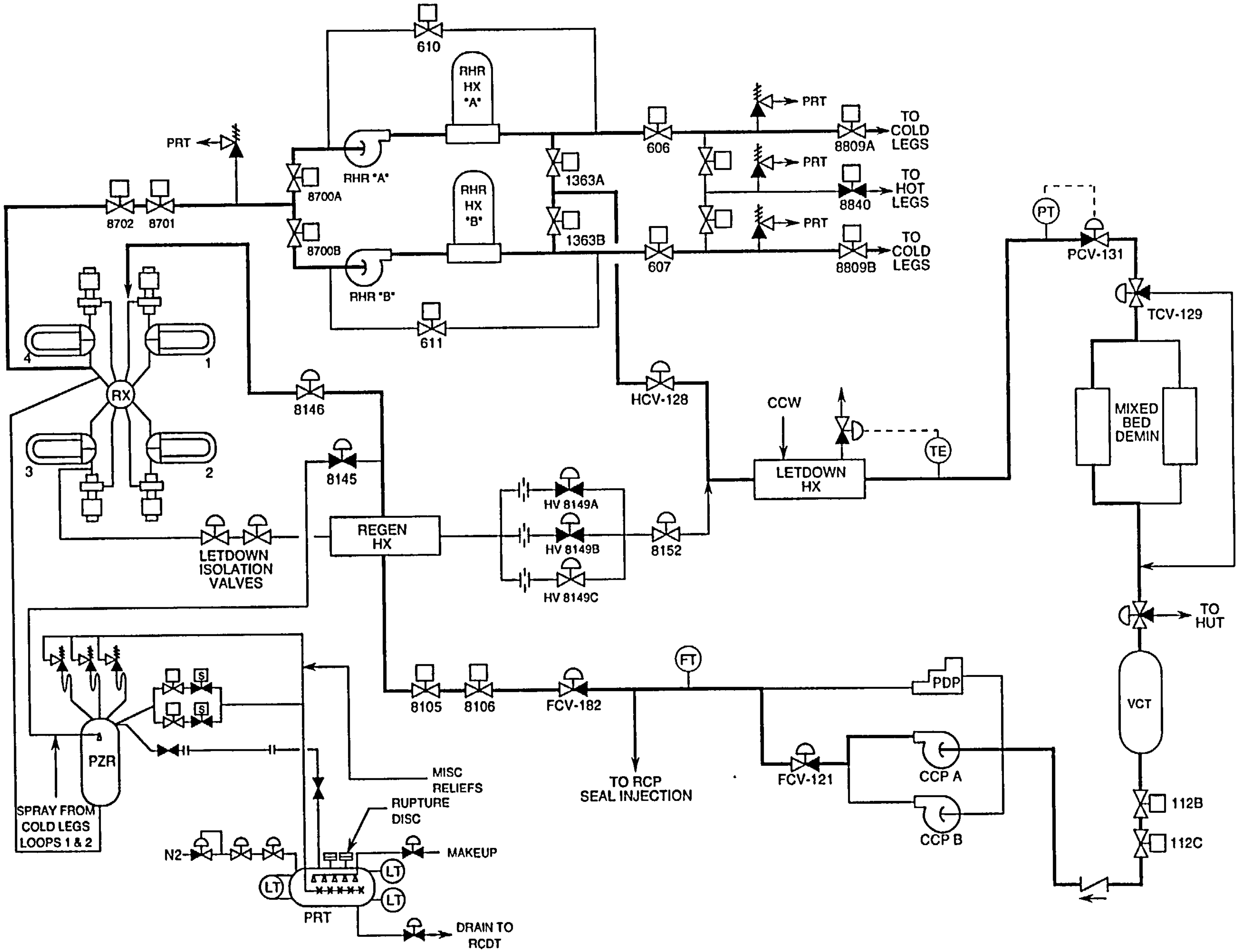

Figure 17-1 Solid Plant Pressure Control

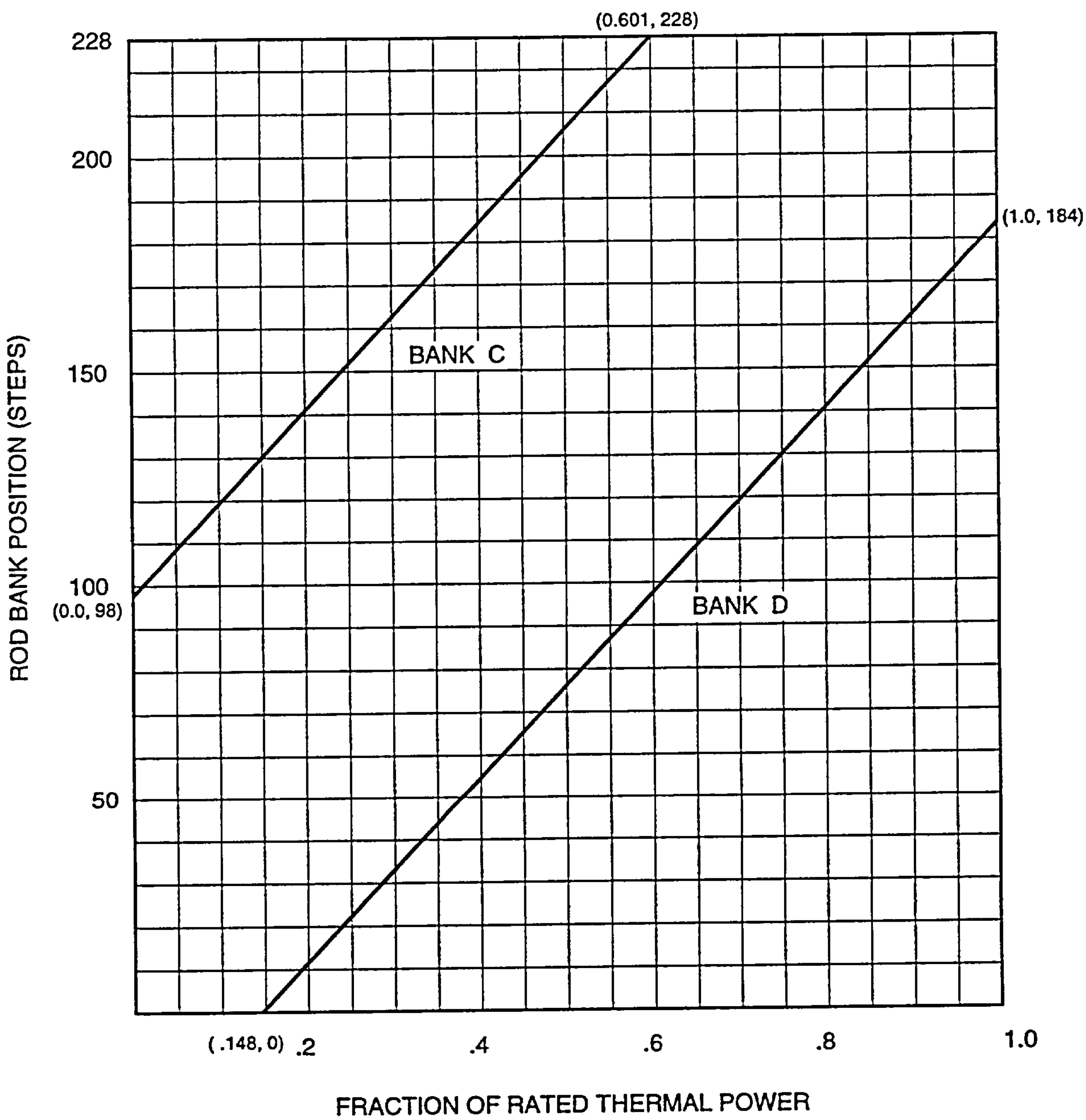

Figure 17-2 Control Bank Insertion Limits, 4-Loop Operations

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Chapter 18.0

Overview and Comparision of U.S. Commercial Nuclear Power Plants

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Overview and Comparison of U.S. Commercial Nuclear Power Plants

Nuclear Power Plant System Sourcebook

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Reactor Plant	City	State	Utility	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
ANO-1	Russellville	AR	Arkansas Power & Light Co.	PWR	B&W	Bechtel	2568	836	MDC
ANO-2	Russellville	AR	Arkansas Power & Light Co	PWR	C-E	Bechtel	2815	858	MDC
Beaver Valley 1	Shippingport	PA	Duquesne Light Co	PWR	W	Stone & Webster	2652	810	MDC
Beaver Valley 2	Shippingport	PA	Duquesne Light Co	PWR	W	Stone & Webster	2652	833	MDC
Bellefonte 1	Scottsboro	AL	Tennessee Valley Authority	PWR	B&W	TVA	3413	1213	DER
Bellefonte 2	Scottsboro	AL	Tennessee Valley Authority	PWR	B&W	TVA	3413	1213	DER
Big Rock Point	Charlevoix	MI	Consumers Power Co.	BWR	GE	Bechtel	240	69	MDC
Braidwood 1	Braidwood	IL	Commonwealth Edison Co.	PWR	W	Sargent & Lundy	3411	1120	MDC
Braidwood 2	Braidwood	IL	Commonwealth Edison Co.	PWR	W	Sargent & Lundy	3411	1120	MDC
Browns Ferry 1	Decatur	AL	Tennessee Valley Authority	BWR	GE	TVA	3293	1065	MDC
Browns Ferry 2	Decatur	AL	Tennessee Valley Authority	BWR	GE	TVA	3293	1065	MDC
Browns Ferry 3	Decatur	AL	Tennessee Valley Authority	BWR	GE	TVA	3293	1065	MDC
Brunswick 1	Southport	NC	Carolina Power & Light Co.	BWR	GE	UE & C	2436	790	MDC
Brunswick 2	Southport	NC	Carolina Power & Light Co.	BWR	GE	UE & C	2436	790	MDC
Byron 1	Byron	IL	Commonwealth Edison Co.	PWR	W	Sargent & Lundy	3411	1105	MDC
Byron 2	Byron	IL	Commonwealth Edison Co.	PWR	W	Sargent & Lundy	3411	1105	MDC
Callaway	Fulton	MO	Union Electric Co.	PWR	W	Bechtel	3565	1145	MDC
Calvert Cliffs 1	Lusby	MD	Baltimore Gas & Electric Co.	PWR	C-E	Bechtel	2700	825	MDC
Calvert Cliffs 2	Lusby	MD	Baltimore Gas & Electric Co.	PWR	C-E	Bechtel	2700	825	MDC
Catawba 1	Clover	SC	Duke Power Co.	PWR	W	Duke Power Co.	3411	1129	MDC
Catawba 2	Clover	SC	Duke Power Co.	PWR	W	Duke Power Co.	3411	1129	MDC
Clinton 1	Clinton	IL	Illinois Power Co.	BWR	GE	Sargent & Lundy	2894	930	DER
Comanche Peak 1	Glen Rose	TX	Texas Utilities Electric Co.	PWR	W	Gibbs & Hill	3425	1150	DER
Comanche Peak 2	Glen Rose	TX	Texas Utilities Electric Co.	PWR	W	Gibbs & Hill	3425	1150	DER

Table 2-1. General Plant Data - Sorted by Plant Name (Continued)

Reactor Plant	City	State	Utility	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Cooper	Brownville	NE	Nebraska Public Power District	BWR	GE	Burns & Row	2381	764	MDC
Crystal River 3	Red Level	FL	Florida Power Corp.	PWR	B&W	Gilbert	2544	821	MDC
D C. Cook 1	Bridgman	MI	Indiana/Michigan Power Co.	PWR	W	AEP	3250	1020	MDC
D C. Cook 2	Bridgman	MI	Indiana/Michigan Power Co.	PWR	W	AEP	3411	1060	MDC
Davis-Besse	Oak Harbor	OH	Toledo Edison Co.	PWR	B&W	Bechtel	2772	860	MDC
Diablo Canyon 1	Avila Beach	CA	Pacific Gas & Electric Co.	PWR	W	Pacific Gas & Electric	3338	1073	MDC
Diablo Canyon 2	Avila Beach	CA	Pacific Gas & Electric Co.	PWR	W	Pacific Gas & Electric	3411	1087	MDC
Dresden 2	Morris	IL	Commonwealth Edison Co.	BWR	GE	Sargent & Lundy	2527	772	MDC
Dresden 3	Morris	IL	Commonwealth Edison Co.	BWR	GE	Sargent & Lundy	2527	773	MDC
Duane Arnold	Palo	IA	Iowa Electric Light & Power Co.	BWR	GE	Bechtel	1658	515	MDC
Farley 1	Dothan	AL	Alabama Power Co.	PWR	W	Bechtel	2652	813	MDC
Farley 2	Dothan	AL	Alabama Power Co.	PWR	W	Bechtel	2652	823	MDC
Fermi 2	Newport	MI	Detroit Edison Co.	BWR	GE	Detroit Edison	3292	1093	MDC
Fitzpatrick	Scriba	NY	New York Power Authority	BWR	GE	Stone & Webster	2436	778	MDC
Fort Calhoun 1	Fort Calhoun	NE	Omaha Public Power District	PWR	C-E	Gibbs & Hill	1500	478	MDC
Fort St. Vrain	Platteville	CO	Public Services Company of Colorado	HTGR	GA	Sargent & Lundy	842	330	MDC
Ginna	Ontario	NY	Rochester Gas & Electric Corp.	PWR	W	Gilbert	1520	470	MDC
Grand Gulf 1	Port Gibson	MS	System Energy Resources, Inc.	BWR	GE	Bechtel	3833	1142	MDC
Grand Gulf 2	Port Gibson	MS	System Energy Resources, Inc.	BWR	GE	Bechtel	3833	1250	DER
Haddam Neck	Haddam Neck	CT	Connecticut Yankee Atomic Power Co.	PWR	W	Stone & Webster	1825	569	MDC
Hatch 1	Baxley	GA	Georgia Power Co.	BWR	GE	SCS / Bechtel	2436	756	MDC
Hatch 2	Baxley	GA	Georgia Power Co.	BWR	GE	SCS / Bechtel	2436	768	MDC
Hope Creek 1	Salem	NJ	Public Services Electric & Gas Co.	BWR	GE	Bechtel	3293	1067	MDC
Indian Point 2	Indian Point	NY	Consolidated Edison Co.	PWR	W	UE & C	2758	849	MDC

Table 2-1. General Plant Data - Sorted by Plant Name (Continued)

Reactor Plant	City	State	Utility	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Indian Point 3	Indian Point	NY	New York Power Authority	PWR	W	UE & C	3025	965	MDC
Kewaunee	Carlton	WI	Wisconsin Public Service Corp.	PWR	W	Pioneer	1650	503	MDC
LaSalle 1	Seneca	IL	Commonwealth Edison Co.	BWR	GE	Sargent & Lundy	3323	1036	MDC
LaSalle 2	Seneca	IL	Commonwealth Edison Co.	BWR	GE	Sargent & Lundy	3323	1036	MDC
Limerick 1	Pottstown	PA	Philadelphia Power & Light Co.	BWR	GE	Bechtel	3293	1055	MDC
Limerick 2	Pottstown	PA	Philadelphia Power & Light Co.	BWR	GE	Bechtel	3293	1065	DER
Maine Yankee	Wiscasset	ME	Maine Yankee Atomic Power Co.	PWR	C-E	Stone & Webster	2630	810	MDC
McGuire 1	Cornelius	NC	Duke Power Co.	PWR	W	Duke Power Co.	3411	1129	MDC
McGuire 2	Cornelius	NC	Duke Power Co.	PWR	W	Duke Power Co.	3411	1129	MDC
Millstone 1	Waterford	CT	Northeast Utilities	BWR	GE	Ebasco	2011	654	MDC
Millstone 2	Waterford	CT	Northeast Utilities	PWR	C-E	Bechtel	2700	863	MDC
Millstone 3	Waterford	CT	Northeast Utilities	PWR	W	Stone & Webster	3411	1142	MDC
Monticello	Monticello	MN	Northern States Power Co.	BWR	GE	Bechtel	1670	536	MDC
Nine Mile Point 1	Scriba	NY	Niagara Mowhawk Power Co.	BWR	GE	Niagara Mohawk	1850	610	MDC
Nine Mile Point 2	Scriba	NY	Niagara Mowhawk Power Co.	BWR	GE	Stone & Webster	3323	[illegible]80	MDC
North Anna 1	Mineral	VA	Virginia Power Co.	PWR	W	Stone & Webster	2893	915	MDC
North Anna 2	Mineral	VA	Virginia Power Co.	PWR	W	Stone & Webster	2893	915	MDC
Oconee 1	Seneca	SC	Duke Power Co.	PWR	B&W	Duke/ Bechtel	2568	846	MDC
Oconee 2	Seneca	SC	Duke Power Co.	PWR	B&W	Duke/ Bechtel	2568	846	MDC
Oconee 3	Seneca	SC	Duke Power Co.	PWR	B&W	Duke/ Bechtel	2568	846	MDC
Oyster Creek 1	Forked River	NJ	GPU Nuclear Corp.	BWR	GE	Burns & Roe/GE	1930	620	MDC
Palisades	South Haven	MI	Consumers Power Co.	PWR	C-E	Bechtel	2530	730	MDC
Palo Verde 1	Wintersburg	AZ	Arizona Public Service Co.	PWR	C-E	Bechtel	3800	1221	MDC
Palo Verde 2	Wintersburg	AZ	Arizona Public Service Co.	PWR	C-E	Bechtel	3800	1221	MDC

Table 2-1. General Plant Data - Sorted by Plant Name (Continued)

Reactor Plant	City	State	Utility	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Palo Verde 3	Wintersburg	AZ	Arizona Public Service Co.	PWR	C-E	Bechtel	3800	1221	MDC
Peach Bottom 2	Peach Bottom	PA	Philadelphia Power & Light Co.	BWR	GE	Bechtel	3293	1051	MDC
Peach Bottom 3	Peach Bottom	PA	Philadelphia Power & Light Co.	BWR	GE	Bechtel	3293	1035	MDC
Perry 1	North Perry	OH	The Cleveland Electric Illuminating Co.	BWR	GE	Gilbert	3579	1205	MDC
Perry 2	North Perry	OH	Cleveland Electric Illuminating Co.	BWR	GE	Gilbert	3579	1205	DER
Pilgrim 1	Plymouth	MA	Boston Edison Co.	BWR	GE	Bechtel	1998	670	MDC
Point Beach 1	Two Creeks	WI	Wisconsin Electric Power Co.	PWR	W	Bechtel	1518	485	MDC
Point Beach 2	Two Creeks	WI	Wisconsin Electric Power Co.	PWR	W	Bechtel	1518	485	MDC
Prairie Island 1	Red Wing	MN	Northern States Power Co.	PWR	W	Pioneer	1650	503	MDC
Prairie Island 2	Red Wing	MN	Northern States Power Co.	PWR	W	Pioneer	1650	503	MDC
Quad Cities 1	Cordova	IL	Commonwealth Edison/Iowa-Illinois Gas & Electric	BWR	GE	Sargent & Lundy	2511	769	MDC
Quad Cities 2	Cordova	IL	Commonwealth Edison/Iowa-Illinois Gas & Electric	BWR	GE	Sargent & Lundy	2511	769	MDC
Rancho Seco	Clay Station	CA	Sacramento Municipal Utility District	PWR	B&W	Bechtel	2772	873	MDC
River Bend 1	St. Francisville	LA	Gulf States Utilities Co.	BWR	GE	Stone & Webster	2894	936	MDC
Robinson 2	Hartsville	SC	Carolina Power & Light Co.	PWR	W	Ebasco	2300	665	MDC
Salem 1	Salem	NJ	Public Services Electric & Gas Co.	PWR	W	Pacific Gas & Electric	3411	1106	MDC
Salem 2	Salem	NJ	Public Services Electric & Gas Co.	PWR	W	Pacific Gas & Electric	3411	1106	MDC
San Onofre 1	San Clemente	CA	Southern California Edison/San Diego Gas & Electric	PWR	W	Bechtel	1347	436	MDC
San Onofre 2	San Clemente	CA	Southern California Edison/San Diego Gas & Electric	PWR	C-E	Bechtel	3390	1070	MDC
San Onofre 3	San Clemente	CA	Southern California Edison/San Diego Gas & Electric	PWR	C-E	Bechtel	3390	1080	MDC
Seabrook 1	Seabrook	NH	New Hampshire Yankee	PWR	W	UE & C	3411	1150	MDC
Sequoyah 1	Soddy-Daisy	TN	Tennessee Valley Authority	PWR	W	TVA	3411	1148	MDC
Sequoyah 2	Soddy-Daisy	TN	Tennessee Valley Authority	PWR	W	TVA	3411	1148	MDC
Shearon Harris 1	New Hill	NC	Carolina Power & Light Co	PWR	W	Ebasco	2775	860	MDC

Table 2-1. General Plant Data - Sorted by Plant Name (Continued)

Reactor Plant	City	State	Utility	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Shoreham	Brookhaven	NY	Long Island Lighting Co.	BWR	GE	Stone & Webster	2436	820	DER
South Texas 1	Palacios	TX	Houston Lighting & Power Co.	PWR	W	Brown & Root	3800	1250	MDC
South Texas 2	Palacios	TX	Houston Lighting & Power Co.	PWR	W	Brown & Root	3800	1250	DER
St. Lucie 1	Hutchinson Island	FL	Florida Power & Light Co.	PWR	C-E	Ebasco	2700	839	MDC
St. Lucie 2	Hutchinson Island	FL	Florida Power & Light Co.	PWR	C-E	Ebasco	2700	839	MDC
Summer	Parr	SC	South Carolina Electric & Gas Co.	PWR	W	Gilbert	2775	885	MDC
Surry 1	Gravel Neck	VA	Virginia Power Co.	PWR	W	Stone & Webster	2441	781	MDC
Surry 2	Gravel Neck	VA	Virginia Power Co.	PWR	W	Stone & Webster	2441	781	MDC
Susquehanna 1	Berwick	PA	Pennsylvania Power & Light Co.	BWR	GE	Bechtel	3293	1032	MDC
Susquehanna 2	Berwick	PA	Pennsylvania Power & Light Co.	BWR	GE	Bechtel	3293	1032	MDC
Three Mile Island 1	Londonderry Twp	PA	GPU Nuclear Corp.	PWR	B&W	Gilbert	2535	776	MDC
Trojan	Prescott	OR	Portland General Electric Co.	PWR	W	Bechtel	3411	1095	MDC
Turkey Point 3	Florida City	FL	Florida Power & Light Co.	PWR	W	Bechtel	2200	666	MDC
Turkey Point 4	Florida City	FL	Florida Power & Light Co.	PWR	W	Bechtel	2200	666	MDC
Vermont Yankee	Vernon	VT	Vermont Yankee Nuclear Power Corp.	BWR	GE	Ebasco	1593	504	MDC
Vogtle 1	Waynesboro	GA	Georgia Power Co.	PWR	W	Bechtel	3411	1079	MDC
Vogtle 2	Waynesboro	GA	Georgia Power Co.	PWR	W	Bechtel	3411	1079	MDC
Waterford 3	Taft	LA	Louisiana Power & Light Co.	PWR	C-E	Ebasco	3390	1075	MDC
Watts Bar 1	Spring City	TN	Tennessee Valley Authority	PWR	W	TVA	3411	1165	DER
Watts Bar 2	Spring City	TN	Tennessee Valley Authority	PWR	W	TVA	3411	1165	DER
WNP-1	Richland	WA	Washington Public Power Supply System	PWR	B&W	UE & C	3760	1266	DER
WNP-2	Richland	WA	Washington Public Power Supply System	BWR	GE	Burns & Roe	3323	1095	MDC
WNP-3	Satsop	WA	Washington Public Power Supply System	PWR	C-E	Ebasco	3800	1242	DER
Wolf Creek	Burlington	KS	Wolf Creek Nuclear Operating Corp.	PWR	W	Bechtel	3411	1128	MDC

Table 2-1. General Plant Data - Sorted by Plant Name (Continued)

Reactor Plant	City	State	Utility	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Yankee Rowe	Rowe	MA	Yankee Atomic Electric Co.	PWR	W	Stone & Webster	600	167	MDC
Zion 1	Zion	IL	Commonwealth Edison Co.	PWR	W	Sargent & Lundy	3250	1040	MDC
Zion 2	Zion	IL	Commonwealth Edison Co.	PWR	W	Sargent & Lundy	3250	1040	MDC

Notes: MDC - Maximum Dependable Capacity
DER - Design Electric Rating

Table 2-4. General Reactor Site Data

Plant Name	Location	Water Source	Ult. Heat Sink	SSE		Tornado Wind Speed (MPH)
				Horiz. G's	Vert. G's	
ANO-1	Russelville, Arkansas	Dardanelle Resevoir	Same	0.2	0.133	360
ANO-2	Russelville, Arkansas	Dardanelle Resevoir	Nat. Cooling tower	0.2	0.133	360
Beaver Valley 1 & 2	25 Mi. NW Pittsburgh, Pa	Ohio River	Nat. Cooling Towers	0.12	0.08	360
Bellefonte 1 & 2	7 Mi. ENE Scotsboro, Ala	Guntersville Resevoir	Nat. Cooling Towers	0.18	0.12	360
Big Rock Point	4 Mi. NE Charlevoix, Mi	Lake Michigan	Same	0.05	0.05	210
Braidwood 1 & 2	2 Mi. S Braidwood, Ill.	Kanakee River	Braidwood Lake	0.2	0.133	360
Browns Ferry 1, 2, & 3	10 Mi. NW Decatur, Ala.	Tennessee River	River/Mech. Cooling Towers	0.2	0.133	300
Brunswick 1 & 2	19 Mi. S Wilmington, NC	Atlantic Ocean	Cape Fear River	0.16	0.107	360
Byron 1 & 2	4 Mi. S Bryon, Ill	Rock River	Nat. Cooling Towers	0.2	0.133	360
Callaway	10 Mi. SE Fulton, MO	Missouri River	Nat. Cooling Tower	0.2	0.133	360
Calvert Cliffs 1 & 2	40 Mi. S Annapolis, MD	Chesapeke Bay	Same	0.15	0.1	360
Catawba 1 & 2	19 Mi. SW Charlotte	Lake Wylie	Mech. Cooling Towers	0.15	0.1	360
Clinton 1	Harp Township, Ill.	Salt Creek (N. Fork)	Lake Clinton (Manmade)	0.25	0.25	360
Comanche Peak 1 & 2	40 Mi. SW Ft. Worth, TX	Squaw Creek Resevoir	Same	0.12	0.08	360
Cooper	23 Mi. S Nebraska City, NB	Missouri River	Same	0.2	0.133	360
Crystal River 3	7 Mi. NW Crystal River, Fla	Gulf Of Mexico	Same	0.1	0.067	360

Table 2-4. General Reactor Site Data (Continued)

Plant Name	Location	Water Source	Ult. Heat Sink	SSE		Tornado Wind Speed (MPH)
				Horiz. G's	Vert. G's	
D.C. Cook 1 & 2	10 Mi. S St.Joseph, Mi	Lake Michigan	Same	0.2	0.133	360
Davis Besse	21 Mi. E Toledo, OH	Lake Erie	Nat. Cooling Tower	0.15	0.1	360
Diablo Canyon 1 & 2	12 Mi. W San Luis Obispo, CA	Pacific Ocean	Same	0.75	0.5	200
Dresden 2 & 3	9 Mi. E Morris, Ill.	Kanakee River	Cooling Lake	0.2	0.133	200
Duane Arnold	8 Mi. NW Cedar Rapids, IO	Cedar River	Mech. Cooling Towers	0.12	0.096	360
Farley 1 & 2	16 Mi. E Dothan, Ala.	Woodruff Resevoir	Mech. Cooling Towers	0.1	0.067	360
Fermi 2	30 Mi. SW Detroit, MI	Lake Erie	Nat. Cooling Towers	0.15	0.1	360
Fitzpatrick	6 Mi. NE Oswego, NY	Lake Ontario	Same	0.15	0.1	360
Fort Calhoun 1	19 Mi. N Omaha, NB	Missouri River	Same	0.17	0.113	360
Ginna	15 Mi. NE Rochester, NY	Lake Ontario	Same	0.2	0.133	132
Grand Gulf 1 & 2	25 Mi. S Vicksburg	Mississippi River	Nat. Cooling Towers	0.15	0.1	360
Haddam Neck	13 Mi. E Meriden, CT	Connecticut River	Same	0.15	0.1	360
Hatch 1 & 2	11 Mi. N Baxley, GA	Altahama River	Mech. Cooling Towers	0.15	0.1	360
Hope Creek 1 & 2	8 Mi. SW Salem, NJ	Delaware River	Nat. Cooling Towers	0.2	0.2	360
Indian Point 2 & 3	25 Mi. N New York City, NY	Hudson River	Same	0.15	0.1	300
Kewaunee	20 Mi. N Manitowoc, Wi	Lake Michigan	Same	0.12	0.08	360

Table 2-4. General Reactor Site Data (Continued)

Plant Name	Location	Water Source	Ult. Heat Sink	SSE Horiz. G's	SSE Vert. G's	Tornado Wind Speed (MPH)
LaSalle 1 & 2	12 Mi.. W Morris, Ill.	Illinois River	2058 Acre Cooling Lake	0.2	0.133	360
Limerick 1 & 2	30 Mi. NW Philadelphia, Pa	Schuylkill River	Nat. Cooling Towers	0.15	0.1	360
Maine Yankee	Wicasset, Maine	Black River	Montsweag Bay	0.1	0.067	360
McGuire 1 & 2	17 Mi. NW Charlotte, NC	Lake Norman	Same	0.15	0.1	360
Millstone 1, 2, & 3	5 Mi. SW New London, CT	Long Island Sound	Niantic Bay	0.17	0.113	300
Monticello	30 Mi. NW Minneapolis, MN	Mississippi River	Mech. Cooling Towers	0.12	0.08	360
Nine Mile Point 1	8 Mi. NE Oswego, NY	Lake Ontario	Same	0.11	0.055	360
Nine Mile Point 2	8 Mi. NE Oswego, NY	Lake Ontario	Same	0.15	0.1	360
North Anna 1 & 2	40 Mi. NW Richmond, VA	North Anna Resevoir	Cooling Pond	0.12	0.08	360
Oconee 1, 2, & 3	30 Mi. W Greenville, SC	Lake Keowee	Same	0.1	0.067	360
Oyster Creek 1	9 Mi. S Toms River, NJ	Atlantic Ocean	Barnegat Bay	0.17	0.113	360
Palisades	35 Mi. W Kalamazoo, MI	Lake Michigan	Mech. Cooling Towers	0.2	0.113	360
Palo Verde 1, 2, & 3	2 Mi. S Wintersberg, AZ	Domestic Water	Mech. Cooling Towers	0.27	0.18	300
Peach Bottom 2 & 3	19 Mi. S Lancaster, PA	Susquehana River	River/Mech. Cooling Towers	0.12	0.08	360
Perry 1 & 2	37 Mi. E Cleveland, OH	Lake Erie	Nat. Cooling Towers	0.15	0.1	360
Pilgrim 1	35 Mi. SE Boston, MA	Cape Cod Bay	Same	0.15	0.1	300

Table 2-4. General Reactor Site Data (Continued)

Plant Name	Location	Water Source	Ult. Heat Sink	SSE		Tornado Wind Speed (MPH)
				Horiz. G's	Vert. G's	
Point Beach 1 & 2	15 Mi. N Manitowoc, WI	Lake Michigan	Same	0.12	0 08	360
Praine Island 1 & 2	40 Mi. SE Minneapolis, MN	Mississippi River	Mech. Cooling Towers	0.12	0.08	360
Quad Cities 1 & 2	20 Mi. NE Moline, Ill.	Mississippi River	Same	0.24	0.16	200
Rancho Seco	25 Mi. SE Sacramento, CA	Folsom South Canal	Nat. Cooling Towers	0.25	0.167	175
River Bend 1	25 Mi. N Batton Rouge, LA	Mississippi River	Mech. Cooling Towers	0.1	0.1	360
Robinson 2	6 Mi. NW Hartsville, SC	Lake Robinson	Water Discharge Tunnel	0.2	0.133	300
Salem 1 & 2	8 Mi. SW Salem, NJ	Delaware River	Same	0.2	0.133	360
San Onofre 1	5 Mi. S San Clemente, CA	Pacific Ocean	Same	0.67	0.44	75
San Onofre 2 & 3	5 Mi. S San Clemente, CA	Pacific Ocean	Same	0.67	0.44	260
Seabrook 1	Seabrook, NH	Atlantic Ocean	Same	0.2	0.133	360
Sequoyah 1 & 2	18 Mi. NE Chattanooga, TN	Tennessee River	River/Nat. Cooling Tower	0.18	0.12	360
Shearon Harris 1	20 Mi. SW Raleigh, NC	Cape Fear River	Nat. Cooling Tower	0.15	0.1	350
Shoreham	12 Mi. NW Riverhead, NY	Long Island Sound	Same	0.2	0.133	360
South Texas 1 & 2	12 Mi. SW Bay City, TX	Colorado River	7000 Acre Cooling Pond	0.1	0.067	360
St. Lucie 1 & 2	12 Mi. SE Ft.Pierce, FL	N/A	N/A	0.1	0.067	360
Summer	26 Mi. NW Columbia, SC	Lake Monticello	Same	0.15	0.1	360
Surry 1 & 2	8 Mi. S Williamsburg, VA	James River	Same	0.15	0.1	360

Table 2-4. General Reactor Site Data (Continued)

Plant Name	Location	Water Source	Ult. Heat Sink	SSE		Tornado Wind Speed (MPH)
				Horiz. G's	Vert. G's	
Susquehanna 1 & 2	7 Mi. NE Berwick, PA	Susquehanna River	Nat. Cooling Towers	0.1	0.067	360
TMI-1	10 Mi. SE Harrisburg, PA	Susquehanna River	Nat. & Mech. Cooling Towers	0.12	0.08	360
Trojan	30 Mi. NW Portland, OR	Columbia River	Nat. Cooling Tower	0.25	0.167	200
Turkey Point 3 & 4	25 Mi. S Miami, FL	Biscayne Bay	Canals	0.15	0.1	225
Vermont Yankee	5 Mi. S Battleboro, VT	Connecticut River	River/Mech. Cooling Towers	0.14	0.093	360
Vogtle 1 & 2	39 Mi. SE Augusta, GA	Savannah River	Nat. Cooling Towers	0.2	0.133	360
Waterford 3	Taft, LA	Mississippi River	Same	0.1	0.067	360
Watts Bar 1 & 2	8 Mi. E Spring City, TN	Chickamunga Lake	Nat. Cooling Towers	0.18	0.12	360
WNP-1	Hanford, WA	Columbia River	Mech. Cooling Towers	0.25	0.167	360
WNP-2	Hanford, WA	Columbia River	Mech. Cooling Towers	0.25	0.167	360
WNP-3	Satsop, WA	N/A	Nat. Cooling Tower	0.25	0.167	360
Wolf Creek	4 Mi. NE Burlington, KA	Wolf Creek Cooling Lake	6000 Acre Cooling Lake	0.12	0.08	360
Yankee Rowe	20 Mi. NW Greengield, MA	Sherman Pond	Same	0.1	0.067	110
Zion 1 & 2	6 Mi. N Waukegan, Ill.	Lake Michigan	Same	0.17	0.113	360

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name

NRC Dockett # (50- - -)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
313	ANO-1	PWR	B&W	IV	12/6/68	5/21/74	2568	12/6/08	Yes	12/19/74	
368	ANO-2	PWR	C-E	IV	12/6/72	12/14/78	2815	12/6/12	Yes	4/26/80	
334	Beaver Valley 1	PWR	W	I	6/26/70	7/2/76	2652	1/29/16	Yes	10/1/76	
412	Beaver Valley 2	PWR	W	I	5/3/74	8/14/87	2652	5/27/27	Yes	11/17/87	
438	Bellefonte 1	PWR	B&W	II	12-24-74	N/A	0	N/A	Indefinite	N/A	1
439	Bellefonte 2	PWR	B&W	II	12/24/74	N/A	0	N/A	Indefinite	N/A	1
210	Big Rock Point	BWR	GE	III	5/31/60	5/1/64	240	5/31/00	Yes	3/29/63	
456	Braidwood 1	PWR	W	III	12/31/75	7/2/87	3411	10/17/26	Yes	7/29/88	
457	Braidwood 2	PWR	W	III	12/31/75	5/20/88	3411	12/18/27	Yes	10/17/88	
259	Browns Ferry 1	BWR	GE	II	5/10/67	12/20/73	3293	5/10/07	Yes	8/1/74	
260	Browns Ferry 2	BWR	GE	II	5/10/67	8/2/74	3293	5/10/07	Yes	3/1/75	
296	Browns Ferry 3	BWR	GE	II	7/31/68	8/18/76	3293	7/21/08	Yes	3/1/77	
325	Brunswick 1	BWR	GE	II	2/7/70	11/12/76	2436	2/7/10	Yes	3/18/77	
324	Brunswick 2	BWR	GE	II	2/7/70	12/27/74	2436	12/6/10	Yes	11/3/75	
454	Byron 1	PWR	W	III	12/31/75	2/14/85	3411	10/31/24	Yes	9/16/85	
455	Byron 2	PWR	W	III	12/31/75	1/30/87	3411	11/6/26	Yes	8/21/87	
483	Callaway	PWR	W	III	4/16/76	10/18/84	3565	10/18/24	Yes	12/19/84	
317	Calvert Cliffs 1	PWR	C-E	I	7/7/69	7/31/74	2700	7/31/14	Yes	5/8/75	
318	Calvert Cliffs 2	PWR	C-E	I	7/7/69	11/30/76	2700	8/31/16	Yes	4/1/77	

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name (Continued)

NRC Dockett # (50-___)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
413	Catawba 1	PWR	W	II	8/7/75	1/17/85	3411	12/6/24	Yes	6/29/85	
414	Catawba 2	PWR	W	II	8/7/75	5/15/86	3411	2/24/26	Yes	8/19/86	
461	Clinton 1	BWR	GE	III	2/24/76	4/17/87	2894	9/29/26	Yes	11/24/87	
445	Comanche Peak 1	PWR	W	IV	12/19/74	N/A	0	N/A	Expected '95	N/A	5
446	Comanche Peak 2	PWR	W	IV	12/19/74	N/A	0	N/A	Indefinite	N/A	3
298	Cooper	BWR	GE	IV	6/4/68	1/18/74	2381	6/4/08	Yes	7/1/74	
302	Crystal River 3	PWR	B&W	III	9/25/68	1/28/77	2544	12/3/16	Yes	3/13/77	
315	D.C. Cook 1	PWR	W	III	3/25/69	10/25/74	3250	3/25/09	Yes	8/28/75	
316	D.C. Cook 2	PWR	W	III	3/25/69	12/23/77	3411	N/A	Yes	7/1/78	
346	Davis-Besse	PWR	B&W	III	3/24/71	4/22/77	2772	3/24/11	Yes	7/31/78	
275	Diablo Canyon 1	PWR	W	V	4/23/68	11/2/84	3338	4/23/08	Yes	5/7/85	
323	Diablo Canyon 2	PWR	W	V	12/9/70	8/26/85	3411	12/9/10	Yes	3/13/86	
237	Dresden 2	BWR	GE	III	1/10/66	12/22/69	2527	12/22/72	Yes	6/9/70	4
249	Dresden 3	BWR	GE	III	10/14/66	3/2/71	2527	10/14/06	Yes	11/16/71	
331	Duane Arnold	BWR	GE	III	6/22/70	2/22/74	1658	6/21/10	Yes	2/1/75	
348	Farley 1	PWR	W	II	8/16/72	6/25/77	2652	8/16/12	Yes	12/1/77	
364	Farley 2	PWR	W	II	8/16/72	3/31/81	2652	8/16/12	Yes	7/30/81	
341	Fermi 2	BWR	GE	III	9/26/72	7/15/85	3292	3/20/25	Yes	1/23/88	
333	Fitzpatrick	BWR	GE	I	5/20/70	10/17/74	2436	5/20/10	Yes	7/28/75	

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name (Continued)

NRC Dockett # (50-)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
285	Fort Calhoun 1	PWR	C-E	IV	6/7/68	8/9/73	1500	6/7/08	Yes	6/20/74	
267	Fort St. Vrain	HTGR	GA	IV	9/17/68	12/21/73	842	9/17/08	Yes	7/1/79	
244	Ginna	PWR	W	I	4/25/66	12/10/84	1520	4/25/06	Yes	7/1/70	
416	Grand Gulf 1	BWR	GE	II	9/4/74	11/1/84	3833	6/16/22	Yes	7/1/85	
417	Grand Gulf 2	BWR	GE	II	9/4/74	N/A	0	N/A	Indefinite	N/A	1
213	Haddam Neck	PWR	W	I	5/26/64	12/27/74	1825	5/26/04	Yes	1/1/68	
321	Hatch 1	BWR	GE	II	9/30/69	10/13/74	2436	9/30/09	Yes	12/31/75	
366	Hatch 2	BWR	GE	II	12/27/72	6/13/78	2436	12/27/12	Yes	9/5/79	
354	Hope Creek 1	BWR	GE	I	11/4/74	7/25/86	3293	4/11/26	Yes	12/20/86	
247	Indian Point 2	PWR	W	I	10/14/66	9/28/73	2758	9/28/13	Yes	8/1/74	
286	Indian Point 3	PWR	W	I	8/13/69	4/5/76	3025	8/13/09	Yes	8/30/76	
305	Kewaunee	PWR	W	III	8/6/68	12/21/73	1650	8/6/08	Yes	6/16/74	
373	LaSalle 1	BWR	GE	III	9/10/73	8/13/82	3323	5/17/22	Yes	1/1/84	
374	LaSalle 2	BWR	GE	III	9/10/73	3/23/84	3323	12/16/23	Yes	10/19/84	
352	Limerick 1	BWR	GE	I	6/19/74	8/8/85	3293	10/26/24	Yes	2/1/86	
353	Limerick 2	BWR	GE	I	6/19/74				Yes	1/8/90	
309	Maine Yankee	PWR	C-E	I	10/21/68	6/29/73	2630	10/21/08	Yes	12/28/72	
369	McGuire 1	PWR	W	II	2/23/73	7/8/81	3411	6/12/21	Yes	12/1/81	
370	McGuire 2	PWR	W	II	2/23/73	5/27/83	3411	3/3/23	Yes	3/1/84	

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name (Continued)

NRC Dockett # (50-___)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
245	Millstone 1	BWR	GE	I	5/19/66	10/31/86	2011	5/19/06	Yes	3/1/71	
326	Millstone 2	PWR	C-E	I	12/11/70	9/30/75	2700	7/31/15	Yes	12/26/75	
423	Millstone 3	PWR	W	I	8/9/74	1/31/86	3411	11/25/25	Yes	4/23/86	
263	Monticello	BWR	GE	III	6/19/67	1/9/81	1670	9/8/10	Yes	6/30/71	
220	Nine Mile Point 1	BWR	GE	I	4/12/65	12/26/74	1850	4/11/05	Yes	12/1/69	
410	Nine Mile Point 2	BWR	GE	I	6/24/74	7/2/87	3323	10/31/26	Yes	4/5/88	
338	North Anna 1	PWR	W	II	2/19/71	4/1/78	2893	4/1/18	Yes	6/6/78	
339	North Anna 2	PWR	W	II	2/19/71	8/21/80	2893	8/21/20	Yes	12/14/80	
269	Oconee 1	PWR	B&W	II	11/6/67	2/6/73	2568	2/6/13	Yes	7/15/73	
270	Oconee 2	PWR	B&W	II	11/6/67	10/6/73	2568	10/6/13	Yes	9/9/74	
287	Oconee 3	PWR	B&W	II	11/6/67	7/19/74	2568	7/19/14	Yes	12/16/74	
219	Oyster Creek 1	BWR	GE	I	12/15/64	8/1/69	1930	4/9/72	Yes	12/1/69	4
255	Palisades	PWR	C-E	III	3/14/67	10/16/72	2530	3/1/74	Yes	12/31/71	4
528	Palo Verde 1	PWR	C-E	V	5/25/76	6/1/85	3800	12/31/24	Yes	1/28/86	
529	Palo Verde 2	PWR	C-E	V	5/25/76	4/24/86	3800	12/9/25	Yes	9/19/86	
530	Palo Verde 3	PWR	C-E	V	5/25/76	11/25/87	3800	3/25/27	Yes	1/8/88	
277	Peach Bottom 2	BWR	GE	I	1/31/68	7/2/74	3293	1/31/08	Yes	12/23/74	
278	Peach Bottom 3	BWR	GE	I	1/31/68	7/2/74	3293	1/31/08	Yes	12/23/74	
440	Perry 1	BWR	GE	III	5/3/77	11/13/86	3579	3/18/26	Yes	11/18/87	

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name (Continued)

NRC Dockett # (50-___)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
441	Perry 2	BWR	GE	III	5/3/77	N/A	0	N/A	Indefinite	N/A	1
293	Pilgrim 1	BWR	GE	I	8/26/68	9/15/72	1998	8/26/08	Yes	12/1/72	
266	Point Beach 1	PWR	W	III	7/19/67	10/5/70	1518	10/5/10	Yes	12/21/70	
301	Point Beach 2	PWR	W	III	7/25/68	3/8/73	1518	3/8/13	Yes	10/1/72	
282	Prairie Island 1	PWR	W	III	6/25/68	4/5/74	1650	8/9/13	Yes	12/16/73	
306	Prairie Island 2	PWR	W	III	6/25/68	10/29/74	1650	10/29/14	Yes	12/21/74	
254	Quad Cities 1	BWR	GE	III	2/15/67	12/14/72	2511	2/15/07	Yes	2/18/73	
265	Quad Cities 2	BWR	GE	III	2/15/67	12/14/72	2511	2/15/07	Yes	3/10/73	
312	Rancho Seco	PWR	B&W	V	10/11/68	8/16/74	2772	10/11/08	Yes	4/17/75	
458	River Bend 1	BWR	GE	IV	3/25/77	11/20/85	2894	8/29/25	Yes	6/16/86	
261	Robinson 2	PWR	W	II	4/13/67	9/23/70	2300	4/13/07	Yes	3/7/71	
272	Salem 1	PWR	W	I	9/25/68	12/1/76	3411	9/25/08	Yes	6/30/77	
311	Salem 2	PWR	W	I	9/25/68	5/20/81	3411	9/25/08	Yes	10/13/81	
206	San Onofre 1	PWR	W	V	3/2/64	3/27/67	1347	9/27/2072	Yes	1/1/68	4
361	San Onofre 2	PWR	C-E	V	10/18/73	9/7/82	3390	10/18/13	Yes	8/8/83	
362	San Onofre 3	PWR	C-E	V	10/18/73	9/16/83	3390	10/18/13	Yes	4/1/84	
443	Seabrook 1	PWR	W	I	7/7/76	10/17/86	3411	10/17/26	No	N/A	5
327	Sequoyah 1	PWR	W	II	5/27/70	9/17/80	3411	5/27/10	Yes	7/1/81	
328	Sequoyah 2	PWR	W	II	5/27/70	9/15/81	3411	5/27/10	Yes	6/1/82	

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name (Continued)

NRC Dockett # (50- - -)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
400	Shearon Harris 1	PWR	W	II	1/27/78	1/12/87	2775	10/24/26	Yes	5/2/87	
322	Shoreham	BWR	GE	I	4/14/73	7/3/85	121	4/13/13	No	N/A	5
498	South Texas 1	PWR	W	IV	12/22/75	3/22/88	3800	8/20/27	Yes	8/25/88	
499	South Texas 2	PWR	W	IV	12/22/75	12/16/88	190	12/16/28			5
335	St. Lucie 1	PWR	C-E	II	7/1/70	3/1/76	2700	3/1/16	Yes	12/21/76	
389	St. Lucie 2	PWR	C-E	II	5/2/77	6/10/83	2700	4/6/23	Yes	8/8/83	
395	Summer	PWR	W	II	3/21/73	11/12/82	2775	3/21/23	Yes	1/1/84	
280	Surry 1	PWR	W	II	6/25/68	5/25/72	2441	5/25/12	Yes	12/22/72	
281	Surry 2	PWR	W	II	6/25/68	1/29/73	2441	1/29/13	Yes	5/1/73	
387	Susquehanna 1	BWR	GE	I	11/2/73	11/12/82	3293	7/17/22	Yes	6/8/83	
388	Susquehanna 2	BWR	GE	I	11/2/73	6/27/84	3293	3/23/24	Yes	2/12/85	
289	Three Mile Island 1	PWR	B&W	I	5/18/68	4/19/74	2535	5/18/08	Yes	9/2/74	
344	Trojan	PWR	W	V	2/8/71	11/21/75	3411	2/8/11	Yes	5/20/76	
250	Turkey Point 3	PWR	W	II	4/27/67	7/19/72	2200	4/27/07	Yes	12/14/72	
251	Turkey Point 4	PWR	W	II	4/27/67	4/10/73	2200	4/27/07	Yes	9/7/73	
271	Vermont Yankee	BWR	GE	I	12/11/67	2/28/73	1593	12/11/07	Yes	11/30/72	
424	Vogtle 1	PWR	W	II	6/28/74	3/16/87	3411	1/16/27	Yes	6/1/87	
425	Vogtle 2	PWR	W	II	6/28/74	N/A	0	N/A	Expected '95	N/A	2
382	Waterford 3	PWR	C-E	IV	11/14/74	3/16/85	3390	12/18/24	Yes	9/24/85	

Table 2-5. Summary of General Licensing Data - Sorted by Plant Name (Continued)

NRC Dockett # (50-___)	Reactor Plant	Reactor Type	NSSS	NRC Region	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
390	Watts Bar 1	PWR	W	II	1/23/73	N/A	0	N/A	Expected '95	N/A	2
391	Watts Bar 2	PWR	W	II	1/23/73	N/A	0	N/A	Expected '95	N/A	2
460	WNP-1	PWR	B&W	V	12/24/75	N/A	0	N/A	Indefinite	N/A	1
397	WNP-2	BWR	GE	V	3/19/73	4/13/84	3323	12/20/23	Yes	12/13/84	
508	WNP-3	PWR	C-E	V	4/11/78	N/A	0	N/A	Indefinite	N/A	1
482	Wolf Creek	PWR	W	IV	5/31/77	6/4/85	3411	3/11/25	Yes	9/3/85	
29	Yankee Rowe	PWR	W	I	11/4/57	12/24/63	600	7/9/00	Yes	7/1/61	
295	Zion 1	PWR	W	III	12/26/68	10/19/73	3250	12/26/08	Yes	12/31/73	
304	Zion 2	PWR	W	III	12/26/68	11/14/73	3250	12/26/08	Yes	9/17/94	

NOTES:

1. Construction halted
2. Under active construction
3. Construction deferred
4. License not expired under 10 CFR 2.109
5. Low power license
6. May be decommissioned

3. PRESSURIZED WATER REACTOR (PWR) SYSTEM OVERVIEW

Reactor plant systems may be broadly classified as safety-related or as non-safety-related. Light water reactor (LWR) safety-related systems typically are considered to be those that are required to perform any of the following safety functions:

- Control reactivity
- Provide reactor core cooling and heat removal from the primary system
- Maintain reactor coolant system integrity
- Maintain containment integrity
- Control radioactive releases

In order to ensure the performance of these "front-line" safety functions, additional safety-related systems are required to perform the following support functions:

- Provide adequate motive power (i.e. electric, pneumatic or hydraulic motive power, direct steam turbine or diesel engine drive)
- Provide adequate control and instrumentation power (i.e., AC or DC electrical control power)
- Provide adequate cooling of safety-related equipment (i.e., cooling water, room air cooling)
- Provide other support functions needed by front-line or support systems to establish and maintain a safe shutdown condition

In their present form, the Nuclear Power Plant System Sourcebook series focuses on front-line safety systems and on electric power and cooling water support systems.

In this section, an overview of PWR systems is provided, focusing on basic system functions and interfaces. In Sections 4 to 6, more detailed comparative information is presented on the different product lines of the three U.S. commercial PWR Nuclear System Supply System (NSSS) vendors: Westinghouse, Combustion Engineering, and Babcock & Wilcox. Comparative data summaries for PWR systems are found in Sections 4.5, 5.5, and 6.5 (for individual PWR vendors) and in Section 7 (compilation for all PWRs). The reader should refer to the available Nuclear Power Plant System Sourcebooks identified in Section 1 for summary information on safety systems at specific nuclear power plants.

3.1 Introduction to the Pressurized Water Reactor

The PWR reactor coolant system (RCS) transports heat generated in a low-enrichment, light-water cooled and moderated core to the secondary coolant system via external primary coolant loops with steam generators. Control and removal of heat from the reactor and conversion of this heat into usable electrical power requires a broad spectrum of operating and auxiliary systems. Additionally, safety systems are required to ensure that postulated accidents at the PWR do not cause undue risk to the health and safety of the public. The spectrum of "generic" PWR systems is listed in Table 3.1-1. As indicated in this table, some systems normally are supplied by the Nuclear System Supply System (NSSS) vendor. The remaining systems, or the Balance-of-Plant (BOP), are supplied by the architect-engineer (A-E) who is responsible for the detailed integrated design of the plant.

3.2 PWR Primary System

The PWR NSSS is the primary system, or reactor coolant system (RCS), which consists of the reactor vessel and two to four external primary coolant loops, each containing one steam generator and one or two primary coolant pumps. The three U.S. PWR vendors have produced seven basic plant configurations, as summarized below:

Vendor	RCS Configuration	Number of Plants
Westinghouse	2-loop	6
	3-loop	14
	4-loop	35
Combustion Engineering	2-loop	14
	3-loop	1
Babcock & Wilcox	"lowered" 2-loop	7
	"raised" 2-loop	3
	Total PWRs	80

The three models of the Westinghouse NSSS are shown in Figure 3.2-1 (2-loop), Figure 3.2-2 (3-loop), and Figure 3.2-3 (4-loop). The Combustion Engineering 2-loop NSSS is shown in Figure 3.2-4. The two basic models of the Babcock & Wilcox NSSS are shown in Figure 3.2-5 ("lowered" 2-loop) and Figure 3.2-6 ("raised" 2-loop).

A pressurizer is connected to the "hot leg" of one of the primary coolant loops and serves to control primary system pressure by means of electric heaters (to increase the steam volume in the pressurizer and raise pressure) and spray (to condense the steam bubble in the pressurizer and lower pressure). RCS coolant inventory is measured by pressurizer water level, which is controlled by the combined letdown to and makeup from the Chemical and Volume Control System (CVCS).

3.3 Reactor Core and Fuel Assemblies

The PWR generates heat in a low-enrichment, light-water cooled and moderated core. All PWR fuel assemblies consist of a square array of fuel and burnable poison rods. The general fuel assembly configurations used by the three PWR vendors are listed below:

Vendor	Fuel Assembly Configuration	Application
Westinghouse	9 x (6 x 6)	Yankee-Rowe only
	14 x 14	2-loop plants and San Onofre 1
	15 x 15	Some 3-loop and 4-loop plants
	17 x 17	Most 3-loop and 4-loop plants, replacing the 15 x 15 fuel elements
Combustion Engineering	15 x 15	Palisades only
	14 x 14	Earlier plants
	16 x 16	Later plants, replacing 14 x 14 fuel elements in earlier plants
Babcock & Wilcox	15 x 15	All plants except Bellefonte
	17 x 17	Bellefonte only. Can replace 15 x 15 fuel elements in earlier
	plants	

The general trend is toward the denser arrays (i.e., 16 x 16, 17 x 17) which have greater surface area and hence lower linear heat rates and surface heat flux. The result is a greater margin to departure from nucleate boiling (DNB), lower clad temperature and peak centerline (fuel) temperature.

3.4 Reactivity Control Systems

Reactivity control is provided by two independent systems; the control rod system and the Chemical and Volume Control System (CVCS). The control rod system provides control for short-term reactivity changes (e.g., startup, shutdown and rapid transients) and is used for rapid shutdown (e.g., reactor trip or scram). All PWRs except Yankee-Rowe (Westinghouse) and Palisades (Combustion Engineering) have multi-finger control rod assemblies that insert into thimbles in the fuel assemblies. The multiple control rod fingers are joined at the top by a "spider" assembly connected to an extension shaft that can be engaged by a control rod drive mechanism (CRDM) in the reactor vessel head. Yankee-Rowe and Palisades have cruciform control rods that are inserted between the fuel assemblies. The cruciform control rods also are driven by means of extension shafts that are engaged by CRDMs in the reactor vessel head. Westinghouse and Combustion Engineering PWRs have magnetic jack CRDMs that provide rod motion in small steps. Babcock & Wilcox PWRs have roller-nut CRDMs that can provide continuous rather than stepped rod motion. Typically, there are 45 to 83 CRDMs in a PWR.

An automatic reactor trip is initiated by the Reactor Protection System (RPS) when monitored plant conditions reach specified safety system setpoints. As indicated in Figure 3.4-1, the RPS causes a reactor trip by opening the circuit breakers supplying power to the rod control system. As a result, the CRDMs are deenergized, allowing the control rods to fall into the reactor core.

The CVCS continuously adjusts boron concentration in the primary coolant to compensate for long-term reactivity changes during normal operation (e.g., fuel burnup, effects of xenon). The CVCS integrates the process of adjusting the primary coolant boron concentration with the RCS coolant inventory control function. The principal CVCS flow paths and interfaces are shown in Figure 3.4-1. The CVCS can take the reactor subcritical without use of control rods by significantly increasing the boron concentration in the primary coolant.

3.5 Heat Transfer Systems for Power Operation

When the reactor is operating at power, the normal heat transfer path is by means of three fluid system loops as illustrated in Figure 3.5-1. The first heat transfer loop is the RCS. This is a closed, single-phase, high-pressure (2200 psig) loop which circulates hot primary coolant from the reactor core, through the steam generators, and returns "cold" primary coolant to the reactor core via the reactor coolant pumps. In this heat transfer loop, the reactor core is the heat source and the steam generators are the heat sink.

The second heat transfer loop is formed by the Steam and Power Conversion System which is a closed, two-phase, lower pressure secondary coolant loop. During power operation, this secondary coolant system removes heat from the RCS by boiling water in the steam generators at about 700 to 1000 psig. The main turbine generators extract power from the steam to generate electricity and exhaust to the main condenser which operates under partial vacuum conditions (20 inches mercury vacuum). Heat is transferred in the main condenser to the tertiary circulating water cooling loop and the condensed steam is returned to the steam generators via the main condensate and feedwater systems which together increase the secondary coolant pressure back up to 1000 psig.

The tertiary coolant loop is the circulating water system which rejects plant waste heat to the ultimate heat sink. This is a low-pressure, high flowrate, single-phase coolant loop that may operate on an open cycle, closed cycle, or combined cycle. In an open cycle system, the circulating water pumps draw cooling water from a body of water (i.e. an ocean, lake, river) and return all of the heated water back to the body of water. In comparison, a closed cycle system recirculates the condenser cooling water and utilizes cooling towers or other heat exchangers to reject heat to the atmosphere. Water from a nearby source is needed to provide makeup for evaporation. In a combined cycle cooling water system, part of the plant waste heat is rejected to the atmosphere via cooling towers before the circulating water is returned to the water source.

3.6 Heat Transfer Systems for Shutdown Cooling at High RCS Pressure

During a normal shutdown, initial shutdown cooling is accomplished by using the main turbine bypass system to direct steam to the main condenser, and the condensate and feedwater systems to return the secondary coolant to the steam generators. The circulating water system completes the heat transfer path to the ultimate heat sink. This essentially is the same heat transport path as is used during power operation (see Section 3.5) except that the main turbine is tripped and bypassed and the steam, condensate, and feedwater systems are operating at a greatly reduced flow rate.

When the Steam and Power Conversion System is not available, heat may be removed from the RCS by the combined operation of the Auxiliary Feedwater (AFW) System and the secondary steam relief system (SSRS). This heat transfer path involves two cooling loops. In the first loop, heat is transferred from the reactor core to the steam generators by forced circulation, or by natural circulation when the reactor coolant pumps are unavailable. In the secondary cooling loop, the AFW system takes water from a condensate storage tank or other suitable water source and delivers it to the steam generators where it is boiled and vented to atmosphere via atmospheric dump valves in the SSRS. The atmosphere is the ultimate heat sink in this case. Core heat removal by the AFW system and the SSRS is illustrated in Figure 3.6-1.

3.7 Heat Transfer Systems for Shutdown Cooling at Low RCS Pressure

The Residual Heat Removal (RHR) System provides for post-shutdown core cooling of the RCS after an initial cooldown and depressurization to about 350°F and 425 psig by the Steam and Power Conversion System or the AFW system and the SSRS. As illustrated in Figure 3.7-1, the RHR system establishes a new closed-loop, low-pressure, single-phase primary heat transfer loop by diverting reactor coolant from an RCS hot leg to the RHR heat exchangers. In most PWRs, the RHR system is a multi-mode system that also performs the low-pressure safety injection (LPSI) function as part of the emergency core cooling system (ECCS).

Heat is transferred from the RHR system to a secondary cooling loop and the reactor coolant is returned to an RCS cold leg. The Component Cooling Water System (CCWS) forms the secondary cooling loop. This is a closed-loop, single-phase, low-pressure system that also provides cooling for other safety-related components. Heat is transferred from the CCWS to a tertiary loop that rejects heat to the ultimate heat sink. The tertiary loop is a service water system that may operate on an open, closed, or combined cycle. The service water and the circulating water systems may operate on different cooling cycles (i.e., a closed cycle service water system and an open cycle circulating water system).

3.8 RCS Overpressure Protection System

RCS overpressure protection is provided by power-operated relief valves (PORVs) and/or safety valves mounted on the pressurizer. The safety valves lift mechanically on high RCS pressure. A typical pressurizer safety valve is shown in Figure 3.8-1. The PORVs can be controlled to open at lower pressures, thereby reducing the frequency of challenges to the safety valves. The PORVs may also play a role in feed-and-bleed, or bleed-and-feed core cooling (see Section 3.9). The pressurizer safety valves and the PORVs discharge to a "quench tank" located inside the containment. The quench tank is partially filled with water and is sized to handle modest blowdowns from the RCS. Rupture disks are generally used to provide overpressure protection for the quench tank.

3.9 Emergency Core Cooling Systems

Following a breach in the reactor coolant system pressure boundary, water is lost from the RCS at a rate that is determined by several factors, including break size and location. The Emergency Core Cooling System (ECCS) is a multi-mode system that injects makeup water into the RCS during a loss-of-coolant accident (LOCA) and recirculates water through the core following a LOCA to provide for long-term post-accident core cooling. In all PWRs, the ECCS includes pressurized safety injection tanks (SITs) and high- and low-pressure safety injection (HPSI and LPSI) pumps. The RCS injection points for these ECCS subsystems vary by PWR vendor as follows:

Vendor	HPSI	LPSI	SITs
Westinghouse	Cold legs (initially) Hot legs (later)	Cold legs (initially) Hot legs (later)	Cold legs
Combustion Engineering	Cold legs	Cold legs	Cold legs
Babcock & Wilcox	Cold legs	Reactor Vessel	Reactor Vessel

In addition, the ECCS in some Westinghouse plants can be aligned to inject into the upper head of the reactor vessel.

3.9.1 ECCS Injection Phase

During the injection phase of operation following a large LOCA, the ECCS operates as an open-loop system and provides rapid injection of borated water to the RCS to ensure reactor shutdown and adequate core cooling. Following a large LOCA, the RCS is rapidly depressurized, and makeup is initially provided by the safety injection accumulators as RCS pressure drops below accumulator pressure (i.e., 650 psig). Both the high- and low-pressure safety injection pumps are aligned to take a suction on the Refueling Water Storage Tank (RWST) and deliver makeup water to the reactor vessel via the RCS cold legs. Water lost from the RCS during the LOCA is collected in the containment sump. The coolant injection and heat transport paths associated with large LOCA mitigation are shown in Figure 3.9-1.

Following a small LOCA, the RCS may slowly depressurize or remain at or near normal operating pressure. RCS pressure behavior will be determined by many factors, including the size of the small break and the availability of the steam generators as a heat sink. An RCS heat balance will be established between the heat generated in the reactor core and heat lost via the small break, the steam generators, and if necessary, the primary power-operated relief valves (PORVs) and safety valves located on the pressurizer. Maximum RCS pressure is limited by the primary safety valves. In some PWR plants, makeup to the RCS can be provided by the ECCS high-pressure safety injection pumps at pressures up to the primary safety valve setpoint. In these plants, it is a relatively straightforward matter to control RCS coolant inventory following the small LOCA.

In some PWR plants, the ECCS high-pressure safety injection pumps have a shutoff head in the range from 1400 to 1800 psig and, therefore, are not capable of providing makeup at full RCS pressure. In these plants, RCS makeup at high pressure is limited to the capacity (and availability) of the normal charging pumps, therefore it is necessary to depressurize the RCS to enable the high-pressure injection pumps to provide

RCS makeup. RCS depressurization can be accomplished by means of heat transfer to the steam generators using the AFW system and the SSRS as described previously. Alternatively, it may be possible to reduce RCS pressure by opening the PORVs on the pressurizer (i.e., bleed-and-feed).

3.9.2 ECCS Recirculation Phase

After the RWST makeup water supply has been exhausted, the ECCS is placed in the recirculation mode of operation by aligning the suctions of the low-pressure safety injection pumps to the containment sump and isolating the suction path from the RWST. In most PWR plants, the high-pressure safety injection pumps cannot be aligned to take a suction directly from the containment sump. At the time recirculation is initiated, the normally dry containment sump is full of water that has collected from the LOCA and from the operation of the containment spray system.

Following a large LOCA, the RCS is depressurized to the point that the low-pressure safety injection pumps can provide continuous makeup to the RCS and the high-pressure pumps may be stopped. If available, heat exchangers in the low-pressure safety injection system may be used during the recirculation phase to transfer heat to the ultimate heat sink via the CCWS and the service water system. The low-pressure ECCS recirculation loop is comparable to the RHR shutdown cooling loop described in Section 3.7, with the exception that the low-pressure pumps are aligned to take a suction from the containment sump.

During a small LOCA, RCS pressure may remain high, precluding injection by the low-pressure safety injection pumps which typically have a shutoff head on the order of 300 to 400 psig. In this case, the high-pressure recirculation flow path is established with the low-pressure and high-pressure safety injection pumps operating in tandem. The low-pressure pumps take a suction on the containment sump and are aligned to deliver the water to the suction of the high-pressure pumps which then inject water into the RCS via the cold legs (initially) or the hot legs (later). Water returns to the containment sump through the RCS break that caused the LOCA. Heat exchangers in the low-pressure safety injection system may be used during high-pressure recirculation to transfer heat to the ultimate heat sink via the CCWS and the service water system.

3.9.3 High-Pressure Feed-and-Bleed Cooling

Some PWRs have the capability to use the high-pressure ECCS pumps to implement a post-transient decay heat removal method called feed-and-bleed cooling. In essence, this is little more than small LOCA mitigation with the pressurizer PORV substituting for a break in the primary system. If the steam generator is unavailable as a post-transient heat sink, RCS pressure will increase to the point that the pressurizer safety valves and/or the PORVs will lift. The RCS will remain at high pressure and a heat balance will be established between decay heat generated in the core and heat carried off via the pressurizer safety valves and/or the PORVs. As shown in Figure 3.9-2, feed-and-bleed cooling is implemented by aligning a high-pressure makeup pump to maintain RCS inventory and modulating the PORVs to control RCS cooldown rate. Normally a discharge from the pressurizer safety valves and/or the PORVs is contained in the pressurizer quench tank. This tank is not sized for continuous feed-and-bleed operation, therefore, rupture disks on the tank will burst, venting the tank to the containment. The containment cooling systems are needed to complete the heat transport path to the ultimate heat sink. Normally, RCS coolant inventory is measured by the water level in the pressurizer. During feed-and-bleed cooling, pressurizer water level may not be an accurate indication of RCS coolant inventory. Furthermore, repeated cycling of the pressurizer safety valves and/or the PORVs may result in valve failure and an actual LOCA due to a stuck-open valve.

3.10 Containment and Containment Auxiliary Systems

The containment structure is a physical boundary against the release of fission products to the environment following a release from the RCS. There are three functionally different types of primary containments used in U.S. PWRs:

- Large, dry (atmospheric) containment
- (Large, dry) subatmospheric containment
- Ice condenser (pressure suppression) containment

These primary containment designs may be constructed of steel or concrete and may be used with or without a secondary containment. PWR containment types are summarized by functional design in Figures 3.10-1. PWR containments are not inerted.

3.10.1 Large, Dry Containment

The large, dry, atmospheric containment is the predominant type of PWR containment, being found in 53 of 80 PWR plants. All Combustion Engineering and Babcock & Wilcox PWRs have large, dry containments. Example of large, dry containment configurations are shown in Figure 3.10-2 (Yankee-Rowe steel sphere), 3.10-3 (Davis-Besse steel cylinder with concrete shield building), 3.10-4 (Diablo Canyon reinforced concrete cylinder with steel liner) and 3.10-5 (Zion post-tensioned concrete cylinder with steel liner). Design pressures for large, dry containments vary considerably, but generally are in the range from 40 to 61 psig. Among plants with large, dry containments, Yankee-Rowe has the lowest containment design pressure at 34 psig.

3.10.2 Subatmospheric Containment

Subatmospheric containments are only found at seven Westinghouse PWR plants, six 3-loop plants, and one 4-loop plant. All subatmospheric containments are constructed of reinforced concrete with a steel liner. An example of the configuration of a subatmospheric containment is shown in Figure 3.10-6 (Millstone 3 4-loop PWR). Design pressures for subatmospheric containments vary considerably, but generally are in the range from 45 to 60 psig.

3.10.3 Ice Condenser Containment

Ice condenser containments are only found at ten Westinghouse 4-loop plants. Examples of ice condenser containments are showing Figure 3.10-7 (Catawba, Sequoyah, and Watts Bar steel cylinder with concrete shield building), and 3.10-8 (typical of D.C. Cook and McGuire reinforced concrete cylinder with steel liner). An isometric view of the ice condenser is shown in Figure 3.10-9. Due to the pressure suppression effects of the ice condenser, these containments have lower design pressures than either large, dry containments or subatmospheric containments. Typical design pressures for ice condenser containments are in the range from 12 to 30 psig.

3.10.4 Containment Auxiliary Systems

Regardless of the containment type, all PWR containment designs have auxiliary systems to accomplish the functions of containment isolation, containment pressure control and heat removal, containment fission product cleanup, and combustible gas control. Systems related to these functions are described below.

A. Containment Isolation

During normal operation, PWR containments typically are closed, or have only a limited amount of "purge" airflow. Containment cooling during normal operation typically is provided by a recirculating ventilation system, therefore, large diameter ventilation lines penetrating containment can remain isolated. Following a LOCA, the containment isolation system causes isolation valves and dampers to close in certain lines that penetrate the containment boundary, including any open containment purge lines.

B. **Containment Pressure Control, Heat Removal, and Fission Product Cleanup**

The functions of containment pressure control, heat removal, and fission product cleanup are integrated in the containment spray system in most PWRs. Containment pressure and fission product concentration in the containment atmosphere are reduced by a containment spray system. The design of this system varies with containment type, but a spray system is found in large, dry containments, subatmospheric containments, and ice condenser containments. In most PWRs, the containment spray system initially injects water from the RWST into the containment via spray headers located in the dome of the containment. A chemical additive usually is added to the spray water to enhance its fission product removal capability. In the ice condenser plants, the ice beds perform a pressure suppression function to limit maximum containment pressure and also provide some "scrubbing" of fission products in the containment atmosphere. When the RWST has been emptied in a plant with a large, dry containment or an ice condenser containment, the containment spray pump suction is aligned to the containment sump and the suction path from the RWST is isolated. In a subatmospheric containment, there typically are two spray systems; an injection spray system that functions as described above, and a recirculation spray system. When the RWST has been emptied in a plant with a subatmospheric containment, the injection spray system is secured and the recirculation spray system is started.

In large, dry containments, post-LOCA containment heat removal is accomplished by heat exchangers in the containment spray system or the residual heat removal system and/or containment fan coolers. The fan coolers include filter beds for fission product removal. A simplified diagram of these containment cooling system heat transport paths is shown in Figure 3.10-10. The heat transfer path from the containment spray (or RHR) heat exchangers and the containment fan coolers to the ultimate heat sink is completed by one or two cooling water loops (i.e. the CCWS and/or the service water system). Plants with subatmospheric containments or ice condenser containments typically do not have containment fan coolers for post-LOCA containment heat removal.

C. **Containment Combustible Gas Control**

PWR containments are not inerted. Post-LOCA combustible gas concentration in the containment can be controlled by hydrogen recombiners and igniters.

3.11 Component Cooling Systems

The Component Cooling Water System (CCWS) is a low-temperature, low-pressure, single-phase cooling system that provides cooling for a wide range of safety-related components. As illustrated in Figure 3.11-1, the CCWS may provide for component, area, or system cooling in a variety of ways:

A. **Direct component cooling**

This type of cooling arrangement is illustrated by cooling paths A-A' and applies to cooling for pump bearings and seals.

B. Area cooling by means of fan cooler units

This type of cooling arrangement is illustrated by cooling paths B-B' and applies to equipment room coolers that are required to maintain normal and post-accident environmental conditions within limits necessary for long-term operation of components in safety systems.

C. Fluid system heat removal

This type of cooling arrangement is illustrated by cooling paths C-C' and applies to CCWS cooling for a variety of systems including the RHR system, containment spray system, and the spent fuel pool cooling system.

D. Area cooling by means of HVAC units

This type of cooling arrangement is illustrated by cooling paths D-D' and applies to CCWS cooling for normal and emergency heating, ventilating, and air-conditioning (HVAC) systems. In this case, the chiller system forms a closed-loop heat transfer system between the CCWS and the area(s) being cooled.

The CCWS rejects heat to the ultimate heat sink through the cooling loop indicated as the service water system in Figure 3.11-1. This service water system may operate on an open, closed, or combined cycle as described previously.

3.12 Safety System Actuation

The role of the Safety System is to actuate components and systems needed to mitigate the consequences of events that challenge limits established for normal plant operation. The Safety System consists of two major subsystems: the Reactor Protection System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). As described previously, the function of the RPS is to initiate a reactor scram when needed. The ESFAS provides for automatic actuation of a wide variety of components and systems based on the detection of abnormal conditions in the reactor plant. As appropriate, the ESFAS can actuate systems necessary for RCS coolant inventory control and/or core cooling, containment isolation and cooling, radioactive release control, emergency power, and component cooling.

As illustrated in Figure 3.12-1, the ESFAS includes provisions for manual actuation at the system level (typically from the control room) or at the actuation-train level (typically from the ESFAS output logic cabinets). A manual trip from the control room actuates all components that would be actuated by an automatic ESFAS actuation signal. A manual trip from ESFAS output logic cabinets actuates only the components that are controlled by the respective ESFAS train.

The relationship between the ESFAS and other means of actuation is also shown in Figure 3.12-1. Individual remote-manual component controls, which do not use any part of the ESFAS logic, generally are provided in the control room and/or at some other alternate control location. In addition, most motor-driven components can be manually actuated by manipulating their circuit breaker on the respective switchgear panel or motor control center. Other types of power-operated valves often can be controlled locally by manual manipulation of the pilot valves on the pneumatic or hydraulic actuator.

3.13 Onsite Electric Power System

The onsite electric power system consists of two parts; the non-Class 1E system which supplies non-safety loads, and the Class 1E system which supplies safety systems. During normal operation, the entire onsite electric power system is supplied from the output of the main generator and/or the offsite grid. Diesel generators are standby AC power sources for the Class 1E portion of the onsite power system, and batteries are standby DC power sources. During normal operation, the diesel generators are idle, and the batteries are maintained fully charged by battery chargers which also supply the DC power loads.

Large Class 1E AC electrical loads (i.e. large pumps and fans) typically are supplied from 6.9 or 4.16 kV switchgear. Smaller Class 1E AC loads (i.e. motor-operated valves, small pumps and fans, battery chargers) are supplied from 480 VAC motor control centers. A representative onsite 4.16 kV and 480 VAC power system (Callaway) is shown in Figure 3.13-1.

Most DC loads are supplied from 125 VDC panels, although some plants may have 250 VDC distribution systems to support DC-powered motor-operated valves or other relatively large DC-powered components. Instrumentation power typically is supplied from a 120 VAC system that normally is powered from the 125 VDC system with backup power from the 480 VAC system. A representative 125 VDC and 120 VAC system (Callaway) is shown in Figure 3.13-2.

Loss of the normal (preferred) source of offsite power typically causes an automatic shift to the alternate source of offsite power and starts the respective standby diesel generator(s). If both sources of offsite power are unavailable, the non-Class 1E and the Class 1E portions of the onsite electric power system are separated by opening circuit breakers, and the diesel generators are aligned to supply the Class 1E system. The standby diesel generators and batteries can provide adequate power to enable other safety systems to establish and maintain a safe shutdown condition.

The diesel generators are complex systems with integrated diesel and generator control systems that interface with a load-sequencing system that re-energizes selected loads in prescribed sequences when the diesel generator is ready for loading. Diesel generator starting is dependent on a source of DC power (usually the station batteries) for the control systems and generator field flashing. In addition, the following support systems typically are needed for diesel generator operation:

- Fuel oil system (including the day tank which is the short-term fuel source)
- Fuel oil storage and transfer system (long-term fuel source)
- Air start system
- Lubricating oil system
- Jacket cooling water system
- Combustion air intake and exhaust system
- Diesel room cooling system

Simplified schematics for these systems are shown in Figures 3.13-3 and 3.13-4. As shown in Figure 3.13-4, heat from the diesel generator jacket cooling water system and lubricating oil system is transferred to the ultimate heat sink via a service water system. In a few plants, the jacket cooling water system may incorporate a radiator (i.e., a water-to-air heat exchanger) and use the atmosphere as a heat sink for diesel generator operation. A significant amount of heat from diesel generator operation is transferred directly to the atmosphere by the diesel exhaust system and the diesel room ventilation system.

Table 3.1-1. Summary of PWR Systems

PWR System	NSSS Scope	BOP Scope
Reactor	X	
Reactivity Control System	X	
Reactor Coolant System	X	
Shutdown Cooling System	X	
Reactor Water Cleanup System		X
Containment		X
Emergency Core Cooling System	X	
Habitability Systems		X
Containment Spray Systems		X
ESF Filter Systems		X
Reactor Trip System	X	
Engineered Safety Feature Actuation System	X	X
Safety Related Display Instrumentation	X	X
Non-Safety Control Systems		X
On-Site Electric Power System		X
Off-Site Electric Power System		X
New Fuel Storage System		X
Spent Fuel Storage System		X
Spent Fuel Pool Cooling and Cleanup System		X
Fuel Handling System	X	
Service Water System		X
Component Cooling Water System		X
Ultimate Heat Sink		X
Compressed Air System		X
Process Sampling System		X
Chemical and Volume Control System	X	
Non-safety HVAC System		X
Fire Protection System		X
Diesel Generator Fuel Oil Storage and Transfer System		X
Diesel Generator Cooling Water System		X
Diesel Generator Starting System		X
Diesel Generator Lubrication System		X
Diesel Generator Combustion Air Intake and Exhaust System		X
Main Steam System		X
Turbine Generator and Auxiliaries		X
Main Feedwater and Condensate System		X
Circulating Water System		X
Auxiliary Feedwater System		X
Radioactive Liquid Waste System		X
Radioactive Gaseous Waste System		X
Radioactive Solid Waste System		X
Radiation Protection System		X

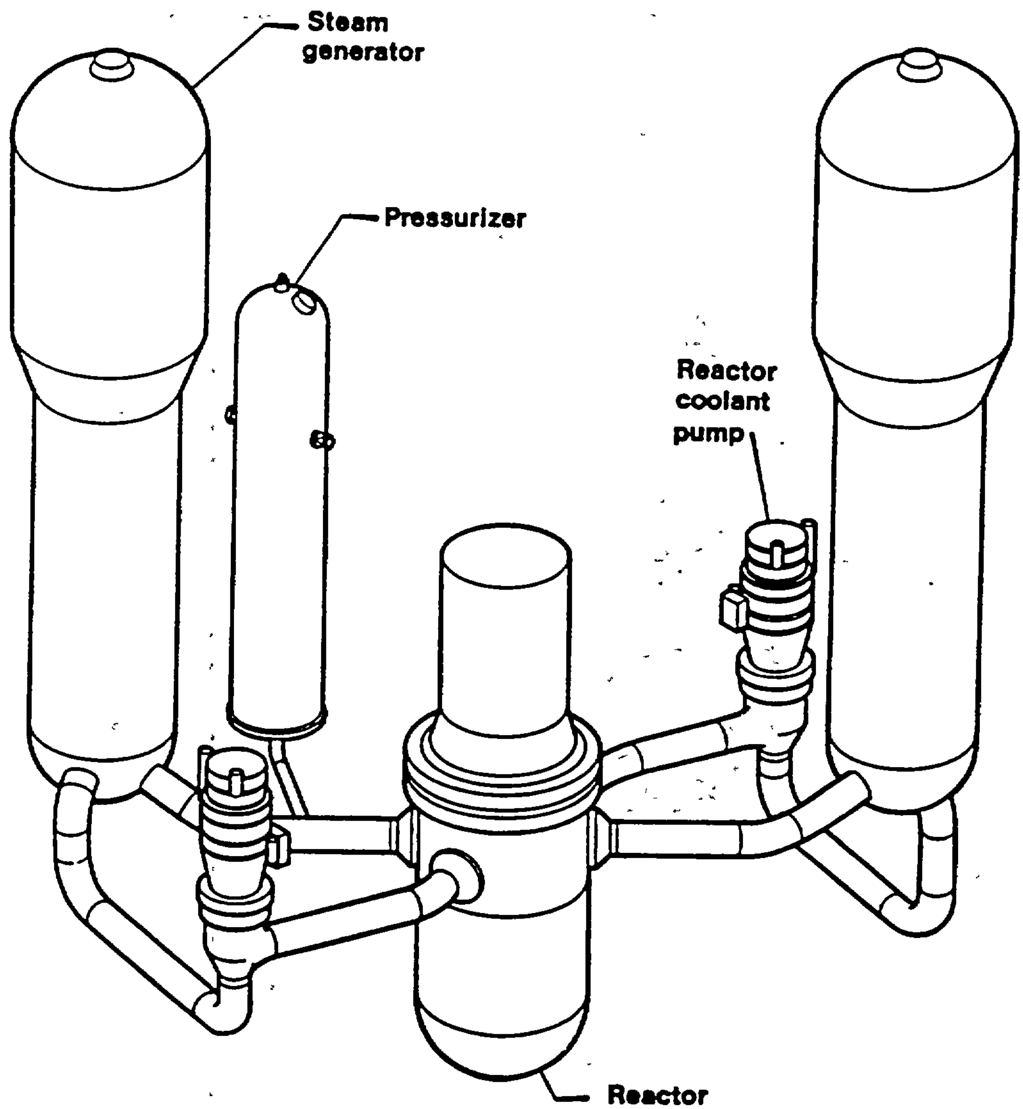


Figure 3.2-1. Westinghouse 2-Loop PWR NSSS

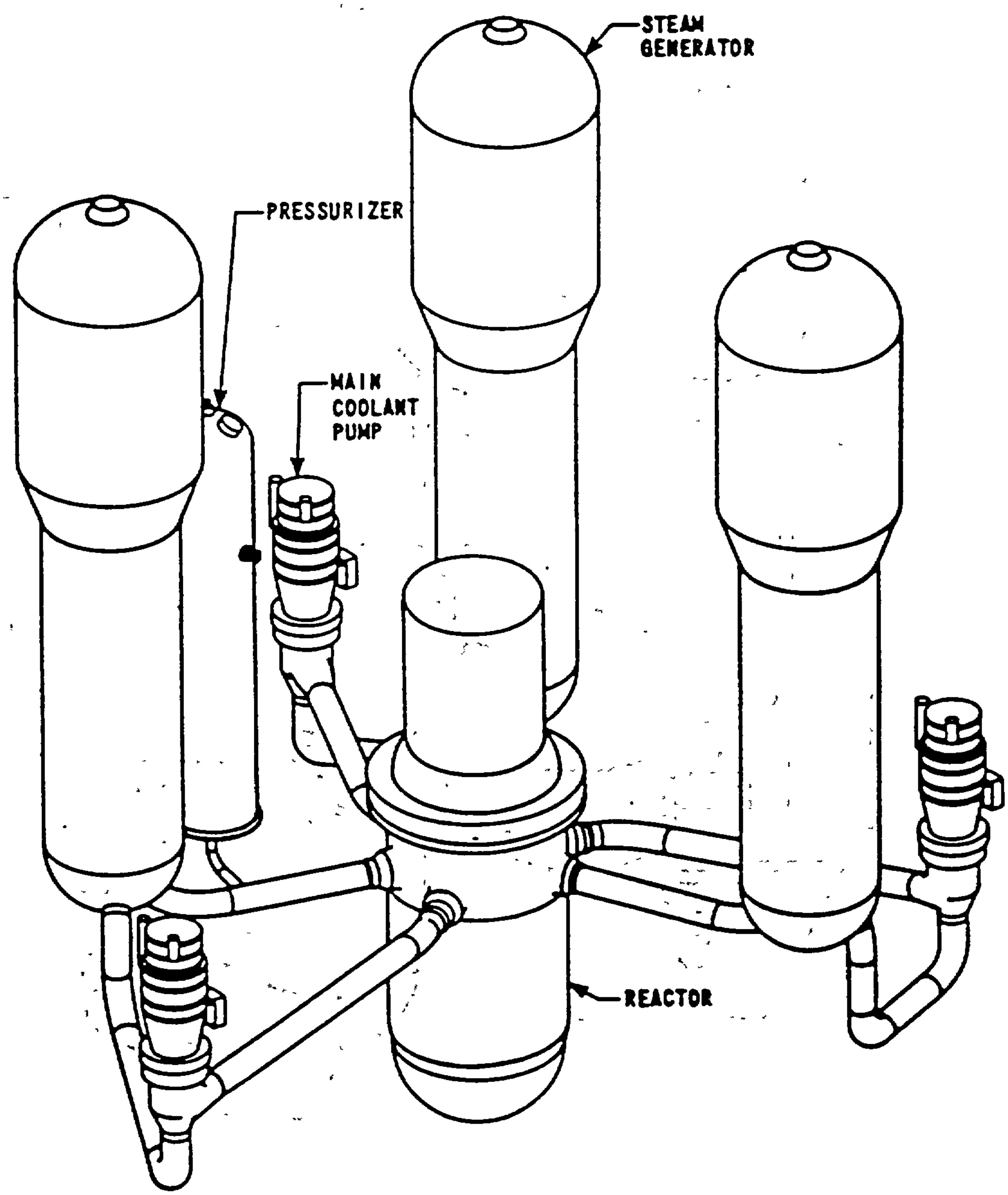


Figure 3.2-2. Westinghouse 3-Loop PWR NSSS

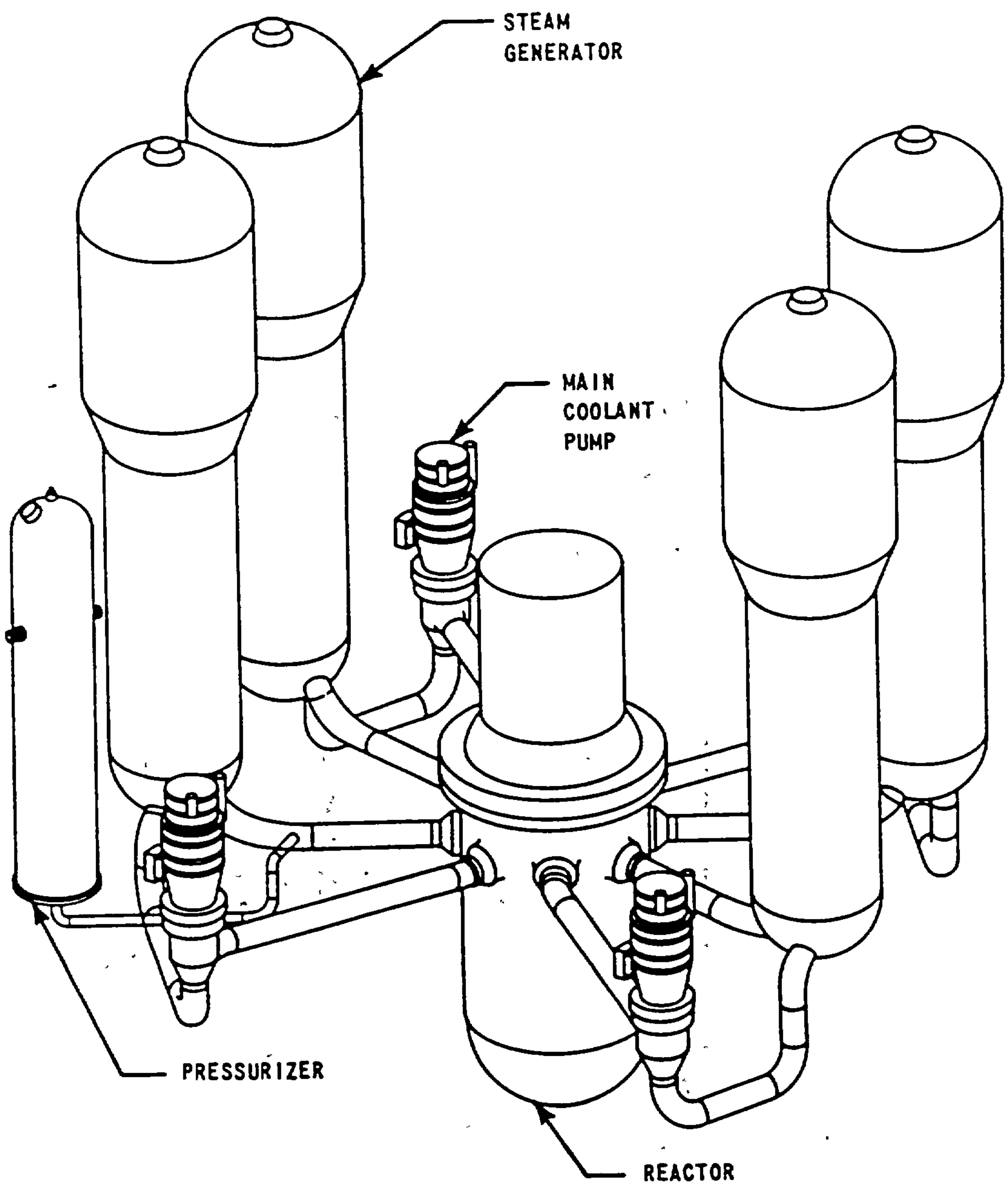


Figure 3.2-3. Westinghouse 4-Loop PWR NSSS

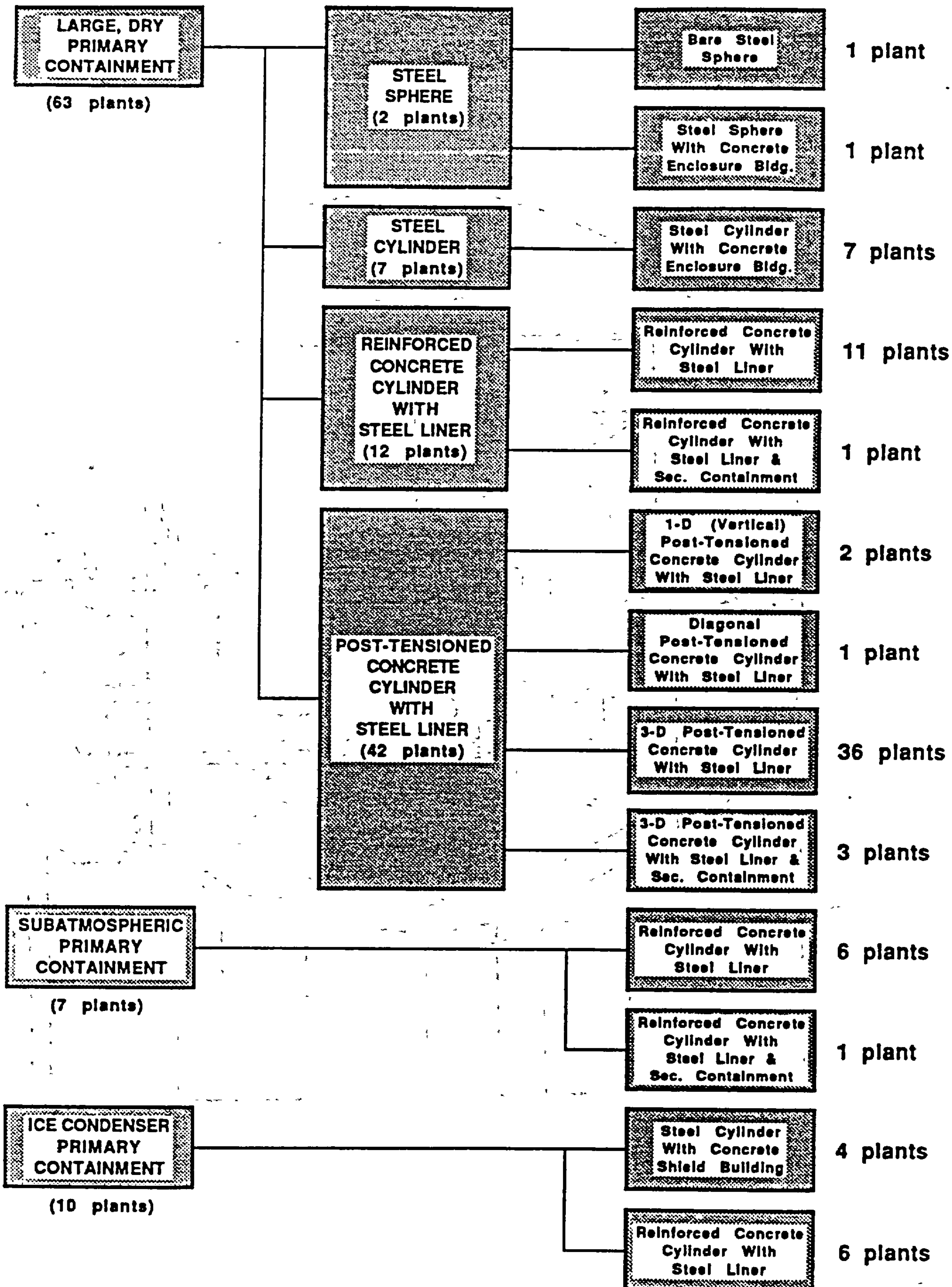


Figure 3.10-1. Distribution of PWR Containment Types

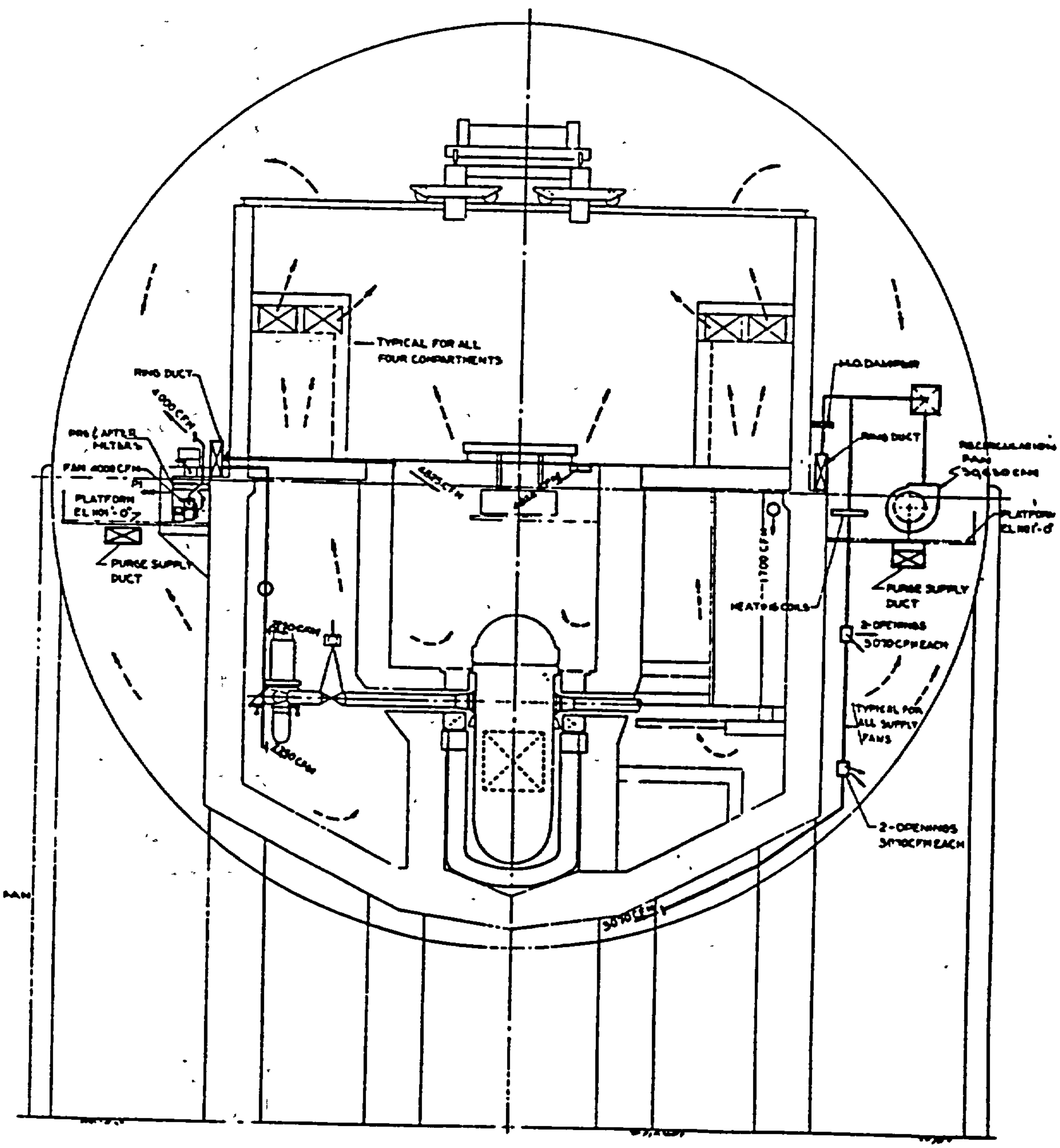


Figure 3.10-2. Yankee-Rowe Large, Dry Containment (Steel Sphere)

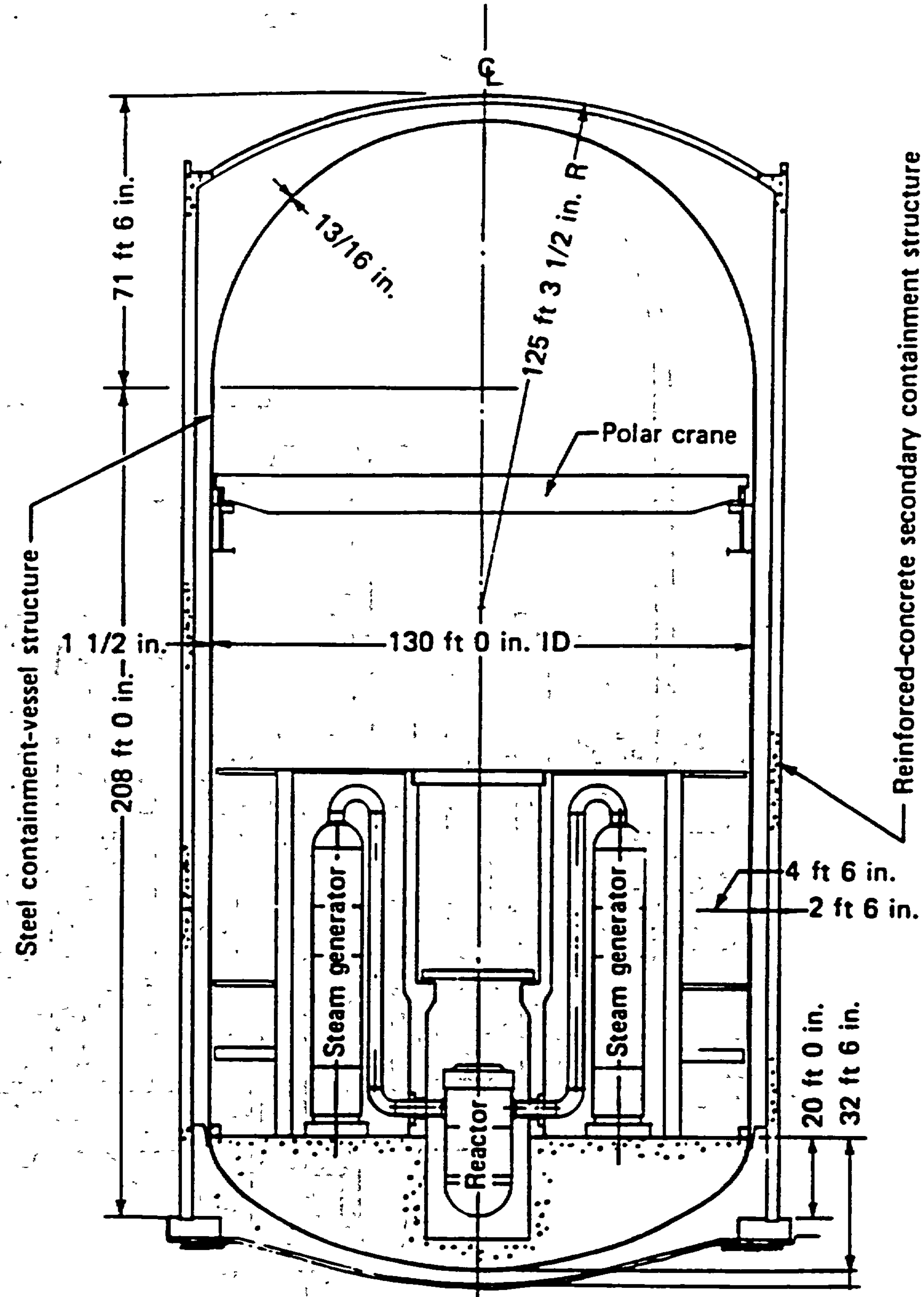


Figure 3.10-3. Davis-Besse Large, Dry Containment (Steel Cylinder with Concrete Shield Building)

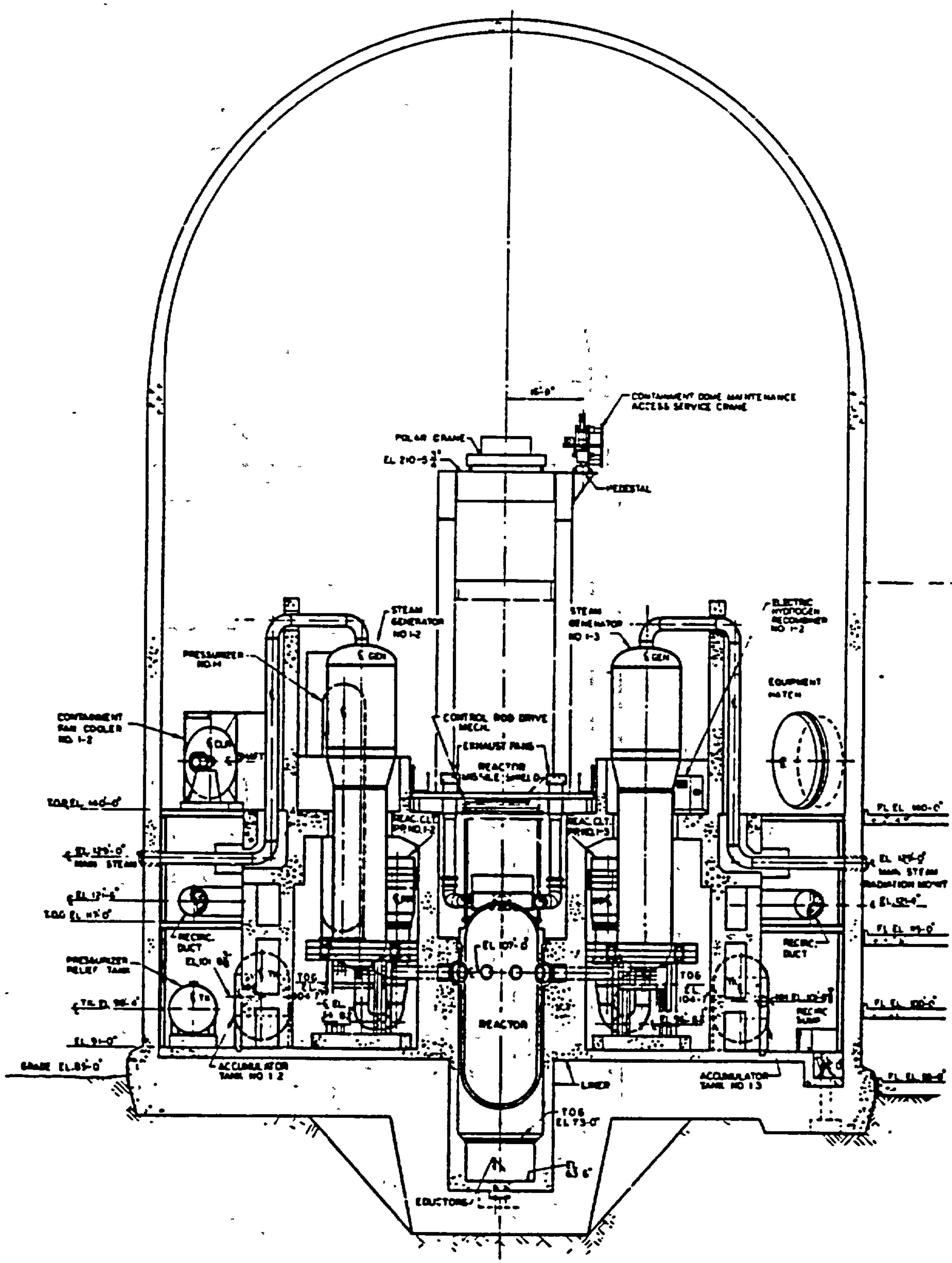


Figure 3.10-4. Diablo Canyon Large, Dry Containment (Reinforced Concrete with Steel Liner)

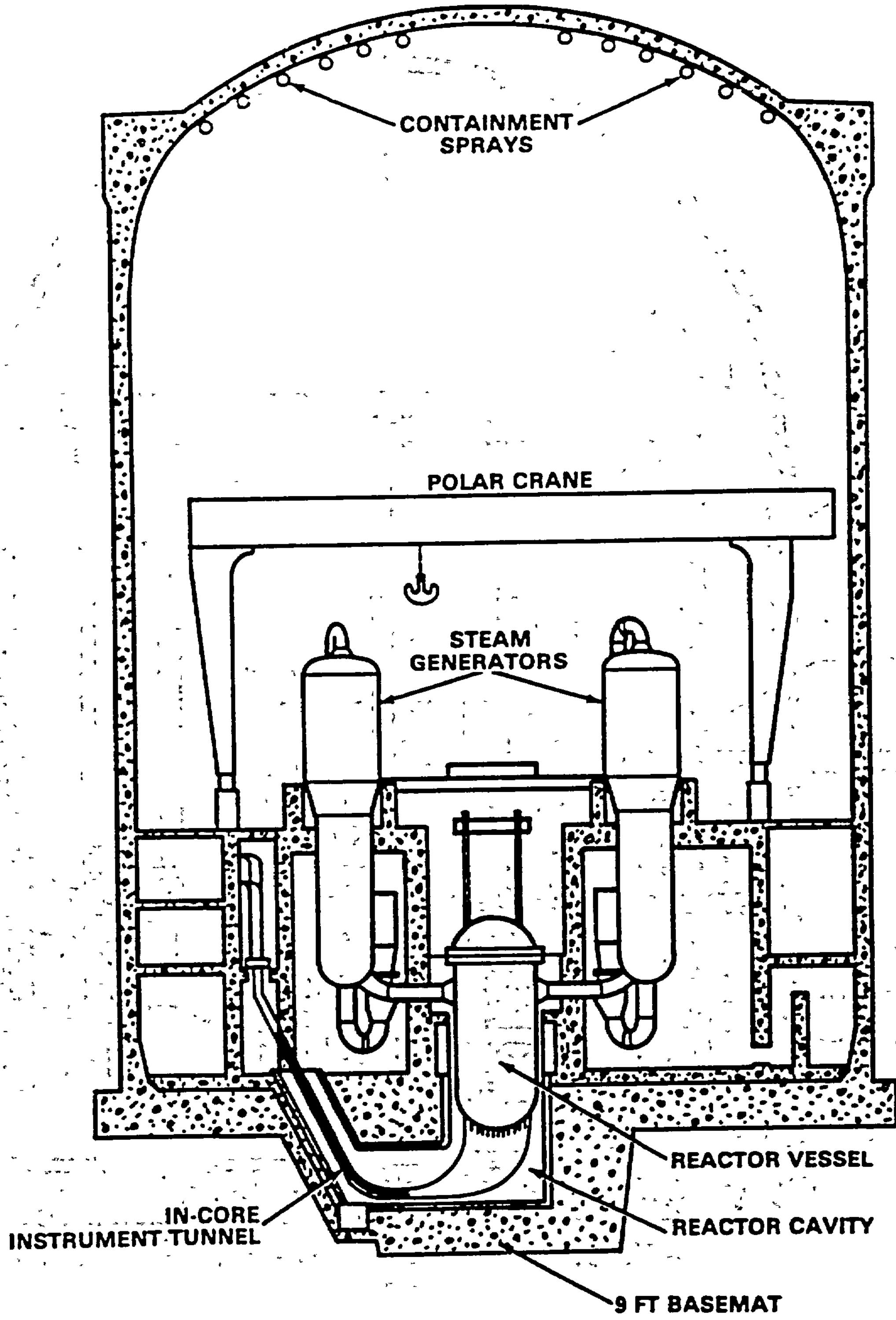


Figure 3.10-5. Zion Large, Dry Containment (Post-Tensioned Concrete with Steel Liner)

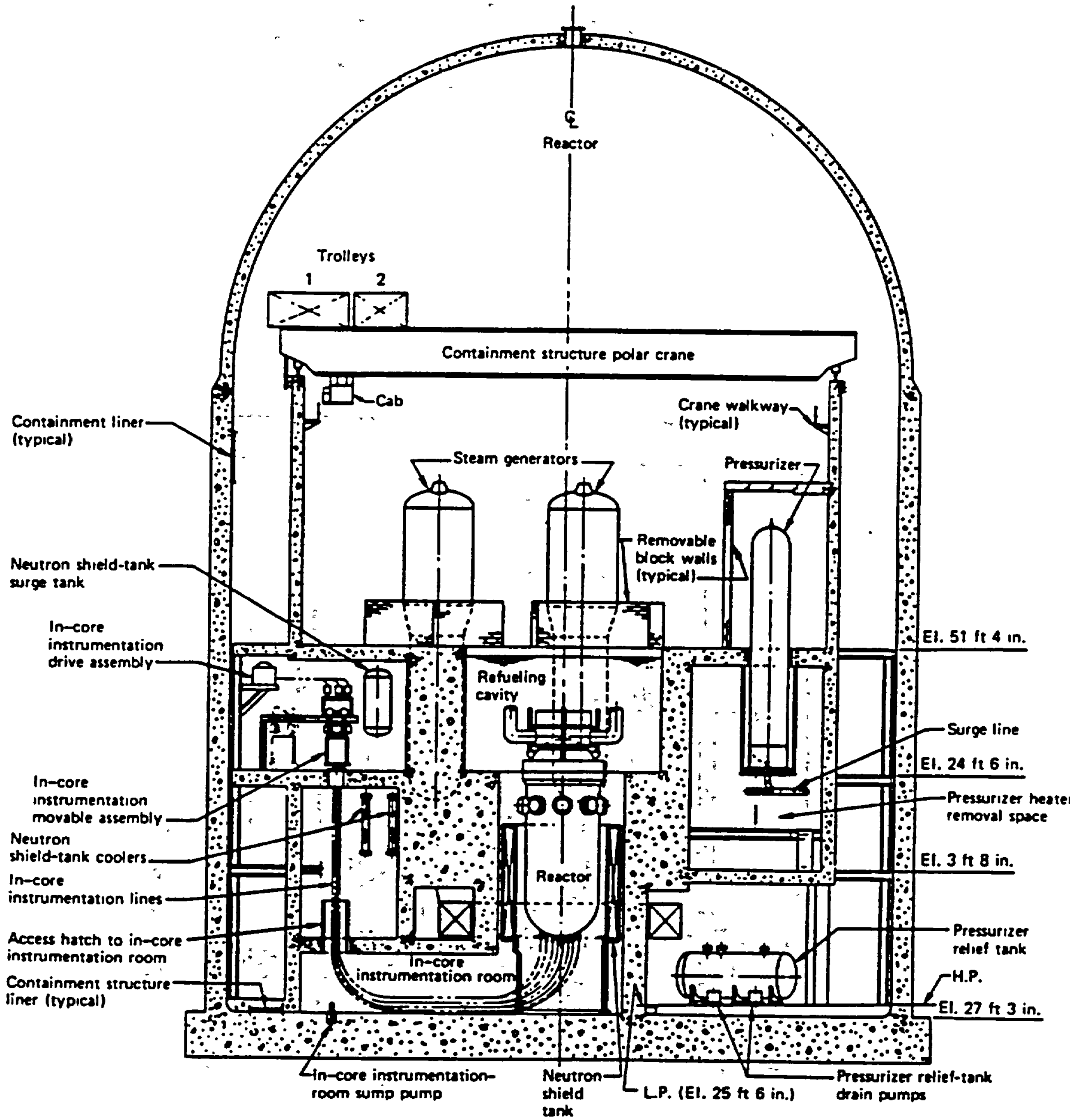


Figure 3.10-6. Millstone 3 Subatmospheric Containment (Reinforced Concrete with Steel Liner)

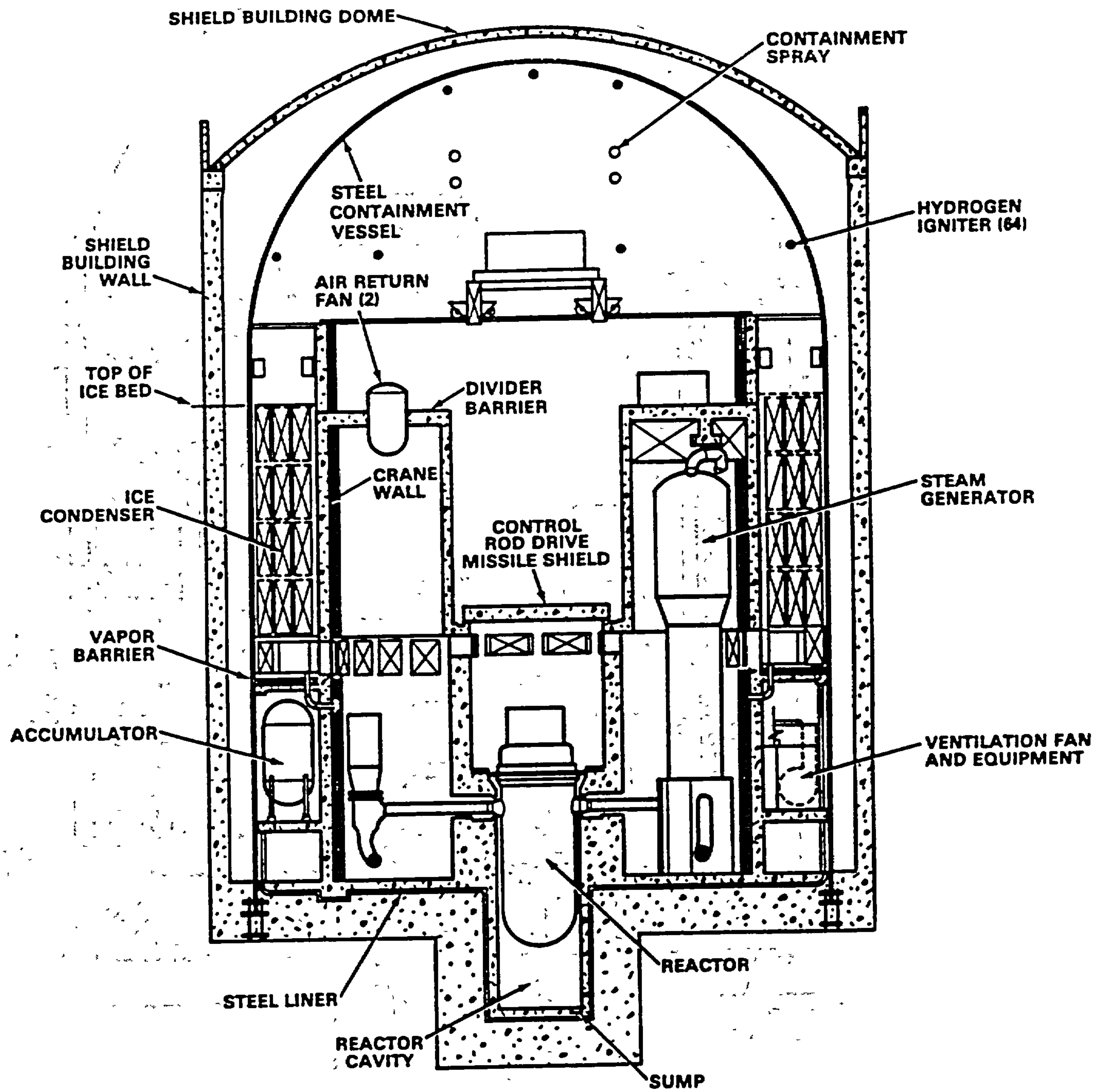


Figure 3.10-7. Sequoyah Ice Condenser Containment (Steel Cylinder with Concrete Shield Building)

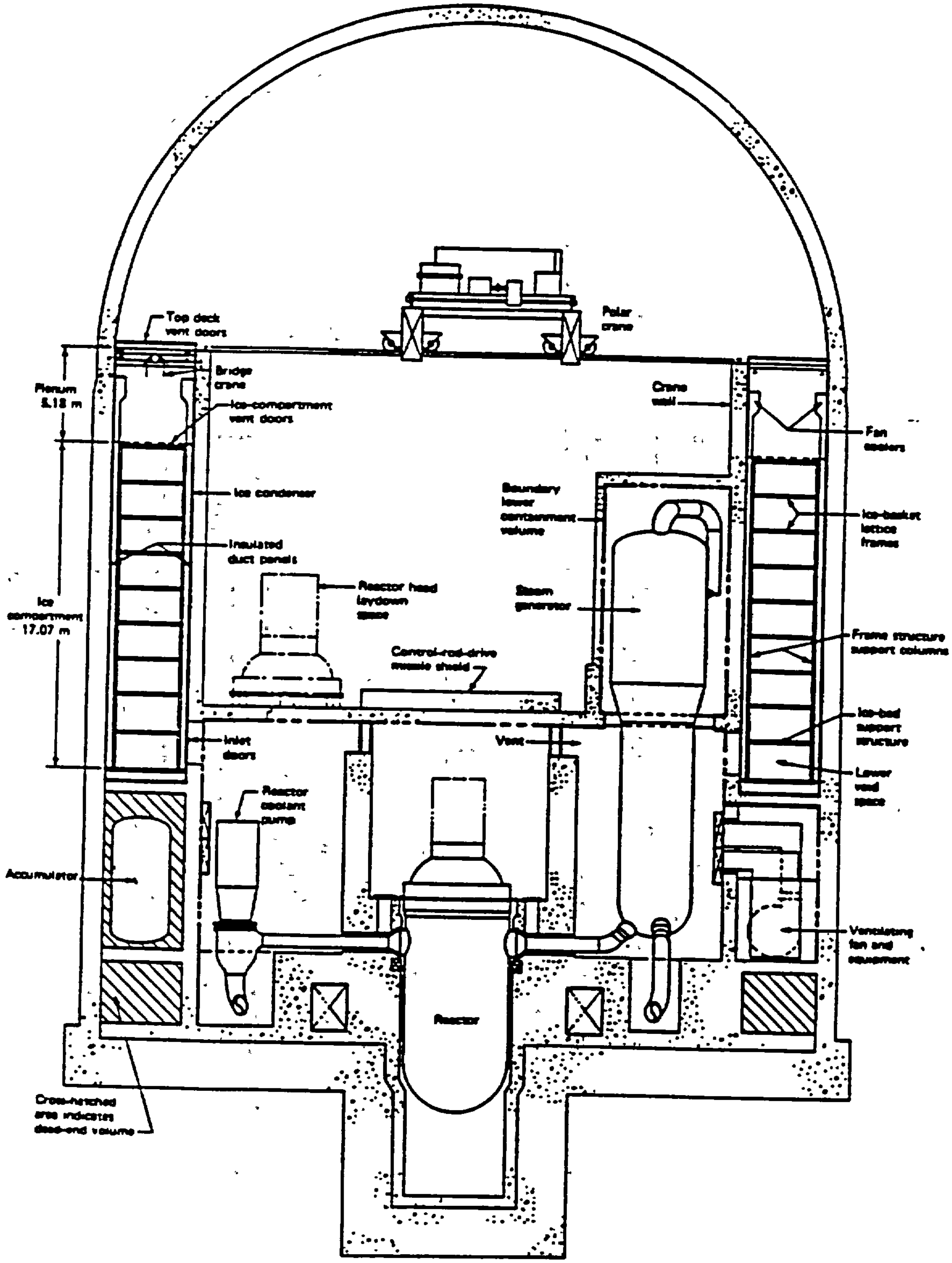


Figure 3.10-8. Ice Condenser Containment with Reinforced Concrete Structure and Steel Liner

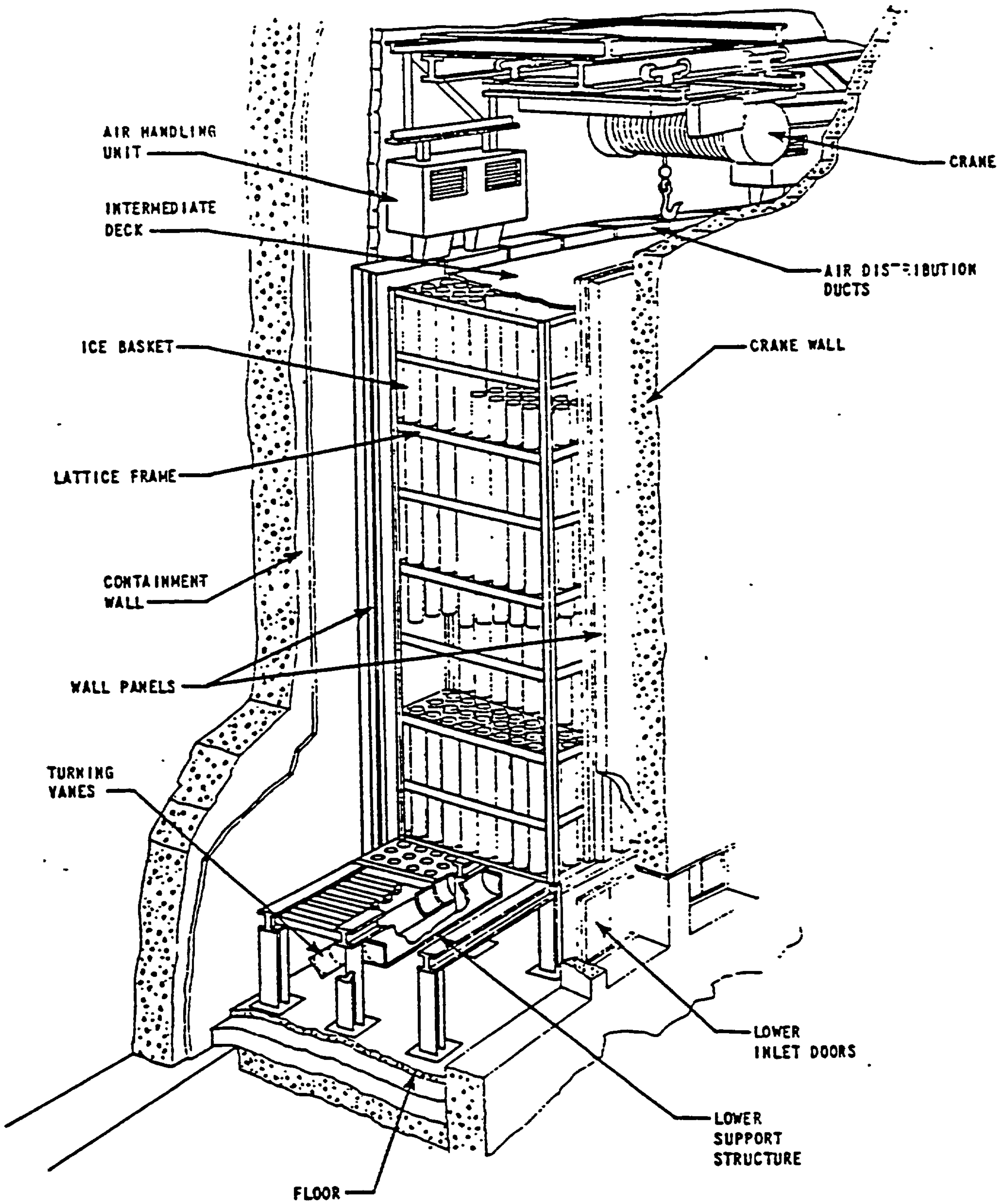


Figure 3.10-9. General Arrangement of an Ice Condenser

4. WESTINGHOUSE PRESSURIZED WATER REACTORS (PWRs)

In the U.S., Westinghouse has produced 55 PWRs with two, three, and four primary loops. As of March 1990, the numbers of units of each type are as follows:

- 2-loop	6 units
- 3-loop	14 units
- 4-loop	35 units

A general orientation to PWR systems is presented in Section 3. Expanding on this introductory material, an overview and brief comparison of major features of Westinghouse PWRs is presented in Section 4.1. The 2-, 3-, and 4-loop Westinghouse plants are described in Sections 4.2 to 4.4, and numerous detailed comparative tables are presented in Section 4.5. In Section 7, the comparative tables for the Westinghouse PWRs are compiled with similar tables for Combustion Engineering and Babcock & Wilcox PWRs.

4.1 Westinghouse PWR Overview

4.1.1 Primary System

In all Westinghouse PWRs, a primary loop consists of a U-tube steam generator, a single vertical, centrifugal reactor coolant pump, and connecting loop piping. The pressurizer is connected to one of the RCS hot legs. The general configuration of the Westinghouse reactor vessel and internals is shown in Figure 4.1-1. The U-tube steam generator is shown in Figure 4.1-2 and the pressurizer is shown in Figure 4.1-3. Typical reactor vessel sizes for the three Westinghouse PWR models are as follows:

- 2-loop	132 inch i.d.
- 3-loop	156 to 159 inch i.d.
- 4-loop	173 inch i.d.

There are four principal models of U-tube steam generators in Westinghouse plants: 27-series, 44-series, 51-series, and Model F. All models have integral moisture separators and steam dryers. Basic design characteristics of Westinghouse steam generators are listed in Table 4.1-1.

The Westinghouse RCS is designed to operate with nearly constant cold leg temperature (T_{cold}). Hot leg temperature (T_{hot}) and average loop temperature (T_{ave}) increase with power level as shown in Figure 4.1-4. This is the same RCS temperature control scheme used in Combustion Engineering PWRs.

4.1.2 Reactor Core and Fuel Assemblies

With the exception of Yankee-Rowe and Shippingport, all Westinghouse commercial PWRs are designed to operate with rod-type slightly-enriched fuel in a 14 x 14 fuel assembly array, a 15 x 15 array, or a 17 x 17 array. Yankee-Rowe has a unique rod-type fuel assembly design, and Shippingport had unique plate-type and rod-type fuel assemblies. All Westinghouse PWRs except Yankee-Rowe and Shippingport also use multi-finger control rods that insert into channels in the fuel assemblies. Yankee-Rowe and Shippingport both were designed with cruciform control rods. A comparison of basic Westinghouse core parameters is presented in Section 4.5. A brief description of each rod-type fuel assembly design is provided below.

A. Yankee-Rowe Fuel Assembly (9 x (6 x 6))

The first-generation of Westinghouse PWR fuel assembly, used only in Yankee-Rowe, consisted of nine 6 x 6 subassemblies arranged in a square array (i.e. essentially a 36 x 36 array). The assemblies measure 7.61 inches square.

Table 4.1-1: General Characteristics of Westinghouse Steam Generators

Design Parameters	27 Series	44 Series	51 Series	Model F
Tube Side (primary)				
Design Pressure (psig)	2,485	2,485	2,485	2,485
Nom. Operating Press. (psig)	2,000	2,235	2,235	unk.
Design Temp. (°F)	unk.	650	650	650
Design Flow Rate (lb/hr)	2.50E+07	3.40E+07	3.40E+07	3.55E+07
Total Primary Side Vol. (ft3)	553	944	1,080	962
Shell Side (secondary)				
Design Pressure (psig)	1,035	1,085	1,085	1,185
Full Power Pressure (psig)	570	755	797 to 960	1,000
Design Temp. (°F)	unk.	600	600	600
Full Power Temp. (°F)	unk.	514	518 to 540	544 to 559
Steam Flow Rate (lb/hr) @ Full Power	1.55E+06	3.32E+06	3.76E+06 to 4.06E+06	3.78E+06
Total Secondary Side Vol. (ft3)	2,592	4,580	5,868	3,559

with an overall length of 111.25 inches. As shown in Figure 4.1-5, the cruciform control rods were inserted between the fuel assemblies. Rubbing straps on the outside edges of the fuel assemblies protected the outer fuel rods from wear by the control rods. The fuel rod cladding was thick-wall stainless steel, and spacing between rods was established by ferrules brazed to the fuel rods.

The later-generation Yankee-Rowe fuel continued the geometry of the original fuel assembly, but changed to thinner-wall cold worked stainless steel cladding or zircaloy. In addition, fuel rod spacing was established by a grid structure.

B. 14 x 14 Fuel Assembly

The 14 x 14 fuel assembly was introduced and is still used in several early Westinghouse plants (i.e., San Onofre 1, Point Beach) along with a corresponding change in the design of the control rods. The use of "rod cluster control assemblies" (RCCAs) distributed control rod poison more uniformly by means of multi-finger control rods that can be inserted into full-length thimbles in each fuel assembly. The 14 x 14 fuel assembly is designed for use with a 16 "finger" RCCAs. The insertion of an RCCA into a 14 x 14 fuel assembly is shown in Figure 4.1-6. The thimbles are structural assemblies that join the end pieces of the fuel assembly. The use of RCCAs instead of cruciform control rods reduced the occurrence of local hot spots when the rods were withdrawn. When first introduced, the 14 x 14 fuel assemblies used stainless steel cladding and structural parts, but changed to zircaloy cladding. Gradual replacement of stainless steel by zircaloy in fuel assembly structural parts continued into the 1970s.

C. 15 x 15 Fuel Assembly

Use of the 15 x 15 fuel assembly was introduced in 1967 with stainless steel cladding and structural parts, but changed to zircaloy cladding in 1968. Gradual replacement of stainless steel by zircaloy in fuel assembly structural parts continued into the 1970s. The 15 x 15 fuel assembly is designed for use with a 20 "finger" rod cluster control assembly (RCCA). Most plants that have used the 15 x 15 fuel assemblies are transitioning to the 17 x 17 fuel assemblies.

D. 17 x 17 Fuel Assembly

The Westinghouse 17 x 17 fuel assembly is shown in Figures 4.1-7 and 4.1-8. The 17 x 17 fuel assembly was designed to fit in the same geometric and power envelope as the 15 x 15 fuel assembly. The use of the 17 x 17 array resulted in numerous advantages over the 15 x 15 fuel assembly, including:

- Core average power density reduced from 7.03 to 5.43 kW/liter (for a 3411 MWt core)
- Reduced linear power rating
- 12% lower surface heat flux, hence increased margin to Departure from Nucleate Boiling (DNB)
- About 500°F (278°C) reduction in peak clad temperature under LOCA conditions.

Nominal fuel enrichment is in the range from 2.1 to 3.1 weight percent U-235. The 17 x 17 fuel assembly is designed for use with a 24 "finger" rod cluster control assembly (RCCA). The center thimble is reserved for in-core instrumentation. This fuel assembly is mostly zircaloy except for the bottom nozzle which is 304 stainless steel, and various springs and bolts which are Inconel 600 or 718.

4.1.3 Reactivity Control System

Core reactivity is controlled by full-length and part-length rod cluster control assemblies (RCCAs) burnable poison rods, and soluble boron in the coolant. The neutron absorber in the control rods typically is Ag-In-Cd (silver-indium-cadmium), while the burnable poison is B_4C (boron carbide) that may be in the form of borosilicate glass. Some Westinghouse plants use B_4C for the control rod poison.

As shown in Figure 4.1-9, the RCCAs consist of multiple control rods that are joined at the top by a "spider" assembly and attached to the control rod drive mechanism (CRDM) extension shaft. Magnetic jack-type CRDMs are used to control the full-length and part-length RCCAs. These CRDMs consist of a set of five magnetic coils outside of the CRDM pressure housing, and solenoid-operated plungers and "gripper latches" inside the pressure housing to engage the grooved drive rod extension shafts and hold, insert, or withdraw the control rods. The five sets of CRDM coils are: (a) the stationary gripper coil, (b) the movable gripper coil, (c) the lift coil, (d) the push-down coil, and (e) the load-transfer coil. The action of these coils is programmed so that stationary and movable grippers are alternately engaged with the grooved drive shaft. The stationary gripper holds the drive shaft while the movable gripper is moving to its new position to raise or lower the control rod through steps of about 3/8 inch. The CRDMs for the full-length rods are designed so that, upon loss of electrical power to the magnetic jack coils, the RCCA is released and falls by gravity into the core. The CRDMs for the part-length RCCAs are designed to hold the rods "as-is" upon loss of power. Details of a magnetic jack CRDM are shown in Figure 4.1-10 and the location of the CRDMS with respect to the reactor vessel and fuel assemblies is shown in Figure 4.1-11. The magnetic jack CRDMs are air-cooled.

4.1.4 Containment

Westinghouse PWRs have been built with a greater variety of containment designs than either C-E or B&W PWRs, and include the only PWRs with subatmospheric containments and ice condenser containments. The distribution of containment types for Westinghouse plants is shown in Figure 4.1-12. Examples of containments for Westinghouse PWRs are included in the following sections.

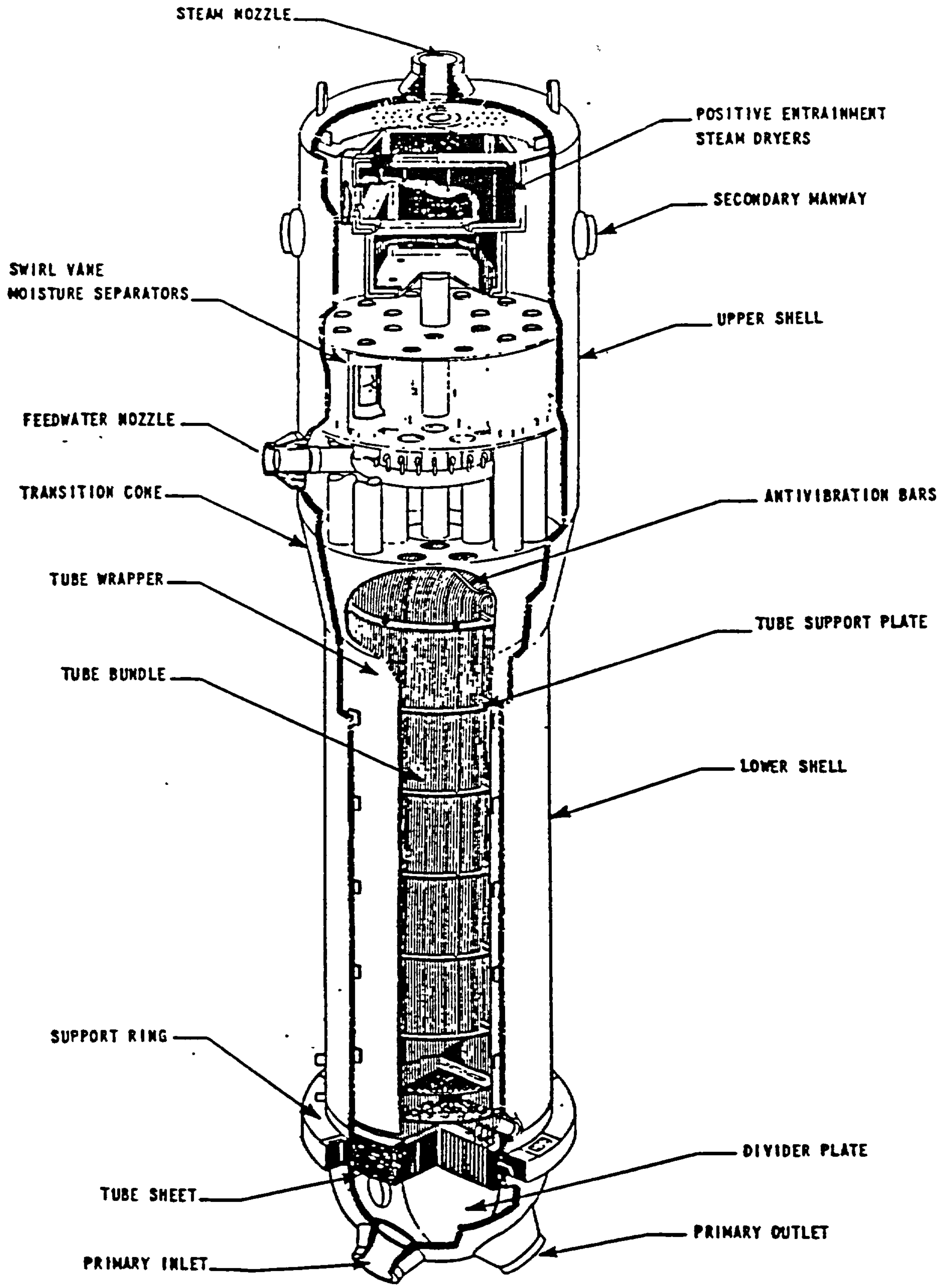
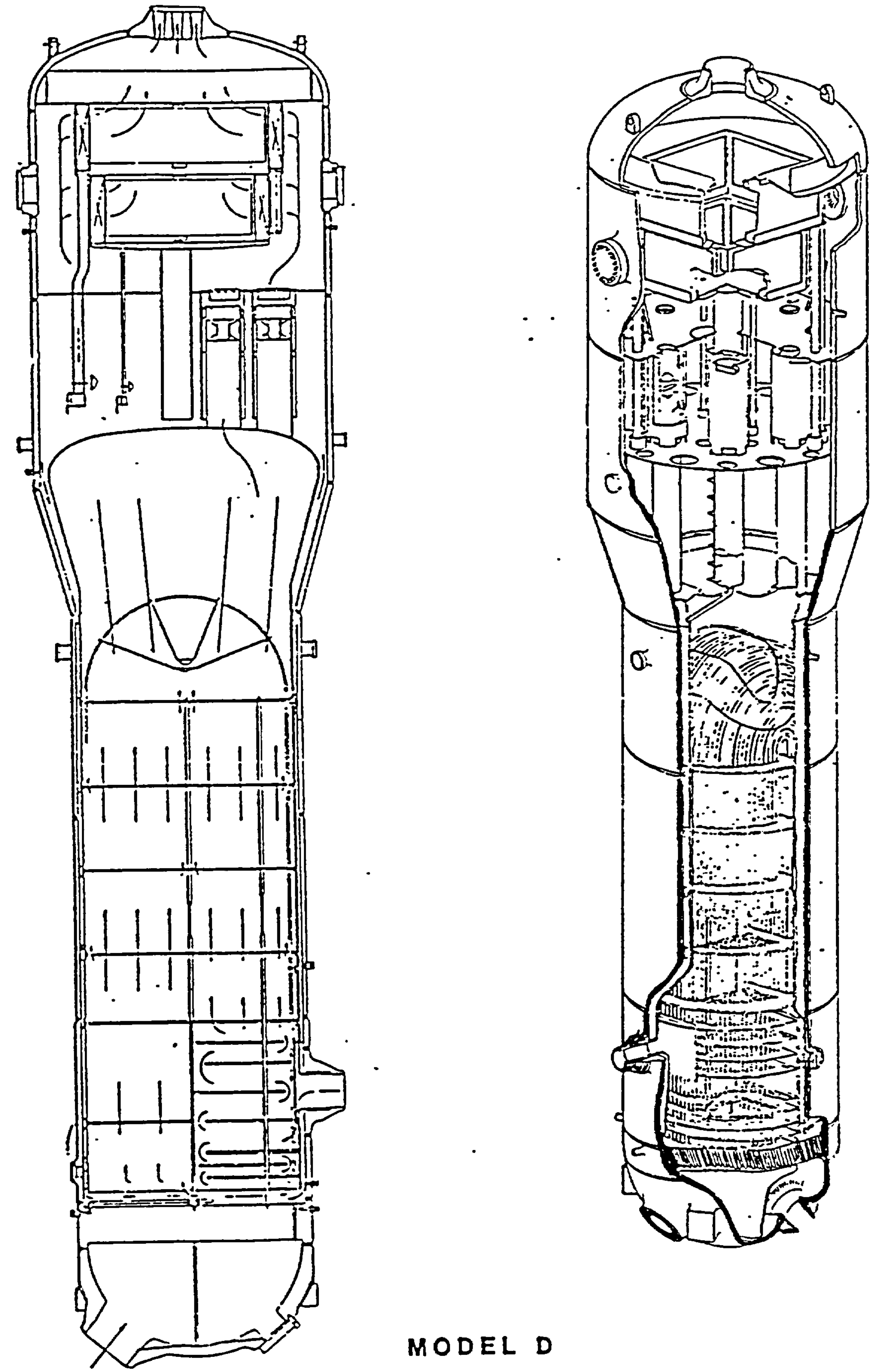
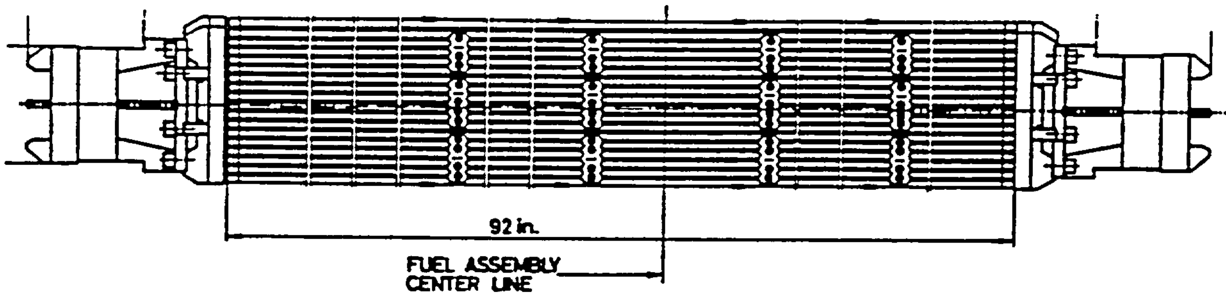


Figure 4.1-2. Westinghouse U-Tube Steam Generator (Typical of Series 44 and 51 and Model F)



MODEL D

STEAM GENERATOR



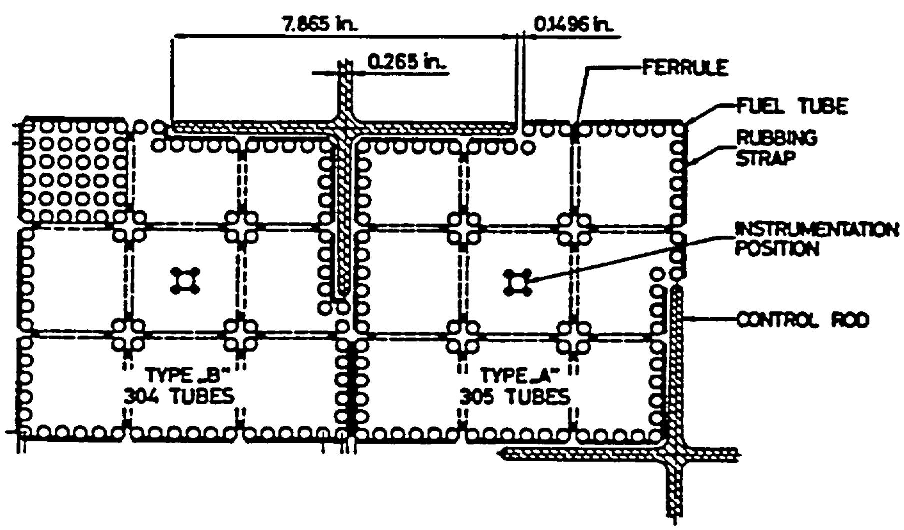


Figure 4.1-5. General Arrangement of the Yankee-Rowe 9 X (6 X 6) Fuel Assembly

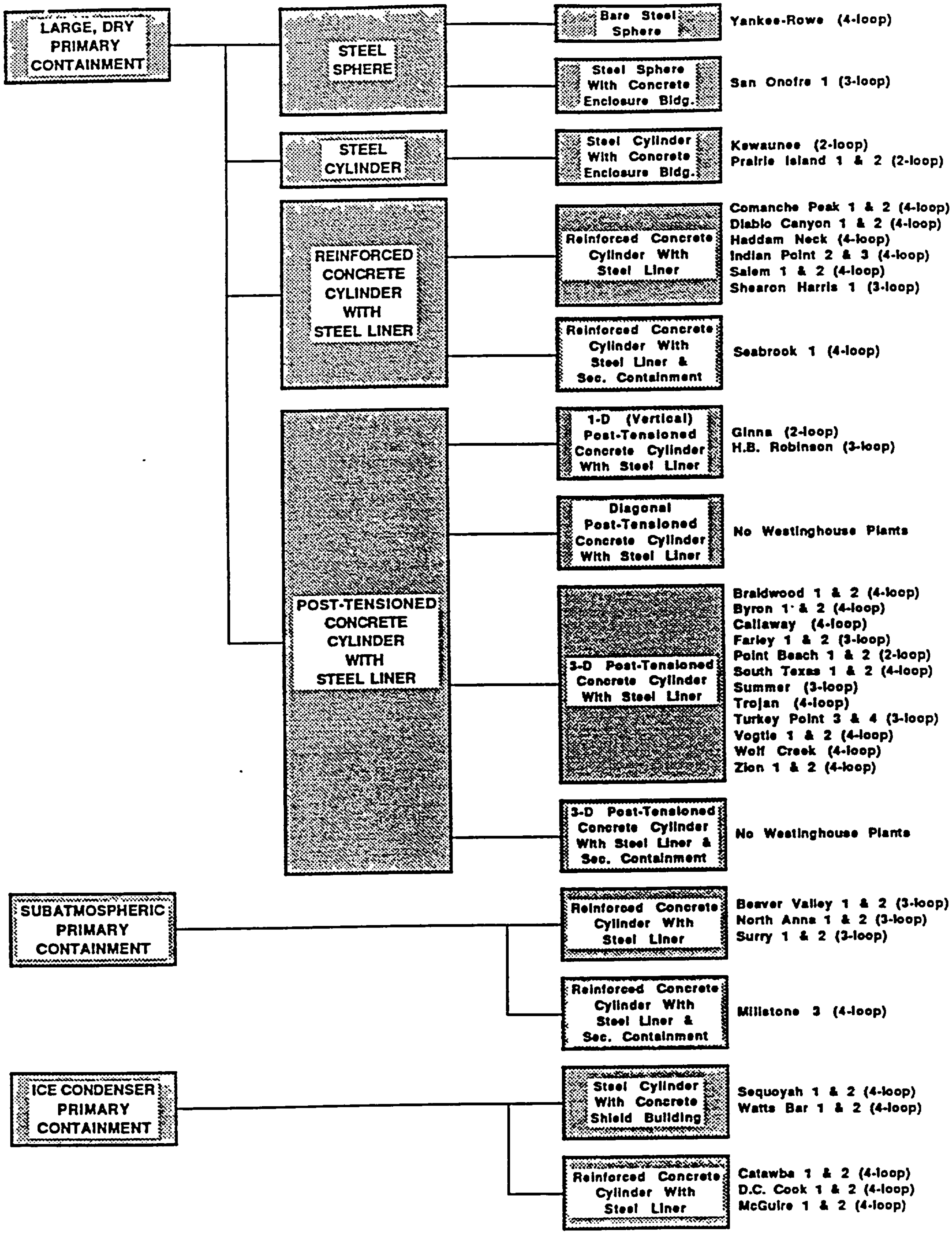


Figure 4.1-12. Distribution of Containment Types for Westinghouse Reactors

4.2 2-Loop Westinghouse PWRs

Westinghouse 2-loop plants in the United States include the following:

- Ginna
- Kewaunee
- Point Beach 1 and 2
- Prairie Island 1 and 2

All of these plants had full power operating licenses as of 2/89.

4.2.1 Reactor Core and Fuel Assemblies

All 2-loop Westinghouse plants have cores comprised of 121 fuel assemblies that yield power levels in the range from 1518 to 1650 MWt. All 2-loop plants use 14 x 14 fuel assemblies and all have power densities in the range from 87 to 96 kW/liter.

4.2.2 Reactivity Control System

Typically there are 29 to 33 rod cluster control assemblies (RCCAs) in 2-loop plants.

4.2.3 Reactor Coolant System

The general configuration of a 2-loop Westinghouse reactor coolant system is shown in Figure 4.2-1. All 2-loop plants have a reactor vessel with a 132 inch inside diameter.

4.2.4 Steam Generators

Ginna and Point Beach 1 and 2 use the intermediate-size 44-series steam generators while Kewaunee and Prairie Island 1 and 2 use the large 51-series steam generators.

4.2.5 Shutdown Cooling Systems

Shutdown cooling is accomplished by the Residual Heat Removal (RHR) system which also operates in the Low-Pressure Safety Injection (LPSI) mode as part of the ECCS.

4.2.6 Emergency Core Cooling Systems

A representative ECCS for a 2-loop Westinghouse PWR is comprised of the following subsystems:

- Three High-Pressure Safety Injection (HPSI) pump trains that inject into the cold legs via a boron injection tank.
- Two RHR trains which perform the Low-Pressure Safety Injection (LPSI) function, each with a pump and heat exchanger
- Three Safety Injection Accumulators, each connected to an RCS cold leg

The shutoff head of the HPSI pumps is about 1750 psig, therefore, these pumps are unable to provide makeup to the RCS at normal operating pressure. In the event of a small LOCA which leaves the RCS at high pressure, it is necessary to first depressurize the RCS before the HPSI pumps can provide makeup. Depressurization can be accomplished by heat transfer from the RCS to the steam generators or by opening the power-operated relief valves on the pressurizer.

The HPSI pumps cannot be aligned to take a suction directly on the containment sump. If needed, the RHR pumps can be aligned in tandem with the HPSI pumps for high-pressure recirculation.

The normal charging system which has three low-capacity positive displacement pumps, is not part of the ECCS.

4.2.7 Containment

All Westinghouse 2-loop plants have large, dry containments of various designs, as summarized below.

Containment Construction	Applicable Plants
- Steel cylinder with concrete enclosure building	Kewaunee Prairie Island 1 and 2
- One-dimension (vertical) post-tensioned concrete cylinder with a steel liner	Ginna
- Three-dimension post-tensioned concrete cylinder with a steel liner	Point Beach 1 and 2

The general arrangement of the Ginna containment is shown in Figure 4.2-2 (section views) and Figures 4.2-3 and 4.2-4 (plan views).

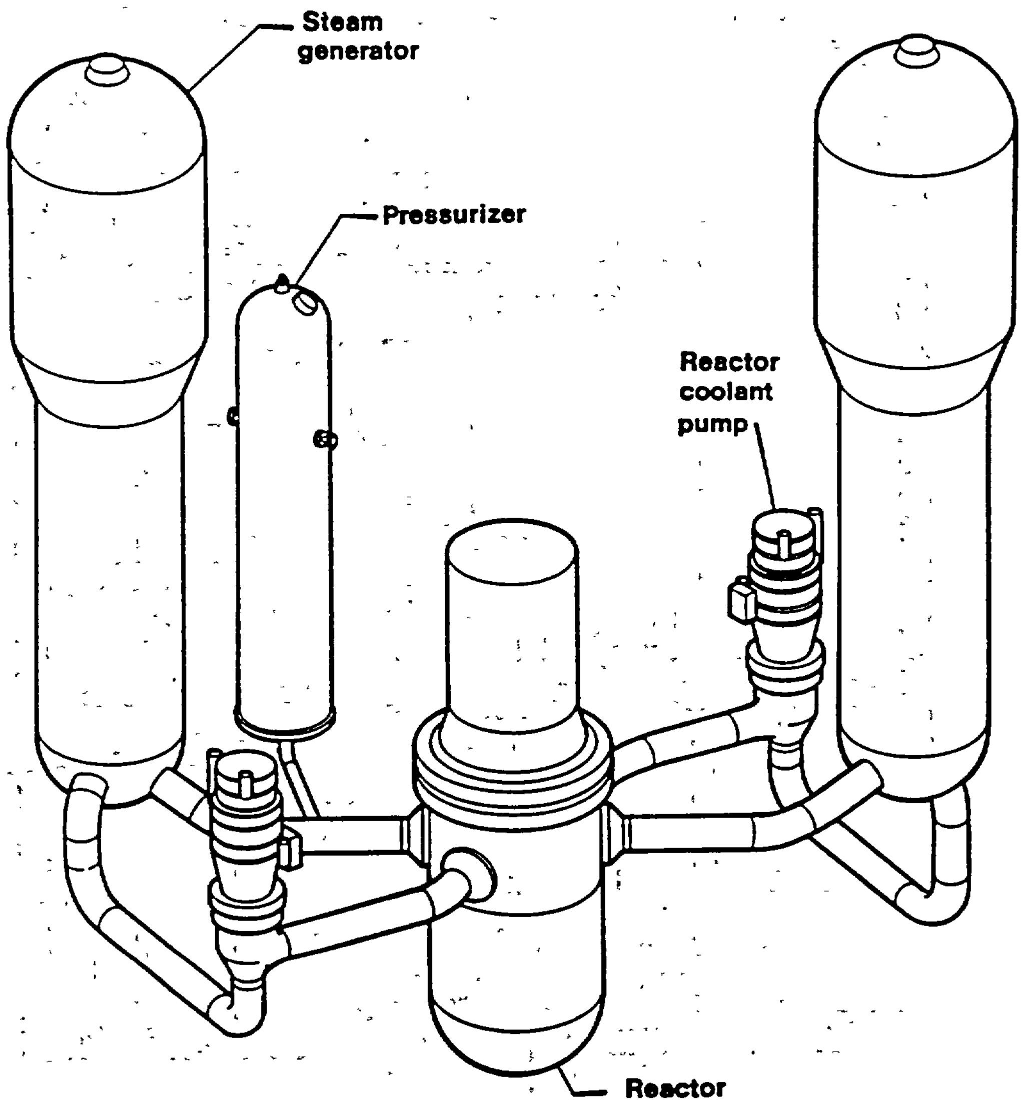


Figure 4.2-1. Westinghouse 2-Loop NSSS

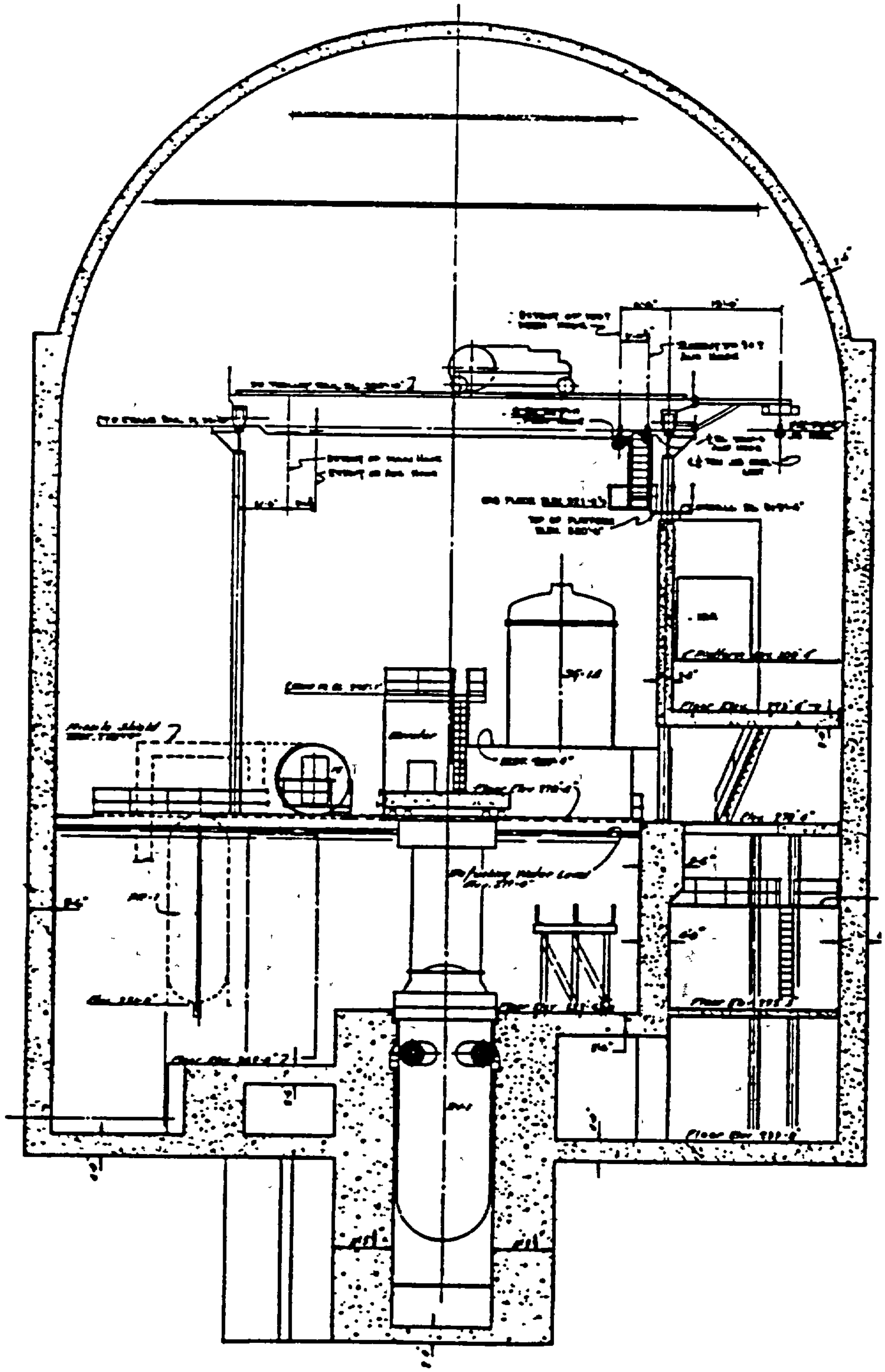


Figure 4.2-2. Section Views of the Ginna Large, Dry Containment (Sheet 1 of 2)

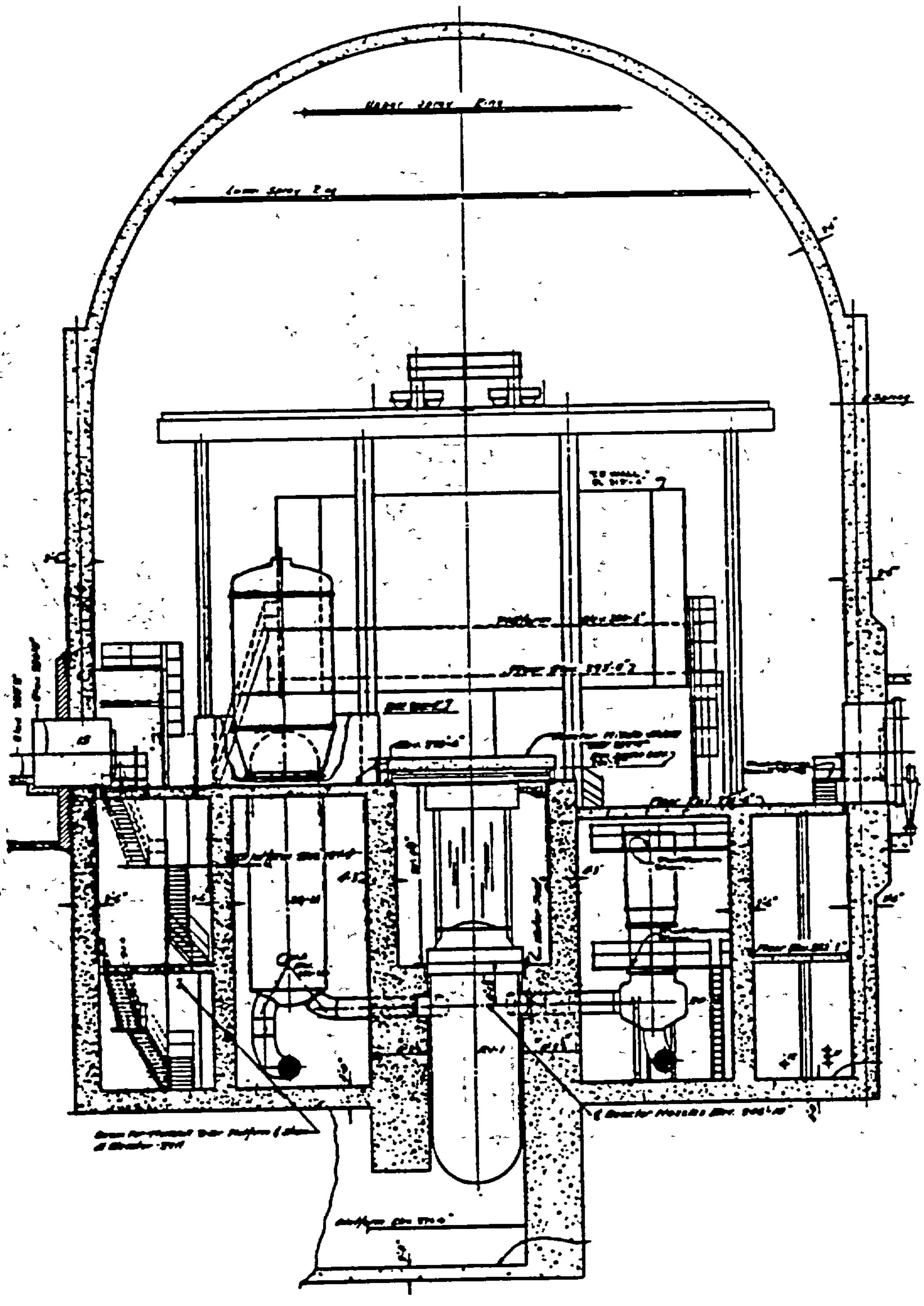


Figure 4.2-2. Section Views of the Ginna Large, Dry Containment (Sheet 2 of 2)

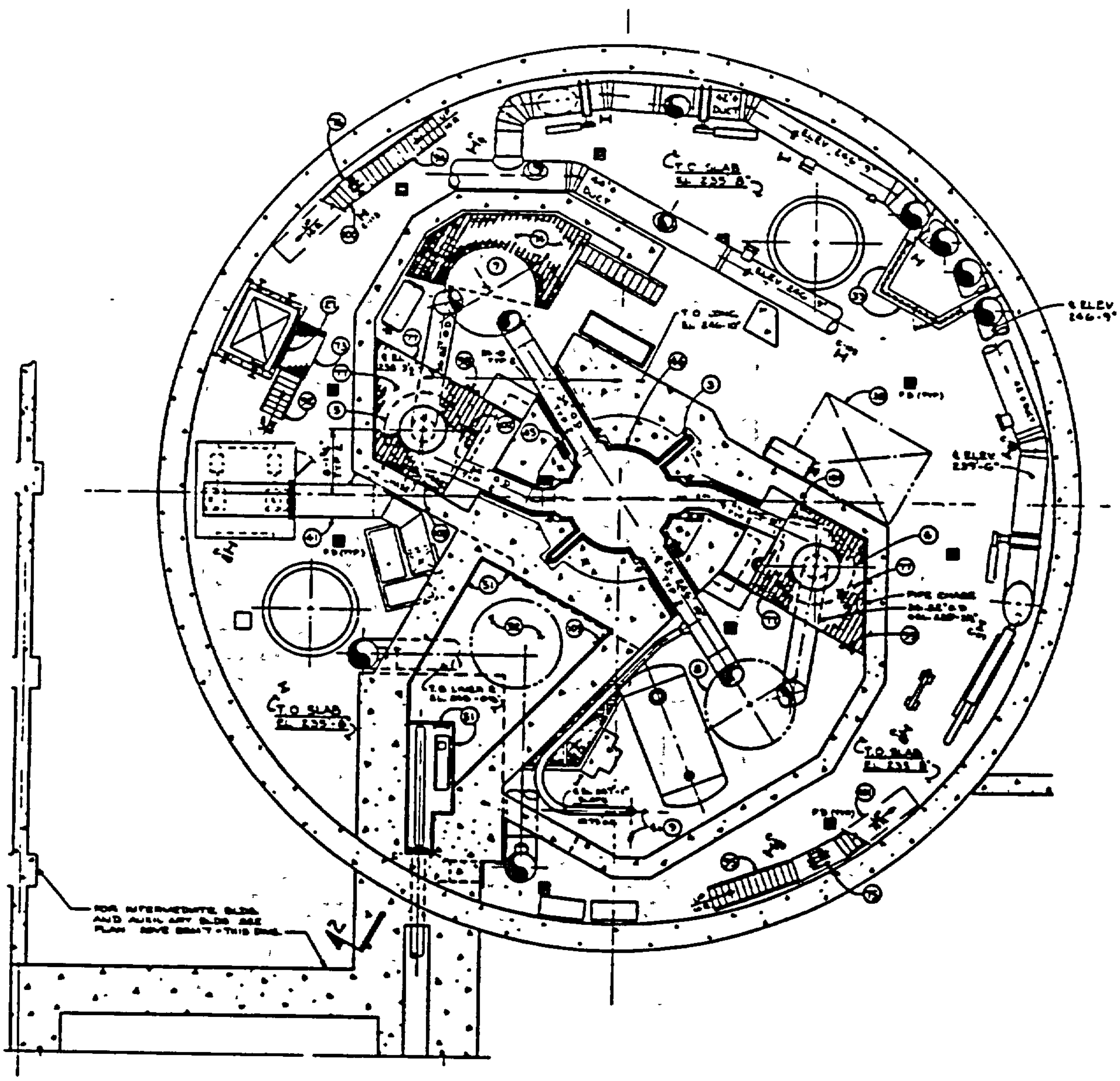


Figure 4.2-3. Plan View of the Ginna Large, Dry Containment Below the Elevation of the Operating Floor

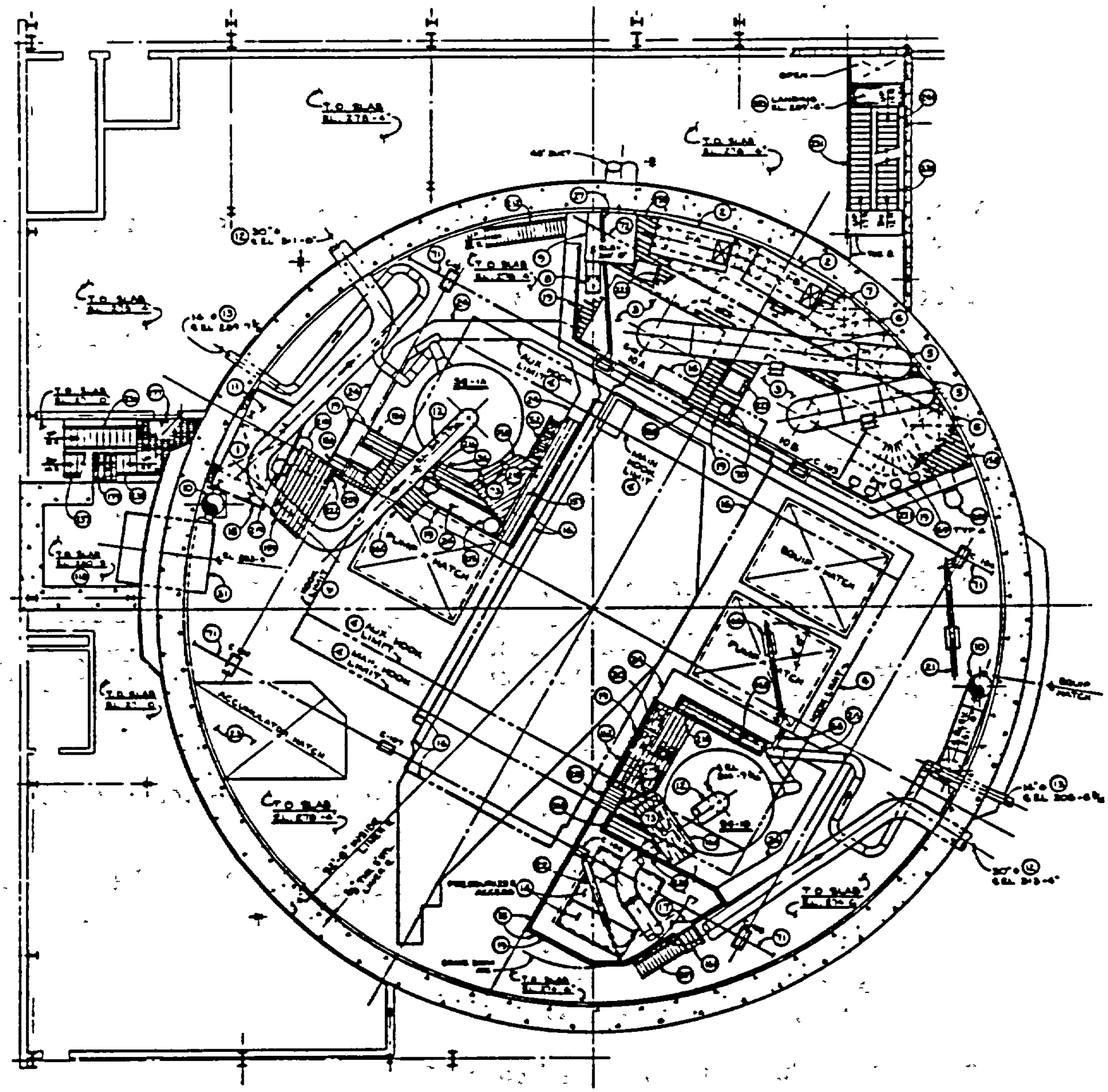


Figure 4.2-4. Plan View of the Ginna Large, Dry Containment, Above the Operating Floor

4.3 3-Loop Westinghouse PWRs

Westinghouse 3-loop plants in the United States include the following:

- Beaver Valley 1 and 2
- Farley 1 and 2
- H.B. Robinson 2
- North Anna 1 and 2
- San Onofre 1
- Shearon Harris 1
- Summer
- Surry 1 and 2
- Turkey Point 3 and 4

All of these plants had full power operating licenses as of 2/89 except San Onofre 1 which had a provisional operating license.

4.3.1 Reactor Core and Fuel Assemblies

The 3-loop Westinghouse plants can be grouped into the following categories based on core thermal output:

- 1347 MWt	San Onofre 1
- 2208 to 2441 MWt	Robinson, Surry 1 and 2, and Turkey Point 3 and 4
- 2660 to 2775 MWt	Beaver Valley 1 and 2, Farley 1 and 2, North Anna 1 and 2, Shearon Harris 1, and Summer

All 3-loop cores are comprised of 157 fuel assemblies. San Onofre 1 uses 14 x 14 fuel assemblies and has an average power density of about 70 kW/liter. The intermediate power level group uses 15 x 15 fuel assemblies and has power densities in the range from 82 to 92 kW/liter. Plants in the high power group use 17 x 17 fuel assemblies to yield power densities of 100 to 108 kW/liter.

4.3.2 Reactivity Control System

The number of full-length and part-length RCCAs for selected 3-loop Westinghouse plants are summarized below:

Plant	Fuel Elements	Full-Length RCCAs	Part-Length RCCAs
Robinson	157	45	8
Surry	157	48	5
Beaver Valley 1 and 2	157	48	5

A more complete listing is provided in Section 4.5.

4.3.3 Reactor Coolant System

The general configuration of a 3-loop Westinghouse reactor coolant system is shown in Figure 4.3-1. Most 3-loop plants have a reactor vessel with a 156 to 159 inch inside diameter. The only exceptions are the following:

- 144 inch vessel	San Onofre 1
- 172 inch vessel	Summer and Turkey Point 3 and 4

4.3.4 Steam Generators

Three vintages of steam generators can be found among the 3-loop Westinghouse plants, as follows:

- 27-series	San Onofre 1
- 44-series	Robinson and Turkey Point 3 and 4
- 51-series	Beaver Valley 1 and 2, Farley 1 and 2, North Anna 1 and 2, Shearon Harris 1, Summer, and Surry 1 and 2

A comparison of Westinghouse steam generator design parameters is included in Section 4.1.

4.3.5 Shutdown Cooling Systems

For the following 3-loop plants, shutdown cooling is accomplished by the Residual Heat Removal (RHR) system which also operates in the Low-Pressure Safety Injection (LPSI) mode as part of the ECCS:

- Farley 1 and 2
- Robinson
- Shearon Harris 1
- Summer
- Turkey Point 3 and 4

The San Onofre 1 and the plants with subatmospheric containment (Beaver Valley, North Anna, and Surry) appear to have separate RHR systems.

4.3.6 Emergency Core Cooling Systems

In most of the 3-loop Westinghouse plants, the centrifugal charging pumps perform the HPSI function and are capable of providing RCS makeup at the PORV setpoint pressure. These pumps inject into the cold legs via a boron injection tank. Plants in this category are:

- Beaver Valley 1 and 2
- Farley 1 and 2
- North Anna 1 and 2
- Shearon Harris 1
- Summer
- Surry 1 and 2

The remaining 3-loop plants (Robinson, San Onofre 1, and Turkey Point 3 and 4) have separate HPSI pumps and positive displacement charging pumps. The shutoff head of the HPSI pumps is about 1700 psig, therefore, these pumps are unable to provide makeup to the RCS at normal operating pressure. In the event of a small LOCA which leaves the RCS at high pressure, it is necessary to first depressurize the RCS before the HPSI pumps can provide makeup. Depressurization can be accomplished by heat transfer from the RCS to the steam generators or by opening the power-operated relief valves on the pressurizer.

Two different low-pressure ECCS subsystems are found in the 3-loop plants based on containment design. In the plants with a large, dry containment, the LPSI function is performed by the RHR system. In some of the subatmospheric containment plants, the low pressure injection/recirculation system is separate from the RHR system and does not include heat exchangers in the flow path to the RCS.

The centrifugal charging pumps, and the separate HPSI pumps, cannot be aligned to take a suction directly on the containment sump. If needed, the low-pressure ECCS subsystem can be aligned in tandem with the high-pressure ECCS subsystem for high-pressure recirculation.

4.3.7 Containment

Westinghouse 3-loop plants have either large, dry containments or subatmospheric containments of various designs, as described below.

A. Large, Dry Containment

Eight of fourteen 3-loop plants have large, dry containments. The types of construction used in these containments is summarized below.

Containment Construction	Applicable Plants
Steel sphere with concrete enclosure building	San Onofre 1
Reinforced concrete cylinder with a steel liner	Shearon Harris 1
One-dimension (vertical) post-tensioned concrete cylinder with a steel liner	Robinson
Three-dimension post-tensioned concrete cylinder with a steel liner	Farley 1 and 2 Summer Turkey Point 3 and 4

Examples of large, dry containments for 3-loop Westinghouse plants are shown in Figures 4.3-2 and 4.3-3 (Robinson) and Figures 4.3-4 and 4.3-5 (Summer).

B. Subatmospheric Containment

The following six of the fourteen 3-loop plants have subatmospheric cylindrical containments constructed of reinforced concrete with a steel liner.

- Beaver Valley 1 and 2
- North Anna 1 and 2
- Surry 1 and 2

Only one other PWR in the U.S. has a subatmospheric containment: Millstone 3, a 4-loop Westinghouse plant. The North Anna subatmospheric containment is shown in Figures 4.3-6 and 4.3-7.

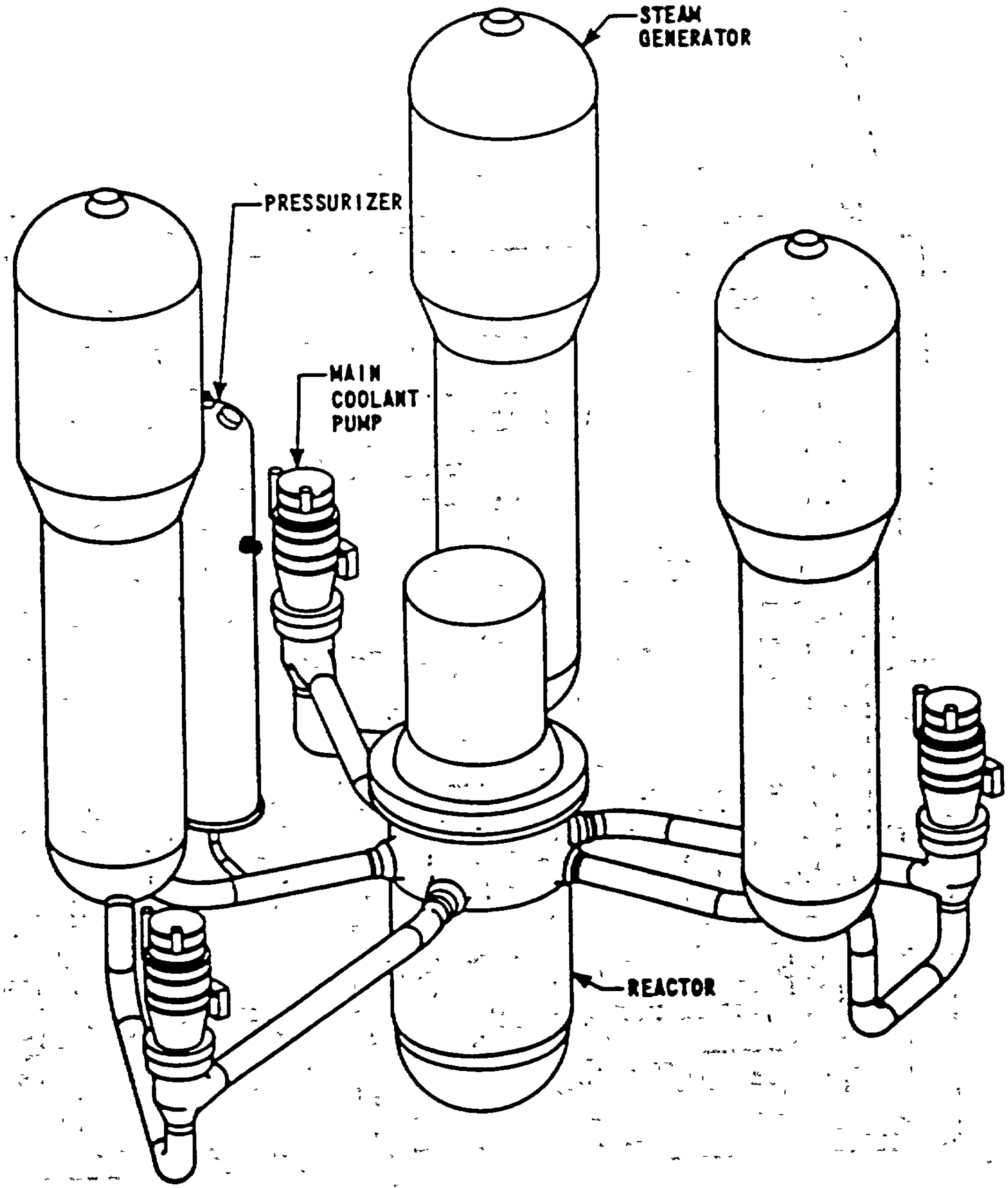


Figure 4.3-1. Westinghouse 3-Loop NSSS

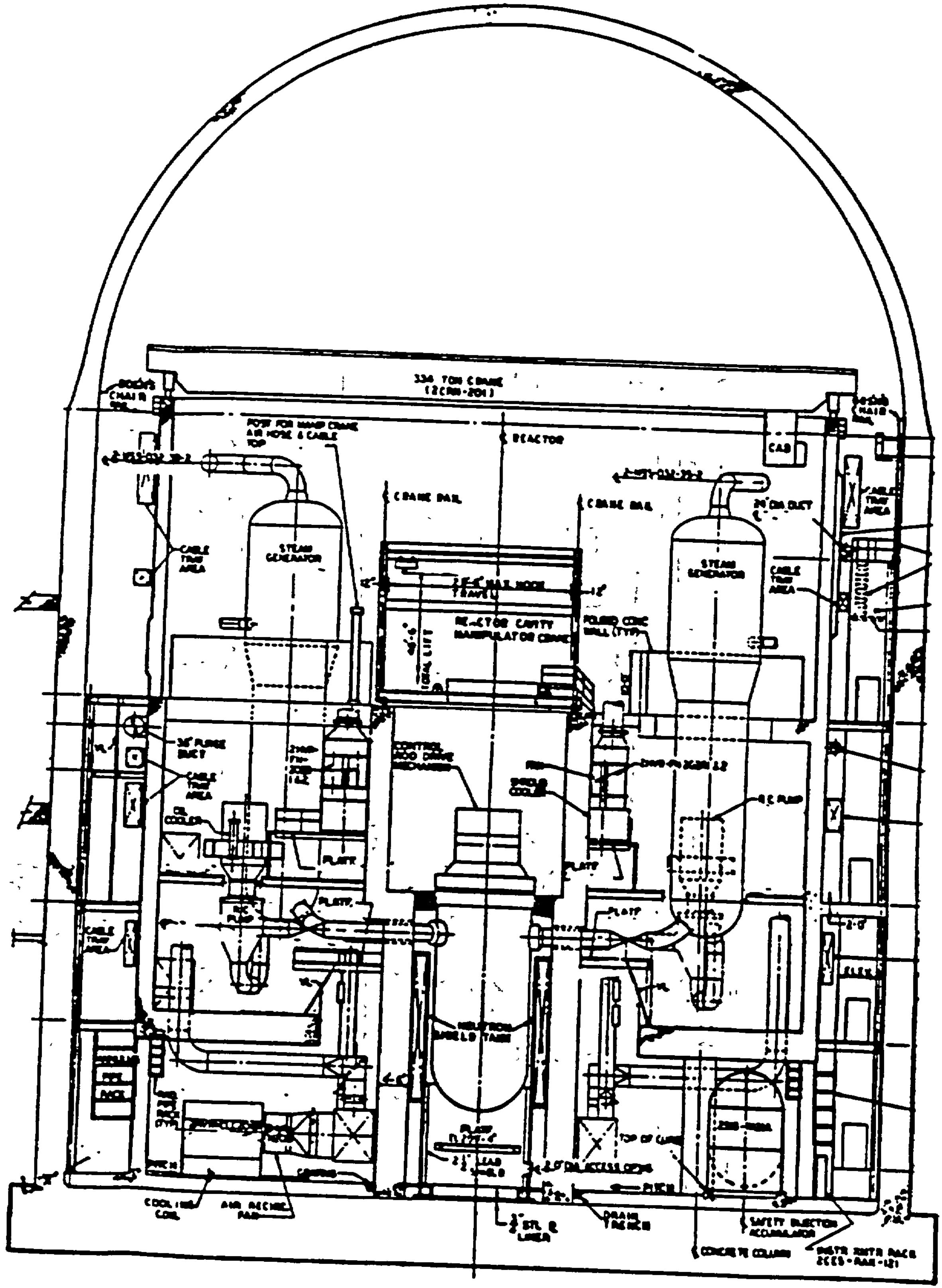


Figure 4.3-2. Section View of the H.B. Robinson Large, Dry Containment (1-D Post-Tensioned Concrete)

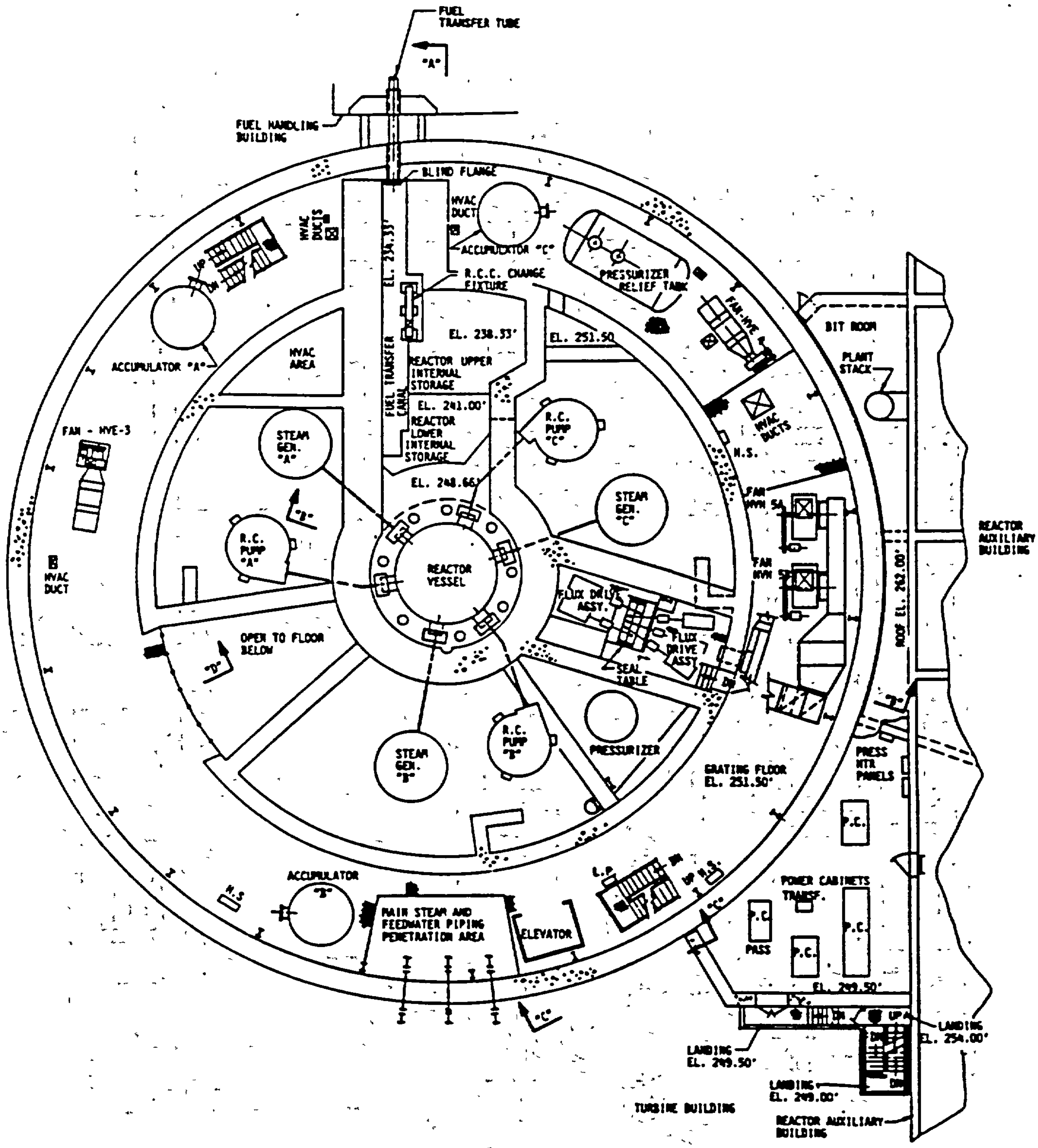


Figure 4.3-3. Plan View of the H.B. Robinson Large, Dry Containment (1-D Post-Tensioned Concrete)

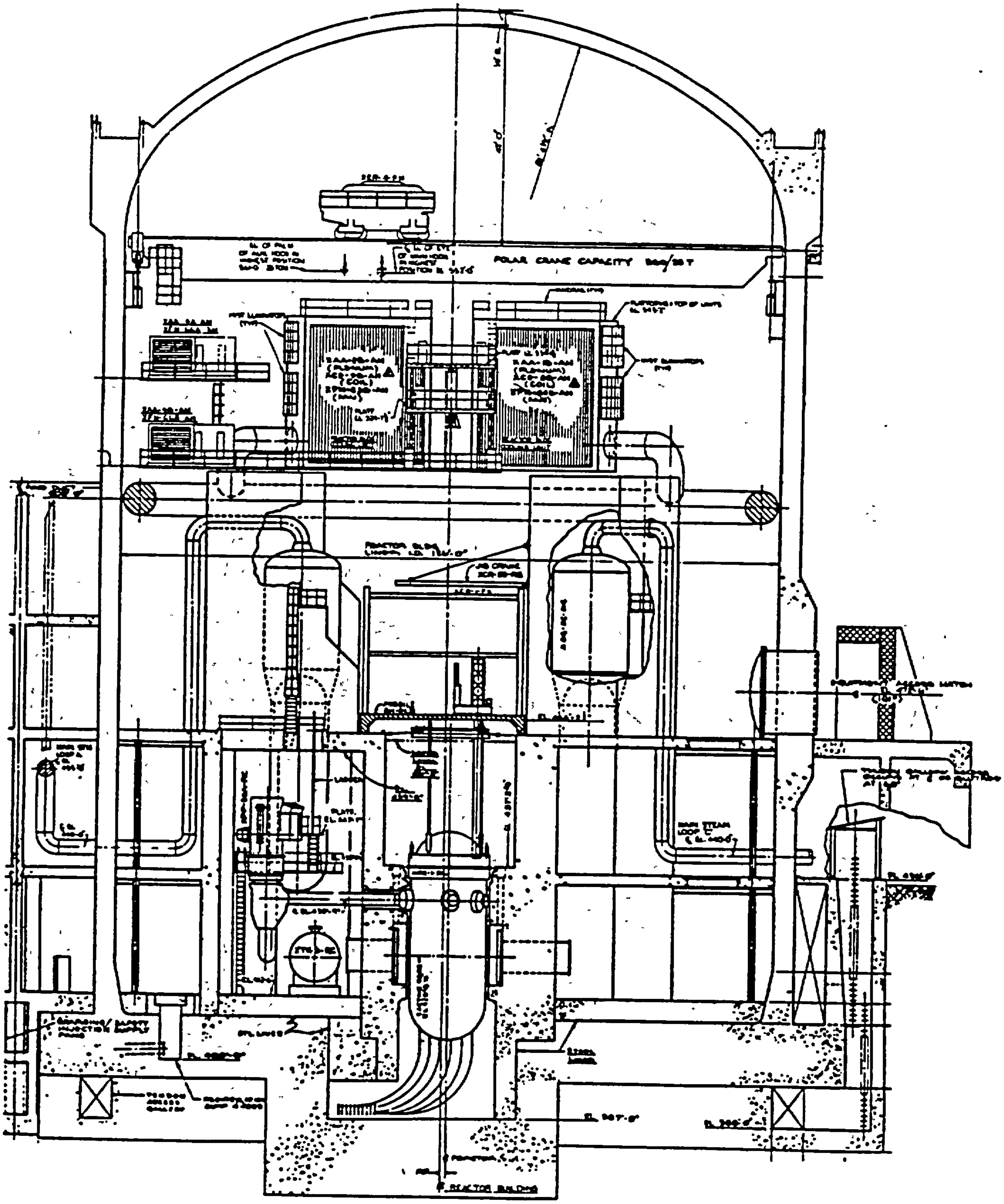


Figure 4.3-4. Section Views of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete) (Sheet 1 of 2)

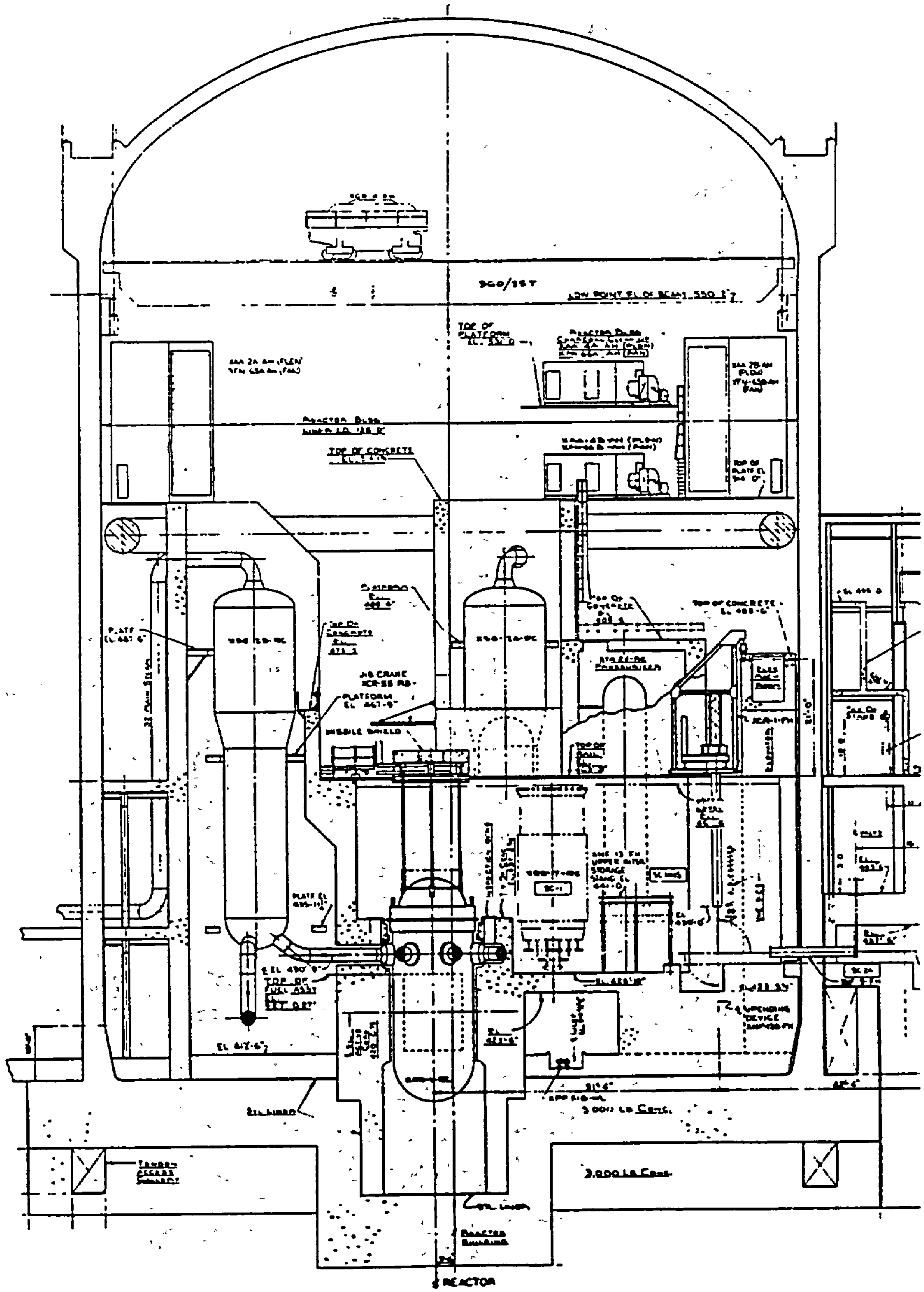


Figure 4.3-4. Section Views of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete) (Sheet 2 of 2)

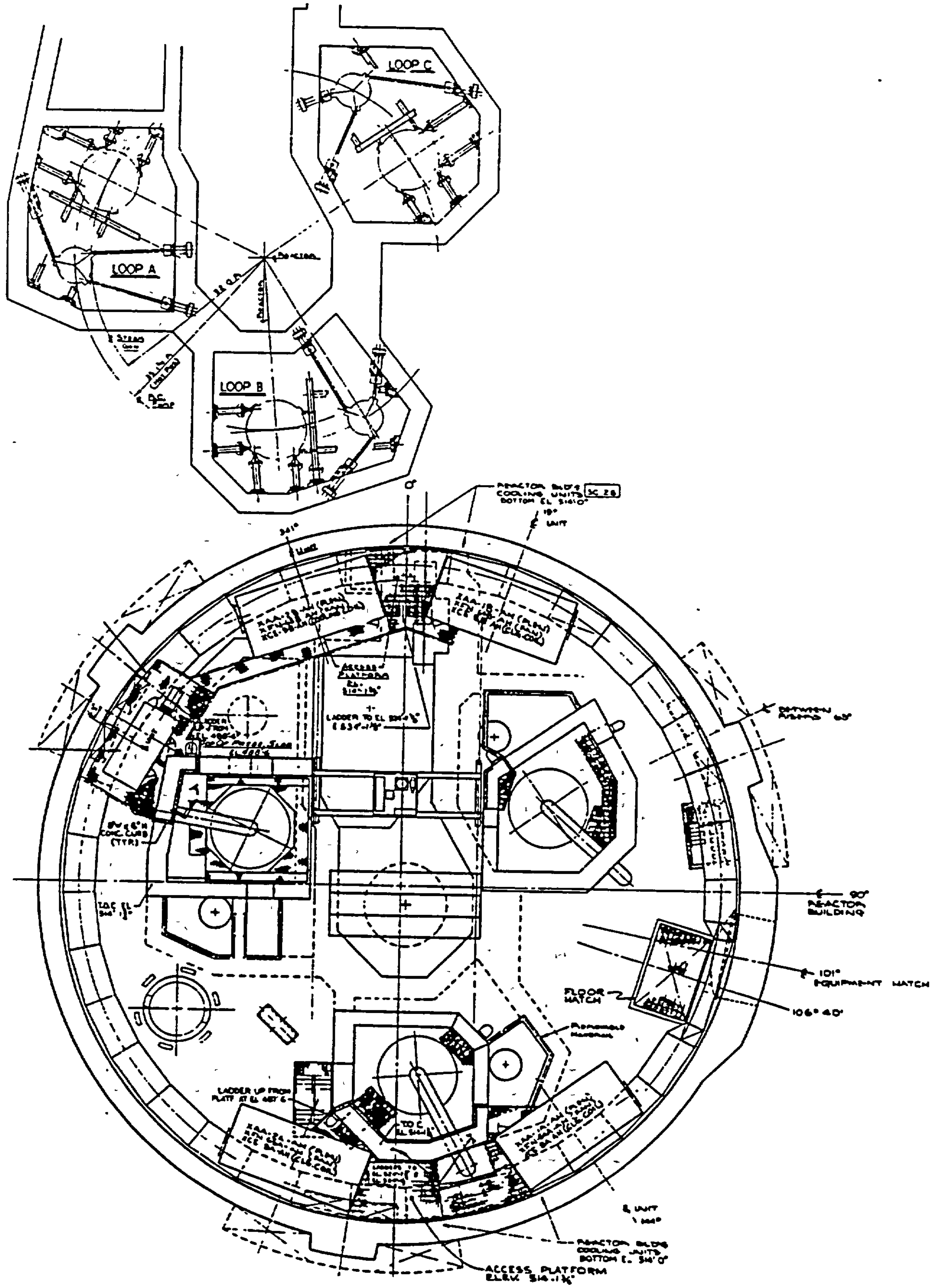


Figure 4.3-5. Plan View of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete)

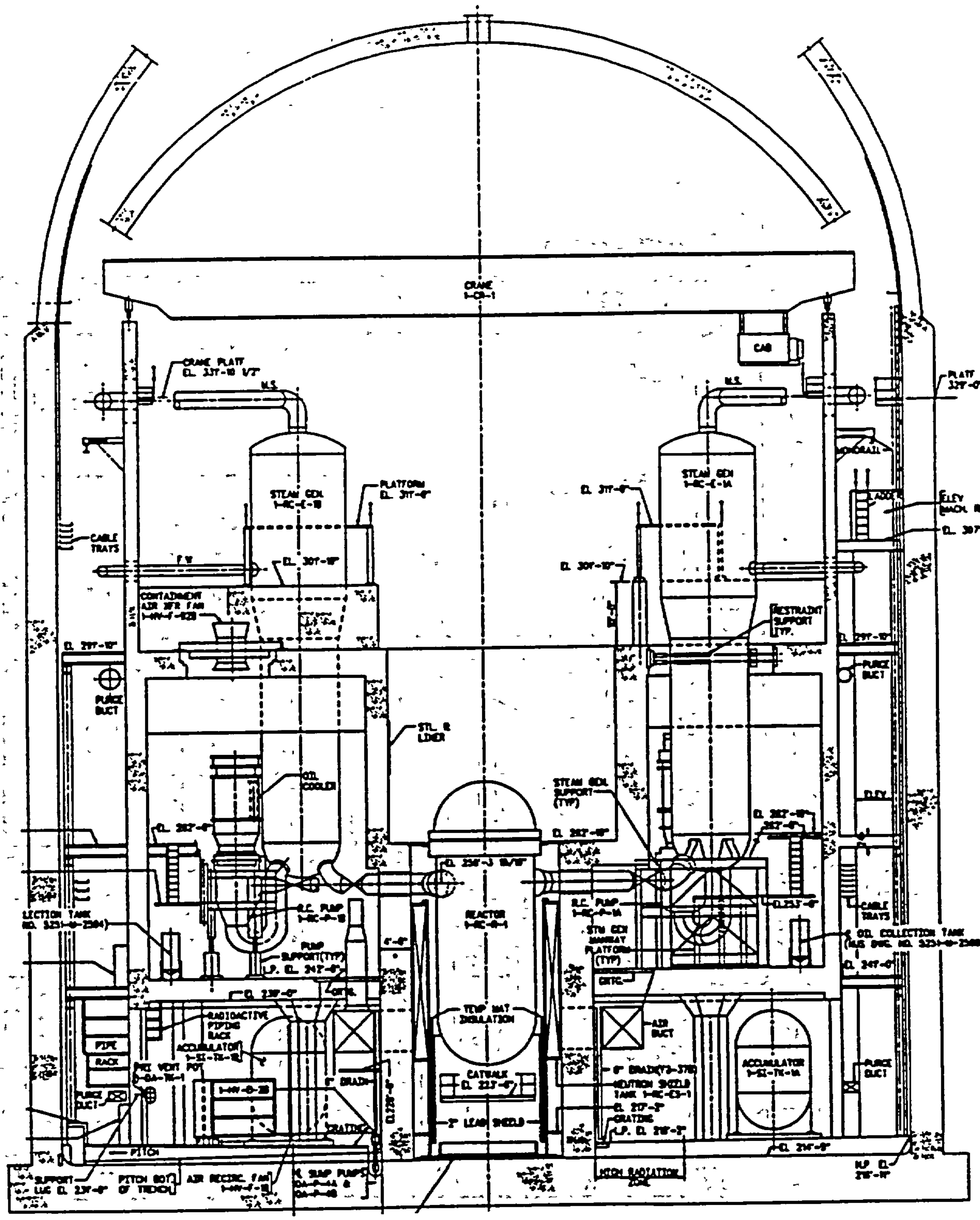


Figure 4.3-6. Section Views of the North Anna Subatmospheric Containment (Sheet 1 of 2)

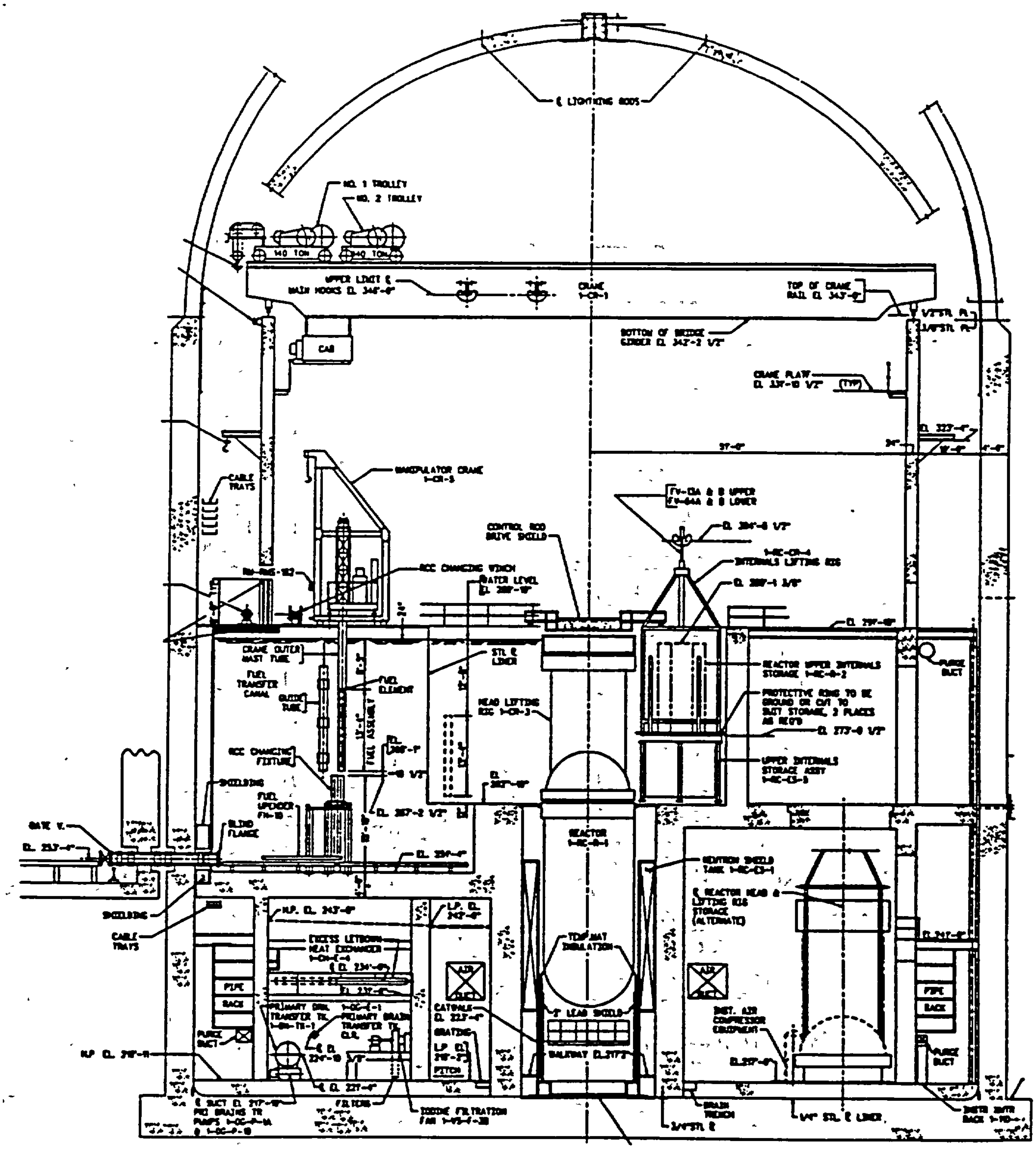


Figure 4.3-6. Section Views of the North Anna Subatmospheric Containment (Sheet 2 of 2)

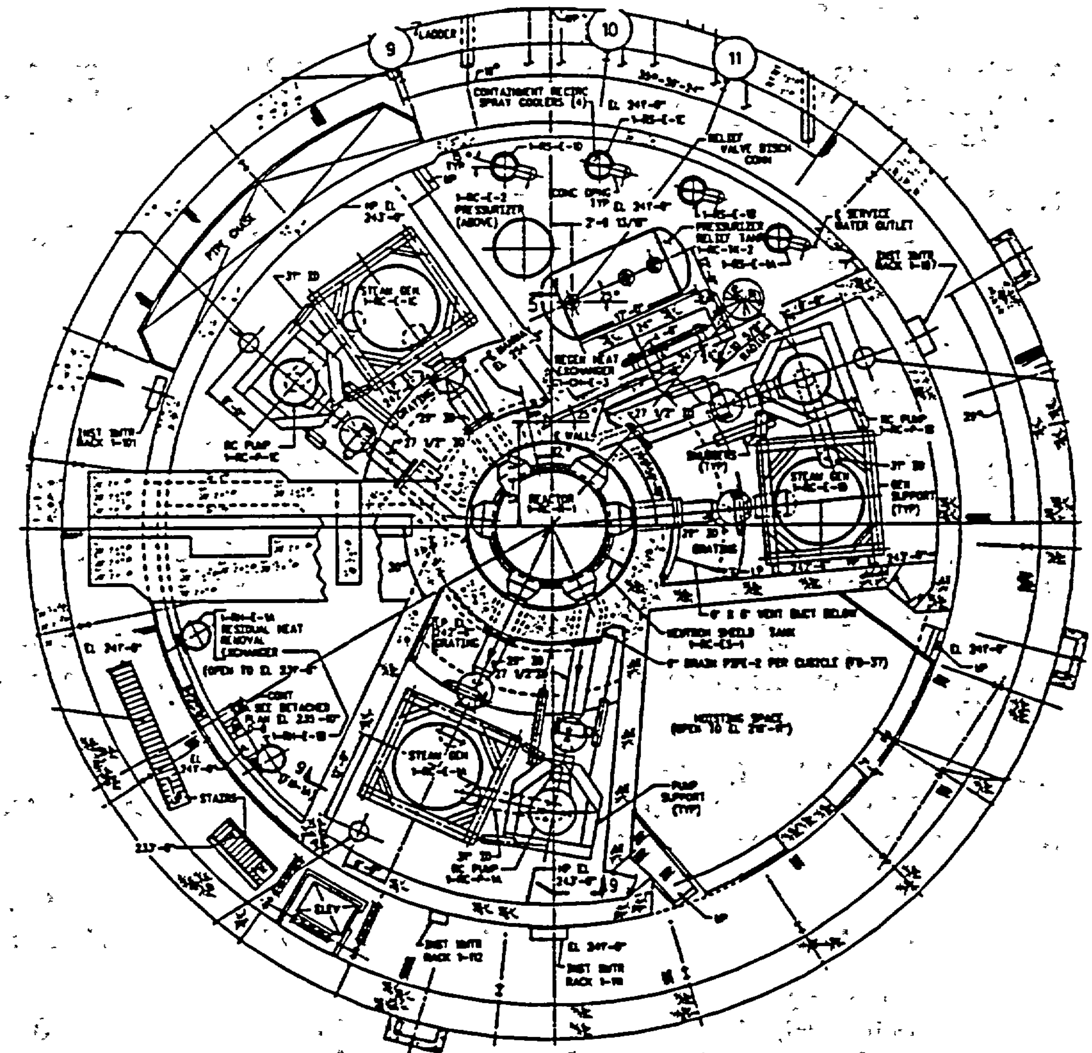


Figure 4.3-7. Plan View of the North Anna Subatmospheric Containment

4.4 4-Loop Westinghouse PWRs

Westinghouse 4-loop plants in the United States include the following:

- Braidwood 1 and 2
- Byron 1 and 2
- Callaway
- Catawba 1 and 2
- Comanche Peak 1 and 2
- D.C. Cook 1 and 2
- Diablo Canyon 1 and 2
- Haddam Neck
- Indian Point 2 and 3
- McGuire 1 and 2
- Millstone 3
- Salem 1 and 2
- Seabrook
- South Texas 1 and 2
- Trojan
- Vogtle 1 and 2
- Watts Bar 1 and 2
- Yankee-Rowe
- Zion 1 and 2

All of these plants had full power operating licenses as of 2/89 except for Comanche Peak 1 and 2, Seabrook, South Texas 2, Vogtle 2, and Watts Bar 1 and 2. The Westinghouse 4-loop NSSS is shown in Figure 4.4-1.

4.4.1 Reactor Core and Fuel Assemblies

The 4-loop Westinghouse plants can be grouped into the following categories based on core thermal output:

- 600 MWt — Yankee-Rowe
- 1825 MWt — Haddam Neck
- 2758 to 3817 MWt — All other 4-loop plants

The Yankee-Rowe core is comprised of 76 unique 9 x (6 x 6) fuel assemblies and uses cruciform control rods as shown in Figure 4.4-2. The majority of the other 4-loop plants use 17 x 17 fuel assemblies, with a few still using 15 x 15 fuel assemblies. The Haddam Neck core has 157 fuel assemblies and all others have 193 fuel assemblies. The general arrangement of a 193 fuel assembly core is shown in Figure 4.4-3.

The average power densities of the earlier 4-loop plants (i.e., Yankee-Rowe, Haddam Neck, and Indian Point 2 and 3) are in the range from 82 to 93 kW/liter. All of the later plants using the 17 x 17 fuel elements have power densities in the range from 98 to 109 kW/liter.

4.4.2 Reactivity Control System

The number of full-length and part-length RCCAs for selected 4-loop Westinghouse plants are summarized below:

Plant	Fuel Elements	Full-Length RCCAs	Part-Length RCCAs
Haddam Neck	157	45	0
Indian Point 3	193	53	0
McGuire 1 and 2	193	53	8
Seabrook 1	193	58	0

A more complete listing is provided in Section 4.5.

Yankee-Rowe is the only Westinghouse plant to use cruciform control rods. There are 24 cruciform control rods operated by magnetic jack CRDMs plus 8 cruciform fixed "shim" rods. The position of the shim rods can be changed only during refueling.

4.4.3 Reactor Coolant System

The general configuration of a 4-loop Westinghouse reactor coolant system is shown in Figure 4.4-1. Most 4-loop plants have a reactor vessel with a 173 inch inside diameter. The only exceptions are the following:

- 109 inch vessel	Yankee-Rowe
- 154 inch vessel	Haddam Neck
- 167 inch vessel	Comanche Peak

4.4.4 Steam Generators

All vintages of steam generators can be found among the 4-loop Westinghouse plants, as follows:

- 27-series	Yankee-Rowe
- 44-series	Indian Point 2 and 3
- 51-series	Most other 4-loop plants
- Model F	Seabrook, Wolf Creek, Callaway (late-model 4-loop plants)

A comparison of Westinghouse steam generator design parameters is included in Section 4.1.

4.4.5 Shutdown Cooling Systems

In most of the 4-loop Westinghouse plants, shutdown cooling is accomplished by the Residual Heat Removal (RHR) system which also operates in the Low-Pressure Safety Injection (LPSI) mode as part of the ECCS.

4.4.6 Emergency Core Cooling Systems

Almost all 4-loop Westinghouse plants have two high-pressure ECCS subsystems; the high-pressure safety injection (HPSI) system and the centrifugal charging pumps which are part of the ECCS. In these plants, the shutoff head of the HPSI pumps typically is on the order of 1600 to 1700 psig. Each centrifugal charging pump pump can provide approximately 150 gpm makeup to the RCS at the PORV setpoint pressure.

The only exceptions to this high-pressure injection capability are the Yankee-Rowe and Indian Point 2 and 3 plants. These plants have an intermediate-pressure HPSI system and low-capacity positive displacement charging pumps which are not part of the ECCS. In the event of a small LOCA which leaves the RCS at high pressure, it is necessary to first depressurize the RCS before the HPSI pumps can provide makeup. Depressurization can be accomplished by heat transfer from the RCS to the steam generators or by opening the power-operated relief valves on the pressurizer.

4.4.7 Containment

Westinghouse 4-loop plants have either large, dry containments, a subatmospheric containment, or ice condenser containments of various designs, as described below. The ice condenser containment is unique to Westinghouse 4-loop PWRs.

A. Large, Dry Containment

Twenty-four of thirty-five 4-loop plants have large, dry containments. The types of construction used in these containments is summarized below.

Containment Construction	Applicable Plants
Bare steel sphere	Yankee-Rowe
Reinforced concrete cylinder with a steel liner	Comanche Peak 1 and 2 Diablo Canyon 1 and 2 Haddam Neck Indian Point 2 and 3 Salem 1 and 2
Reinforced concrete cylinder with a steel steel liner and secondary containment	Seabrook 1
Three-dimension post-tensioned concrete cylinder with a steel liner	Braidwood 1 and 2 Byron 1 and 2 Callaway South Texas 1 and 2 Trojan Vogtle 1 and 2 Wolf Creek Zion 1 and 2

The Yankee-Rowe steel sphere large, dry containment is shown in Figures 4.4-4 and 4.4-5. A reinforced concrete large, dry containment (Diablo Canyon) is shown in Figures 4.4-6 and 4.4-7. The South Texas three-dimension post-tensioned concrete large, dry containment is shown in Figures 4.4-8 and 4.4-9.

B. Subatmospheric Containment

Millstone 3 is the only 4-loop Westinghouse PWR with a subatmospheric containment. The Millstone 3 containment is shown in Figure 4.4-10. Construction is of reinforced concrete cylinder with a steel liner and a secondary containment. All other subatmospheric containments in the U.S are found in Westinghouse 3-loop plants.

C. Ice Condenser (Pressure Suppression) Containment

Ice condenser containments are unique to 4-loop Westinghouse PWRs, and ten of the thirty-five 4-loop plants have this type of containments. The types of construction used in these containments is summarized below.

Containment Construction	Applicable Plants
Steel cylinder with concrete shield building	Catawba 1 and 2 Sequoyah 1 and 2 Watts Bar 1 and 2
Reinforced concrete cylinder with steel liner	D.C. Cook 1 and 2 McGuire 1 and 2

The Catawba ice condenser containment is shown in Figures 4.4-11 and 4.4-12, and the Watts Bar containment is shown in Figures 4.4-13 to 4.4-15. Additional details on the configuration of the ice condenser units can be found in Section 3.

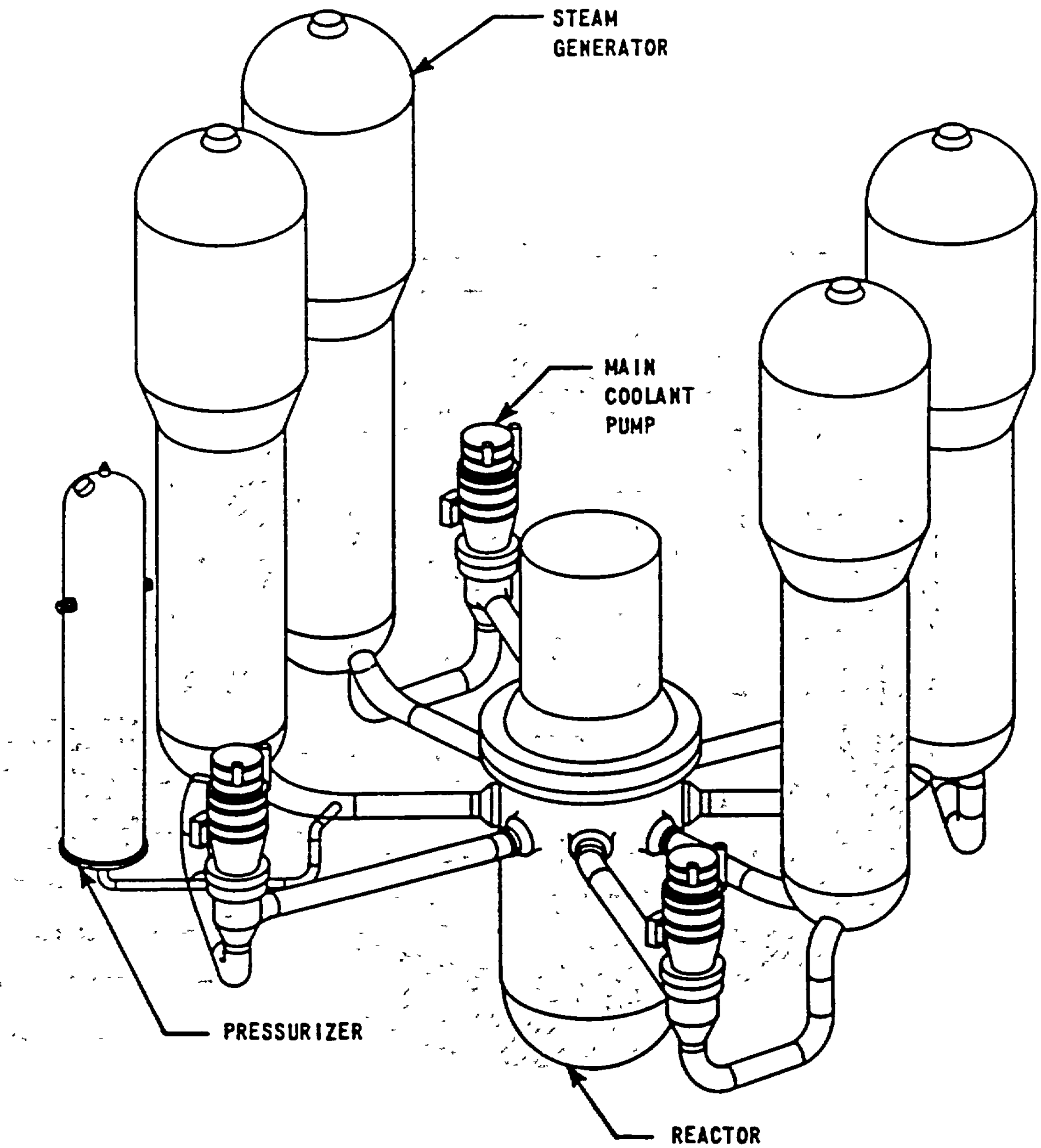


Figure 4.4-1. Westinghouse 4-Loop NSSS

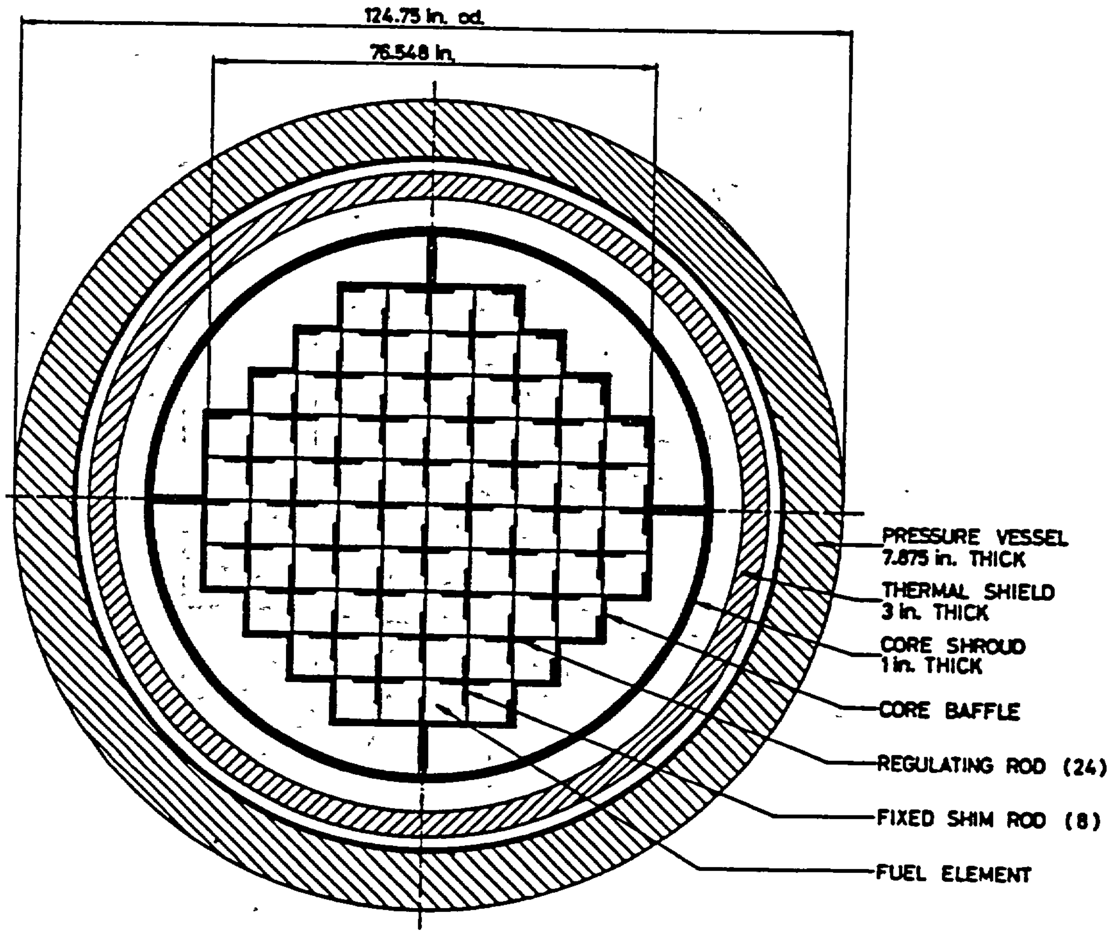
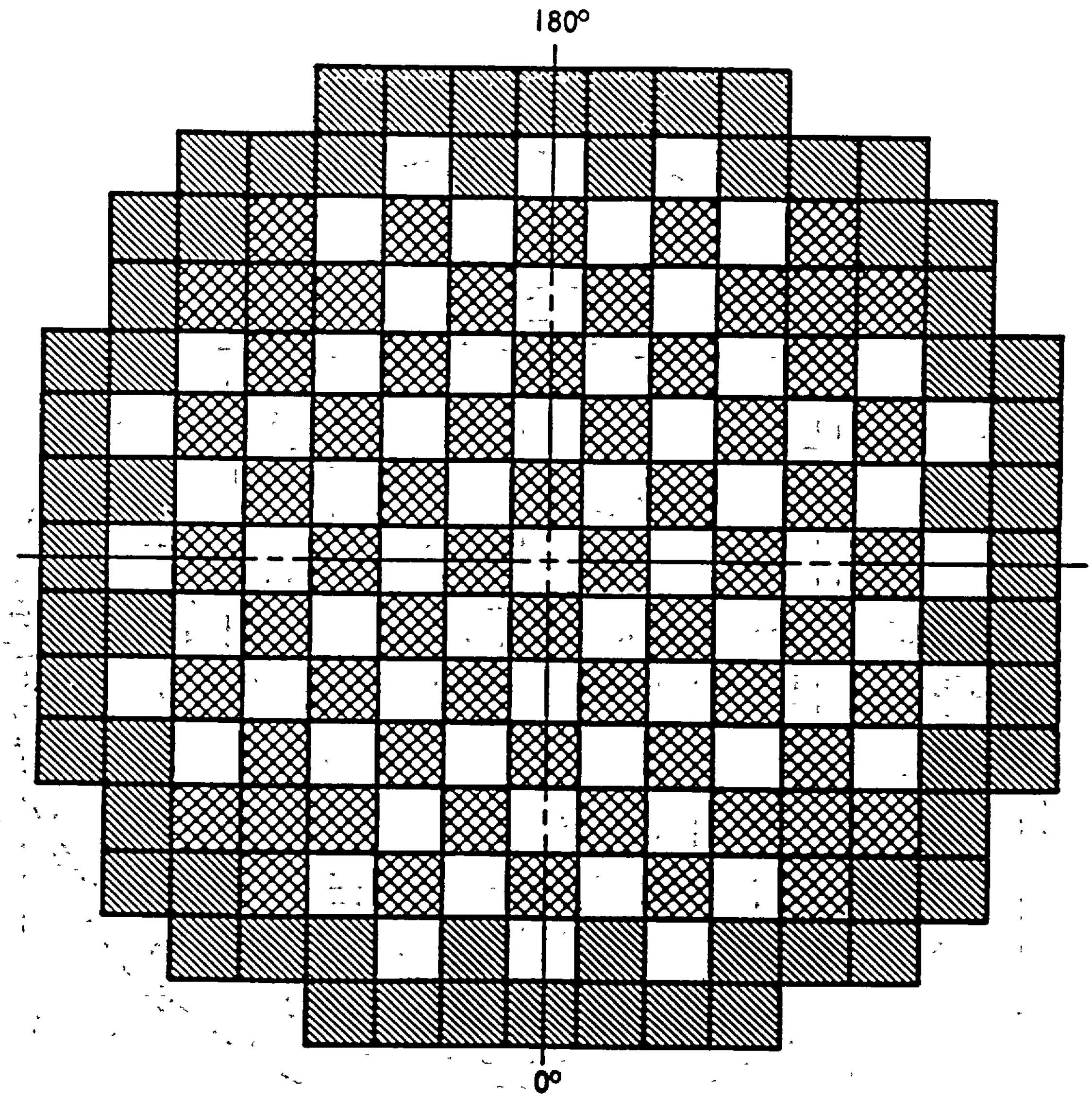


Figure 4.4-2. General Arrangement of the Yankee-Rowe Core



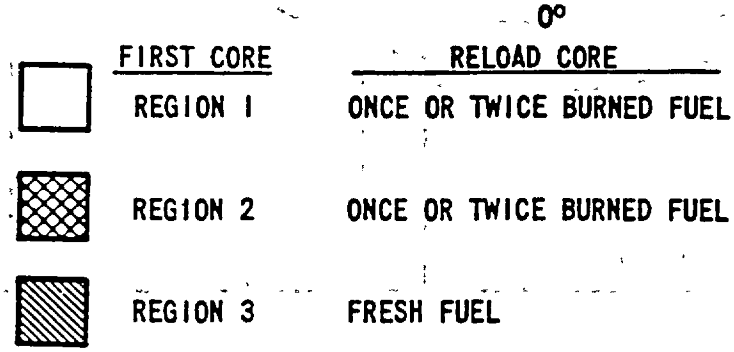


Figure 4.4-3. General Arrangement of a 193 Fuel Assembly Core

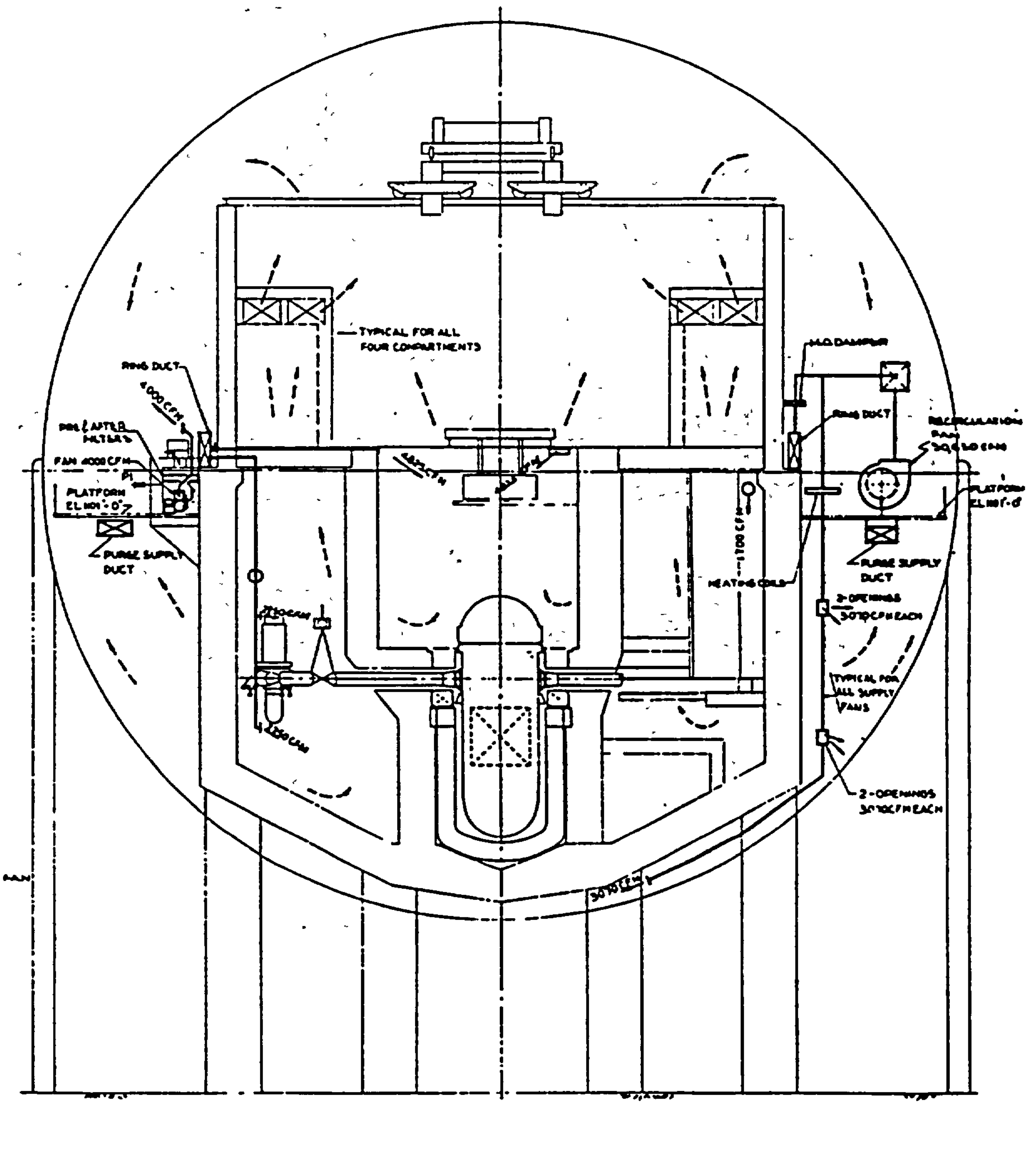


Figure 4.4-4. Section View of the Yankee-Rowe Large, Dry Containment (Steel Sphere)

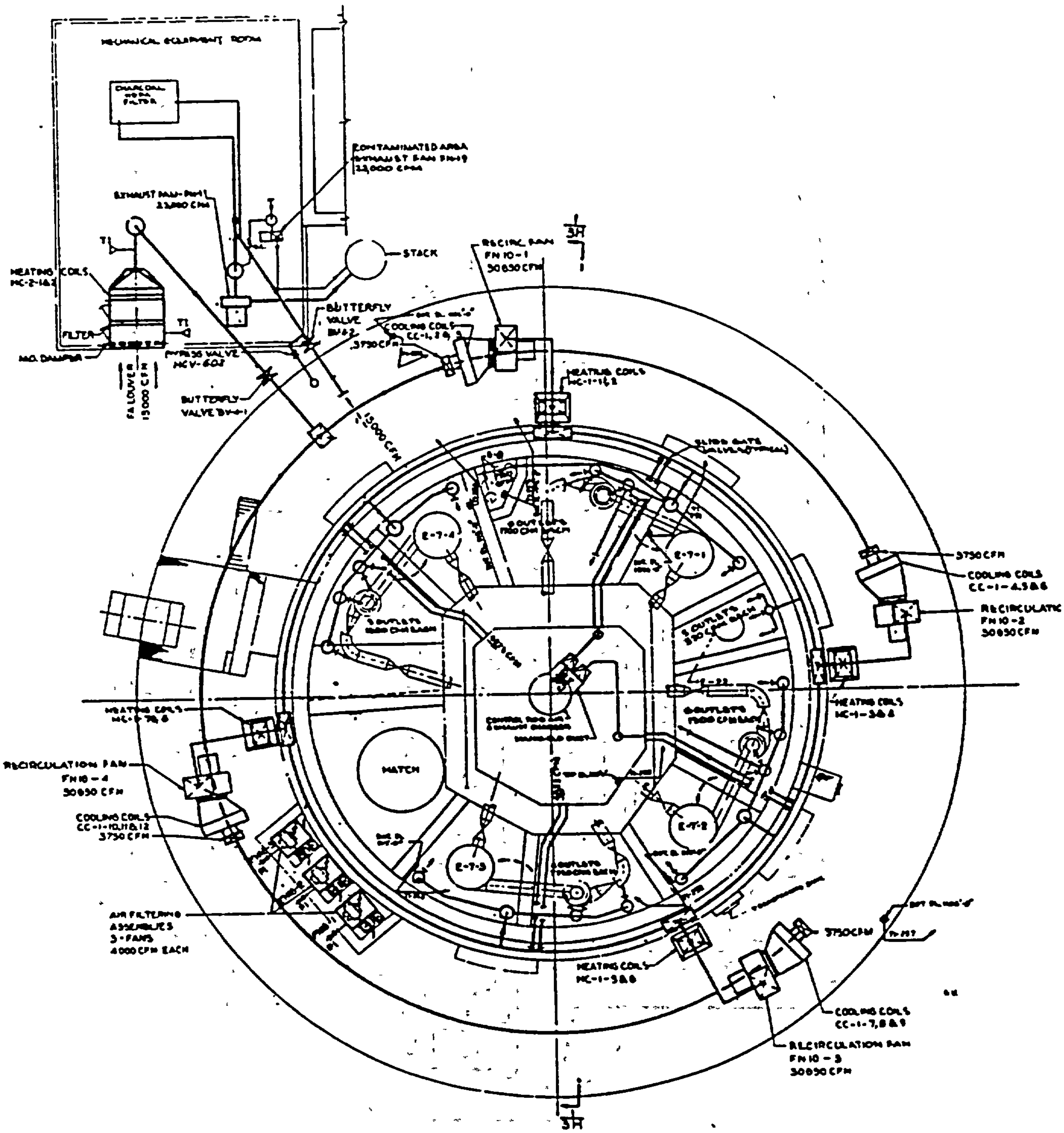


Figure 4.4-5. Plan View of the Yankee-Rowe Large, Dry Containment (Steel Sphere)

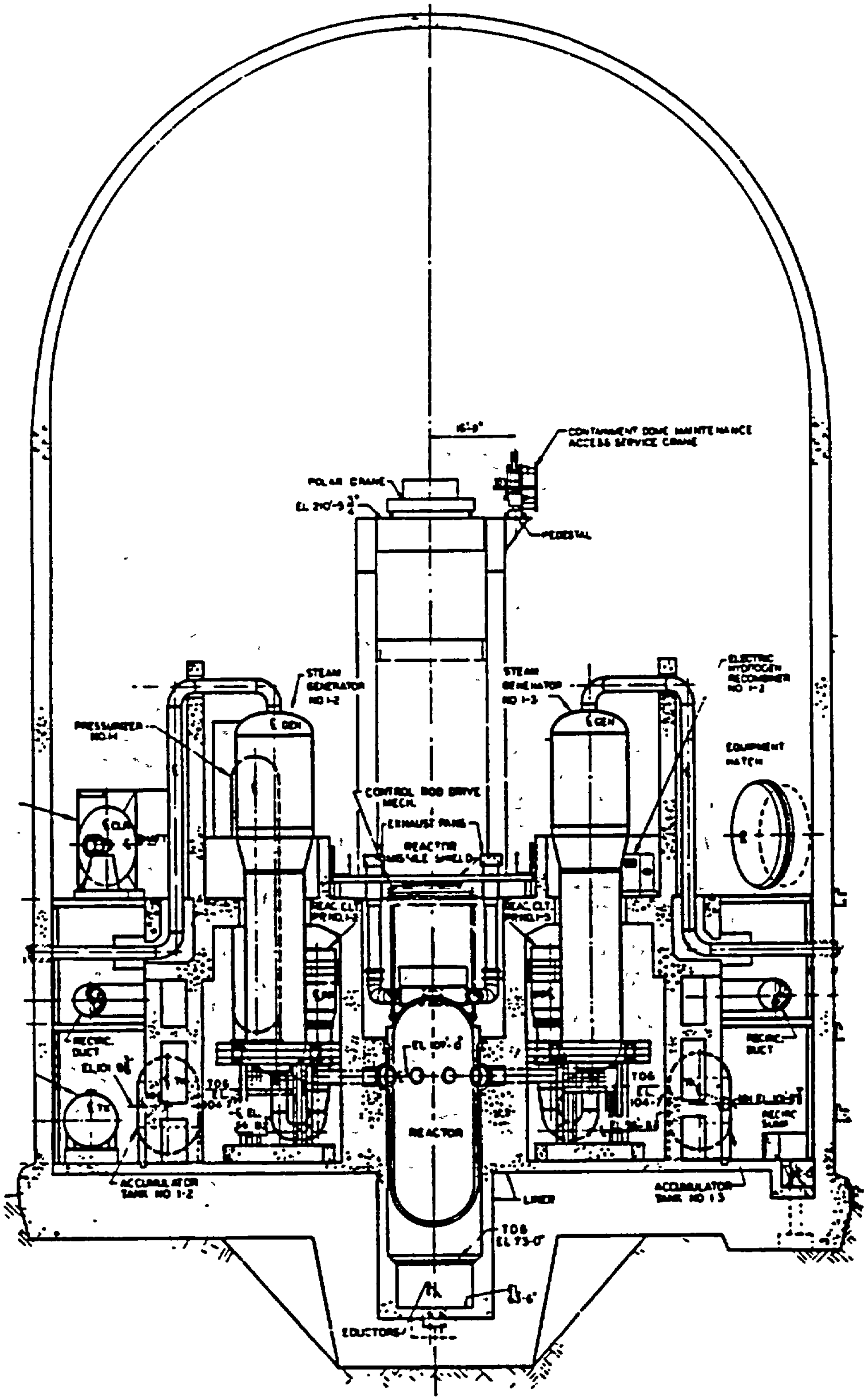


Figure 4.4-6. Section Views of the Diablo Canyon Large, Dry Containment (Reinforced Concrete)

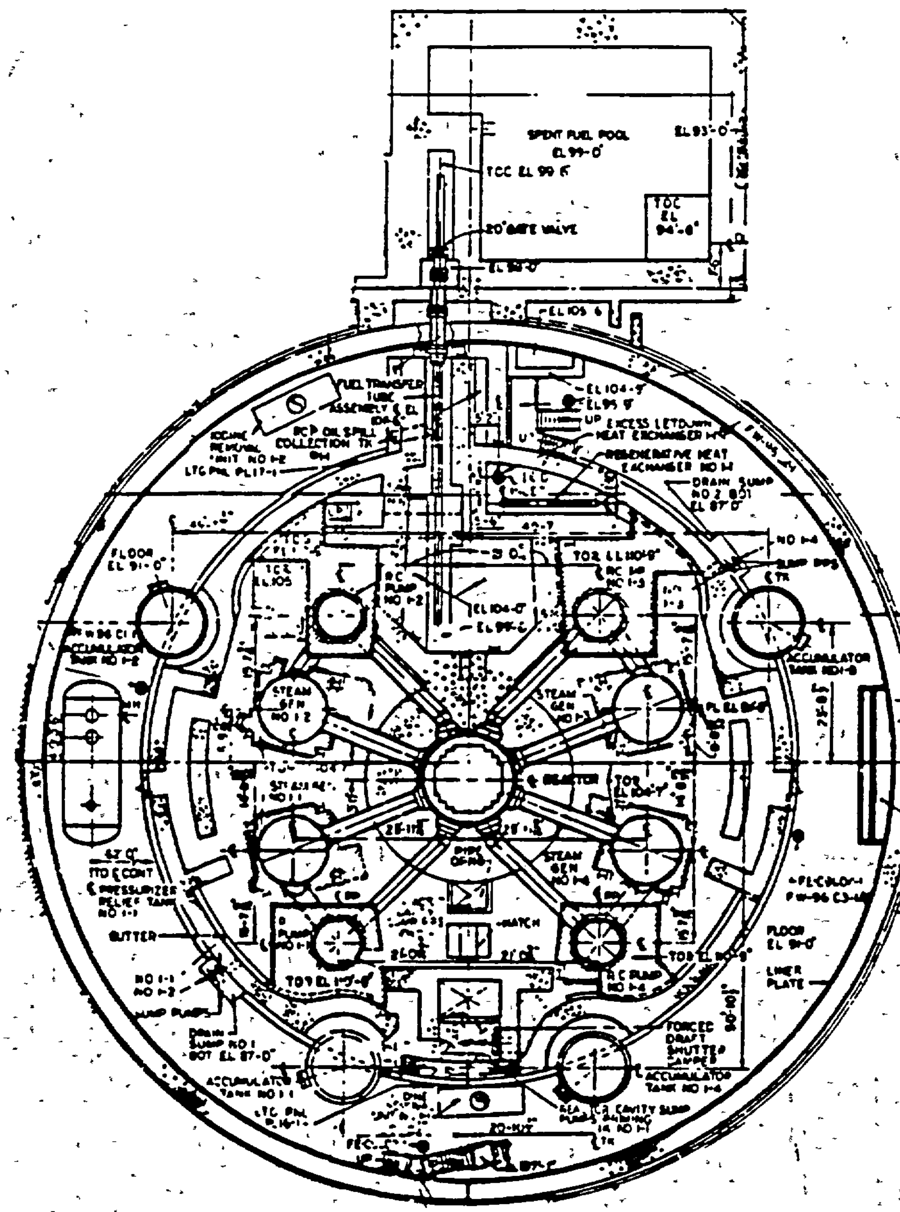


Figure 4.4-7. Plan View of the Diablo Canyon Large, Dry Containment (Reinforced Concrete)

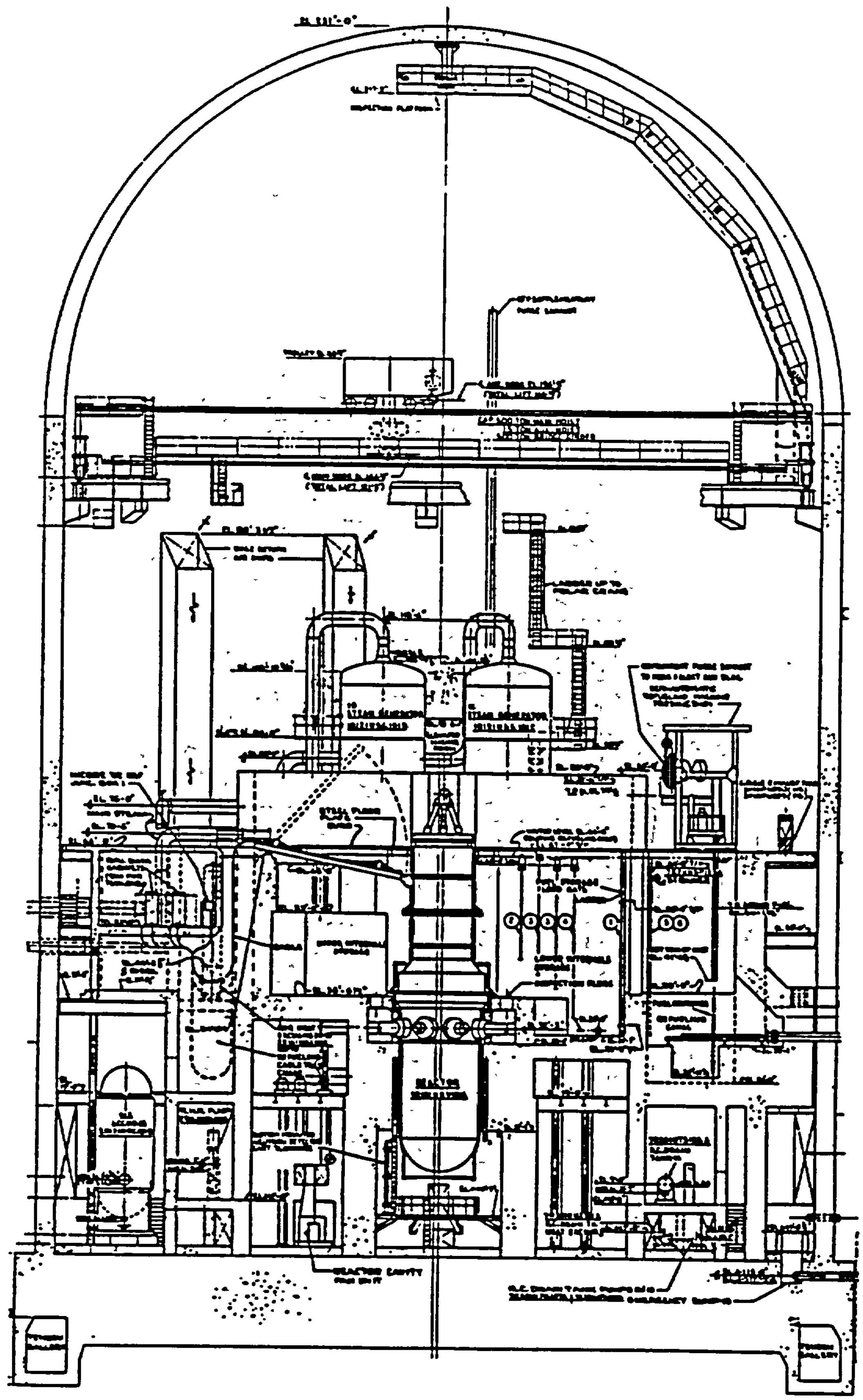


Figure 4.4-8. Section View of South Texas Large, Dry Containment (3-D Post-Tensioned Concrete)

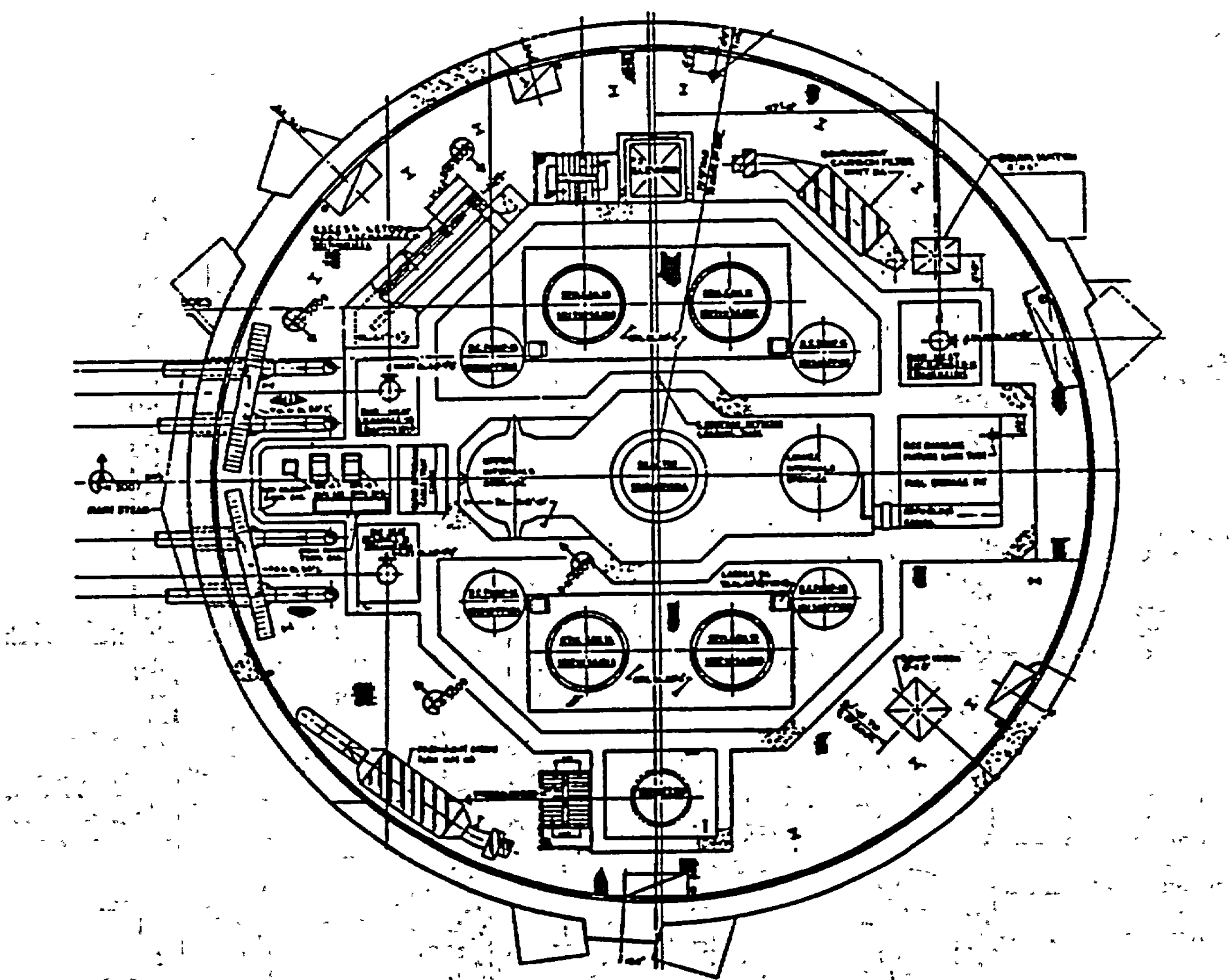


Figure 4.4-9. Plan View of South Texas Large, Dry Containment (3-D Post-Tensioned Concrete)

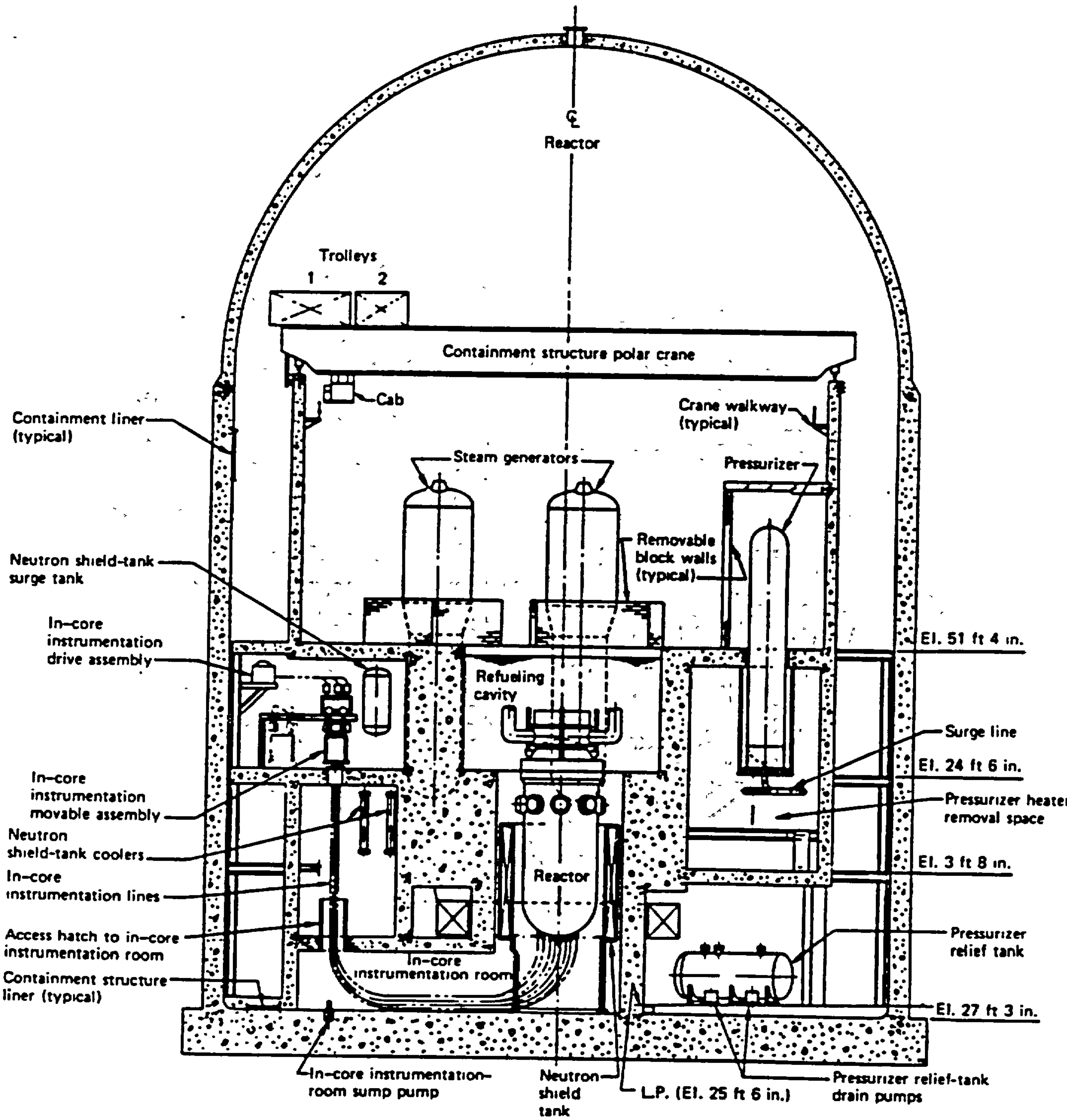


Figure 4.4-10. Section View of the Millstone 3 Subatmospheric Containment

4.5 Westinghouse PWR Comparative Data

This section contains the following tables which present comparative system data for Westinghouse PWRs:

- Table 4.5-1 Design Parameters for Representative Westinghouse PWRs
- Table 4.5-2 Comparison of Westinghouse PWR Vessel and Core Parameters
- Table 4.5-3 Westinghouse PWR System Comparison - RCS, AFW, Charging and HPSI
- Table 4.5-4 Comparison of Westinghouse PWR Pressurizer Relief Capacity
- Table 4.5-5 Comparison of Westinghouse PWR Containments
- Table 4.5-6 Comparison of Westinghouse PWR Backup Electric Power Systems
- Table 4.5-7 Comparison of Westinghouse PWR Power Conversion Systems

Table 4.5-1. Design Parameters for Representative Westinghouse PWRs

REACTOR PLANT CHARACTERISTICS	WESTINGHOUSE 2-LOOP PLANT GINNA	WESTINGHOUSE 3-LOOP PLANT H.B. ROBINSON	WESTINGHOUSE EARLY 4-LOOP PLANT TROJAN	WESTINGHOUSE LATE 4-LOOP PLANT SOUTH TEXAS 1 AND 2
Overall				
Number of loops	2 without isolation	3 without isolation	4 without isolation	4 without isolation
Thermal capacity	1520 MWt	2308 MWt	3411 MWt	3817 MWt
Electric capacity	470 MWe	665 MWe	1095 MWe	1 x 1250 MWe
Efficiency (net)	30.92%	29.81%	32.10%	32.75%
Coolant pressure in primary circuit at exit from reactor	2235 psig	2235 psig	2235 psig	2235 psig
Coolant temperature at inlet	551.9 °F	546 2°F	552.5 °F	560°F
Coolant temperature at exit from reactor	634°F	642°F	620 °F	628 8°F
Coolant flow rate through reactors (total)	66.7E6 lb/hr	101.5E6 lb/hr	132.7E6 lb/hr	1.39E7 lb/hr
Core				
Height of active core	12 ft.	12 ft.	11.98 ft.	14 ft.
Equivalent diameter	8 04 ft.	9.96 ft.	11.06 ft.	11.1 ft.
Number of fuel assemblies	121	157	193	193
Number of control rods assemblies	29	53	53	57
Number of fuel elements in assembly	179	204	264	264
Diameter of fuel element	0.3669 in.	0.3659 in.	0.3225 in.	0.374 in.
Area of heat transfer surface	28,715 sq ft.	42,460 sq.ft.	59,700 sq ft.	69,700 sq.ft.
Mean specific heat flux	150,500 Btu/hr-sq ft.	171,600 Btu/hr-sq ft.	Unk.	181,200 Btu/hr-sq ft.
Number of fuel rods	21,659	32,028	50,913	50,913
Core loading	3 region non-uniform	3 region non-uniform	3 region non-uniform	3 region non-uniform
Average burnup (first cycle)	~14,126 MWd/MTu	13,000 MWd/MTu	Unk.	15,000 MWd/MTu
Fuel weight (as U02)	120,130 lbs.	175,400 lbs.	222,739 lbs.	259,860 lbs

Table 4.5-1. Design Parameters for Representative Westinghouse PWRs (Continued)

REACTOR PLANT CHARACTERISTICS	WESTINGHOUSE 2-LOOP PLANT GINNA	WESTINGHOUSE 3-LOOP PLANT H.B. ROBINSON	WESTINGHOUSE EARLY 4-LOOP PLANT TROJAN	WESTINGHOUSE LATE 4-LOOP PLANT SOUTH TEXAS 1 AND 2
Reactor Vessel				
Vessel height	39.11 ft.	41.5 ft.	43.83 ft.	43.75 ft
Inner diameter	11 ft.	12.96 ft.	13.92 ft.	14.4 ft
Number of openings for inlet and outlet Nozzles	2 x 2	2 x 3	2 x 4	2 x 4
Steam Generator				
Number of units	2	3	4	4
Thermal power per unit	650 MWt	769.3 MWt	Unk.	954 MWt
Shell side operating pressure (steam)	989 psig	1005 psig	1073 psig	1073 psi
Tube side operating pressure	2235 psig	2235 psig	2235 psig	2235 psi
Tube side design flow	33.63 E6 lb/hr	33.93 E6 lb/hr	Unk.	Unk.
Piping				
Hot leg inner dia.	29 in.	29 in.	29 in.	29 in.
Cold leg inner dia.	27.5 in.	27.5 in.	27.5 in.	27.5 in.
Between pump and steam generator	31 in.	31 in.	31 in.	32 in.
Safety Valves				
First opening pressure	2485 psig	2485 psig	2485 psig	2485 psig
Capacity	189,500 lb/hr each of 2	139,300 lb/hr each of 3	123,100 lb/hr each of 2	504,953 lb/hr each of 3
	117.8 lb/hr/MWt each PORV of 2	95.5 lb/hr/MWt each PORV of 2	61.6 lb/hr/MWt each PORV of 2	210,000 lb/hr each PORV of 3
Primary Coolant Pumps				
Number	2	3	4	4
Pump capacity	90,000 gpm	88,500 gpm	88,500 gpm	102,500 gpm
Coolant temperature	555.1 °F	546.5°F	552.5°F	650°F
Pressure rise	252 ft. head	261 ft. head	277 ft. head	Unk.
Design pressure	2485 psig	2485 psig	2485 psig	2485 psig
Design temperature	650 °F	650 °F	650 °F	650 °F
Motor rating (nameplate)	5,500 hp.	6,000 hp.	6,000 hp.	6,000 hp.

Table 4.5-2. Comparison of Westinghouse PWR Vessel and Core Parameters

PWR Vendor	PWR Type	Reactor Plant Name	Core Power (MWt)	Reactor Vessel I.D. (in)	Core Equivalent Diam. (in)	Core Active Height (in)	Core Average Power Density (kW/liter)	Number of Fuel Elements	Fuel Element Geometry	Number of Control Rods (Full/Part Length)
W	2-loop	Ginna	1520	132	96.5	144	89.00	121	14 x 14	29 F/4 P
W	2-loop	Point Beach 1 & 2	1518	132	96.5	144	87.00	121	14 x 14	37 (total)
W	2-loop	Kewaunee	1650	132	96.5	144	94.90	121	14 x 14	33 (total)
W	2-loop	Prairie Island 1 & 2	1650	132	96.5	144	95.90	121	14 x 14	29 F/4 P
W	3-loop	San Onofre 1	1347	144	119.5	144	70.40	157	14 x 14	45 (total)
W	3-loop	H.B. Robinson 2	2200	156	119.5	144	82.60	157	15 x 15	48 F/5 P
W	3-loop	Surry 1 & 2	2441	159	119.5	144	92.00	157	15 x 15	48 F/5 P
W	3-loop	Turkey Point 3 & 4	2208	172	119.5	144	82.80	157	15 x 15	48 F/5 P
W	3-loop	Beaver Valley 1	2660	157	119.5	143.7	100.00	157	17 x 17	48 F/5 P
W	3-loop	Beaver Valley 2	2660	157	119.5	143.7	100.00	157	17 x 17	48 F/5 P
W	3-loop	Farley 1 & 2	2652	157	119.5	144	101.10	157	17 x 17	45 (total)
W	3-loop	North Anna 1 & 2	2775	157	119.5	144	108.70	157	17 x 17	48 (total)
W	3-loop	Shearon Harris 1	2785	157	119.5	144	105.00	157	17 x 17	52 (total)
W	3-loop	Summer	2785	172	119.5	144	104.50	157	17 x 17	48 (total)
W	4-loop	Yankee Rowe	600	109	75.7	91.9	89.3 to 90.1	76	9 x (6 x 6)	24 (total)
W	4-loop	Haddam Neck	1825	154	119.6	121.8	82.00	157	15 x 15	45 (total)
W	4-loop	Braidwood 1 & 2	3411	173	132.7	143.7	104.50	193	17 x 17	53 (total)
W	4-loop	Byron 1 & 2	3411	173	132.7	143.7	104.50	193	17 x 17	53 (total)
W	4-loop	Callaway	3425	173	132.7	143.7	109.20	193	17 x17	53 (total)
W	4-loop	Catawba 1 & 2	3411	173	132.7	143.7	103.5 to 44.6	193	17 x 17	53 (total)

Table 4.5-2. Comparison of Westinghouse PWR Vessel and Core Parameters (Continued)

PWR Vendor	PWR Type	Reactor Plant Name	Core Power (MWt)	Reactor Vessel I.D. (in)	Core Equivalent Diam. (in)	Core Active Height (in)	Core Average Power Density (kW/liter)	Number of Fuel Elements	Fuel Element Geometry	Number of Control Rods (Full/Part Length)
W	4-loop	Comanche Peak 1 & 2	3425	167	132.7	143.7	103.3 to 104.5	193	17 x 17	53 (total)
W	4-loop	D. C. Cook 1 & 2	3250	173	132.7	143.7	98.0 to 103.8	193	17 x 17	53 (total)
W	4-loop	Diablo Canyon 1 & 2	3338	173	132.7	143.7	102.3 to 104.5	193	17 x 17	53 F/8 P
W	4-loop	Indian Point 2	2758	173	132.7	144	85.00	193	15 x 15	53 F/8 P
W	4-loop	Indian Point 3	3025	173	132.7	144	92.70	193	15 x 15	53 F/8 P
W	4-loop	McGuire 1 & 2	3425	173	132.7	143.7	103.50	193	17 x 17	53 (total)
W	4-loop	Millstone 3	3411	173	132.7	143.7	104.50	193	17 x17	53 F/8 P
W	4-loop	Salem 1 & 2	3338	173	132.7	143.7	102.60	193	17 x 17	53 (total)
W	4-loop	Seabrook 1	3411	173	132.7	143.7	104.50	193	17 x 17	53 (total)
W	4-loop	Sequoyah 1 & 2	3411	173	132.7	143.7	103.50	193	17 x 17	53 F/8 P
W	4-loop	South Texas 1 & 2	3817	173	132.7	168	105 (est.)	193	17 x 17	57 (total)
W	4-loop	Trojan	3411	173	132.7	143.7	105.50	193	17 x 17	53 F/8 P
W	4-loop	Vogtle 1 & 2	3411	173	132.7	143.7	104.50	193	17 x 17	57 (total)
W	4-loop	Watts Bar 1 & 2	3411	173	132.7	143.7	103.50	193	17 x 17	53 F/8 P
W	4-loop	Wolf Creek	3411	173	132.7	143.7	101.90	193	17 x 17	57 (total)
W	4-loop	Zion 1 & 2	3250	173	132.7	144	100.00	193	15 x 15	57 (total)

Table 4.5-3. Westinghouse PWR System Comparison - RCS, AFW, Charging and HPSI

Plant Name	NSSS Vendor	Reactor Coolant System: Core MWt	# RCS Loops	# RCS PORV/SV	S/G Model	Auxiliary Feedwater System: # AFW Pumps	Type Drive	Capacity gpm @ psig	Charging System: # Pumps	Type Pump	Capacity gpm @ psig	Capacity gpm @ PORV	High Pressure Injection System: # Pumps	Type Pump	Capacity gpm @ psig	Capacity gpm @ PORV	Notes
Beaver Valley 1 & 2	W	2660	3	3/3	51	2 1	M T	350 @ 1169 700 @ 1169	3	Cent	150 @ 2514	150	Same as cent. charging pumps				(a)
Braidwood 1 & 2	W	3411	4	2/3	51	1 1	M D	890 @ 1452 840 @ 1452	2 1	Cent PD	150 @ 2526 98	150 98	2	Cent	400 @ 1106	0	(a)
Byron 1 & 2	W	3411	4	2/3	51	1 1	M D	890 @ 1452 840 @ 1452	2 1	Cent PD	150 @ 2526 98	150 98	2	Cent	400 @ 1106	0	(a)
Callaway	W	3425	4	2/3	F	2 1	M T	600 @ 1387 1200 @ 1387	2 1	Cent PD	150 98	150 98	2	Cent	425 @ 1162	0	(a)
Catawba 1 & 2	W	3411	4	3/3	51	2 1	M T	500 @ 1392 100 @ 1395	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	400 @ 1750	0	(a)
Comanche Peak 1 & 2	W	3425	4	2/3	F	2 1	M T	470 @ 1107 900 @ 1107	2 1	Cent PD	unk.	unk.	2	Cent	unk	0	(a)
D C Cook 1	W	3250	4	3/3	51	1 1	M T	450 @ 1177 900 @ 1177	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	400 @ 1700	0	(a) (b)
D.C. Cook 2	W	3250	4	2/3	51F	1 1	M T	450 @ 1177 900 @ 1177	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	400 @ 1700	0	(a) (b)
Diablo Canyon 1 & 2	W	3338	4	3/3	UT	2 1	M T	440 @ 1300 880 @ unk	2 1	Cent PD	150 @ 2514 98 @ 2514	150 98	2	Cent	425 @ 1084	0	
Farley 1 & 2	W	2652	3	2/3	51	2 1	M M	350 @ 1214 700 @ 1214	3	Cent	150 @ 2800	150	Same as cent charging pumps				(a)
Ginna	W	1520	2	2/2	44	2 1 2	M T M	200 @ 1344 400 @ 1344 200 @ 1080	3	PD	60	60	3	Cent	300 @ 1170	0	(a) (c)
Haddam Neck	W	1825	4	2/3	27	2	T	450 @ 1000	2 1	Cent PD	360 @ 2300 30	360 30	2	Cent	970 @ 1750	0	(a)
Indian Point 2	W	2758	4	2/3	44F	2 1	M T	400 @ 1350 800 @ 1350	3	PD	98	98	3	Cent	400 @ 1180	0	
Indian Point 3	W	3025	4	2/3	44F	2 1	M T	400 @ 1350 800 @ 1350	3	PD	98	98	3	Cent	400 @ 1080	0	(a)
Kewaunee	W	1650	2	2/2	51	2 1	M T	240 @ 1235 240 @ 1235	3	PD	60	60	2	Cent	700 @ 1082	0	(a)
McGuire 1 & 2	W	3425	4	3/3	51	2 1	M T	450 @ 1655 900 @ 1730	2 1	Cent PD	150 @ 2514 55	150 55	2	Cent	400 @ 1106	0	(a)
Millstone 3	W	3411	4	2/3	51	2 1	M T	575 @ 1290 1150 @ 1290	3	Cent	150 @ 2800	150	2	Cent	425 @ 1500	0	(a)
North Anna 1 & 2	W	2775	3	2/3	51F	2 1	M T	350 @ 1214 700 @ 1214	3	Cent	150 @ 2500	150	Same as cent. charging pumps				(a)
Point Beach 1 & 2	W	1518	2	2/2	44F	1 1	M T	200 @ 1192 400 @ 1192	3	PD	60.5	60.5	2	Cent	700 @ 1750	0	(a) (b)
Prairie Island 1 & 2	W	1650	2	2/2	51	1 1	M T	200 @ 1200 200 @ 1200	3	PD	60.5	60 5	2	Cent	700 @ 1082	0	(a) (d)
Robinson	W	2200	3	2/3	44F	2 1	M T	300 @ 1300 600 @ 1300	3	PD	77	77	3	Cent	375 @ 1750	0	(a)
Salem 1 & 2	W	3338	4	2/3	51	2 1	M T	440 @ 1300 880 @ 1650	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	unk.	0	(a)
San Onofre 1	W	1347	3	2/2	27	1 1	M T	235 @ 1035 300 @ 1110	2	Cent	0 @ 2400	0	Same as cent. charging pumps				(a)
Seabrook	W	3411	4	2/3	F	1 1	M T	710 @ 1322 710 @ 1322	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	425 @ 1750	0	(a)

Table 4.5-3. Westinghouse PWR System Comparison - RCS, AFW, Charging and HPSI (Continued)

		Reactor Coolant System				Auxiliary Feedwater System			Charging System				High Pressure Injection System				
Plant Name	NSSS Vendor	Core MWt	# RCS Loops	# RCS PORV/SV	S/G Model	# AFW Pumps	Type Drive	Capacity gpm @ psig	# Pumps	Type Pump	Capacity gpm @ psig	Capacity gpm @ PORV	# Pumps	Type Pump	Capacity gpm @ psig	Capacity gpm @ PORV	Notes
Sequoyah 1 & 2	W	3411	4	2/3	51	2 1	M T	440 @ 1257 880 @ 1257	2 1	Cent PD	150 @ 2514 55	150 55	2	Cent	425 @ 1084	0	
Shearon Harris 1	W	2785	3	3/3	UT	2 1	M T	400 @ 1265 900 @ 1265	3	Cent	150 @ 2514	150	Same as cent. charging pumps				
South Texas 1 & 2	W	3817	4	2/3	F	3 1	M T	540 @ 1435 540 @ 1435	2 1	Cent PD	160 @ 2513 35	160 35	3	Cent	800 @ 1235	0	
Summer	W	2785	3	unk.	UT	2 1	M T	400 @ 1211 1010 @ 1211	3	Cent	150 @ unk	150	Same as cent. charging pumps				
Surry 1 & 2	W	2441	3	2/3	51F	2 1	M T	350 @ 1183 700 @ 1183	3	Cent	150 @ 2485	150	Same as cent. charging pumps				(a)
Trojan	W	3411	4	2/3	51	1 1	T D	960 @ 1474 960 @ 1474	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	425 @ 1700	0	(a)
Turkey Point 3 & 4	W	2208	3	2/3	44F	2	T	600 @ 1203	3	PD	77	77	2	Cent	300 @ 1750	0	(a)
Vogtle 1 & 2	W	3411	4	2/2	51	2 1	M T	630 @ 1517 1117 @ 1517	2 1	Cent PD	150 @ 2514 98 @ unk.	150 98	2	Cent	425 @ 1162	0	(a)
Watts Bar 1 & 2	W	3411	4	2/3	51	2 1	M T	470 @ 1600 940 @ 1600	2 1	Cent PD	150 @ 2514 98 @ 3200	150 98	2	Cent	unk.	0	(a)
Wolf Creek	W	3411	4	2/3	F	2 1	M T	600 @ 1387 1200 @ 1387	2 1	Cent PD	150 @ 2514 98 @ 2514	150 98	2	Cent	425 @ 1161	0	(a)
Yankee Rowe	W	600	4	1/2	27	1	T	80 @ 1200	3	Cent	33	33	3	Cent	187 @ 650	0	(a) (e)
Zion 1 & 2	W	3250	4	2/3	51	2 1	M T	450 @ 1343 900 @ 1343	2 1	Cent PD	150 @ 2800 98	150 98	2	Cent	400 @ 1084	0	(a)

Table 4.5-3. Westinghouse PWR System Comparison - RCS, AFW, Charging and HPSI (Continued)

General Note:

All pump capacities are stated on a per-pump basis. AFW pump capacity is stated in terms of rated capacity. Charging and high pressure injection pump capacity is stated in terms of rated capacity and approximate capacity when RCS pressure is at the PORV setpoint (i.e. for "feed and bleed" operation).

Codes used in this table include:

Type drive:	M = electric motor T = steam turbine
Type pump:	PD = positive displacement Cent = centrifugal
RCS PORV/SV:	number of RCS power-operated valves (first number) and safety valves (second number).
S/G model:	UT = U-tube (see note (a)) OT = once-through

Notes:

(a) The Westinghouse small inventory steam generators (series 27 and 44) require twice the feedwater flow to prevent dryout as compared to the later versions (series 51 and F). The exception is Yankee-Rowe (ref. NUREG/CR-3713, Section 3).
(b) At Point Beach, and D.C. Cook, the motor-driven AFW pump in each unit can feed steam generators in both units.
(c) Ginna has a main AFW system with two motor-driven and one turbine-driven pump as well as a standby AFW system which has tow motor-driven pumps located in a separate area.
(d) At Prairie Island, the motor driven AFW pump at each unit normally supplies the opposite unit.
(e) At Yankee Rowe, charging and SIS provide backup for AFW.

Table 4.5-4. Comparison of Westinghouse PWR Pressurizer Relief Capacity

Plant Name	NSSS Vendor	#RCS PORV's	Manufacturer	Capacity (lb/hr/MWt)	Lowest Setpoint (psig)	# RCS SV's	Capacity (Klb/hr)	Lowest Setpoint (psig)
Beaver Valley 1 & 2	W	3	Masoneilan 38-20771	79.9	2335	3	345	2485
Braidwood 1 & 2	W	2	Unk.	210*	2335	3	420	2485
Byron 1 & 2	W	2	Unk.	210*	2335	3	420	2485
Callaway	W	2	Unk.	210*	2335	3	420	2485
Catawba 1 & 2	W	3	Unk.	210*	2485	3	420	Unk.
Comanche Peak 1 & 2	W	2	Unk.	210*	2185	3	420	2485
D.C. Cook 1	W	3	Masoneilan	64.6	2335	3	129.2	2485
D.C. Cook 2	W	2	38-20721	61.8	2335	3	123.5	2485
Diablo Canyon 1 & 2	W	3	Unk.	Unk.	Unk.	3	Unk.	Unk.
Farley 1 & 2	W	2	Copes-Vulcan D-100-160	79.2	2335	3	130.1	2485
Ginna	W	2	Copes-Vulcan D-100-160	117.8	2335	2	189.5	2485
Haddam Neck	W	2	Copes-Vulcan D-100-160	115.1	2270	3	160.7	2485
Indian Point 2	W	2	Copes-Vulcan D-100-160	78.7	2335	3	147.9	2485
Indian Point 3	W	2	Copes-Vulcan D-100-160	78.7	2335	3	147.9	2485
Kewaunee	W	2	Copes-Vulcan D-100-160	106	2335	2	209.1	2485

Table 4.5-4. Comparison of Westinghouse PWR Pressurizer Relief Capacity (Continued)

Plant Name	NSSS Vendor	#RCS PORV's	Manufacturer	Capacity (lb/hr/MWt)	Lowest Setpoint (psig)	# RCS SV's	Capacity (Klb/hr)	Lowest Setpoint (psig)
McGuire 1 & 2	W	3	Unk.	210*	2335	3	420	2485
Millstone 3	W	2	Unk.	210*	2335	3	420	2485
North Anna 1 & 2	W	2	Masoneilan 38-20721	76	2335	3	137	2485
Point Beach 1 & 2	W	2	Copes-Vulcan D-100-160	117.9	2335	2	189.7	2485
Prairie Island 1 & 2	W	2	Copes-Vulcan D-100-160	5	2335	2	209.1	2485
Robinson	W	2	Copes-Vulcan D-100-160	95.5	2335	3	130.9	2485
Salem 1 & 2	W	2	Copes-Vulcan D-100-160	63	2350	3	125.8	2485
San Onofre 1	W	2	ACF Industries 70-18-9 DRTX	80	2190	2	178.2	2500
Seabrook	W	2	Unk.	Unk.	Unk.	3	Unk.	Unk.
Sequoyah 1 & 2	W	N/A	Unk.	Unk.	Unk.	Unk.	Unk.	Unk.
Shearon Harris 1	W	3	Unk.	210*	2335	3	380	2485
South Texas 1 & 2	W	2	Unk.	210*	2485	3	505	2485
Summer	W	Unk.	Unk.	Unk.	Unk.	Unk.	Unk.	Unk.
Surry 1 & 2	W	2	Copes-Vulcan IA58RGP	86	2335	3	120.5	2360
Trojan	W	2	Copes-Vulcan D-100-160	61.6	2350	3	123.1	2485

Table 4.5-4. Comparison of Westinghouse PWR Pressurizer Relief Capacity (Continued)

Plant Name	NSSS Vendor	#RCS PORV's	Manufacturer	Capacity (lb/hr/MWt)	Lowest Setpoint (psig)	# RCS SV's	Capacity (Klb/hr)	Lowest Setpoint (psig)
Turkey Point 3 & 4	W	2	Copes-Vulacn 5-131642	95.1	2335	3	132.8	2485
Vogtle 1 & 2	W	2	Unk.	210*	2235	2	420	2485
Watts Bar 1 & 2	W	2	Unk.	Unk.	Unk.	3	Unk.	Unk.
Wolf Creek	W	2	N/A	210*	2235	3	420	2485
Yankee Rowe	W	1	Dresser 31533 VX	118	2400	2	153	2485
Zion 1 & 2	W	2	Copes-Vulcan D-100-160	64.6	2335	3	129.2	2485

* Klb/hr rating

Table 4.5-5. Comparison of Westinghouse PWR Containments

Plant Name	NSSS Vendor	Arch./ Engineer	Prim. Cont. Type	Construction	Concrete Construction Subtype	Internal Diameter (feet)	Containment Free Volume (ft3)	Design Pressure (psig)	Design Leak Rate % vol/day	Enclosure Building?
Beaver Valley 1 & 2	W	Stone & Webster	Sub Atm	Concrete Cylinder w/ Steel Liner	Reinforced	126	1.80E+06	54	0.1	No
Braidwood 1 & 2	W	Sargent & Lundy	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	140	2.90E+06	61	0.1	No
Byron 1 & 2	W	Sargent & Lundy	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	140	2.90E+06	61	0.1	No
Callaway	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	140	2.50E+06	60	0.1	No
Catawba 1 & 2	W	Duke Power	Ice Cond.	Steel Cylinder	Reinforced	115	1.22E+06	30	0.2	Yes
Comanche Peak 1 & 2	W	Gibbs & Hill	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	135	2.98E+06	50	0.1	No
D.C. Cook 1 & 2	W	AEP	Ice Cond.	Concrete Cylinder w/ Steel Liner	Reinforced	115	unk.	12	0.25	No
Diablo Canyon 1 & 2	W	Pac. Gas & Elect	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	140	2.63E+06	47	0.1	No
Farley 1 & 2	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.03E+06	54	0.3	No
Ginna	W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	1-D Vert. Prestressed	105	9.97E+05	60	0.1	No
Haddam Neck	W	Stone & Webster	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	136	1.71E+06	40	0.1	No
Indian Point 2	W	UEC	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	135	2.61E+06	47	0.1	No
Indian Point 3	W	UEC	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	135	2.61E+06	47	0.1	No
Kewaunee	W	Pioneer	Dry	Steel Cylinder	- - -	108	unk.	46	0.5	Yes
McGuire 1 & 2	W	Duke Power	Ice Cond.	Concrete Cylinder w/ Steel Liner	Reinforced	115	unk.	28	0.2	No
Millstone 3	W	Stone & Webster	Sub Atm	Concrete Cylinder w/ Steel Liner	Reinforced	140	1.03E+07	45	0.9	Yes
North Anna 1 & 2	W	Stone & Webster	Sub Atm.	Concrete Cylinder w/ Steel Liner	Reinforced	126	unk.	45	0.1	No
Point Beach 1 & 2	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	105	unk.	60	0.4	No

Table 4.5-5. Comparison of Westinghouse PWR Containments (Continued)

Plant Name	NSSS Vendor	Arch./ Engineer	Prim. Cont. Type	Construction	Concrete Construction Subtype	Internal Diameter (feet)	Containment Free Volume (ft3)	Design Pressure (psig)	Design Leak Rate % vol/day	Enclosure Building?
Prairie Island 1 & 2	W	Pioneer.	Dry	Steel Cylinder	---	105	unk.	41	0.5	Yes
Robinson 2	W	Ebasco	Dry	Concrete Cylinder w/ Steel Liner	1-D Vert. Prestressed	130	2.10E+06	42	0.1	No
Salem 1 & 2	W	PSE&G	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	140	2.62E+06	47	0.1	No
San Onofre 1	W	Betchel	Dry	Steel Sphere	---	140	1.44E+06	47	0.5	Yes
Seabrook 1	W	UEC	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	140	2.70E+06	65	0.5	Yes
Sequoyah 1 & 2	W	TVA	Ice Cond.	Steel Cylinder	---	106	unk.	10.8	0.5	Yes
Shearon Harris 1	W	Ebasco	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	130	2.50E+06	45	0.3	No
South Texas 1 & 2	W	Brown	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	150	3.30E+06	56	0.3	No
Summer	W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	126	unk.	55	0.2	No
Surry 1 & 2	W	Stone & Webster	Sub Atm.	Concrete Cylinder w/ Steel Liner	Reinforced	126	1.80E+06	60	0.1	No
Trojan	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	124	2.00E+06	60	0.2	No
Turkey Point 3 & 4	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.55E+06	59	0.25	No
Vogtle 1 & 2	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	140	2.70E+06	52	0.1	No
Watts Bar 1 & 2	W	TVA	Ice Cond.	Steel Cylinder	---	115	unk.	15	0.5	Yes
Wolf Creek	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	135	2.50E+06	60	0.1	No
Yankee Rowe	W	Stone & Webster	Dry	Bare Steel Sphere	---	125	1.02E+06	34	3	No
Zion 1 & 2	W	Sargent & Lundy	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	141	2.86E+06	47	0.1	No

Table 4.5-6. Comparison of Westinghouse PWR Backup Electric Power Systems

Reactor Plant	NSSS Vendor	Shared Diesels per Plant	Dedicated Diesels per Unit	Continuous Rating (kW)	Diesel Manufacturer	# of Batteries per Plant	Voltage	Notes
Beaver Valley 1 & 2	W	None	2	2600	Gen. Motors	5 * 4 **	125	*Unit 1 **Unit 2 (2 are diesel batts.)
Braidwood 1 & 2	W	None	2	5500	unk.	2	125	
Byron 1 & 2	W	None	2	5500	unk.	2	125	
Callaway	W	None	2	6201	unk.	2	125	
Catawba 1 & 2	W	1 **	2	7000	unk.	7 ** 2 *	125	*Diesel batt. **1 DG and 3 125V DC batts for safe shtdwn.
Comanche Peak 1 & 2	W	None	2	7000	unk.	2 *	125	
D.C. Cook 1 & 2	W	None	2	3500* 3600**	Worthington Worthington	4	250	*Unit 1 **Unit 2
Diablo Canyon 1 & 2	W	1	2	2600	Alco	6	125	
Farley 1 & 2	W	3	2	2600 4600	Fairbanks-Morse Fairbanks-Morse	7	125	
Ginna	W	None	2 *	1950	Alco	2	120	*480 VAC diesel generator
Haddam Neck	W	None	2	2850	unk.	2	125	
Indian Point 2	W	None	3	1750	Alco	4	125	
Indian Point 3	W	None	3	1750	Alco	3	125	
Kewaunee	W	None	2	2850	Gen. Motors	2	125	
McGuire 1 & 2	W	None	2	4000	unk.	4	125	
Millstone 3	W	None	2	4986	Fairbanks-Morse	4	125	
North Anna 1 & 2	W	None	2	2750	Fairbanks-Morse	8	125	
Point Beach 1 & 2	W	2		2850	Gen. Motors	2	125	

Table 4.5-6. Comparison of Westinghouse PWR Backup Electric Power Systems (Continued)

Reactor Plant	NSSS Vendor	Shared Diesels per Plant	Dedicated Diesels per Unit	Continuous Rating (kW)	Diesel Manufacturer	# of Batteries per Plant	Voltage	Notes
Prairie Island 1 & 2	W	2		2850	Gen. Motors (model 999-20)	4	125	
Robinson 2	W	None	2 1 *	2450 2500	Fairbanks-Morse Fairbanks-Morse	2	125	* Dedicated shutdown diesel generator
Salem 1 & 2	W	 1 *	3	2600 40000*	Alco	6 2	125 250	* Gas turbine
San Onofre 1	W	None	2	unk.	unk.	2	125	
Seabrook 1	W	None	2 *	6083	Fairbanks-Morse	4	125	
Sequoyah 1 & 2	W	None	2	3600	Bruce GM	unk.	unk.	
Shearon Harris 1	W	None	2	6500	unk.	2	125	
South Texas 1 & 2	W	None	3	5935	Cooper Energy Services	unk.	unk.	
Summer	W	None	2	4250	Fairbanks-Morse	unk.	unk.	
Surry 1 & 2	W	1	1	2850	Gen. Motors	4	125	
Trojan	W	None	2	4418	Gen. Motors	2 1	125 250	
Turkey Point 3 & 4	W	2		2500	Schoonmaker GM	4	125	
Vogtle 1 & 2	W	None	2	7000	unk.	4	125	
Watts Bar 1 & 2	W	None	2	4750	Fairbanks-Morse	unk.	unk.	
Wolf Creek	W	None	2	6201	unk.	4	125	
Yankee Rowe	W	None	3 *	400	Gen. Motors	3	125	*480 VAC diesel generator
Zion 1 & 2	W	1	2	4000	Cooper-Bessemer	5	125	

Table 4.5-7. Comparison of Westinghouse PWR Power Conversion Systems

Plant Name	NSSS Vendor	Architect/ Engineer	Constructor	Turbine Gen. Cap. (MWe)	Turbine Bypass Capability (%)	Condenser Cooling Type	Normal Heat Sink	# Main FW Pumps	FW Pump Drive Type	Shutoff Head (psig)	Capacity (gpm)
Beaver Valley 1 & 2	W	Stone & Webster	Duquesne Light Co	833	85	Closed Loop	Nat. Cooling Tower	2	AC	unk.	unk
Braidwood 1 & 2	W	Sargent & Lundy	Commonwealth Edison	1120	40	Once Through	Braidwood Lake	2 (50%) 1 (50%)	turbine AC	unk. unk	unk. unk
Byron 1 & 2	W	Sargent & Lundy	Commonwealth Edison	1120	40	Closed Loop	Nat. Cooling Towers	2 (50%) 1 (50%)	turbine AC	unk unk	unk. unk
Callaway	W	Bechtel	Daniel	1120	40	Closed Loop	Nat. Cooling Tower	2 (67%)	turbine	unk.	unk.
Catawba 1 & 2	W	Duke Power	Duke Power	1129	100	Closed Loop	Mech. Cooling Towers	2 (50%)	AC	unk	18,400
Comanche Peak 1 & 2	W	Gibbs & Hill	Brown & Root	1150	unk.	Once Through	Squaw Creek Resvoir	2 (50%)	turbine	986	19,800
D.C. Cook 1 & 2	W	AEP	AEP	1060	85	Once Through	Lake Michigan	2 (70%)	turbine	1,138	16,750
Diablo Canyon 1 & 2	W	Pacific Gas & Electric	Pacific Gas & Electric	1087	40	Once Through	Pacific Ocean	2	turbine	unk.	unk.
Farley 1 & 2	W	SCS/Bechtel	Daniel	823	40	Closed Loop	Mech. Cooling Towers	2	turbine	1,474	15,000 @5250 RPM
Ginna	W	Gilbert	Bechtel	470	40	Once Through	Lake Ontario	2 (50%)	AC	1,165	14,000 @853 psig
Haddam Neck	W	Stone & Webster	Stone & Webster	582	40	Once Through	Connecticut River	2 (50%)	AC	1,100	9,600
Indian Point 2	W	UE & C	Wedco	849	40	Once Through	Hudson River	2	turbine	unk.	1,530 @970 psig
Indian Point 3	W	UE & C	Wedco	965	45	Once Through	Hudson River	2	turbine	unk.	1,530 @970 psig
Kewaunee	W	FEI	FEI	503	40	Once Through	Lake Michigan	2 (60%)	AC	2,276	10,000
McGuire 1 & 2	W	Duke Power	Duke Power	1129	100	Once Through	Lake Norman	2	turbine	unk.	18,000
Millstone 3	W	Stone & Webster	Stone & Webster	1142	40	Once Through	Niantic Bay	2(50%) 1 (50%)	turbine AC	1,235 unk	19,650 unk
North Anna 1 & 2	W	Stone & Webster	Stone & Webster	915	40	Once Through	Cooling Lake	3 (50%)	AC	986	16,250
Point Beach 1 & 2	W	Bechtel	Bechtel	485	40	Once Through	Lake Michigan	2 (50%)	AC	1,062	780 @941 psig
Prairie Island 1 & 2	W	FEI	Northern States Power	503	10	Closed Loop	Mech Cooling Towers	2 (65%)	AC	1,050	unk
Robinson 2	W	Ebasco	Ebasco	665	40	Once Through	Lake Robinson	2	AC	1,040	12,690
Salem 1 & 2	W	Public Service Electric& Gas	UE & C	1106	40	Once Through	Delaware River	2 (50%)	turbine	unk.	18,613 @884 psig
San Onofre 1	W	Bechtel	Bechtel	436	10	Once Through	Pacific Ocean	2 (50%)	AC	1,165	14,000 @853 psig
Seabrook 1	W	UE & C	UE & C	1150	50	Once Through	Atlantic Ocean	2 (50%)	turbine	unk.	17,200 @ 1019 psig
Sequoyah 1 & 2	W	TVA	TVA	1148	unk.	Combined Cycle	Nat. Cooling Tower & Tennessee River	unk.	unk.	unk.	unk

Table 4.5-7. Comparison of Westinghouse PWR Power Conversion Systems (Continued)

Plant Name	NSSS Vendor	Architect/ Engineer	Constructor	Turbine Gen. Cap. (MWe)	Turbine Bypass Capability (%)	Condenser Cooling Type	Normal Heat Sink	# Main FW Pumps	FW Pump Drive Type	Shutoff Head (psig)	Capacity (gpm)
Shearon Harris 1 & 2	W	Ebasco	Daniel	860	81	Closed Loop	Nat Cooling Tower	2 (50%)	AC	1,517	15,115 @ 1031 psig
South Texas 1 & 2	W	Bechtel	Ebasco	1250	unk.	Closed Loop	7000 Acre Cooling Pond	unk.	unk.	unk.	unk.
Summer	W	Daniel	Gilbert	885	unk	Once Through	James River	unk.	unk	unk.	unk
Surry 1 & 2	W	Stone & Webster	Stone & Webster	781	40	Once Through	James River	2 (50%)	AC	unk.	13,800
Trojan	W	Bechtel	Bechtel	1095	40	Closed Loop	Nat. Cooling Tower	2 (70%)	turbine	1,409	19,800 @876 psig
Turkey Point 3 & 4	W	Bechtel	Bechtel	666	40	Once Through	Mech. Cooling Towers	2 (60%)	AC	1,149	13,000 @ 815 psig
Vogtle 1 & 2	W	Bechtel	Georgia Power Co.	1079	40	Closed Loop	Nat. Cooling Towers	2	turbine	unk.	24,400 @ 1300 psig
Watts Bar 1 & 2	W	TVA	TVA	1177	40	Closed Loop	Nat. Cooling Towers	2	turbine	unk.	23,600 @ 819 psig
Wolf Creek	W	Bechtel/ S & L	Daniel	1128	unk.	Once Through	6000 Acre Cooling Pond	2(67%)	turbine	unk.	unk.
Yankee Rowe	W	Stone & Webster	Stone & Webster	167	unk.	Once Through	Sherman Pond	3	AC	unk.	2,160 @945 psig
Zion 1 & 2	W	Stone & Webster	Commonwealth Edison	1040	40	Once Through	Lake Michigan	2 (50%) 1 (50%)	turbine AC	1,600 unk.	15,800 @1160 psig unk

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Chapter 19.0

Combustion Engineering Plant Description

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19.0 COMBUSTION ENGINEERING PLANT DESCRIPTION

19.1 Introduction

This chapter provides a basic introduction to the Combustion Engineering (CE) technology by discussing the major differences between a Westinghouse design and a CE design. The first part of the discussion will be about the mechanical systems, specifically the reactor coolant system, the steam generator, the emergency core cooling systems, the control element assembly, and the control element drive mechanism. The second part will discuss plant protection and monitoring systems.

19.2 Mechanical Systems

19.2.1 Reactor Coolant System

The reactor coolant system consists of two heat transport loops, each of which has two reactor coolant pumps and one steam generator. The reactor coolant exits the reactor vessel and is transported through hot leg (T_h) piping to the steam generators. The reactor coolant leaves the steam generator through two cold legs (T_c), each containing a reactor coolant pump. In each loop, the coolant is returned to the reactor vessel.

Figure 19-1 shows an elevation view of the reactor coolant system. Figure 19-2 shows a plan view of the system. The hot leg piping is 42" in diameter, and the cold leg piping is 30". The reactor coolant system is designed to 2500 psia, with normal operating pressure around 2250 psia. T_{avg} at 100% power is 583°F.

19.2.2 Steam Generator

The CE steam generators are vertical, inverted, U-tube, tube and shell heat exchangers similar to the Westinghouse design. Each of the two steam generators in a CE plant are much larger than those in a four loop Westinghouse plant with the same rated electrical output. Each CE steam generator has 8,400 tubes providing 86,000 square feet of heat transfer area. Figure 19-3 shows the design features of a CE steam generator.

19.2.3 Emergency Core Cooling Systems

The emergency core cooling systems (Figure 19-4) consist of the high head injection system (HPSI), the low head injection system (LPSI), and the safety injection tanks (SITs).

The high head injection system consists of two trains. Borated water is taken from the refueling water storage tank during the injection phase or from the containment sump during the recirculation phase and pumped to the cold legs through motor operated valves. The HPSI pumps have a discharge pressure of 1600 psig. Three non-safety related positive displacement pumps in the chemical and volume control system provide normal makeup to the RCS. These pumps charge water from boric acid makeup tanks into the RCS during an accident, but since they are non-safety related, this flow is not taken credit for in the FSAR accident analysis.

The low pressure injection system, or shutdown cooling system, consists of two trains. Water is taken from the refueling water storage tank during the injection phase. The LPSI pumps have a discharge pressure of 150 psig.

The LPSI pumps are capable of taking a suction from the recirculation sump, but the HPSI system is designed to perform the recirculation function. When the LPSI system is aligned for shutdown cooling, the LPSI pump takes a suction on the RCS hot leg and discharges the water through the shutdown cooling heat exchangers to the RCS cold legs. Note that the shutdown cooling heat exchangers are normally aligned in the containment spray flowpath.

There are four safety injection tanks, one on each cold leg. The SITs are filled with borated water and pressurized with nitrogen. The normal pressure in the tanks is approximately 600 psig.

19.2.4 Control Element Assembly and Drive Mechanism

A CE control element assembly (CEA) has a spider and hub design with five fingers which are nearly one inch in diameter and consist of boron carbide pellets. A CEA is shown in Figure 19-5. The control element drive mechanism is a magnetic jack design (Figure 19-6), except five coils are used instead of three. A control element drive mechanism control system (CEDMCS) is used to automatically or manually move the CEAs.

19.3 Plant Protection and Monitoring Systems

19.3.1 Reactor Protection System (RPS)

A simplified CE RPS is shown in Figure 19-7. First of all, CE uses separate instruments for protection and control. If one of the protection channel parameters exceeds its trip value, the associated bistable will trip. This will deenergize the trip relay in that channel. The six logic matrices consist of a series-parallel contact network (Figure 19-8) and are used to determine whether the two out of four coincidence trip logic has been satisfied.

When a logic matrix determines that the trip coincidence is satisfied, the associated logic matrix relays deenergize, opening the associated trip path contacts. When these contacts open, all circuit breaker control relays deenergize and all reactor trip circuit breakers open. Eight reactor trip circuit breakers are in the circuit between the motor generator sets and the CEDM coils. One pair of breakers on each side must open for the CEAs to trip into the core.

The engineered safety features actuation system operates very similar to the RPS described above.

19.3.2 Core Protection Calculators (CPC)

Core protection calculators (Figure 19-9) have been added to the newer CE plants to generate reactor trip signals based upon local power density and DNBR, which prevents these limits from being exceeded during anticipated operational occurrences. The CPC is a digital computer that continuously calculates a conservative value of plant local power density and DNBR using safety channel inputs from RCS flow, RCS pressure, RCS temperatures, reactor power, and flux distribution.

19.3.3 Core Operating Limits Supervisory System (COLSS)

The core operating limits supervisory system (Figure 19-10) is a plant computer program which provides comprehensive and continuously updated information. The program consists of on-line power distribution, DNBR correlation, calorimetric power, and maximum linear power generation rate calculations. When the COLSS is operable, the plant Technical Specifications allows the plant to operated closer to the kw/ft and DNBR limits.

19.4 Summary

This chapter discussed the major differences between a Westinghouse design plant and a Combustion Engineering design plant. The CE plant has two reactor coolant loops, each of which has two reactor coolant pumps and one steam generator.

The emergency core cooling systems in a CE plant consist of a high pressure injection system which is also used for the recirculation phase, a low pressure injection system which is also used for shutdown cooling, and four safety injection tanks.

The CE reactor protection system uses a two out of four coincidence logic for reactor trips and engineered safety features actuations. The core protection calculator and the core operating limits supervisory system allows the plant to operate closer to the kw/ft and DNBR safety limits.

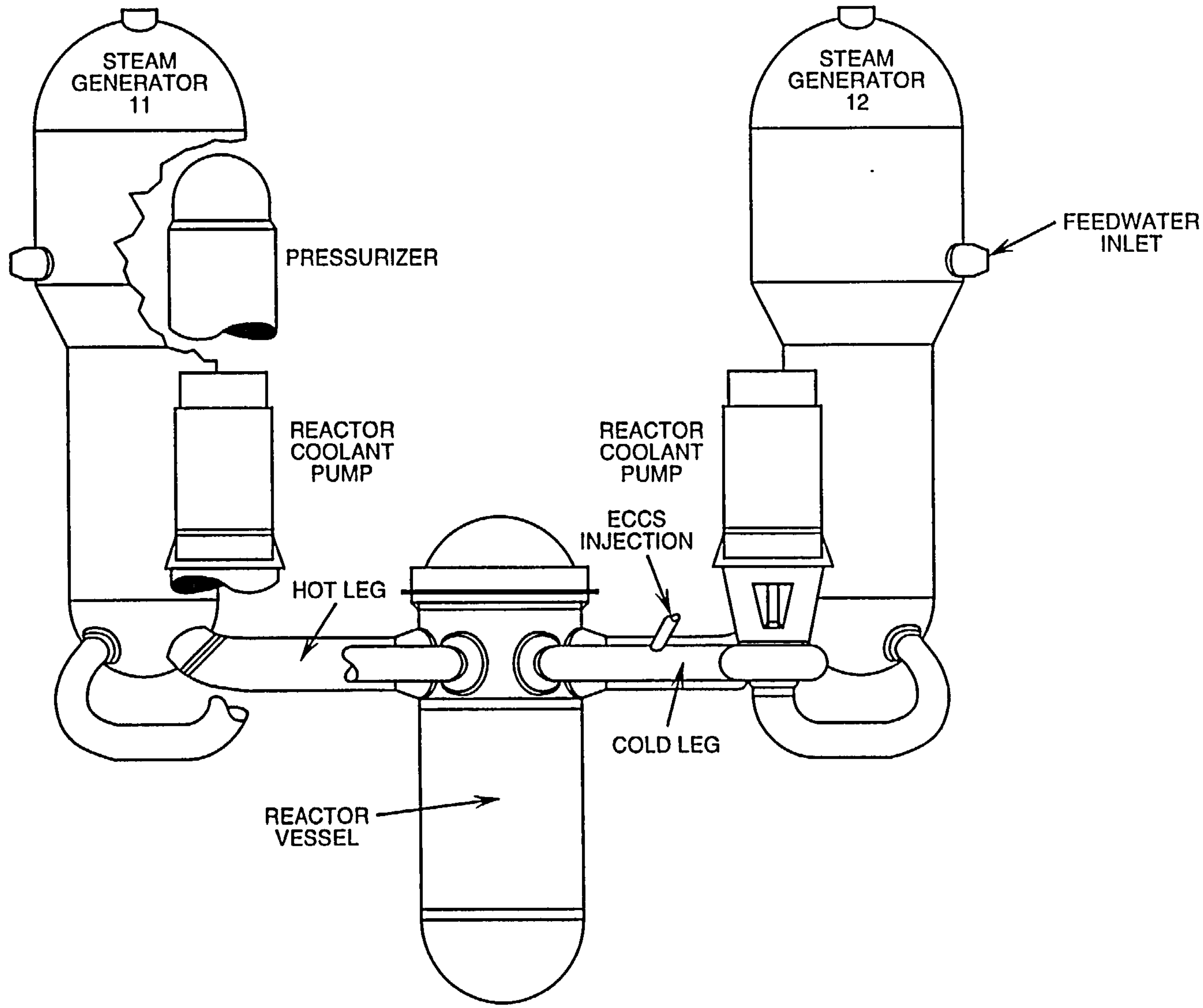

Figure 19-1 Reactor Coolant System - Elevation View

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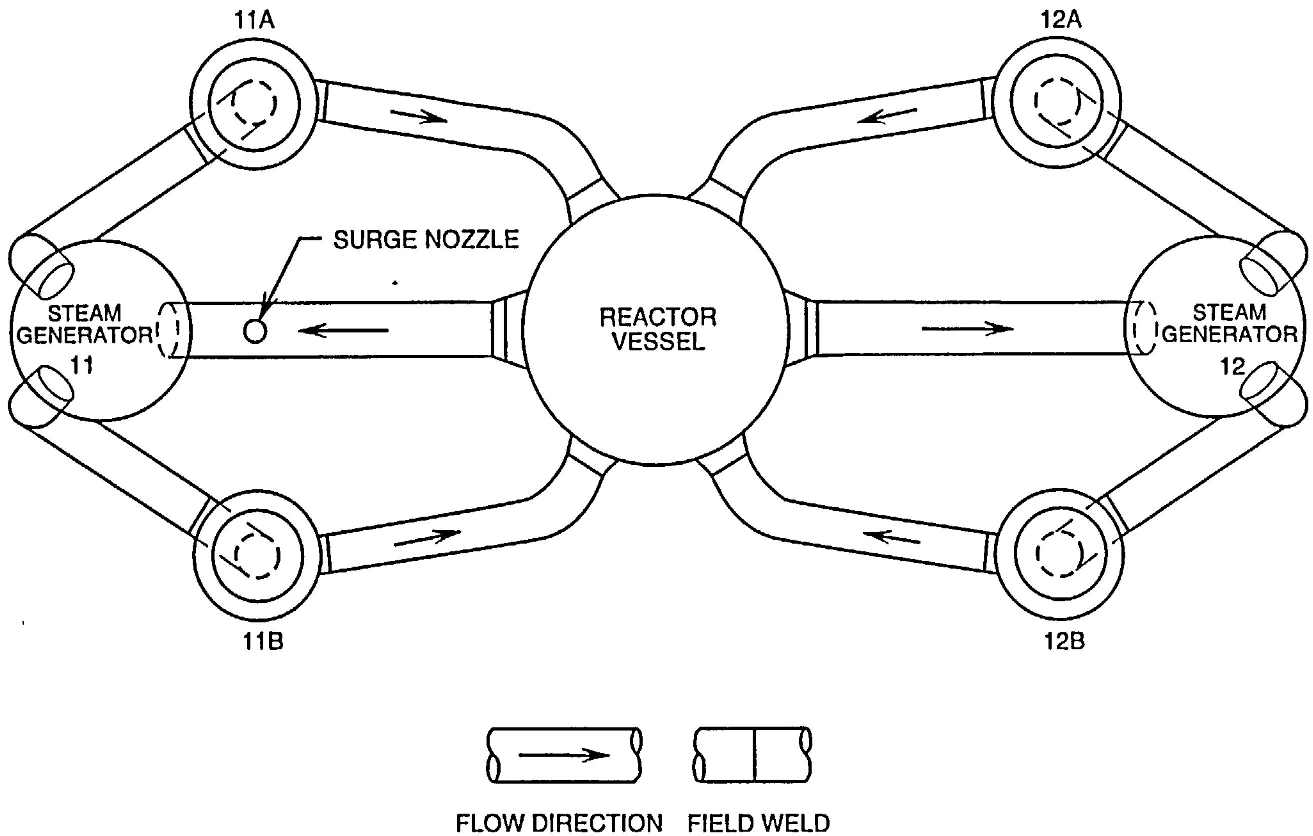

Figure 19-2 Reactor Coolant System - Plan View

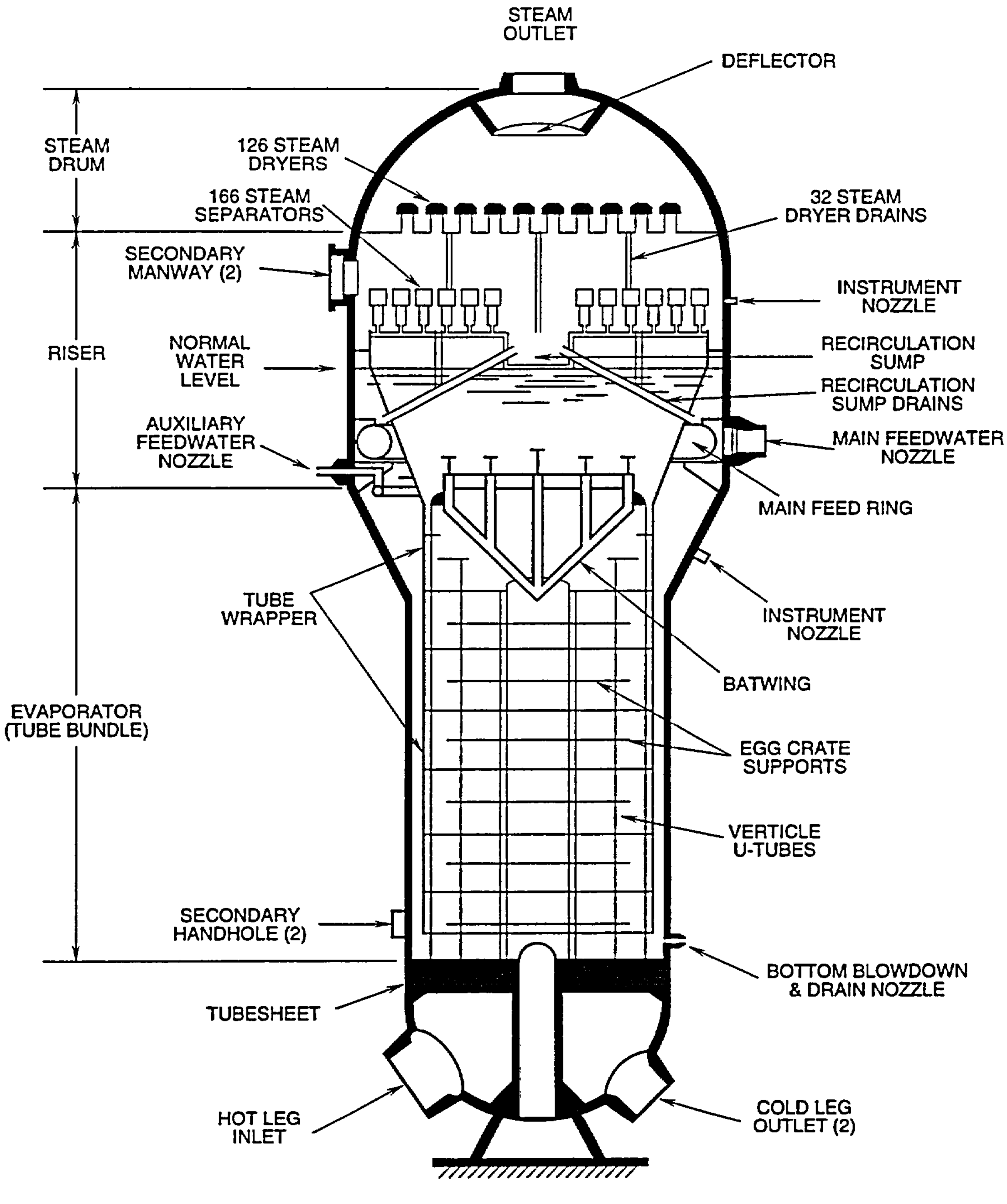

Figure 19-3 Steam Generator Secondary Side

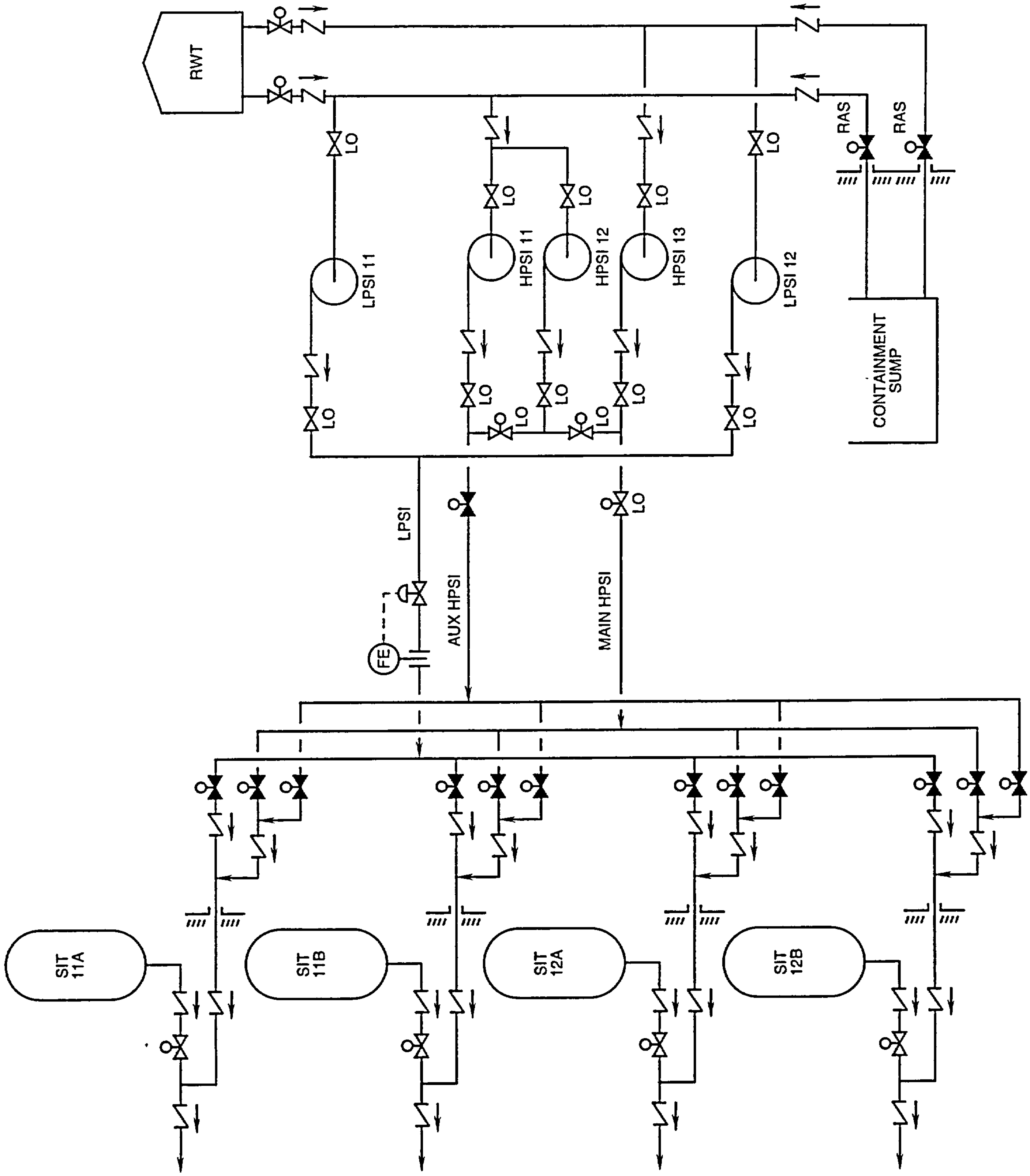

Figure 19-4 Emergency Core Cooling Systems

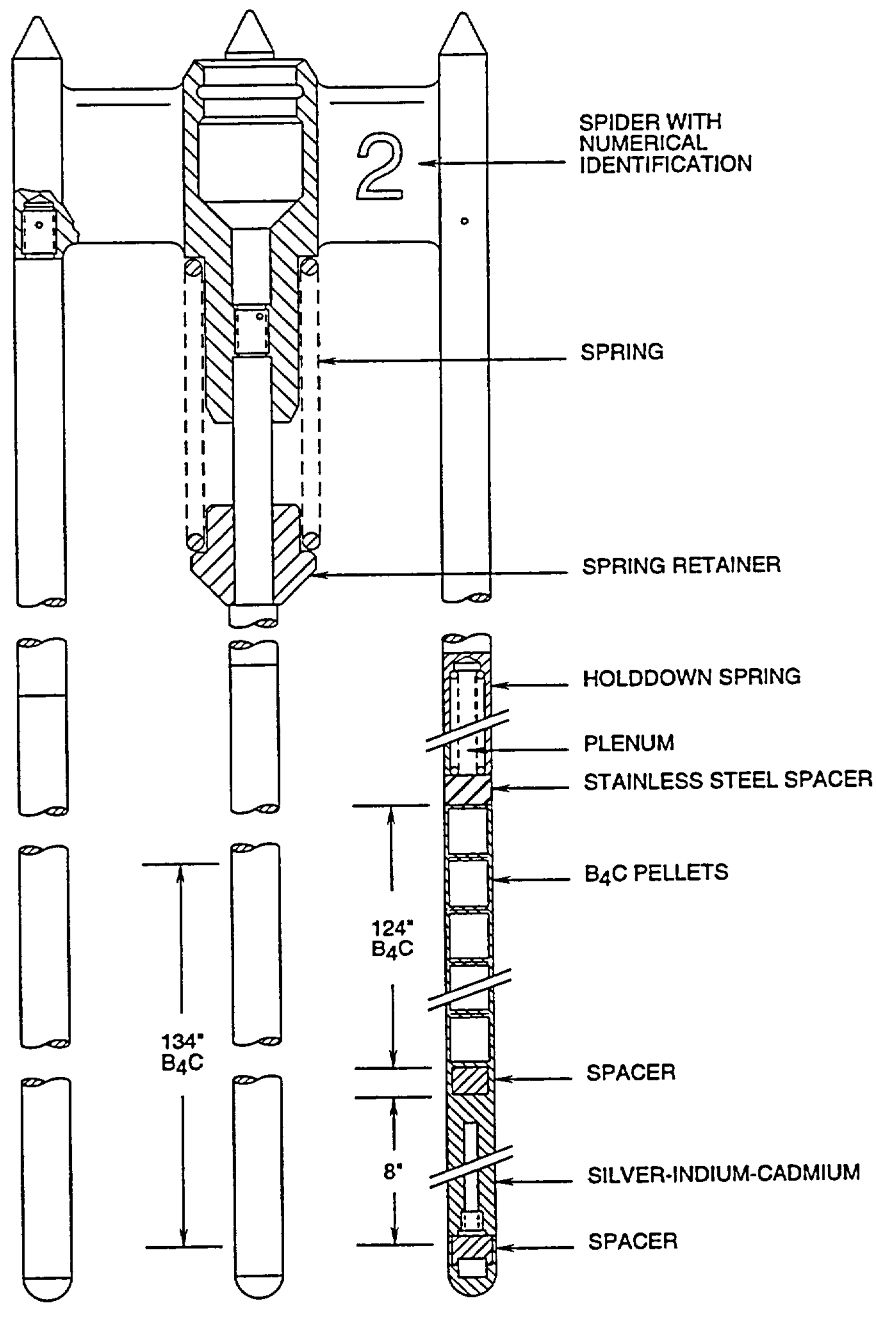

Figure 19-5 Full Length CEA

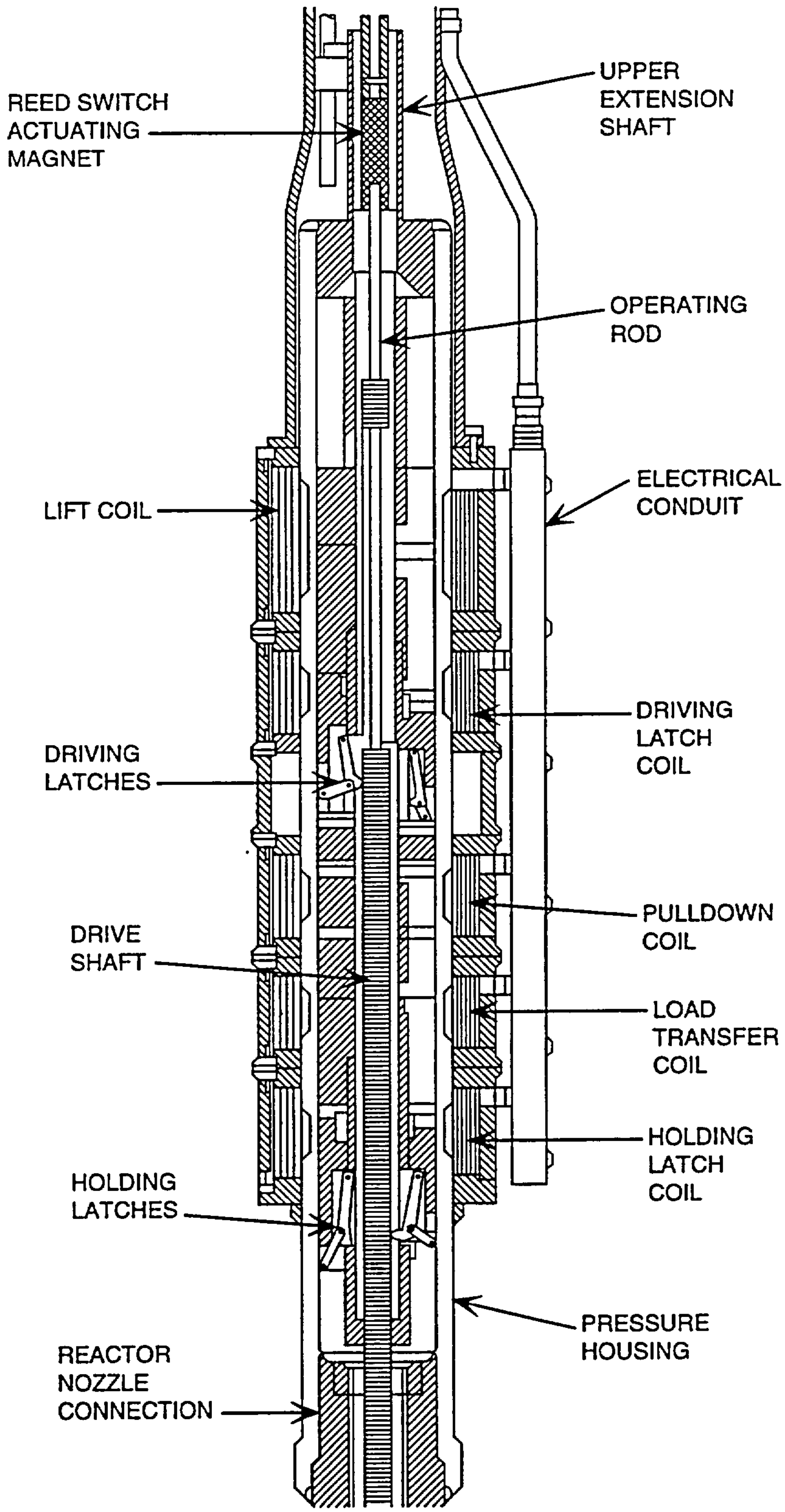

Figure 19-6 Control Element Drive Mechanism

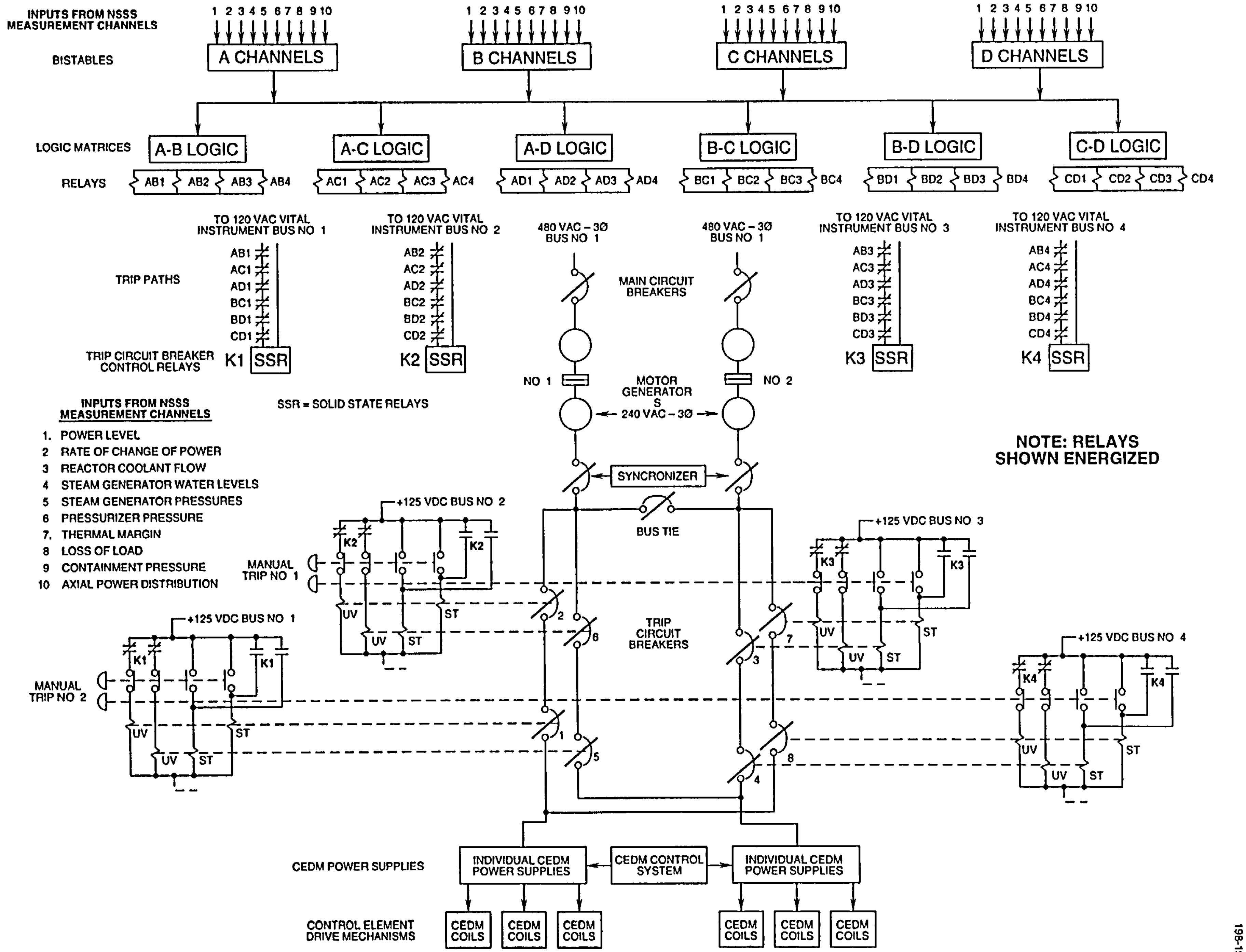

Figure 19-7 Simplified Reactor Protection System

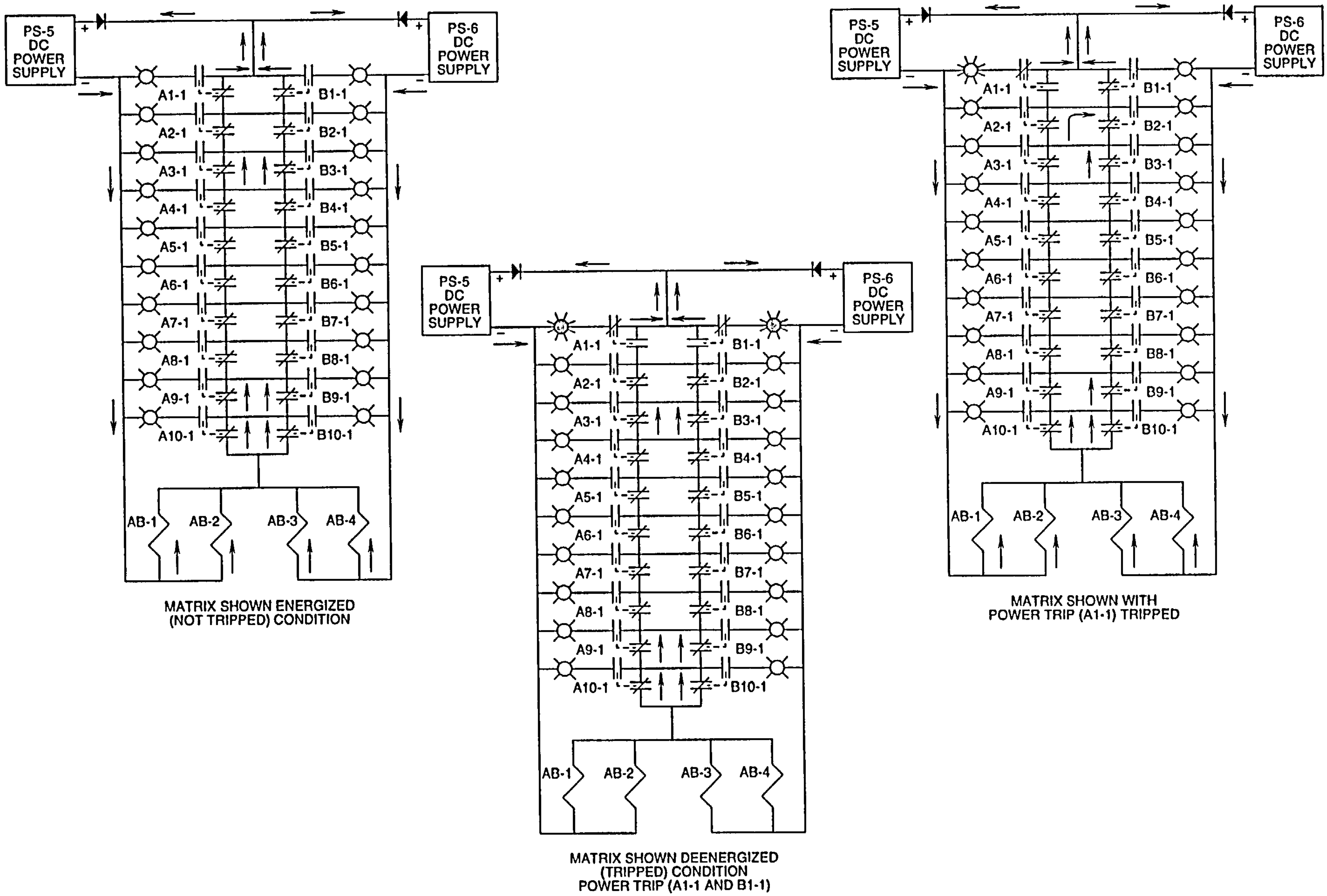

Figure 19-8 Coincidence Logic Matrix AB

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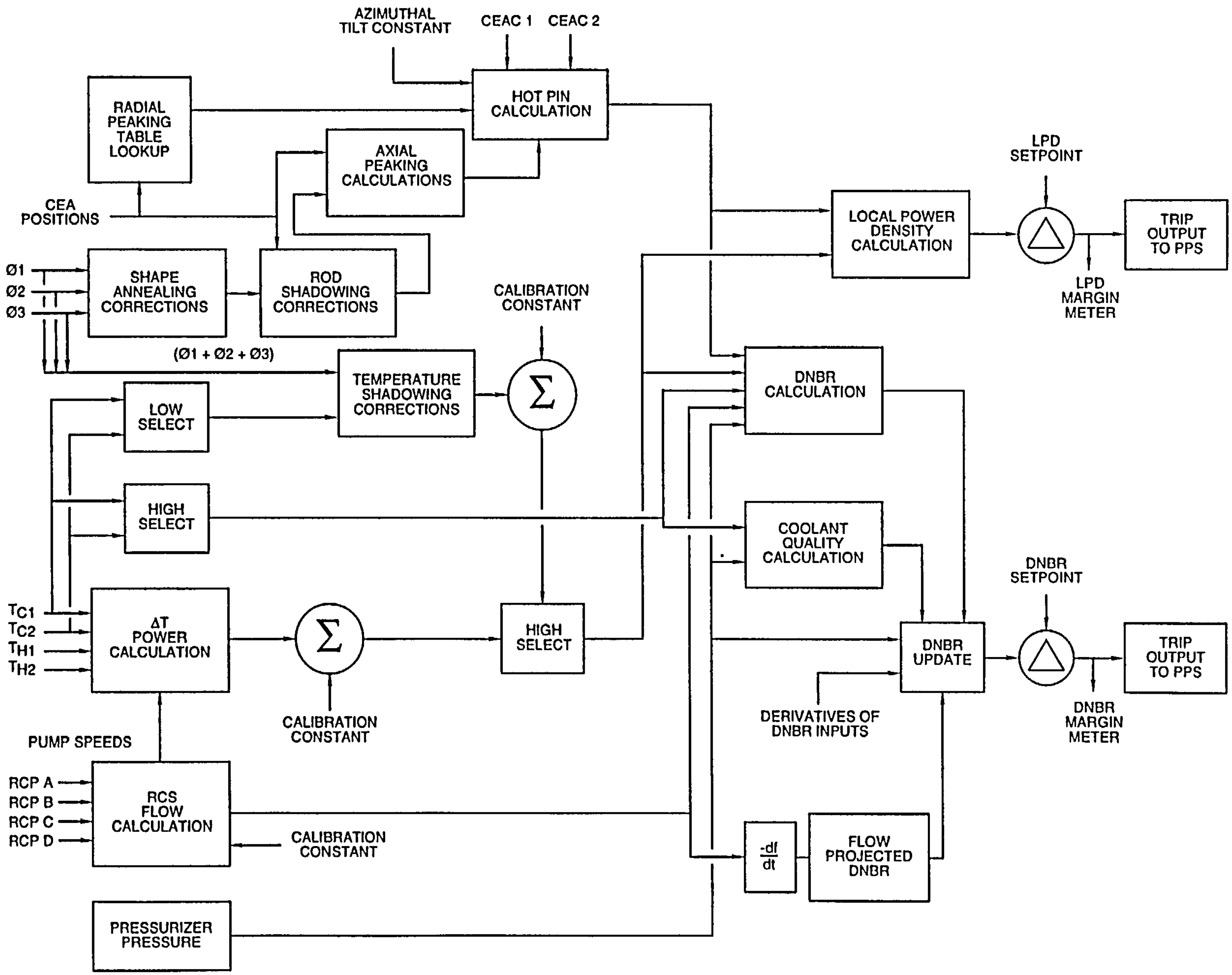

Figure 19-9 CPC Software Block Diagram

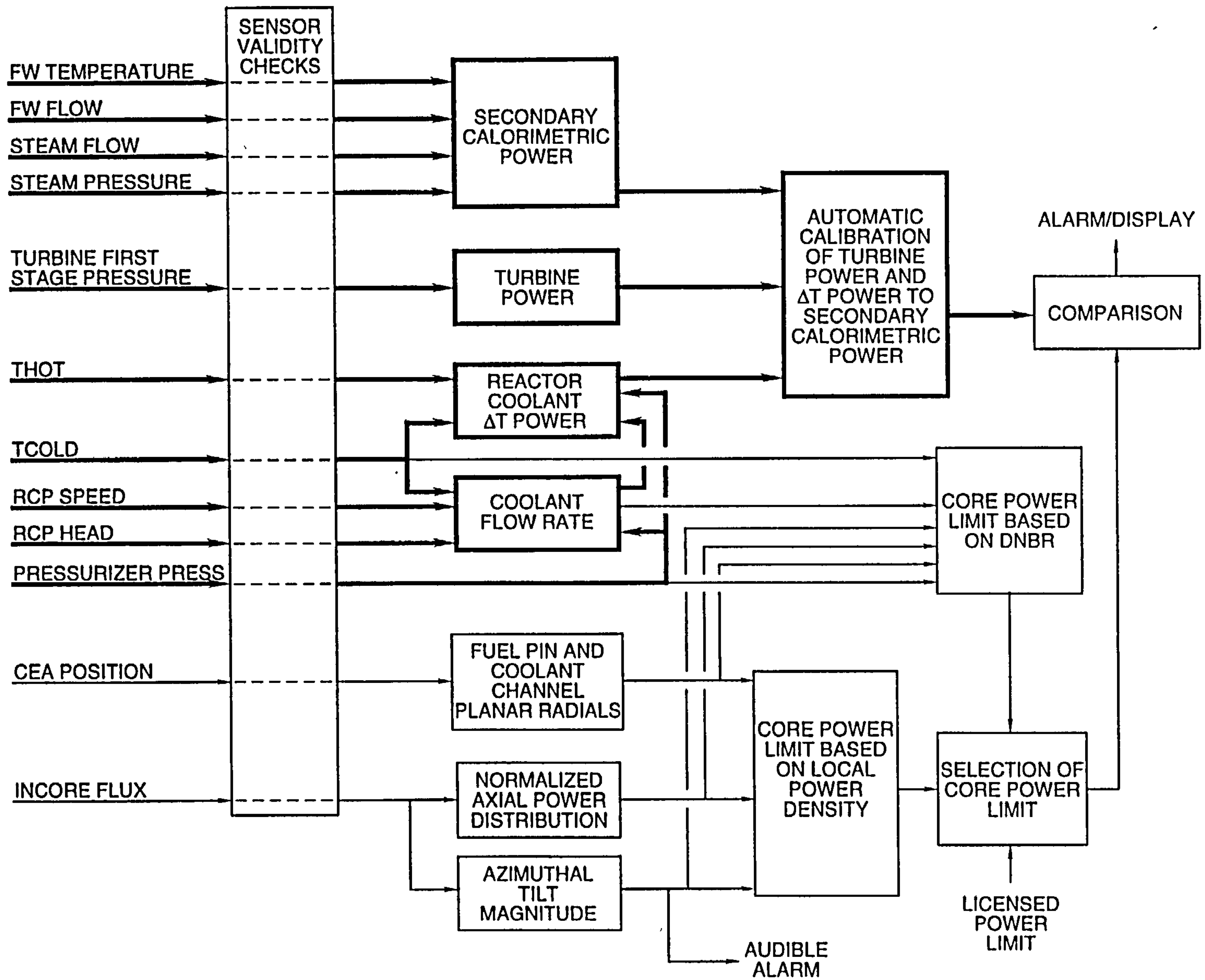

Figure 19-10 Core Operating Limits Supervisory System Block Diagram

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Chapter 20.0

Babcock and Wilcox Plant Description

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20.0 BABCOCK & WILCOX PLANT DESCRIPTION

20.1 Introduction

This chapter provides a basic introduction to the Babcock & Wilcox technology by discussing the major differences between a Westinghouse design and a B&W design. The first part of the discussion will be about the mechanical systems, specifically the reactor coolant system, the steam generator, the emergency core cooling systems, and the control rod drive mechanism. The second part will discuss the control systems, specifically the integrated control system and the reactor protection system.

20.2 Mechanical Systems

20.2.1 Reactor Coolant System

The reactor coolant system consists of two heat transport loops, each of which has two reactor coolant pumps and one steam generator. The reactor coolant is transported through hot leg (T_h) piping connecting the reactor vessel to the steam generators. The heat generated in the core inside the reactor vessel is transferred to the secondary system in the steam generators. The coolant leaves the steam generator through two cold leg (T_c) connections, each containing a reactor coolant pump. In each loop, the coolant is returned to the reactor vessel.

Figure 20-1 shows the major components of a raised loop design of the reactor coolant system. The hot leg piping is 38" in diameter, and the cold leg piping is 32". The reactor coolant system is designed to 2500 psig, with normal operating pressure around 2195 psig. T_{avg} at 100% power is 601°F.

20.2.2 Once Through Steam Generator

The purpose of the steam generator is to take the heat from the primary coolant flowing inside of the tubes and make steam using the secondary water flowing around the tubes. This purpose is accomplished in the once through steam generators (OTSG), which is a slightly different design than the Westinghouse U-tube design.

Instead of having U shaped tubes, the OTSG (Figure 20-2) uses a straight tube design. There are approximately 16,000 tubes in the OTSG. The OTSG is a counterflow heat exchanger. That is, the primary coolant enters the tubes at the top of the OTSG and flows straight through the tubes to the bottom of the OTSG. The feedwater enters the OTSG at the bottom and flows to the top of the tube bundle. At the primary outlet, the flow splits into two paths, each going to a reactor coolant pump.

The operation of the OTSG is also slightly different from a U-tube steam generator design. The steam generated in a U-tube steam generator is saturated steam. Also, the amount of heat transfer area is constant with power. In an OTSG, the steam at the outlet of the OTSG has a minimum of 50°F superheat, and the heat transfer area varies with power.

At the bottom of the OTSG (Figure 20-3), the feedwater is heated to approximately a saturated condition in the subcooled region. The water begins to boil in the nucleate boiling region, and at the outlet of this region is about 95% steam. The steam then enters the film boiling region, where it is heated into saturated steam. Finally, the steam enters the superheat region and receives enough heat to provide the minimum of 50°F superheat.

The sizes of these heat transfer regions change with power. As power increases, the feedwater flow increases. The subcooled region will increase in size. The nucleate boiling region will also increase greatly in size. The size of the film boiling region is approximately a constant over power. The increase in size of the subcooled region and the nucleate boiling region results in a decrease in the size of the superheat region. However, the steam at the outlet of the OTSG still has a minimum of 50°F superheat. These changes in the amount of heat transfer area allows the operator to actually control primary temperature with feedwater if the control rods are not available.

20.2.3 Emergency Core Cooling Systems

The emergency core cooling systems consist of the high head injection system, the low head injection system, and the core flood system.

The high head injection system (Figure 20-4) consists of two trains. Water is taken from the borated water storage tank and pumped to the cold legs through motor operated valves. The valves can be throttled to control high pressure injection flow. The pumps in the high head system are used as the makeup pumps during normal operation.

The low pressure injection system (Figure 20-5), or decay heat removal system, consists of two trains. Water can be taken from the borated water storage tank during the injection phase or from the recirculation sump during the recirculation phase. The pumps discharge to the core flood nozzles on the reactor vessel.

The core flood system (Figure 20-6) consists of two tanks. The core flood tanks are filled with borated water and pressurized with nitrogen. The normal pressure in the tanks is approximately 600 psig. The tanks discharge into the core flood nozzles on the reactor vessel.

20.2.4 Control Rod Drive Mechanism

The control rod drive mechanism for a B&W plant is also slightly different from a Westinghouse drive mechanism. Instead of using a stepping motor, the mechanism uses a leadscrew and roller nut assembly.

The major parts of the B&W drive mechanism are shown in Figure 20-7. A synchronous reluctance motor is used to provide the driving force for the control rod drive mechanism. The motor stator is located outside of the motor tube and the rotor on the inside of the motor tube. When energized, the upper part of the segmented arms of the rotor are pulled out, which pivots the roller nuts on the opposite end of the arms into the leadscrew. For every rotation of the roller nuts around the leadscrew, the leadscrew will move 0.750 inches. To prevent the leadscrew from rotating during rod motion, there is a torque taker on the top of the leadscrew. The torque taker transmits the torque to the torque tube and prevents rotation of the leadscrew. The torque taker also has a permanent magnet on it to close reed switches for rod position indication.

The control rod drive mechanisms are designed to drop the rods upon a loss of power. With no power to the drive mechanism, the segmented arms will pivot to the inward position due to springs. This causes the roller nuts to disengage the leadscrew, and the rod will fall.

20.3 Control Systems

20.3.1 Integrated Control System

The B&W plants use an integrated control system (ICS) to simultaneously control the main turbine, main feedwater flow control valves, main feedwater pumps, and the control rods. The ICS is shown in simplified form in Figure 20-8 and more detailed in Figure 20-9. The basic function of the system is to match generated megawatts to desired megawatts.

There are four major subassemblies in the ICS. These are:

- Unit load demand,
- Integrated master,
- Feedwater demand, and
- Reactor demand.

The unit load demand subassembly acts as the setpoint generator for the ICS. The operator can input the desired load and the desired rate of load change into this subassembly, and these signals are transmitted to the remainder of the ICS.

There are several functions of the integrated master subassembly. First, this subassembly controls the load of the turbine generator by positioning the turbine control valves. Another function is to feed the demand signal to the feedwater and reactor demand subassemblies. To do this, the integrated master modifies the signal being sent. This subassembly also controls the position of the steam dump valves. The final purpose is to maintain a constant load on the turbine, even when plant conditions are changing. For example, if circulating water temperature is higher than normal, the vacuum in the main condenser will be lower (higher absolute pressure). The output of the turbine will be less due to the loss of efficiency. The number of megawatts generated will be less than the desired megawatts. The error signal will cause an increase in the output of the feedwater and reactor demand subassemblies. The integrated master performs its functions by controlling at a constant steam pressure. If pressure goes up, the turbine valves will open to lower pressure (pick up more load), and vice versa.

The feedwater demand signal originates in the unit load demand and is modified by the integrated master. There is a separate control for each OTSG. The demand signal controls the position of the startup feedwater regulating valve and the main feedwater regulating valve, which are operated in sequence. That is, the startup feedwater regulating valve opens first and then the main valve. To maintain the proper differential pressure across the feedwater regulating valves, the feedwater demand subassembly also controls the speed of the main feed pumps.

The reactor demand subassembly controls the position of the control rods for the purpose of controlling reactor coolant system temperature. The demand signal again comes from the unit load demand and is modified by the integrated master.

20.3.2 Reactor Protection System

The reactor protection system for a B&W plant is significantly different from that of a Westinghouse plant.

The reactor protection system is shown in Figure 20-10. If one of the monitored parameters exceeds its trip value, the associated contact in that channel will open. This will deenergize the trip relay in that channel, which tells the other channels that one channel has seen a trip condition. If a second channel receives a trip signal (from the same or a different parameter), the reactor will trip.

Therefore, the reactor protection system for a B&W unit is two out of four reactor protection system channels, and not based upon a certain coincidence of only one parameter.

20.4 Summary

This chapter discussed the major differences between a Westinghouse design plant and a Babcock and Wilcox design plant. The B&W plant has two reactor coolant loops, each of which has two reactor coolant pumps and one steam generator. B&W plants use once through steam generators.

The emergency core cooling systems in a B&W plant consist of a high pressure injection system (which is also used for normal makeup to the reactor coolant system), a low pressure injection system (which is also used for decay heat removal), and a core flood system.

An integrated control system is used to control the main turbine, main feedwater flow control valves, main feedwater pumps, and the control rods in the B&W design plant. The reactor protection system coincidence for a B&W unit is two out of four channels of any combination of monitored parameters exceeding their setpoints.

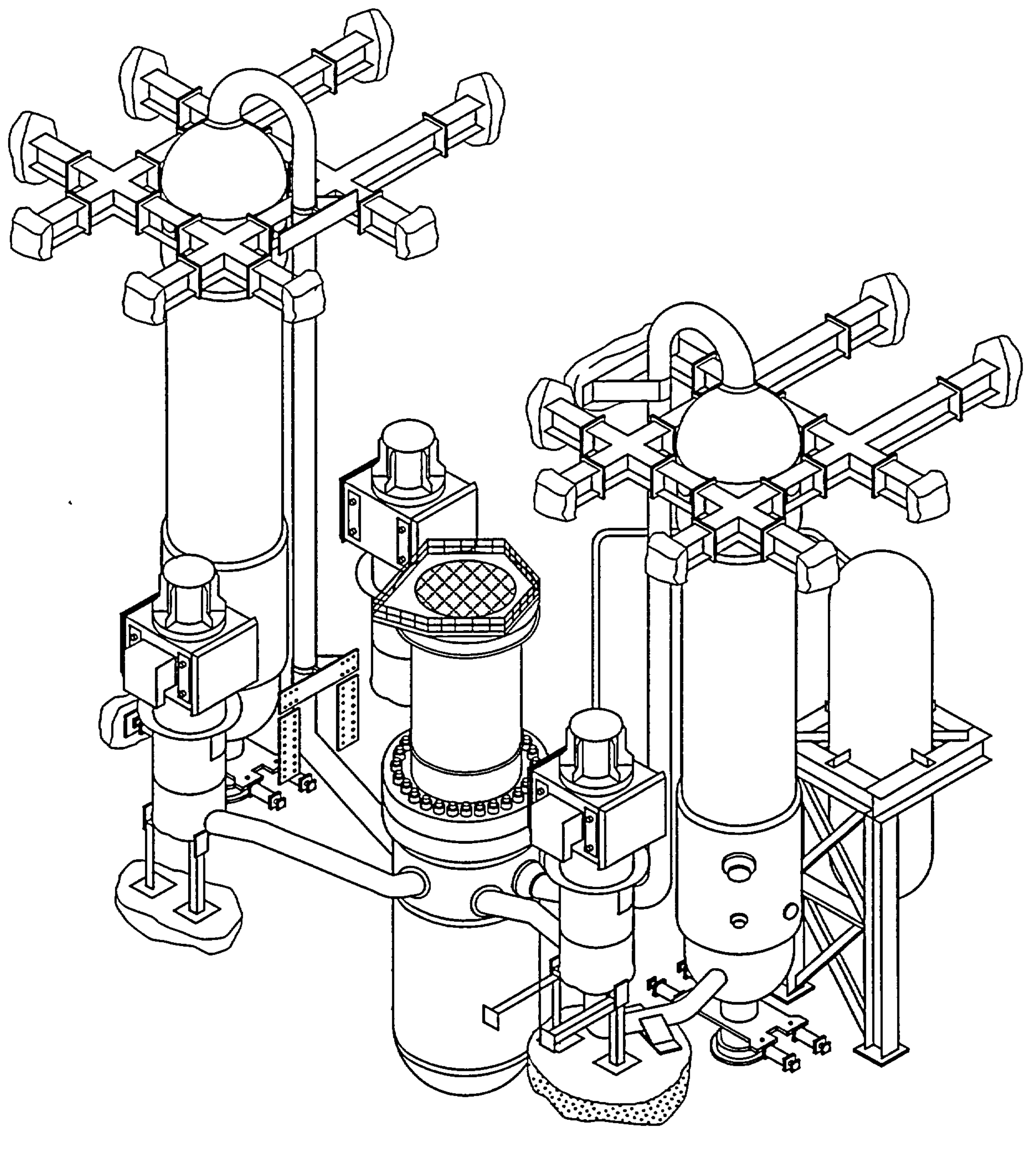

Figure 20-1 Reactor Coolant System Supports and Restraints

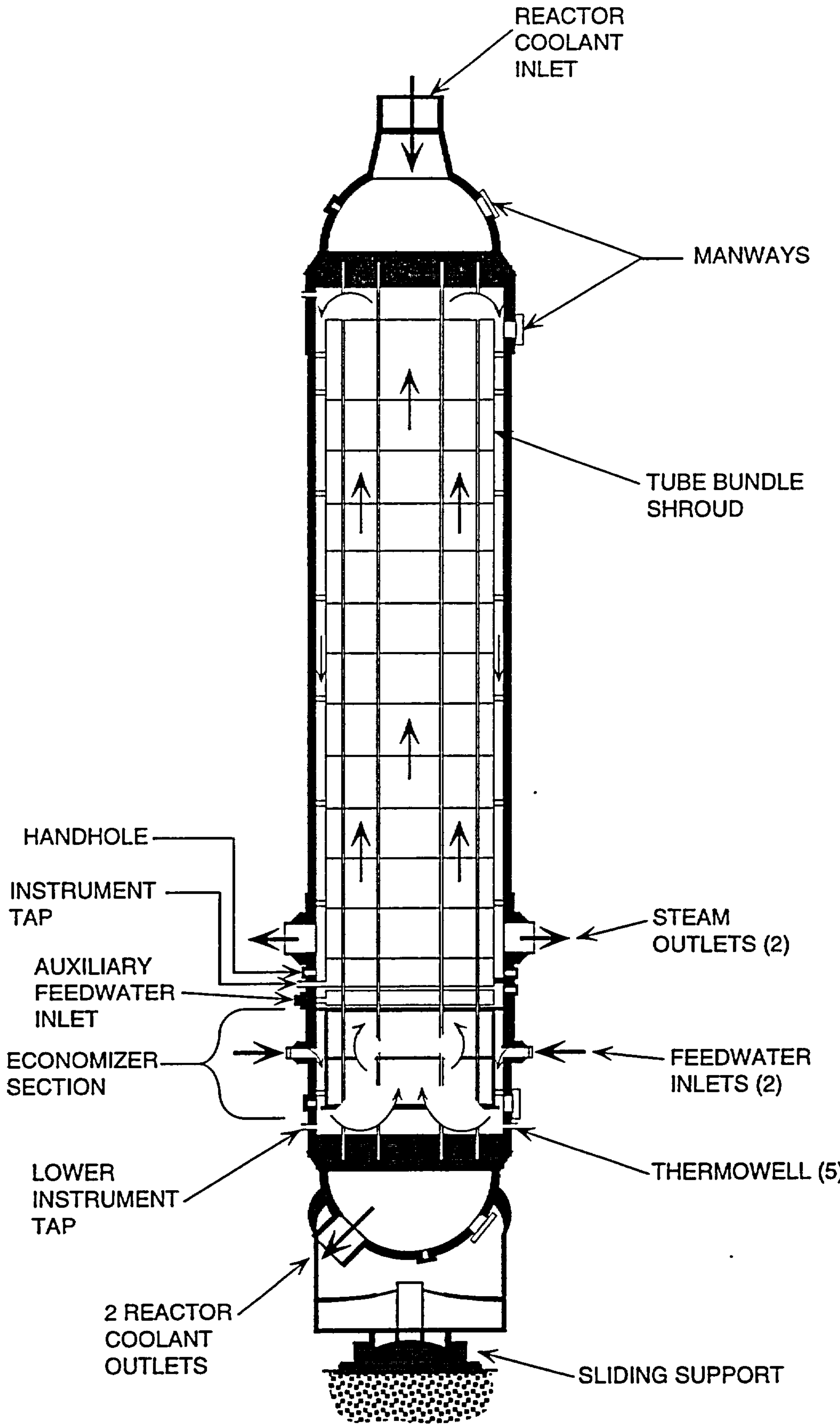

Figure 20-2 Integral Economizer Once-Through Steam Generator

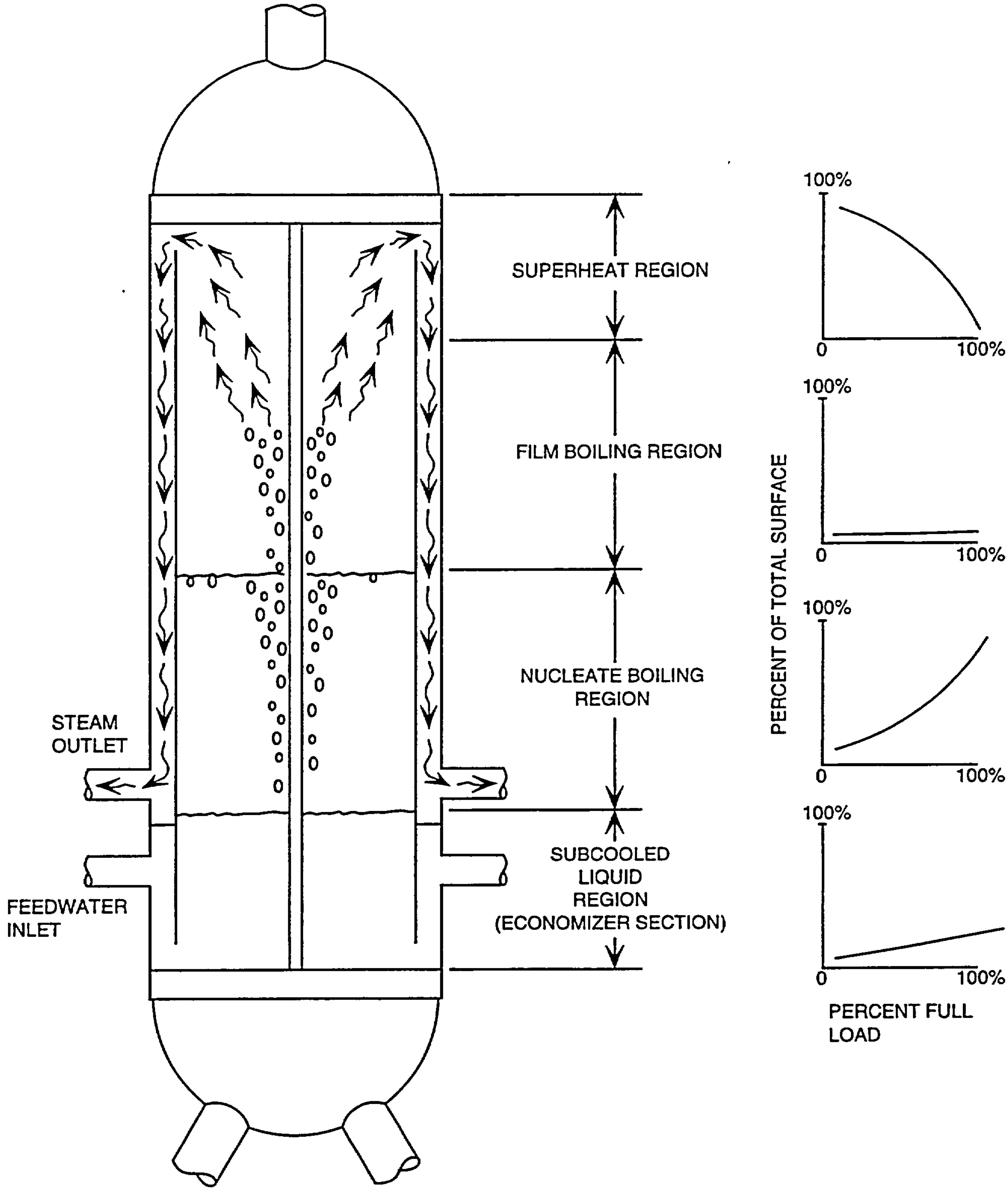

Figure 20-3 Heat Transfer Areas

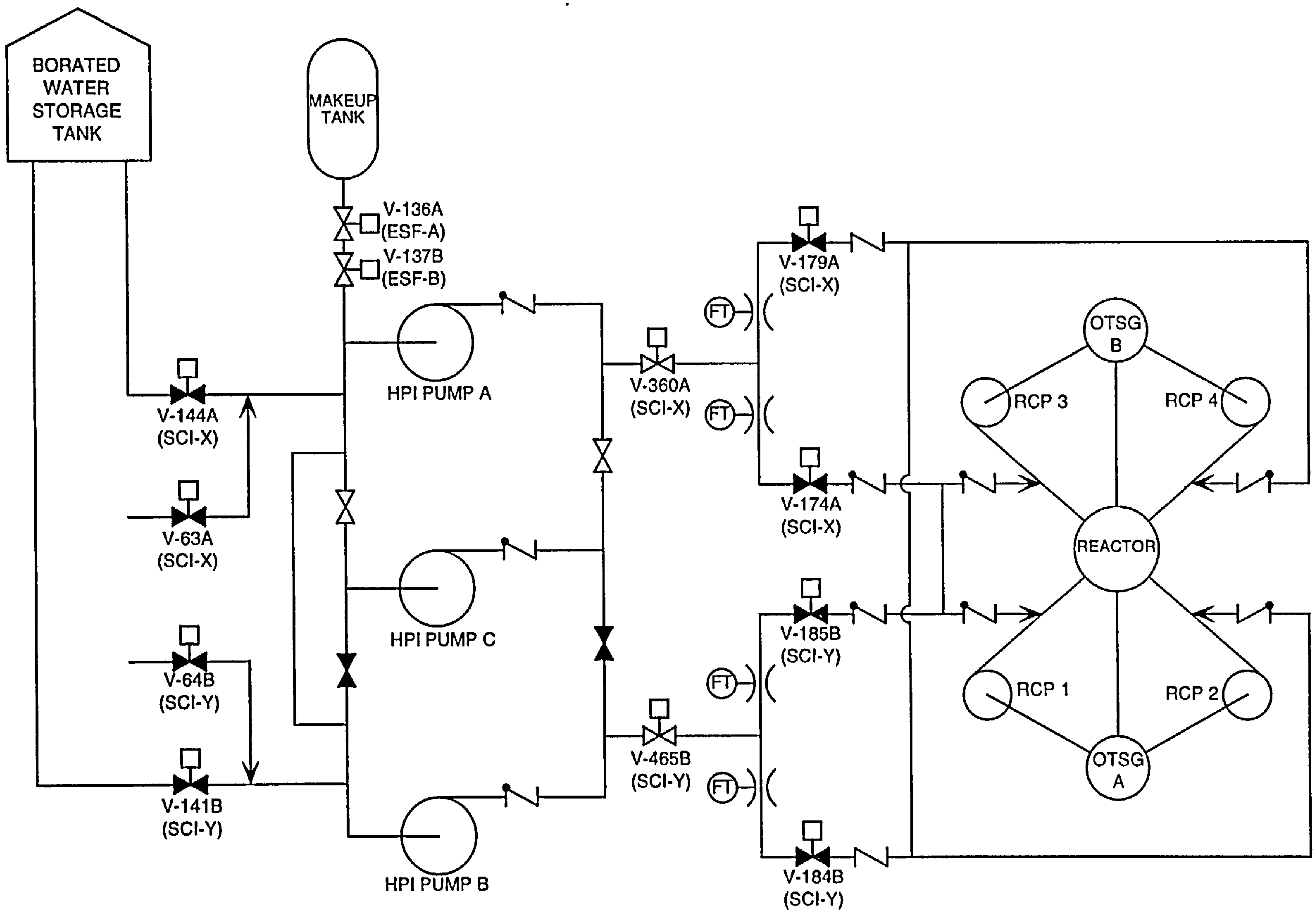

Figure 20-4 High Pressure Injection System

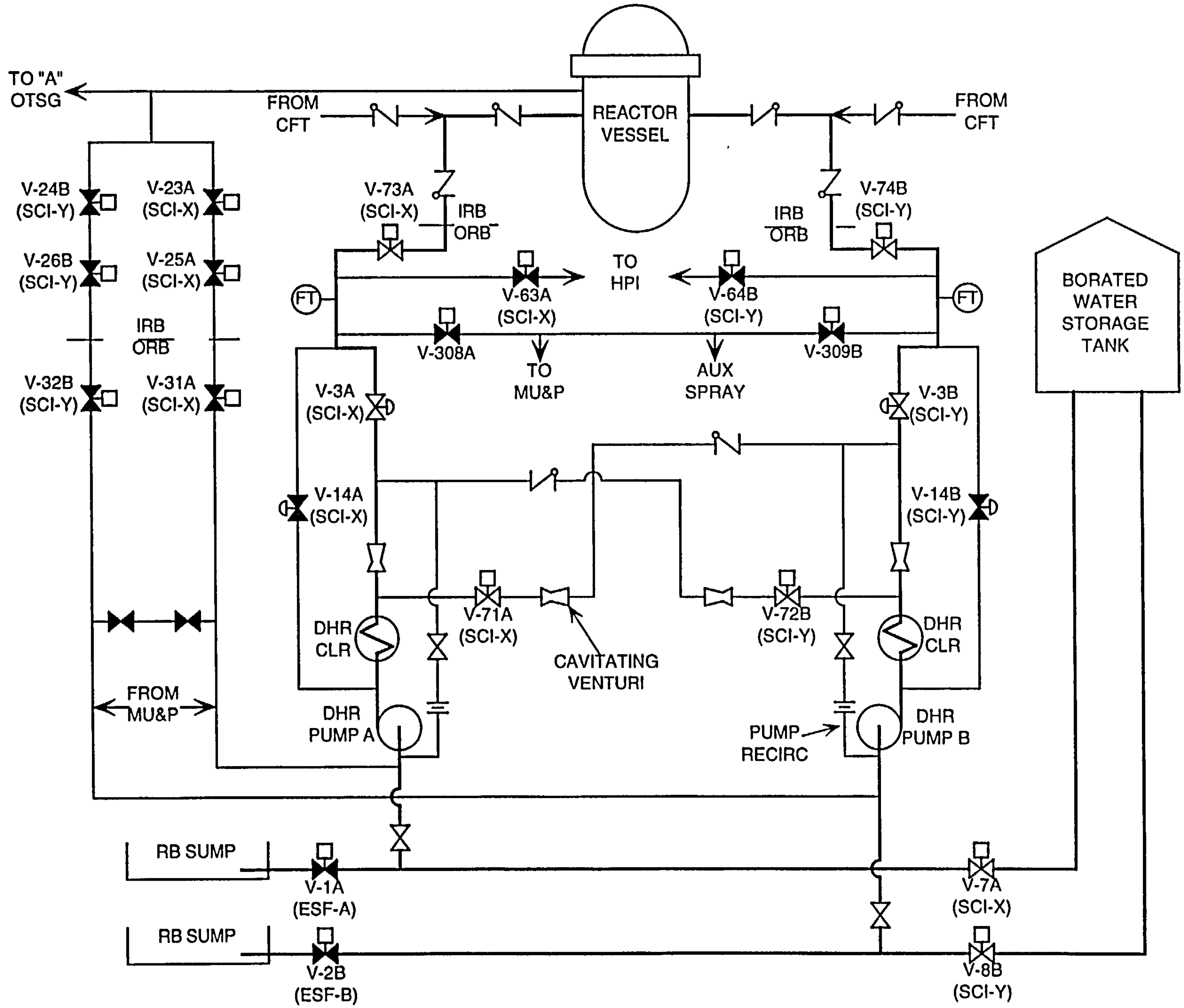

Figure 20-5 Decay Heat Removal System

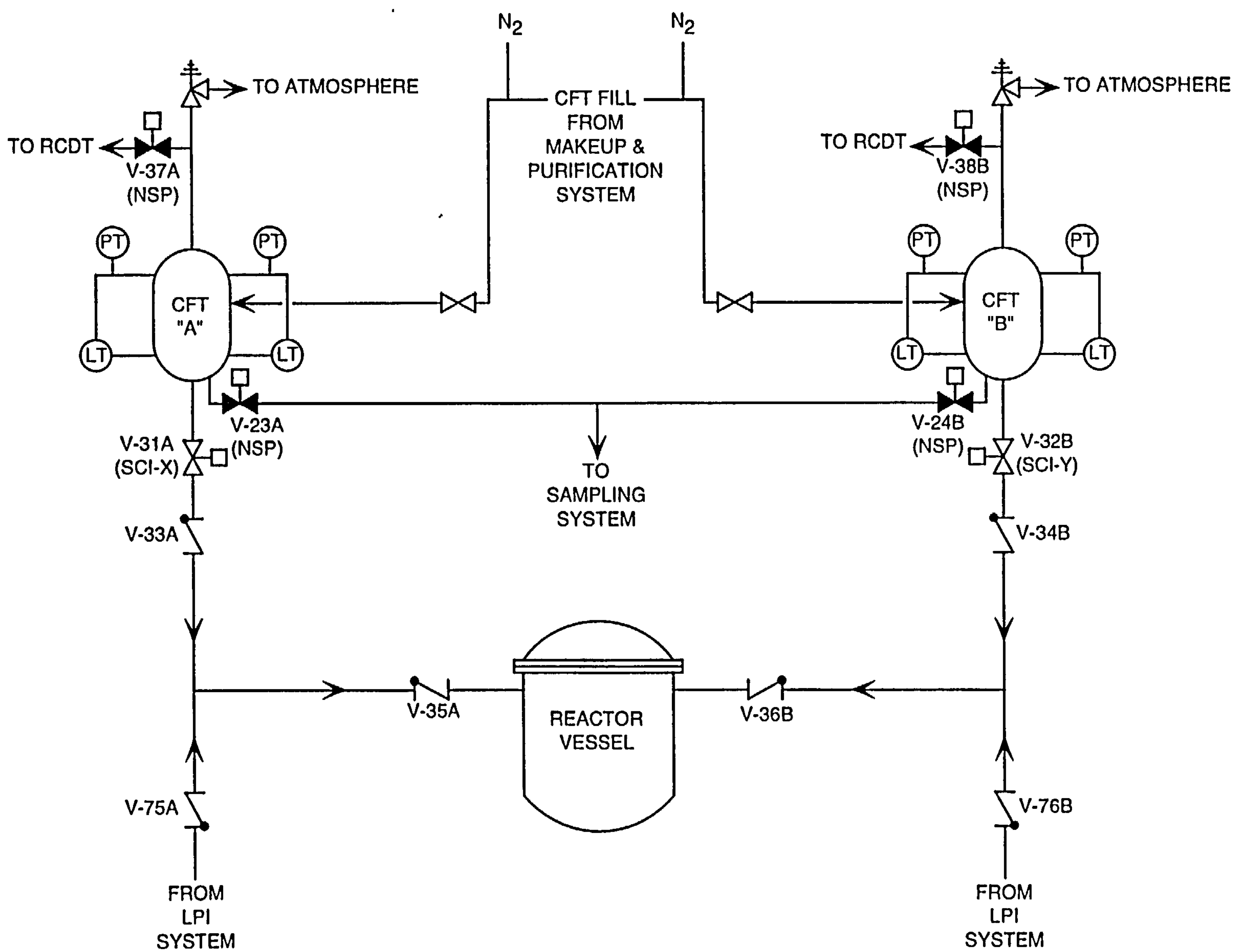

Figure 20-6 Core Flooding System

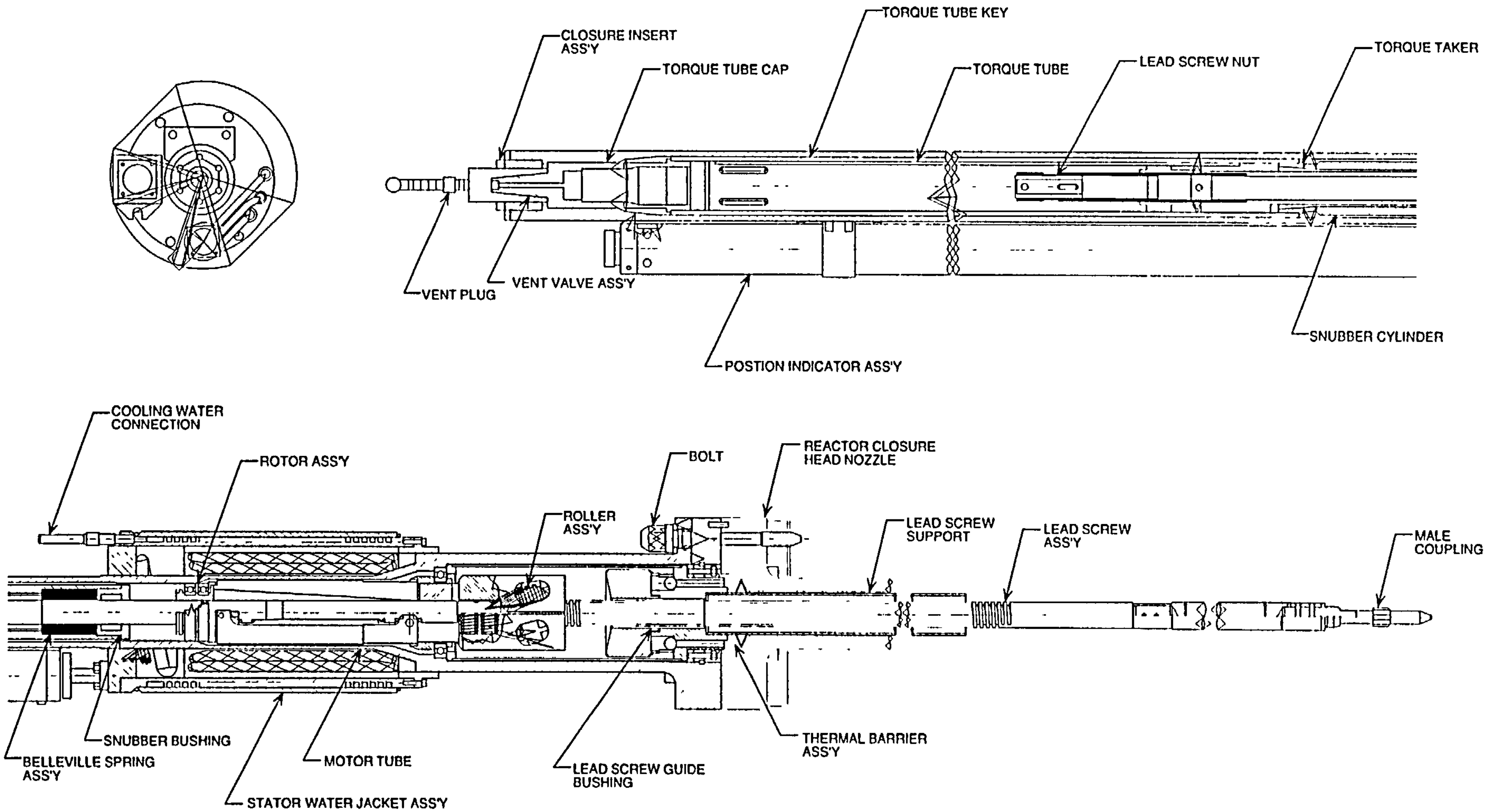

Figure 20-7 Control Rod Drive Mechanism Cross Section

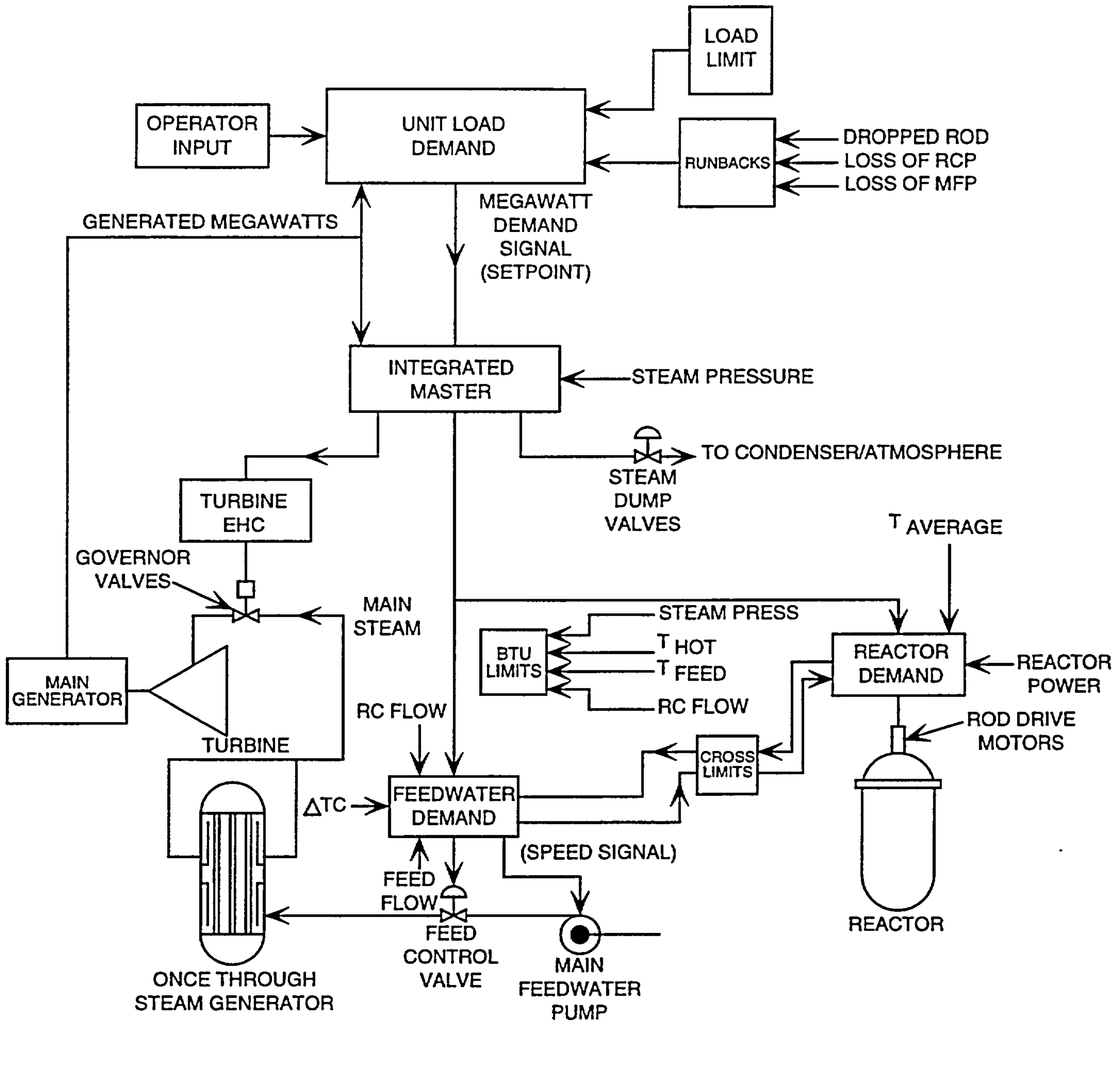

Figure 20-8 Simplified Integrated Control System

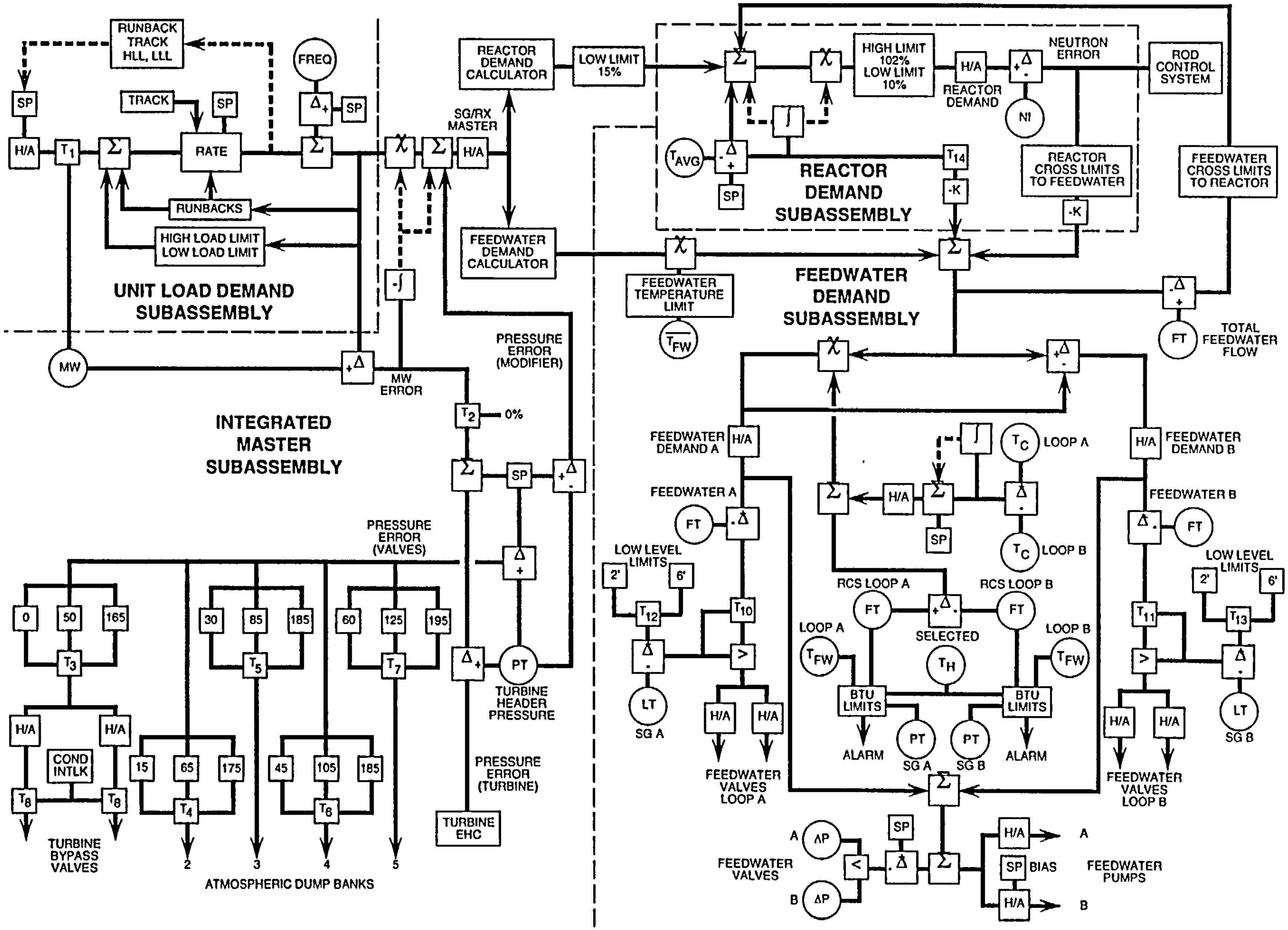

Figure 20-9 Integrated Control System (Detailed)

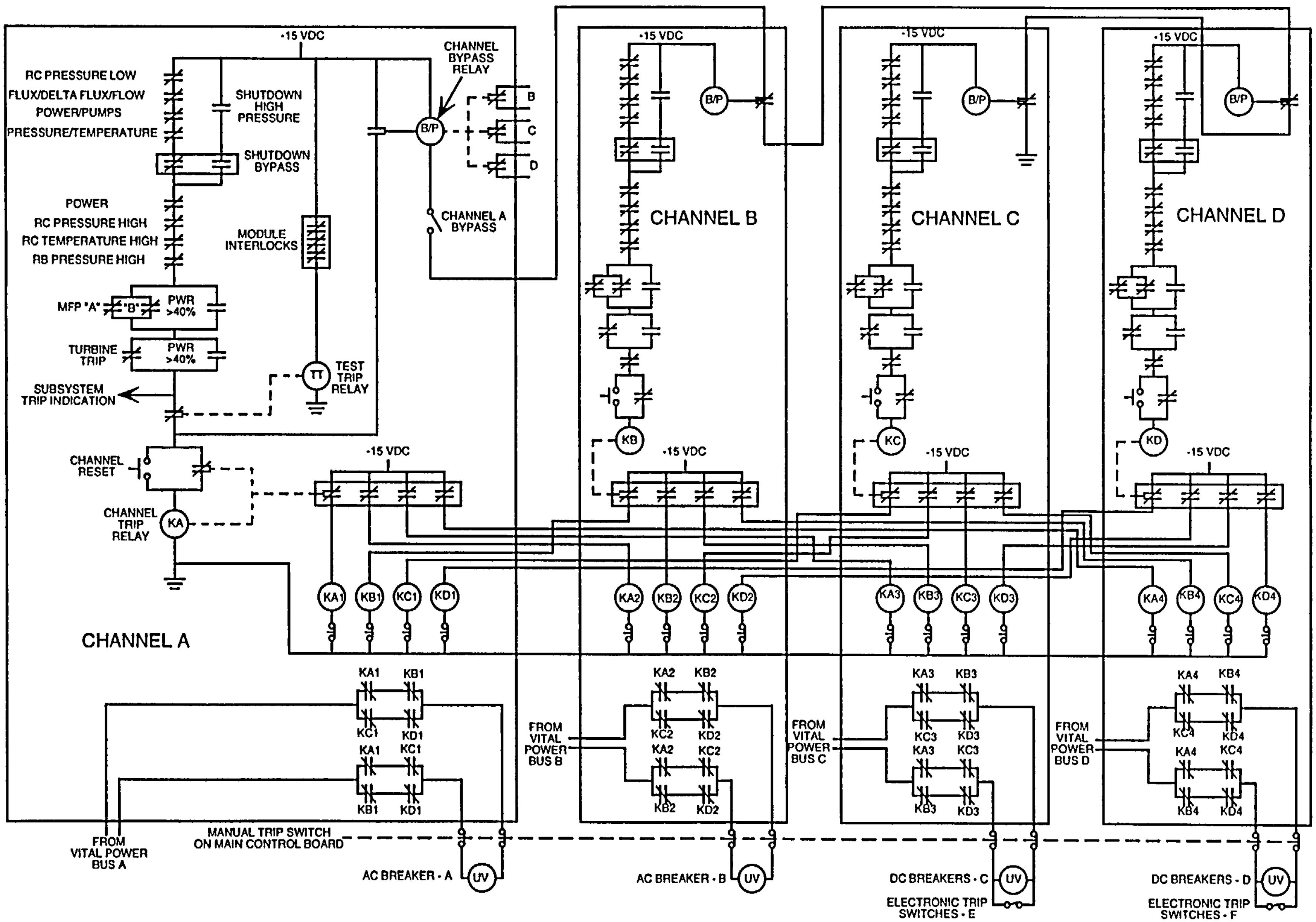

Figure 20-10 Reactor Protection System Channel Logic