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

Nuclear Power Plant System Sourcebook

Prepared by P. Lobner, C. Donahoe, C. Cavallin

Science Applications International Corporation

Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

This report is the introductory volume to the Nuclear Power Plant Sourcebook Series and is intended as a source of current summary and comparative information on U.S. commercial light water reactors (LWRs). The summary and comparative information is organized into the following four parts: (a) general U.S. LWRs, (b) pressurized water reactors (PWRs), (c) boiling water reactors (BWRs), and (d) bibliographies of general PWR and BWR references, plant-specific references, system-specific references, and component-specific references. This report is supplemented by a set of Sourcebooks that provides more detailed information on specific U.S. LWR plants.

TABLE OF CONTENTS

Section			Page
1.	Introduction		1-1
	1.1	Organization of This Introductory Volume	1-1
	1.2	Available Nuclear Power Plant System Sourcebooks	1-2
	1.3	Organization of a Nuclear Power Plant System Sourcebook	1-2
2.	General Comparative Data for U.S. Commercial Nuclear Power Plants		2-1
3.	Pressurized Water Reactors (PWR) System Overview		3-1
	3.1	Introduction to the Pressurized Water Reactor	3-1
	3.2	PWR Primary System	3-1
	3.3	Reactor Core and Fuel Assemblies	3-2
	3.4	Reactivity Control Systems	3-3
	3.5	Heat Transfer Systems for Power Operation	3-3
	3.6	Heat Transfer Systems for Shutdown Cooling at High RCS Pressure	3-4
	3.7	Heat Transfer Systems for Shutdown Cooling at Low RCS Pressure	3-4
	3.8	RCS Overpressure Protection System	3-4
	3.9	Emergency Core Cooling Systems	3-5
		3.9.1 ECCS Injection Phase	3-5
		3.9.2 ECCS Recirculation Phase	3-6
		3.9.3 High-Pressure Feed-and-Bleed Cooling	3-6
	3.10	Containment and Containment Auxiliary Systems	3-7
		3.10.1 Large, Dry Containment	3-7
		3.10.2 Subatmospheric Containment	3-7
		3.10.3 Ice Condenser Containment	3-7
		3.10.4 Containment Auxiliary Systems	3-7
	3.11	Component Cooling Systems	3-8
	3.12	Safety System Actuation	3-9
	3.13	Onsite Electric Power System	3-9
4.	Westinghouse Pressurized Water Reactors (PWRs)		4-1
	4.1	Westinghouse PWR Overview	4-1
	4.2	2-Loop Westinghouse PWRs	4-17
	4.3	3-Loop Westinghouse PWRs	4-24
	4.4	4-Loop Westinghouse PWRs	4-36
	4.5	Westinghouse PWR Comparative Data	4-55
5.	Combustion Engineering (C-E) Pressurized Water Reactors (PWRs)		5-1
	5.1	Combustion Engineering PWR Overview	5-1
	5.2	2-Loop (Pre-System 80) Combustion Engineering PWRs	5-12
	5.3	2-Loop System 80 Combustion Engineering PWRs	5-27
	5.4	3-Loop Combustion Engineering PWRs	5-37
	5.5	Combustion Engineering PWR Comparative Data	5-45

TABLE OF CONTENTS
(Continued)

Section			Page
6.	Babcock & Wilcox (B&W) Pressurized Water Reactors (PWRs)		6-1
	6.1	Babcock & Wilcox PWR Overview	6-1
	6.2	177 Lowered-Loop Babcock & Wilcox PWRs	6-14
	6.3	177 Raised-Loop Babcock & Wilcox PWRs	6-25
	6.4	205 Raised-Loop Babcock & Wilcox PWRs	6-29
	6.5	Babcock & Wilcox PWR Comparative Data	6-39
7.	Comparative Pressurized Water Reactor (PWR) Data		7-1
8.	Boiling Water Reactors (BWR) System Overview		8-1
	8.1	Introduction to the Boiling Water Reactor	8-1
	8.2	BWR Primary System (the "Nuclear Boiler")	8-2
	8.3	Reactor Core and Fuel Assemblies	8-3
		8.3.1 12 x 12 (Big Rock Point) Fuel Assembly	8-3
		8.3.2 7 x 7 Fuel Assembly	8-3
		8.3.3 8 x 8 Fuel Assembly	8-3
		8.3.4 9 x 9 Fuel Assembly	8-4
	8.4	Reactivity Control Systems	8-4
	8.5	Heat Transfer Systems for Power Operation	8-5
	8.6	Heat Transfer Systems for Shutdown Cooling at High RCS Pressure	8-5
	8.7	Heat Transfer Systems for Shutdown Cooling at Low RCS Pressure	8-6
	8.8	RCS Overpressure Protection System and Automatic Depressurization System (ADS)	8-7
	8.9	Emergency Core Cooling Systems	8-7
		8.9.1 High-Pressure ECCS Subsystems	8-7
		8.9.2 Low-Pressure ECCS Subsystems	8-8
	8.10	Containment and Containment Auxiliary Systems	8-8
		8.10.1 Large, Dry Containment	8-9
		8.10.2 Pre-Mark I Pressure Suppression Containment	8-9
		8.10.3 Mark I Pressure Suppression Containment	8-9
		8.10.4 Mark II Pressure Suppression Containment	8-10
		8.10.5 Mark III Pressure Suppression Containment	8-11
	8.11	Component Cooling Systems	8-12
	8.12	Safety System Actuation	8-12
	8.13	Onsite Electric Power System	8-13
9.	General Electric (GE) Boiling Water Reactors (BWRs)		9-1
	9.1	BWR/1	9-1
	9.2	BWR/2	9-10
	9.3	BWR/3	9-17
	9.4	BWR/4	9-22
	9.5	BWR/5	9-28
	9.6	BWR/6	9-35
10.	Comparative Boiling Water Reactor (BWR) Data		10-1

TABLE OF CONTENTS
(Continued)

Appendix

A Bibliography of General PWR and BWR References

B Bibliography of Plant-Specific References For U.S. Commercial Nuclear Power Plants

C Bibliography of Nuclear Power Plant System References

D Bibliography of Nuclear Power Plant Component References

LIST OF FIGURES

Figure		Page
1.3-1	Key to Symbols in Fluid System Drawings	1-14
1.3-2	Key to Symbols in Electrical System Drawings	1-16
3.2-1	Westinghouse 2-Loop PWR NSSS	3-12
3.2-2	Westinghouse 3-Loop PWR NSSS	3-13
3.2-3	Westinghouse 4-Loop PWR NSSS	3-14
3.2-4	Combustion Engineering 2-Loop PWR NSSS	3-15
3.2-5	Babcock & Wilcox 177 "Lowered-Loop" PWR NSSS	3-16
3.2-6	Babcock & Wilcox 177 "Raised-Loop" PWR NSSS	3-17
3.4-1	PWR Reactivity Control Systems: Control Rods and the Chemical and Volume Control System	3-18
3.5-1	Normal PWR Heat Transport Paths During Power Operation	3-19
3.6-1	PWR Heat Transport Paths During Auxiliary Feedwater System Operation	3-20
3.7-1	Normal PWR Heat Transport Paths During Shutdown Cooling	3-21
3.8-1	Typical Pressurizer Safety Valve	3-22
3.9-1	PWR Coolant Injection and Heat Transport Paths During a Large LOCA-ECCS Injection Phase	3-23
3.9-2	PWR Coolant Injection and Heat Transport Paths During Post-Transient, High-Pressure Feed-and-Bleed Cooling	3-24
3.10-1	Distribution of PWR Containment Types	3-25
3.10-2	Yankee-Rowe Large, Dry Containment (Steel Sphere)	3-26
3.10-3	Davis-Besse Large, Dry Containment (Steel Cylinder with Concrete Shield Building)	3-27
3.10-4	Diablo Canyon Large, Dry Containment (Reinforced Concrete with Steel Liner)	3-28
3.10-5	Zion Large, Dry Containment (Post-Tensioned Concrete with Steel Liner)	3-29
3.10-6	Millstone 3 Subatmospheric Containment (Reinforced Concrete with Steel Liner)	3-30

LIST OF FIGURES
(Continued)

Figure		Page
3.10-7	Sequoyah Ice Condenser Containment (Steel Cylinder with Concrete Shield Building)	3-31
3.10-8	Ice Condenser Containment with Reinforced Concrete Structure and Steel Liner	3-32
3.10-9	General Arrangement of an Ice Condenser	3-33
3.10-10	Typical PWR Containment Post-LOCA Removal Paths	3-34
3.11-1	Examples of Component Cooling System Heat Transport Paths	3-35
3.12-1	Example of Actuation System Interfaces	3-36
3.13-1	Representative PWR 4160 VAC and 480 VAC Electric Power Distribution System (Callaway)	3-37
3.13-2	Representative PWR 125 VDC and 120 VAC Electric Power Distribution System (Callaway)	3-38
3.13-3	Diesel Fuel Oil and Air Starting Systems	3-39
3.13-4	Diesel Lubrication, Cooling, and Room Ventilation Systems	3-40
4.1-1	General Arrangement of a Westinghouse Reactor Vessel and Internals	4-5
4.1-2	Westinghouse U-Tube Steam Generator (Typical of Series 44 and 51 and Model F)	4-6
4.1-3	Typical Westinghouse Pressurizer	4-7
4.1-4	Typical Westinghouse RCS Temperature Control Program	4-8
4.1-5	General Arrangement of the Yankee-Rowe 9 x (6 x 6) Fuel Assembly	4-9
4.1-6	Westinghouse 14 x 14 Fuel Assembly and 16-Element Rod Cluster Control Assembly	4-10
4.1-7	Westinghouse 17 x 17 Fuel Assembly, Longitudinal View	4-11
4.1-8	Westinghouse 17 x 17 Fuel Assembly, Cross-Section View	4-12
4.1-9	Westinghouse Rod Cluster Control Assembly (RCCA, Typical for 17 x 17 Fuel Assembly)	4-13
4.1-10	Details of a Westinghouse Magnetic Jack Full-Length Control Rod Drive Mechanism	4-14

LIST OF FIGURES
(Continued)

Figure		Page
4.1-11	General Arrangement of Westinghouse CRDMs and RCCAs	4-15
4.1-12	Distribution of Containment Types for Westinghouse Reactors	4-16
4.2-1	Westinghouse 2-Loop NSSS	4-19
4.2-2	Section Views of the Ginna Large, Dry Containment	4-20
4.2-3	Plan View of the Ginna Large, Dry Containment Below the Elevation of the Operating Floor	4-22
4.2-4	Plan View of the Ginna Large, Dry Containment, Above the Operating Floor	4-23
4.3-1	Westinghouse 3-Loop NSSS	4-27
4.3-2	Section View of the H.B. Robinson Large, Dry Containment (1-D Post-Tensioned Concrete)	4-28
4.3-3	Plan View of the H.B. Robinson Large, Dry Containment (1-D Post-Tensioned Concrete)	4-29
4.3-4	Section Views of the Summer Large, Dry Containment (3-D Post-Tensioned Concr	4-30
4.3-5	Plan View of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete)	4-32
4.3-6	Section Views of the North Anna Subatmospheric Containment	4-33
4.3-7	Plan View of the North Anna Subatmospheric Containment	4-35
4.4-1	Westinghouse 4-Loop NSSS	4-39
4.4-2	General Arrangement of the Yankee-Rowe Core	4-40
4.4-3	General Arrangement of a 193 Fuel Assembly Core	4-41
4.4-4	Section View of the Yankee-Rowe Large, Dry Containment (Steel Sphere)	4-42
4.4-5	Plan View of the Yankee-Rowe Large, Dry Containment (Steel Sphere)	4-43
4.4-6	Section Views of the Diablo Canyon Large, Dry Containment (Reinforced Concrete)	4-44
4.4-7	Plan View of the Diablo Canyon Large, Dry Containment (Reinforced Concrete)	4-45

LIST OF FIGURES
(Continued)

Figure		Page
4.4-8	Section View of South Texas Large, Dry Containment (3-D Post-Tensioned Concrete)	4-46
4.4-9	Plan View of South Texas Large, Dry Containment (3-D Post-Tensioned Concrete)	4-47
4.4-10	Section View of the Millstone 3 Subatmospheric Containment	4-48
4.4-11	Section View of the Catawba Ice Condenser Containment	4-49
4.4-12	Plan View of the Catawba Ice Condenser Containment Lower Compartment	4-50
4.4-13	Section Views of the Watts Bar Ice Condenser Containment	4-51
4.4-14	Plan View of the Watts Bar Ice Condenser Containment Lower Compartment	4-53
4.4-15	Plan View of the Watts Bar Ice Condenser Containment Upper Compartment	4-54
5.1-1	Typical Combustion Engineering RCS Temperature Control Program	5-5
5.1-2	Combustion Engineering 15 x 15 Fuel Assembly and General Arrangement of the Palisades Core	5-6
5.1-3	Combustion Engineering 16 x 16 Fuel Assembly and General Arrangement of a 217 Fuel Assembly Core	5-7
5.1-4	Longitudinal View of a 16 x 16 Fuel Assembly	5-8
5.1-5	Typical Control Element Assembly (CEA) for a Pre-System 80 C-E PWR	5-9
5.1-6	Details of Combustion Engineering Magnetic Jack Control Element Drive Mechanism	5-10
5.1-7	Distribution of Containment Types for Combustion Engineering Reactors	5-11
5.2-1	Combustion Engineering 2-Loop (Pre-System 80) NSSS	5-15
5.2-2	Combustion Engineering 2-Loop (Pre-System 80) NSSS Side View	5-16
5.2-3	Combustion Engineering 2-Loop (Pre-System 80) NSSS Plan View	5-17
5.2-4	General Arrangement of the Palisades Reactor Vessel and Internals	5-18

LIST OF FIGURES
(Continued)

Figure		Page
5.2-5	General Arrangement of the San Onofre 2/3 Reactor Vessel and Internals	5-19
5.2-6	Combustion Engineering U-Tube Steam Generator (Pre-System 80)	5-20
5.2-7	Combustion Engineering Pressurizer (Pre-System 80)	5-21
5.2-8	Section View of the St. Lucie Large, Dry Containment (Steel Cylinder with Concrete Enclosure Building)	5-22
5.2-9	Plan View of the St. Lucie Large, Dry Containment (Steel Cylinder with Concrete Enclosure Building)	5-23
5.2-10	Section View of the San Onofre 2/3 Large, Dry Containment (3-D Post-Tensioned Concrete)	5-24
5.2-11	Plan View of the San Onofre 2/3 Large, Dry Containment (3-D Post-Tensioned Concrete)	5-25
5.2-12	Cutaway View of the San Onofre 2/3 Containment and Fuel Storage Building Showing Refueling Facilities	5-26
5.3-1	Combustion Engineering 2-Loop System 80 NSSS Side View	5-29
5.3-2	Combustion Engineering 2-Loop System 80 NSSS Plan View	5-30
5.3-3	General Arrangement of the System 80 241 Fuel Assembly Core	5-31
5.3-4	Combustion Engineering 8 and 12 Element System 80 CEAs	5-32
5.3-5	General Arrangement of the System 80 Reactor Vessel and Internals	5-33
5.3-6	Combustion Engineering System 80 U-Tube Steam Generator	5-34
5.3-7	Combustion Engineering System 80 Pressurizer	5-35
5.3-8	Section View of the Palo Verde Large, Dry Containment (3-D Post-Tensioned Concrete)	5-36
5.4-1	Maine Yankee Reactor Vessel	5-38
5.4-2	Maine Yankee U-Tube Steam Generator	5-39
5.4-3	Maine Yankee Pressurizer	5-40
5.4-4	Section Views of the Maine Yankee Large, Dry Containment (Reinforced Concrete)	5-41

LIST OF FIGURES
(Continued)

Figure		Page
5.4-5	Plan View of the Maine Yankee Large, Dry Containment at the Level of the Primary Loops	5-43
5.4-6	Plan View of the Main Yankee Large, Dry Containment at the Operating Floor Level	5-44
6.1-1	Typical Babcock & Wilcox RCS Temperature Control Program	6-4
6.1-2	General Arrangement of a B & W 177 Fuel Assembly Core	6-5
6.1-3	General Arrangement of a B & W 205 Fuel Assembly Core	6-6
6.1-4	B & W 15 x 15 (Mark B) Fuel Assembly	6-7
6.1-5	B & W 17 x 17 (Mark C) Fuel Assembly	6-8
6.1-6	B & W 16 Element CRA	6-9
6.1-7	B & W 24 Element CRA	6-10
6.1-8	B & W Roller-Nut Control Rod Drive Mechanism	6-11
6.1-9	B & W Core Flooding Injection Path for Core Flood Tanks and Low-Pressure Injection System	6-12
6.1-10	Distribution of Containment Types for Babcock & Wilcox Reactors	6-13
6.2-1	B & W 177 Lowered-Loop NSSS Side View	6-16
6.2-2	B & W 177 Lowered-Loop NSSS Plan View	6-17
6.2-3	General Arrangement of the B & W 177 Plant Reactor Vessel and Internals	6-18
6.2-4	Flow Paths in the B & W 177 Plant Once-Through Steam Generator	6-19
6.2-5	B & W 177 Plant Pressurizer	6-20
6.2-6	Section Views of the Three Mile Island 1 Large, Dry Containment (3-D Post-Tensioned Concrete)	6-21
6.2-7	Plan View of the Three Mile Island 1 Large, Dry Containment at the Level of the Primary Loops	6-23
6.2-8	Plan View of the Three Mile Island 1 Large, Dry Containment at the Level of the Operating Floor	6-24
6.3-1	B & W 177 Raised-Loop NSSS Side View	6-26

LIST OF FIGURES
(Continued)

Figure		Page
6.3-2	B & W 177 Raised-Loop NSSS Plan View	6-27
6.3-3	Section View of the Davis-Besse Large, Dry Containment (Steel Cylinder with Concrete Shield Building)	6-28
6.4-1	B & W 205 Raised Loop NSSS	6-30
6.4-2	B & W 205 Raised-Loop NSSS Showing Component Supports	6-31
6.4-3	B & W 205 Raised Loop NSSS Side View	6-32
6.4-4	B & W 205 Raised-Loop NSSS Plan View	6-33
6.4-5	General Arrangement of the B & W 205 Plant Reactor Vessel and Internals	6-34
6.4-6	B & W 205 Plant Once-Through Steam Generator	6-35
6.4-7	B & W 205 Plant Pressurizer	6-36
6.4-8	Section View of the Bellefonte Large, Dry Containment (3-D Post-Tensioned Concrete)	6-37
6.4-9	Plan View of the Bellefonte Large, Dry Containment (3-D Post-Tensioned Concrete)	6-38
8.2-1	BWR/2 Non-Jet Pump BWR Nuclear Boiler	8-22
8.2-2	Representative BWR/3 to BWR/6 Jet Pump Nuclear Boiler	8-23
8.2-3	Jet Pump Recirculation System Elevation and Isometric (Representative of BWR/5 and BWR/6 Plants)	8-24
8.2-4	Typical Jet Pump Installation	8-25
8.2-5	Jet Pump Details	8-26
8.2-6	Core Flooding Capability with Jet Pump Recirculation System	8-27
8.2-7	Representative BWR/2 to BWR/4 Recirculation Pump Variable Speed Control System	8-28
8.2-8	Representative BWR/5 and BWR/6 Recirculation Flow Control System	8-29
8.2-9	Steam Separator	8-30
8.2-10	Steam Dryer	8-31

LIST OF FIGURES
(Continued)

Figure		Page
8.2-11	Simplified Diagram of a Yarway Reactor Vessel Water Level Measuring Instrument	8-32
8.2-12	Example of BWR Vessel Water Level Instrument Ranges	8-33
8.3-1	Big Rock Point Fuel Assembly and Control Rod Geometry	8-34
8.3-2	Details of Big Rock Point (BWR/1) Fuel Assembly (Fuel Channel Removed)	8-35
8.3-3	General Arrangement of a 7 x 7 Fuel Assembly	8-36
8.3-4	8 x 8 Fuel Assembly and Control Rod Geometry	8-37
8.4-1	BWR Control Rod Assembly	8-38
8.4-2	BWR Hydraulic Control Rod Drive	8-39
8.4-3	Control Rod Drive Hydraulic Control Unit	8-40
8.4-4	Simplified Diagram of Portions of the Control Rod Drive Hydraulic System That are Related to the Scram Function	8-41
8.5-1	Normal BWR Heat Transport Paths During Power Operation	8-42
8.6-1	BWR/3 Isolation Condenser Heat Transport Paths	8-43
8.6-2	BWR/4 to BWR/6 Reactor Core Isolation Cooling System Heat Transport Paths	8-44
8.6-3	BWR/4 to BWR/6 Steam Condensing Reactor Core Isolation Cooling System Heat Transport Paths	8-45
8.7-1	BWR/4, BWR/5 and BWR/6 Heat Transport Paths Using Multi-Mode RHR System in the Shutdown Cooling Mode	8-46
8.8-1	Typical BWR Safety Valve	8-47
8.8-2	BWR/2 Power-Actuated Relief Valve (PARV)	8-48
8.8-3	BWR Three-Stage Pilot-Operated Safety/Relief Valve	8-49
8.8-4	BWR Crosby Safety/Relief Valve	8-50
8.9-1	BWR/3 to BWR/4 High Pressure Coolant Injection (HPCI) Coolant Injection and Heat Transport Paths	8-51

LIST OF FIGURES
(Continued)

Figure		Page
8.9-2	BWR/5 to BWR/6 High Pressure Coolant Injection (HPCI) Coolant Injection and Heat Transport Paths	8-52
8.9-3	BWR/2 to BWR/6 (Low Pressure) Core Spray System Coolant Injection Path	8-53
8.9-4	BWR Core Spray System Sparger Location Above the Reactor Core	8-54
8.10-1	Distribution of BWR Containments by Model	8-55
8.10-2	Distribution of BWR Containments by Construction Type	8-56
8.10-3	BWR Pressure Suppression Containments	8-57
8.10-4	Big Rock Point Large, Dry Containment (Steel Sphere)	8-58
8.10-5	Humboldt Bay Pre-Mark I Pressure Suppression Containment	8-59
8.10-6	General Arrangement of a BWR Mark I Containment	8-60
8.10-7	BWR Mark I Free-Standing Steel Primary Containment	8-61
8.10-8	Details of the Mark I Containment Drywell	8-62
8.10-9	Typical BWR Standby Gas Treatment System	8-63
8.10-10	General Arrangement of a BWR Mark II Containment	8-64
8.10-11	General Arrangement of a BWR Mark III Containment	8-65
8.10-12	Definition of Primary and Secondary Containment Boundaries in a Mark III Containment	8-66
8.11-1	Examples of Component Cooling System Heat Transport Paths	8-68
8.12-1	Example of Actuation System Interfaces	8-69
8.13-1	Representative BWR/4 4160, 480 and 120 VAC Electric Power Distribution System (Cooper)	8-70
8.13-2	Representative BWR/4 125 and 250 VDC Electric Power Distribution System (Cooper)	8-71
8.13-3	Representative BWR/6 4160 and 480 VAC Electric Power Distribution System (River Bend)	8-72
8.13-4	Representative BWR/6 125 VDC and 120 VAC Electric Power Distribution System (River Bend)	8-73

LIST OF FIGURES
(Continued)

Figure		Page
8.13-5	Diesel Fuel Oil and Air Starting Systems	8-74
8.13-6	Diesel Lubrication, Cooling and Room Ventilation Systems	8-75
9.1-1	Simplified Diagram of Big Rock Point Power Conversion System	9-5
9.1-2	Big Rock Point Reactor Vessel and Intervals	9-6
9.1-3	Cutaway View of the Big Rock Point Large, Dry Containment	9-7
9.1-4	Section View of the Big Rock Point Large, Dry Containment	9-8
9.1-5	[illegible]iew of the Big Rock Point Large, Dry Containment	9-9
9.2-1	Ty[illegible] BWR/2 Nuclear Boiler	9-12
9.2-2	Typical BWR/2 Vessel Water Level Instrument Ranges	9-13
9.2-3	General Arrangement of the Oyster Creek Mark I Containment	9-14
9.2-4	Cutaway View of the Nine Mile Point 1 Reactor Building	9-16
9.3-1	Typical BWR/3 Nuclear Boiler	9-20
9.3-2	Typical BWR/3 Vessel Water Level Instrument Ranges	9-21
9.4-1	Typical BWR/4 Nuclear Boiler	9-25
9.4-2	General Arrangement of the Fitzpatrick Mark I Containment	9-26
9.4-3	General Arrangement of the Limerick Mark II Containment	9-27
9.5-1	Typical BWR/5 Vessel Water Level Instrument Ranges	9-31
9.5-2	General Arrangement of the Nine Mile Point 2 Mark II Containment	9-32
9.5-3	Plan View of the Nine Mile Point 2 Mark II Containment at the Elevation of the Reactor Core	9-33
9.5-4	Plan View of the Nine Mile Point 2 Mark II Containment at the Level of the Suppression Pool	9-34
9.6-1	Typical BWR/6 Nuclear Boiler	9-37
9.6-2	Water Level Trip and Alarm Elevations Associated with a BWR/6	9-38
9.6-3	General Arrangement of the River Bend Mark III Containment	9-39

LIST OF FIGURES
(Continued)

Figure		Page
9.6-4	Plan View of the River Bend Mark III Containment at the Elevation of the Suppression Pool	9-40
9.6-5	General Arrangement of the Perry Mark III Containment	9-41
9.6-6	General Arrangement of the Grand Gulf Mark III Containment	9-42

LIST OF TABLES

Table		Page
1.2-1	Sourcebook Status as of 8/15/90, Ordered by Reactor Type and Vendor	1-5
1.2-2	Sourcebook Status as of 8/15/90, Ordered by Plant Name	1-6
1.3-1	Comparison of PWR Systems Included in the Sourcebooks	1-8
1.3-2	Comparison of BWR Systems Included in the Sourcebooks	1-11
2-1	General Plant Data - Sorted by Plant Name	2-2
2-2	General Plant Data - Sorted by Utility Name	2-8
2-3	General Plant Data - Sorted by NSSS Vendor and A-E	2-14
2-4	General Reactor Site Data	2-20
2-5	Summary of General Licensing Data - Sorted by Plant Name	2-25
2-6	Summary of General Licensing Data - Sorted by NRC Region	2-32
3.1-1	Summary of PWR Systems	3-11
4-1.1	General Characteristics of Westinghouse Steam Generators	4-2
4.5-1	Design Parameters for Representative Westinghouse PWRs	4-56
4.5-2	Comparison of Westinghouse PWR Vessel and Core Parameters	4-58
4.5-3	Westinghouse PWR System Comparison - RCS, AFW, Charging and HPSI	4-60
4.5-4	Comparison of Westinghouse PWR Pressurizer Relief Capacity	4-63
4.5-5	Comparison of Westinghouse PWR Containments	4-66
4.5-6	Comparison of Westinghouse PWR Backup Electric Power Systems	4-68
4.5-7	Comparison of Westinghouse PWR Power Conversion Systems	4-70
5.1-1	General Characteristics of C-E Steam Generators	5-2
5.5-1	Design Parameters for Representative Combustion Engineering PWRs	5-46
5.5-2	Comparison of Combustion Engineering PWR Vessel and Core Parameters	5-48
5.5-3	Combustion Engineering PWR System Comparison - RCS, AFW, Charging and HPSI	5-49

LIST OF TABLES
(Continued)

Table		Page
5.5-4	Comparison of Combustion Engineering PWR Pressurizer Relief Capacity	5-51
5.5-5	Comparison of Combustion Engineering PWR Containments	5-52
5.5-6	Comparison of Combustion Engineering PWR Backup Electric Power Systems	5-53
5.5-7	Comparison of Combustion Engineering PWR Power Conversion Systems	5-54
6.1-1	General Characteristics of Babcock and Wilcox Steam Generators	6-2
6.5-1	Design Parameters for Representative Babcock and Wilcox PWRs	6-40
6.5-2	Comparison of Babcock & Wilcox PWR Vessel and Core Parameters	6-42
6.5-3	Babcock & Wilcox PWR System Comparison - RCS, AFW, Charging and HPSI	6-43
6.5-4	Comparison of Babcock & Wilcox PWR Pressurizer Relief Capacity	6-45
6.5-5	Comparison of Babcock & Wilcox PWR Containments	6-46
6.5-6	Comparison of Babcock & Wilcox PWR Backup Electric Power Systems	6-47
6.5-7	Comparison of Babcock & Wilcox PWR Power Conversion Systems	6-48
7-1	Comparison of PWR Vessel and Core Parameters	7-2
7-2	PWR System Comparison - RCS, AFW, Charging and HPSI	7-5
7-3	Comparison of PWR Pressurizer Relief Capacity	7-9
7-4	Comparison of PWR Containments	7-13
7-5	Comparison of PWR Backup Electric Power Systems	7-16
7-6	Comparison of PWR Power Conversion Systems	7-20
8.1-1	Summary of BWR Systems	8-15
8.2-1	Summary of BWR Vessel Sizes	8-16
8.3-1	Comparison of BWR Core Parameters	8-17

LIST OF TABLES
(Continued)

Table		Page
8.10-1	Comparison of BWR Pressure Suppression Containment Design Parameters	8-18
8.13-1	Comparison of BWR AC Load Distribution	8-20
8.13-2	Comparison of BWR DC Load Distribution	8-21
9-1	General Electric BWR Comparison	9-3
10-1	Design Parameters for Representative General Electric BWRs	10-2
10-2	Comparison of General Electric BWR Vessel and Core Parameters	10-5
10-3	General Electric Boiling Water Reactor Plant System Comparison	10-7
10-4	Comparison of General Electric BWR Recirculation and Shutdown Cooling	10-9
10-5	Comparison of General Electric BWR Primary System Relief Capacity	10-11
10-6	Comparison of General Electric BWR Emergency Coolant Injection and Spray Systems	10-13
10-7	Comparison of General Electric BWR Containments	10-15
10-8	Comparison of General Electric BWR Backup Electric Power Systems	10-17
10-9	Comparison of General Electric BWR Power Conversion Systems	10-19

CAUTION

The information in this report has been developed over an extended period of time based on a site visit, the Final Safety Analysis Report, system and layout drawings, and other published information. To the best of our knowledge, it accurately reflects the plant configuration at the time the information was obtained, however, the information in this document has not been independently verified by the licensee or the NRC.

NOTICE

This sourcebook will be periodically updated with new and/or replacement pages as appropriate to incorporate additional information on this reactor plant. Technical errors in this report should be brought to the attention of the following:

Mr. Mark Rubin
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Systems Technology
Mail stop 7E2
Washington, D.C. 20555

Corrections and other recommended changes should be submitted in the form of marked up copies of the affected text, tables or figures. Supporting documentation should be included or referenced if possible.

1. INTRODUCTION

This introductory volume to the Nuclear Power Plant System Sourcebook series is intended as a source of current comparative information on U.S. commercial light water reactors (LWRs). As such, this volume should help the user to relate the more detailed information in an individual Nuclear Power Plant System Sourcebook to similar plants and to the U.S. commercial nuclear power industry at large.

A Nuclear Power Plant System Sourcebook is an individual report that provides summary information on the safety systems at a particular U.S. commercial nuclear power plant. The Sourcebook is intended to serve as an accurate, consistently formatted, integrated information resource that is more convenient to use than other information resources currently available to the U.S. Nuclear Regulatory Commission.

1.1 Organization of This Introductory Volume

This introductory volume for the Nuclear Power System Sourcebook series provides background information on the Sourcebooks and provides summary and comparative information on U.S. commercial LWRs. The summary and comparative information is organized into four parts as follows:

- **General Nuclear Industry Comparative Information**
  - Tabular industry-wide data comparisons of U.S. nuclear power plants (Section 2)
- **Pressurized Water Reactor (PWR) Comparative Information**
  - Overview of PWR systems (Section 3)
  - Descriptive comparisons of the product lines of U.S. commercial PWR vendors
    - Westinghouse PWRs (Section 4)
    - Combustion Engineering PWRs (Section 5)
    - Babcock & Wilcox PWRs (Section 6)
  - Tabular system data comparisons of all U.S. PWR plants (Section 7)
- **Boiling Water Reactor (BWR) Comparative Information**
  - Overview of BWR systems (Section 8)
  - Descriptive comparisons of the General Electric BWR product lines (Section 9)
  - Tabular system data comparisons of all U.S. BWR plants (Section 10)
- **Bibliographies**
  - General PWR and BWR references (Appendix A)
    Plant-specific references (Appendix B)
  - System-specific references (Appendix C)
  - Component specific references (Appendix D)

The overviews of PWR systems in Sections 3 and BWR systems in Section 8 focus on basic system functions and interfaces and include simple system schematic drawings to illustrate major system flow paths, with no attempt to accurately portray system redundancies. These sections are intended as high-level introductions to the principal types of systems found at U.S. commercial LWR plants, with comparisons to highlight significant differences among plants.

The descriptive comparisons in Sections 4, 5, 6, and 9 broadly define the product lines of the four U.S. commercial LWR vendors and identify significant

similarities and differences among the product lines. Comparative system data are compiled and presented in tabular form in Sections 4.5, 5.5 and 6.5 (Westinghouse, C-E, and B&W PWRs, respectively), and in Sections 7 (all PWRs) and 10 (all BWRs).

For more details on safety systems of specific nuclear power plants, the reader is referred to the available Nuclear Power Plant System Sourcebooks identified in Section 1.3. Additional sources of information are identified in the bibliographies in Appendices A to D of this introductory volume to the Sourcebook series.

1.2 Available Nuclear Power Plant System Sourcebooks

The Nuclear Power Plant System Sourcebooks that are currently available are identified in Tables 1.2-1 and 1.2-2. There are 120 LWR power plants identified in these tables, with a breakdown among vendors as follows:

- Westinghouse 55 PWRs
- Combustion Engineering 15 PWRs
- Babcock & Wilcox 10 PWRs
- General Electric 40 BWRs

As of August 1990, Sourcebooks are available on 72 LWR power plants (60 percent of all U.S. LWR power plants) with a breakdown among vendors as follows:

- Westinghouse 26 PWRs
- Combustion Engineering 9 PWRs
- Babcock & Wilcox 3 PWRs
- General Electric 34 BWRs

1.3 Organization of a Nuclear Power Plant System Sourcebook

Each Sourcebook is organized in five sections as follows:

- Section 1: Summary Data on the Plant
- Section 2: Identification of Similar Nuclear Power Plants
- Section 3: System Information
- Section 4: Plant Information
- Section 5: Plant-Specific Bibliography

Sections 1 and 2 are brief summaries that introduce Sections 3 and 4 where the bulk of the plant-specific information is presented. Section 3 begins with a tabular summary of the plant systems as identified in the Final or Updated Safety Analysis Report (FSAR or USAR) and an indication of which systems are covered in the Sourcebook. Also included is a high-level block diagram that indicates the major interfaces that exist among cooling water systems at the plant.

Section 3 continues with system summaries that describe the design and operating characteristics of major systems, focusing on those systems that have a role in mitigating the effects of transients and/or a loss-of-coolant-accident (LOCA). The Sourcebook system summaries typically cover the following systems:

- Reactor Coolant System (RCS)
- Emergency Core Cooling System (ECCS)
- Electric Power System
- Instrumentation and Control Systems
- Component Cooling Water System
- Service Water System
- Auxiliary Feedwater System (PWRs only)

- Reactor Core Isolation Cooling (RCIC) or Isolation Condenser System (BWRs only)
- Charging System (PWRs only)
- Control Rod Drive Hydraulic System (BWRs only)

In addition, some Sourcebooks also include system summaries for containment cooling systems and reactor shutdown cooling systems. The specific systems covered in the available Sourcebooks are shown in Tables 1.3-1 (PWRs) and 1.3-2 (BWRs).

An individual system summary in Section 3 of a Sourcebook includes the following information:

- **System Function:**
  A brief statement of the basic function(s) which the system is intended to perform.
- **System Definition:**
  A narrative overview of the major elements or components of the system, supplemented by system diagrams and one or more system data summary tables.
- **System Drawing(s):**
  Typically, two versions of the system drawings are presented; the first version is the basic simplified system drawing, and the second version indicates general component locations by means of shaded zones that are superimposed on the basic drawing. The principle flow paths through fluid systems also are highlighted in the second version of the system drawing.
- **System Data Summary Table:**
  For the major components in the system drawings, the data summary table lists the component identification number, component type, general location, and electric power source.
- **System Operation:**
  A brief description of how the system operates during normal plant operation and during accident conditions.
- **System Success Criteria:**
  A concise statement of the requirements for component or loop (train) success in order to accomplish the system function(s).
- **Component Information:**
  Listings of basic design or operating parameters of major components in the system.
- **Support Systems and Interfaces:**
  A brief description of support system interfaces related to system operation. Interfaces with the following types of support systems typically are described: (a) actuation and control, (b) motive power sources, (c) component cooling, (d) equipment room cooling, and (e) other support systems as appropriate.

The plant information presented in Section 4 of a Sourcebook includes a brief description of the site and major buildings, a simplified site drawing, building elevation and simplified layout drawings, and summary data tables. The simplified site and layout drawings are intended to provide a general orientation to the plant. These are not architectural scale drawings, and no attempt has been made to show all of the minor rooms and hallways that may exist in the plant. In general, component and equipment locations are not shown in the simplified layout drawings. Location codes are assigned and used consistently in the simplified layout drawings, system drawings, and data tables. This should enable Sourcebook users to develop a general understanding of the physical arrangement in the plant of the major portions of each system.

The bibliography included as Section 5 of a Sourcebook lists other sources of information on the plant in question. These Sourcebook bibliographies have been included in the compilation of plant-specific references in Appendix B of this introductory volume to the Sourcebook series.

Each Sourcebook includes two appendices that define the standard symbols used in the Sourcebook system and layout drawings and the terms used in the various data tables appearing in the Sourcebook. The standard symbols used in the system drawings are shown in Figures 1.3-1 and 1.3-2.

The information presented in a Sourcebook is compiled from a variety of sources, including the following:

- The licensee's Final or Updated Safety Analysis Report (FSAR or USAR).
- System piping and instrumentation drawings (P&IDs), one-line electrical drawings, and site and building layout drawings.
- Relevant NUREG and NUREG/CR reports.
- Other available information sources.

While this introductory volume and the individual Sourcebooks are believed to accurately reflect the plant configurations at the time the information was obtained, it should be noted that these reports have not been independently verified by the NRC or the licensee. Please refer to the notice at the front of this volume for instructions on reporting any technical errors or recommending other changes in this report to the NRC.

Table 1.2-1. Sourcebook Status as of 8/15/90, Ordered by Reactor Type and Vendor

PWR Type	Reactor Plant Name	Sourcebook
B&W (L-loop)	ANO-1	None Planned
B&W (L-loop)	Crystal River 3	None Planned
B&W (L-loop)	Oconee 1, 2 & 3	None Planned
B&W (L-loop)	Rancho Seco	SAIC-89/1027
B&W (L-loop)	Three Mile Island 1	SAIC-89/1526
B&W (R-loop)	Bellefonte 1 & 2	New 1990
B&W (R-loop)	Davis-Besse	SAIC-88/1021
C-E (3-loop)	Maine Yankee	SAIC-88/1022
C-E (2-loop)	ANO-2	None Planned
C-E (2-loop)	Calvert Cliffs 1 & 2	None Planned
C-E (2-loop)	Fort Calhoun 1	SAIC-88/1982
C-E (2-loop)	Millstone 2	SAIC-89/1018
C-E (2-loop)	Palisades	None Planned
C-E (2-loop)	Palo Verde 1, 2 & 3	SAIC-88/1017
C-E (2-loop)	San Onofre 2 & 3	None Planned
C-E (2-loop)	St. Lucie 1 & 2	SAIC-89/1527
C-E (2-loop)	Waterford 3	SAIC-88/1019
W (2-loop)	Ginna	SAIC-89/1539
W (2-loop)	Kewaunee	None Planned
W (2-loop)	Point Beach 1 & 2	None Planned
W (2-loop)	Prairie Island 1 & 2	None Planned
W (3-loop)	Beaver Valley 1	SAIC-89/1529
W (3-loop)	Beaver Valley 2	SAIC-90/1428
W (3-loop)	Farley 1 & 2	None Planned
W (3-loop)	H.B. Robinson 2	SAIC-89/1026
W (3-loop)	North Anna 1 & 2	None Planned
W (3-loop)	San Onofre 1	None Planned
W (3-loop)	Shearon Harris 1	SAIC-89/1536
W (3-loop)	Summer	None Planned
W (3-loop)	Surry 1 & 2	None Planned
W (3-loop)	Turkey Point 3 & 4	None Planned
W (4-loop)	Braidwood 1 & 2	SAIC-89/1016
W (4-loop)	Byron 1 & 2	SAIC-89/1015
W (4-loop)	Callaway	SAIC-88/1997
W (4-loop)	Catawba 1 & 2	SAIC-88/1995
W (4-loop)	Comanche Peak 1 &	New 1990
W (4-loop)	D. C. Cook 1 & 2	None Planned
W (4-loop)	Diablo Canyon 1 & 2	None Planned
W (4-loop)	Haddam Neck	New 1990
W (4-loop)	Indian Point 2	None Planned
W (4-loop)	Indian Point 3	SAIC-89/1037
W (4-loop)	McGuire 1 & 2	SAIC-89/1528
W (4-loop)	Millstone 3	SAIC-89/1019
W (4-loop)	Salem 1 & 2	None Planned
W (4-loop)	Seabrook 1	SAIC-90/1429
W (4-loop)	Sequoyah 1 & 2	None Planned
W (4-loop)	South Texas 1 & 2	SAIC-89/1032
W (4-loop)	Trojan	SAIC-88/1023
W (4-loop)	Vogtle 1 & 2	SAIC-89/1540
W (4-loop)	Watts Bar 1 & 2	None Planned
W (4-loop)	Wolf Creek	SAIC-88/1996
W (4-loop)	Yankee Rowe	SAIC-88/1993
W (4-loop)	Zion 1 & 2	SAIC-89/1532

BWR Type	Reactor Plant Name	Sourcebook
BWR/1	Big Rock Point	SAIC-89/1028
BWR/2	Nine Mile Point 1	SAIC-89/1534
BWR/2	Oyster Creek 1	SAIC-89/1530
BWR/3	Dresden 2 & 3	SAIC-89/1533
BWR/3	Millstone 1	SAIC-89/1017
BWR/3	Monticello	SAIC-89/1025
BWR/3	Pilgrim 1	None Planned
BWR/3	Quad Cities 1 & 2	SAIC-89/1537
BWR/4	Browns Ferry 1	SAIC-89/1029
BWR/4	Browns Ferry 2	SAIC-89/1030
BWR/4	Browns Ferry 3	SAIC-89/1031
BWR/4	Brunswick 1 & 2	SAIC-89/1011
BWR/4	Cooper	SAIC-88/1994
BWR/4	Duane Arnold	None Planned
BWR/4	Fermi 2	SAIC-89/1036
BWR/4	Fitzpatrick	SAIC-89/1024
BWR/4	Hatch 1 & 2	SAIC-89/1034
BWR/4	Hope Creek 1 & 2	SAIC-89/1033
BWR/4	Limerick 1 & 2	SAIC-89/1023
BWR/4	Peach Bottom 2 & 3	SAIC-89/1020
BWR/4	Shoreham	None Planned
BWR/4	Susquehanna 1 & 2	SAIC-89/1035
BWR/4	Vermont Yankee	None Planned
BWR/5	LaSalle 1 & 2	None Planned
BWR/5	Nine Mile Point 2	SAIC-89/1538
BWR/5	WNP-2	SAIC-89/1022
BWR/6	Clinton 1	SAIC-89/1021
BWR/6	Grand Gulf 1	SAIC-89/1007
BWR/6	Perry 1 & 2	SAIC-89/1535
BWR/6	River Bend 1	SAIC-89/1531

Table 1.2-2. Sourcebook Status as of 8/15/90, Ordered by Plant Name

Reactor Plant Name	Reactor Type	Reactor Vendor and Vintage	Sourcebook
ANO-1	PWR	B&W (L-loop)	**None Planned**
ANO-2	PWR	C-E (2-loop)	**None Planned**
Beaver Valley 1	PWR	W (3-loop)	SAIC-89/1529
Beaver Valley 2	PWR	W (3-loop)	SAIC-90/1428
Bellefonte 1 & 2	PWR	B&W (R-loop)	New 1990
Big Rock Point	BWR	BWR/1	SAIC-89/1028
Braidwood 1 & 2	PWR	W (4-loop)	SAIC-89/1016
Browns Ferry 1	BWR	BWR/4	SAIC-89/1029
Browns Ferry 2	BWR	BWR/4	SAIC-89/1030
Browns Ferry 3	BWR	BWR/4	SAIC-89/1031
Brunswick 1 & 2	BWR	BWR/4	SAIC-89/1011
Byron 1 & 2	PWR	W (4-loop)	SAIC-89/1015
Callaway	PWR	W (4-loop)	SAIC-88/1997
Calvert Cliffs 1 & 2	PWR	C-E (2-loop)	**None Planned**
Catawba 1 & 2	PWR	W (4-loop)	SAIC-88/1995
Clinton 1	BWR	BWR/6	SAIC-89/1021
Comanche Peak 1 & 2	PWR	W (4-loop)	New 1990
Cooper	BWR	BWR/4	SAIC-88/1994
Crystal River 3	PWR	B&W (L-loop)	**None Planned**
D. C. Cook 1 & 2	PWR	W (4-loop)	**None Planned**
Davis-Besse	PWR	B&W (R-loop)	SAIC-88/1021
Diablo Canyon 1 & 2	PWR	W (4-loop)	**None Planned**
Dresden 2 & 3	BWR	BWR/3	SAIC-89/1533
Duane Arnold	BWR	BWR/4	**None Planned**
Farley 1 & 2	PWR	W (3-loop)	**None Planned**
Fermi 2	BWR	BWR/4	SAIC-89/1036
Fitzpatrick	BWR	BWR/4	SAIC-89/1024
Fort Calhoun 1	PWR	C-E (2-loop)	SAIC-88/1982
Ginna	PWR	W (2-loop)	SAIC-89/1539
Grand Gulf 1	BWR	BWR/6	SAIC-89/1007
H.B. Robinson 2	PWR	W (3-loop)	SAIC-89/1026
Haddam Neck	PWR	W (4-loop)	New 1990
Hatch 1 & 2	BWR	BWR/4	SAIC-89/1034
Hope Creek 1 & 2	BWR	BWR/4	SAIC-89/1033
Indian Point 2	PWR	W (4-loop)	**None Planned**
Indian Point 3	PWR	W (4-loop)	SAIC-89/1037
Kewaunee	PWR	W (2-loop)	**None Planned**
LaSalle 1 & 2	BWR	BWR/5	**None Planned**
Limerick 1 & 2	BWR	BWR/4	SAIC-89/1023
Maine Yankee	PWR	C-E (3-loop)	SAIC-88/1022
McGuire 1 & 2	PWR	W (4-loop)	SAIC-89/1528
Millstone 1	BWR	BWR/3	SAIC-89/1017
Millstone 2	PWR	C-E (2-loop)	SAIC-89/1018
Millstone 3	PWR	W (4-loop)	SAIC-89/1019
Monticello	BWR	BWR/3	SAIC-89/1025
Nine Mile Point 1	BWR	BWR/2	SAIC-89/1534
Nine Mile Point 2	BWR	BWR/5	SAIC-89/1538
North Anna 1 & 2	PWR	W (3-loop)	**None Planned**
Oconee 1, 2 & 3	PWR	B&W (L-loop)	**None Planned**
Oyster Creek 1	BWR	BWR/2	SAIC-89/1530
Palisades	PWR	C-E (2-loop)	**None Planned**
Palo Verde 1, 2 & 3	PWR	C-E (2-loop)	SAIC-88/1017

Table 1.2-2. Sourcebook Status as of 8/15/90, Ordered by Plant Name (Continued)

Reactor Plant Name	Reactor Type	Reactor Vendor and Vintage	Sourcebook
Peach Bottom 2 & 3	BWR	BWR/4	SAIC-89/1020
Perry 1 & 2	BWR	BWR/6	SAIC-89/1535
Pilgrim 1	BWR	BWR/3	None Planned
Point Beach 1 & 2	PWR	W (2-loop)	None Planned
Prairie Island 1 & 2	PWR	W (2-loop)	None Planned
Quad Cities 1 & 2	BWR	BWR/3	SAIC-89/1537
Rancho Seco	PWR	B&W (L-loop)	SAIC-89/1027
River Bend 1	BWR	BWR/6	SAIC-89/1531
Salem 1 & 2	PWR	W (4-loop)	None Planned
San Onofre 1	PWR	W (3-loop)	None Planned
San Onofre 2 & 3	PWR	C-E (2-loop)	None Planned
Seabrook 1	PWR	W (4-loop)	SAIC-90/1429
Sequoyah 1 & 2	PWR	W (4-loop)	None Planned
Shearon Harris 1	PWR	W (3-loop)	SAIC-89/1536
Shoreham	BWR	BWR/4	None Planned
South Texas 1 & 2	PWR	W (4-loop)	SAIC-89/1032
St. Lucie 1 & 2	PWR	C-E (2-loop)	SAIC-89/1527
Summer	PWR	W (3-loop)	None Planned
Surry 1 & 2	PWR	W (3-loop)	None Planned
Susquehanna 1 & 2	BWR	BWR/4	SAIC-89/1035
Three Mile Island 1	PWR	B&W (L-loop)	SAIC-89/1526
Trojan	PWR	W (4-loop)	SAIC-88/1023
Turkey Point 3 & 4	PWR	W (3-loop)	None Planned
Vermont Yankee	BWR	BWR/4	None Planned
Vogtle 1 & 2	PWR	W (4-loop)	SAIC-89/1540
Waterford 3	PWR	C-E (2-loop)	SAIC-88/1019
Watts Bar 1 & 2	PWR	W (4-loop)	None Planned
WNP-2	BWR	BWR/5	SAIC-89/1022
Wolf Creek	PWR	W (4-loop)	SAIC-88/1996
Yankee Rowe	PWR	W (4-loop)	SAIC-88/1993
Zion 1 & 2	PWR	W (4-loop)	SAIC-89/1532

Table 1.3-1. Comparison of PWR Systems Included in the Sourcebooks

	B & W PWRs				C - E PWRs						W 2 loop	Westinghouse 3-Loop PWRs			
Generic System Name	Rancho Seco	Three Mile Island 1	Davis-Besse	Bellefonte 1 & 2	Maine Yankee	Fort Calhoun 1	St. Lucie 1 & 2	Millstone 2	Waterford 3	Palo Verde 1, 3, and 3	Ginna	Beaver Valley 1	Beaver Valley 2	Robinson	Shearon Harris
Reactor Coolant System				Note 6											
Auxiliary Feedwater System & Secondary Steam Relief				Note 6											
ECCS (HPI)				Note 6											
ECCS (LPI)				Note 6											
Containment Spray System				Note 6	Note 3	Note 3	Note 2	Note 2	Note 2	Note 2					
Containment Fan Coolers or Other Containment Cooling Sys.				Note 6											
Charging System		Note 4		Note 6	Note 4							Note 4			
Shutdown Cooling System	Note 1	Note 1	Note 1	Note 6	Note 3	Note 3	Note 2	Note 2	Note 2	Note 2	Note 1			Note 1	Note 1
Reactor Protection System				Note 6											
Safety Feature Actuation System				Note 6											
Remote Shutdown Features				Note 6											
Class 1E Power System				Note 6											
Safety-related Component Cooling Water System	NCW	NSCC DHCC	CCW	Note 6	PCCW SCCW	ACCW	CCW	RB CCW	CCW	ECWS	CCW	Note 5	CCW	CCS	CCW
Emergency Service Water System	NSRW	ERW NSRW DHRW	SWS	Note 6	SWS	RW	ICW	SWS	ACCW	ESPS	SWS	RWS	SWS	SWS	SWS
Fire Water System															
Emergency Ventilation System															

Table 1.3-1. Comparison of PWR Systems Included in the Sourcebooks (continued)

	Westinghouse 4-Loop PWRs															
Generic System Name	Braidwood 1 & 2	Byron 1 & 2	Callaway	Catawba 1 & 2	Comanche Peak 1 & 2	Haddam Neck	Indian Point 3	McGuire 1 & 2	Millstone 3	Seabrook 1	South Texas 1 & 2	Trojan	Vogtle 1 & 2	Wolf Creek	Yankee-Rowe	Zion 1 & 2
Reactor Coolant System					Note 6	Note 6										
Auxiliary Feedwater System & Secondary Steam Relief					Note 6	Note 6										
ECCS (HPI)					Note 6	Note 6										
ECCS (LPI)					Note 6	Note 6										
Containment Spray System					Note 6	Note 6										
Containment Fan Coolers or Other Containment Cooling Sys					Note 6	Note 6										
Charging System					Note 6	Note 6						Note 4			Note 4	
Shutdown Cooling System	Note 1	Note 1	Note 1	Note 1	Note 6	Note 6	Note 1	Note 1	Note 1	Note 1		Note 1	Note 1	Note 1		Note 1
Reactor Protection System					Note 6	Note 6										
Safety Feature Actuation System					Note 6	Note 6										
Remote Shutdown Features					Note 6	Note 6										
Class 1E Power System					Note 6	Note 6										
Safety-related Component Cooling Water System	CCS	CCS	CCW	CCW	Note 5	Note 6	ACCW	CCW		PCCW	CCW	SCW	CCW	CCW		CCW
Emergency Service Water System	ESW	ESW	ESW	NSW	Note 6	Note 6	ESWS	NSW	ESWS	SWS	ECW	SWS HPSW	NSCW	ESW		SWS
Fire Water System																
Emergency Ventilation System																

Table 1.3-1. Comparison of PWR Systems Included in the Sourcebooks (continued)

Symbols:

= System is addressed in the Sourcebook

XXX = System is addressed in the Sourcebook. More than one system may perform a similar function at the plant in question, so the abbreviation for the plant-specific system name is indicated (i.e., xxx)

N/A = System is not applicable to the plant in question.

Note x = See notes, below, regarding how the system is treated in the Sourcebook.

= System is not covered in the Sourcebook.

Notes:

1. A multi-mode RHR system performs ECCS and normal shutdown (decay) heat removal functions. The RHR system is described primarily in its ECCS role.

2. The LPI, containment spray, and shutdown cooling systems are interrelated and share the same decay heat removal heat exchanger. These systems are described primarily in the ECCS role.

3. The LPI, containment spray, and shutdown cooling systems are interrelated and share the same decay heat removal heat exchanger. These systems are described in the ECCS and CS roles.

4. The charging system is described only as part of the ECCS.

5. Many safety-related components are cooled by the Emergency Service Water System. A separate component cooling water system is not described.

6. This system will be included in the Sourcebook. The Haddam Neck and Bellefonte Sourcebooks are expected to be completed in late 1990, and the Comanche Peak Sourcebook is expected to be completed in early 1991.

Table 1.3-2. Comparison of BWR Systems Included in the Sourcebooks

	BWR 1	BWR/2		BWR/3				BWR/4								
Generic System Name	Big Rock Point	Nine Mile Point 1	Oyster Creek	Dresden 2 & 3	Millstone 1	Monticello	Quad Cities 1 & 2	Browns Ferry 1	Browns Ferry 2	Browns Ferry 3	Brunswick 1 & 2	Cooper	Fermi 2	Fitzpatrick	Hatch 1 & 2	Hope Creek 1 & 2
Reactor Coolant System																
Isolation Condenser System						N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
RCIC System	N/A	N/A	N/A	N/A	N/A											
Feedwater Coolant Injection	N/A		N/A	N/A		N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
ECCS (HPCI / HPCS)	N/A	N/A	N/A		N/A											
ECCS (Core Spray / LPCS)																
ECCS (LPCI)	N/A	N/A	N/A													
ECCS (ADS)																
Shutdown Cooling System						Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1
Control Rod Drive Hydraulic System																
Suppression Pool Cooling/ Containment Spray System				Note 2	Note 2	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1
Reactor Protection System																
Safety Feature Actuation System																
Remote Shutdown Features																
Class 1E Power System																
Safety-related Component Cooling Water System	RCW				SCCW	Note 3	Note 3	Note 3	Note 3	Note 3	Note 3	RB CCW	EECW	Note 3	Note 3	SACS
Emergency Service Water System	SWS	CSRW	ESWS	DGCW CCSW	ESWS	ESW	DGCW	EECW	EECW	EECW	SWS	SWS	EESW	ESWS	PSW	SSWS
RHR (High-Pressure) Service Water System	RCW					RHR SW	RHR SW	RHR SW	RHR SW	RHR SW	RHR SW	RHR SW	RHR SW	RHR SW	RHR SW	SACS
Fire Water System																
Reactor Water Cleanup System																
Emergency Ventilation System																
Independent Safe Shutdown System	N/A	N/A	N/A	N/A	N/A	N/A		N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table 1.3-2. Comparison of BWR Systems Included in the Sourcebooks (continued)

	BWR/4 (Continued)			BWR/5		BWR/6			
Generic System Name	Limerick 1 & 2	Peach Bottom 2 & 3	Susquehanna 1 & 2	Nine Mile Point 2	WNP-2	Clinton 1	Grand Gulf 1	Perry 1 & 2	River Bend 1
Reactor Coolant System									
Isolation Condenser System	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
RCIC System									
Feedwater Coolant Injection	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
ECCS (HPCI / HPCS)									
ECCS (Core Spray / LPCS)									
ECCS (LPCI)									
ECCS (ADS)									
Shutdown Cooling System	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1
Control Rod Drive Hydraulic System									
Suppression Pool Cooling/ Containment Spray System	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1	Note 1
Reactor Protection System									
Safety Feature Actuation System									
Remote Shutdown Features									
Class 1E Power System									
Safety-related Component Cooling Water System	Note 3	Note 3	Note 3		Note 3	Note 3	Note 3	ECCW	Note 3
Emergency Service Water System	ESW	ESW	ESW	SWS	SSW	SSW	SSW	ESW	SSW
RHR (High-Pressure) Service Water System	RHR SW	HPSW	RHR SW	SWS	SSW	SSW	SSW	ESW	SSW
Fire Water System									
Reactor Water Cleanup System									
Emergency Ventilation System									
Independent Safe Shutdown System	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table 1.3-2. Comparison of BWR Systems Included in the Sourcebooks (continued)

Symbols:

= System is addressed in the Sourcebook

xxx = System is addressed in the Sourcebook. More than one system may perform a similar function at the plant in question, so the abbreviation for the plant-specific system name is indicated (i.e., xxx)

N/A = System is not applicable to the plant in question.

Note x = See notes, below, regarding how the system is treated in the Sourcebook.

= System is not covered in the Sourcebook.

Notes:

1. A multi-mode RHR system performs ECCS, containment cooling, and normal shutdown (decay) heat removal functions. The RHR system is described primarily in its ECCS role.

2. A multi-mode LPCI system performs ECCS and containment cooling functions. This system is described primarily in its ECCS role.

3. Many safety-related components are cooled by the Emergency Service Water System. A separate component cooling water system is not described.

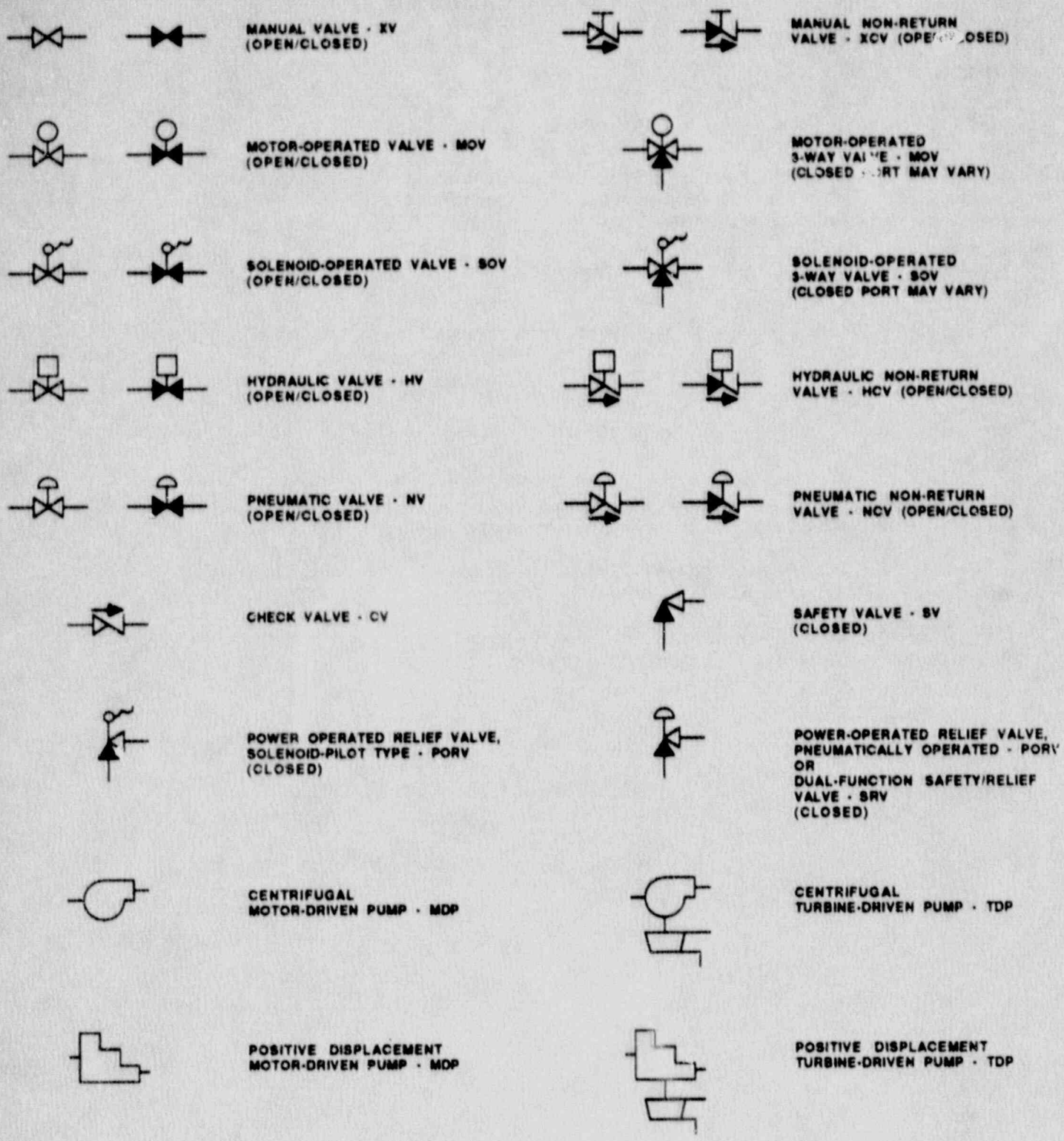
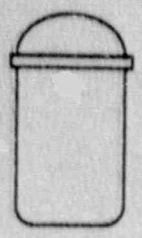
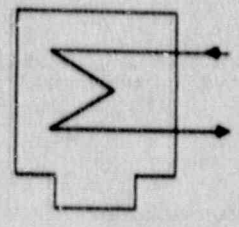


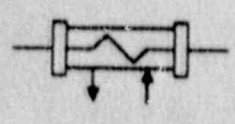
Figure 1.3-1. Key To Symbols In Fluid System Drawings



PWR/BWR
REACTOR VESSEL - RV



MAIN CONDENSER - COND

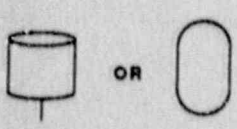


HEAT EXCHANGER - HX

MECHANICAL DRAFT
COOLING TOWER

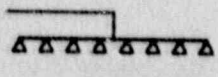
STEAM-TO-WATER
OR WATER-TO-STEAM HEAT
EXCHANGER (I.E. FEEDWATER
HEATER, DRAIN COOLER, ETC.) - HX

AIR COOLING UNIT - ACU



OR

TANK - TK



SPRAY NOZZLES - SN

RUPTURE DISK - RD

FILTER - FLT

ORIFICE - OR

Figure 1.3-1. Key To Symbols In Fluid System Drawings (Continued)

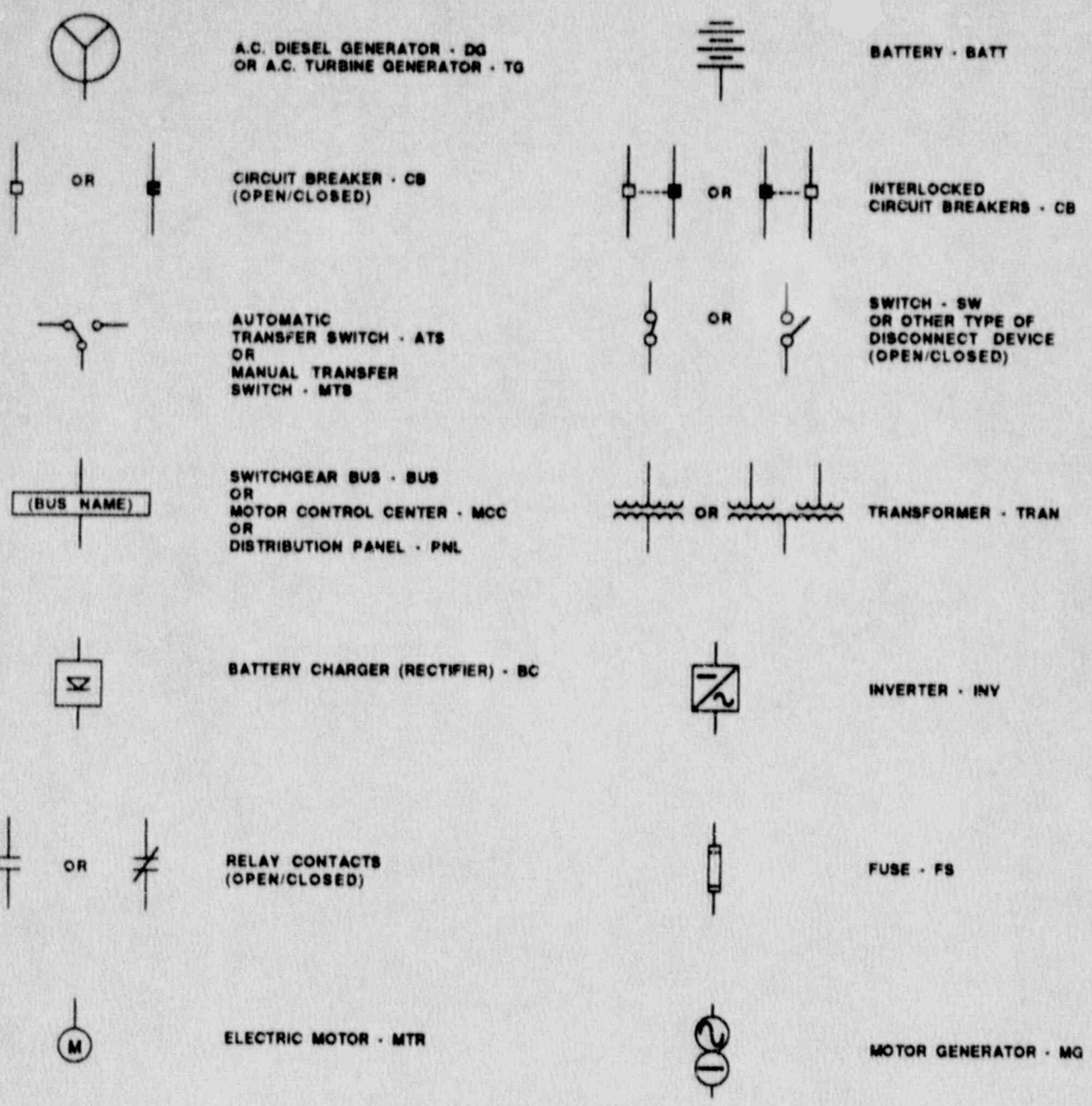


Figure 1.3-2. Key To Symbols In Electrical System Drawings

2. GENERAL COMPARATIVE DATA FOR U.S. COMMERCIAL NUCLEAR POWER PLANTS

This section contains the following tables which present general comparative data for U.S. commercial nuclear power plants:

- Table 2-1 General Plant Data - Sorted by Plant Name
- Table 2-2 General Plant Data - Sorted by Utility Name
- Table 2-3 General Plant Data - Sorted by NSSS Vendor and A-E
- Table 2-4 General Reactor Site Data
- Table 2-5 Summary of General Licensing Data - Sorted by Plant Name
- Table 2-6 Summary of General Licensing Data - Sorted by NRC Region

Table 2-1. General Plant Data - Sorted by Plant Name

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 & Roe	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	820	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 [illegible]	NY	New York Power Authority	PWR	W	UE & C	3025	965	MDC
Kewaunee	Carlton	WI	Wisconsin Public Service Corp.	PWR	W	Pioneer	1650	5[illegible]3	MDC
LaSalle 1	Seneca	IL	Commonwealth Edison Co.	BWR	GE	Sargent & Lundy	3323	103[illegible]	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	8[illegible]0	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	[illegible]
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	[illegible]
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 2-2. General Plant Data - Sorted by Utility Name

Utility	Reactor Plant	City	State	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Alabama Power Co.	Farley 1	Dothan	AL	PWR	W	Bechtel	2652	813	MDC
Alabama Power Co.	Farley 2	Dothan	AL	PWR	W	Bechtel	2652	813	MDC
Arizona Public Service Co.	Palo Verde 1	Wintersburg	AZ	PWR	C-E	Bechtel	3800	1221	MDC
Arizona Public Service Co.	Palo Verde 2	Wintersburg	AZ	PWR	C-E	Bechtel	3800	1221	MDC
Arizona Public Service Co.	Palo Verde 3	Wintersburg	AZ	PWR	C-E	Bechtel	3800	1221	MDC
Arkansas Power & Light Co.	ANO-1	Russellville	AR	PWR	B&W	Bechtel	2568	836	MDC
Arkansas Power & Light Co.	ANO-2	Russellville	AR	PWR	C-E	Bechtel	2815	858	MDC
Baltimore Gas & Electric Co.	Calvert Cliffs 1	Lusby	MD	PWR	C-E	Bechtel	2700	825	MDC
Baltimore Gas & Electric Co.	Calvert Cliffs 2	Lusby	MD	PWR	C-E	Bechtel	2700	825	MDC
Boston Edison Co.	Pilgrim 1	Plymouth	MA	BWR	GE	Bechtel	1998	670	MDC
Carolina Power & Light Co.	Brunswick 1	Southport	NC	BWR	GE	UE&C	2436	790	MDC
Carolina Power & Light Co.	Brunswick 2	Southport	NC	BWR	GE	UE&C	2436	790	MDC
Carolina Power & Light Co.	Robinson 2	Hartsville	SC	PWR	W	Ebasco	2300	665	MDC
Carolina Power & Light Co.	Shearon Harris 1	New Hill	NC	PWR	W	Ebasco	2775	860	MDC
Cleveland Electric Illuminating Co.	Perry 2	North Perry	OH	BWR	GE	Gilbert	3579	1205	DER
Commonwealth Edison Co.	Braidwood 1	Braidwood	IL	PWR	W	Sargent & Lundy	3411	1120	MDC
Commonwealth Edison Co.	Braidwood 2	Braidwood	IL	PWR	W	Sargent & Lundy	3411	1120	MDC
Commonwealth Edison Co.	Byron 1	Byron	IL	PWR	W	Sargent & Lundy	3411	1105	MDC
Commonwealth Edison Co.	Byron 2	Byron	IL	PWR	W	Sargent & Lundy	3411	1105	MDC
Commonwealth Edison Co.	Dresden 2	Morris	IL	BWR	GE	Sargent & Lundy	2527	772	MDC
Commonwealth Edison Co.	Dresden 3	Morris	IL	BWR	GE	Sargent & Lundy	2527	773	MDC
Commonwealth Edison Co.	LaSalle 1	Seneca	IL	BWR	GE	Sargent & Lundy	3323	1036	MDC
Commonwealth Edison Co.	LaSalle 2	Seneca	IL	BWR	GE	Sargent & Lundy	3323	1036	MDC

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

Utility	Reactor Plant	City	State	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Commonwealth Edison Co.	Zion 1	Zion	IL	PWR	W	Sargent & Lundy	3250	1040	MDC
Commonwealth Edison Co.	Zion 2	Zion	IL	PWR	W	Sargent & Lundy	3250	1040	MDC
Commonwealth Edison/Iowa-Illinois Gas & Electric	Quad Cities 1	Cordova	IL	BWR	GE	Sargent & Lundy	2511	769	MDC
Commonwealth Edison/Iowa-Illinois Gas & Electric	Quad Cities 2	Cordova	IL	BWR	GE	Sargent & Lundy	2511	769	MDC
Connecticut Yankee Atomic Power Co.	Haddam Neck	Haddam Neck	CT	PWR	W	Stone & Webster	1825	569	MDC
Consolidated Edison Co.	Indian Point 2	Indian Point	NY	PWR	W	UE&C	2758	849	MDC
Consumers Power Co.	Big Rock Point	Charlevoix	MI	BWR	GE	Bechtel	240	69	MDC
Consumers Power Co.	Palisades	South Haven	MI	PWR	C-E	Bechtel	2530	730	MDC
Detroit Edison Co.	Fermi 2	Newport	MI	BWR	GE	Detroit Edison	3292	1093	MDC
Duke Power Co.	Catawba 1	Clover	SC	PWR	W	Duke Power Co.	3411	1129	MDC
Duke Power Co.	Catawba 2	Clover	SC	PWR	W	Duke Power Co.	3411	1129	MDC
Duke Power Co.	McGuire 1	Cornelius	NC	PWR	W	Duke Power Co.	3411	1129	MDC
Duke Power Co.	McGuire 2	Cornelius	NC	PWR	W	Duke Power Co.	3411	1129	MDC
Duke Power Co.	Oconee 1	Seneca	SC	PWR	B&W	Duke/ Bechtel	2568	846	MDC
Duke Power Co.	Oconee 2	Seneca	SC	PWR	B&W	Duke/ Bechtel	2568	846	MDC
Duke Power Co.	Oconee 3	Seneca	SC	PWR	B&W	Duke/ Bechtel	2568	846	MDC
Duquesne Light Co.	Beaver Valley 1	Shippingport	PA	PWR	W	Stone & Webster	2652	810	MDC
Duquesne Light Co.	Beaver Valley 2	Shippingport	PA	PWR	W	Stone & Webster	2652	833	MDC
Florida Power & Light Co.	St. Lucie 1	Hutchinson Island	FL	PWR	C-E	Ebasco	2700	839	MDC
Florida Power & Light Co.	St. Lucie 2	Hutchinson Island	FL	PWR	C-E	Ebasco	2700	839	MDC
Florida Power & Light Co.	Turkey Point 3	Florida City	FL	PWR	W	Bechtel	2200	666	MDC
Florida Power & Light Co.	Turkey Point 4	Florida City	FL	PWR	W	Bechtel	2200	666	MDC
Florida Power Corp.	Crystal River 3	Red Level	FL	PWR	B&W	Gilbert	2544	821	MDC

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

Utility	Reactor Plant	City	State	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Georgia Power Co.	Hatch 1	Baxley	GA	BWR	GE	SCS / Bechtel	2436	756	MDC
Georgia Power Co.	Hatch 2	Baxley	GA	BWR	GE	SCS / Bechtel	2436	768	MDC
Georgia Power Co.	Vogtle 1	Waynesboro	GA	PWR	W	Bechtel	3411	1079	MDC
Georgia Power Co.	Vogtle 2	Waynesboro	GA	PWR	W	Bechtel	3411	1079	MDC
GPU Nuclear Corp.	Oyster Creek 1	Forked River	NJ	BWR	GE	Burns & Roe/GE	1930	620	MDC
GPU Nuclear Corp.	Three Mile Island 1	Londonderry Twp	PA	PWR	B&W	Gilbert	2535	776	MDC
Gulf States Utilities Co.	River Bend 1	St. Francisville	LA	BWR	GE	Stone & Webster	2894	936	MDC
Houston Lighting & Power Co.	South Texas 1	Palacios	TX	PWR	W	Brown & Root	3900	1250	MDC
Houston Lighting & Power Co.	South Texas 2	Palacios	TX	PWR	W	Brown & Root	3800	1250	DER
Illinois Power Co.	Clinton 1	Clinton	IL	BWR	GE	Sargent & Lundy	2894	930	DER
Indiana/Michigan Power Co.	D.C. Cook 1	Bridgman	MI	PWR	W	AEP	3250	1020	MDC
Indiana/Michigan Power Co.	D.C. Cook 2	Bridgman	MI	PWR	W	AEP	3411	1060	MDC
Iowa Electric Light & Power Co.	Duane Arnold	Palo	IA	BWR	GE	Bechtel	1658	515	MDC
Long Island Lighting Co.	Shoreham	Brookhaven	NY	BWR	GE	Stone & Webster	2436	820	DER
Louisiana Power & Light Co.	Waterford 3	Taft	LA	PWR	C-E	Ebasco	3390	1075	MDC
Maine Yankee Atomic Power Co.	Maine Yankee	Wiscasset	ME	PWR	C-E	Stone & Webster	2630	810	MDC
Nebraska Public Power District	Cooper	Brownville	NE	BWR	GE	Burns & Roe	2381	764	MDC
New Hampshire Yankee	Seabrook 1	Seabrook	NH	PWR	W	UE & C	3411	1150	MDC
New York Power Authority	Fitzpatrick	Scriba	NY	BWR	GE	Stone & Webster	2436	778	MDC
New York Power Authority	Indian Point 3	Indian Point	NY	PWR	W	UE & C	3025	965	MDC
Niagara Mowhawk Power Co.	Nine Mile Point 1	Scriba	NY	BWR	GE	Niagara Mohawk	1850	610	MDC
Niagara Mowhawk Power Co.	Nine Mile Point 2	Scriba	NY	BWR	GE	Stone & Webster	3323	1080	MDC
Northeast Utilities	Millstone 1	Waterford	CT	PWR	GE	Ebasco	2011	654	MDC

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

Utility	Reactor Plant	City	State	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Northeast Utilities	Millstone 2	Waterford	CT	PWR	C-E	Bechtel	2700	863	MDC
Northeast Utilities	Millstone 3	Waterford	CT	PWR	W	Stone & Webster	3411	1142	MDC
Northern States Power Co.	Monticello	Monticello	MN	BWR	GE	Bechtel	1670	536	MDC
Northern States Power Co.	Prairie Island 1	Red Wing	MN	PWR	W	Pioneer	1650	503	MDC
Northern States Power Co.	Prairie Island 2	Red Wing	MN	PWR	W	Pioneer	1650	503	MDC
Omaha Public Power District	Fort Calhoun 1	Fort Calhoun	NE	PWR	C-E	Gibbs & Hill	1500	478	MDC
Pacific Gas & Electric Co.	Diablo Canyon 1	Avila Beach	CA	PWR	W	Pacific Gas & Electric	3338	1073	MDC
Pacific Gas & Electric Co.	Diablo Canyon 2	Avila Beach	CA	PWR	W	Pacific Gas & Electric	3411	1087	MDC
Pennsylvania Power & Light Co.	Susquehanna 1	Berwick	PA	BWR	GE	Bechtel	3293	1032	MDC
Pennsylvania Power & Light Co.	Susquehanna 2	Berwick	PA	BWR	GE	Bechtel	3293	1032	MDC
Philadelphia Power & Light Co.	Limerick 1	Pottstown	PA	BWR	GE	Bechtel	3293	1055	MDC
Philadelphia Power & Light Co.	Limerick 2	Pottstown	PA	BWR	GE	Bechtel	3293	1065	DER
Philadelphia Power & Light Co.	Peach Bottom 2	Peach Bottom	PA	BWR	GE	Bechtel	3293	1051	MDC
Philadelphia Power & Light Co.	Peach Bottom 3	Peach Bottom	PA	BWR	GE	Bechtel	3293	1035	MDC
Portland General Electric Co.	Trojan	Prescott	OR	PWR	W	Bechtel	3411	1095	MDC
Public Services Company of Colorado	Fort St. Vrain	Platteville	CO	HTGR	GA	Sargent & Lundy	842	330	MDC
Public Services Electric & Gas Co.	Hope Creek 1	Salem	NJ	BWR	GE	Bechtel	3293	1067	MDC
Public Services Electric & Gas Co.	Salem 1	Salem	NJ	PWR	W	Pacific Gas & Electric	3411	1106	MDC
Public Services Electric & Gas Co.	Salem 2	Salem	NJ	PWR	W	Pacific Gas & Electric	3411	1106	MDC
Rochester Gas & Electric Corp.	Ginna	Ontario	NY	PWR	W	Gilbert	1520	470	MDC
Sacramento Municipal Utility District	Rancho Seco	Clay Station	CA	PWR	B&W	Bechtel	2772	873	MDC
South Carolina Electric & Gas Co.	Summer	Parr	SC	PWR	W	Gilbert	2775	885	MDC
Southern California Edison/San Diego Gas & Electric	San Onofre 1	San Clemente	CA	PWR	W	Bechtel	1347	436	MDC

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

Utility	Reactor Plant	City	State	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Southern California Edison/San Diego Gas & Electric	San Onofre 2	San Clemente	CA	PWR	C-E	Bechtel	3390	1070	MDC
Southern California Edison/San Diego Gas & Electric	San Onofre 3	San Clemente	CA	PWR	C-E	Bechtel	3390	1080	MDC
System Energy Resources, Inc.	Grand Gulf 1	Port Gibson	MS	BWR	GE	Bechtel	3833	1142	MDC
System Energy Resources, Inc.	Grand Gulf 2	Port Gibson	MS	BWR	GE	Bechtel	3833	1250	DER
Tennessee Valley Authority	Bellefonte 1	Scottsboro	AL	PWR	B&W	TVA	3413	1213	DER
Tennessee Valley Authority	Bellefonte 2	Scottsboro	AL	PWR	B&W	TVA	3413	1213	DER
Tennessee Valley Authority	Browns Ferry 1	Decatur	AL	BWR	GE	TVA	3293	1065	MDC
Tennessee Valley Authority	Browns Ferry 2	Decatur	AL	BWR	GE	TVA	3293	1065	MDC
Tennessee Valley Authority	Browns Ferry 3	Decatur	AL	BWR	GE	TVA	3293	1065	MDC
Tennessee Valley Authority	Sequoyah 1	Soddy-Daisy	TN	PWR	W	TVA	3411	1148	MDC
Tennessee Valley Authority	Sequoyah 2	Soddy-Daisy	TN	PWR	W	TVA	3411	1148	MDC
Tennessee Valley Authority	Watts Bar 1	Spring City	TN	PWR	W	TVA	3411	1165	DER
Tennessee Valley Authority	Watts Bar 2	Spring City	TN	PWR	W	TVA	3411	1165	DER
Texas Utilities Electric Co.	Comanche Peak 1	Glen Rose	TX	PWR	W	Gibbs & Hill	3425	1150	DER
Texas Utilities Electric Co.	Comanche Peak 2	Glen Rose	TX	PWR	W	Gibbs & Hill	3425	1150	DER
The Cleveland Electric Illuminating Co.	Perry 1	North Perry	OH	BWR	GE	Gilbert	3579	1205	MDC
Toledo Edison Co.	Davis-Besse	Oak Harbor	OH	PWR	B&W	Bechtel	2772	860	MDC
Union Electric Co.	Callaway	Fulton	MO	PWR	W	Bechtel	3565	1145	MDC
Vermont Yankee Nuclear Power Corp.	Vermont Yankee	Vernon	VT	BWR	GE	Ebasco	1593	504	MDC
Virginia Power Co.	North Anna 1	Mineral	VA	PWR	W	Stone & Webster	2893	915	MDC
Virginia Power Co.	North Anna 2	Mineral	VA	PWR	W	Stone & Webster	2893	915	MDC
Virginia Power Co.	Surry 1	Gravel Neck	VA	PWR	W	Stone & Webster	2441	781	MDC
Virginia Power Co.	Surry 2	Gravel Neck	VA	PWR	W	Stone & Webster	2441	781	MDC

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

Utility	Reactor Plant	City	State	Reactor Type	NSSS Vendor	Architect/ Engineer	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER
Washington Public Power Supply System	WNP-1	Richland	WA	PWR	B&W	UE&C	3760	1266	DER
Washington Public Power Supply System	WNP-2	Richland	WA	BWR	GE	Burns & Roe	3323	1095	MDC
Washington Public Power Supply System	WNP-3	Satsop	WA	PWR	C-E	Ebasco	3800	1242	DER
Wisconsin Electric Power Co.	Point Beach 1	Two Creeks	WI	PWR	W	Bechtel	1518	485	MDC
Wisconsin Electric Power Co.	Point Beach 2	Two Creeks	WI	PWR	W	Bechtel	1518	485	MDC
Wisconsin Public Service Corp.	Kewaunee	Carlton	WI	PWR	W	Pioneer	1650	503	MDC
Wolf Creek Nuclear Operating Corp.	Wolf Creek	Burlington	KS	PWR	W	Bechtel	3411	1128	MDC
Yankee Atomic Electric Co.	Yankee Rowe	Rowe	MA	PWR	W	Stone & Webster	600	167	MDC

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

Table 2-3. General Plant Data - Sorted by NSSS Vendor and A-E

Reactor Type	NSSS Vendor	Architect/ Engineer	Reactor Plant	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER	City	State	Utility
PWR	B&W	Bechtel	ANO-1	2568	836	MDC	Russellville	AR	Arkansas Power & Light Co.
PWR	B&W	Bechtel	Davis-Besse	2772	860	MDC	Oak Harbor	OH	Toledo Edison Co.
PWR	B&W	Bechtel	Rancho Seco	2772	873	MDC	Clay Station	CA	Sacramento Municipal Utility District
PWR	B&W	Duke/ Bechtel	Oconee 1	2568	846	MDC	Seneca	SC	Duke Power Co.
PWR	B&W	Duke/ Bechtel	Oconee 2	2568	846	MDC	Seneca	SC	Duke Power Co.
PWR	B&W	Duke/ Bechtel	Oconee 3	2568	846	MDC	Seneca	SC	Duke Power Co.
PWR	B&W	Gilbert	Crystal River 3	2544	821	MDC	Red Level	FL	Florida Power Corp.
PWR	B&W	Gilbert	Three Mile Island 1	2535	776	MDC	Londonderry Twp.	PA	GPU Nuclear Corp.
PWR	B&W	TVA	Bellefonte 1	3413	1213	DER	Scottsboro	AL	Tennessee Valley Authority
PWR	B&W	TVA	Bellefonte 2	3413	1213	DER	Scottsboro	AL	Tennessee Valley Authority
PWR	B&W	UE&C	WNP-1	3760	1266	DER	Richland	WA	Washington Public Power Supply System
PWR	C-E	Bechtel	ANO-2	2815	858	MDC	Russellville	AR	Arkansas Power & Light Co.
PWR	C-E	Bechtel	Calvert Cliffs 1	2700	825	MDC	Lusby	MD	Baltimore Gas & Electric Co.
PWR	C-E	Bechtel	Calvert Cliffs 2	2700	825	MDC	Lusby	MD	Baltimore Gas & Electric Co.
PWR	C-E	Bechtel	Millstone 2	2700	863	MDC	Waterford	CT	Northeast Utilities
PWR	C-E	Bechtel	Palisades	2530	730	MDC	South Haven	MI	Consumers Power Co.
PWR	C-E	Bechtel	Palo Verde 1	3800	1221	MDC	Wintersburg	AZ	Arizona Public Service Co.
PWR	C-E	Bechtel	Palo Verde 2	3800	1221	MDC	Wintersburg	AZ	Arizona Public Service Co.
PWR	C-E	Bechtel	Palo Verde 3	3800	1221	MDC	Wintersburg	AZ	Arizona Public Service Co.
PWR	C-E	Bechtel	San Onofre 2	3390	1070	MDC	San Clemente	CA	Southern California Edison/San Diego Gas & Electric
PWR	C-E	Bechtel	San Onofre 3	3390	1080	MDC	San Clemente	CA	Southern California Edison/San Diego Gas & Electric
PWR	C-E	Ebasco	St. Lucie 1	2700	839	MDC	Hutchinson Island	FL	Florida Power & Light Co.
PWR	C-E	Ebasco	St. Lucie 2	2700	839	MDC	Hutchinson Island	FL	Florida Power & Light Co.

Table 2-3. General Plant Data - Sorted by NSSS Vendor and A-E (Continued)

Reactor Type	NSSS Vendor	Architect/ Engineer	Reactor Plant	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER	City	State	Utility
PWR	C-E	Ebasco	Waterford 3	3390	1075	MDC	Taft	LA	Louisiana Power & Light Co.
PWR	C-E	Ebasco	WNP-3	3800	1242	DER	Satsop	WA	Washington Public Power Supply System
PWR	C-E	Gibbs & Hill	Fort Calhoun 1	1500	478	MDC	Fort Calhoun	NE	Omaha Public Power District
PWR	C-E	Stone & Webster	Maine Yankee	2630	810	MDC	Wiscasset	ME	Maine Yankee Atomic Power Co.
HTGR	GA	Sargent & Lundy	Fort St. Vrain	842	330	MDC	Platteville	CO	Public Services Company of Colorado
BWR	GE	Bechtel	Big Rock Point	240	69	MDC	Charlevoix	MI	Consumers Power Co.
BWR	GE	Bechtel	Duane Arnold	1658	515	MDC	Palo	IA	Iowa Electric Light & Power Co.
BWR	GE	Bechtel	Grand Gulf 1	3833	1142	MDC	Port Gibson	MS	System Energy Resources, Inc.
BWR	GE	Bechtel	Grand Gulf 2	3833	1250	DER	Port Gibson	MS	System Energy Resources, Inc.
BWR	GE	Bechtel	Hope Creek 1	3293	1067	MDC	Salem	NJ	Public Services Electric & Gas Co.
BWR	GE	Bechtel	Limerick 1	3293	1055	MDC	Pottstown	PA	Philadelphia Power & Light Co.
BWR	GE	Bechtel	Limerick 2	3293	1065	DER	Pottstown	PA	Philadelphia Power & Light Co.
BWR	GE	Bechtel	Monticello	1670	536	MDC	Monticello	MN	Northern States Power Co.
BWR	GE	Bechtel	Peach Bottom 2	3293	1051	MDC	Peach Bottom	PA	Philadelphia Power & Light Co.
BWR	GE	Bechtel	Peach Bottom 3	3293	1035	MDC	Peach Bottom	PA	Philadelphia Power & Light Co.
BWR	GE	Bechtel	Pilgrim 1	1998	670	MDC	Plymouth	MA	Boston Edison Co.
BWR	GE	Bechtel	Susquehanna 1	3293	1032	MDC	Berwick	PA	Pennsylvania Power & Light Co.
BWR	GE	Bechtel	Susquehanna 2	3293	1032	MDC	Berwick	PA	Pennsylvania Power & Light Co.
BWR	GE	Burns & Roe	Cooper	2381	764	MDC	Brownville	NE	Nebraska Public Power District
BWR	GE	Burns & Roe / GE	Oyster Creek 1	1930	620	MDC	Forked River	NJ	GPU Nuclear Corp.
BWR	GE	Burns & Roe	WNP-2	3323	1095	MDC	Richland	WA	Washington Public Power Supply System
BWR	GE	Detroit Edison	Fermi 2	3292	1093	MDC	Newport	MI	Detroit Edison Co.
BWR	GE	Ebasco	Millstone 1	2011	654	MDC	Waterford	CT	Northeast Utilities

Table 2-3. General Plant Data - Sorted by NSSS Vendor and A-E (Continued)

Reactor Type	NSSS Vendor	Architect/ Engineer	Reactor Plant	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER	City	State	Utility
BWR	GE	Ebasco	Vermont Yankee	1593	504	MDC	Vernon	VT	Vermont Yankee Nuclear Power Corp.
BWR	GE	Gilbert	Perry 1	3579	1205	MDC	North Perry	OH	The Cleveland Electric Illuminating Co.
BWR	GE	Gilbert	Perry 2	3579	1205	DER	North Perry	OH	Cleveland Electric Illuminating Co.
BWR	GE	Niagara Mohawk	Nine Mile Point 1	1850	610	MDC	Scriba	NY	Niagara Mohawk Power Co.
BWR	GE	Sargent & Lundy	Clinton 1	2894	930	DER	Clinton	IL	Illinois Power Co.
BWR	GE	Sargent & Lundy	Dresden 2	2527	772	MDC	Morris	IL	Commonwealth Edison Co.
BWR	GE	Sargent & Lundy	Dresden 3	2527	773	MDC	Morris	IL	Commonwealth Edison Co.
BWR	GE	Sargent & Lundy	LaSalle 1	3323	1036	MDC	Seneca	IL	Commonwealth Edison Co.
BWR	GE	Sargent & Lundy	LaSalle 2	3323	1036	MDC	Seneca	IL	Commonwealth Edison Co.
BWR	GE	Sargent & Lundy	Quad Cities 1	2511	769	MDC	Cordova	IL	Commonwealth Edison/Iowa-Illinois Gas & Electric
BWR	GE	Sargent & Lundy	Quad Cities 2	2511	769	MDC	Cordova	IL	Commonwealth Edison/Iowa-Illinois Gas & Electric
BWR	GE	Stone & Webster	Fitzpatrick	2436	778	MDC	Scriba	NY	New York Power Authority
BWR	GE	Stone & Webster	Nine Mile Point 2	3323	1080	MDC	Scriba	NY	Niagara Mohawk Power Co.
BWR	GE	Stone & Webster	River Bend 1	2894	936	MDC	St. Francisville	LA	Gulf States Utilities Co.
BWR	GE	Stone & Webster	Shoreham	2436	820	DER	Brookhaven	NY	Long Island Lighting Co.
BWR	GE	SCS / Bechtel	Hatch 1	2436	756	MDC	Baxley	GA	Georgia Power Co.
BWR	GE	SCS / Bechtel	Hatch 2	2436	768	MDC	Baxley	GA	Georgia Power Co.
BWR	GE	TVA	Browns Ferry 1	3293	1065	MDC	Decatur	AL	Tennessee Valley Authority
BWR	GE	TVA	Browns Ferry 2	3293	1065	MDC	Decatur	AL	Tennessee Valley Authority
BWR	GE	TVA	Browns Ferry 3	3293	1065	MDC	Decatur	AL	Tennessee Valley Authority
BWR	GE	UE & C	Brunswick 1	2436	790	MDC	Southport	NC	Carolina Power & Light Co.
BWR	GE	UE & C	Brunswick 2	2436	790	MDC	Southport	NC	Carolina Power & Light Co.
PWR	W	AEP	D.C. Cook 1	3250	1020	MDC	Bridgman	MI	Indiana/Michigan Power Co.

Table 2-3. General Plant Data - Sorted by NSSS Vendor and A-E (Continued)

Reactor Type	NSSS Vendor	Architect/ Engineer	Reactor Plant	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER	City	State	Utility
PWR	W	AEP	D.C. Cook 2	3411	1060	MDC	Bridgman	MI	Indiana/Michigan Power Co.
PWR	W	Bechtel	Callaway	3565	1145	MDC	Fulton	MO	Union Electric Co.
PWR	W	Bechtel	Farley 1	2652	813	MDC	Dothan	AL	Alabama Power Co.
PWR	W	Bechtel	Farley 2	2652	823	MDC	Dothan	AL	Alabama Power Co.
PWR	W	Bechtel	Point Beach 1	1518	485	MDC	Two Creeks	WI	Wisconsin Electric Power Co.
PWR	W	Bechtel	Point Beach 2	1518	485	MDC	Two Creeks	WI	Wisconsin Electric Power Co.
PWR	W	Bechtel	San Onofre 1	1347	433	MDC	San Clemente	CA	Southern California Edison/San Diego Gas & Electric
PWR	W	Bechtel	Trojan	3411	1095	MDC	Prescott	OR	Portland General Electric Co.
PWR	W	Bechtel	Turkey Point 3	2200	666	MDC	Florida City	FL	Florida Power & Light Co.
PWR	W	Bechtel	Turkey Point 4	2200	666	MDC	Florida City	FL	Florida Power & Light Co.
PWR	W	Bechtel	Vogtle 1	3411	1079	MDC	Waynesboro	GA	Georgia Power Co.
PWR	W	Bechtel	Vogtle 2	3411	1079	MDC	Waynesboro	GA	Georgia Power Co.
PWR	W	Bechtel	Wolf Creek	3411	1128	MDC	Burlington	KS	Wolf Creek Nuclear Operating Corp.
PWR	W	Brown & Root	South Texas 1	3800	1250	MDC	Palacios	TX	Houston Lighting & Power Co.
PWR	W	Brown & Root	South Texas 2	3800	1250	DER	Palacios	TX	Houston Lighting & Power Co.
PWR	W	Duke Power Co.	Catawba 1	3411	1129	MDC	Clover	SC	Duke Power Co.
PWR	W	Duke Power Co.	Catawba 2	3411	1129	MDC	Clover	SC	Duke Power Co.
PWR	W	Duke Power Co.	McGuire 1	3411	1129	MDC	Cornelius	NC	Duke Power Co.
PWR	W	Duke Power Co.	McGuire 2	3411	1129	MDC	Cornelius	NC	Duke Power Co.
PWR	W	Ebasco	Robinson 2	2300	665	MDC	Hartsville	SC	Carolina Power & Light Co.
PWR	W	Ebasco	Shearon Harris 1	2775	860	MDC	New Hill	NC	Carolina Power & Light Co.
PWR	W	Gibbs & Hill	Comanche Peak 1	3425	1150	DER	Glen Rose	TX	Texas Utilities Electric Co.
PWR	W	Gibbs & Hill	Comanche Peak 2	3425	1150	DER	Glen Rose	TX	Texas Utilities Electric Co.

Table 2-3. General Plant Data - Sorted by NSSS Vendor and A-E (Continued)

Reactor Type	NSSS Vendor	Architect/ Engineer	Reactor Plant	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER	City	State	Utility
PWR	W	Gilbert	Ginna	1520	470	MDC	Ontario	NY	Rochester Gas & Electric Corp.
PWR	W	Gilbert	Summer	2775	885	MDC	Parr	SC	South Carolina Electric & Gas Co.
PWR	W	Pacific Gas & Electric	Diablo Canyon 1	3338	1073	MDC	Avila Beach	CA	Pacific Gas & Electric Co.
PWR	W	Pacific Gas & Electric	Diablo Canyon 2	3411	1087	MDC	Avila Beach	CA	Pacific Gas & Electric Co.
PWR	W	Pacific Gas & Electric	Salem 1	3411	1106	MDC	Salem	NJ	Public Services Electric & Gas Co.
PWR	W	Pacific Gas & Electric	Salem 2	3411	1106	MDC	Salem	NJ	Public Services Electric & Gas Co.
PWR	W	Pioneer	Kewaunee	1650	503	MDC	Carlton	WI	Wisconsin Public Service Corp.
PWR	W	Pioneer	Prairie Island 1	1650	503	MDC	Red Wing	MN	Northern States Power Co.
PWR	W	Pioneer	Prairie Island 2	1650	503	MDC	Red Wing	MN	Northern States Power Co.
PWR	W	Sargent & Lundy	Braidwood 1	3411	1120	MDC	Braidwood	IL	Commonwealth Edison Co.
PWR	W	Sargent & Lundy	Braidwood 2	3411	1120	MDC	Braidwood	IL	Commonwealth Edison Co.
PWR	W	Sargent & Lundy	Byron 1	3411	1105	MDC	Byron	IL	Commonwealth Edison Co.
PWR	W	Sargent & Lundy	Byron 2	3411	1105	MDC	Byron	IL	Commonwealth Edison Co.
PWR	W	Sargent & Lundy	Zion 1	3250	1040	MDC	Zion	IL	Commonwealth Edison Co.
PWR	W	Sargent & Lundy	Zion 2	3250	1040	MDC	Zion	IL	Commonwealth Edison Co.
PWR	W	Stone & Webster	Beaver Valley 1	2652	810	MDC	Chippingport	PA	Duquesne Light Co.
PWR	W	Stone & Webster	Beaver Valley 2	2652	833	MDC	Shippingport	PA	Duquesne Light Co.
PWR	W	Stone & Webster	Haddam Neck	1825	569	MDC	Haddam Neck	CT	Connecticut Yankee Atomic Power Co.
PWR	W	Stone & Webster	Millstone 3	3411	1142	MDC	Waterford	CT	Northeast Utilities
PWR	W	Stone & Webster	North Anna 1	2893	915	MDC	Mineral	VA	Virginia Power Co.
PWR	W	Stone & Webster	North Anna 2	2893	915	MDC	Mineral	VA	Virginia Power Co.
PWR	W	Stone & Webster	Surry 1	2441	781	MDC	Gravel Neck	VA	Virginia Power Co.
PWR	W	Stone & Webster	Surry 2	2441	781	MDC	Gravel Neck	VA	Virginia Power Co.

Table 2-3. Gen. al Plant Data - Sorted by NSSS Vendor and A-E (Continued)

Reactor Type	NSSS Vendor	Architect/ Engineer	Reactor Plant	Core Power MWt	Net Electrical Output MWe	MWe Rating MDC or DER	City	State	Utility
PWR	W	Stone & Webster	Yankee Rowe	600	167	MDC	Rowe	MA	Yankee Atomic Electric Co.
PWR	W	TVA	Sequoyah 1	3411	1148	MDC	Soddy-Daisy	TN	Tennessee Valley Authority
PWR	W	TVA	Sequoyah 2	3411	1148	MDC	Soddy-Daisy	TN	Tennessee Valley Authority
PWR	W	TVA	Watts Bar 1	3411	1165	DER	Spring City	TN	Tennessee Valley Authority
PWR	W	TVA	Watts Bar 2	3411	1165	DER	Spring City	TN	Tennessee Valley Authority
PWR	W	UE&C	Indian Point 2	2758	849	MDC	Indian Point	NY	Consolidated Edison Co.
PWR	W	UE&C	Indian Point 3	3025	965	MDC	Indian Point	NY	New York Power Authority
PWR	W	UE&C	Seabrook 1	3411	1150	MDC	Seabrook	NH	New Hampshire Yankee

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
				Horiz. G's	Vert. G's	Speed (MPH)
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		Tornado Wind Speed (MPH)
				Horiz. G's	Vert. G's	
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
Prairie 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 Baton 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	360
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	30 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 Docket # (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	7/31/69	8/18/76	3293	7/21/06	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 Docket # (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 Docket # (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	1/9/90	
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 Docket # (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	
336	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 Docket # (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	10/11/06	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 Docket # (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	2/9/11	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 Docket # (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

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Docket # (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
I	Beaver Valley 1	PWR	W	334	6/26/70	7/2/76	2652	1/29/16	Yes	10/1/76	
I	Beaver Valley 2	PWR	W	412	5/3/74	8/14/87	2652	5/27/27	Yes	11/17/87	
I	Calvert Cliffs 1	PWR	C-E	317	7/7/69	7/31/74	2700	7/31/14	Yes	5/8/75	
I	Calvert Cliffs 2	PWR	C-E	318	7/7/69	11/30/76	2700	8/31/16	Yes	4/1/77	
I	Fitzpatrick	BWR	GE	333	5/20/70	10/17/74	2436	5/20/10	Yes	7/28/75	
I	Ginna	PWR	W	244	4/25/66	12/10/84	1520	4/25/06	Yes	7/1/70	
I	Haddam Neck	PWR	W	213	5/26/64	12/27/74	1825	5/26/04	Yes	1/1/68	
I	Hope Creek 1	BWR	GE	354	11/4/74	7/25/86	3293	4/11/25	Yes	12/20/86	
I	Indian Point 2	PWR	W	247	10/14/66	9/28/73	2758	9/28/13	Yes	8/1/74	
I	Indian Point 3	PWR	W	286	8/13/69	4/5/76	3025	8/13/09	Yes	8/30/76	
I	Limerick 1	BWR	GE	352	6/19/74	8/8/85	3293	10/26/24	Yes	2/1/86	
I	Limerick 2	BWR	GE	353	6/19/74				Yes	1/8/90	
I	Maine Yankee	PWR	C-E	309	10/21/68	6/29/73	2630	10/21/08	Yes	12/28/72	
I	Millstone 1	BWR	GE	245	5/19/66	10/31/86	2011	5/1[illegible]06	Yes	3/1/71	
I	Millstone 2	PWR	C-E	326	12/11/70	9/30/75	2700	7/31/15	Yes	12/26/75	
I	Millstone 3	PWR	W	423	8/9/74	1/31/86	3411	11/25/25	Yes	4/23/86	
I	Nine Mile Point 1	BWR	GE	220	4/12/65	12/26/74	1850	4/11/05	Yes	12/1/69	
I	Nine Mile Point 2	BWR	GE	410	6/24/74	7/2/87	3323	10/31/26	Yes	4/5/88	
I	Oyster Creek 1	BWR	GE	219	12/15/64	8/1/6[illegible]	1[illegible]	4/9/72	Yes	12/1/69	4

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region (Continued)

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Docket # (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
I	Peach Bottom 2	BWR	GE	277	1/31/68	7/2/74	3293	1/31/08	Yes	12/23/74	
I	Peach Bottom 3	BWR	GE	278	1/31/68	7/2/74	3293	1/31/08	Yes	12/23/74	
I	Pilgrim 1	BWR	GE	293	8/26/68	9/15/72	1998	8/26/08	Yes	12/1/72	
I	Salem 1	PWR	W	272	9/25/68	12/ /76	3411	9/25/08	Yes	6/30/77	
I	Salem 2	PWR	W	311	9/25/68	5/ 0/81	3411	9/25/08	Yes	10/13/81	
I	Seabrook 1	PWR	W	443	7/ /76	10/17/86	3411	10/17/26	No	N/A	5
I	Shoreham	BWR	GE	322	4/ /73	7/3/85	121	4/13/13	No	N/A	5
I	Susquehanna 1	BWR	GE	387	11/2/73	11/12/82	3293	7/17/22	Yes	6/8/83	
I	Susquehanna 2	BWR	GE	388	11/2/73	6/27/84	3293	3/23/24	Yes	2/12/85	
I	Three Mile Island 1	PWR	B&W	289	5/18/68	4/19/74	2535	5/18/08	Yes	9/2/74	
I	Vermont Yankee	BWR	GE	271	12/11/67	2/28/73	1593	12/11/07	Yes	11/30/72	
I	Yankee Rowe	PWR	W	29	11/4/57	12/24/63	600	7/9/00	Yes	7/1/61	
II	Bellefonte 1	PWR	B&W	438	12-24-74	N/A	0	N/A	Indefinite	N/A	1
II	Bellefonte 2	PWR	B&W	439	12/24/74	N/A	0	N/A	Indefinite	N/A	1
II	Browns Ferry 1	BWR	GE	259	5/10/67	12/20/73	3293	5/10/07	Yes	8/1/74	
II	Browns Ferry 2	BWR	GE	260	5/10/67	8/2/74	3293	5/10/07	Yes	3/1/75	
II	Browns Ferry 3	BWR	GE	296	7/31/68	8/18/76	3293	7/21/08	Yes	3/1/77	
II	Brunswick	BWR	GE	325	2/7/70	11/12/76	2436	2/7/10	Yes	3/18/77	
II	Brunswick 2	BWR	GE	324	2/7/70	12/27/74	2436	12/6/10	Yes	11/3/75	

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region (Continued)

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Docket # (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
II	Catawba 1	PWR	W	413	8/7/75	1/17/85	3411	12/6/24	Yes	6/29/85	
II	Catawba 2	PWR	W	414	8/7/75	5/15/86	3411	2/24/26	Yes	8/19/86	
II	Farley 1	PWR	W	348	8/16/72	6/25/77	2652	8/16/12	Yes	12/1/77	
II	Farley 2	PWR	W	364	8/16/72	3/31/81	2652	8/16/12	Yes	7/30/81	
II	Grand Gulf 1	BWR	GE	416	9/4/74	11/1/84	3833	6/16/22	Yes	7/1/85	
II	Grand Gulf 2	BWR	GE	417	9/4/74	N/A	0	N/A	Indefinite	N/A	1
II	Hatch 1	BWR	GE	321	9/30/69	10/13/74	2436	9/30/09	Yes	12/31/75	
II	Hatch 2	BWR	GE	366	12/27/72	6/13/78	2436	12/27/12	Yes	9/5/79	
II	McGuire 1	PWR	W	369	2/23/73	7/8/81	3411	6/12/21	Yes	12/1/81	
II	McGuire 2	PWR	W	370	2/23/73	5/27/83	3411	3/3/23	Yes	3/1/84	
II	North Anna 1	PWR	W	338	2/19/71	4/1/78	2893	4/1/18	Yes	6/6/78	
II	North Anna 2	PWR	W	339	2/19/71	8/21/80	2893	8/21/20	Yes	12/14/80	
II	Oconee 1	PWR	B&W	269	11/6/67	2/6/73	2568	2/6/13	Yes	7/15/73	
II	Oconee 2	PWR	B&W	270	11/6/67	10/6/73	2568	10/6/13	Yes	9/9/74	
II	Oconee 3	PWR	B&W	287	11/6/67	7/19/74	2568	7/19/14	Yes	12/16/74	
II	Robinson 2	PWR	W	261	4/13/67	9/23/70	2300	4/13/07	Yes	3/7/71	
II	Sequoyah 1	PWR	W	327	5/27/70	9/17/80	3411	5/27/10	Yes	7/1/81	
II	Sequoyah 2	PWR	W	328	5/27/70	9/15/81	3411	5/27/10	Yes	6/1/82	
II	Shearon Harris 1	PWR	W	400	1/27/78	1/12/87	2775	10/24/26	Yes	5/2/87	

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region (Continued)

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Docket # (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
II	St. Lucie 1	PWR	C-E	335	7/1/70	3/1/76	2700	3/1/16	Yes	12/21/76	
II	St. Lucie 2	PWR	C-E	389	5/2/77	6/10/83	2700	4/6/23	Yes	8/8/83	
II	Summer	PWR	W	395	3/21/73	11/12/82	2775	3/21/23	Yes	1/ /84	
II	Surry 1	PWR	W	280	6/25/68	5/25/72	2441	5/25/12	Yes	12/22/72	
II	Surry 2	PWR	W	281	6/25/68	1/29/73	2441	1/29/13	Yes	5/1/73	
II	Turkey Point 3	PWR	W	250	4/27/67	7/19/72	2200	4/27/07	Yes	12/14/72	
II	Turkey Point 4	PWR	W	251	4/27/67	4/10/73	2200	4/27/07	Yes	9/7/73	
II	Vogtle 1	PWR	W	424	6/28/74	3/16	3411	1/16/27	Yes	6/1/87	
II	Vogtle 2	PWR	W	425	6/28/74	N/A	0	N/A	Expected '95	N/A	2
II	Watts Bar 1	PWR	W	390	1/23/73	N/A	0	N/A	Expected '95	N/A	2
II	Watts Bar 2	PWR	W	391	1/23/73	N/A	0	N/A	Expected '95	N/A	2
III	Big Rock Point	BWR	GE	210	5/31/60	5/1/64	240	5/31/00	Yes	3/29/63	
III	Braidwood 1	PWR	W	456	12/31/75	7/2/87	3411	10/17/26	Yes	7/29/88	
III	Braidwood 2	PWR	W	457	12/31/75	5/20/88	3411	12/18/27	Yes	10/17/88	
III	Byron 1	PWR	W	454	12/31/75	2/14/85	3411	10/31/24	Yes	9/16/85	
III	Byron 2	PWR	W	455	12/31/75	1/30/87	3411	11/6/26	Yes	8/21/87	
III	Callaway	PWR	W	483	4/16/76	10/18/84	3565	10/18/24	Yes	12/19/84	
III	Clinton 1	BWR	GE	461	2/24/76	4/17/87	2894	9/29/26	Yes	11/24/87	
III	Crystal River 3	PWR	B&W	302	9/25/68	1/28/77	2544	12/3/16	Yes	3/13/77	

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region (Continued)

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Docket # (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
III	D.C. Cook 1	PWR	W	315	3/25/69	10/25/74	3250	3/25/09	Yes	8/28/75	
III	D.C. Cook 2	PWR	W	316	3/25/69	12/23/77	3411	N/A	Yes	7/1/78	
III	Davis-Besse	PWR	B&W	346	3/24/71	4/22/77	2772	3/24/11	Yes	7/31/78	
III	Dresden 2	BWR	GE	237	1/10/66	12/22/69	2527	12/22/72	Yes	6/9/70	4
III	Dresden 3	BWR	GE	249	10/14/66	3/2/71	2527	10/14/06	Yes	11/16/71	
III	Duane Arnold	BWR	GE	331	6/22/70	2/22/74	1658	6/21/10	Yes	2/1/75	
III	Fermi 2	BWR	GE	341	9/26/72	7/15/85	3292	3/20/25	Yes	1/23/88	
III	Kewaunee	PWR	W	305	8/6/68	12/21/73	1650	8/6/08	Yes	6/16/74	
III	LaSalle 1	BWR	GE	373	9/10/73	8/13/82	3323	5/17/22	Yes	1/1/84	
III	LaSalle 2	BWR	GE	374	9/10/73	3/23/84	3323	12/16/23	Yes	10/19/84	
III	Monticello	BWR	GE	263	6/19/67	1/9/81	1670	9/8/10	Yes	6/30/71	
III	Palisades	PWR	C-E	255	3/14/67	10/16/72	2530	3/1/74	Yes	12/31/71	4
III	Perry 1	BWR	GE	440	5/3/77	11/13/86	3579	3/18/26	Yes	11/18/87	
III	Perry 2	BWR	GE	441	5/3/77	N/A	0	N/A	Indefinite	N/A	1
III	Point Beach 1	PWR	W	266	7/19/67	10/5/70	1518	10/5/10	Yes	12/21/70	
III	Point Beach 2	PWR	W	301	7/25/68	3/8/73	1518	3/8/13	Yes	10/1/72	
III	Prairie Island 1	PWR	W	282	6/25/68	4/5/74	1650	8/9/13	Yes	12/16/73	
III	Prairie Island 2	PWR	W	306	6/25/68	10/29/74	1650	10/29/14	Yes	12/21/74	
III	Quad Cities 1	BWR	GE	254	2/15/67	12/14/72	2511	2/15/07	Yes	2/18/73	

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region (Continued)

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Doc (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
III	Quad Cities 2	BWR	GE	265	2/15/67	12/14/72	2511	2/15/07	Yes	3/10/73	
III	Zion 1	PWR	W	295	12/26/68	10/19/73	3250	12/26/06	Yes	12/31/73	
III	Zion 2	PWR	W	304	12/26/68	11/14/73	3250	12/26/08	Yes	9/17/74	
IV	ANO-1	PWR	B&W	313	12/6/68	5/21/74	2568	12/6/08	Yes	12/19/74	
IV	ANO-2	PWR	C-E	368	12/6/72	12/14/78	2815	12/6/12	Yes	4/26/80	
IV	Comanche Peak 1	PWR	W	445	12/19/74	N/A	0	N/A	Expected '95	N/A	5
IV	Comanche Peak 2	PWR	W	446	12/19/74	N/A	0	N/A	Indefinite	N/A	3
IV	Cooper	BWR	GE	298	6/4/68	1/18/74	2381	6/4/08	Yes	7/1/74	
IV	Fort Calhoun 1	PWR	C-E	285	6/7/68	8/9/73	1500	6/7/08	Yes	6/20/74	
IV	Fort St. Vrain	HTGR	GA	267	9/17/68	12/21/73	842	9/17/08	Yes	7/1/79	
IV	River Bend 1	BWR	GE	458	3/25/77	11/20/85	2894	8/29/25	Yes	6/16/86	
IV	South Texas 1	PWR	W	498	12/22/75	3/22/88	3800	8/20/27	Yes	8/25/88	
IV	South Texas 2	PWR	W	499	12/22/75	12/16/88	190	12/16/28			5
IV	Waterford 3	PWR	C-E	382	11/14/74	3/16/85	3390	12/18/24	Yes	9/24/85	
IV	Wolf Creek	PWR	W	482	5/31/77	6/4/85	341[illegible]	[illegible]/25	Yes	9/3/85	
V	Diablo Canyon 1	PWR	W	275	4/23/68	11/2/84	3338	[illegible]	Yes	5/7/85	
V	Diablo Canyon 2	PWR	W	323	12/[illegible]/70	[illegible]/26/85	3411	[illegible]10	Yes	3/13/86	
V	Palo Verde 1	PWR	C-E	528	5/25/76	6/1/85	3800	2/31/24	Yes	1/28/86	
V	Palo Verde 2	PWR	C-E	529	5/25/76	4/24/86	3800	12/9/25	Yes	9/19/86	

Table 2-6. Summary of General Licensing Data - Sorted by NRC Region (Continued)

NRC Region	Reactor Plant	Reactor Type	NSSS	NRC Docket # (50-)	Construction Permit	Operating Licence	Lic. Power MWt	Expiration Date	Comm'l Ops ?	Date of Comm'l Ops	Notes
V	Palo Verde 3	PWR	C-E	530	5/25/76	11/ 5/87	3800	3/25/27	Yes	1/8/88	
V	Rancho Seco	PWR	B&W	312	10/11/68	8/16/74	2772	10/11/08	Yes	4/17/75	
V	San Onofre 1	PWR	W	206	3/2/64	3/27/67	1347	9/27/2072	Yes	1/1/68	4
V	San Onofre 2	PWR	C-E	361	10/18/73	9/7/82	3390	10/18/13	Yes	8/8/83	
V	San Onofre 3	PWR	C-E	362	10/18/73	9/16/83	3390	10/18/13	Yes	4/1/84	
V	Trojan	PWR	W	344	2/8/71	11/21/75	3411	2/8/11	Yes	5/20/76	
V	WNP-1	PWR	B&W	460	12/24/75	N/A	0	N/A	Indefinite	N/A	1
V	WNP-2	BWR	GE	397	3/19/73	4/13/84	3323	12/20/23	Yes	12/13/84	
V	WNP-3	PWR	C-E	508	4/11/78	N/A	0	N/A	Indefinite	N/A	1

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 pu ge lines.

B. Containment Pressure Con ol, 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

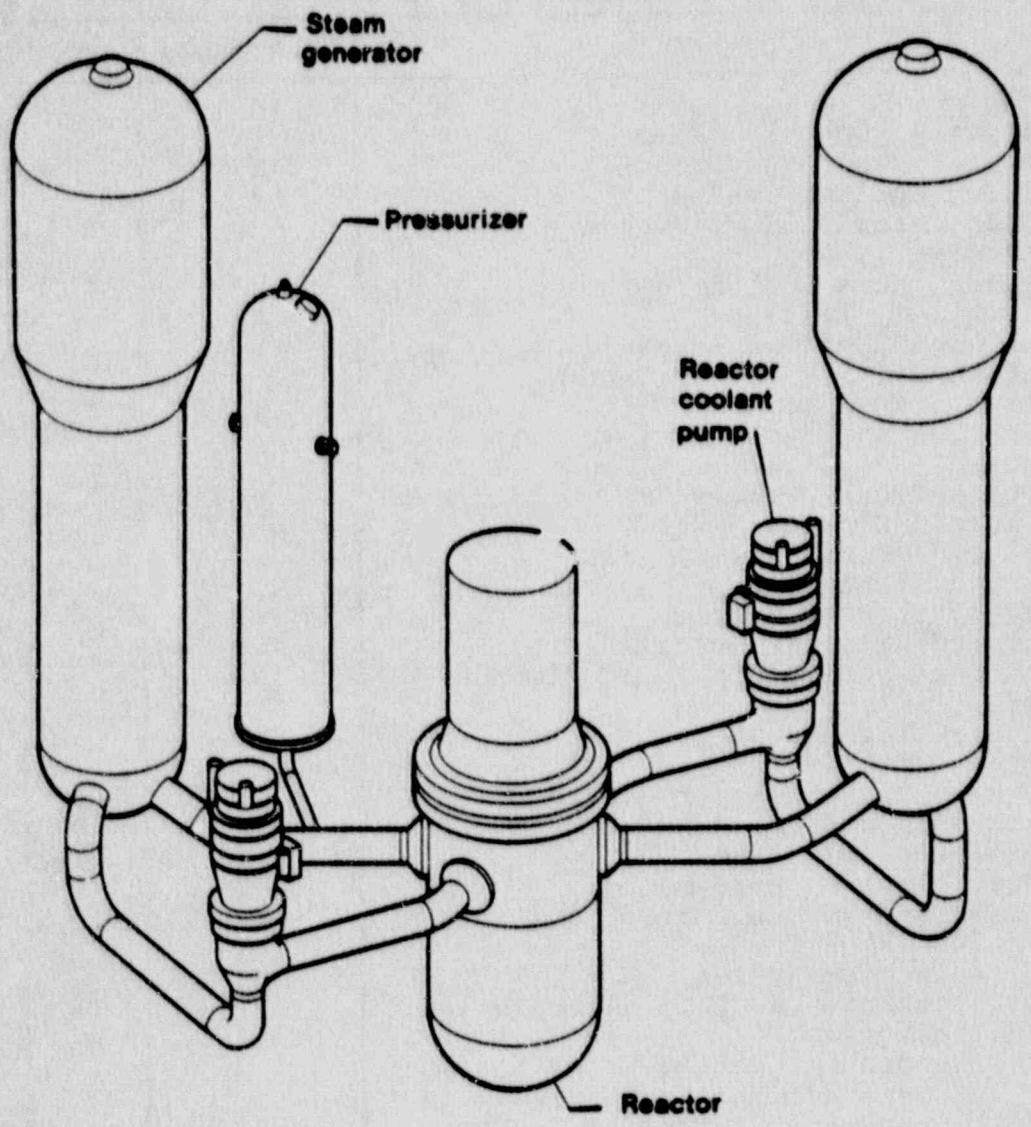


Figure 3.2-1. Westinghouse 2-Loop PWR NSSS

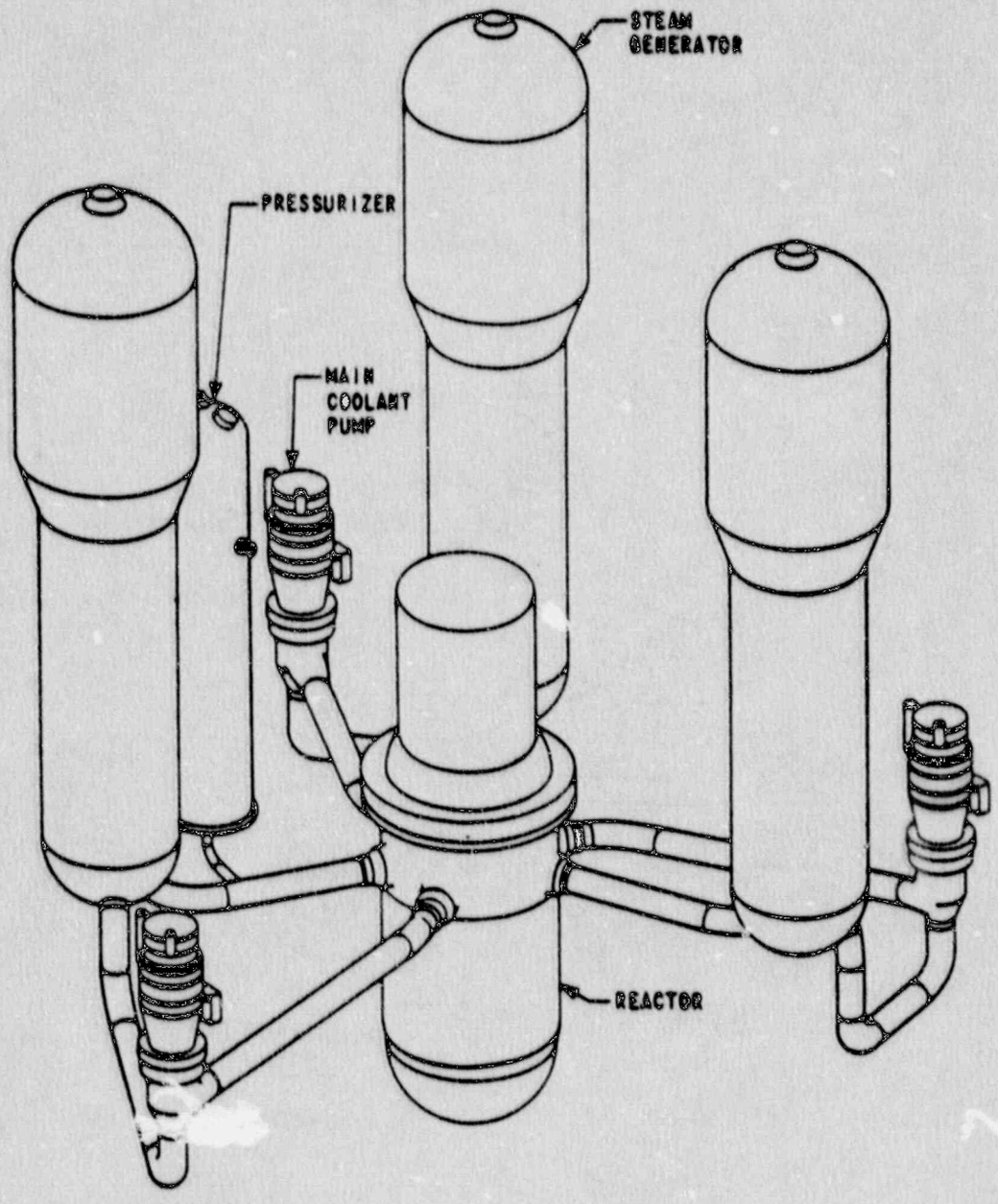


Figure 3.2-2. Westinghouse 3-Loop PWR NSSS

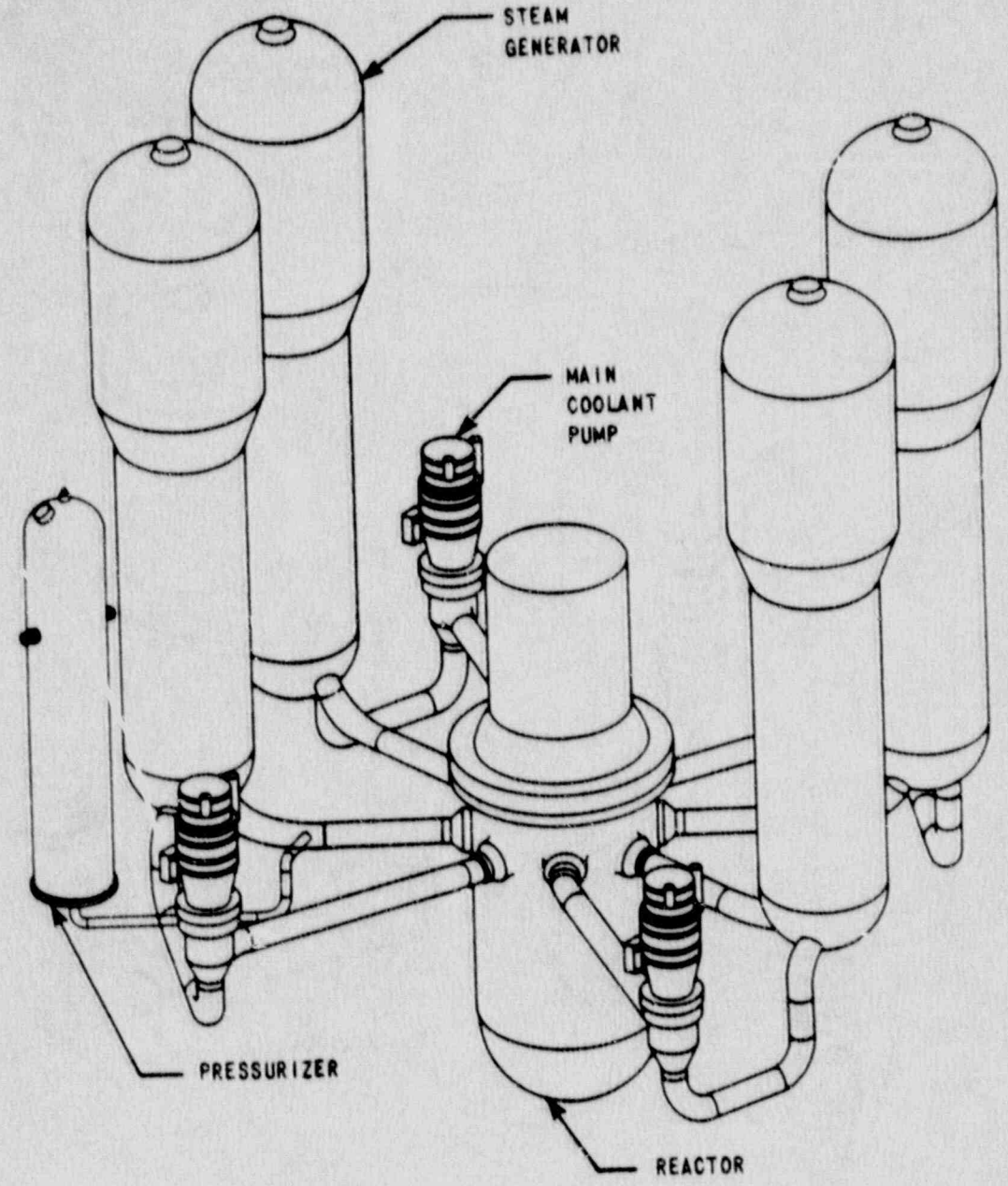


Figure 3.2-3. Westinghouse 4-Loop PWR NSSS

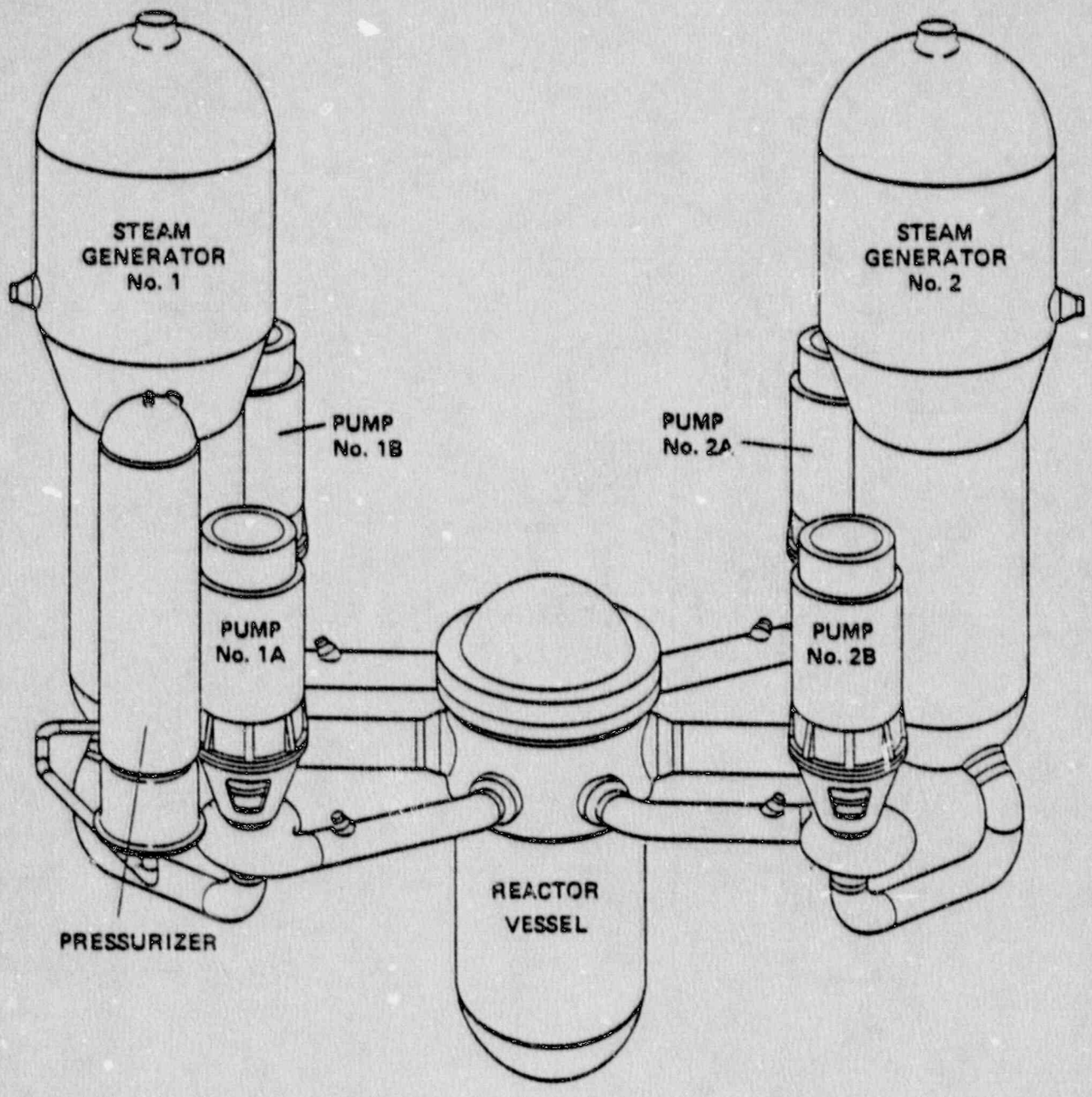


Figure 3.2-4. Combustion Engineering 2-Loop PWR NSSS

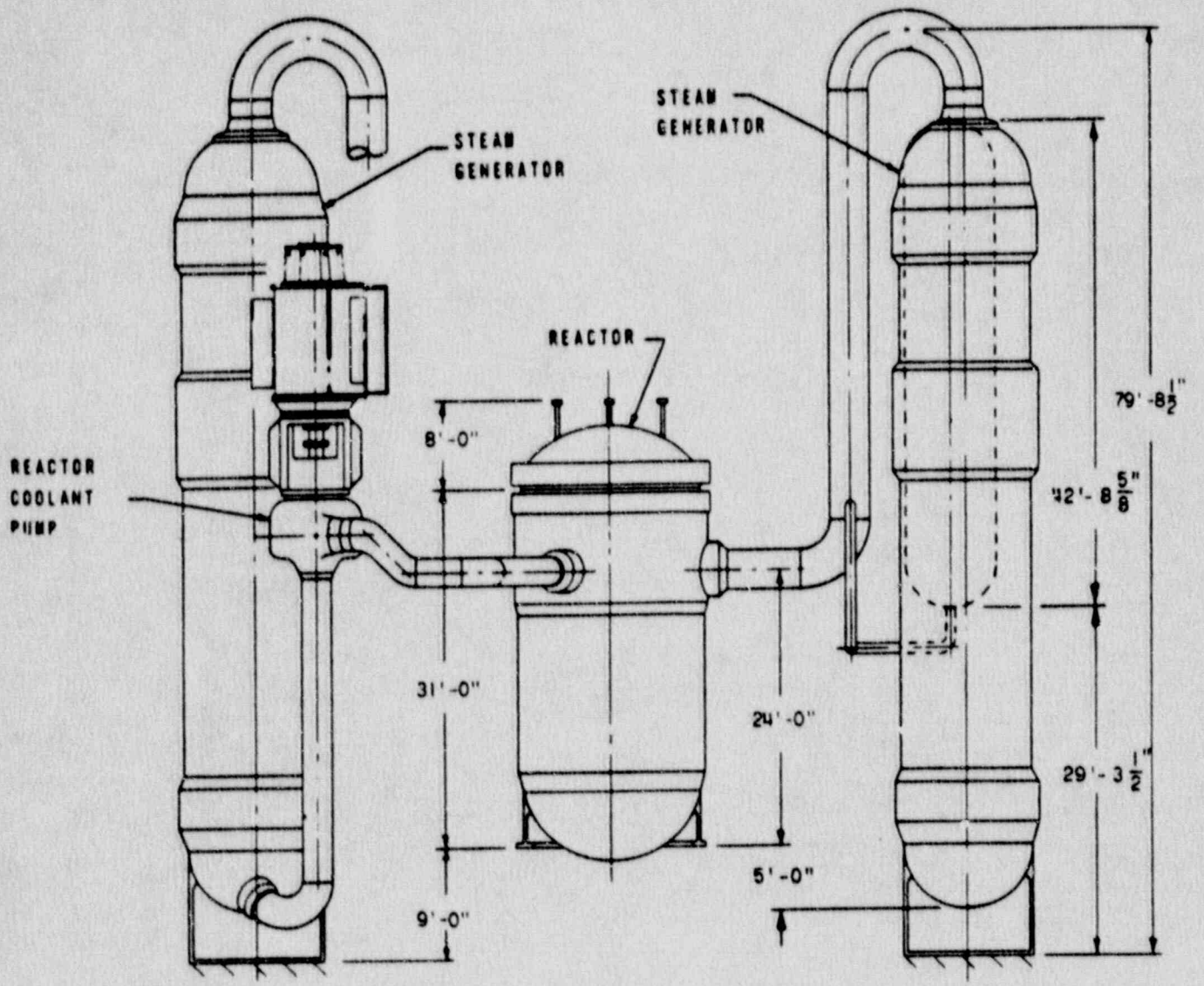


Figure 3.2-5. Babcock & Wilcox 177 "Lowered-Loop" PWR NSSS

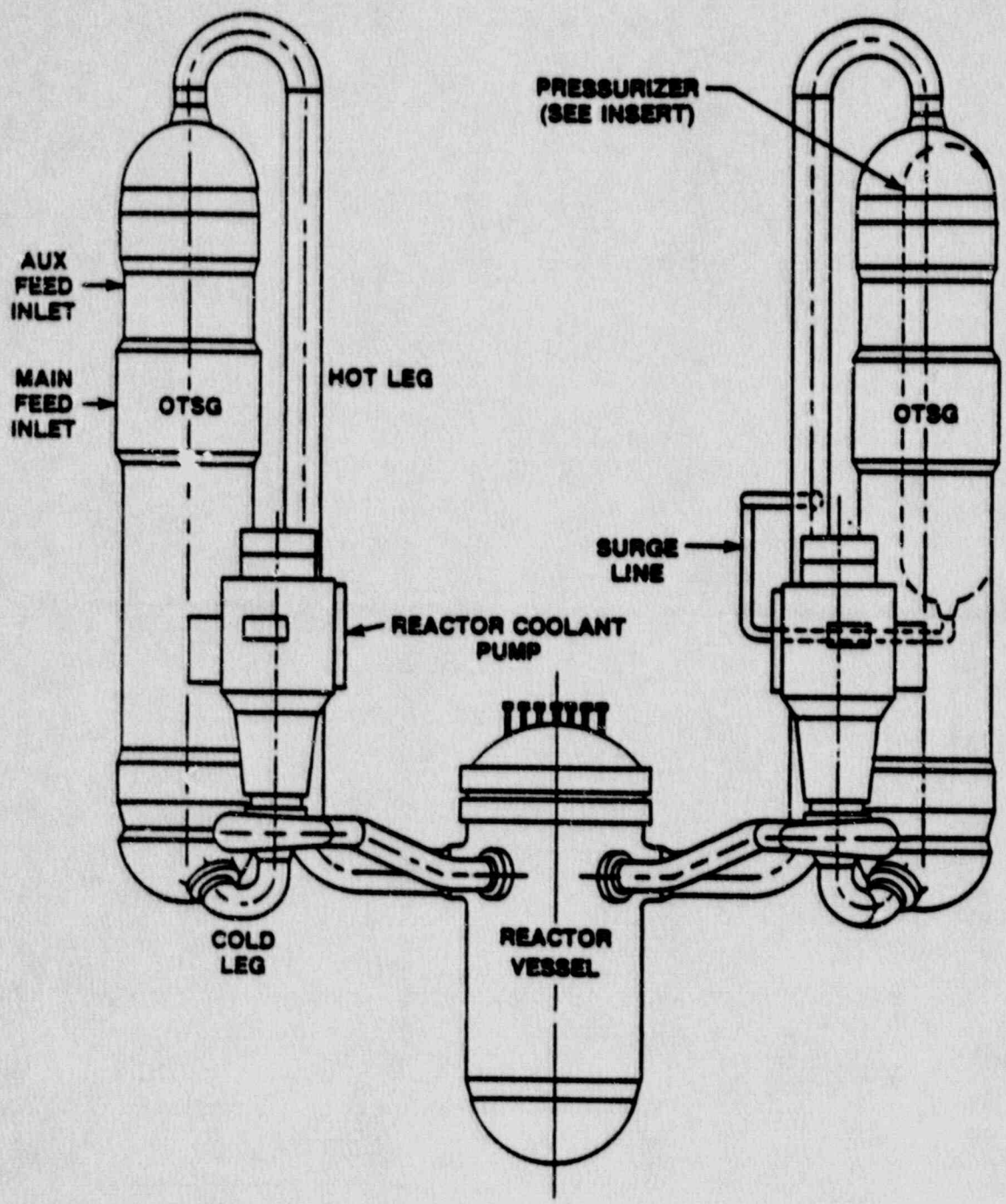


Figure 3.2-6. Babcock & Wilcox 177 "Raised-Loop" PWR NSSS

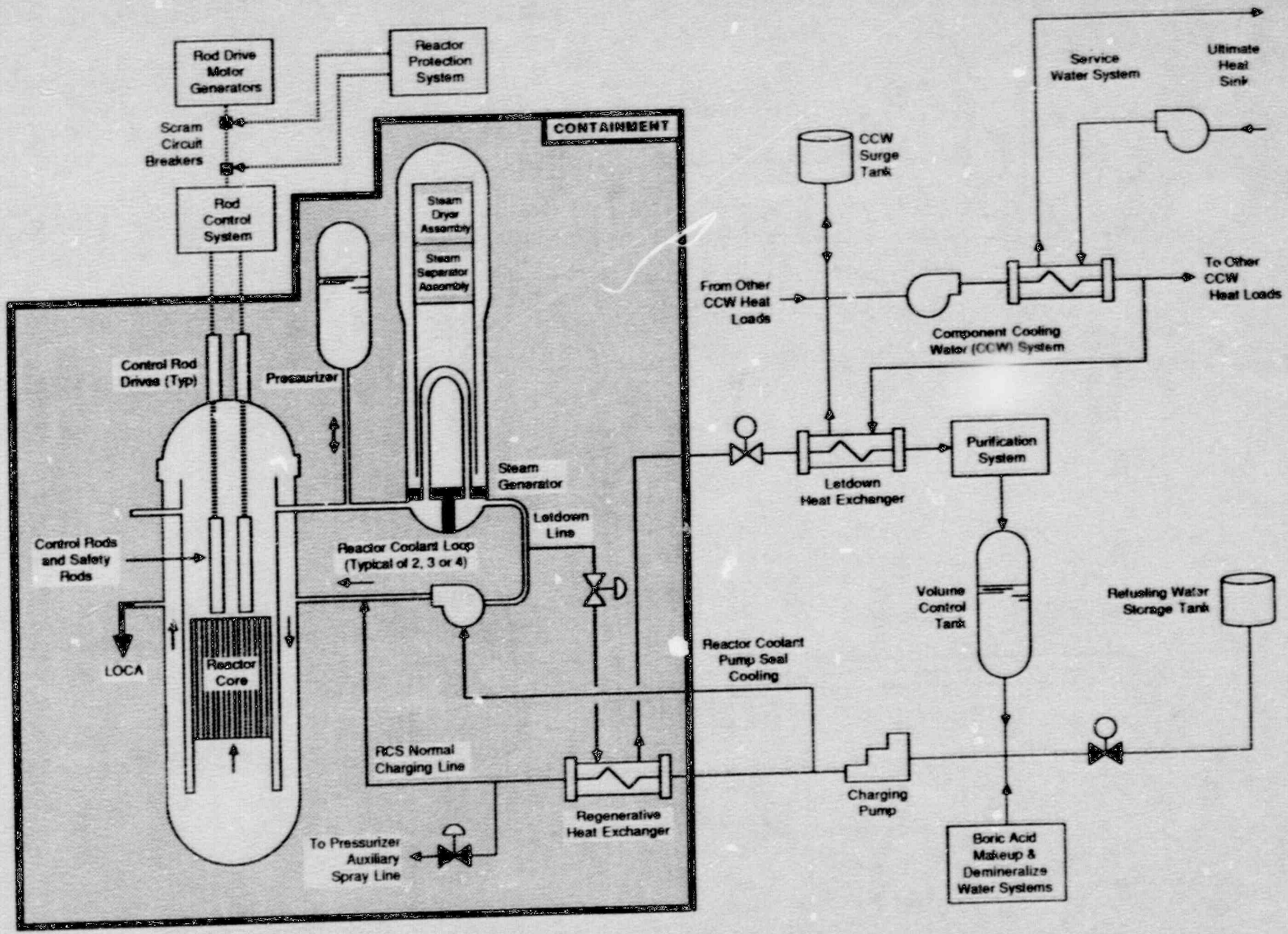


Figure 3.4-1. PWR Reactivity Control Systems: Control Rods and the Chemical and Volume Control System

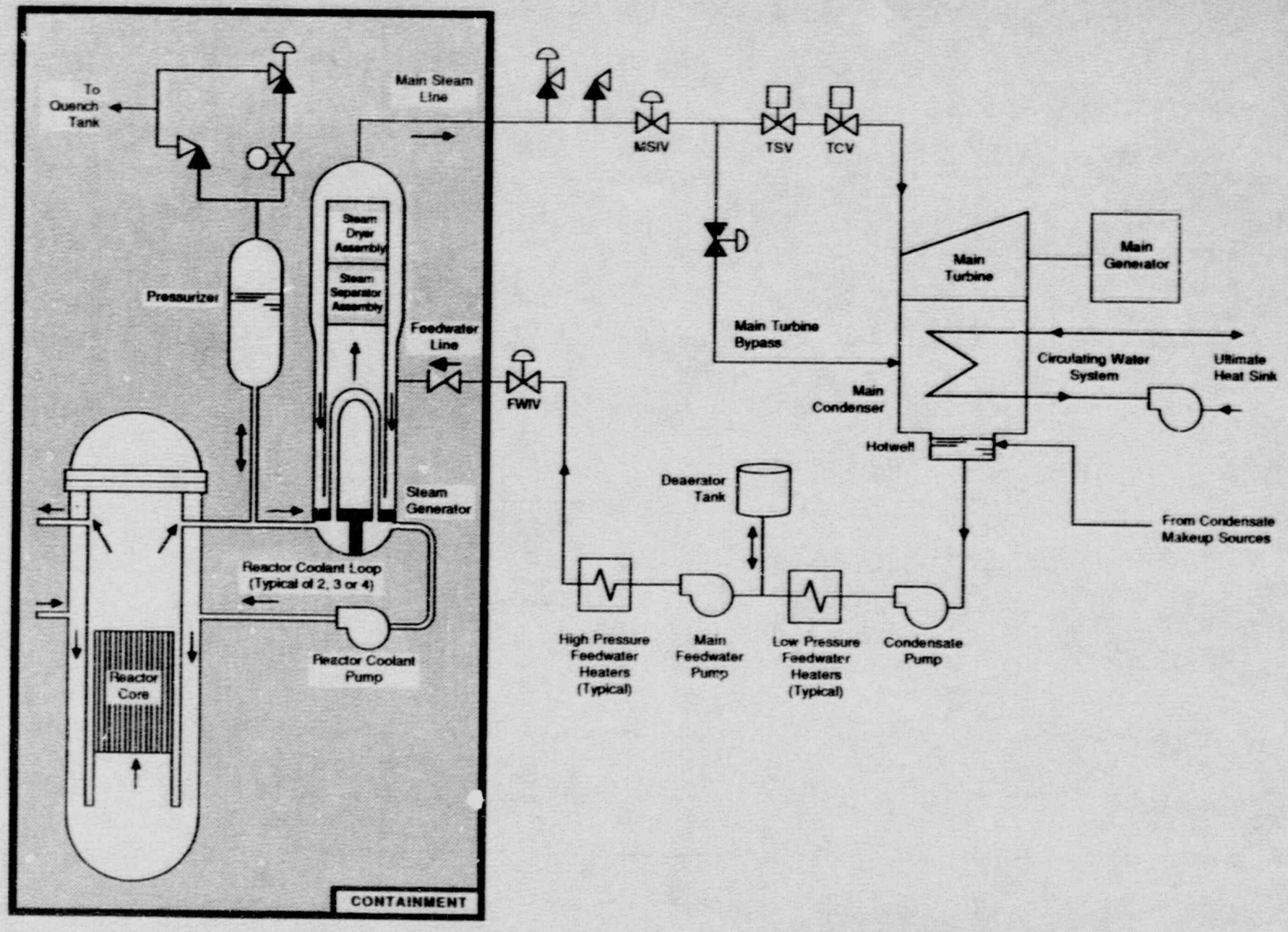


Figure 3.5-1. Normal PWR Heat Transport Paths During Power Operation

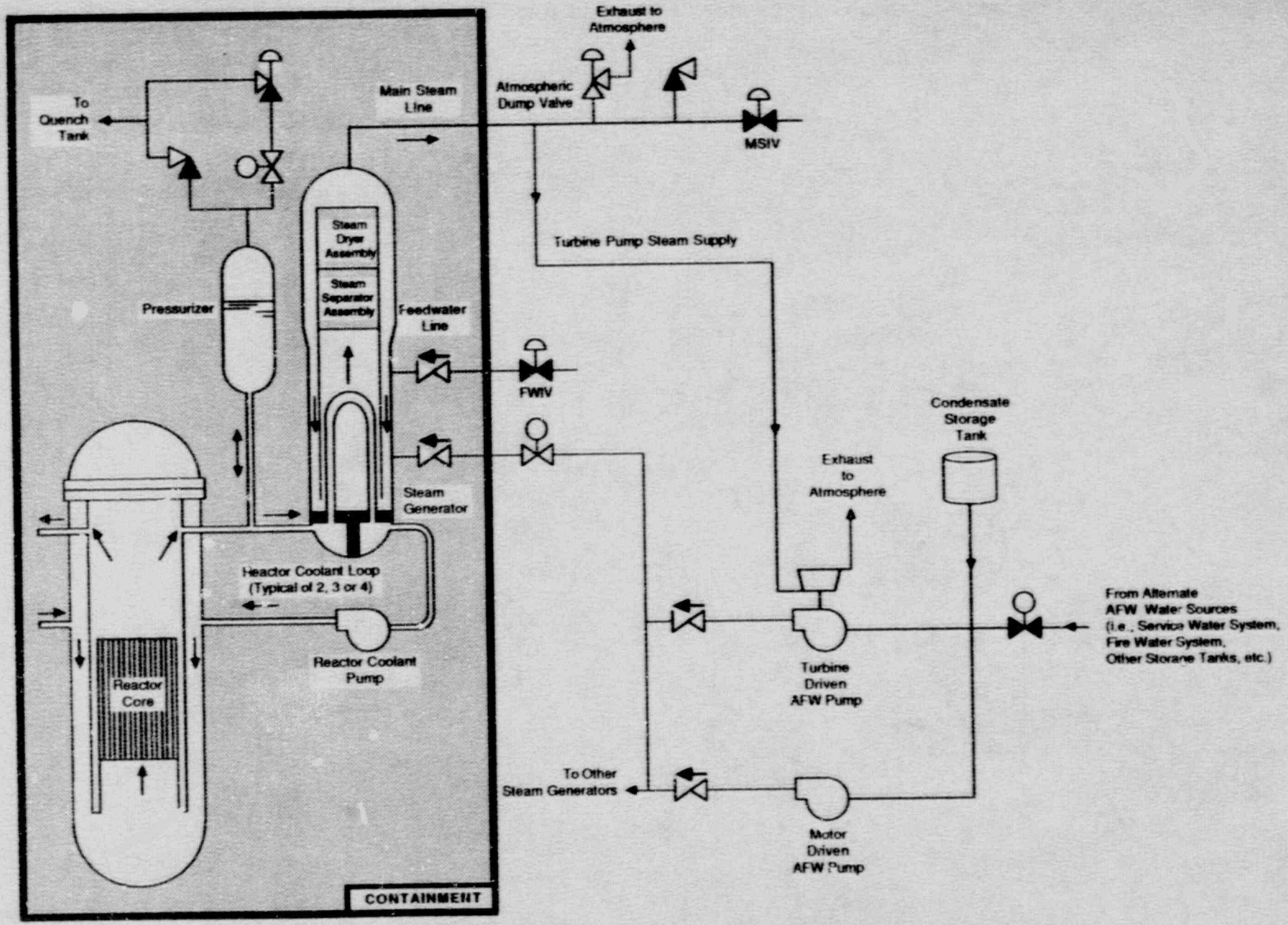


Figure 3.6-1. PWR Heat Transport Paths During Auxiliary Feedwater System Operation

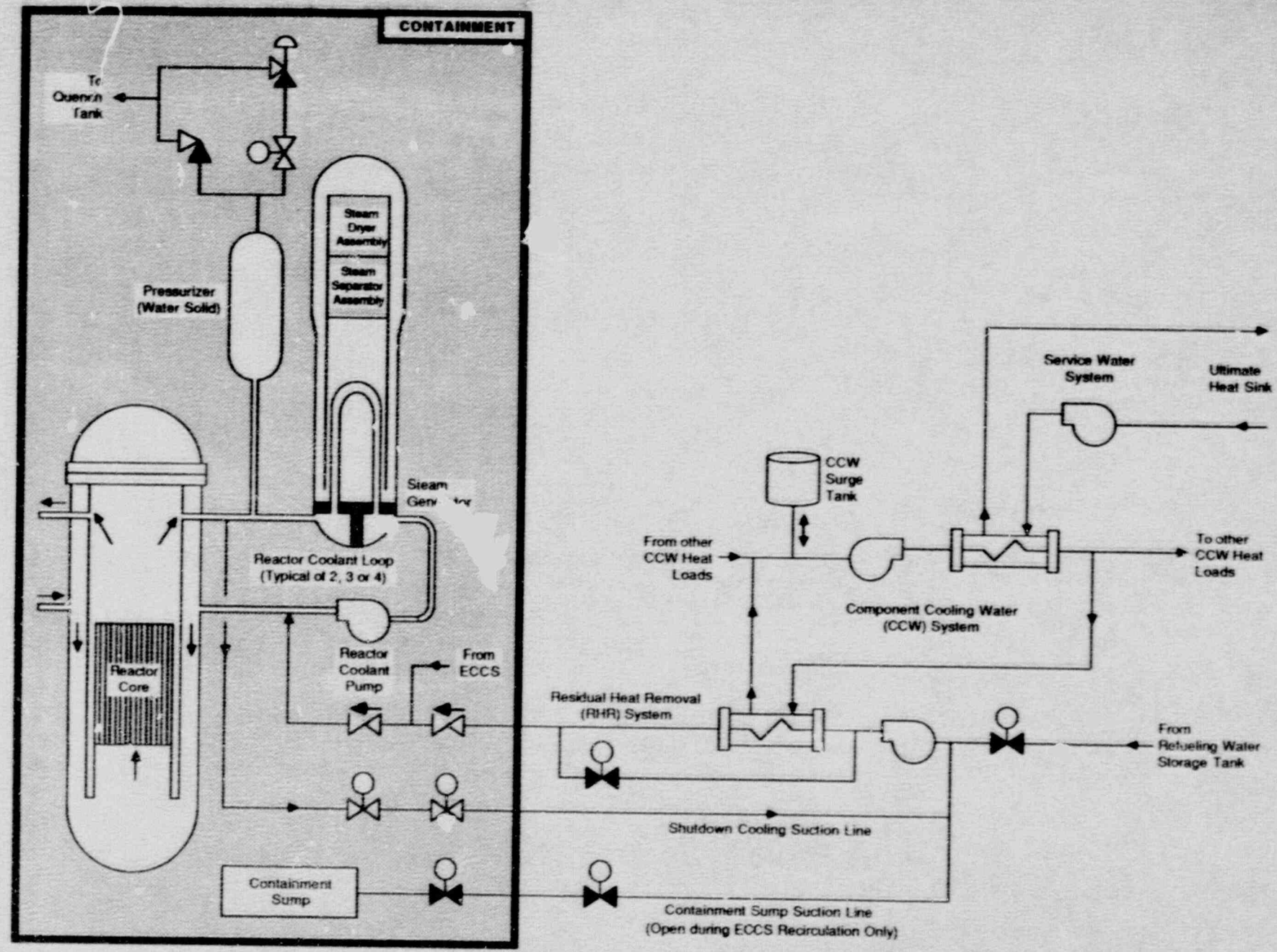


Figure 3.7-1. Normal PWR Heat Transport Paths During Shutdown Cooling

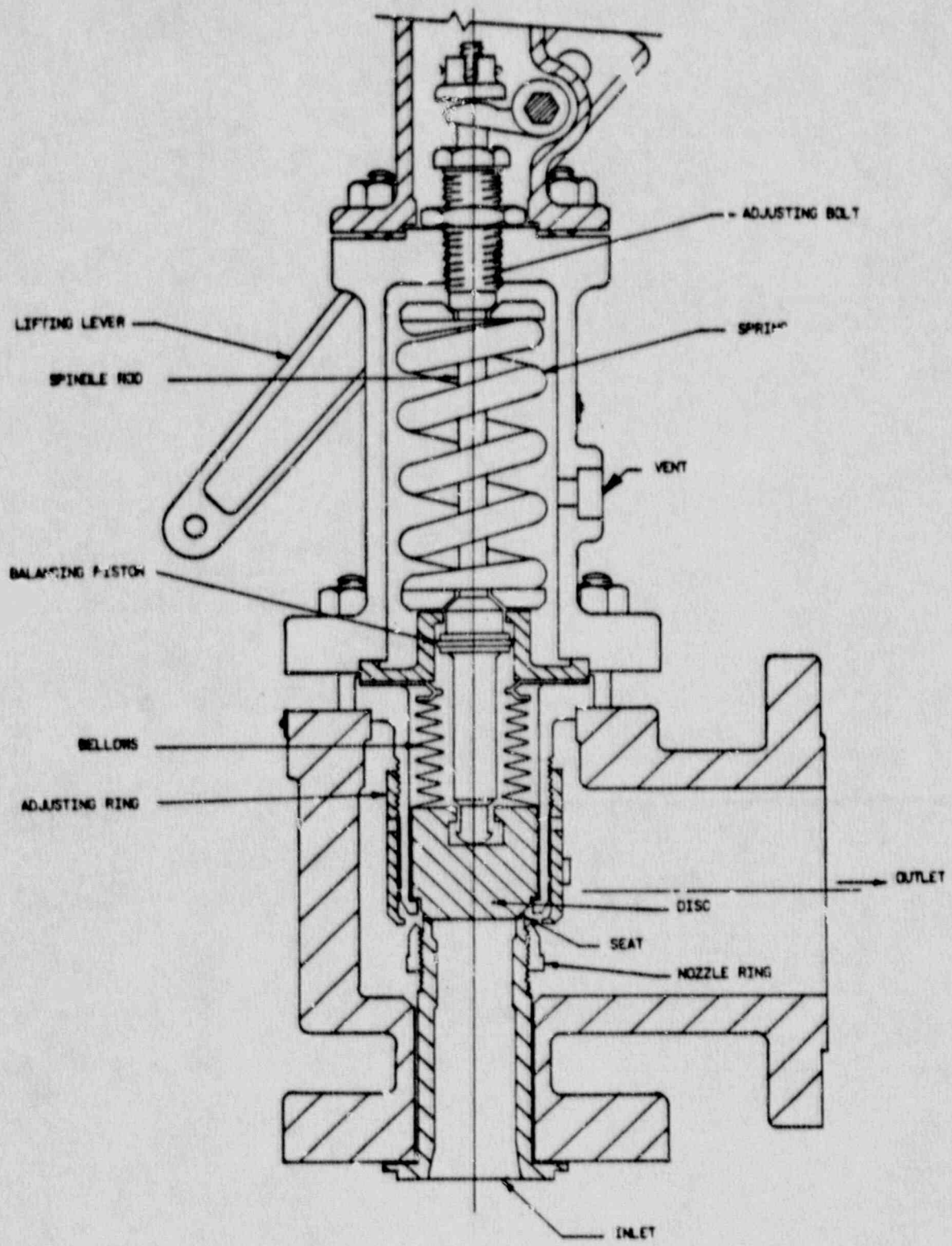


Figure 3.8-1. Typical Pressurizer Safety Valve

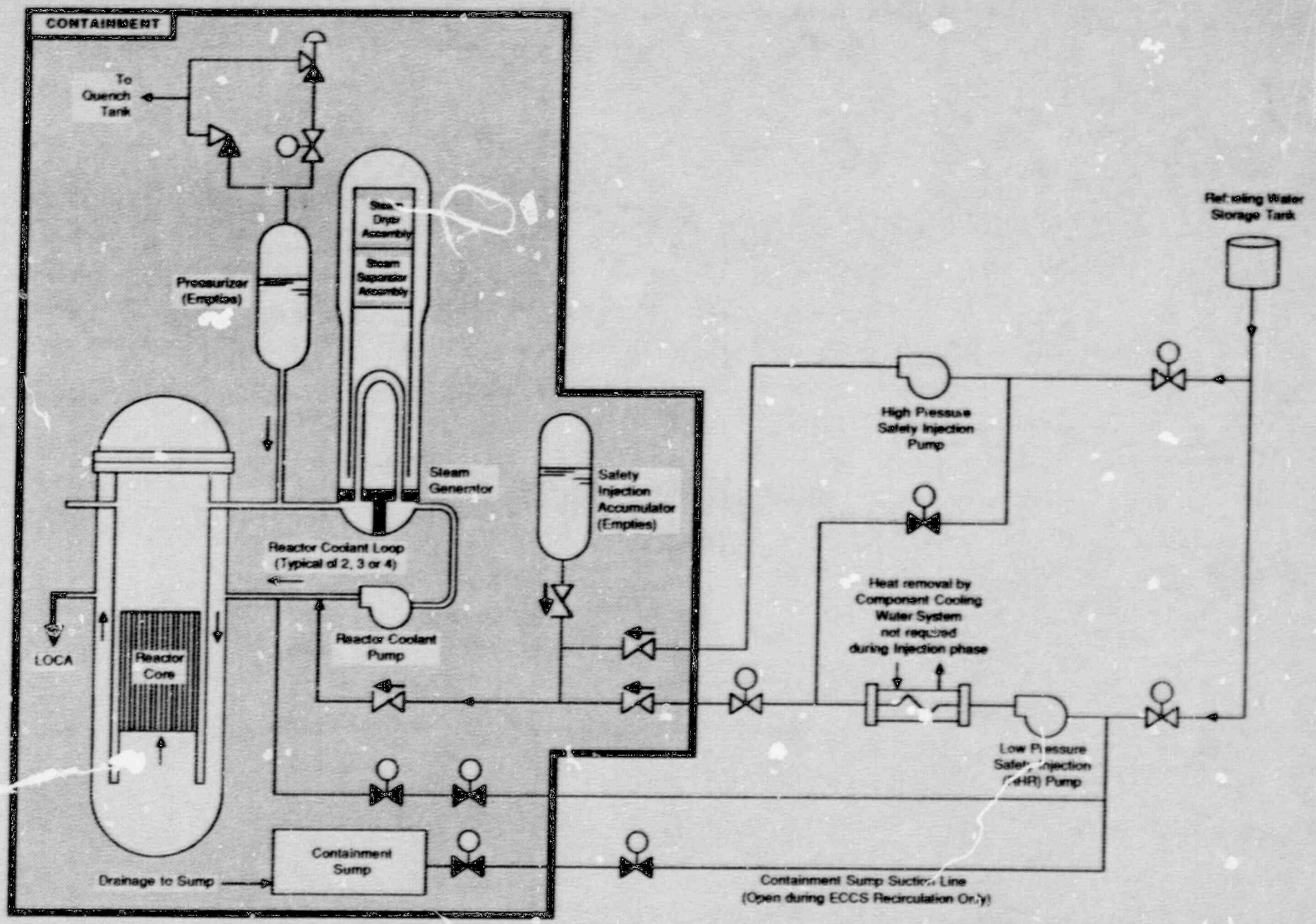


Figure 3.9-1. PWR Coolant Injection and Heat Transport Paths During a Large LOCA-ECCS Injection Phase

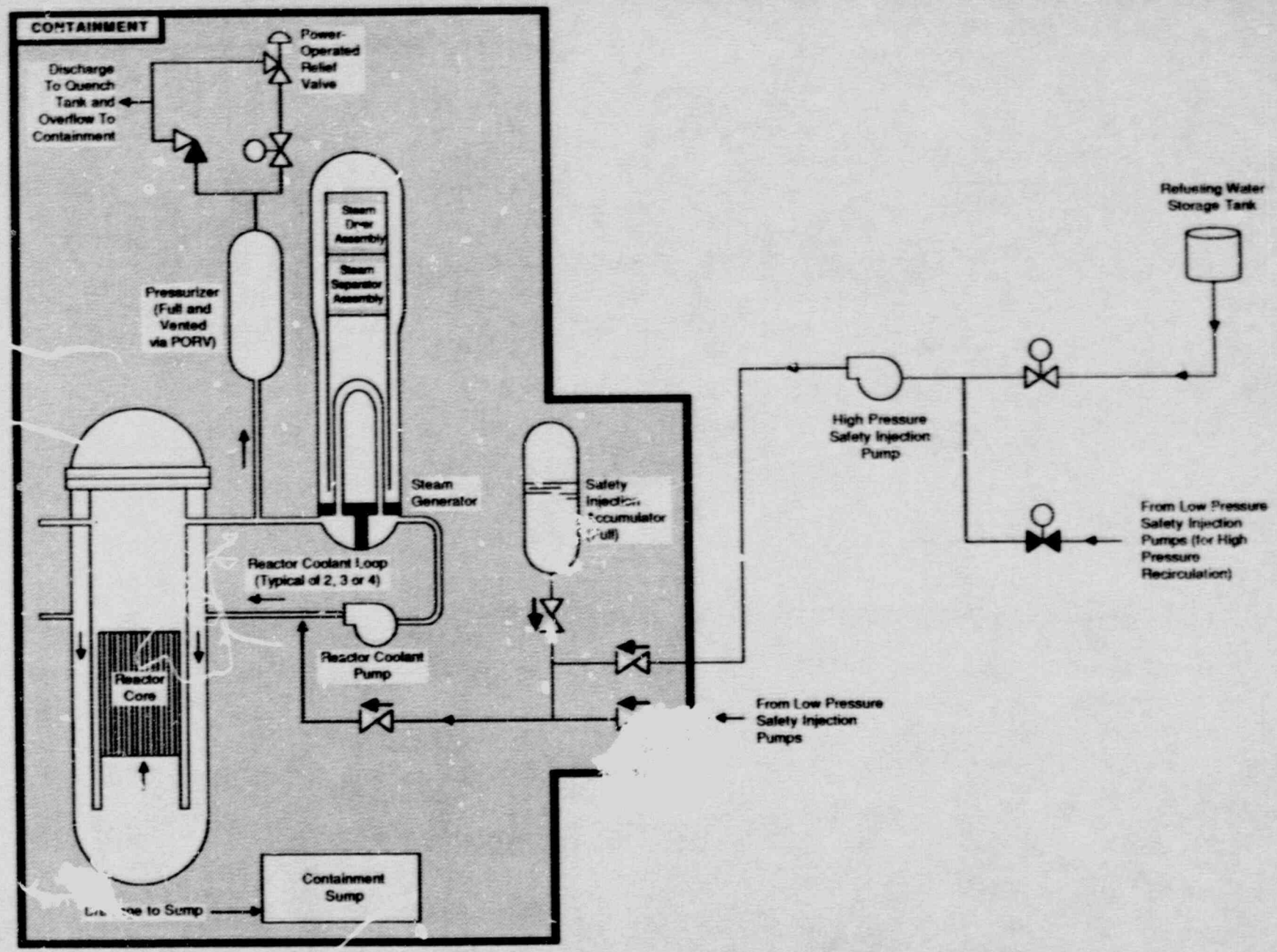


Figure 3.9-2. PWR Coolant Injection and Heat Transport Paths During Post-Transient, High-Pressure Feed-and-Bleed Cooling

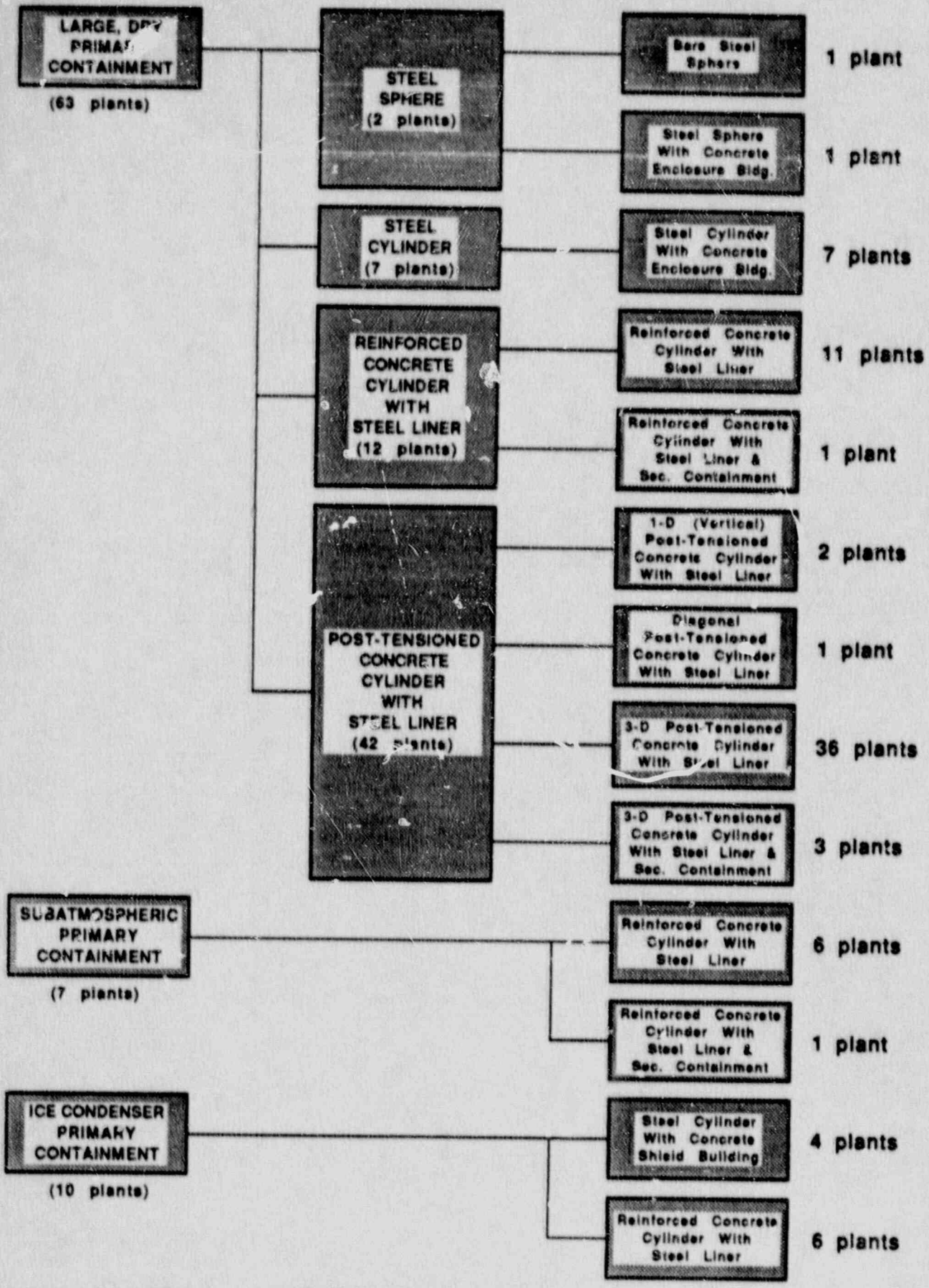


Figure 3.10-1. Distribution of PWR Containment Types

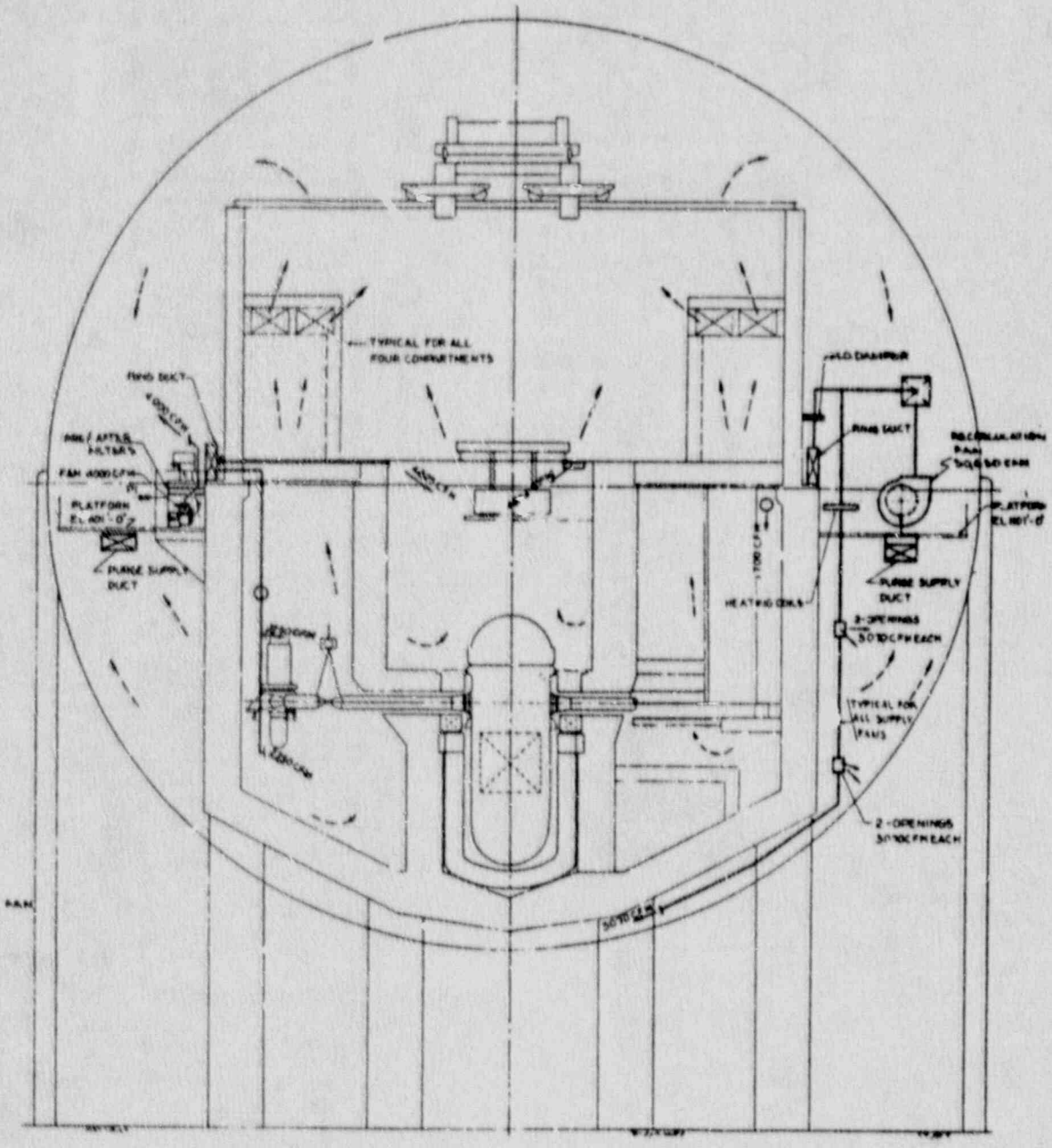


Figure 3.10-2. Yankee-Rowe Large, Dry Containment (Steel Sphere)

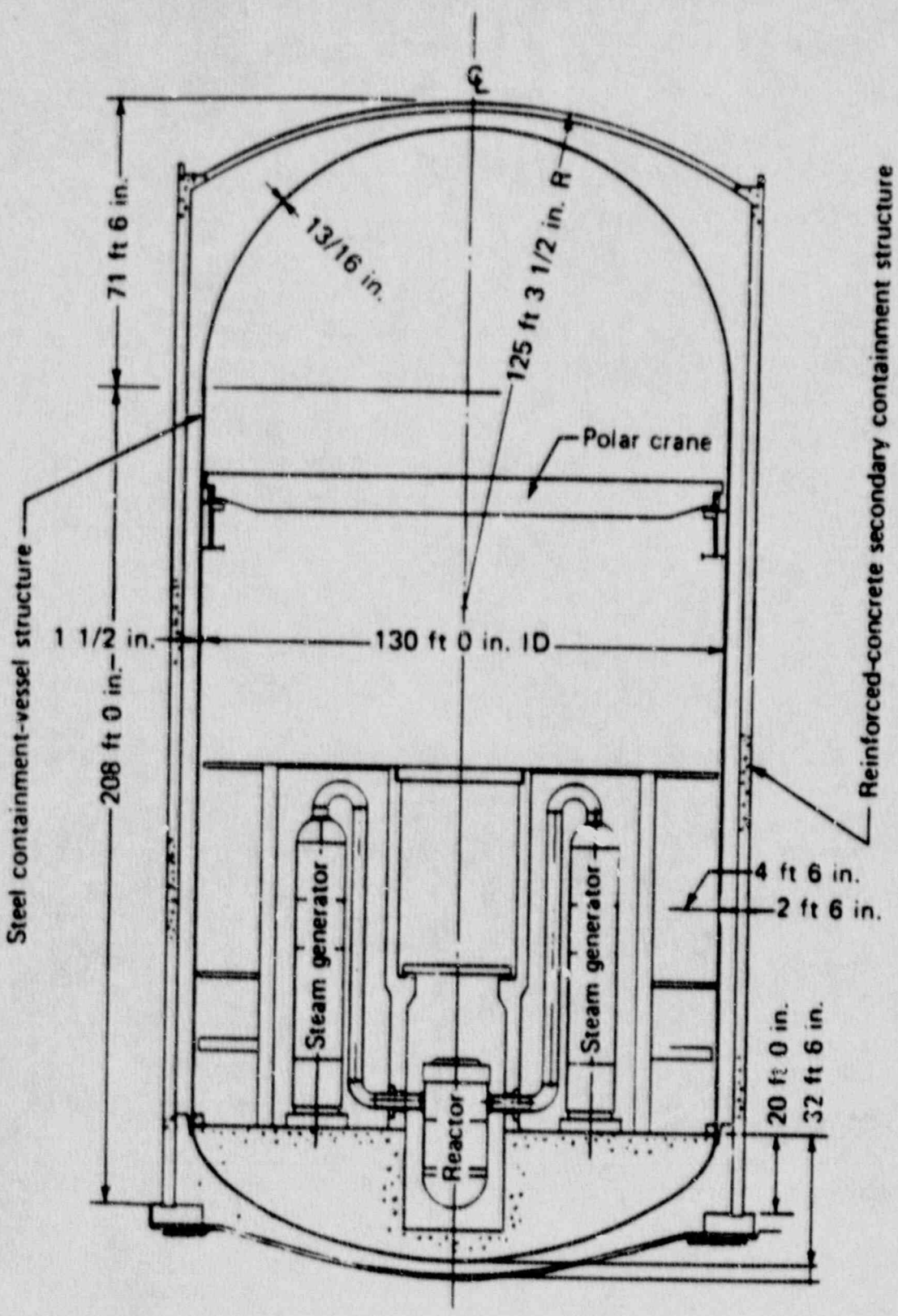


Figure 3.10-3. Davis-Besse Large, Dry Containment (Steel Cylinder with Concrete Shield Building)

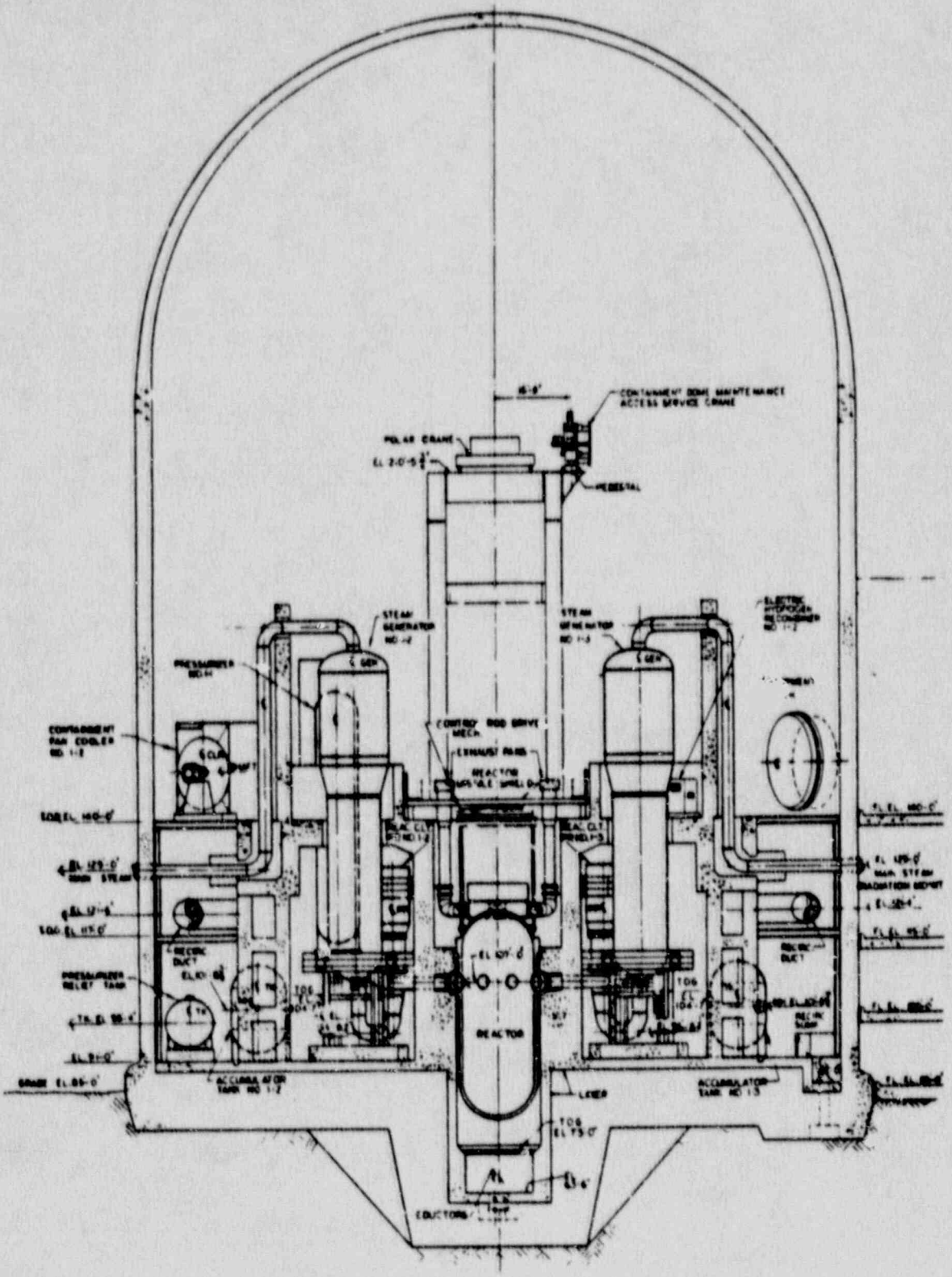


Figure 3.10-4. Diablo Canyon Large, Dry Containment (Reinforced Concrete with Steel Liner)

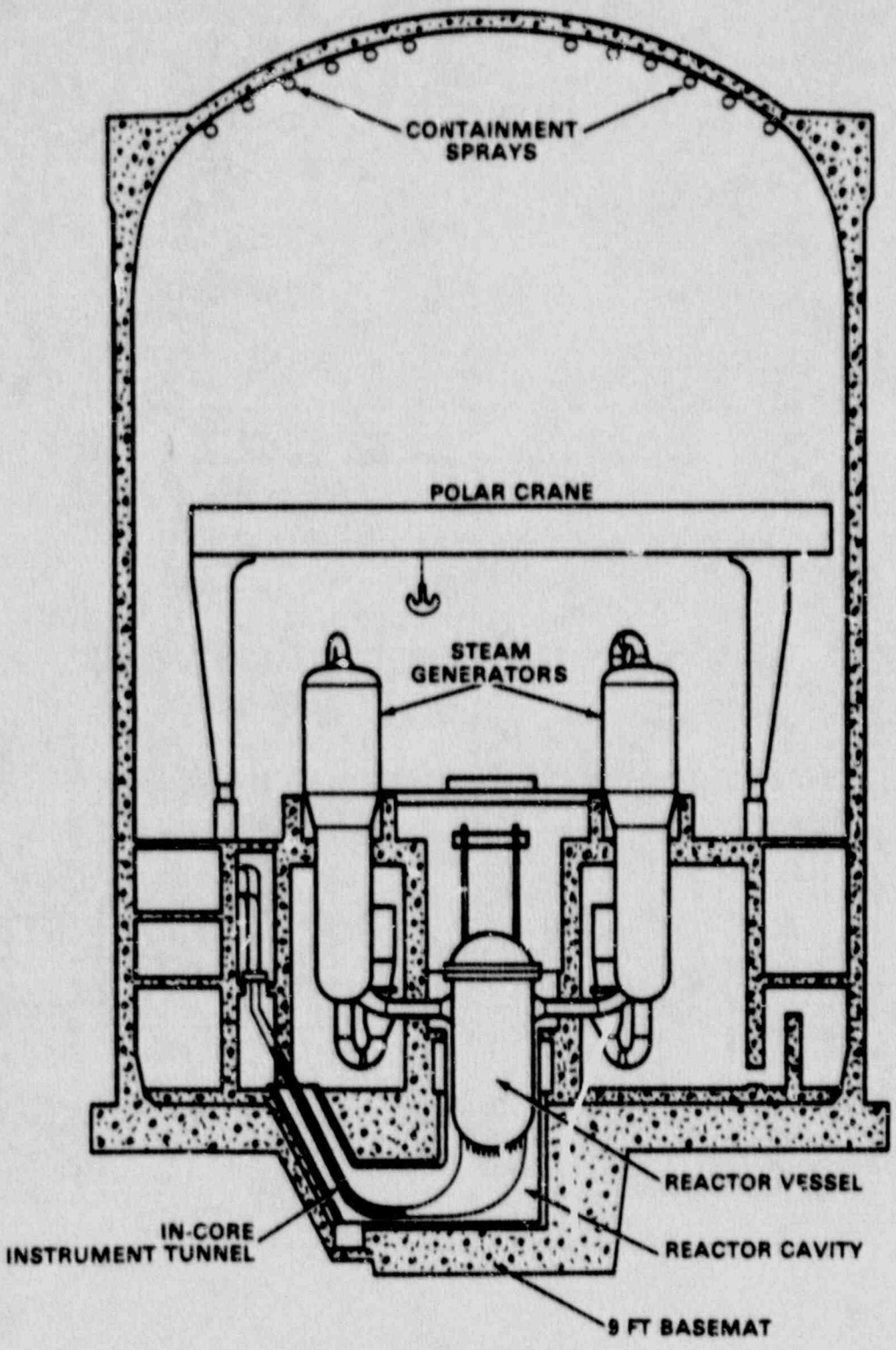


Figure 3.10-5. Zion Large, Dry Containment (Post-Tensioned Concrete with Steel Liner)

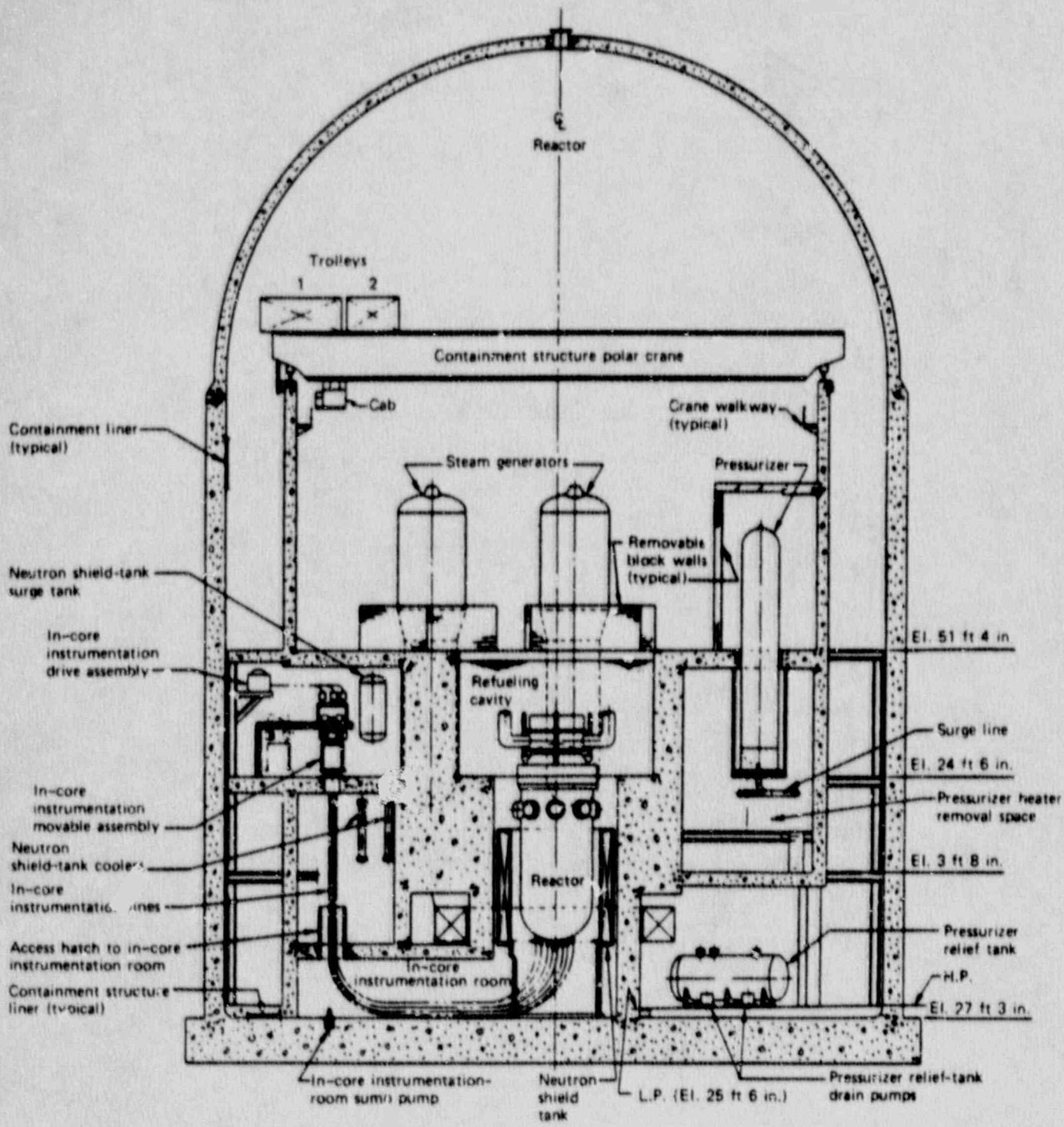


Figure 3.10-6. Millstone 3 Subatmospheric Containment (Reinforced Concrete with Steel Liner)

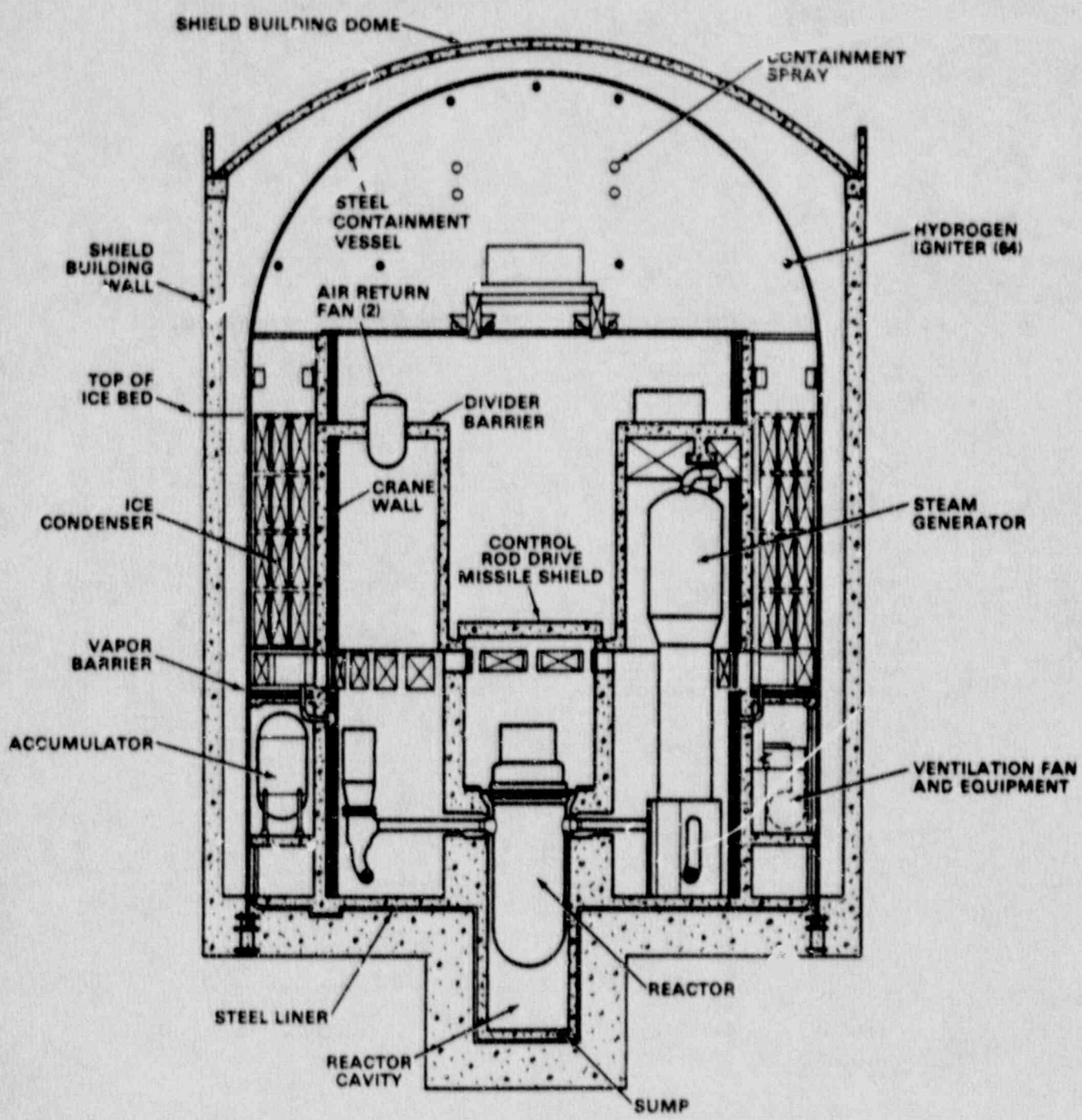


Figure 3.10-7. Sequoyah Ice Condenser Containment (Steel Cylinder with Concrete Shield Building)

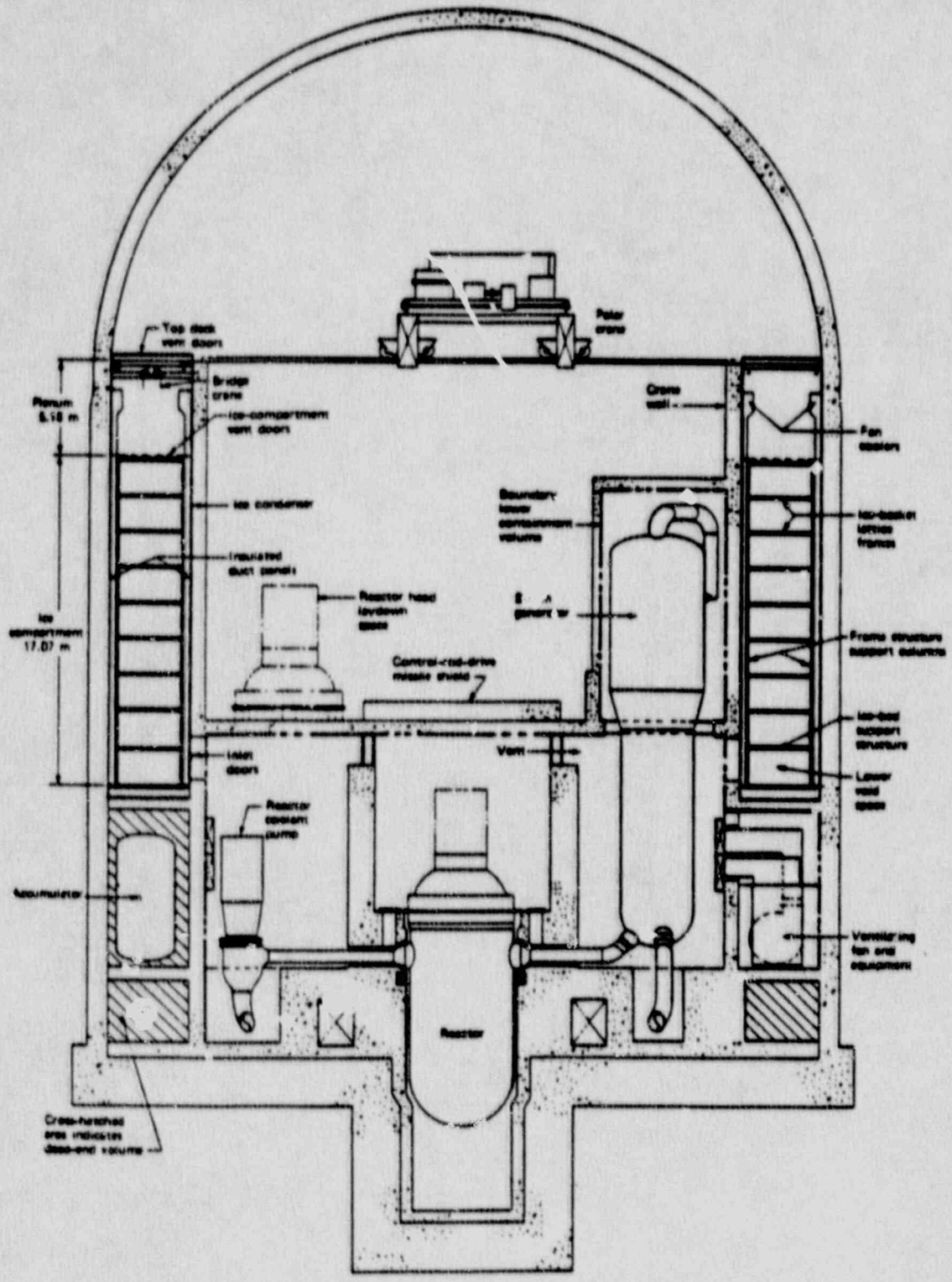


Figure 3.10-8. Ice Condenser Containment with Reinforced Concrete Structure and Steel Liner

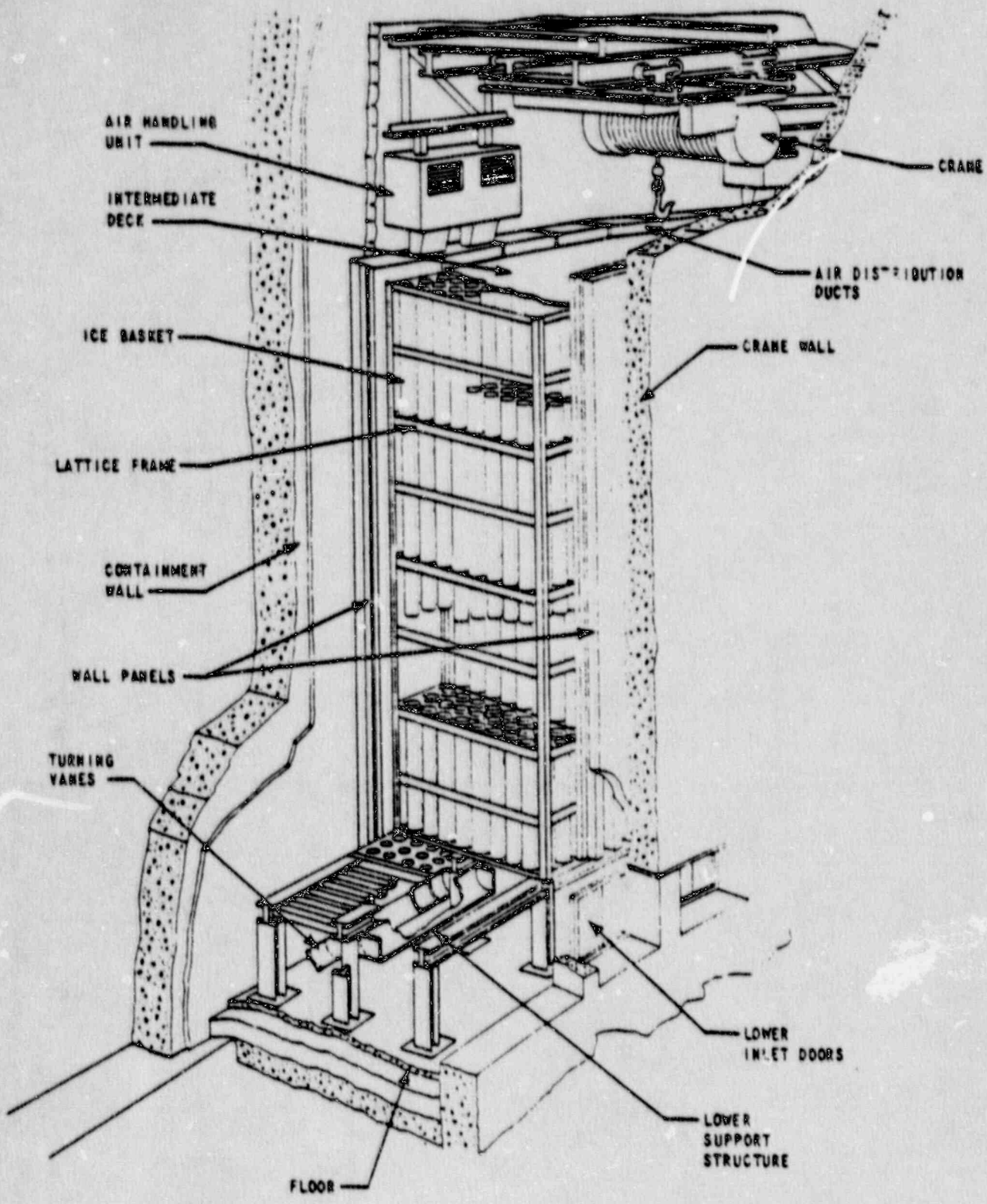


Figure 3.10-9. General Arrangement of an Ice Condenser

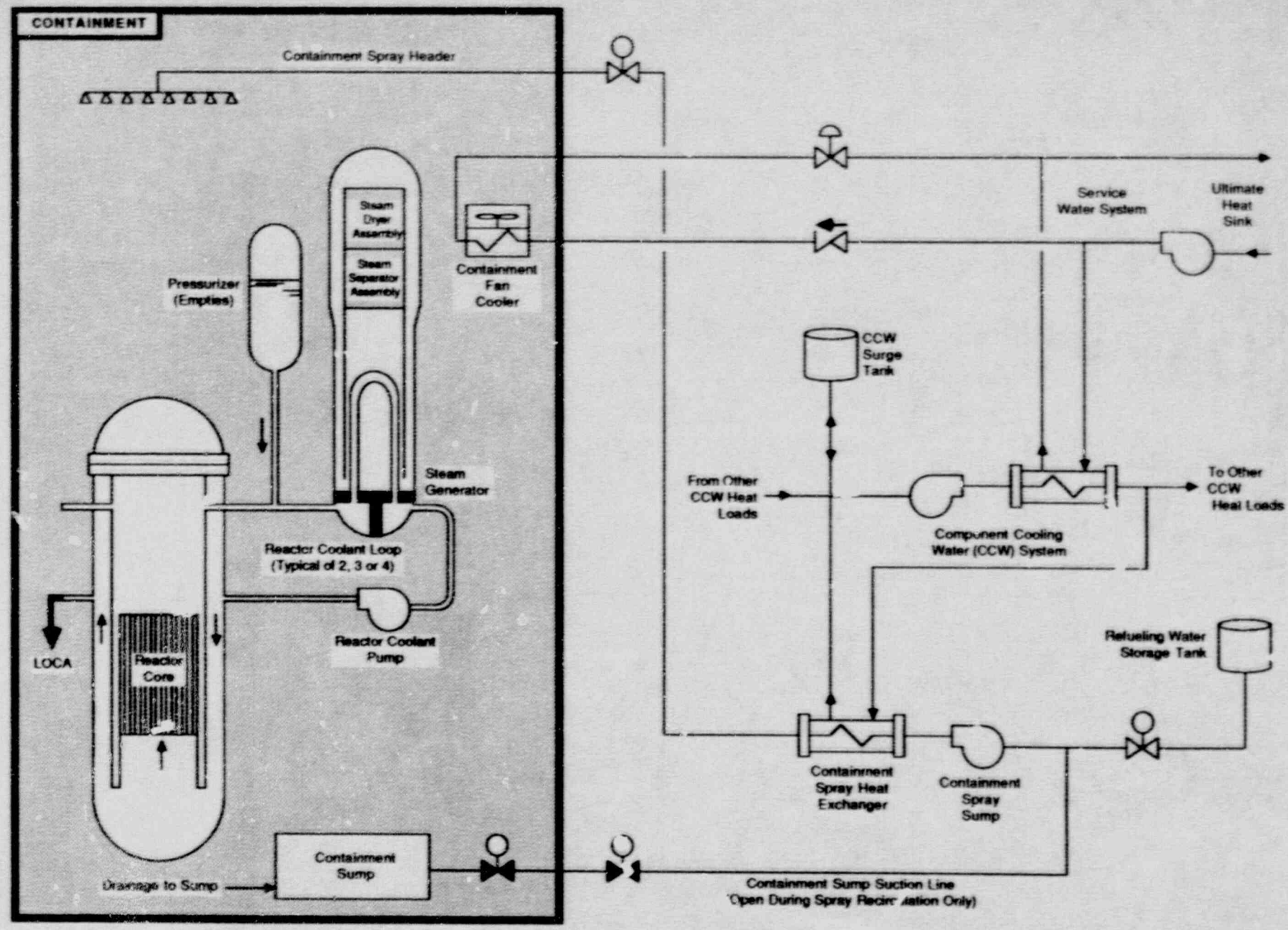


Figure 3.10-10. Typical PWR Containment Post-LOCA Heat Removal Paths

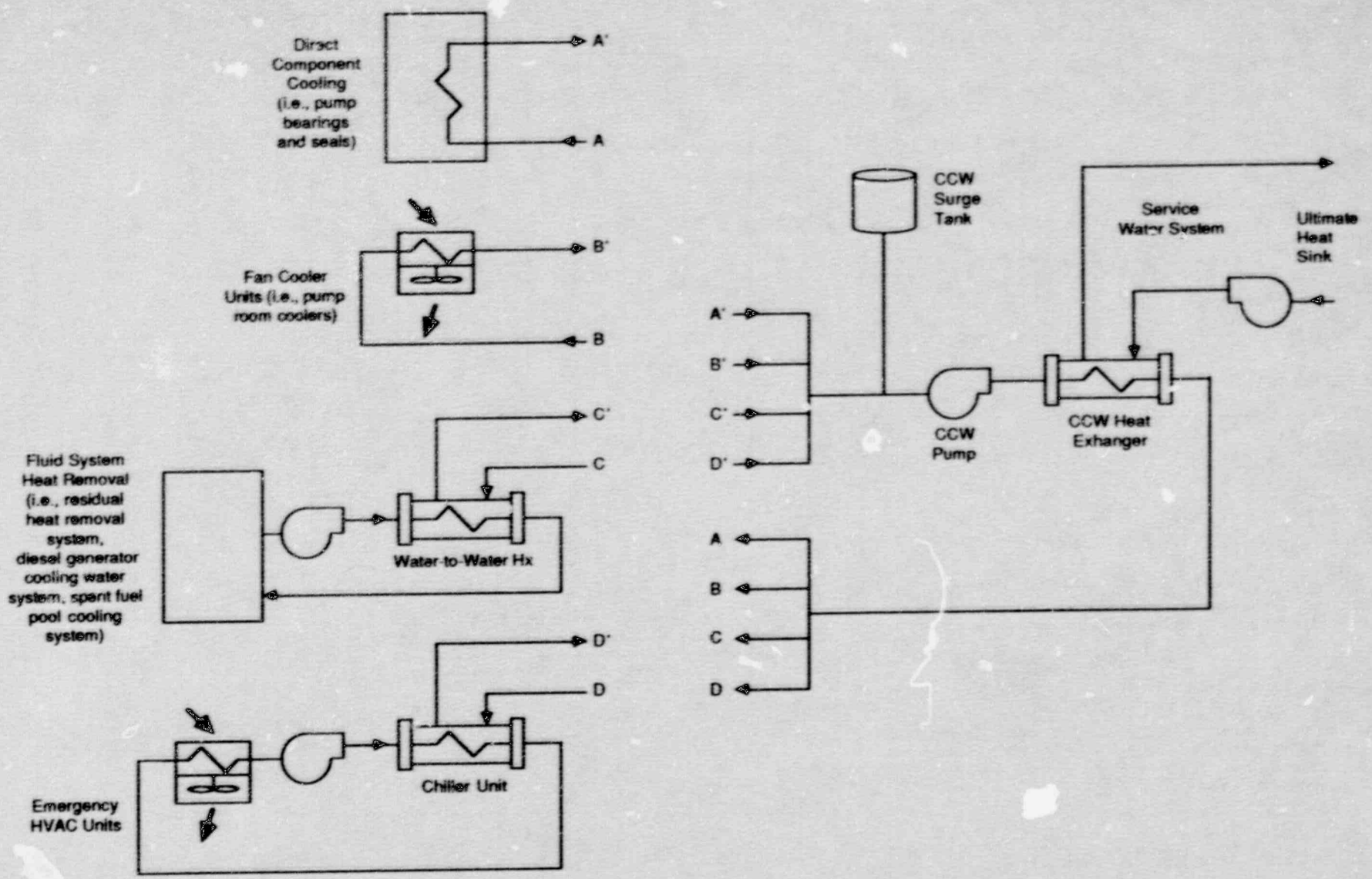
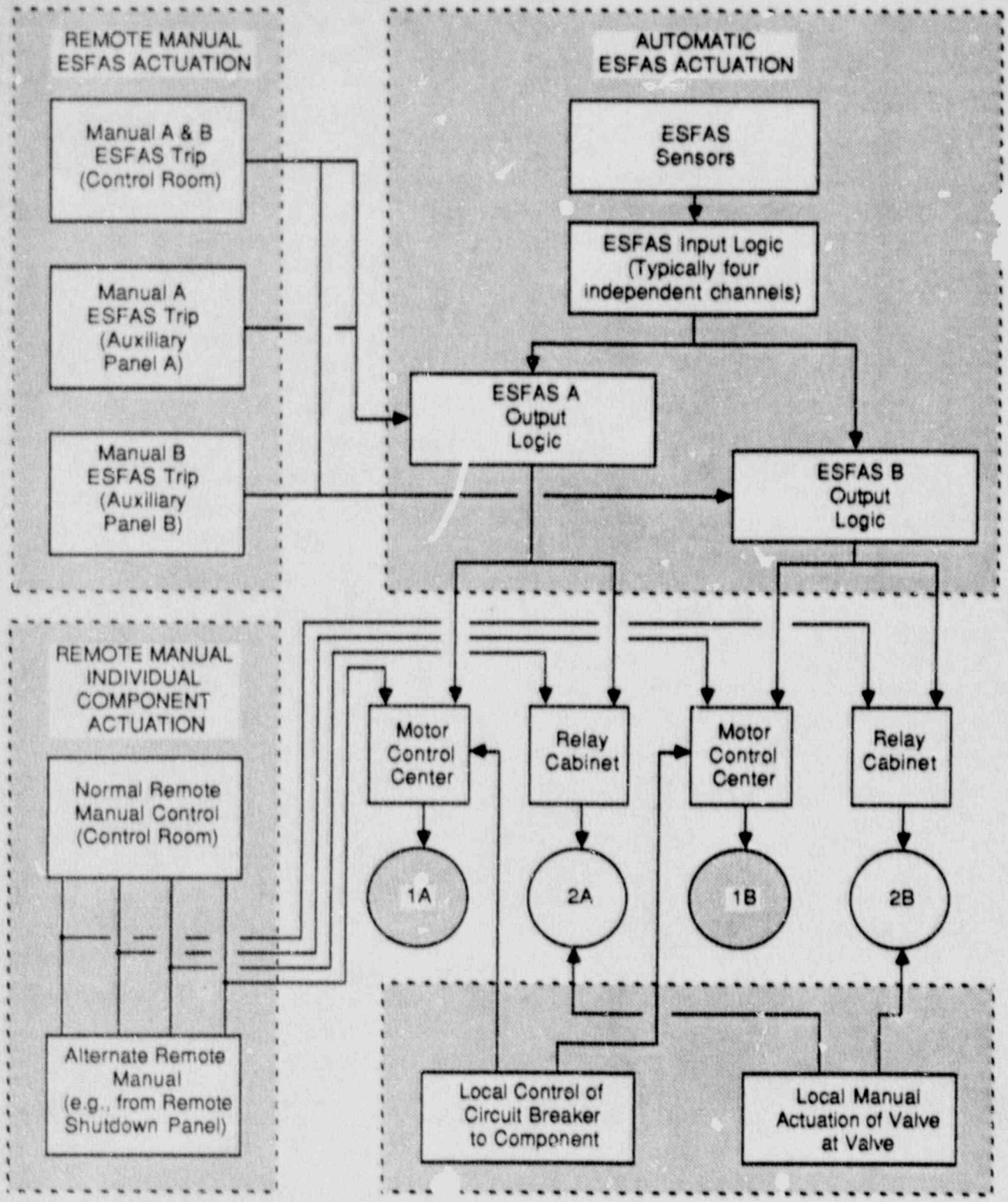


Figure 3.11-1. Examples of Component Cooling System Heat Transport Paths



Note: Components 1A and 1B are typical of motor-driven pumps and motor-operated valves.
Components 2A and 2B are typical of pneumatically or hydraulically-operated valves.

Figure 3.12-1. Example of Actuation System Interfaces

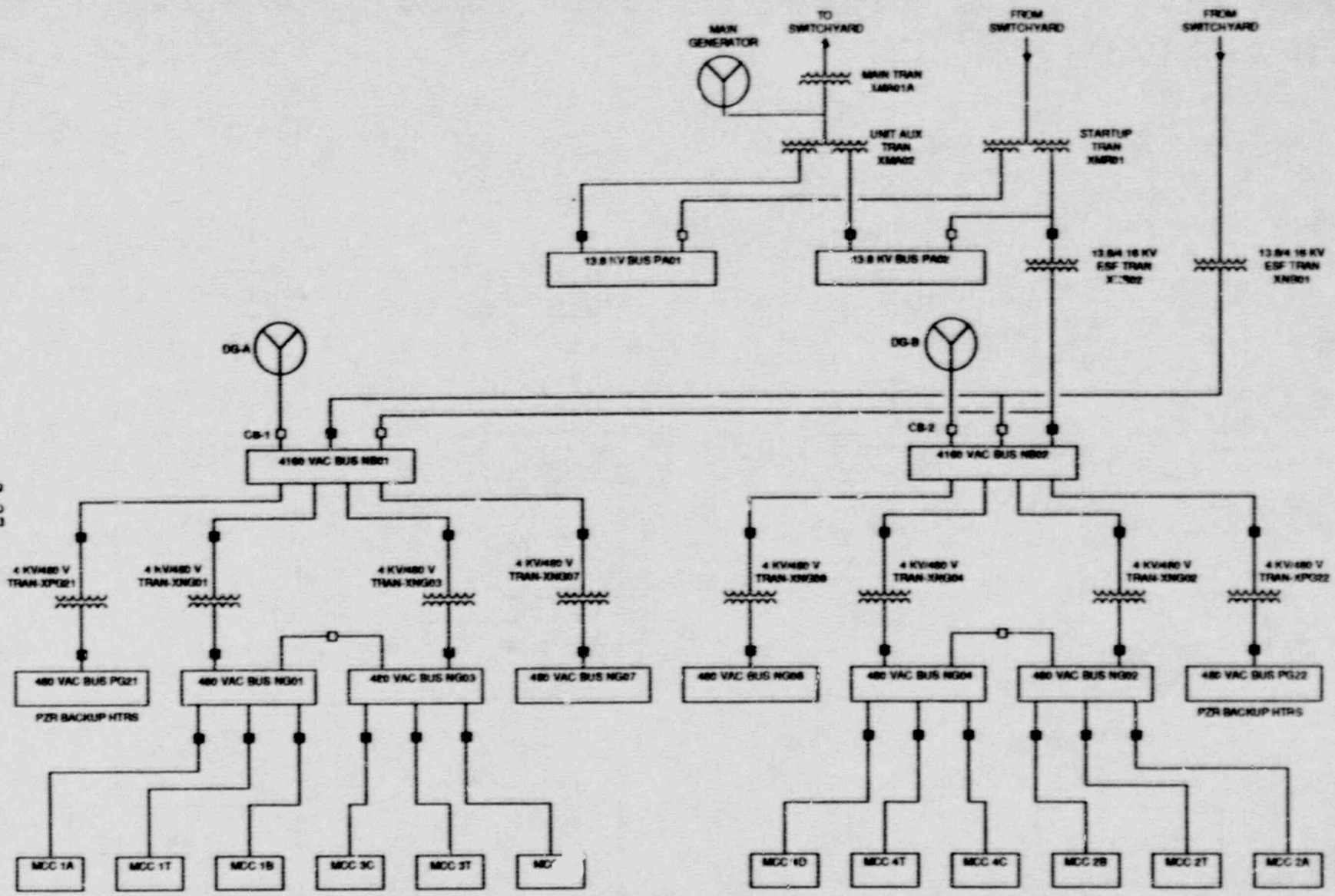


Figure 3.13-1. Representative PWR 4160 VAC and 480 VAC Electric Power Distribution System (Callaway)

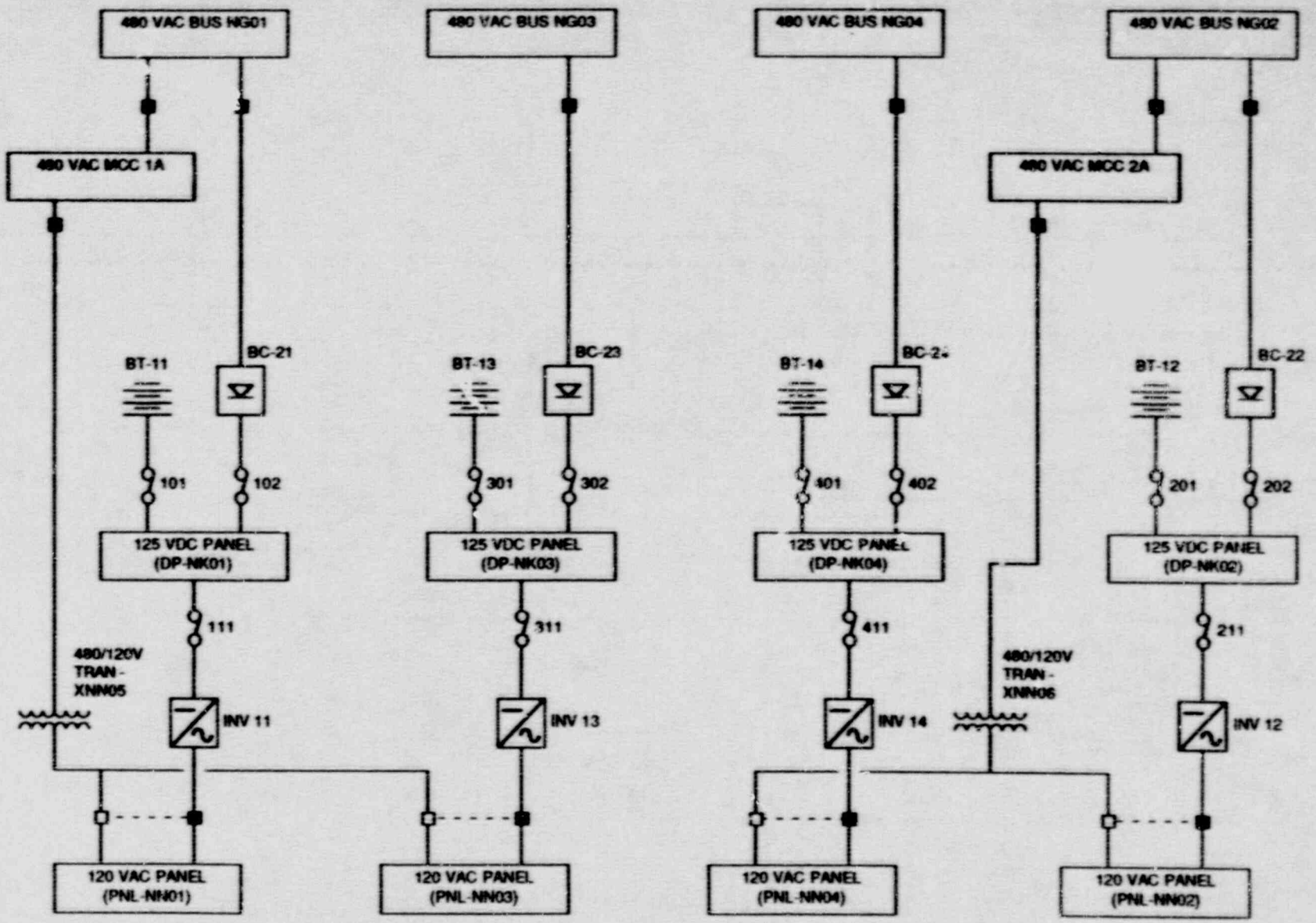


Figure 3.13-2. Representative PWR 125 VDC and 120 VAC Electric Power Distribution System (Callaway)

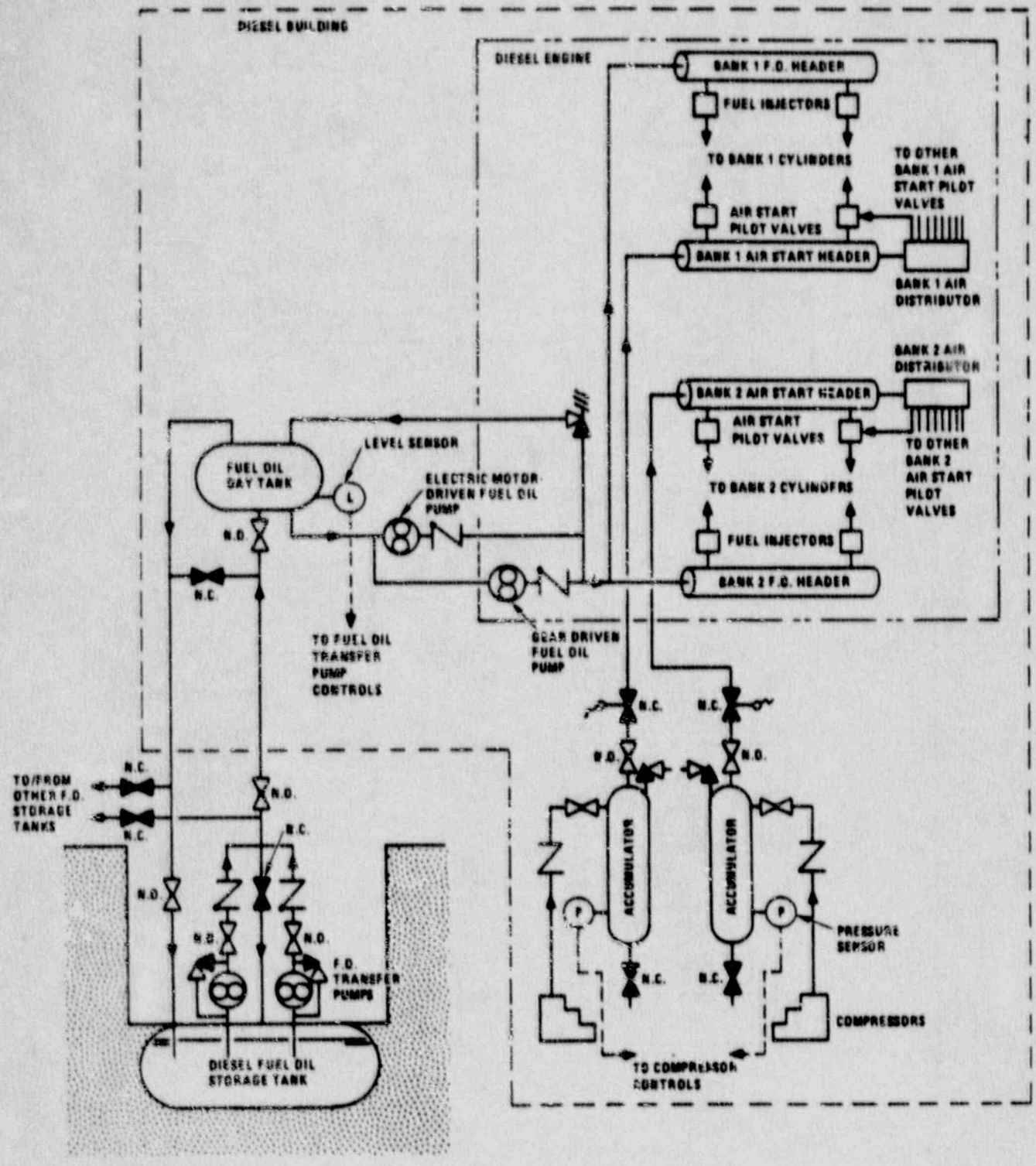


Figure 3.13-3. Diesel Fuel Oil and Air Starting Systems

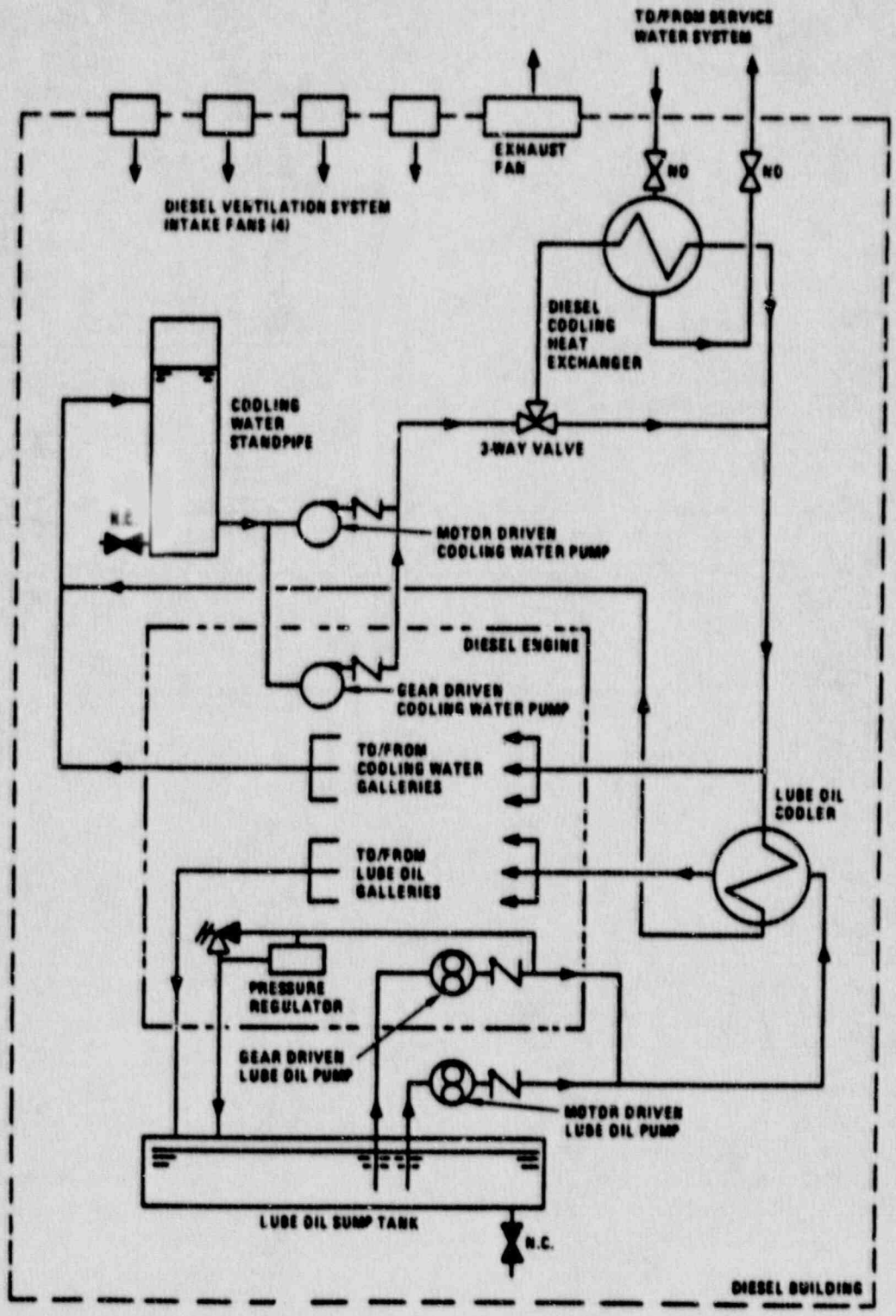


Figure 3.13-4. Diesel Lubrication, Cooling, and Room Ventilation Systems

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 1969. 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.

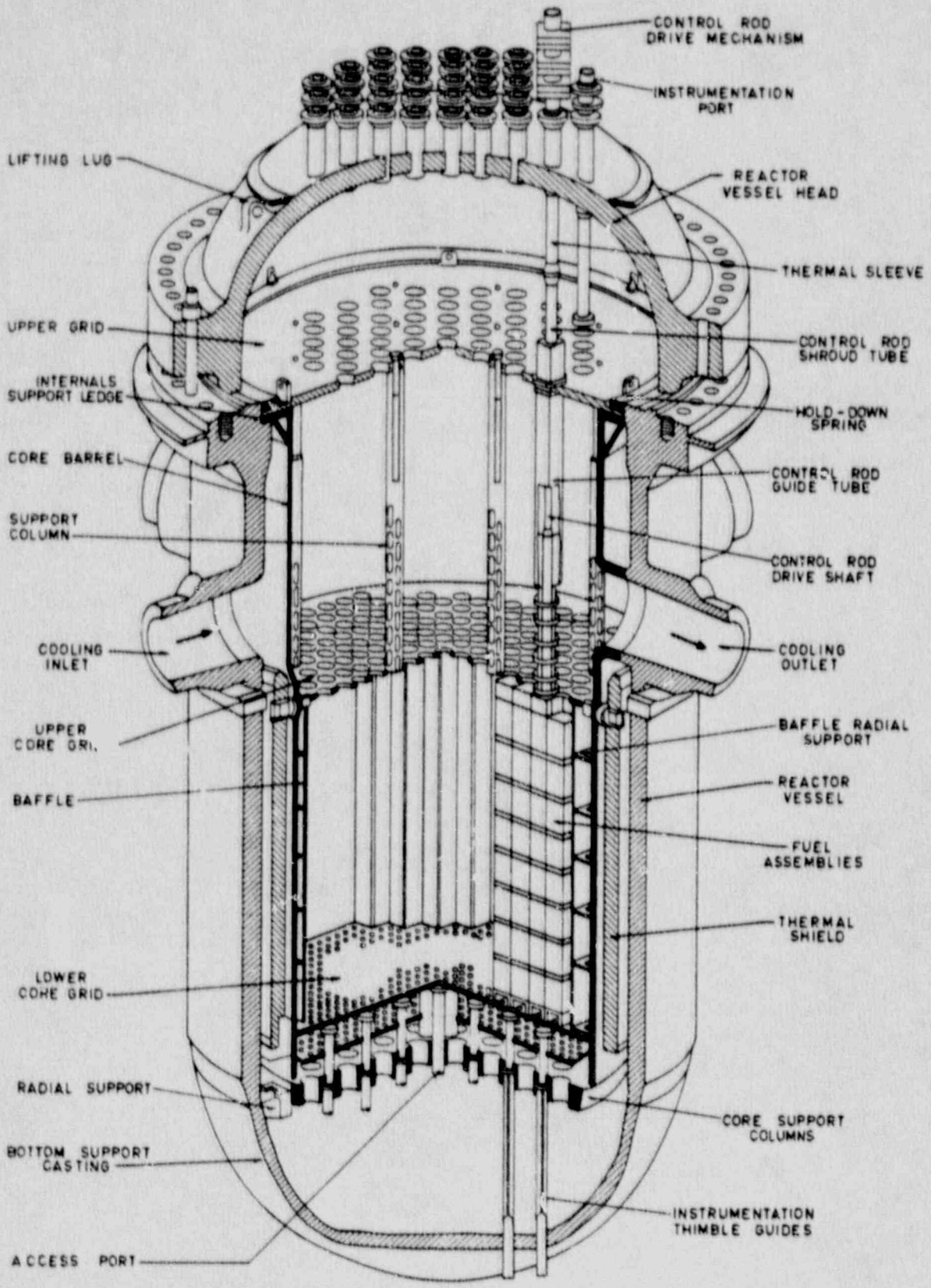


Figure 4.1-1. General Arrangement of a Westinghouse Reactor Vessel and Internals

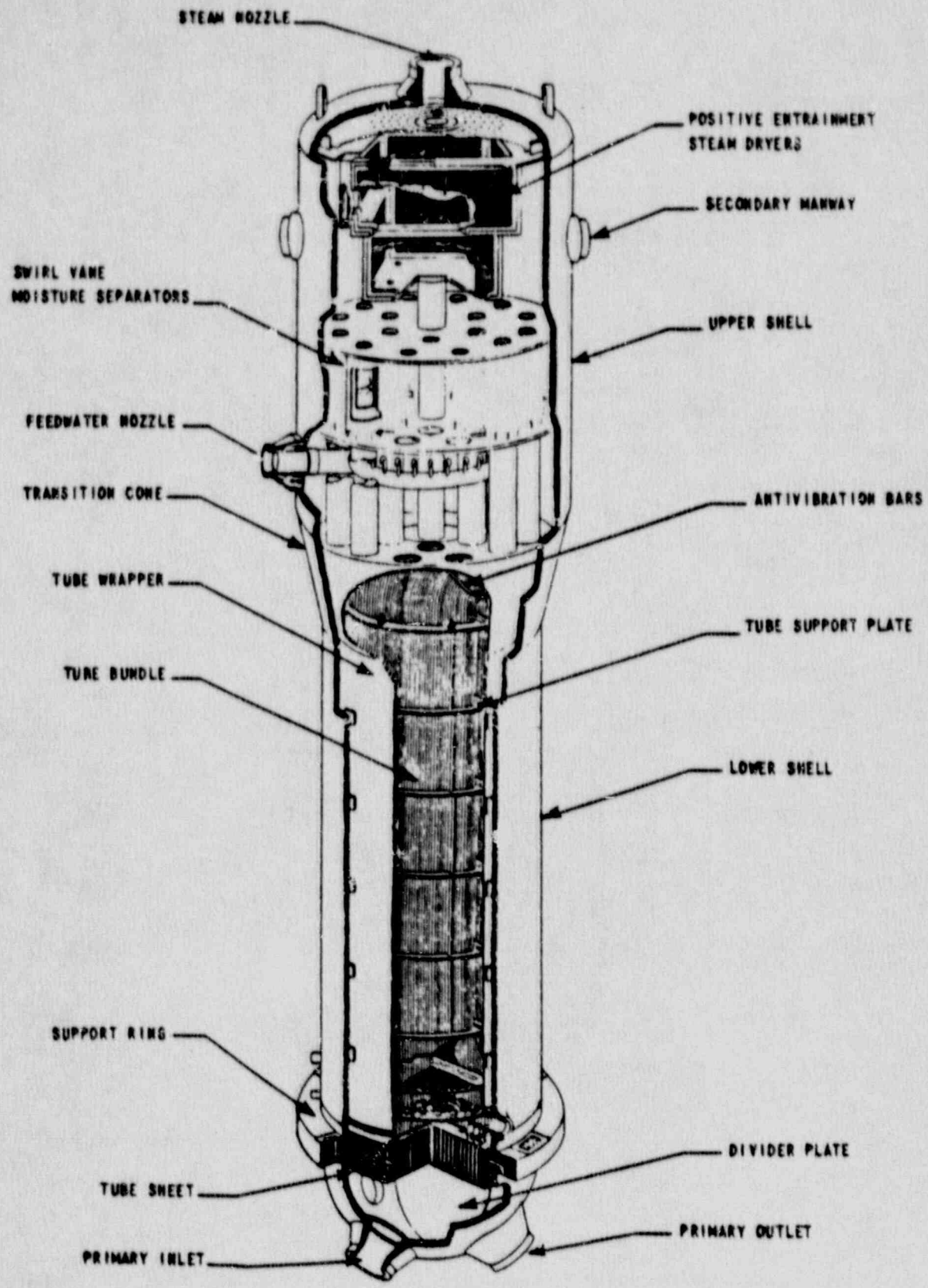


Figure 4.1-2. Westinghouse U-Tube Steam Generator (Typical of Series 44 and 51 and Model F)

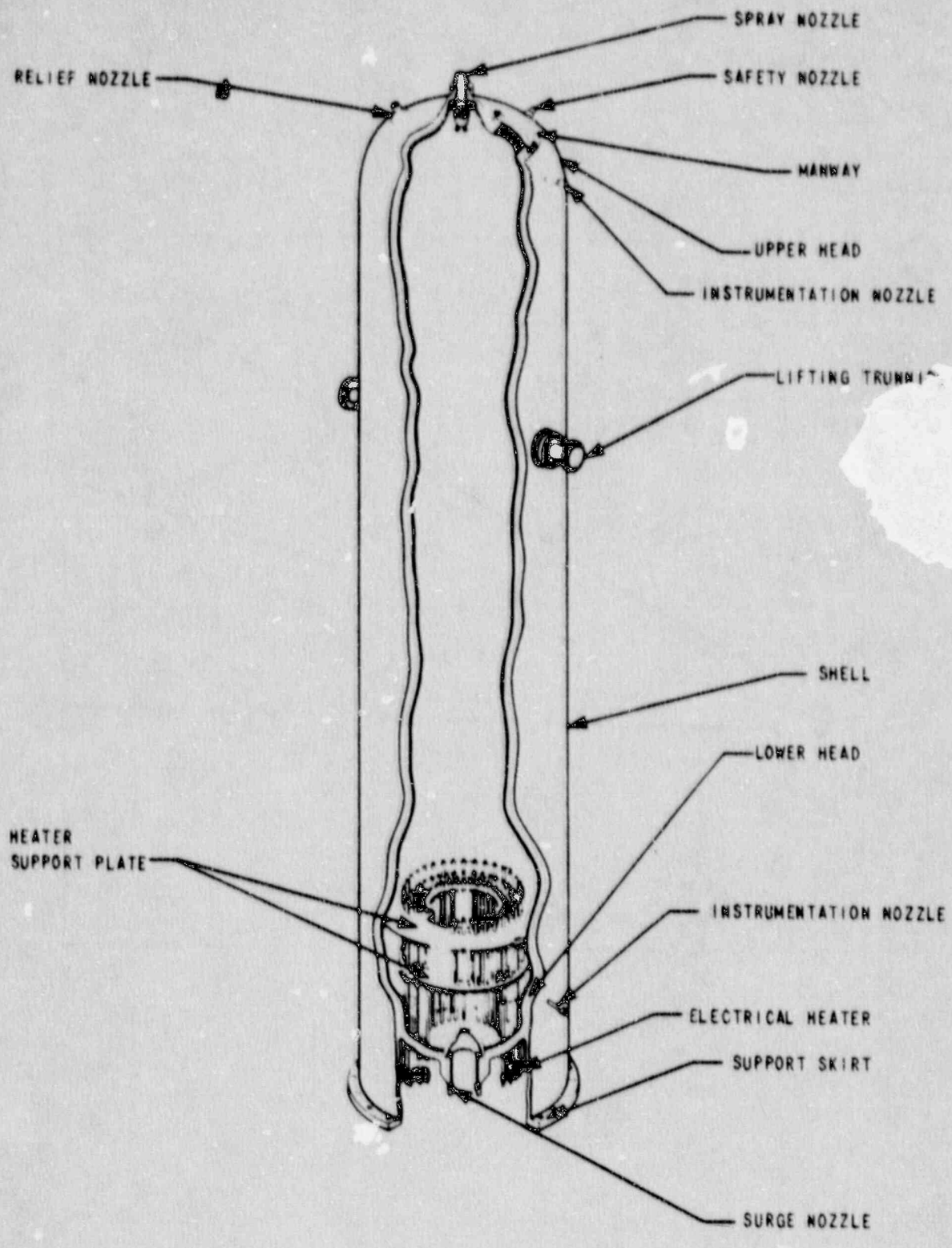


Figure 4.1-3. Typical Westinghouse Pressurizer

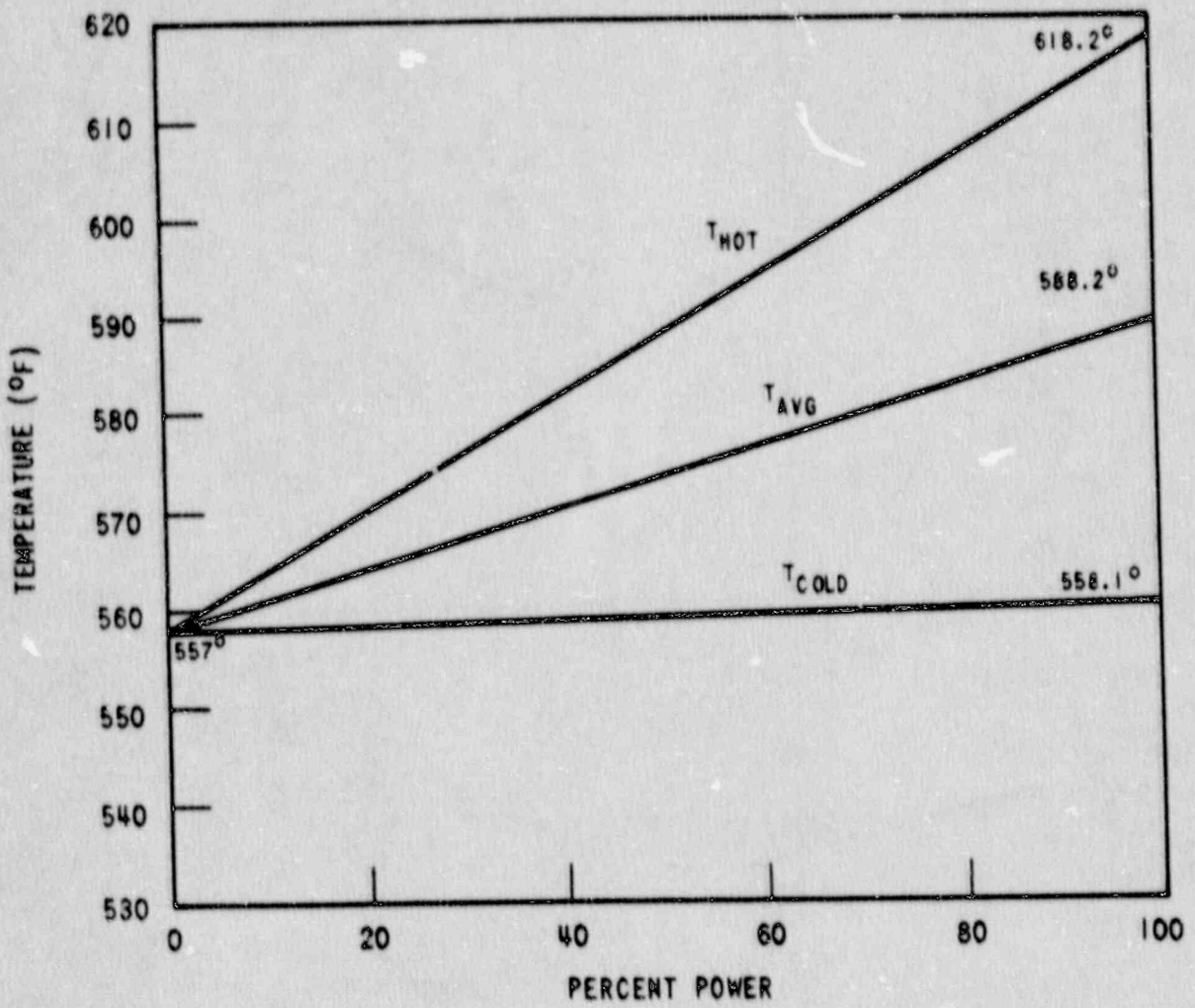
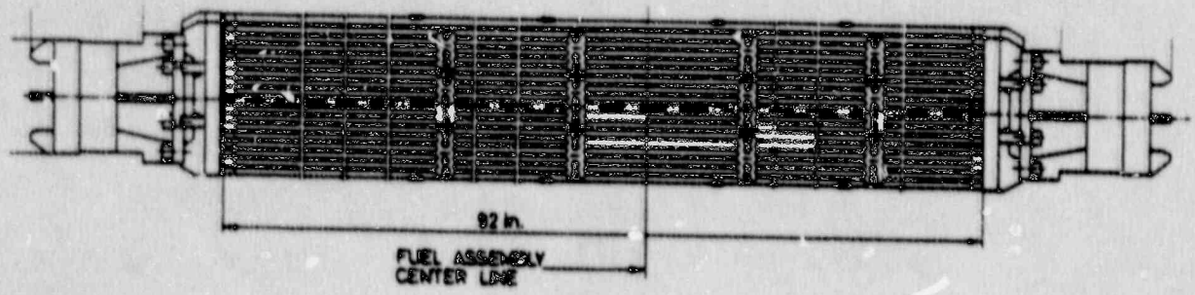


Figure 4.1-4. Typical Westinghouse RCS Temperature Control Program



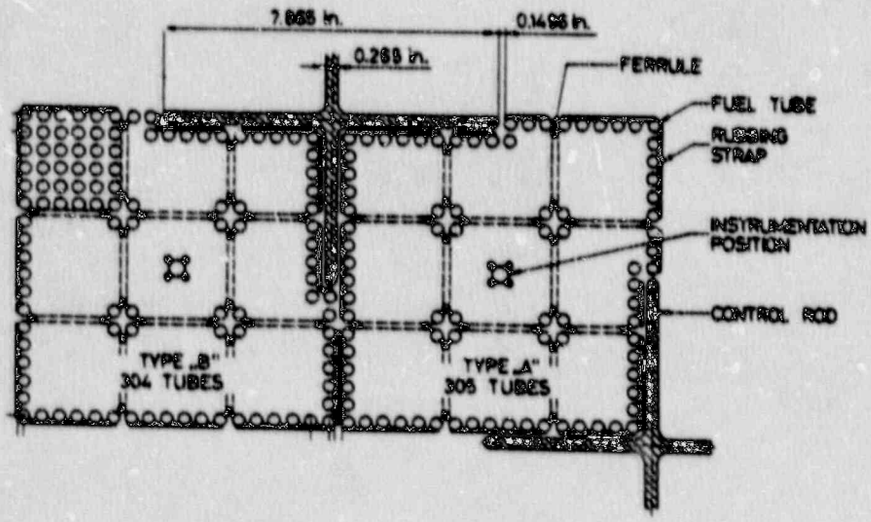


Figure 4.1-5. General Arrangement of the Yankee-Rowe 9 X (6 X 6) Fuel Assembly

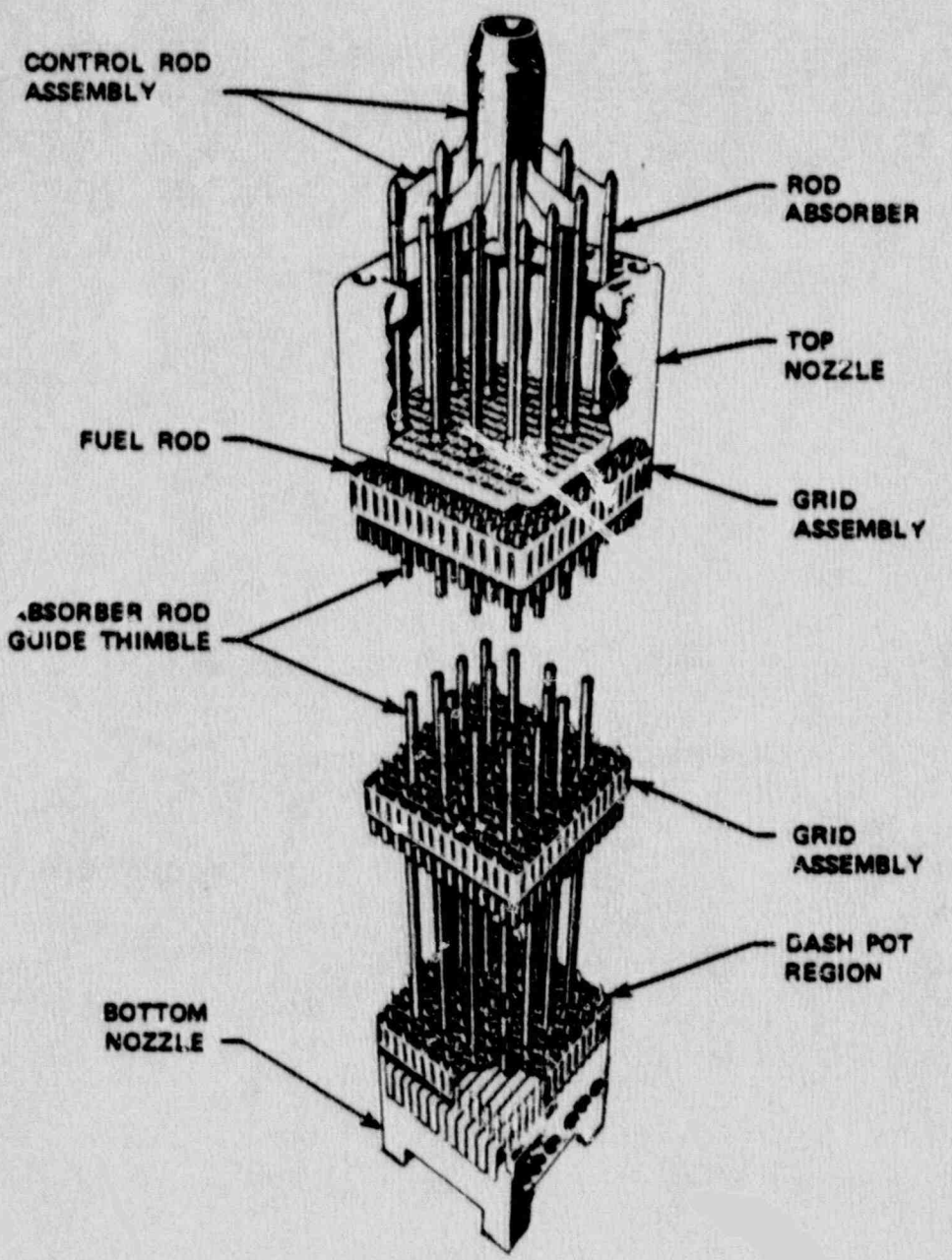


Figure 4.1-6. Westinghouse 14 X 14 Fuel Assembly and 16-Element Rod Cluster Control Assembly

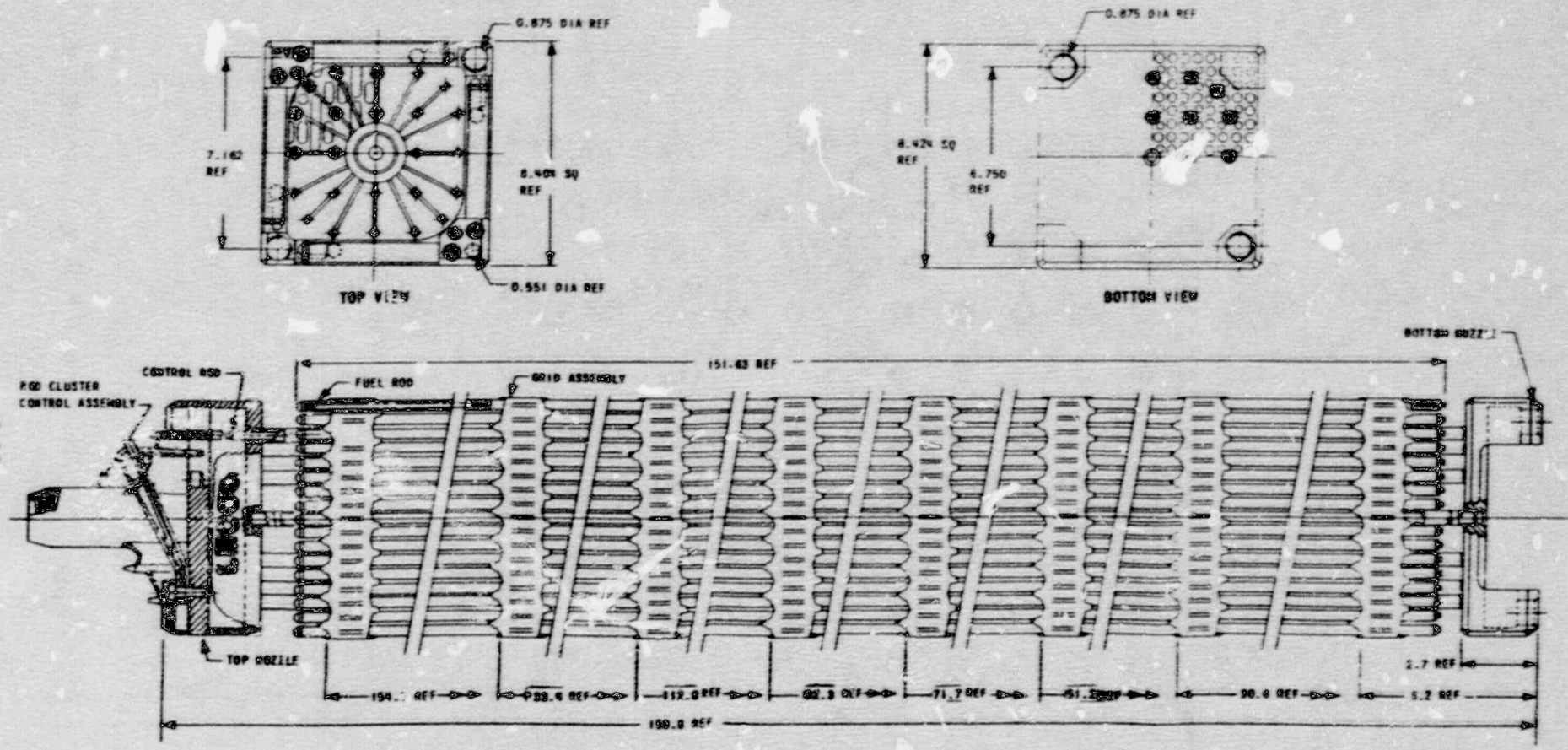


Figure 4.1-7. Westinghouse 17 X 17 Fuel Assembly Longitudinal View

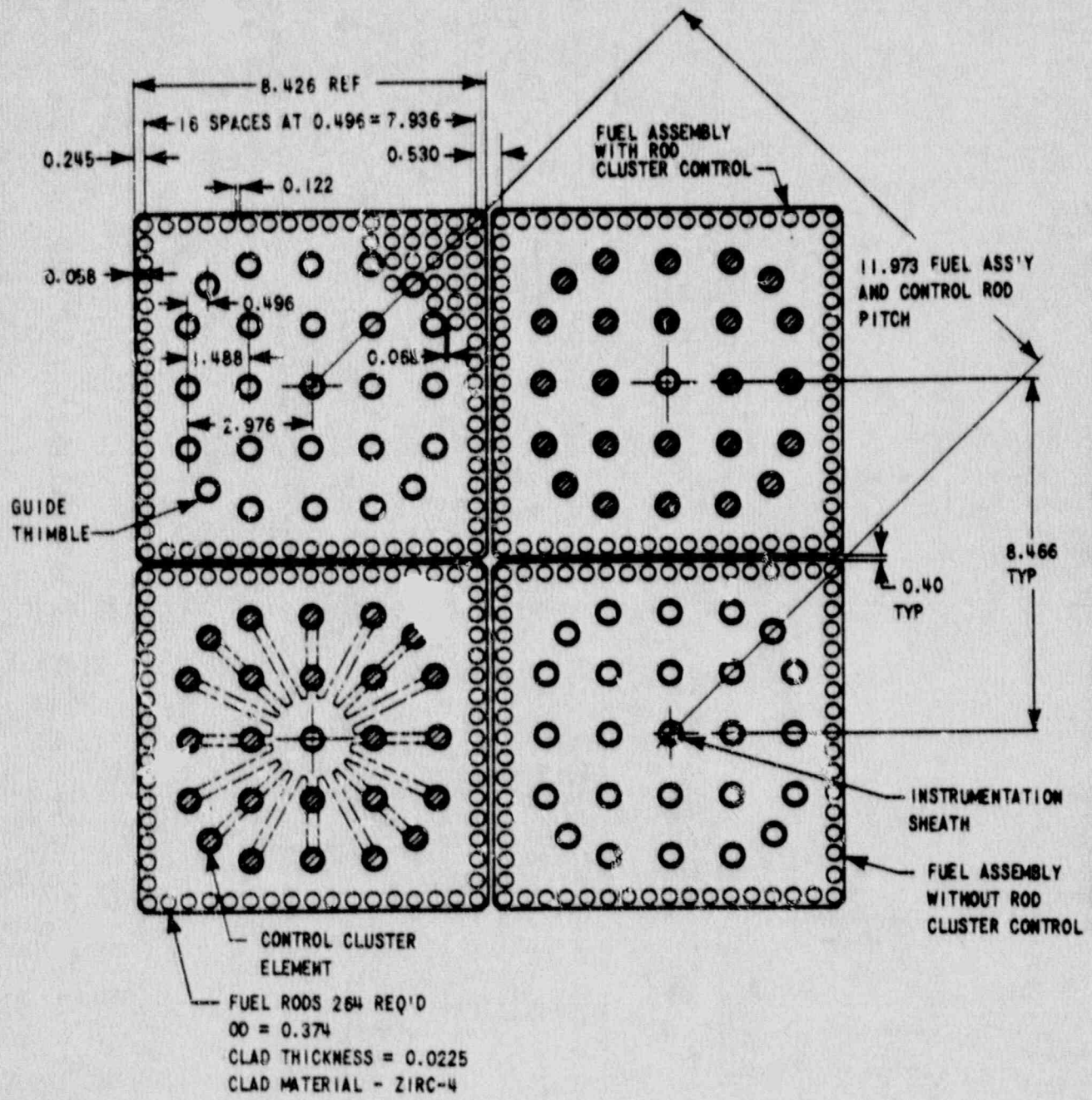


Figure 4.1-8. Westinghouse 17 X 17 Fuel Assembly, Cross-Section View

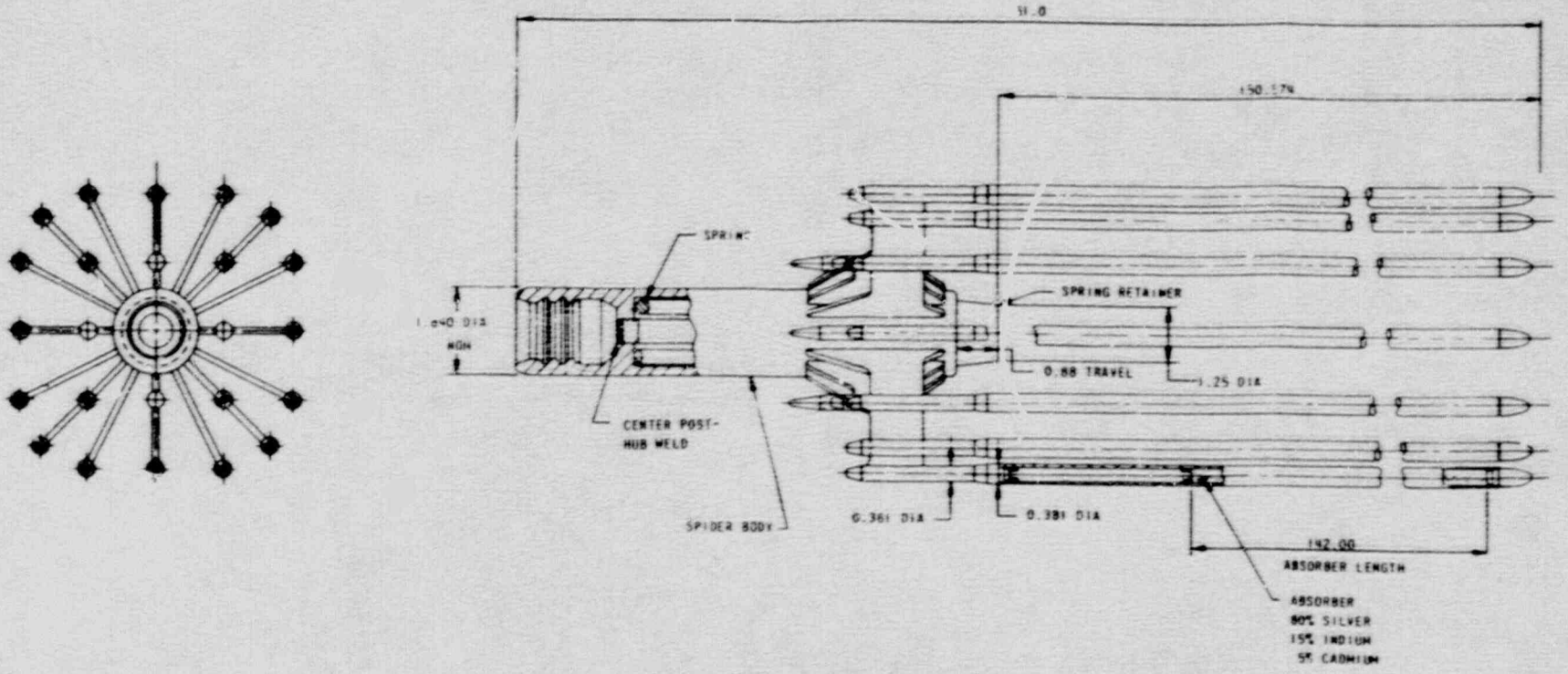


Figure 4.1-9. Westinghouse Rod Cluster Control Assembly (RCCA, Typical for 17 X 17 Fuel Assembly)

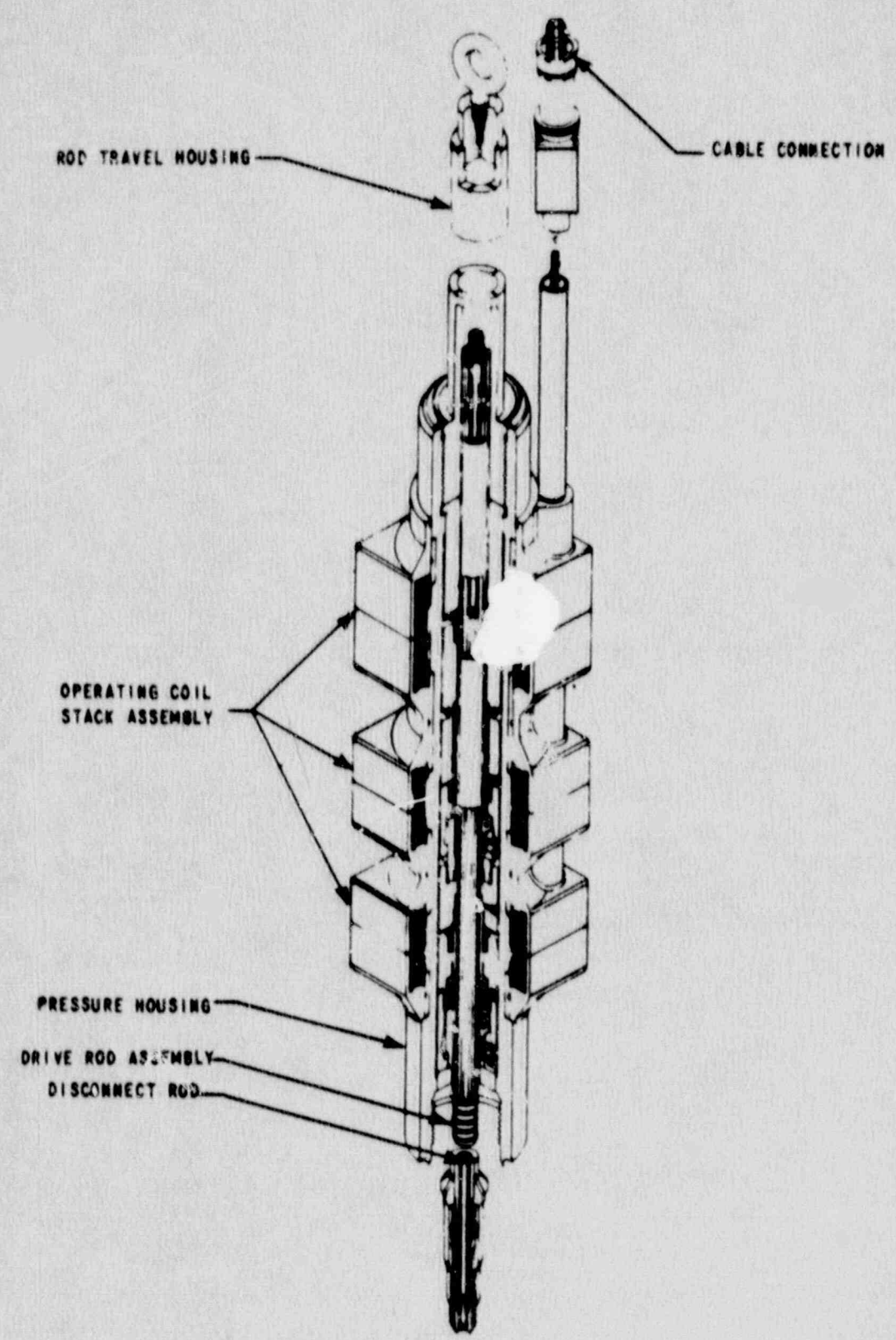


Figure 4.1-10. Details of a Westinghouse Magnetic Jack Full-Length Control Rod Drive Mechanism

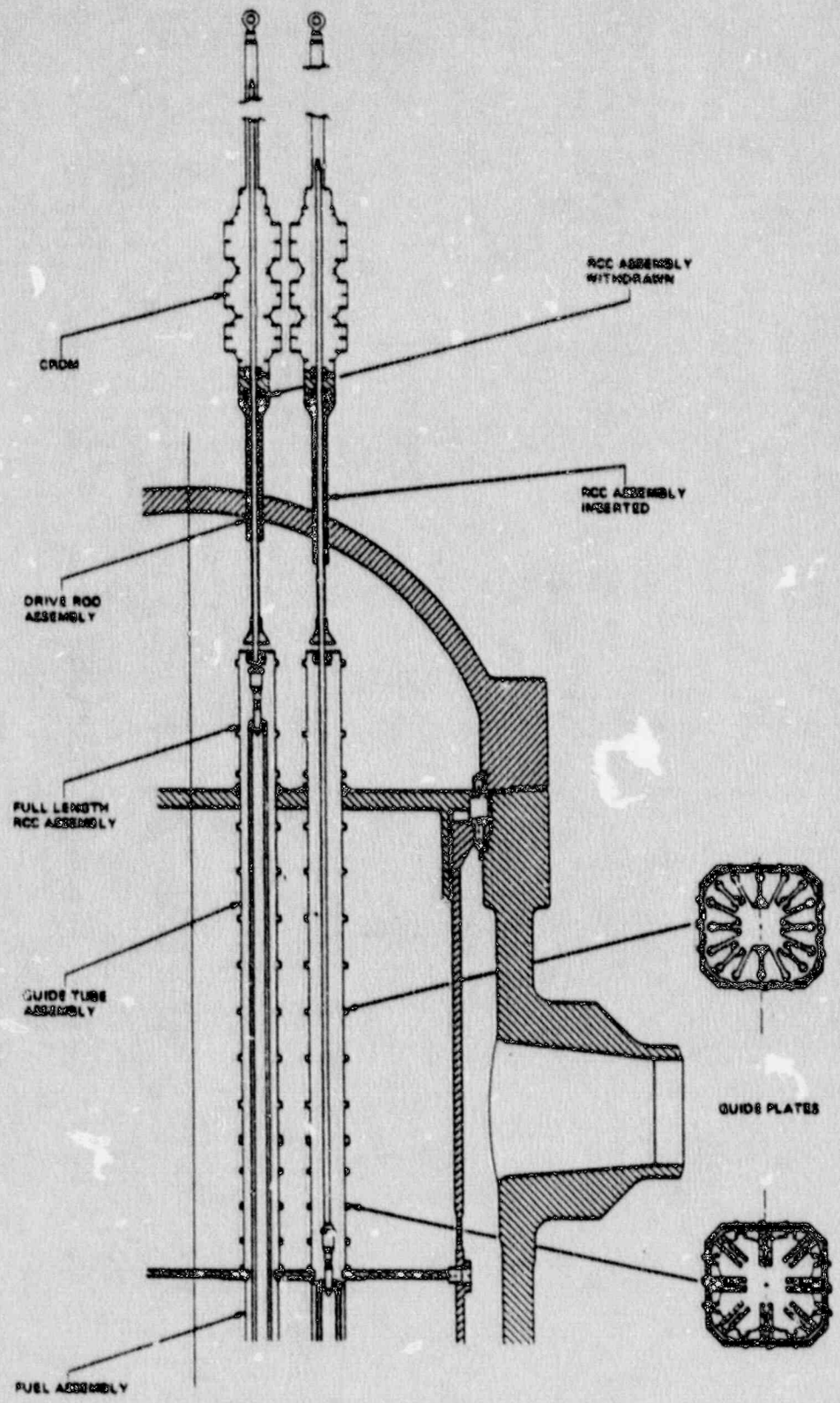


Figure 4.1-11. General Arrangement of Westinghouse CRDMs and RCCAs

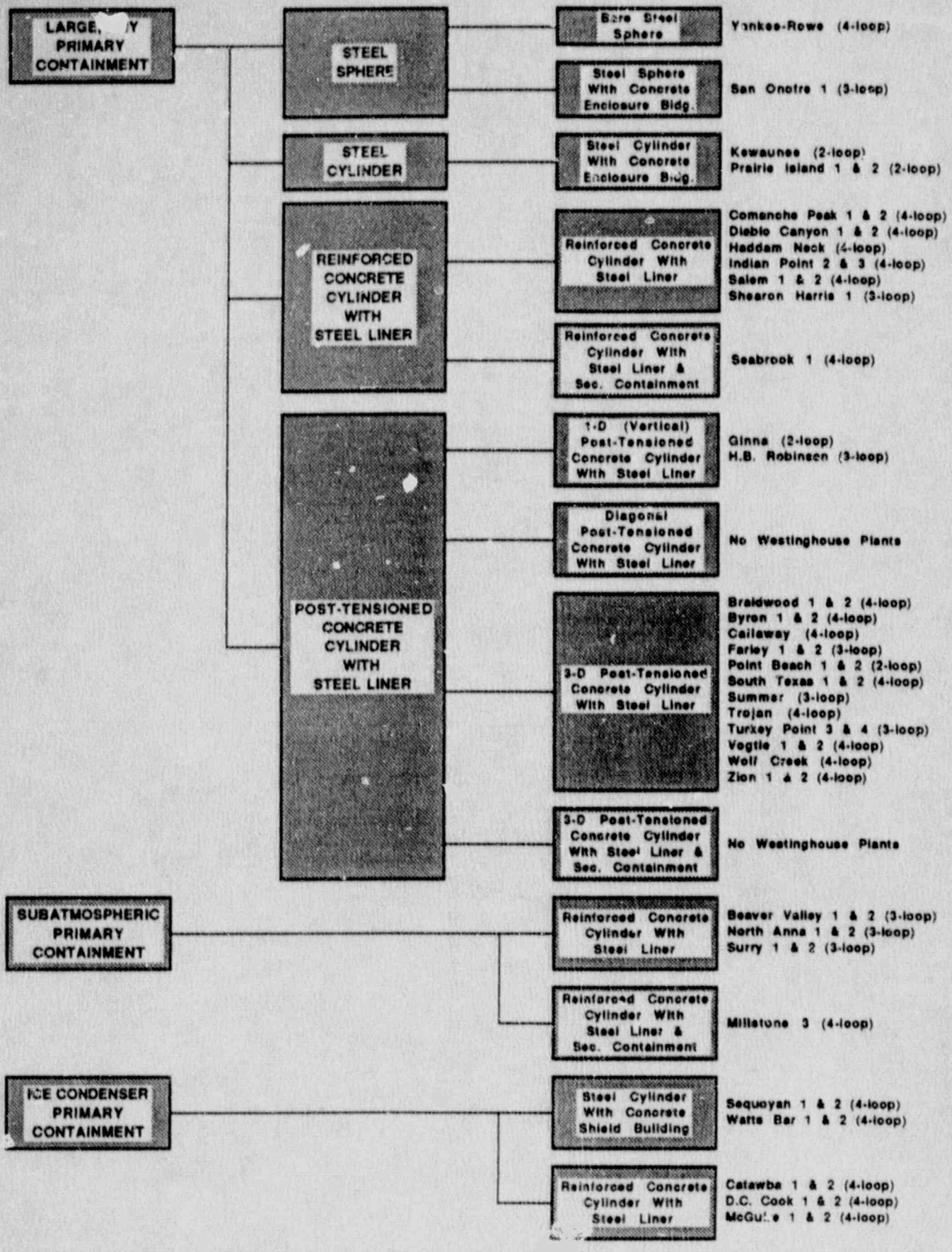


Figure 4.1-12. Distribution of Containment Types for Westinghouse Reactors

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).

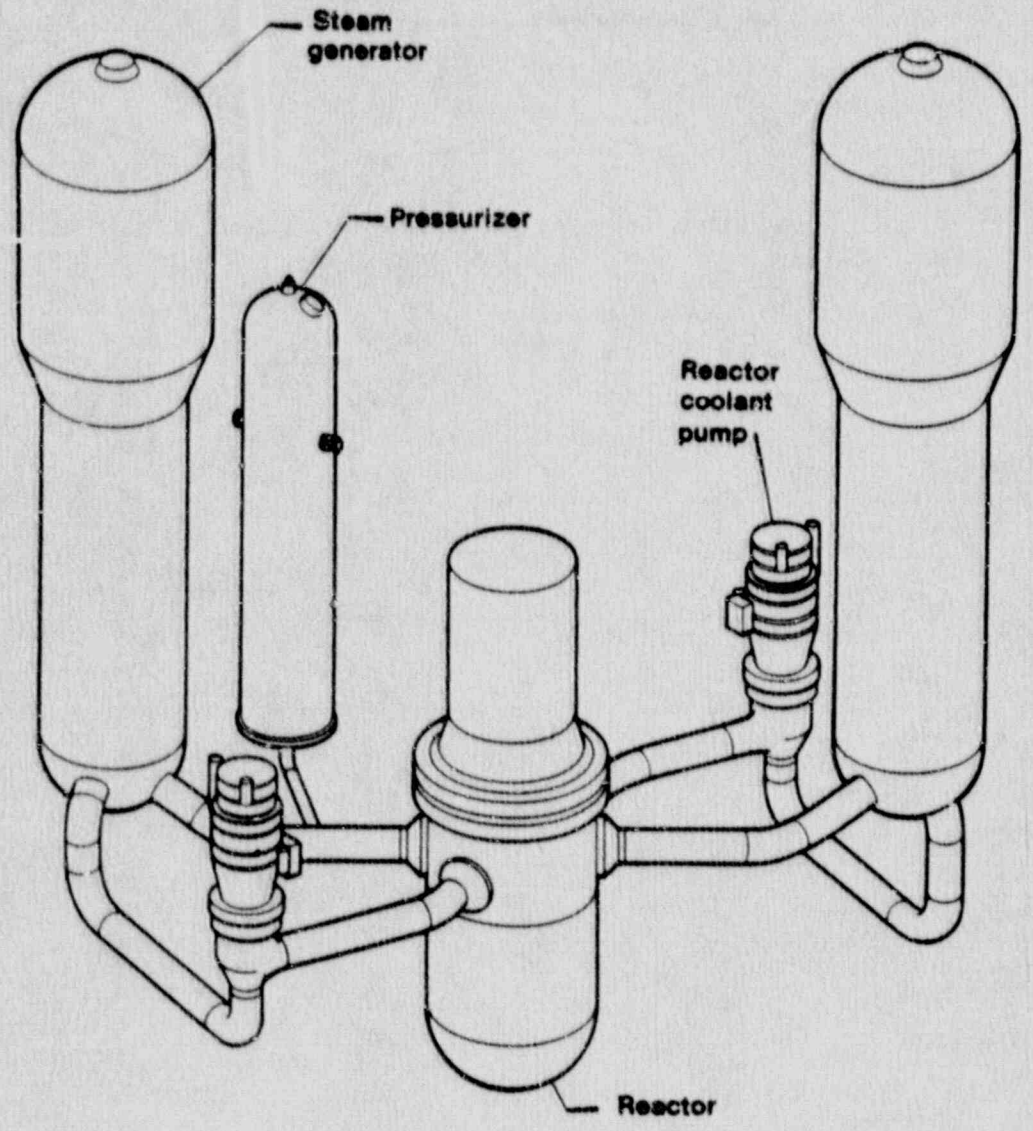


Figure 4.2-1. Westinghouse 2-Loop NSSS

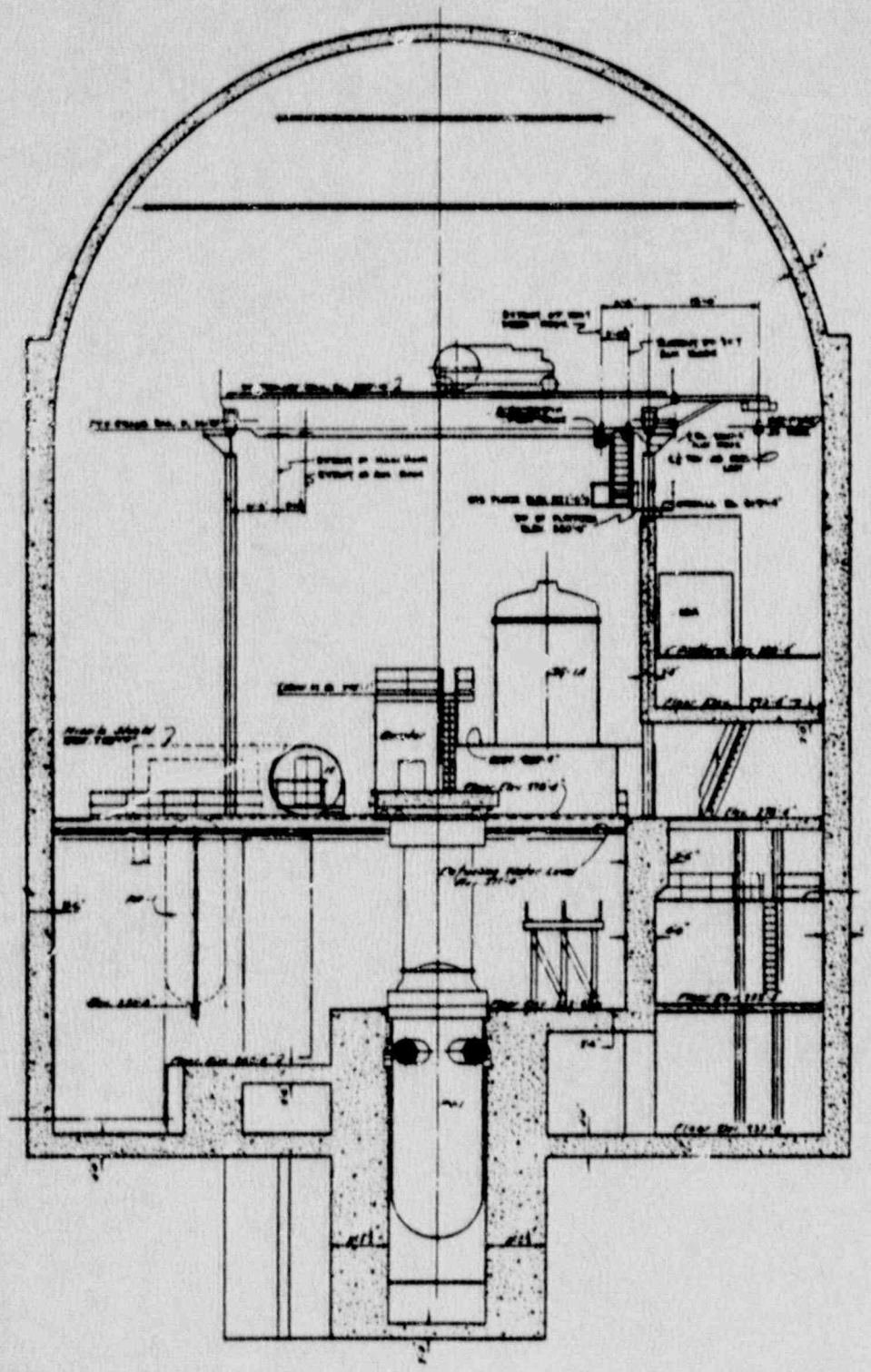


Figure 4.2-2. Section Views of the Ginna Large, Dry Containment (Sheet 1 of 2)

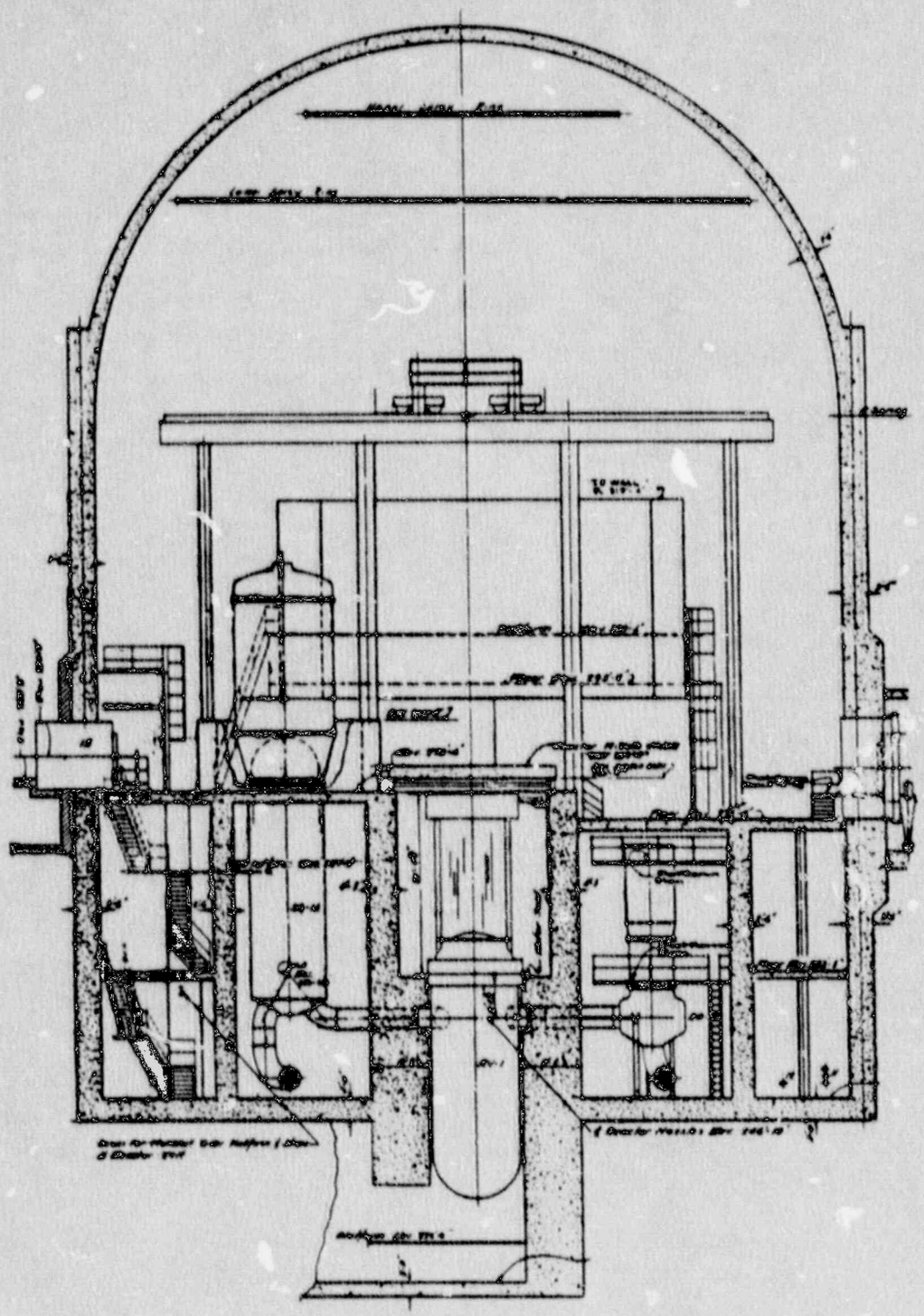


Figure 4.2-2. Section Views of the Ginna Large, Dry Containment (Sheet 2 of 2)

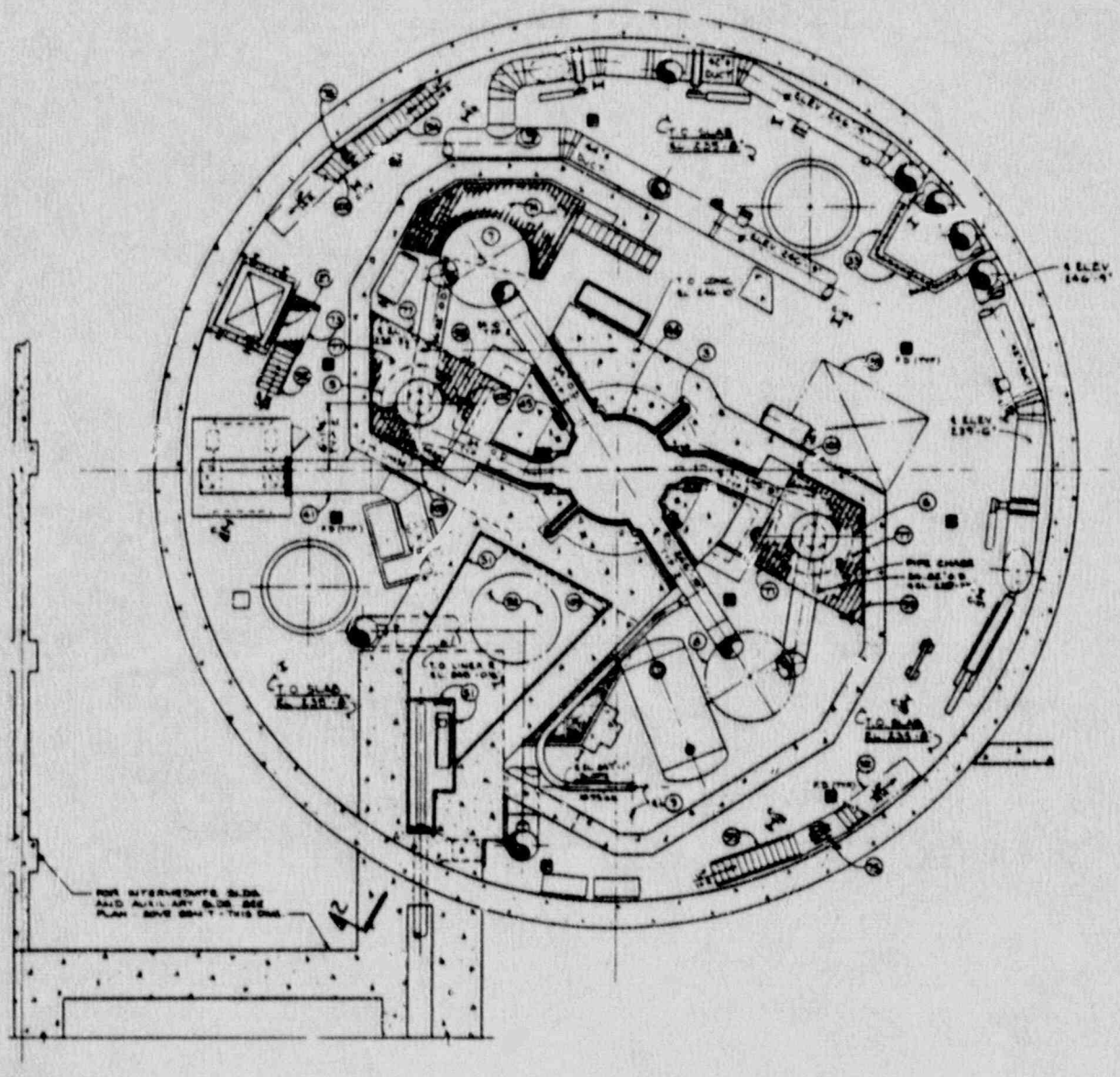


Figure 4.2-3. Plan View of the Ginna Large, Dry Containment Below the Elevation of the Operating Floor

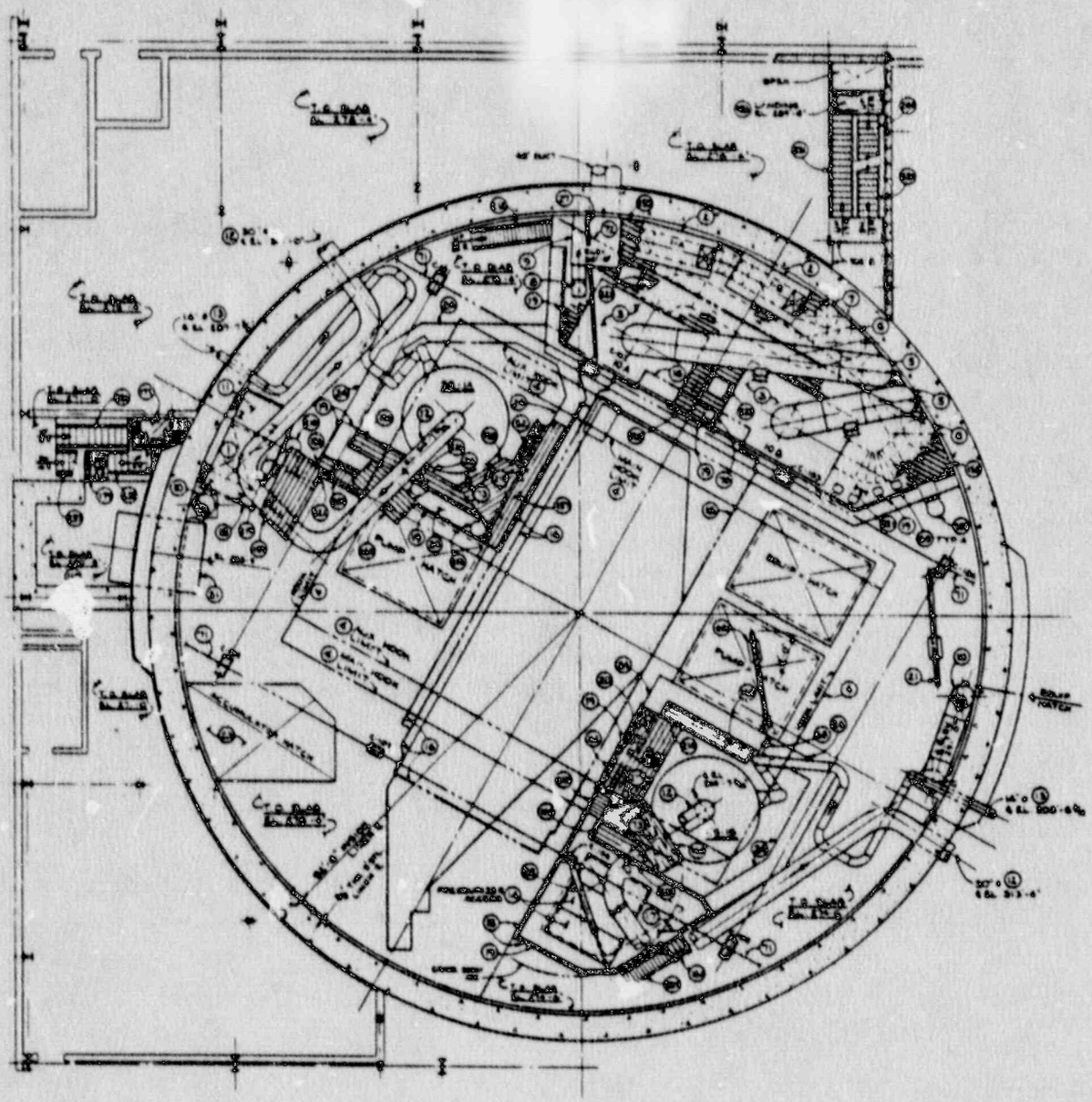


Figure 4.2-4. Plan View of the Ginna Large, Dry Containment, Above the Operating Floor

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.

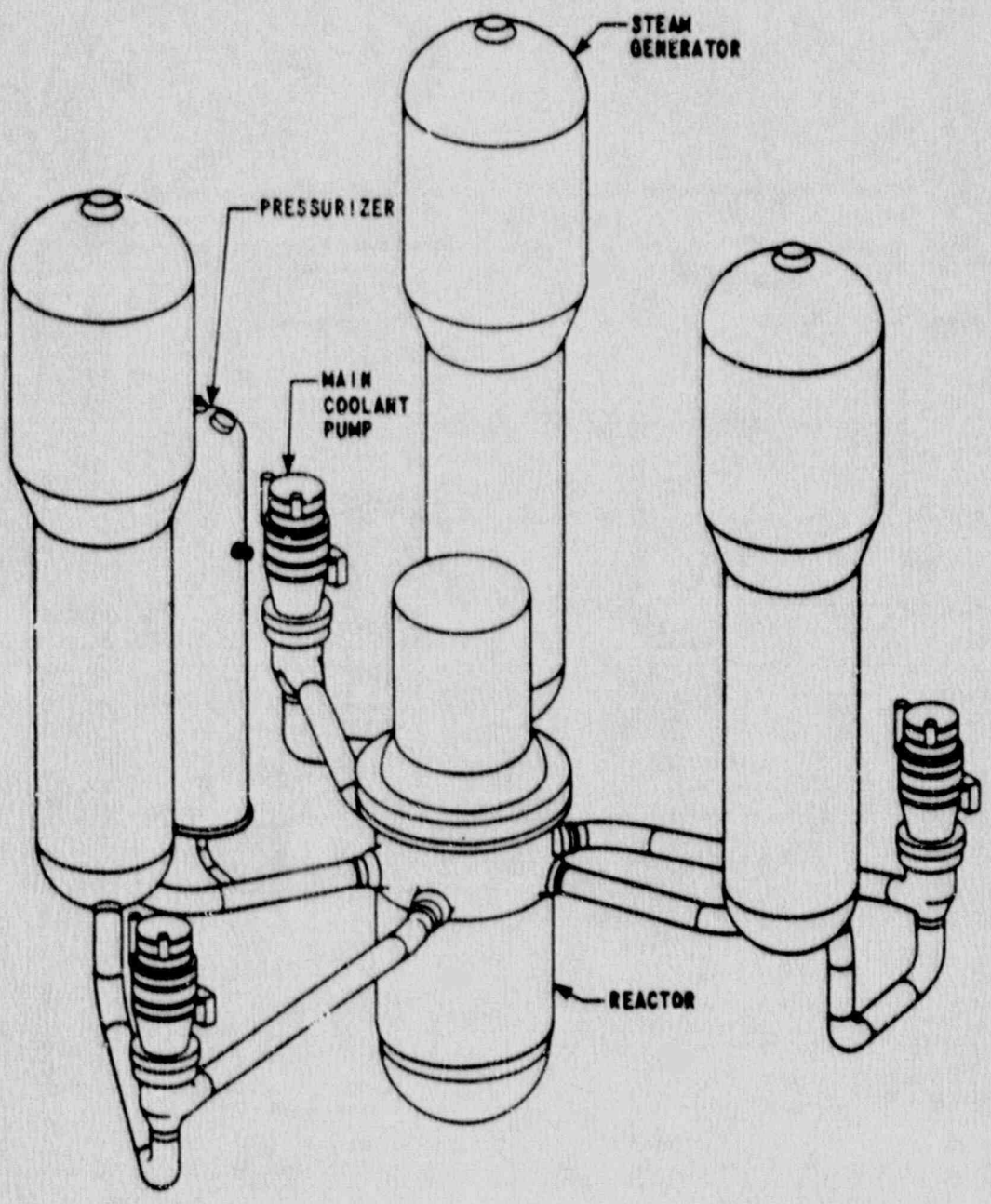


Figure 4.3-1. Westinghouse 3-Loop NSSS

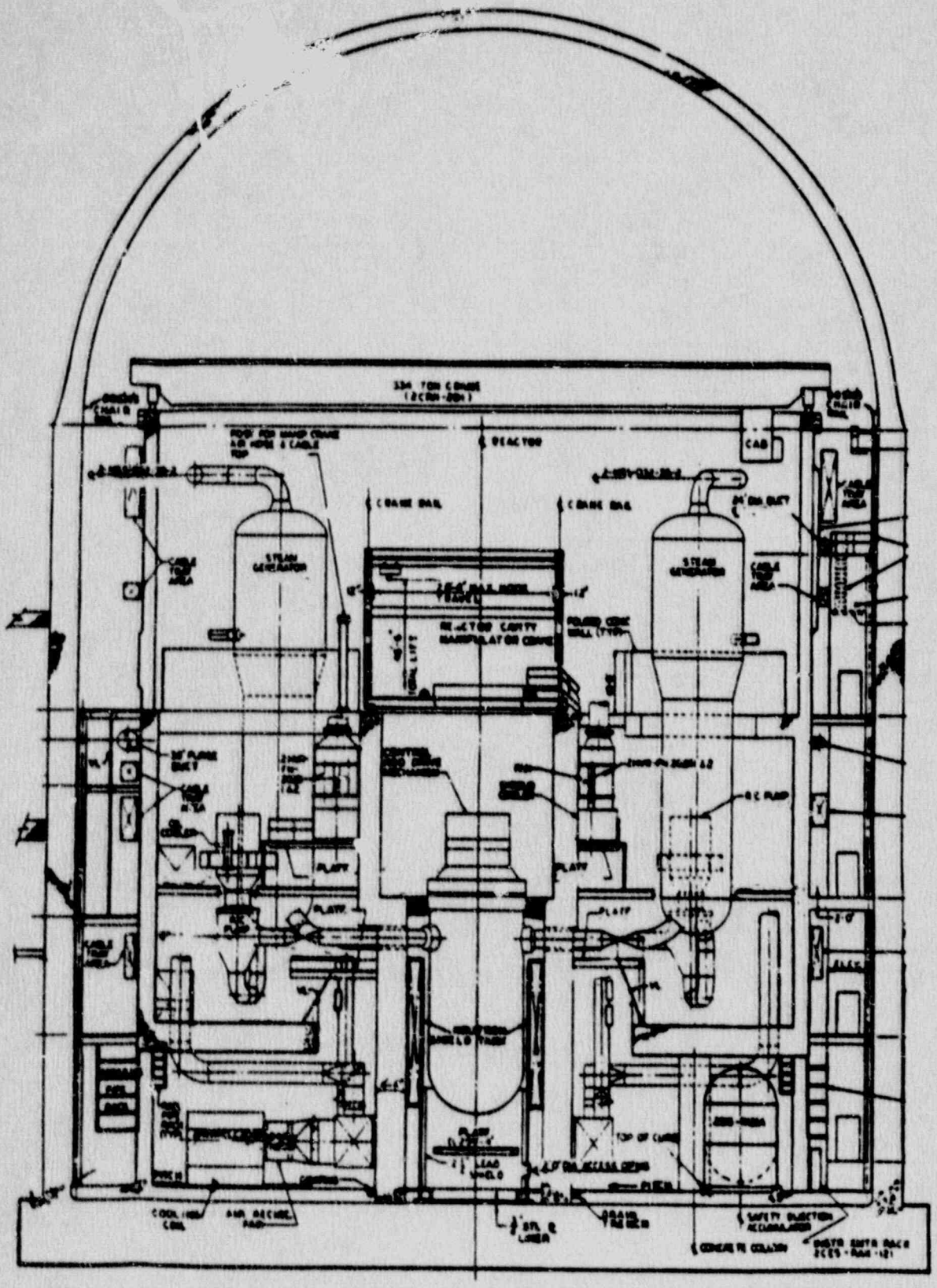


Figure 4.3-2. Section View of the H.B. Robinson Large, Dry Containment (1-D Post-Tensioned Concrete)

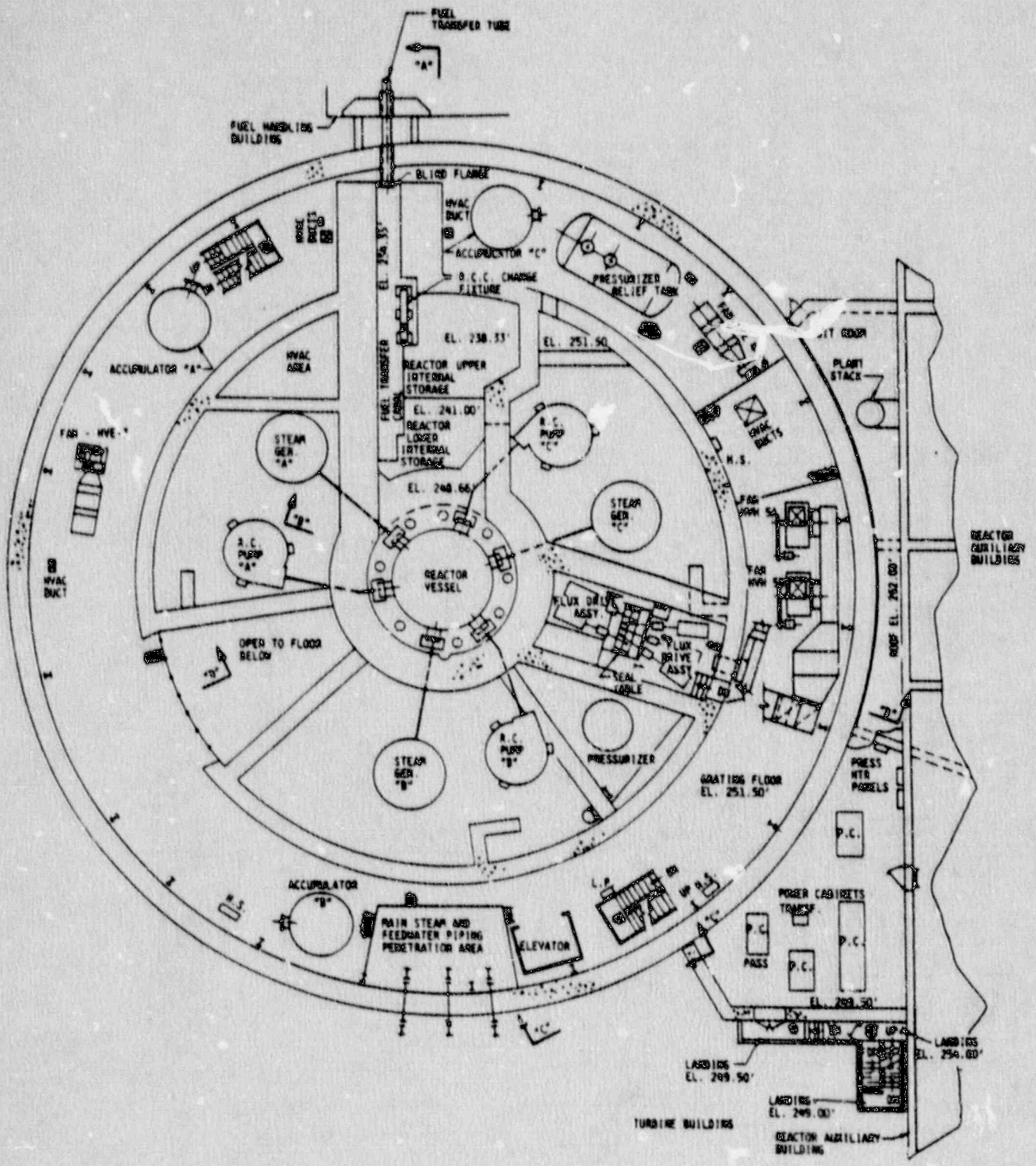


Figure 4.3-3. Plan View of the H.B. Robinson Large, Dry Containment (1-D Post-Tensioned Concrete)

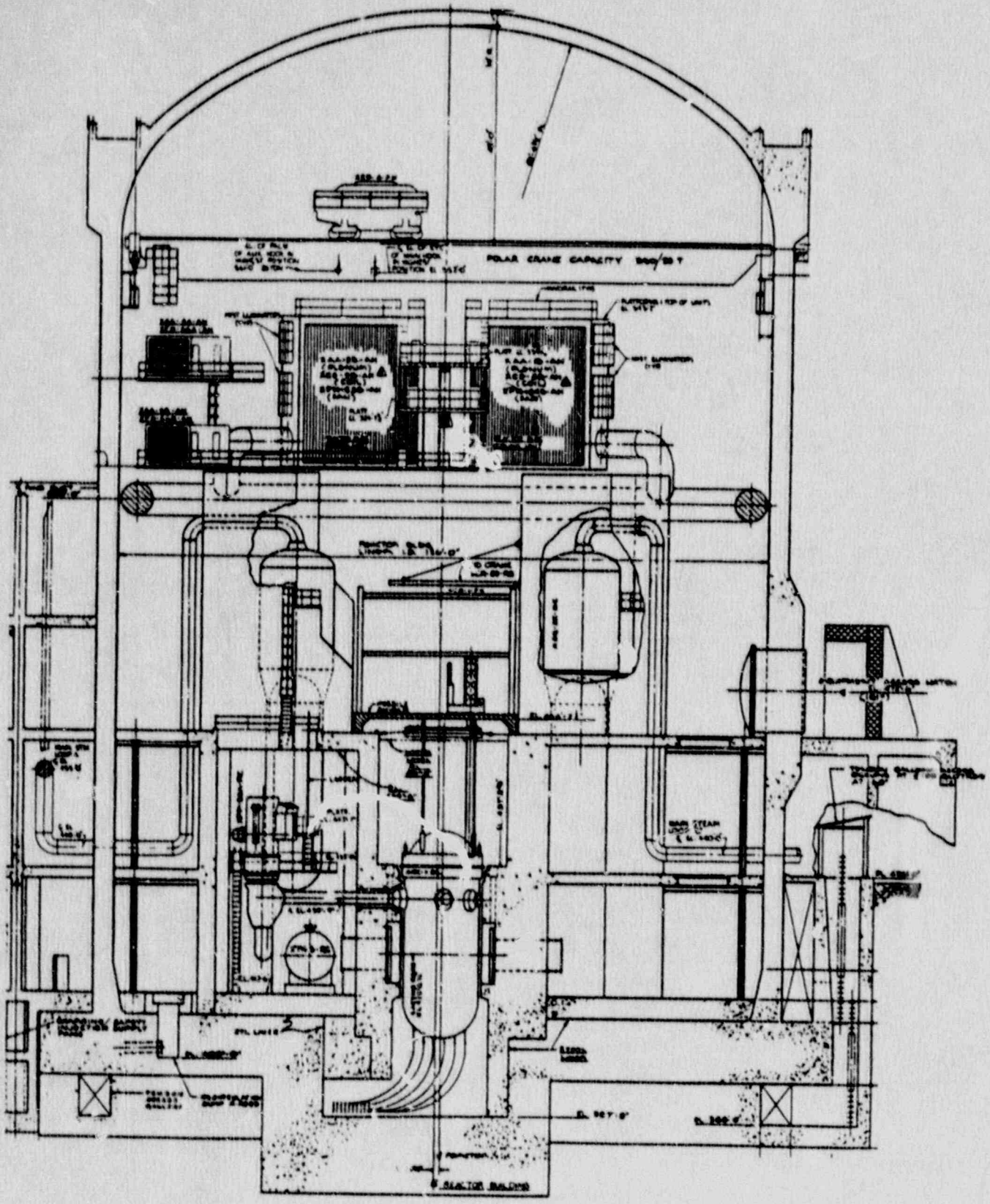


Figure 4.3-4. Section Views of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete) (Sheet 1 of 2)

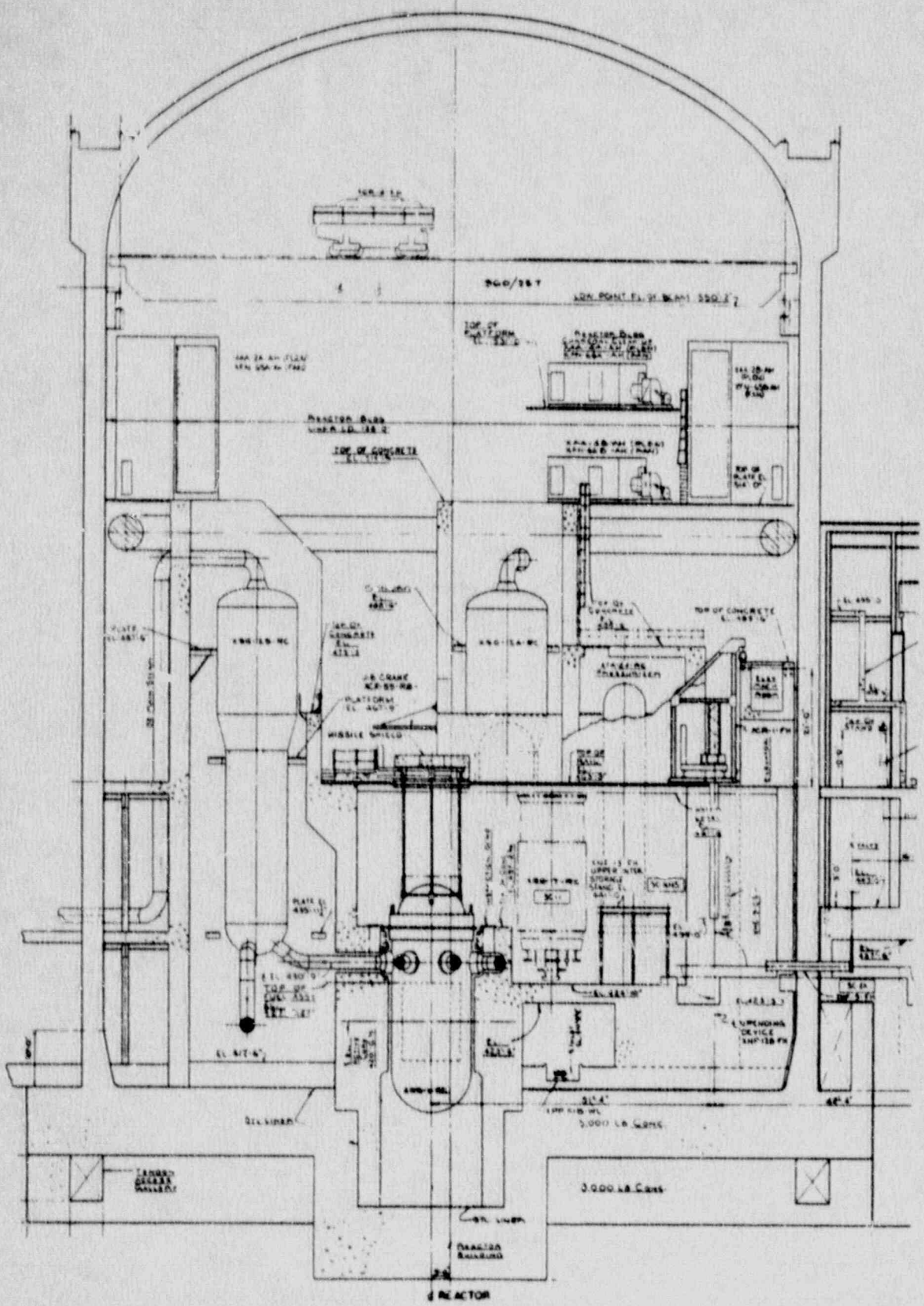


Figure 4.3-4. Section Views of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete) (Sheet 2 of 2)

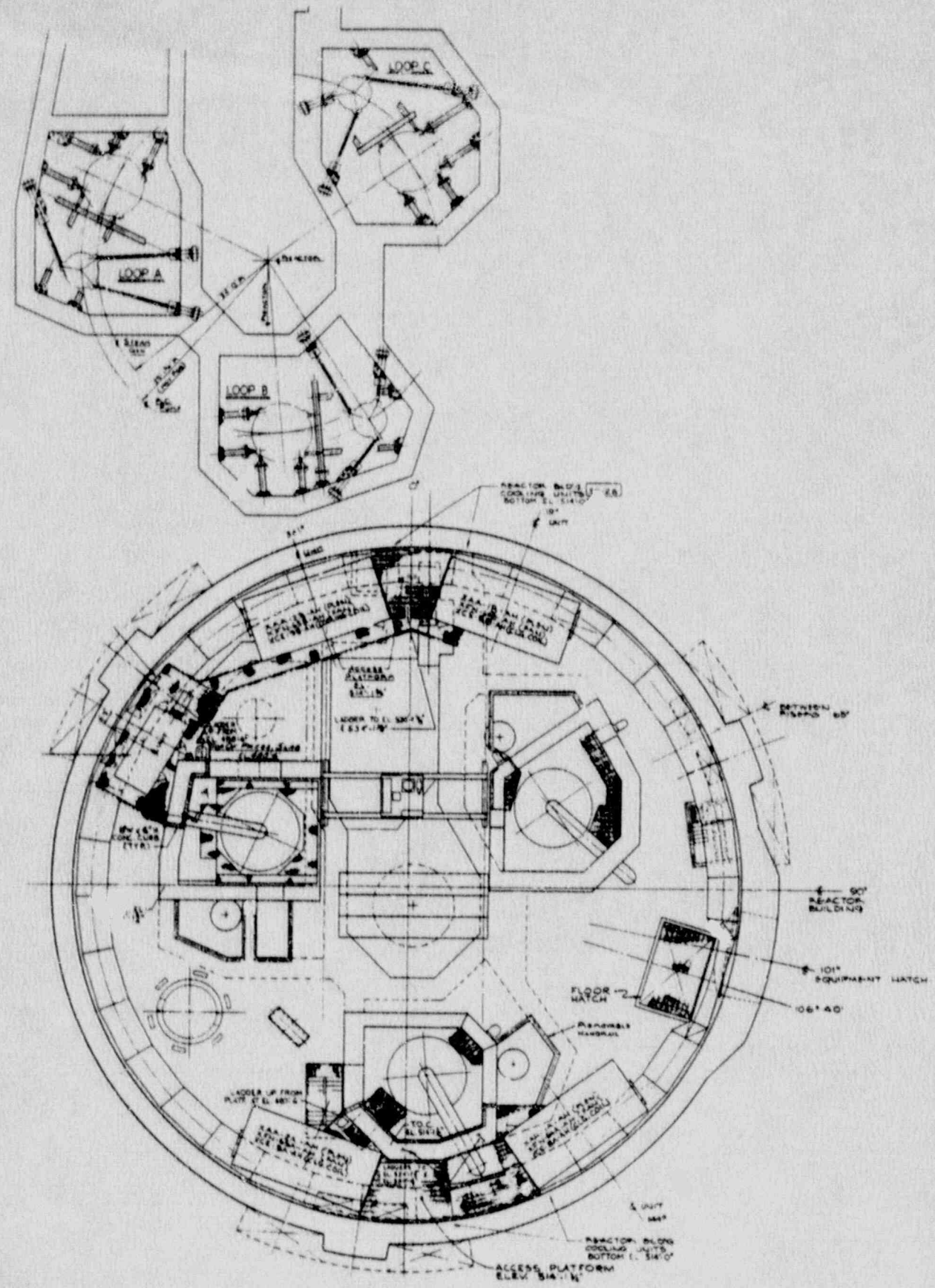


Figure 4.3-5. Plan View of the Summer Large, Dry Containment (3-D Post-Tensioned Concrete)

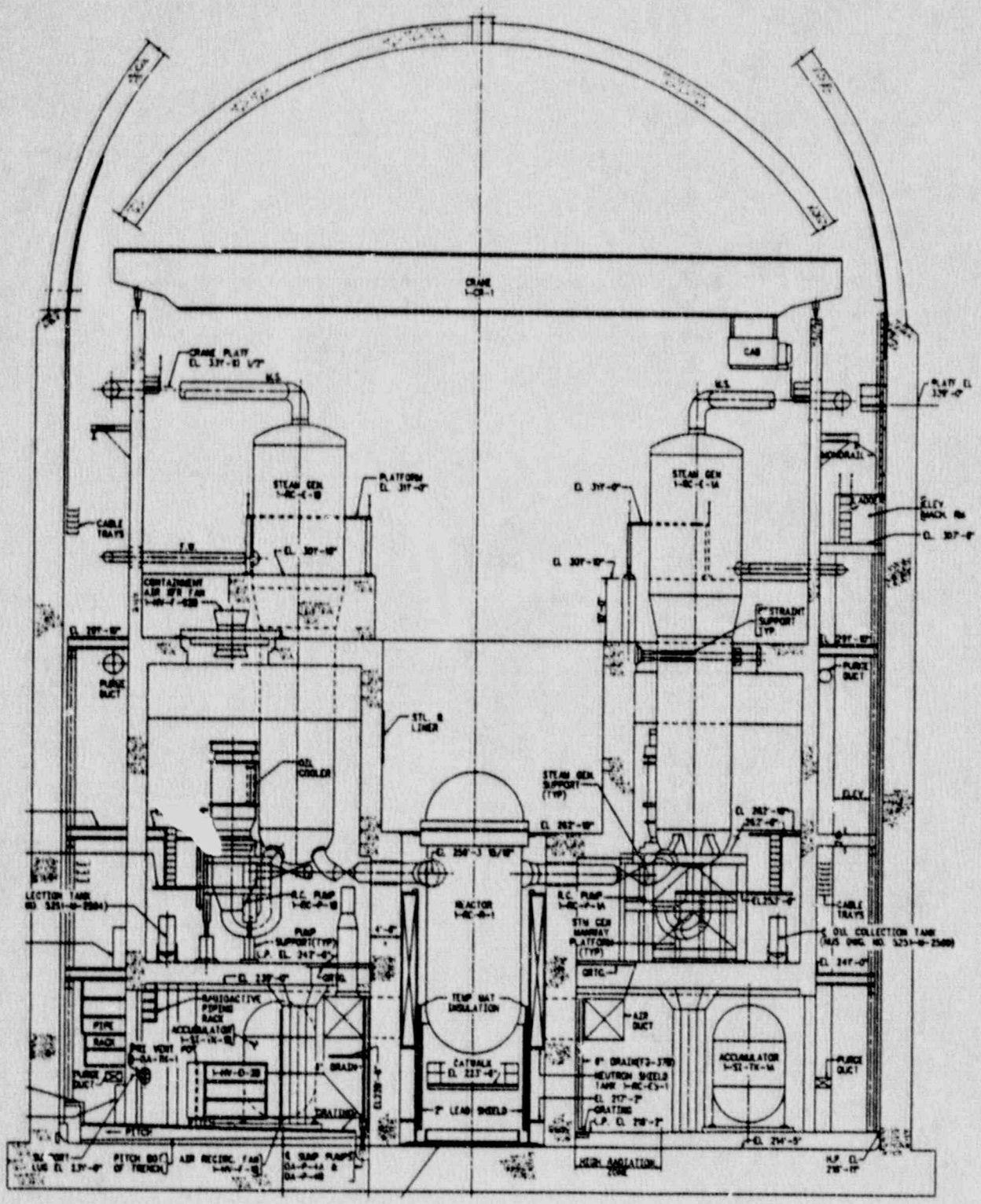


Figure 4.3-6. Section Views of the North Anna Subatmospheric Containment (Sheet 1 of 2)

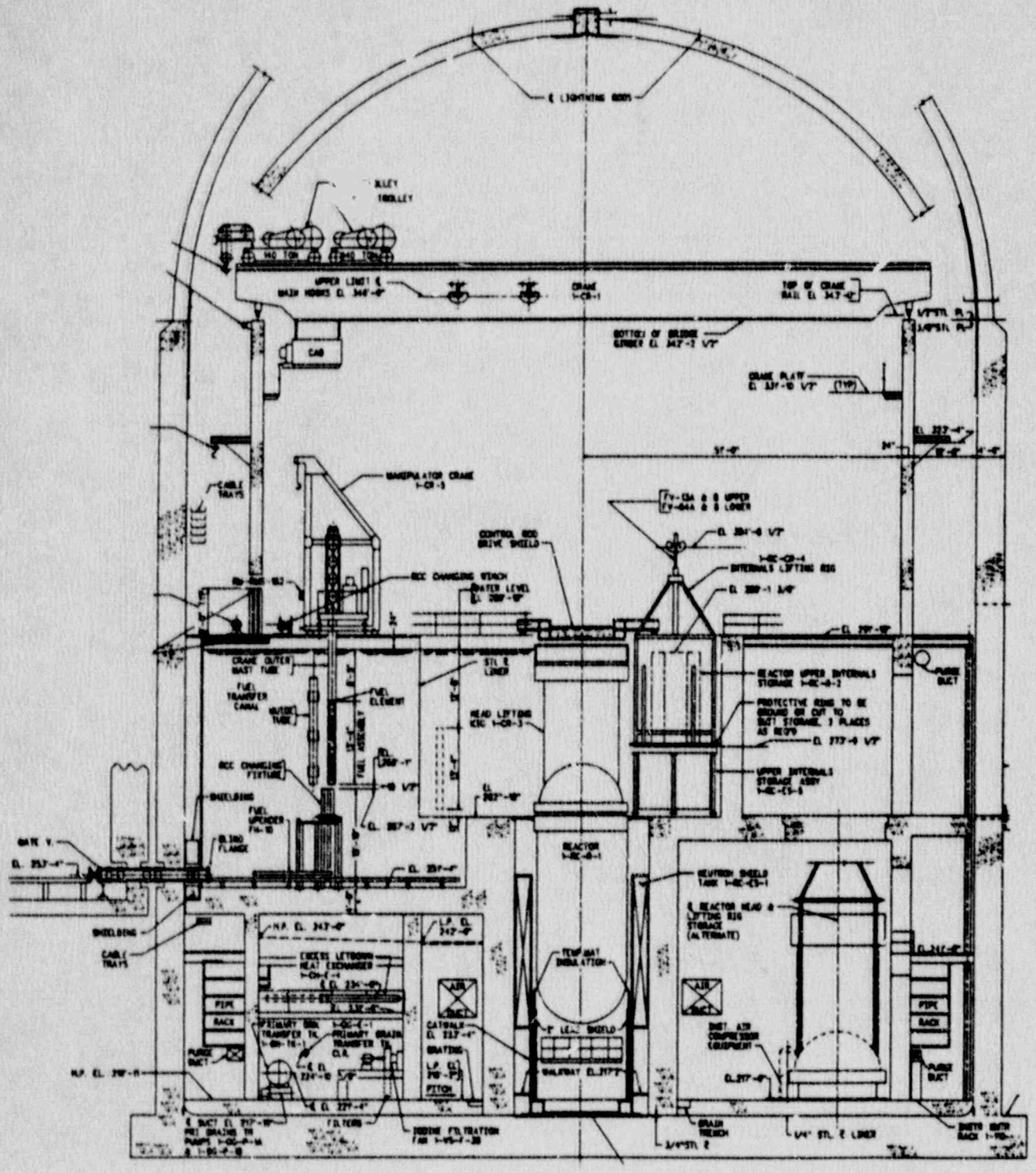


Figure 4.3-6. Section Views of the North Anna Subatmospheric Containment (Sheet 2 of 2)

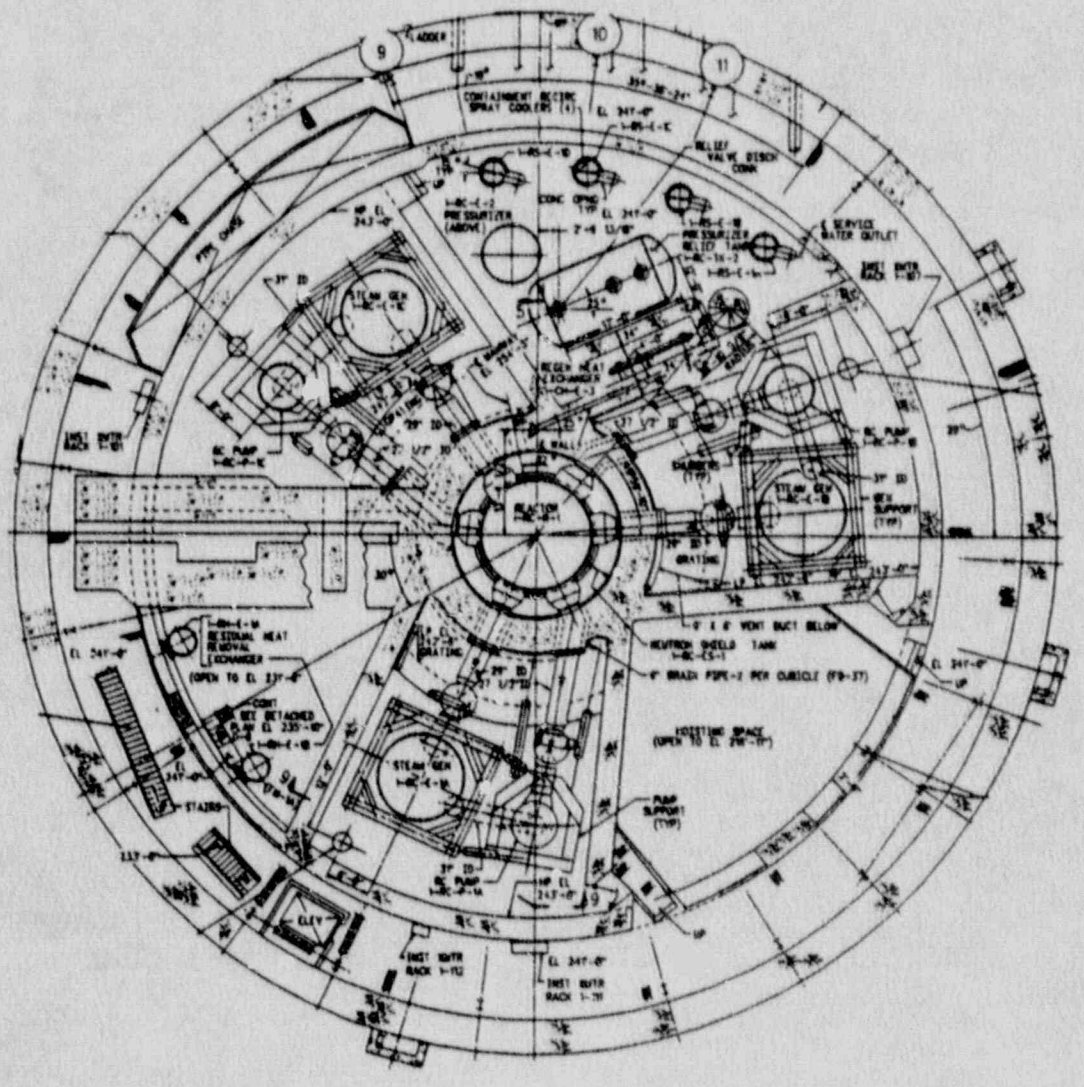


Figure 4.3-7. Plan View of the North Anna Subatmospheric Containment

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.

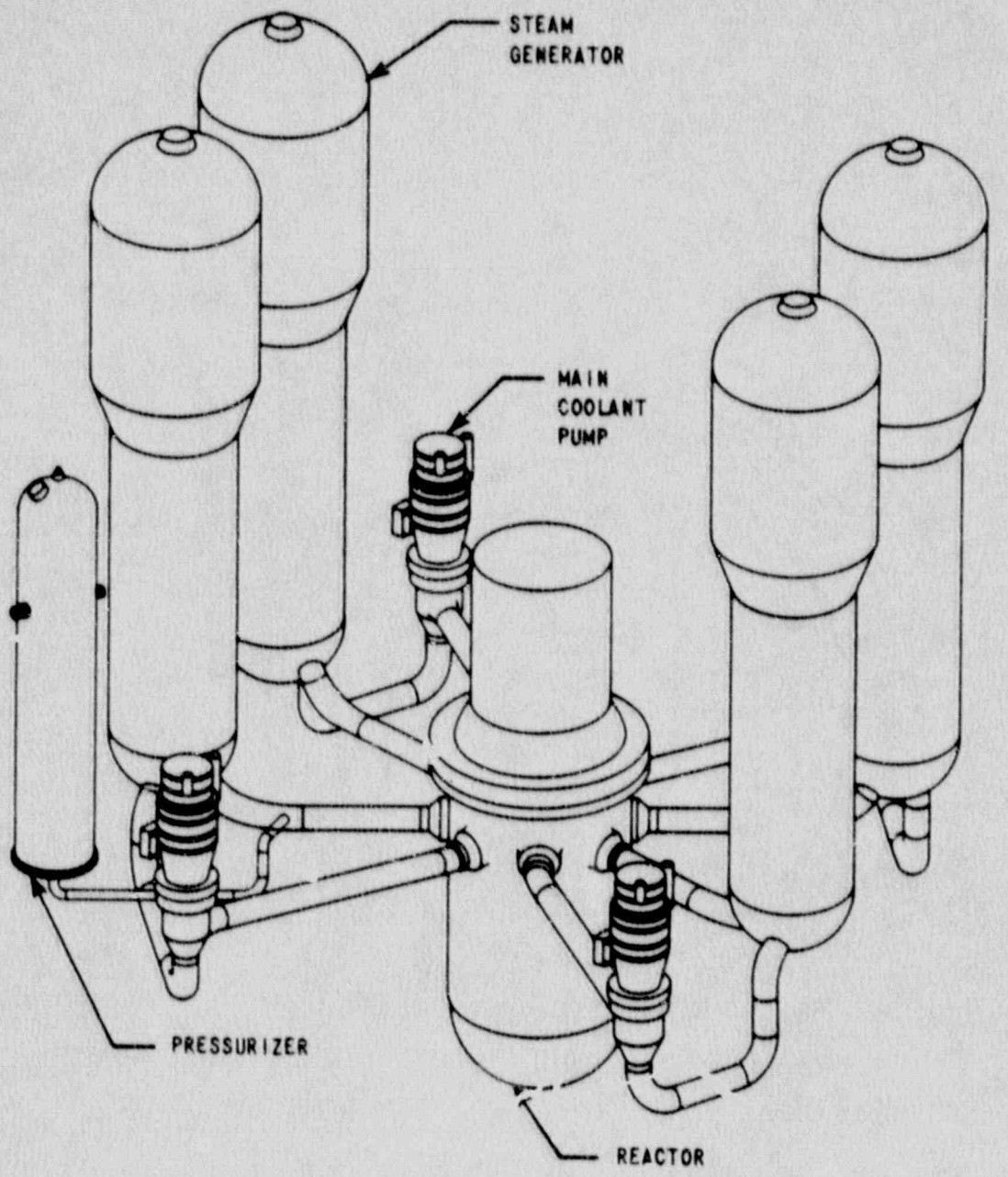


Figure 4.4-1. Westinghouse 4-Loop NSSS

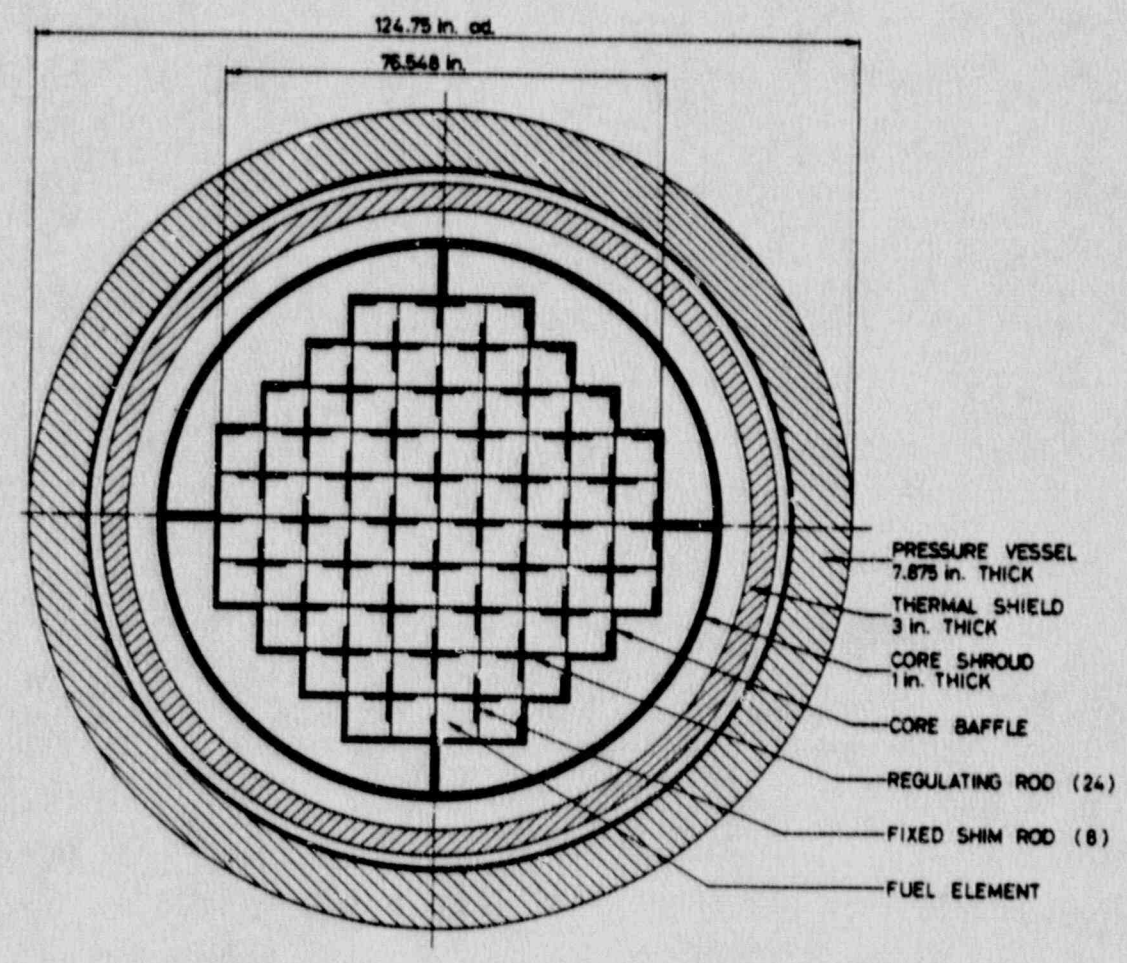


Figure 4.4-2. General Arrangement of the Yankee-Rowe Core

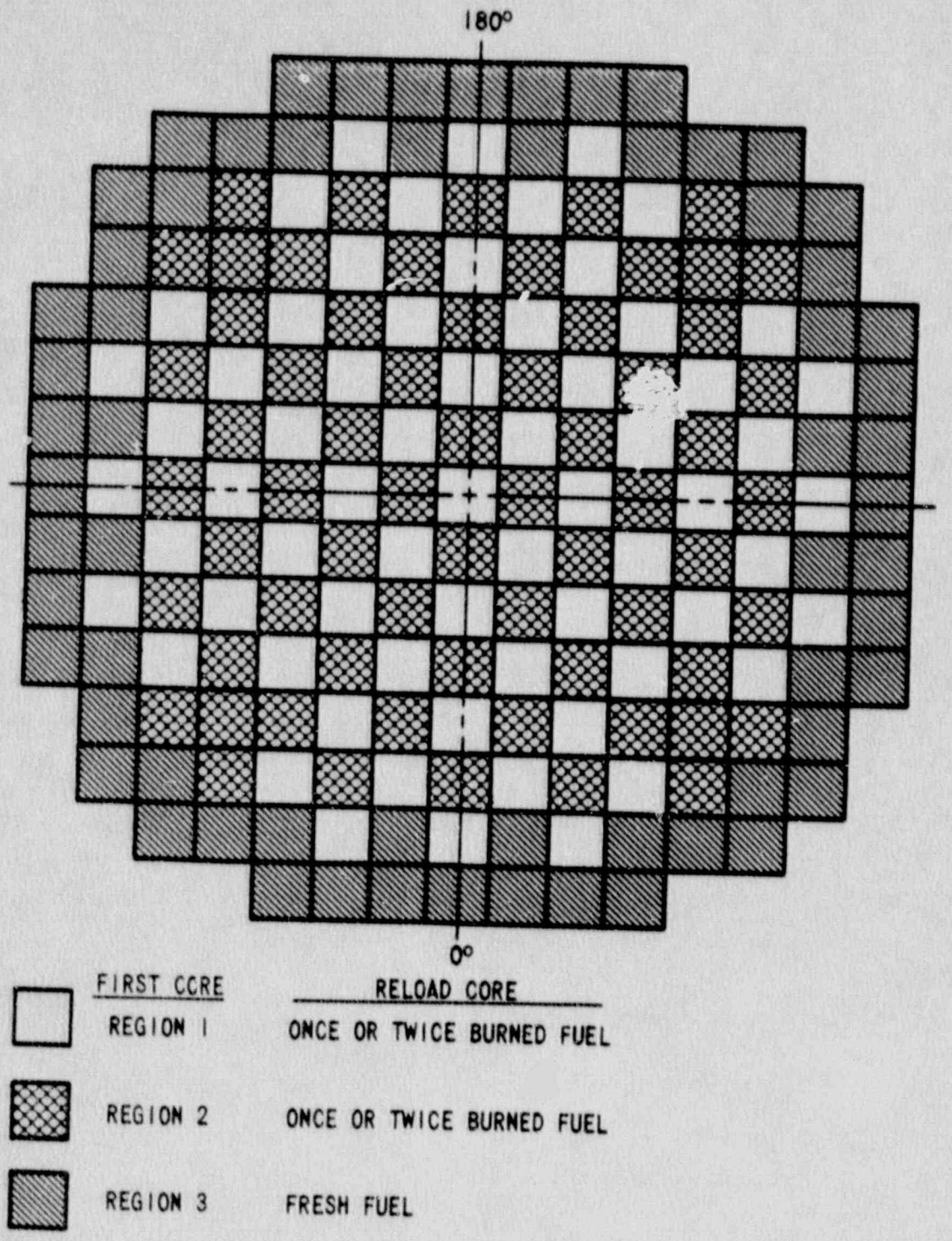


Figure 4.4-3. General Arrangement of a 193 Fuel Assembly Core

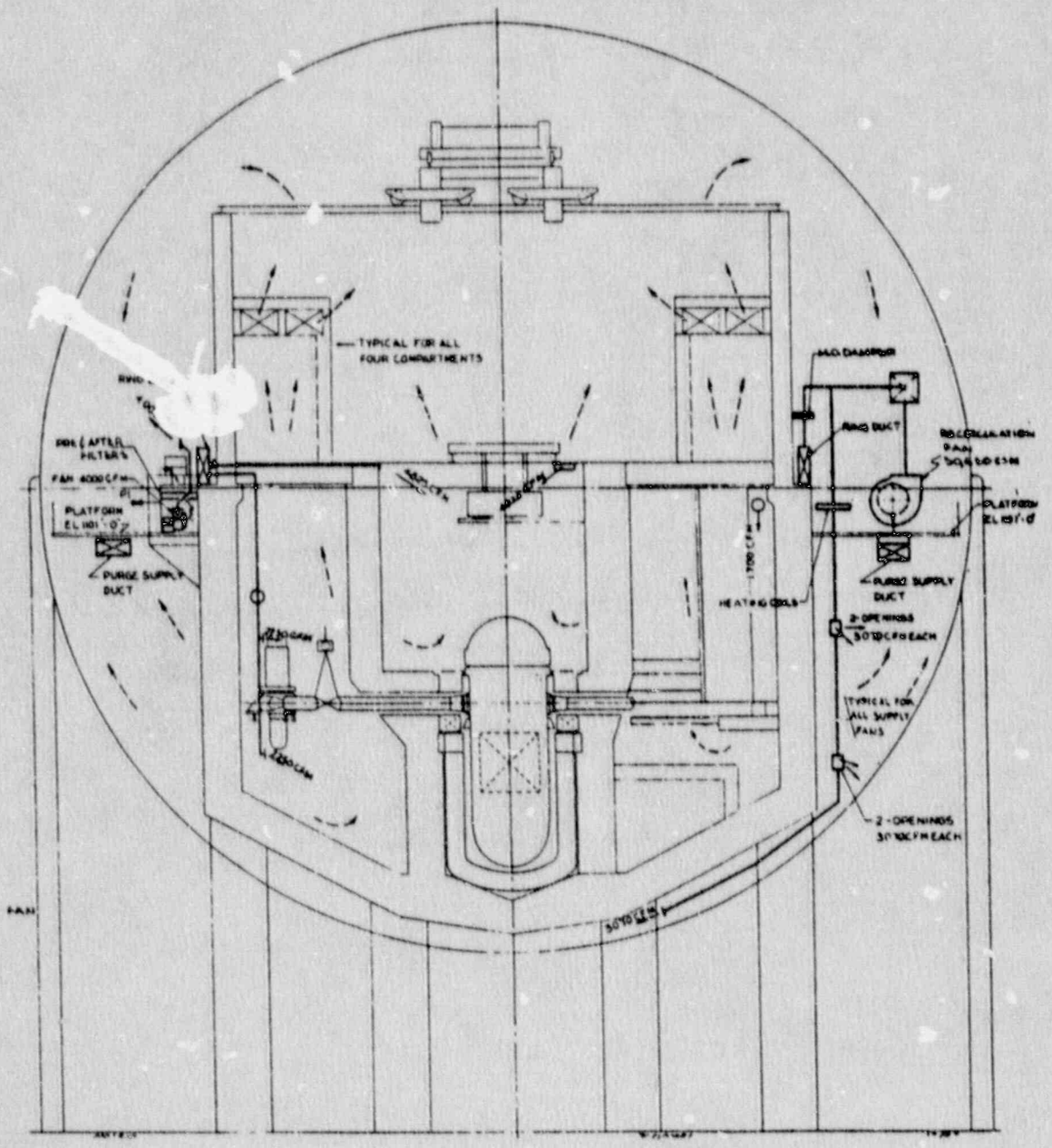


Figure 4.4-4. Section View of the Yankee-Rowe Large, Dry Containment (Steel Sphere)

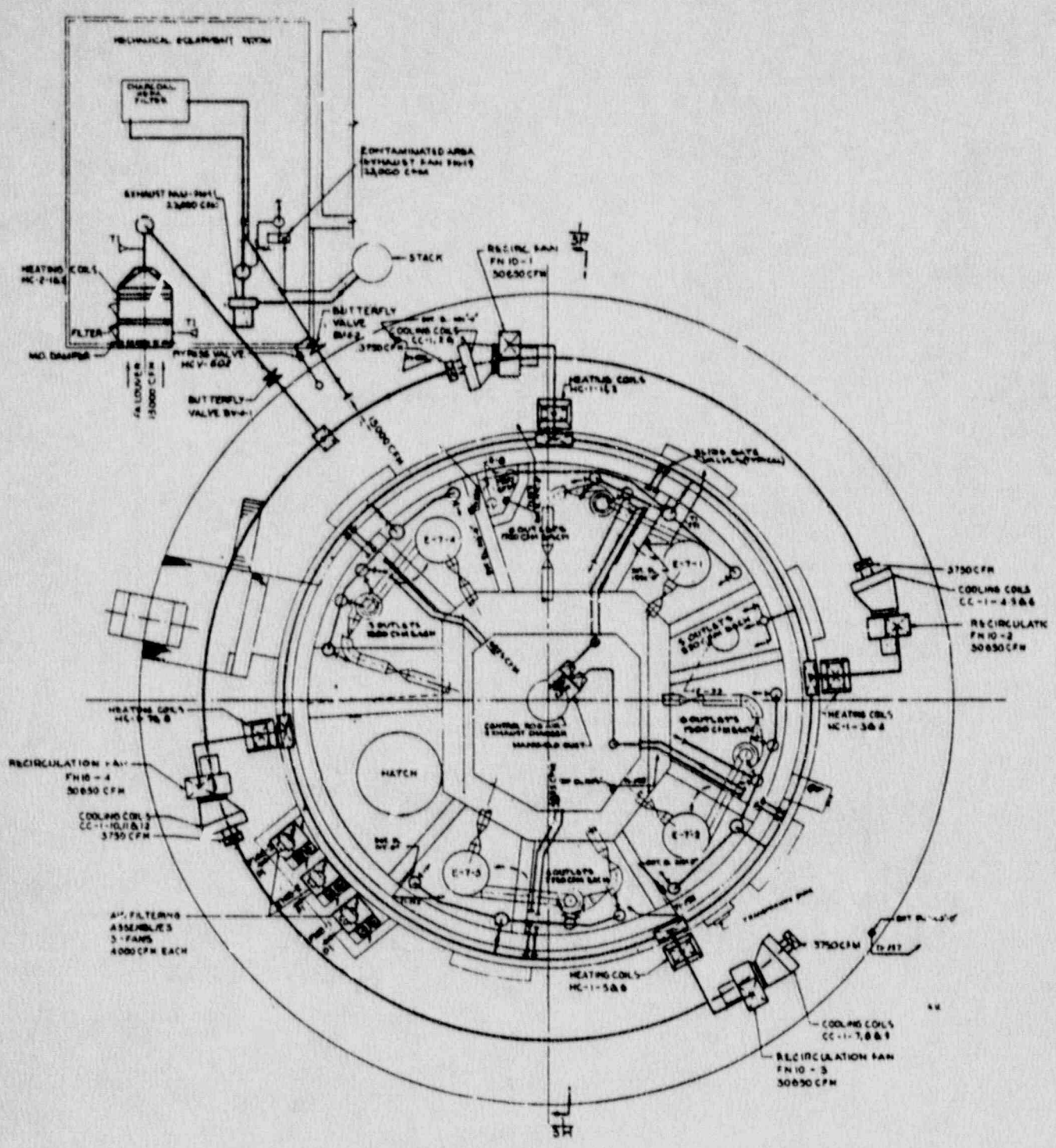


Figure 4.4-5. Plan View of the Yankee-Rowe Large, Dry Containment (Steel Sphere)

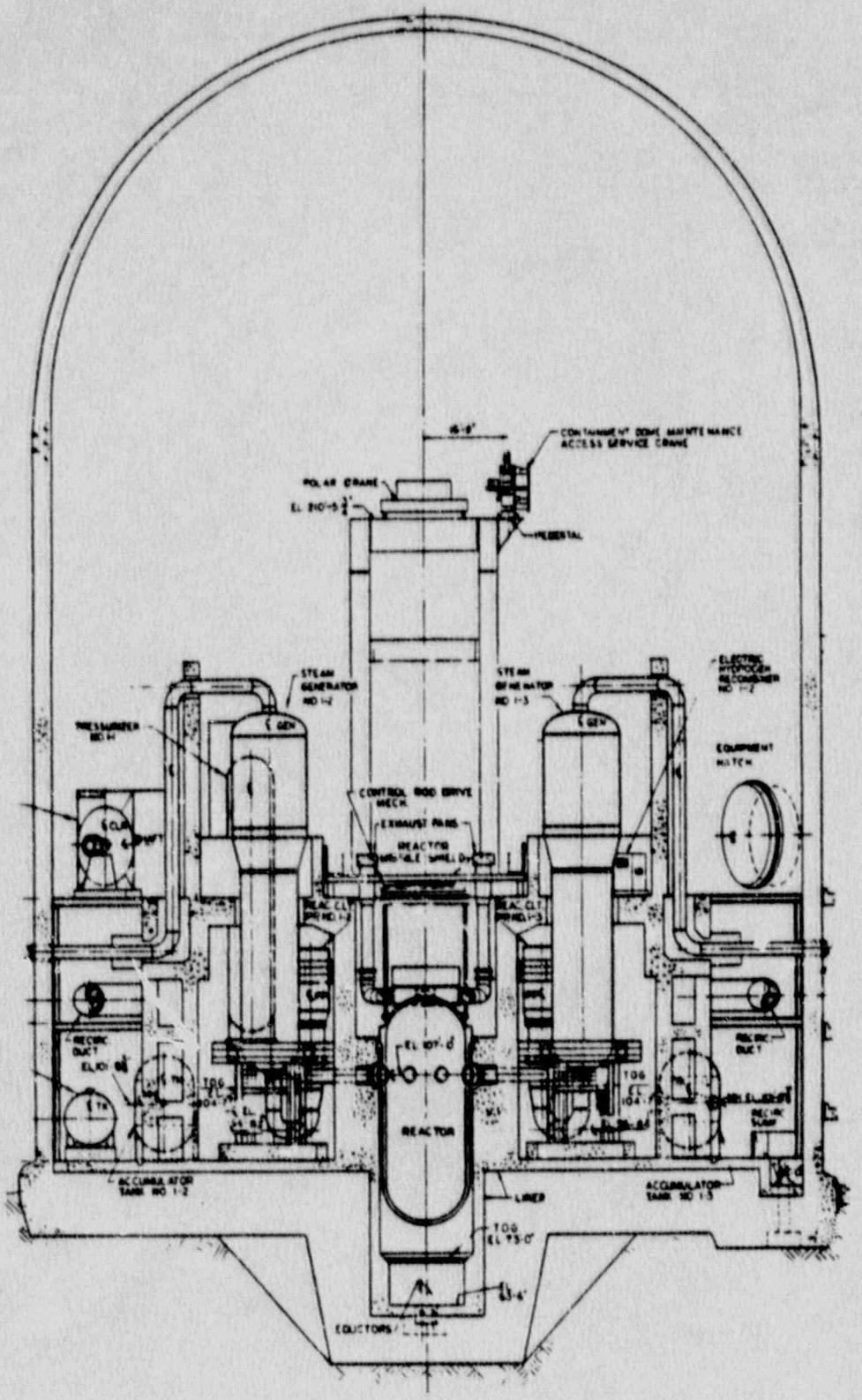


Figure 4.4-6. Section Views of the Diablo Canyon Large, Dry Containment (Reinforced Concrete)

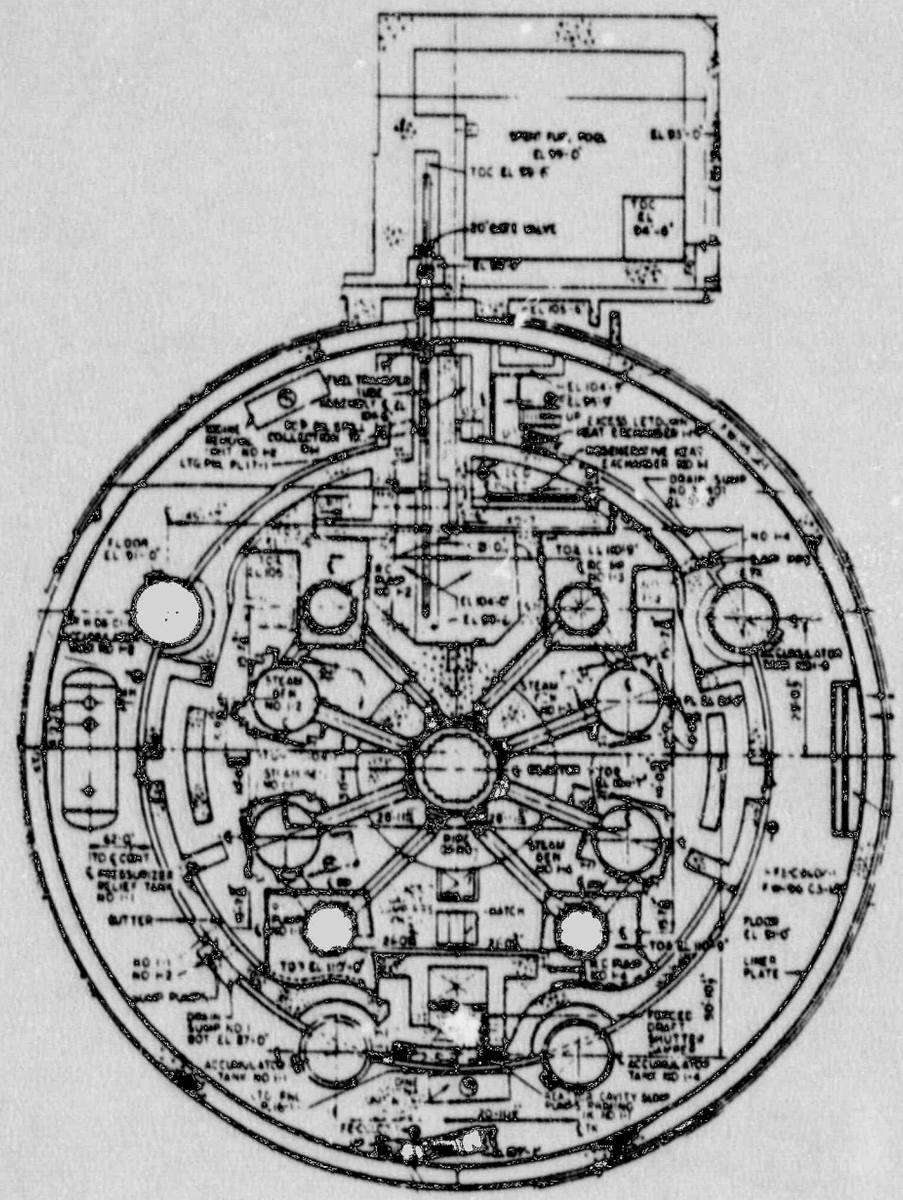


Figure 4.4-7. Plan View of the Diablo Canyon Large, Dry Containment (Reinforced Concrete)

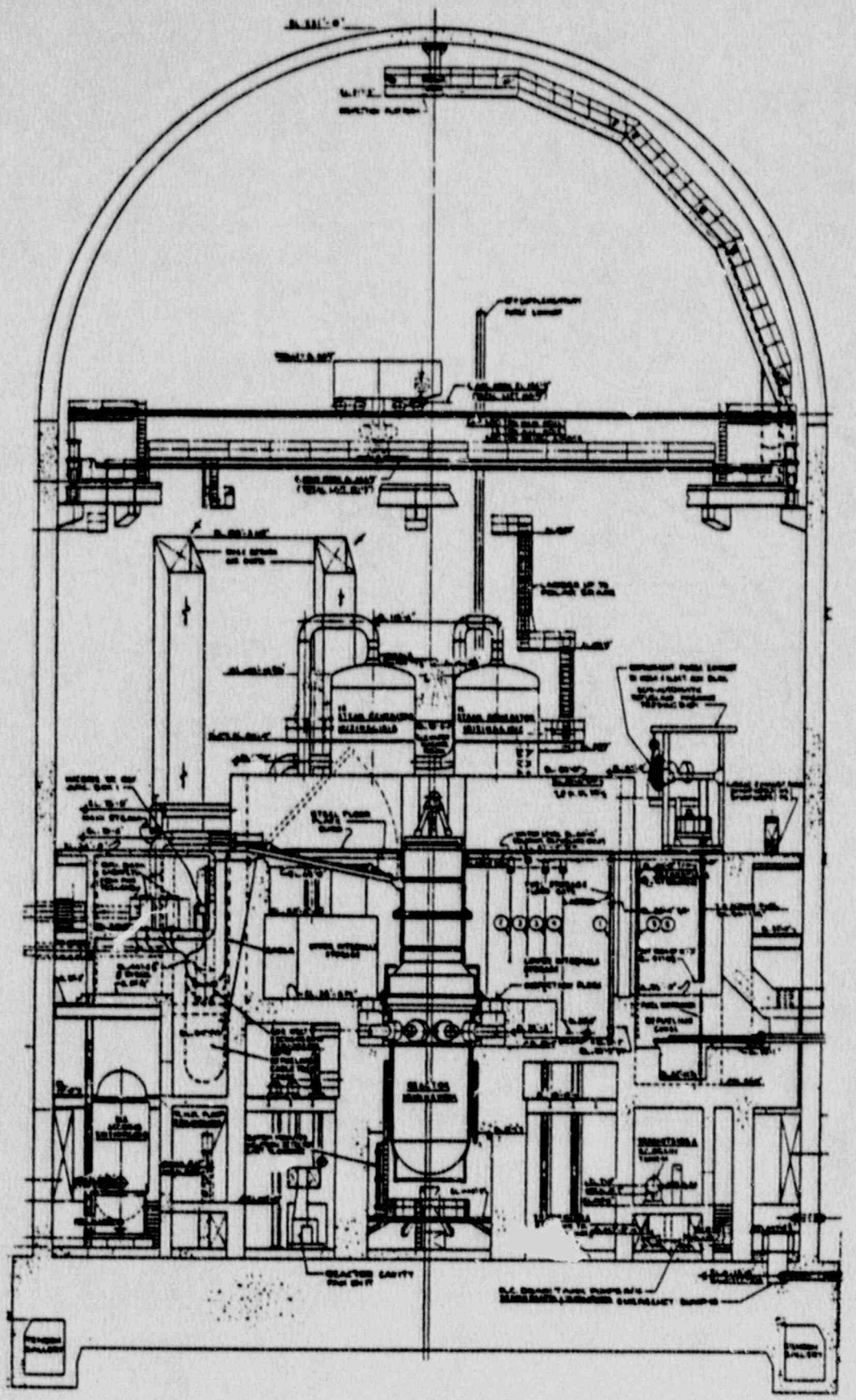


Figure 4.4-8. Section View of South Texas Large, Dry Containment (3-D Post-Tensioned Concrete)

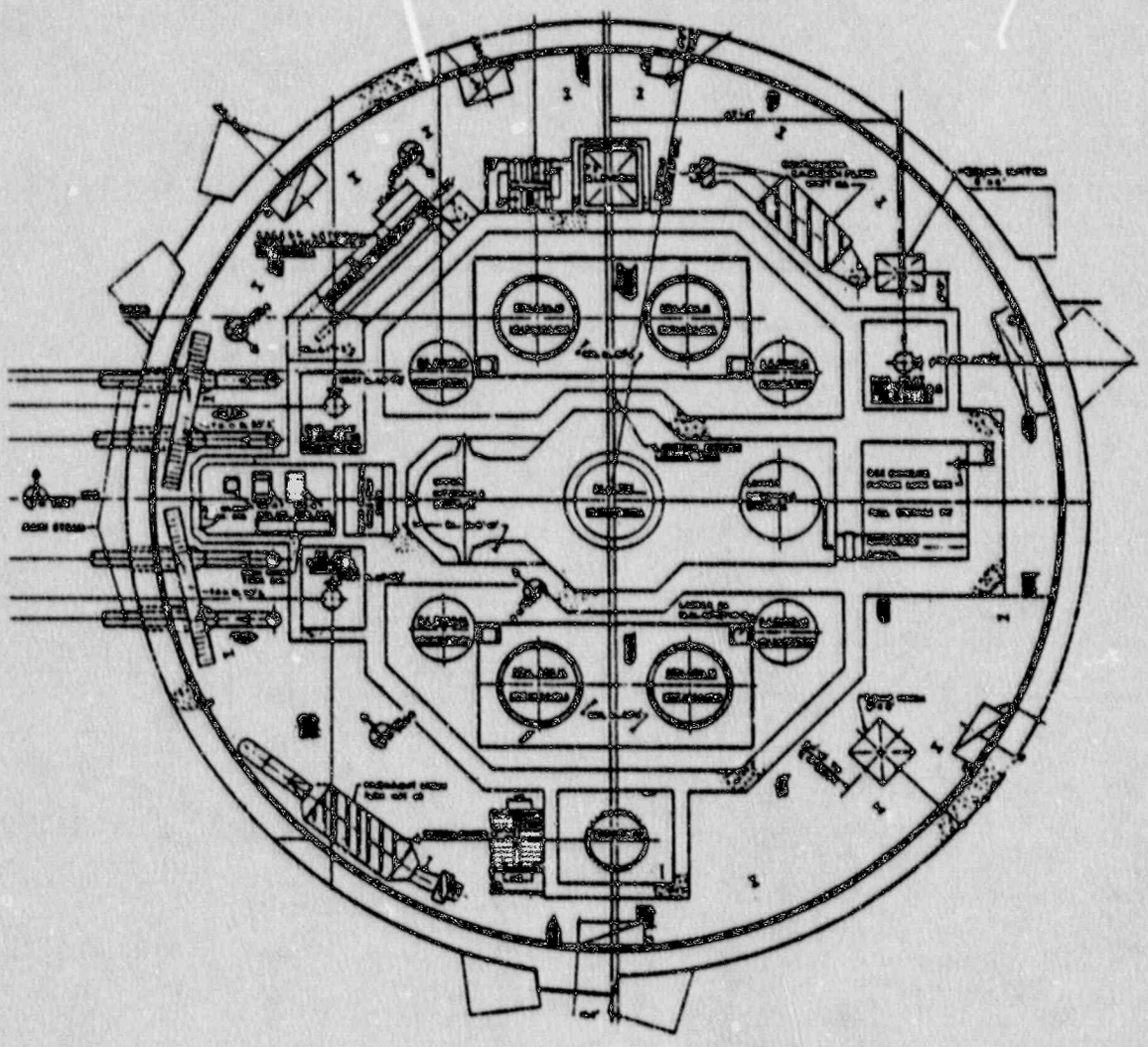


Figure 4.4-9. Plan View of South Texas Large, Dry Containment (3-D Post-Tensioned Concrete)

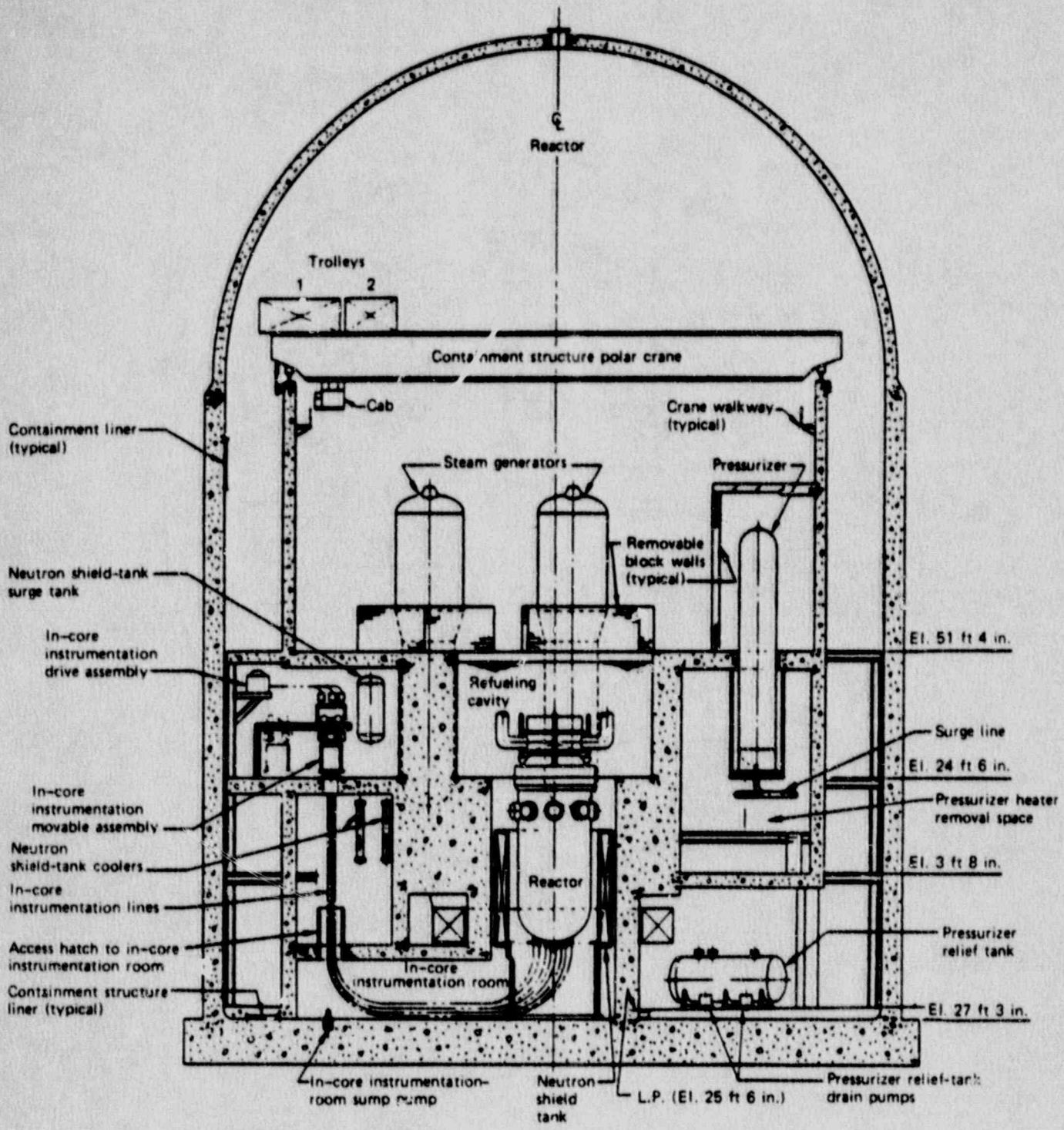


Figure 4.4-10. Section View of the Millstone 3 Subatmospheric Containment

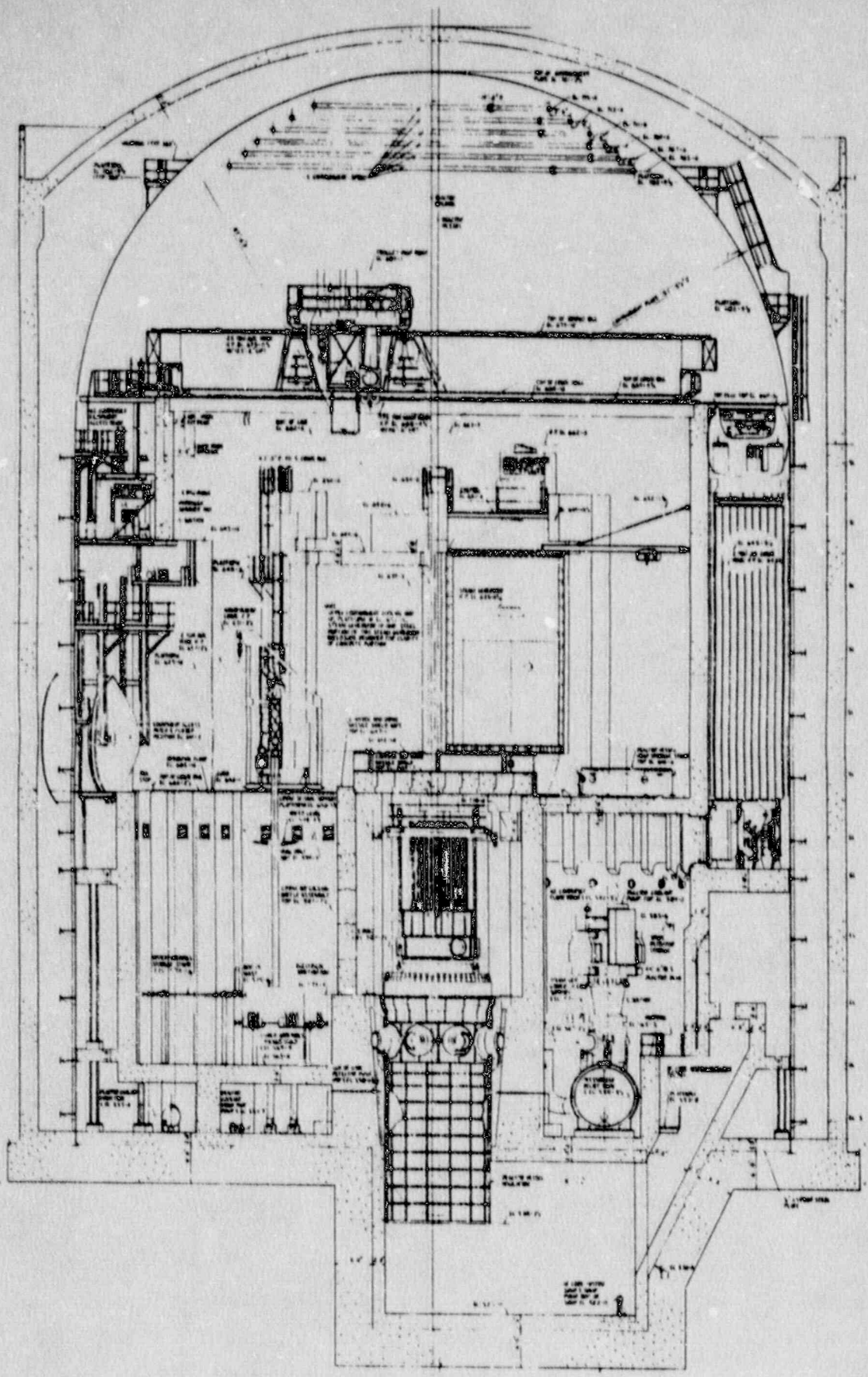


Figure 4.4-11. Section View of the Catawba Ice Condenser Containment

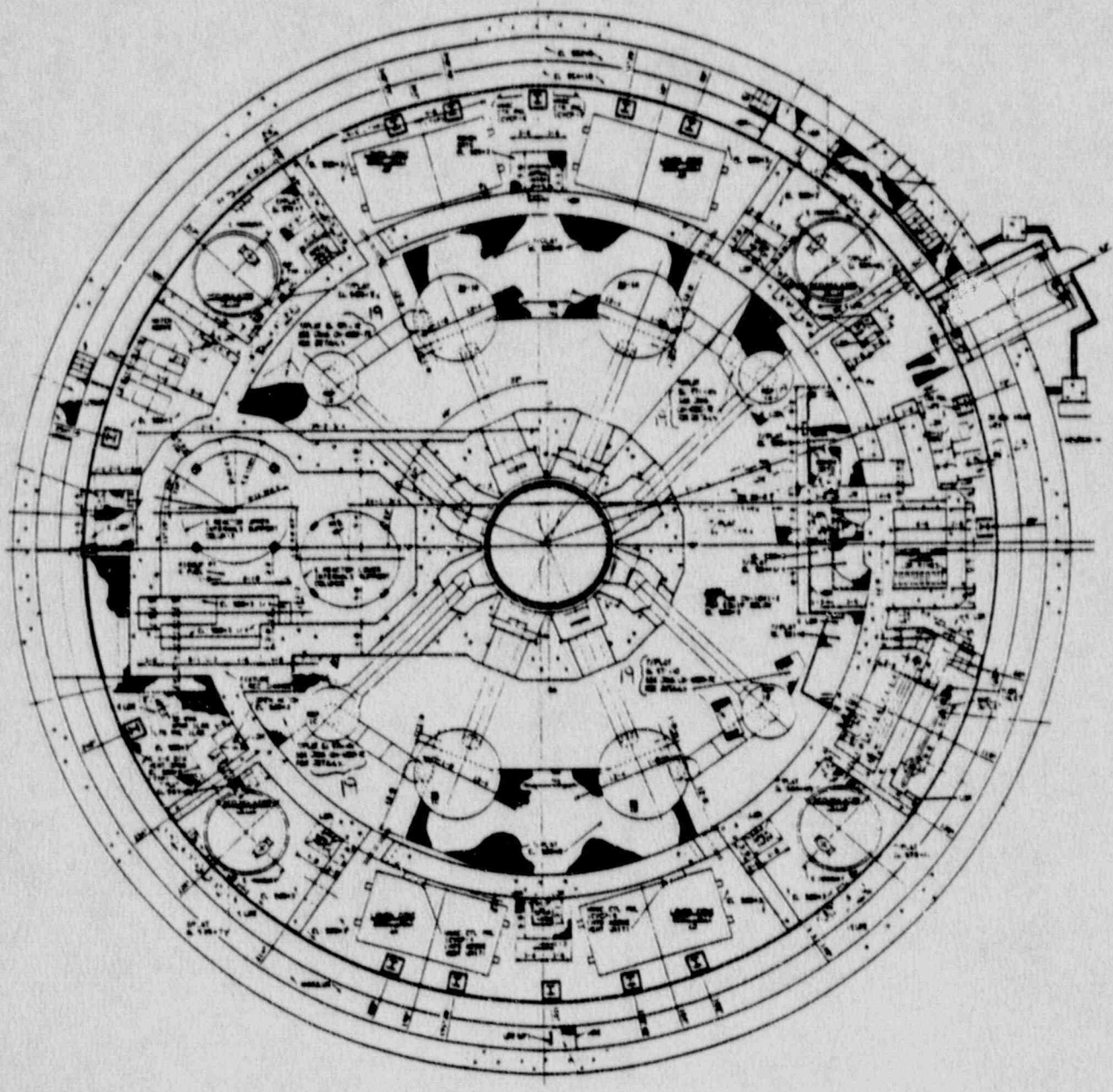


Figure 4.4-12. Plan View of the Catawba Ice Condenser Containment Lower Compartment

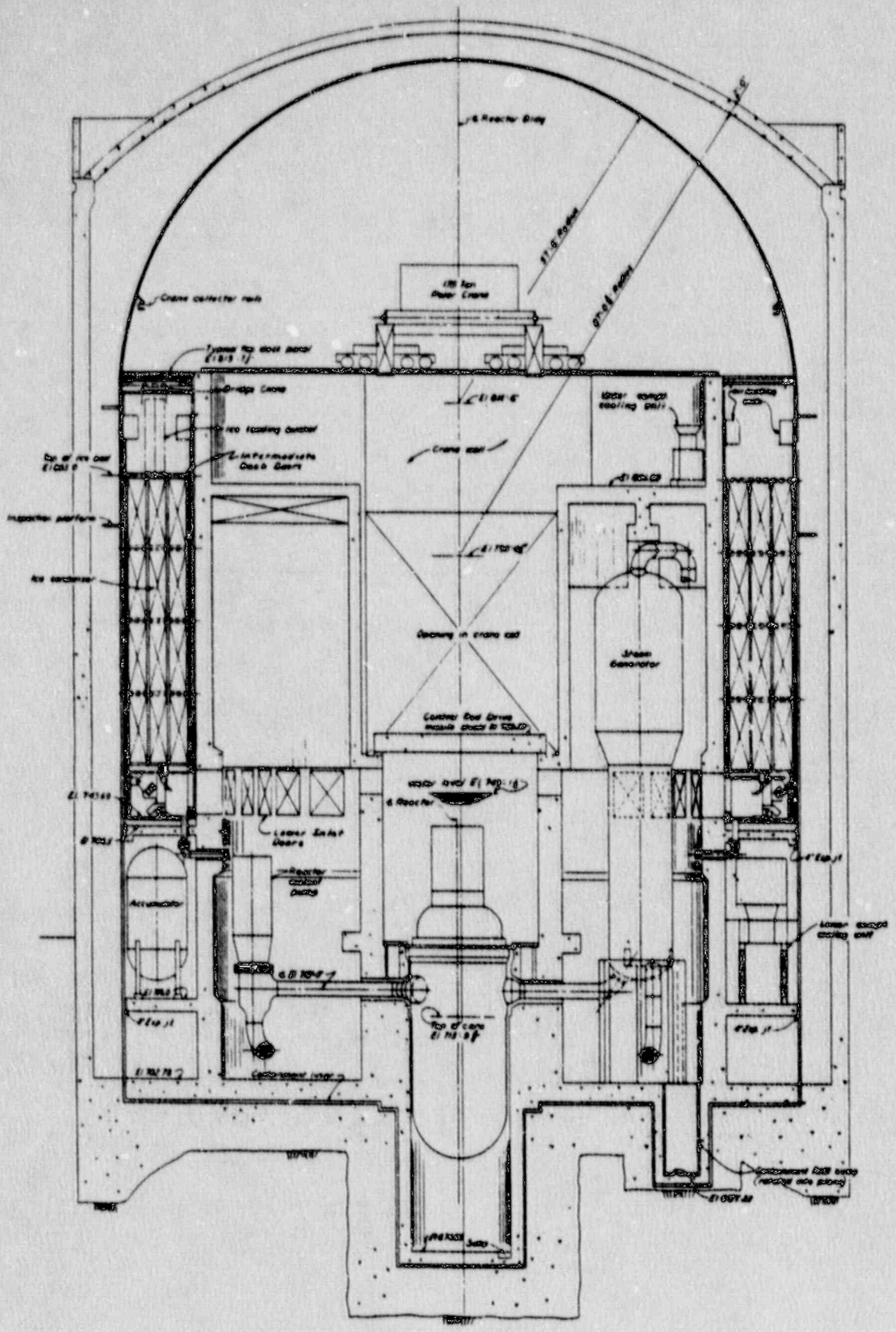


Figure 4.4-13. Section Views of the Watts Bar Ice Condenser Containment (Sheet 1 of 2)

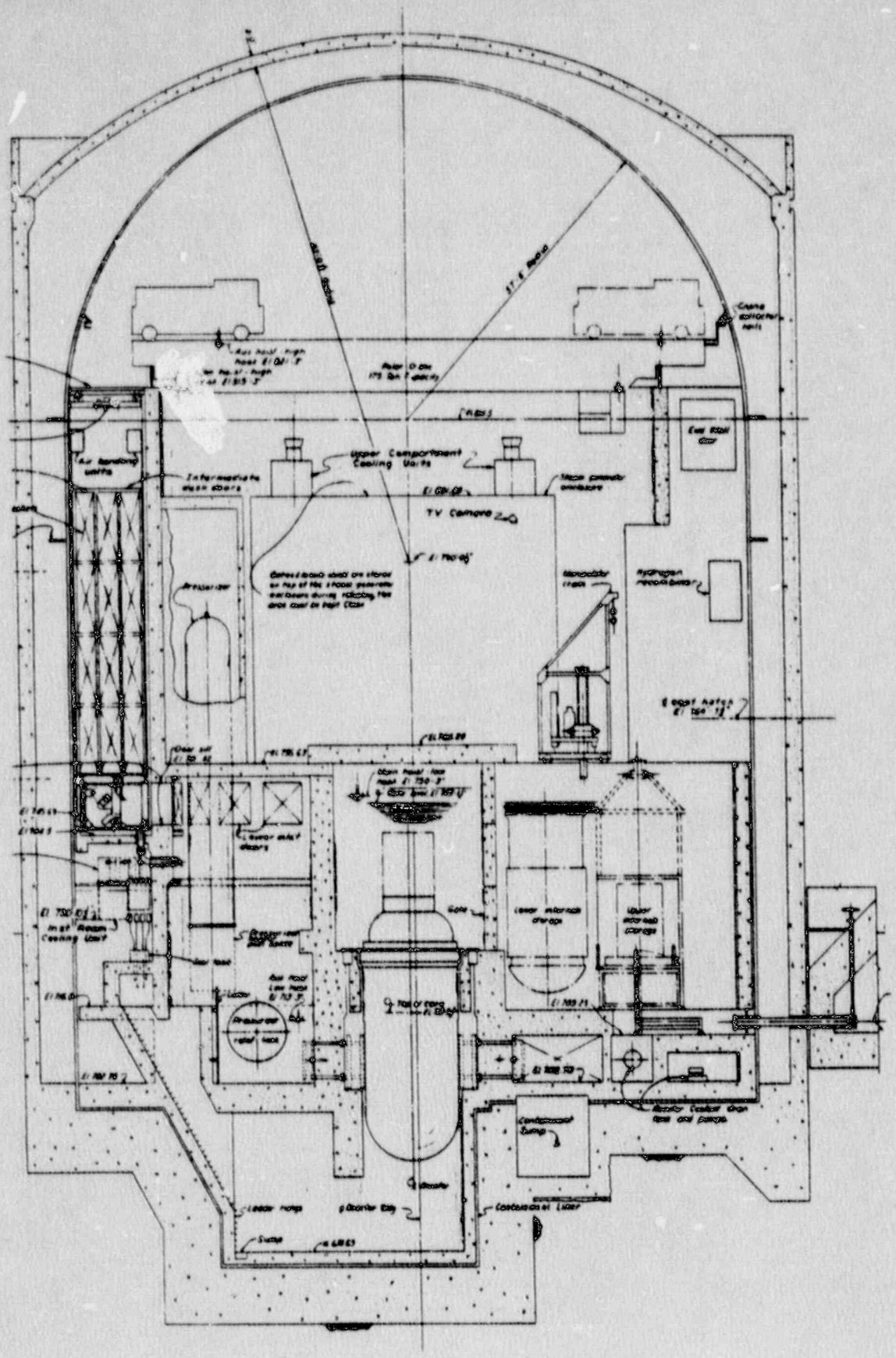


Figure 4.4-13. Section Views of the Watts Bar Ice Condenser Containment (Sheet 2 of 2)

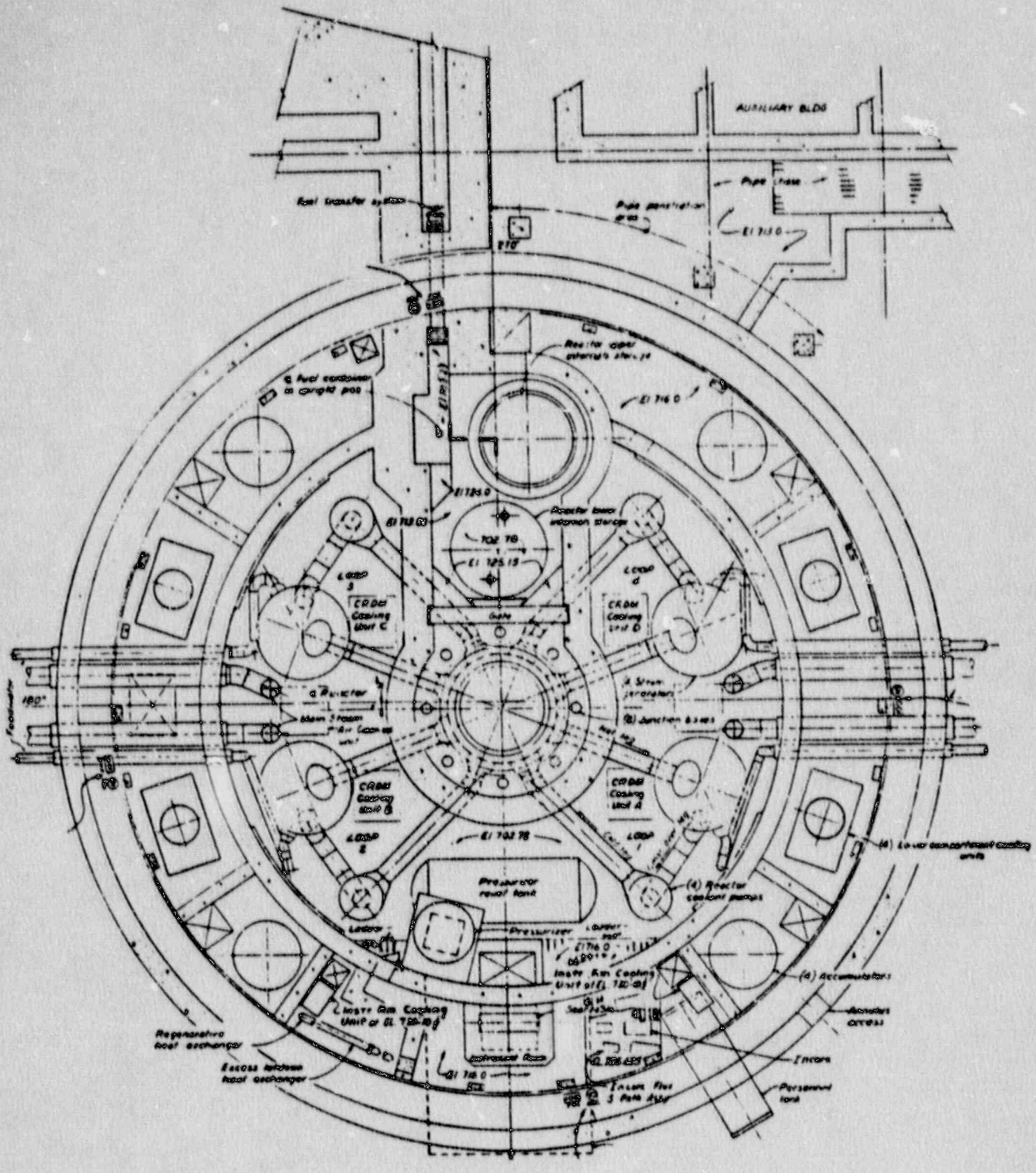


Figure 4.4-14. Plan View of the Watts Bar Ice Condenser Containment Lower Compartment

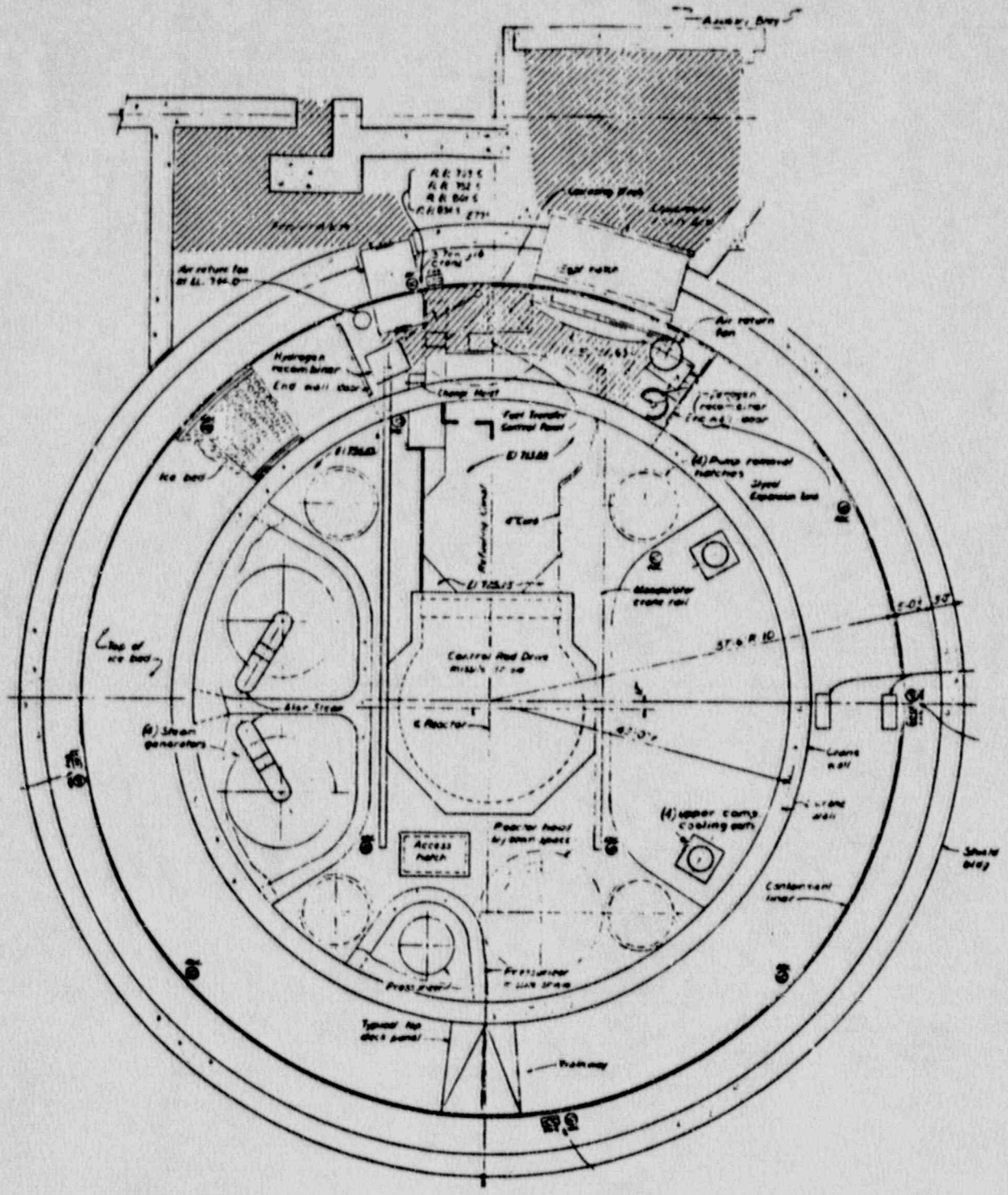


Figure 4.4-15. Plan View of the Watts Bar Ice Condenser Containment Upper Compartment

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 P
W	3-loop	Turkey Point 3 & 4	2208	172	119.5	144	82.80	157	15 x 15	
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 Diar (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

		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
Beaver Valley 1 & 2	W	2660	3	3/3	51	2	M	350 @ 1169	3	Cent	150 @ 2514	150	Same as cent. charging pumps				(a)
						1	T	700 @ 1169									
Braidwood 1 & 2	W	3411	4	2/3	51	1	M	890 @ 1452	2	Cent	150 @ 2526	150	2	Cent	400 @ 1106	0	(a)
						1	D	840 @ 1452	1	PD	98	98					
Byron 1 & 2	W	3411	4	2/3	51	1	M	890 @ 1452	2	Cent	150 @ 2526	150	2	Cent	400 @ 1106	0	(a)
						1	D	840 @ 1452	1	PD	98	98					
Callaway	W	3425	4	2/3	F	2	M	500 @ 1387		Cent	150	150	2	Cent	425 @ 1162	0	(a)
						1	T	1200 @ 1387	1	PD	98	98					
Catawba 1 & 2	W	3411	4	3/3	51	2	M	500 @ 1392	2	Cent	150 @ 2800	150	2	Cent	400 @ 1750	0	(a)
						1	T	100 @ 1395	1	PD	98	98					
Comanche Peak 1 & 2	W	3425	4	2/3	F	2	M	470 @ 1107	2	Cent	unk.	unk.	2	Cent	unk.	0	(a)
						1	T	900 @ 1107	1	PD							
D.C. Cook 1	W	3250	4	3/3	51	1	M	450 @ 1177	2	Cent	150 @ 2800	150	2	Cent	400 @ 1700	0	(a) (b)
						1	T	900 @ 1177	1	PD	98	98					
D.C. Cook 2	W	3250	4	2/3	51F	1	M	450 @ 1177	2	Cent	150 @ 2800	150	2	Cent	400 @ 1700	0	(a) (b)
						1	T	900 @ 1177	1	PD	98	98					
Diablo Canyon 1 & 2	W	3338	4	3/3	UT	2	M	440 @ 1300	2	Cent	150 @ 2514	150	2	Cent	425 @ 1084	0	
						1	T	880 @ unk.	1	PD	98 @ 2514	98					
Farley 1 & 2	W	2652	3	2/3	51	2	M	350 @ 1214	3	Cent	150 @ 2800	150	Same as cent. charging pumps				(a)
						1	M	700 @ 1214									
Ginna	W	1520	2	2/2	44	2	M	200 @ 1344	3	PD	60	60	3	Cent	300 @ 1170	0	(a) (c)
						1	T	400 @ 1344									
						2	M	200 @ 1080									
Haddam Neck	W	1825	4	2/3	27	2	T	450 @ 1000	2	Cent	360 @ 2300	360	2	Cent	970 @ 1750	0	(a)
									1	PD	30	30					
Indian Point 2	W	2758	4	2/3	44F	2	M	400 @ 1350	3	PD	98	98	3	Cent	400 @ 1180	0	
						1	T	800 @ 1350									
Indian Point 3	W	3025	4	2/3	44F	2	M	400 @ 1350	3	PD	98	98	3	Cent	400 @ 1080	0	(a)
						1	T	800 @ 1350									
Kewaunee	W	1650	2	2/2	51	2	M	240 @ 1235	3	PD	60	60	2	Cent	700 @ 1082	0	(a)
						1	T	240 @ 1235									
McGuire 1 & 2	W	3425	4	3/3	51	2	M	450 @ 1655	2	Cent	150 @ 2514	150	2	Cent	400 @ 1106	0	(a)
						1	T	900 @ 1730	1	PD	55	55					
Millstone 3	W	3411	4	2/3	51	2	M	575 @ 1290	3	Cent	150 @ 2800	150	2	Cent	425 @ 1500	0	(a)
						1	T	1150 @ 1290									
North Anna 1 & 2	W	2775	3	2/3	51F	2	M	350 @ 1214	3	Cent	150 @ 2500	150	Same as cent. charging pumps				(a)
						1	T	700 @ 1214									
Point Beach 1 & 2	W	1518	2	2/2	44F	1	M	200 @ 1192	3	PD	60.5	60.5	2	Cent	700 @ 1750	0	(a) (b)
						1	T	400 @ 1192									
Prairie Island 1 & 2	W	1650	2	2/2	51	1	M	200 @ 1200	3	PD	60.5	60.5	2	Cent	710 @ 1082	0	(a) (d)
						1	T	200 @ 1200									
Robinson	W	2200	3	2/3	44F	2	M	300 @ 1300	3	PD	77	77	3	Cent	775 @ 1750	0	(a)
						1	T	600 @ 1300									
Salem 1 & 2	W	3338	4	2/3	51	2	M	440 @ 1300	2	Cent	150 @ 2800	150	2	Cent	unk.	0	(a)
						1	T	880 @ 1550	1	PD	98	98					
San Onofre 1	W	1347	3	2/2	27	1	M	235 @ 1035	2	Cent	0 @ 2400	0	Same as cent. charging pumps				(a)
						1	T	300 @ 1110									
Seabrook	W	3411	4	2/3	F	1	M	710 @ 1322	2	Cent	150 @ 2800	150	2	Cent	425 @ 1750	0	(a)
						1	T	710 @ 1322	1	PD	98	98					

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	800 @ 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 38-20721	64.8	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 !A58RGP	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-1100-160	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	Bechtel	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	Bechtel	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	Bechtel	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	Bechtel	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	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	124	2.00E+06	60	0.2	No
Turkey Point 3 & 4	W	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.55E+06	59	0.25	No
Vogtle 1 & 2	W	Bechtel	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	Bechtel	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 W[illegible]nghouse 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 Reservoir	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 @851 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	1,032	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	10,800 @876 psig
Turkey Point 3 & 4	W	Bechtel	Bechtel	666	40	Once Through	Mech. Cooling Towers	2 (60%)	AC	1,1<9	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.

5. COMBUSTION ENGINEERING (C-E) PRESSURIZED WATER REACTORS (PWRS)

In the U.S., Combustion Engineering has produced 15 PWRs with two or three primary loops. As of March 1990, the numbers of units of each type are as follows:

- 2-loop (pre-System 80) 11 units
- 2-loop System 80 3 units
- 3-loop 1 unit

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 Combustion Engineering PWRs is presented in Section 5.1. The Combustion Engineering plants are described in Sections 5.2 to 5.4, and numerous detailed comparative tables are presented in Section 5.5. In Section 7, the comparative tables for the Combustion Engineering PWRs are compiled with similar tables for Westinghouse and Babcock & Wilcox PWRs.

5.1 Combustion Engineering PWR Overview

5.1.1 Primary System

All C-E PWRs except Maine Yankee have 2-loop reactor coolant systems. In the C-E 2-loop RCS, each primary loop is comprised of one hot leg that enters the bottom of a U-tube steam generator, and two cold legs that return flow to the reactor vessel. One reactor coolant pump is in each cold leg, for a total of four reactor coolant pumps. The pressurizer is connected to one of the hot legs.

In the C-E 3-loop RCS, each primary loop is comprised of one hot leg that enters the bottom of a U-tube steam generator, and one cold leg that return flow to the reactor vessel. One reactor coolant pump is in each cold leg, for a total of three reactor coolant pumps. The pressurizer is connected to one of the hot legs. This arrangement is similar to that of a Westinghouse 3-loop plant.

The C-E RCS is designed to operate with a 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 5.1-1. This is the same RCS temperature control scheme used in Westinghouse PWRs.

The C-E steam generator is a vertical U-tube design incorporating integral moisture separators and steam dryers. Basic design characteristics of Combustion Engineering steam generators are listed in Table 5.1-1.

5.1.2 Reactor Core and Fuel Assemblies

With the exception of Palisades, all C-E commercial PWRs are designed to operate with rod-type slightly-enriched fuel in a 14 x 14 fuel assembly array or a 16 x 16 array, with multi-finger control rods that insert into channels in the fuel assemblies. Palisades uses a unique 15 x 15 fuel assembly array of rod-type fuel and cruciform control rods. C-E fuel assemblies are made entirely of zircaloy. A comparison of basic C-E PWR core parameters is presented in Section 5.5. A brief description of each fuel assembly design is provided below.

A. 15 x 15 Fuel Assembly

The external dimensions of the Palisades 15 x 15 fuel assembly are 8.1135 x 8.1135 inches, with an active fuel length of 136.7 inches. The geometry of the 15 x 15 fuel assembly and the general arrangement of the Palisades core are shown in Figure 5.1-2.

Table 5.1-1. General Characteristics of C-E Steam Generators

Design Parameters	ANO 2	Calvert Cliffs 1 & 2	Fort Calhoun 1	Maine Yankee	Millstone 2	Palo Verde Verde 1, 2 & 3
Tube Side (primary)						
Design Pressure (psig)	2,485	2,485	2,485	2,485	2,485	2,485
Nom. Operating Press. (psig)	2,235	2,235	2,085	2,235	2,235	2,235
Design Temp. (°F)	650	650	650	650	650	650
Design Flow Rate (lb/hr)	6.20E+07	6.10E+07	3.59E+07	4.07E+07	6.10E+07	8.20E+07
Total Primary Side Vol. (ft3)	1,588	1,646	850	1,032	1,527	unk.
Shell Side (secondary)						
Design Pressure (psig)	1,085	985	985	985	1,000	1,270
Full Power Pressure (psig)	900	850	770	815	800	1,070
Design Temp. (°F)	560	550	550	550	550	575
Full Power Temp. (°F)	532	525	514	520	520	553
Steam Flow Rate (lb/hr) @ Full Power	unk.	6.00E+06	unk.	unk.	unk.	unk.
Total Secondary Side Vol. (ft3)	7,957	8,006	4,411	5,139	8,006	unk.

B. 14 x 14 Fuel Assembly

There are 196 lattice positions in the 14 x 14 array. The external dimensions of the 14 x 14 fuel assembly are 7.98 x 7.98 inches, with an active fuel length of 128 or 136.7 inches. Only Fort Calhoun 1 uses fuel assemblies with the shorter active fuel length.

C. 16 x 16 Fuel Assembly

The 16 x 16 fuel assembly is designed to the same physical dimensions as the 14 x 14 fuel assembly: 7.98 x 7.98 inches, however, the active fuel length has been increased to 150 inches. The 16 x 16 fuel assembly is designed to be "reconstitutable", allowing disassembly, removal and replacement of fuel rods, and reassembly. There are 226 lattice positions in the 16 x 16 array. Twenty of the lattice positions are taken by five control element assembly (CEA) guide tubes that each occupy the space of four fuel rods (i.e. a 2 x 2 array). The center guide tube can be used for in-core instrumentation. The 16 x 16 fuel assembly has a significantly lower linear power density than in the prior 14 x 14 design.

Linear Power Density	16 x 16	14 x 14
- Average (kW/ft)	5.2	7.1
- Maximum (kW/ft)	12.5	18.0

The geometry of the 16 x 16 fuel assembly is shown in Figures 5.1-3 and 5.1-4.

5.1.3 Reactivity Control System

In all C-E plants except Palisades, core reactivity is controlled by full-length control element assemblies (CEAs), part-length control element assemblies (PLCEAs), burnable poison rods, and soluble boron in the coolant. The neutron absorber in the CEAs typically is B_4C (boron carbide), the PLCEAs typically use Ag-In-Cd (silver-indium-cadmium), while the burnable poison rods use B_4C (boron carbide).

As shown in Figure 5.1-5, a CEA for a pre-System 80 plant consists of multiple control rod fingers joined at the top by a "spider" assembly and attached to the control element drive mechanism (CEDM) extension shaft. The System 80 plants (Palo Verde) have three different types of CEAs with 4, 8, and 12 control rod fingers.

The control element drive mechanisms (CEDMs) for C-E PWRs are magnetic jack type drives that are capable of inserting, withdrawing, holding, or tripping the CEAs and PLCEAs from any point within the 150 inch stroke of the drives. The magnetic jack CEDM shown Figure 5.1-6 consists of upper and lower pressure housings, the coil stack assembly, reed-switch assembly, and extension shaft assembly. Drive power is provided by the "stack coil" consisting of five DC magnet coils positioned around the CEDM housing. The lifting (rod withdrawal) operation consists of a series of magnetically operated step movements. Within the CEDM pressure housing, two sets of mechanical latches engage a notched extension shaft connected to the CEAs and PLCEAs. The magnetic force to hold and drive a rod is obtained from two DC magnet coils mounted on the outside of the CEDM motor tube. CEDM "hold" is achieved by energizing one magnet coil at reduced current while the other coils are de-energized. A controller-programmer is used to coordinate the operation of the magnet coils to achieve the stepping sequence needed to insert or withdraw the rods. The magnetic jack CEDMs are air-cooled

5.1.4 Containment

All C-E PWRs have large, dry containments that have varied construction styles as shown in Figure 5.1-7. Descriptions of containments for C-E PWRs are included in the following sections.

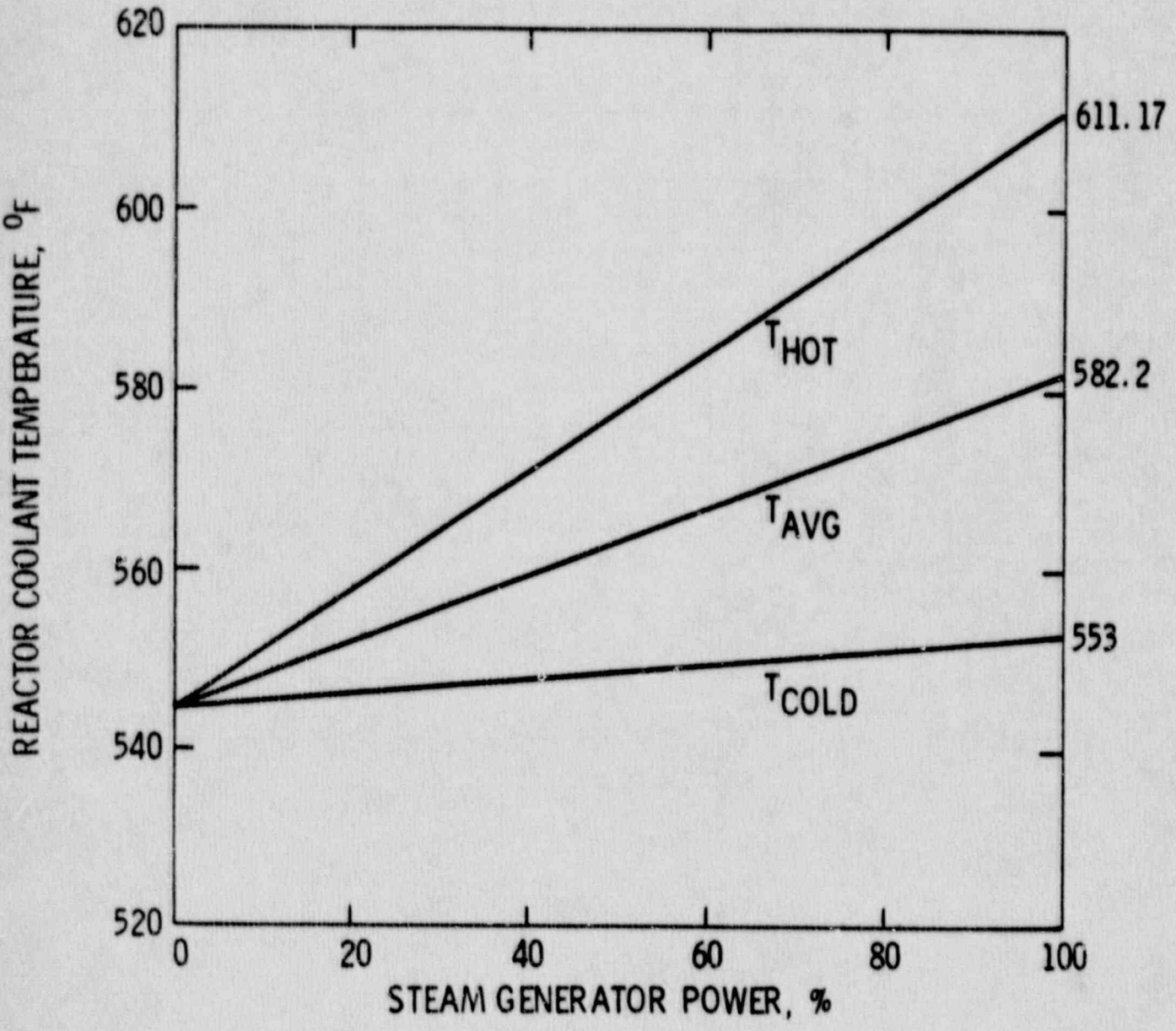


Figure 5.1-1. Typical Combustion Engineering RCS Temperature Control Program

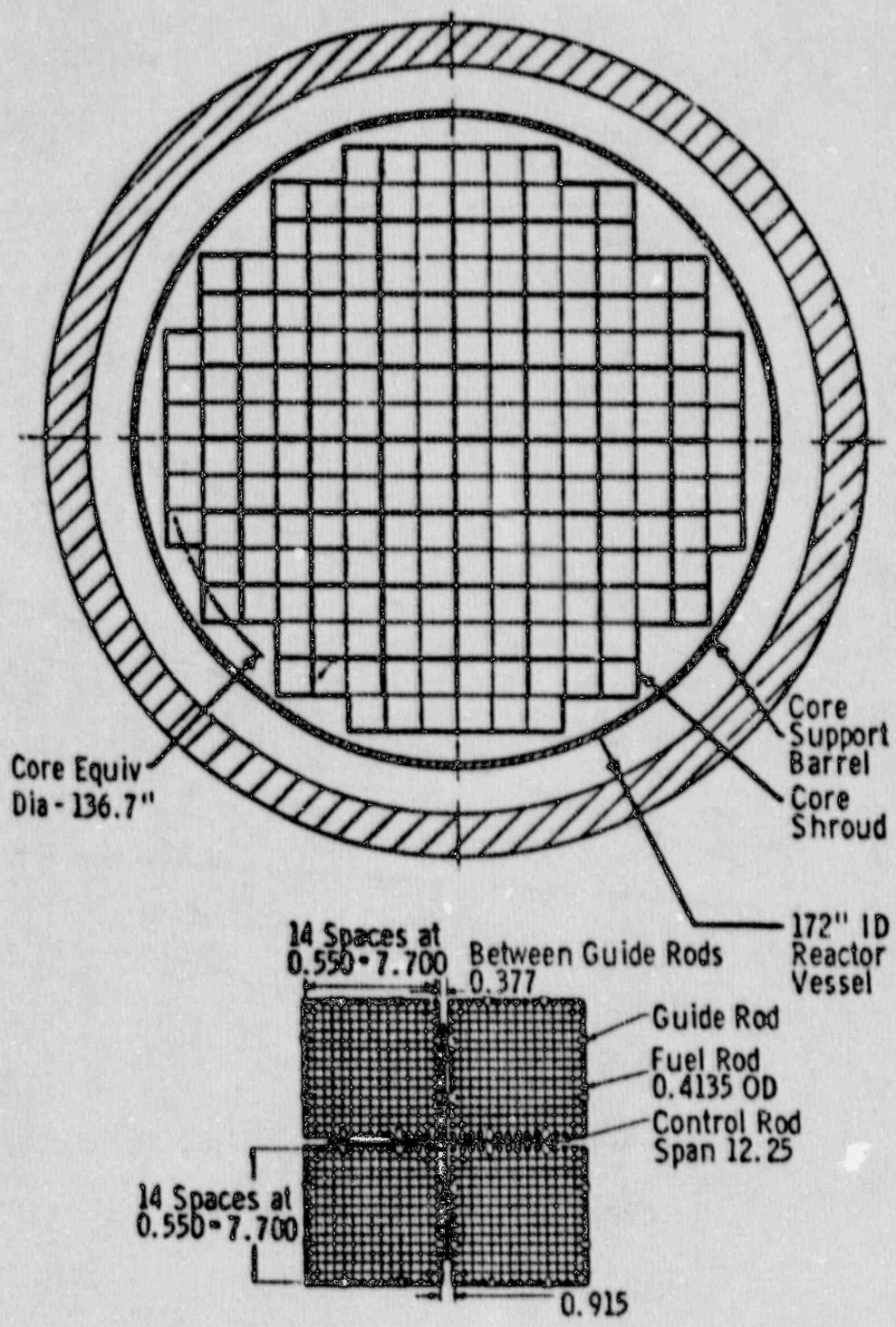


Figure 5.1-2. Combustion Engineering 15 X 15 Fuel Assembly and General Arrangement of the Palisades Core

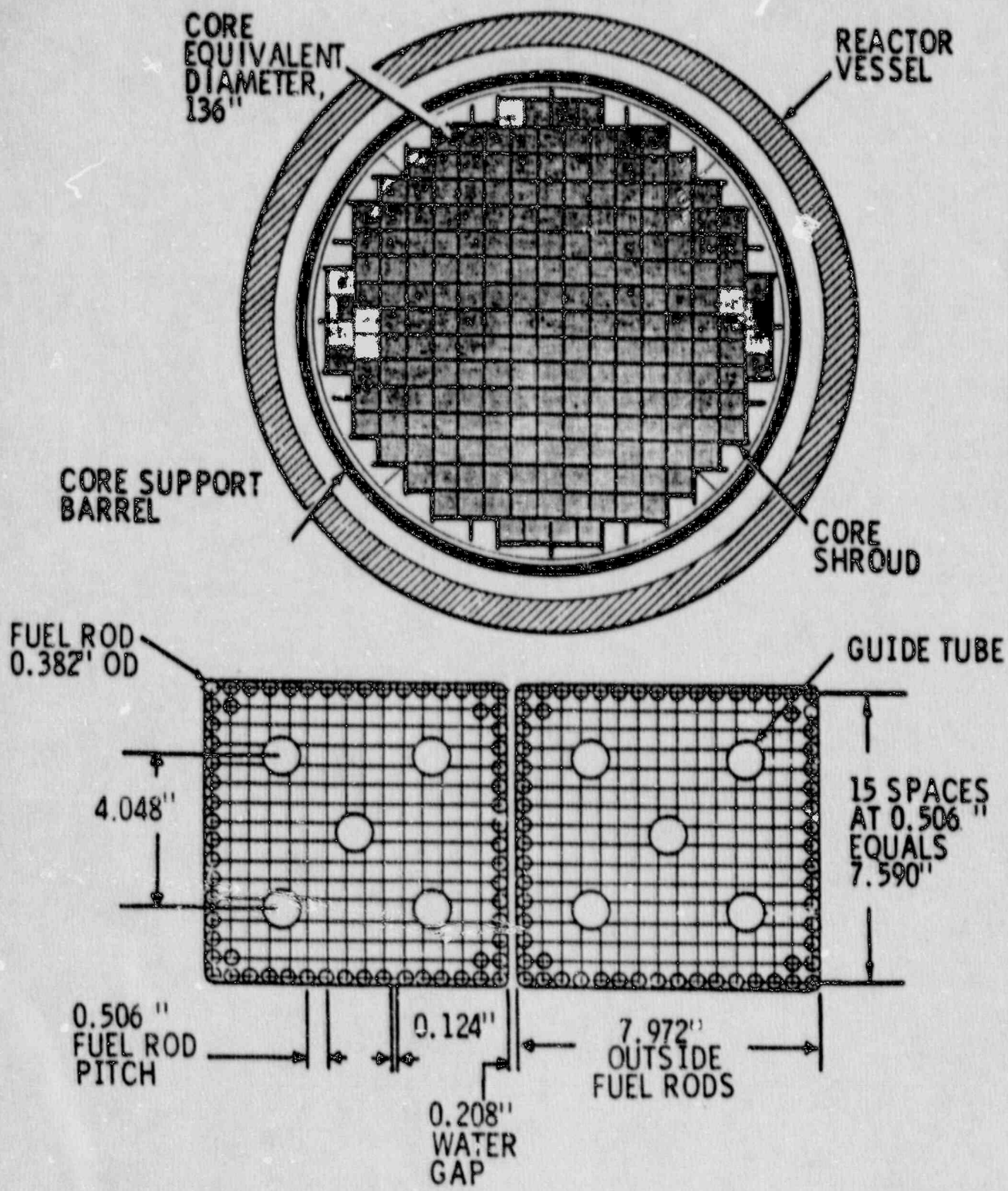


Figure 5.1-3. Combustion Engineering 16 X 16 Fuel Assembly and General Arrangement of a 217 Fuel Assembly Core

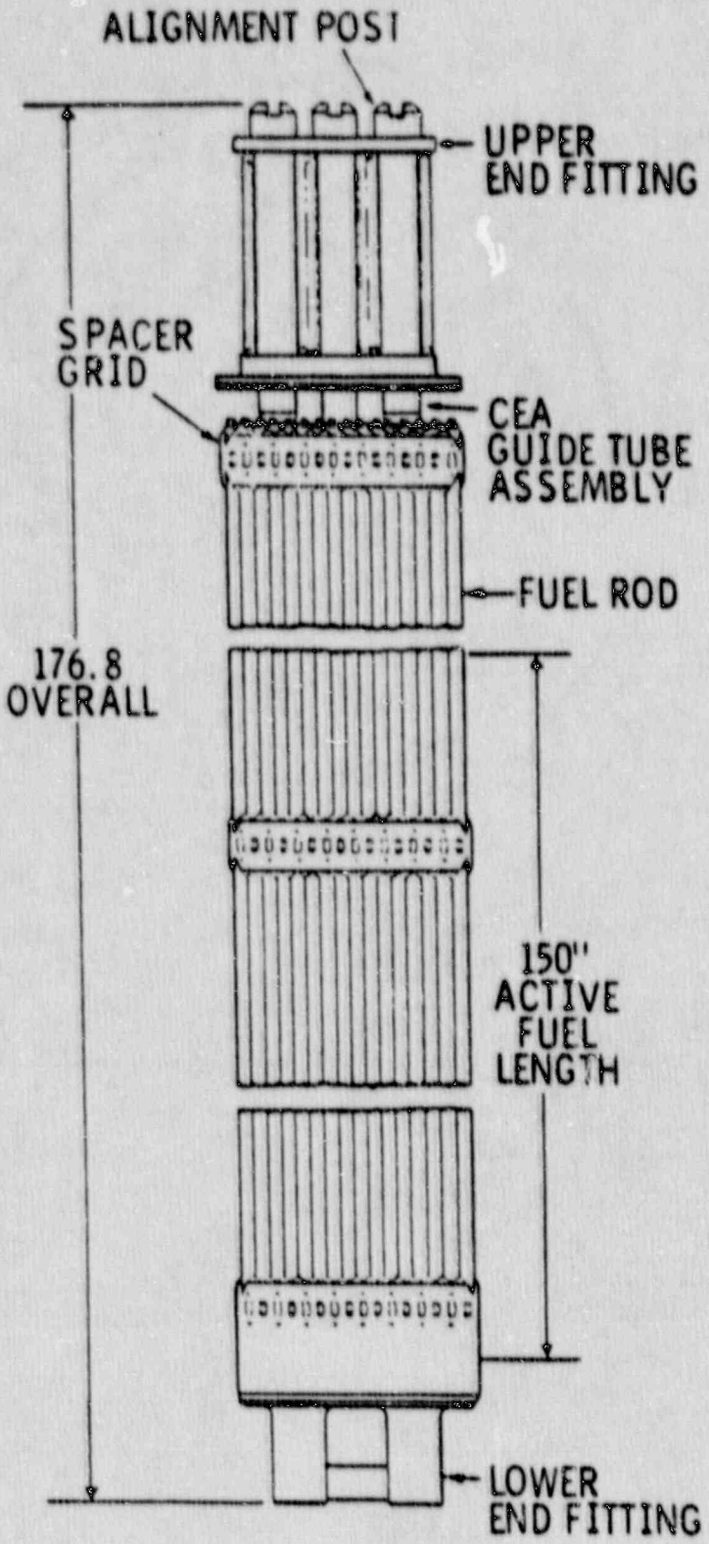


Figure 5.1-4. Longitudinal View of a 16 X 16 Fuel Assembly

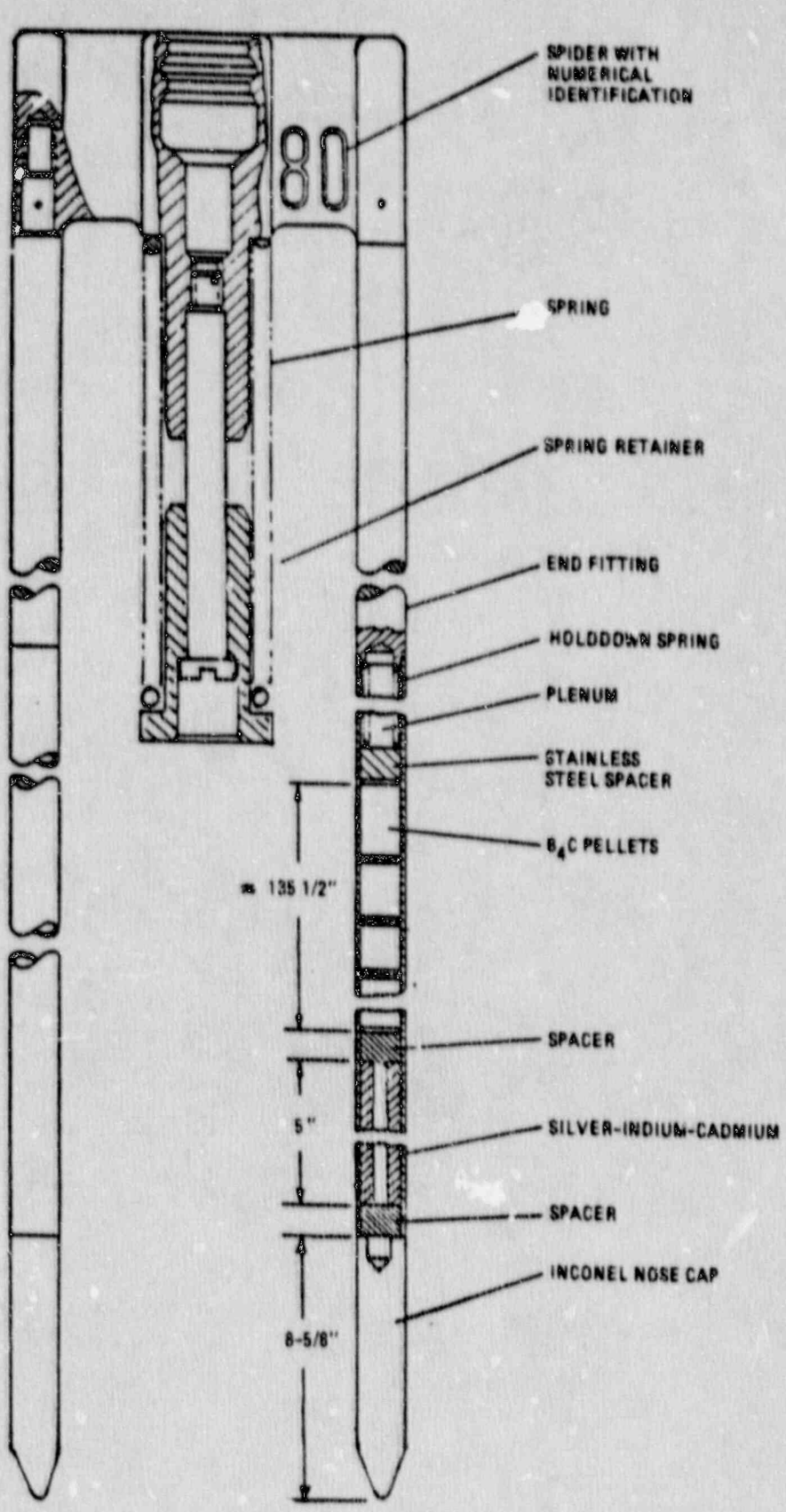


Figure 5.1-5. Typical Control Element Assembly (CEA) for a Pre-System 80 C-E PWR

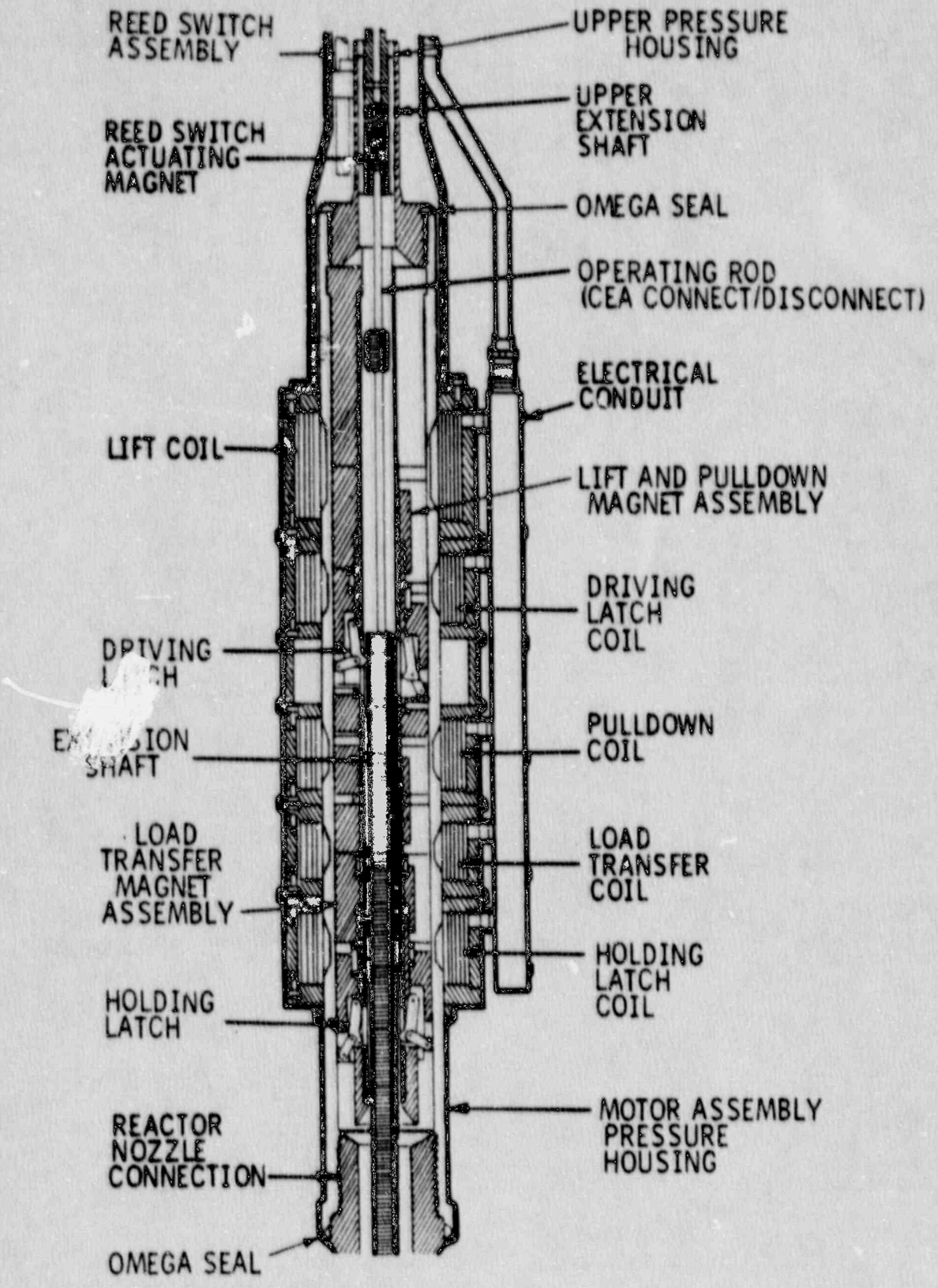


Figure 5.1-6. Details of Combustion Engineering Magnetic Jack Control Element Drive Mechanism

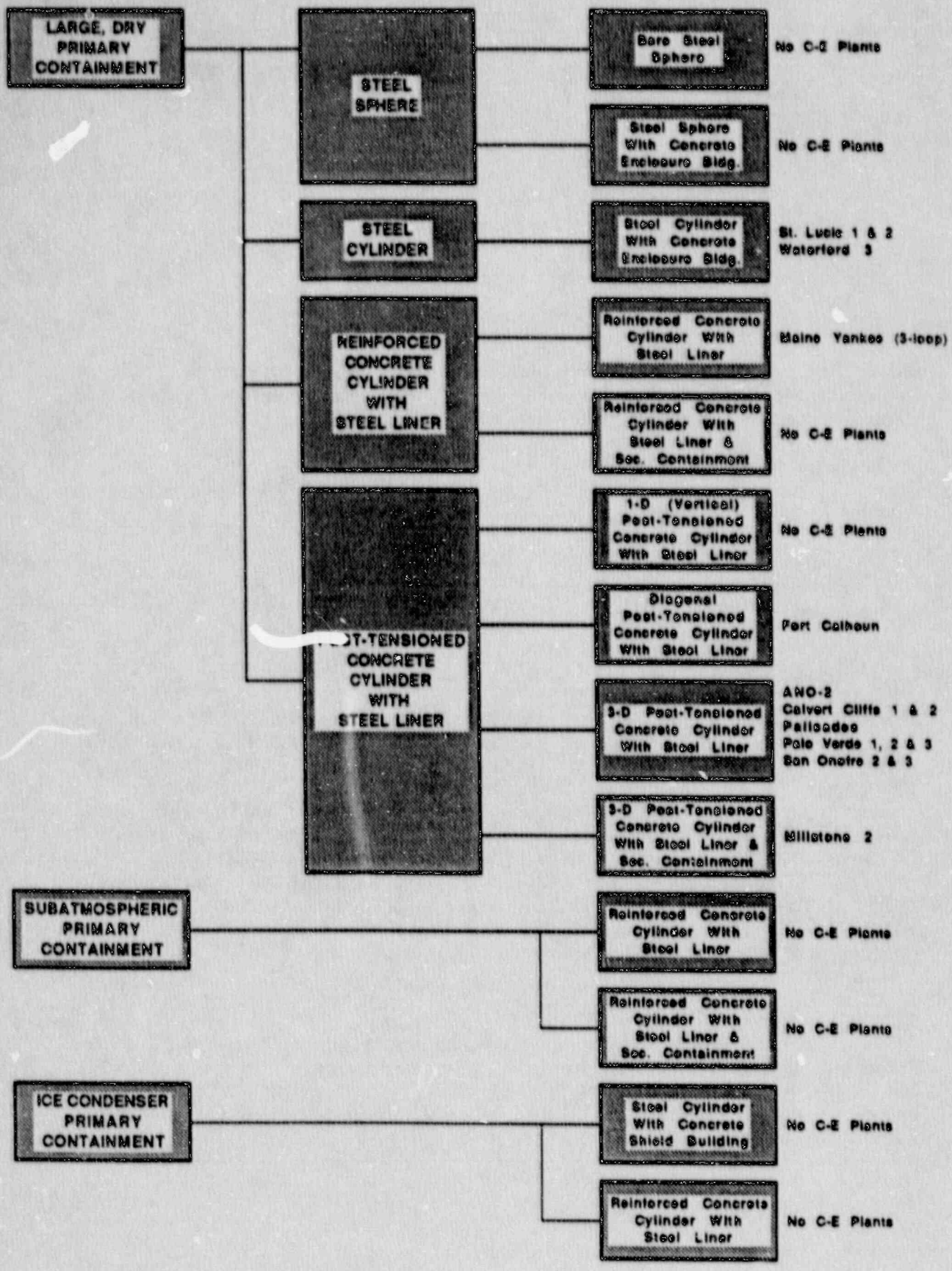


Figure 5.1-7. Distribution of Containment Types for Combustion Engineering Reactors

5.2 2-Loop (Pre-System 80) Combustion Engineering PWRs

Combustion Engineering 2-loop (pre-system 80) plants in the United States include the following:

- Arkansas Nuclear One-2
- Calvert Cliffs 1 and 2
- Fort Calhoun 1
- Millstone 2
- Palisades
- San Onofre 2 and 3
- St. Lucie 1 and 2
- Waterford 3

All of these plants had full power operating licenses as of 2/89 except Palisades which had a provisional operating license. The general arrangement of a Combustion Engineering 2-loop PWR is shown in Figures 5.2-1 to 5.2-3.

5.2.1 Reactor Core and Fuel Assemblies

The 2-loop C-E plants can be grouped into the following categories based on core thermal output and average power density:

- 1420 MWt	Fort Calhoun 1
- 2530 MWt	Palisades
- 2560 to 2815 MWt	Calvert Cliffs 1 and 2, Millstone 2, St. Lucie 1 and 2
- 2815 MWt	ANO-2
- 3390 MWt	San Onofre 2 and 3 and Waterford 3

Plants in the first three groups operate at a relatively lower power density than plants in the latter two groups. Fort Calhoun 1 and Palisades both operate at an average power density of 80 kW/liter. The Fort Calhoun 1 core is comprised of 133 14 x 14 fuel assemblies while Palisades has a unique C-E core design that uses 204 15 x 15 fuel assemblies and cruciform control rods. The Calvert Cliffs 1 and 2, Millstone 2, and St. Lucie 1 and 2 cores all consist of 217 fuel assemblies (either 14 x 14 or 16 x 16 fuel assemblies) and operate at power densities in the range from 78 to 83 kW/liter.

ANO-2 is a unique core design that operates at a relatively high power density of 96.6 kW/liter with 177 16 x 16 fuel assemblies. In the high power group, San Onofre 2 and 3 and Waterford 3 revert to the 217 assembly core with 16 x 16 fuel assemblies that operate at a power density of about 95 kW/liter. The arrangement of the 217 fuel assembly core is shown in Figure 5.1-3.

5.2.2 Reactivity Control System

All pre-System 80 2-loop plants, except Palisades, have CEAs and PLCEAs that consist of five neutron absorber rods assembled in a square array with one rod in the center. The rods are connected to a "spider" structure which is connected to the CEDM extension shaft. The number of CEAs and PLCEAs for selected C-E plants are summarized below:

Plant	Fuel Elements	Full-Length CEAs	Part-Length CEAs
Calvert Cliffs 1 and 2	217	77	8
St. Lucie 1 and 2	217	73	8
ANO-2	177	73	8
San Onofre 2 and 3	217	83	8

A more complete listing is provided in Section 5.5. Some of these plants use B_4C neutron absorber material (St. Lucie and Calvert Cliffs) while others use Ag-In-Cd (silver-indium-cadmium).

Palisades has 41 full-length and 4 part-length cruciform control rods that have Ag-In-Cd neutron absorber material.

5.2.3 Reactor Coolant System

Each RCS loop has one steam generator and two reactor coolant pumps. As illustrated in Figure 5.2-1, a single RCS hot leg supplies each steam generator which divides the return flow into two RCS cold legs. Each cold leg is served by a single reactor coolant pump. All 2-loop pre-System 80 plants have a reactor vessel with a 172 inch inside diameter except Fort Calhoun 1 which has a 140 inch inside diameter reactor vessel.

The general arrangement of the Palisades reactor vessel and internals is shown in Figure 5.2-4, and the arrangement of later 2-loop plants (San Onofre 2 and 3) is shown in Figure 5.2-5. The C-E steam generator is a vertical U-tube design incorporating integral moisture separators and steam dryers as shown in Figure 5.2-6. In the pre-System 80 plants, the feedwater nozzle enters the steam generator above the level of the tube bundle. The configuration of the pressurizer is shown in Figure 5.2-7.

5.2.4 Shutdown Cooling Systems

Shutdown cooling is provided by a multi-mode system that also performs the low-pressure safety injection (LPSI) function as part of the ECCS. The shutdown cooling heat exchangers may also be shared by the containment spray system (i.e. Millstone 2).

5.2.5 Emergency Core Cooling Systems

The C-E ECCS for the 2-loop pre-System 80 plants consists of high-pressure and low-pressure safety injection pump subsystems and four safety injection tanks (accumulators). All of these ECCS subsystems inject into the four RCS cold legs. There is no provision for injecting into the hot legs as in Westinghouse PWRs (i.e., during the latter phase of ECCS operation in the recirculation mode).

The shutoff head of the HPSI pumps is less than RCS normal operating pressure, therefore, the RCS must depressurize before these pumps can provide any makeup. At full RCS pressure, makeup capacity is limited to that provided by three positive displacement charging pumps that are not part of the ECCS. The HPSI pumps can be aligned to take a suction on the containment sump.

5.2.6 Containment

All 2-loop pre-System 80 C-E PWRs have large, dry cylindrical containments. The types of construction used in these containments is summarized below.

Containment Construction	Applicable Plants
Steel cylinder with concrete enclosure building	St. Lucie 1 and 2 Waterford 3
Diagonal post-tensioned concrete cylinder with a steel steel liner	Fort Calhoun 1
Three-dimension post- tensioned concrete cylinder with a steel liner	ANO-2 Calvert Cliffs 1 and 2 Palisades San Onofre 2 and 3
Three-dimension post- tensioned concrete cylinder with a steel liner and a secondary containment	Millstone 2

The St. Lucie steel containment and enclosure building are shown in Figures 5.2-8 and 5.2-9. The San Onofre 2 and 3 three dimension post-tensioned concrete containment is shown in Figures 5.2-10 and 5.2-12.

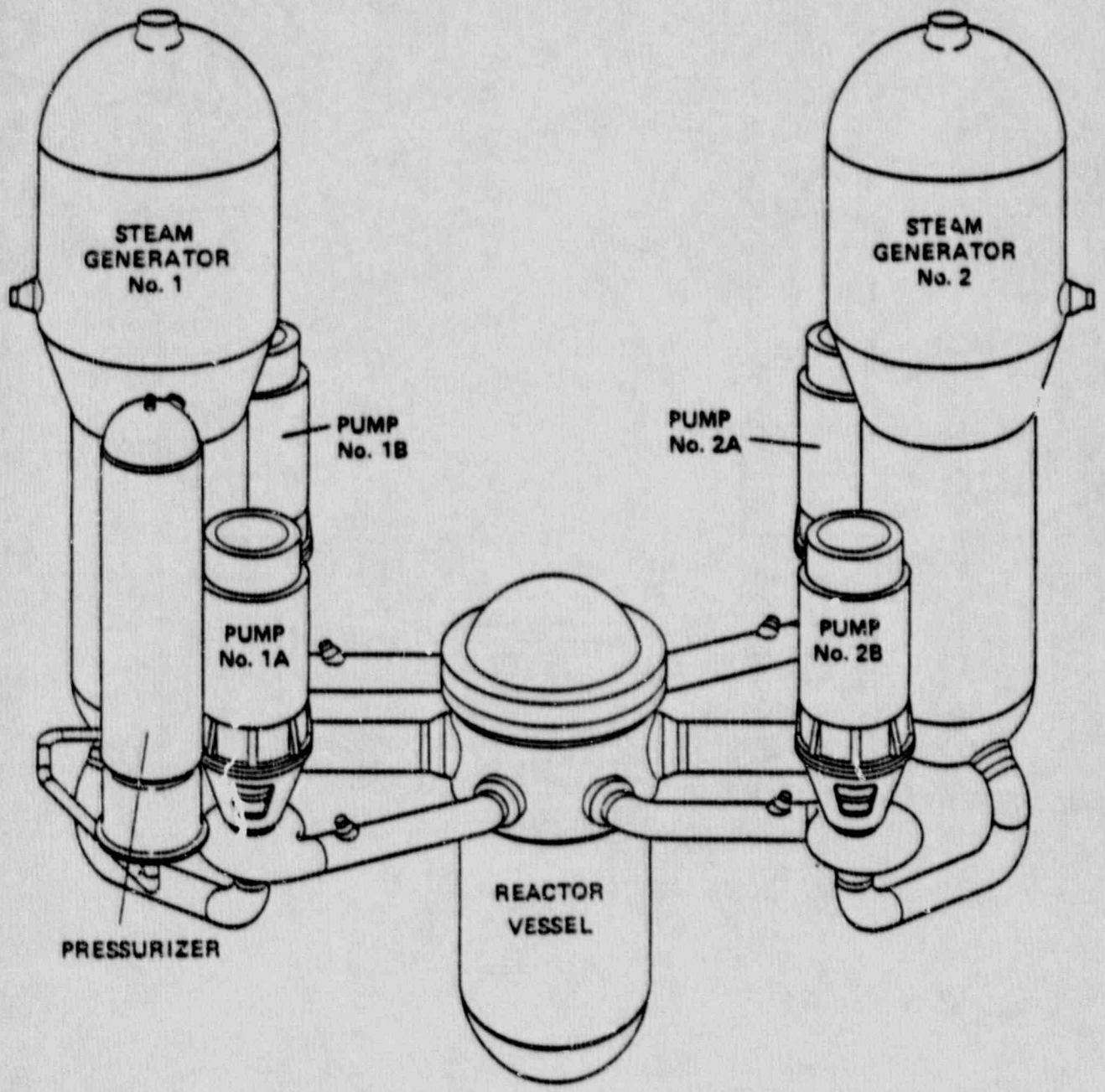


Figure 5.2-1. Combustion Engineering 2-Loop (Pre-System 80) NSSS

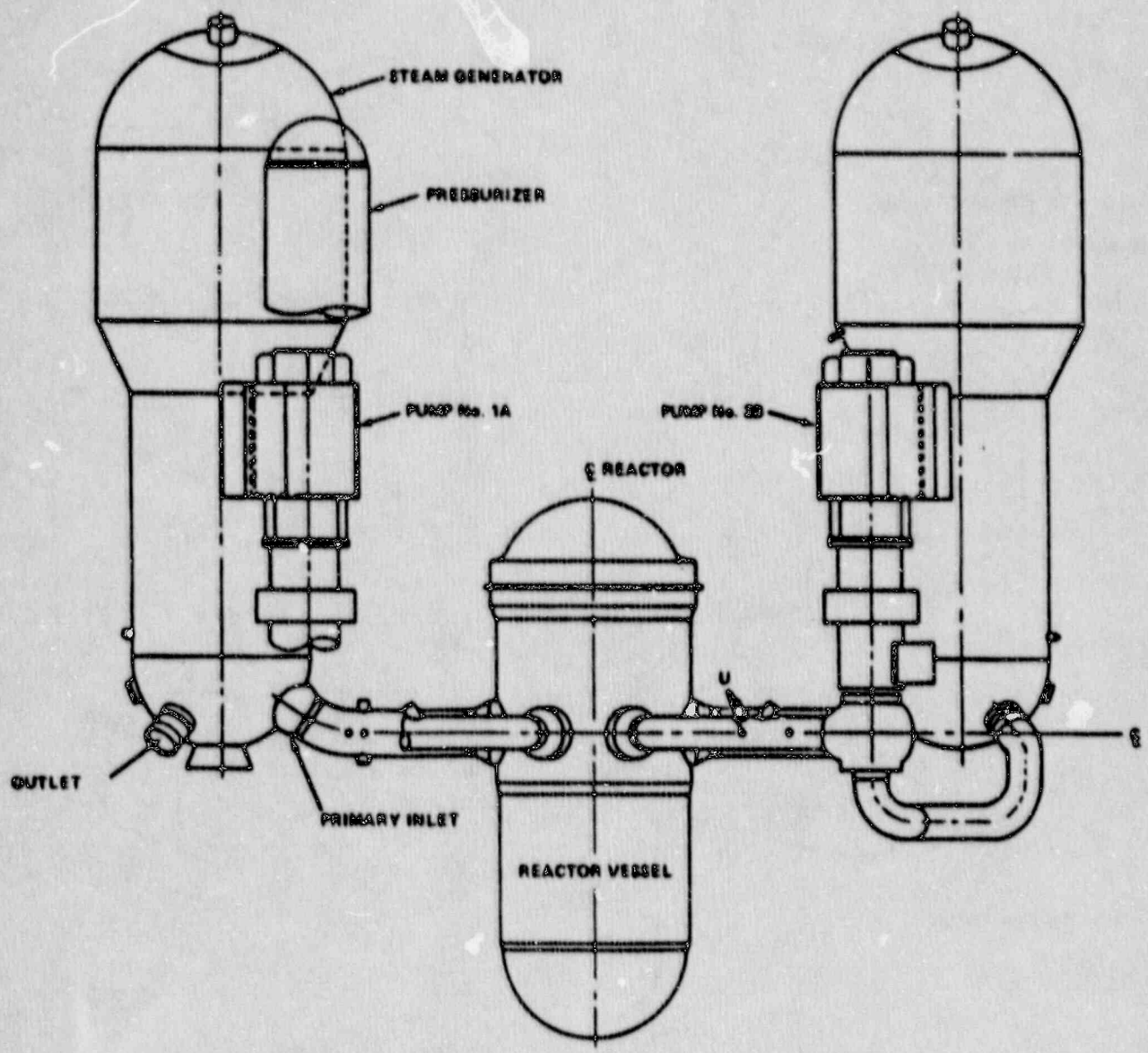


Figure 5.2-2. Combustion Engineering 2-Loop (Pre-System 80) NSSS Side View

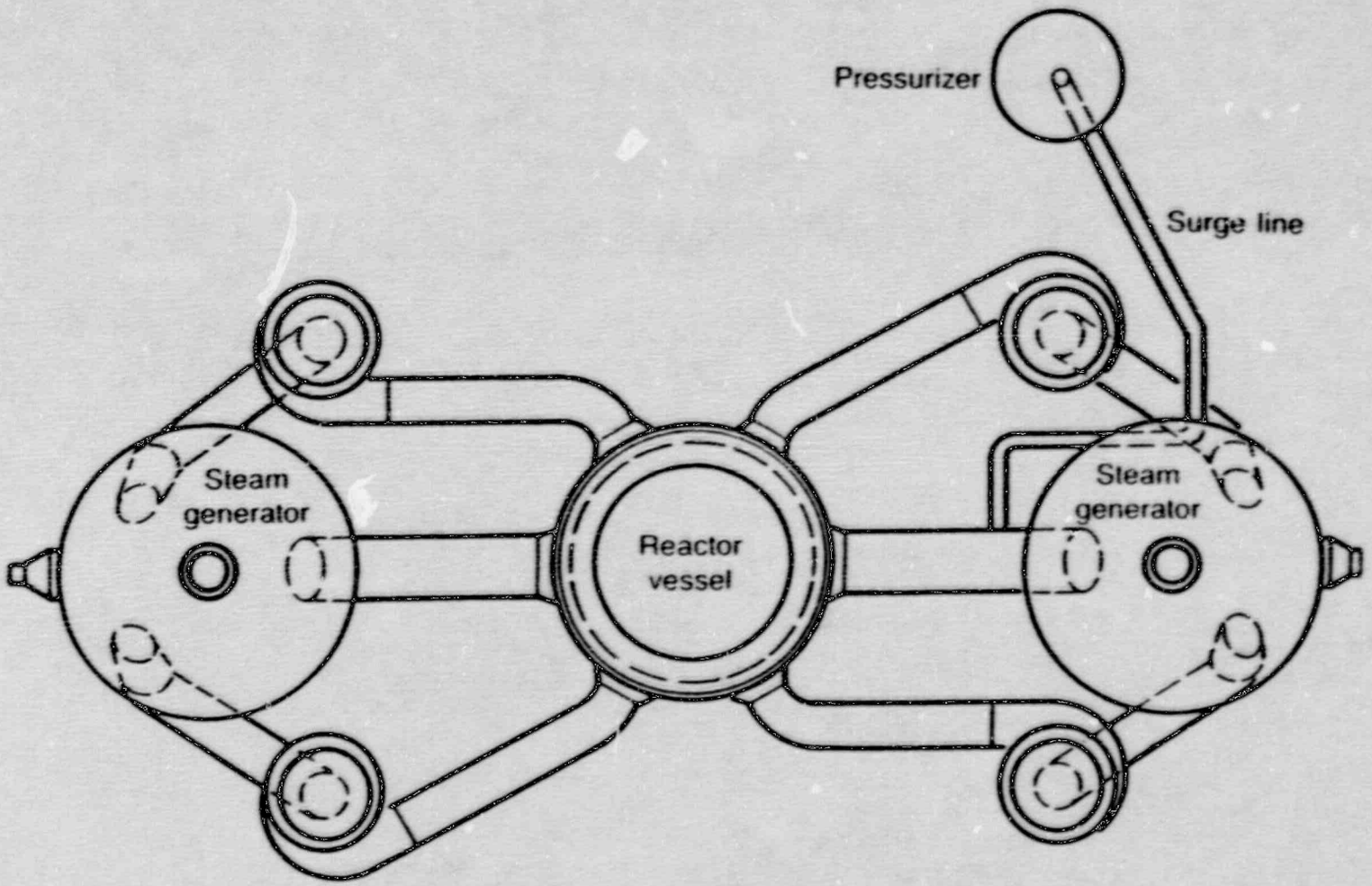


Figure 5.2-3. Combustion Engineering 2-Loop (Pre-System 80) NSSS Plan View

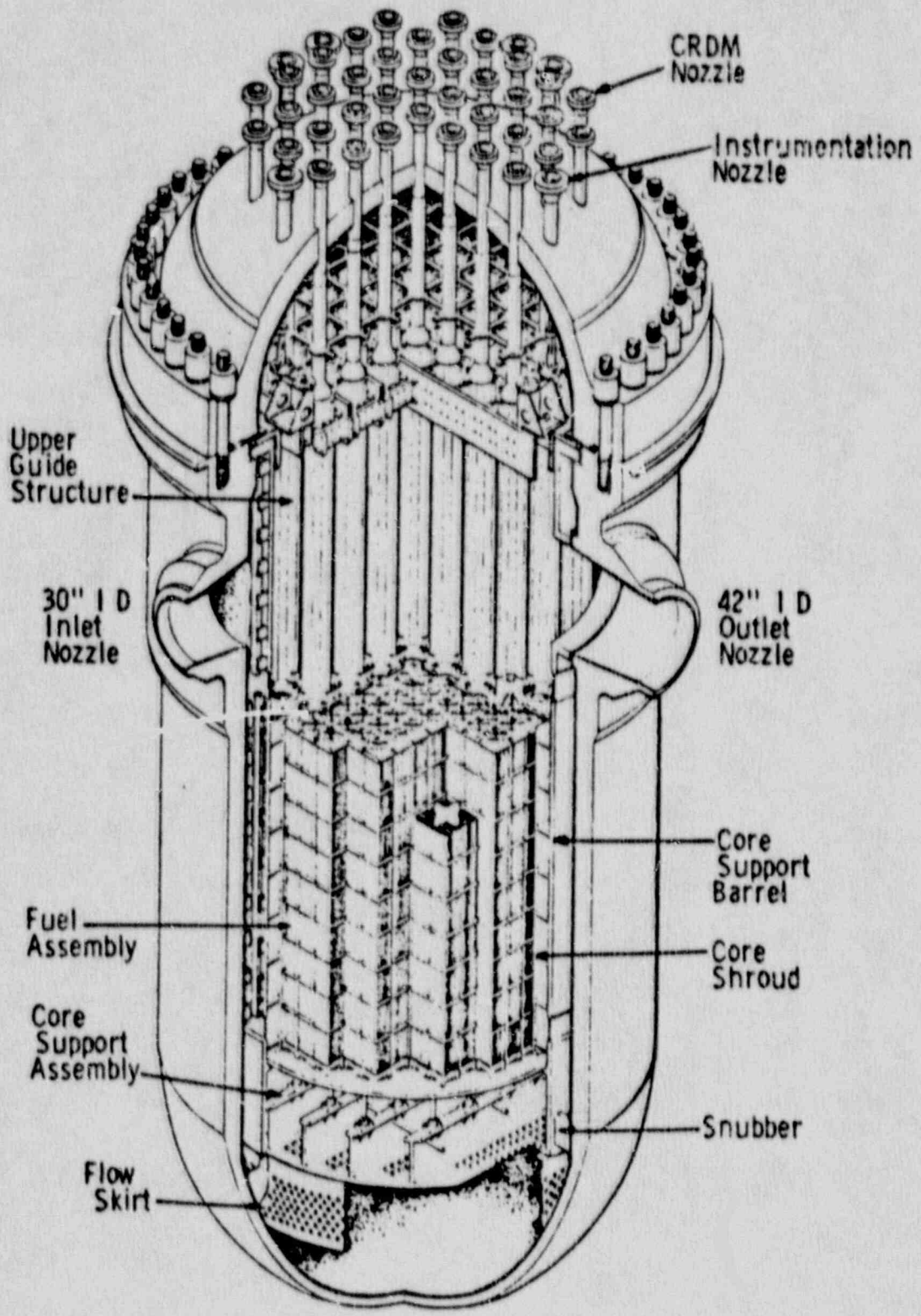


Figure 5.2-4. General Arrangement of the Palisades Reactor Vessel and Internals

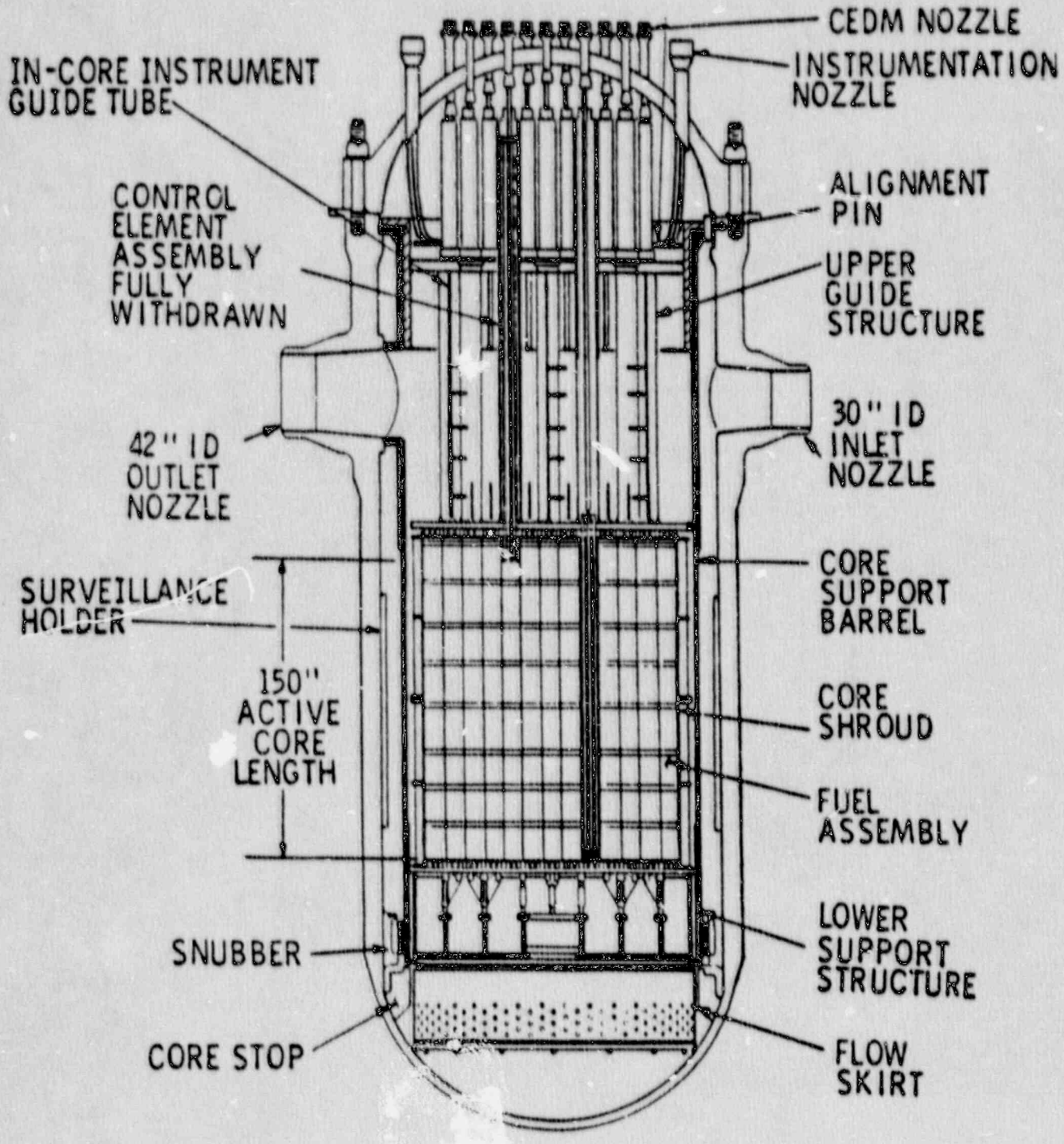
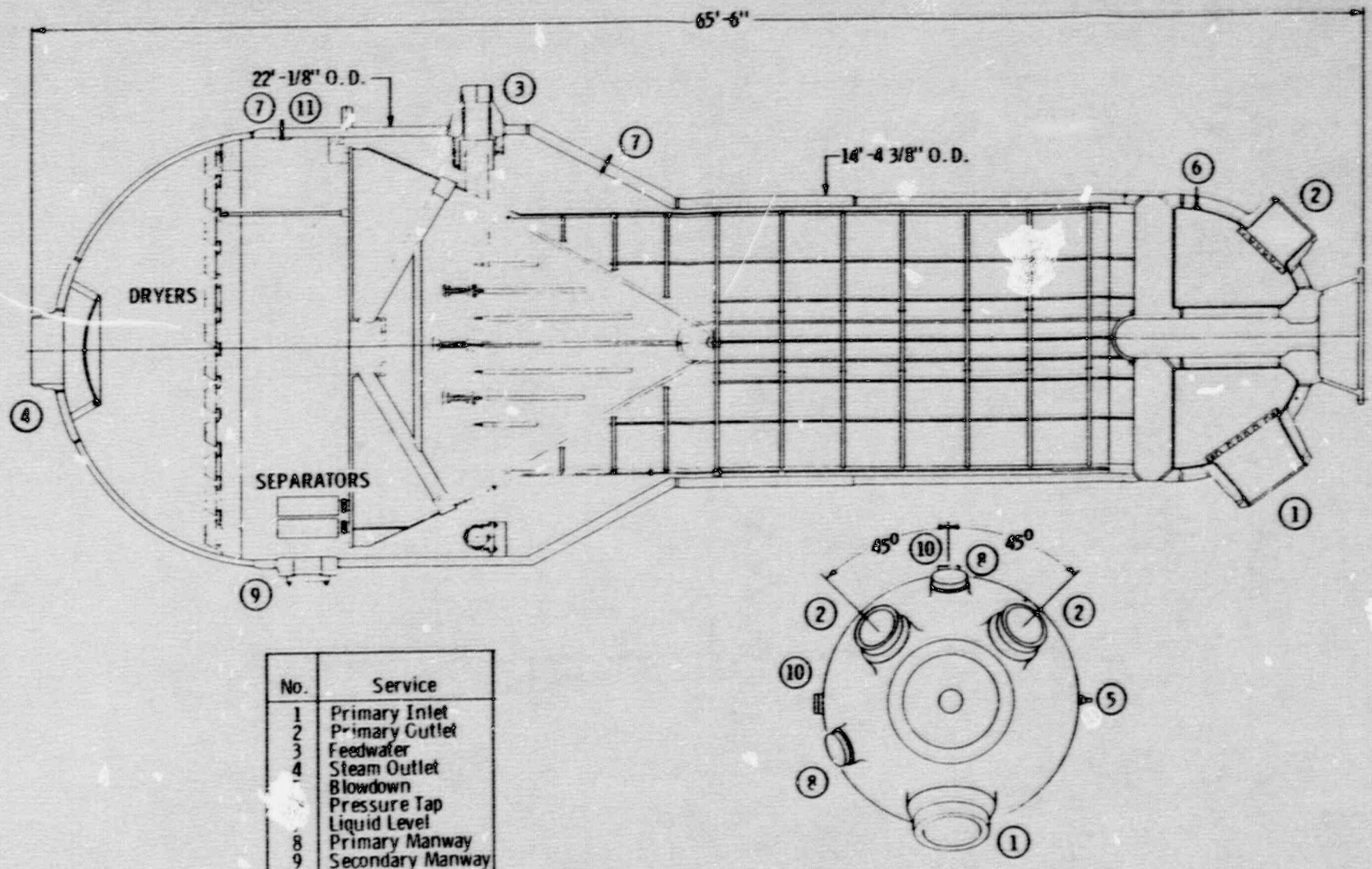
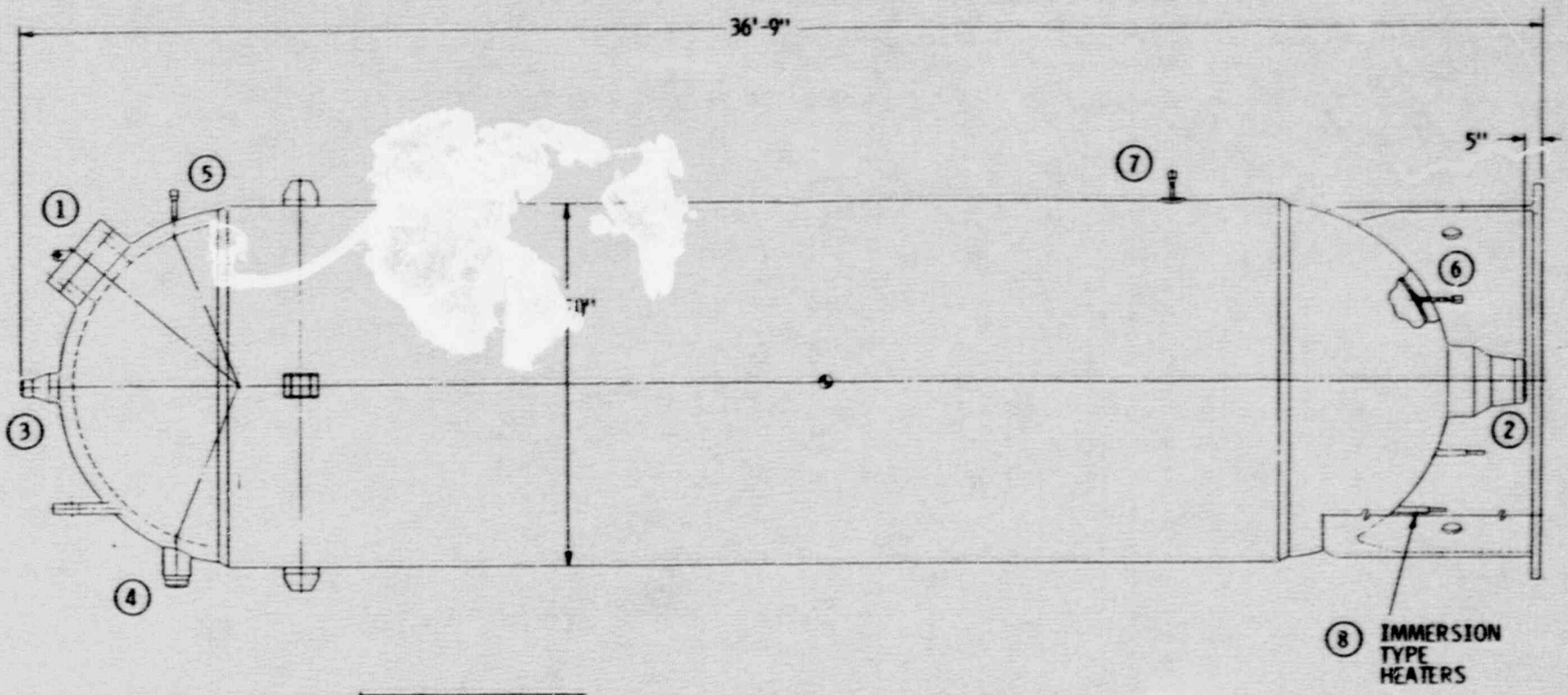


Figure 5.2-5. General Arrangement of the San Onofre 2/3 Reactor Vessel and Internals



No.	Service
1	Primary Inlet
2	Primary Outlet
3	Feedwater
4	Steam Outlet
[illegible]	Blowdown
[illegible]	Pressure Tap
[illegible]	Liquid Level
8	Primary Manway
9	Secondary Manway
10	Handhole
11	Sample

Figure 5.2-6. Combustion Engineering U-Tube Steam Generator (Pre-System 80)



No.	Service
1	Manway
2	Surge
3	Spray
4	Safety Valve
5	Instrument
6	Instrument
7	Temperature
8	Heater

Figure 5.2-7. Combustion Engineering Pressurizer (Pre-System 80)

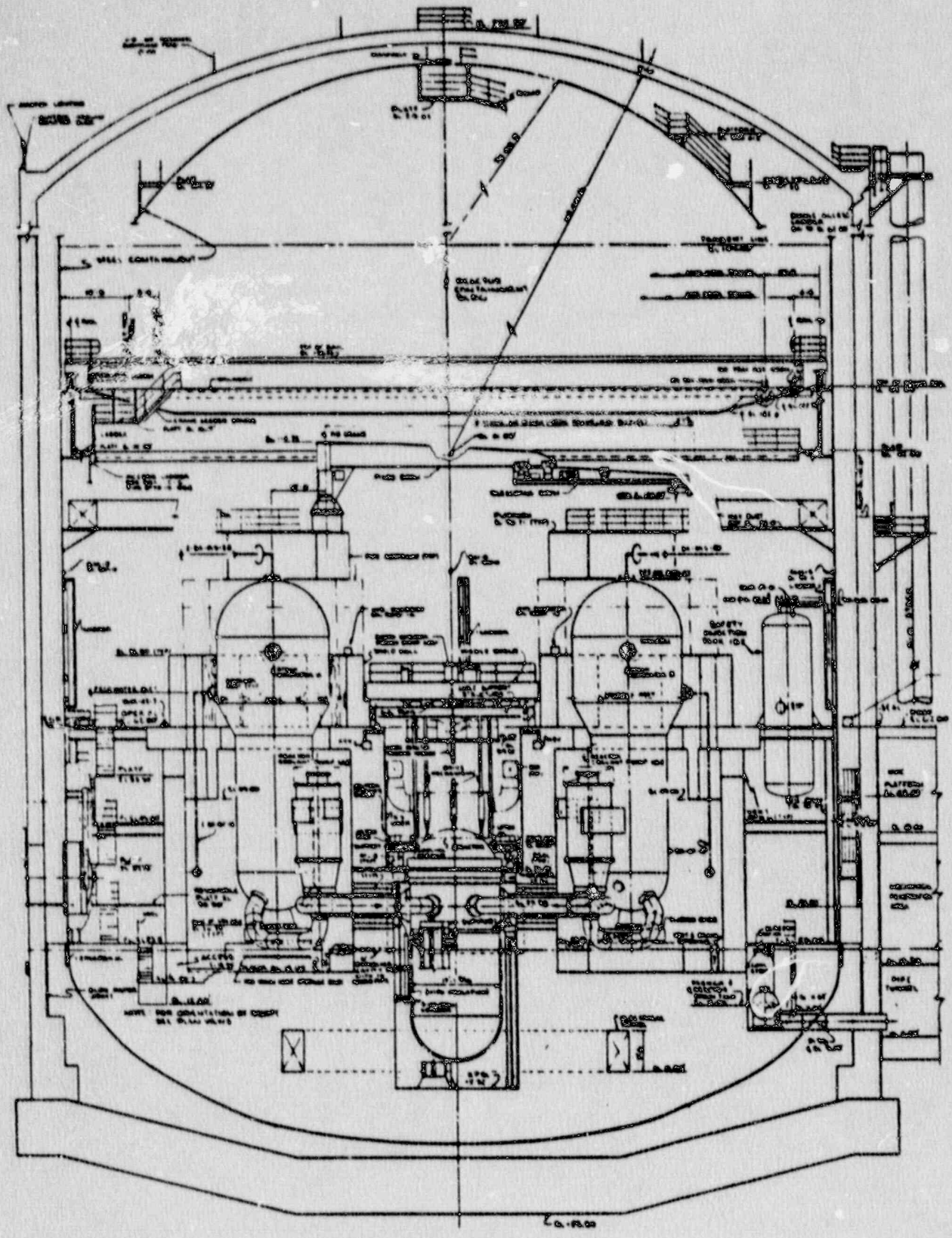


Figure 5.2-8. Section View of the St. Lucie Large, Dry Containment (Steel Cylinder with Concrete Enclosure Building)

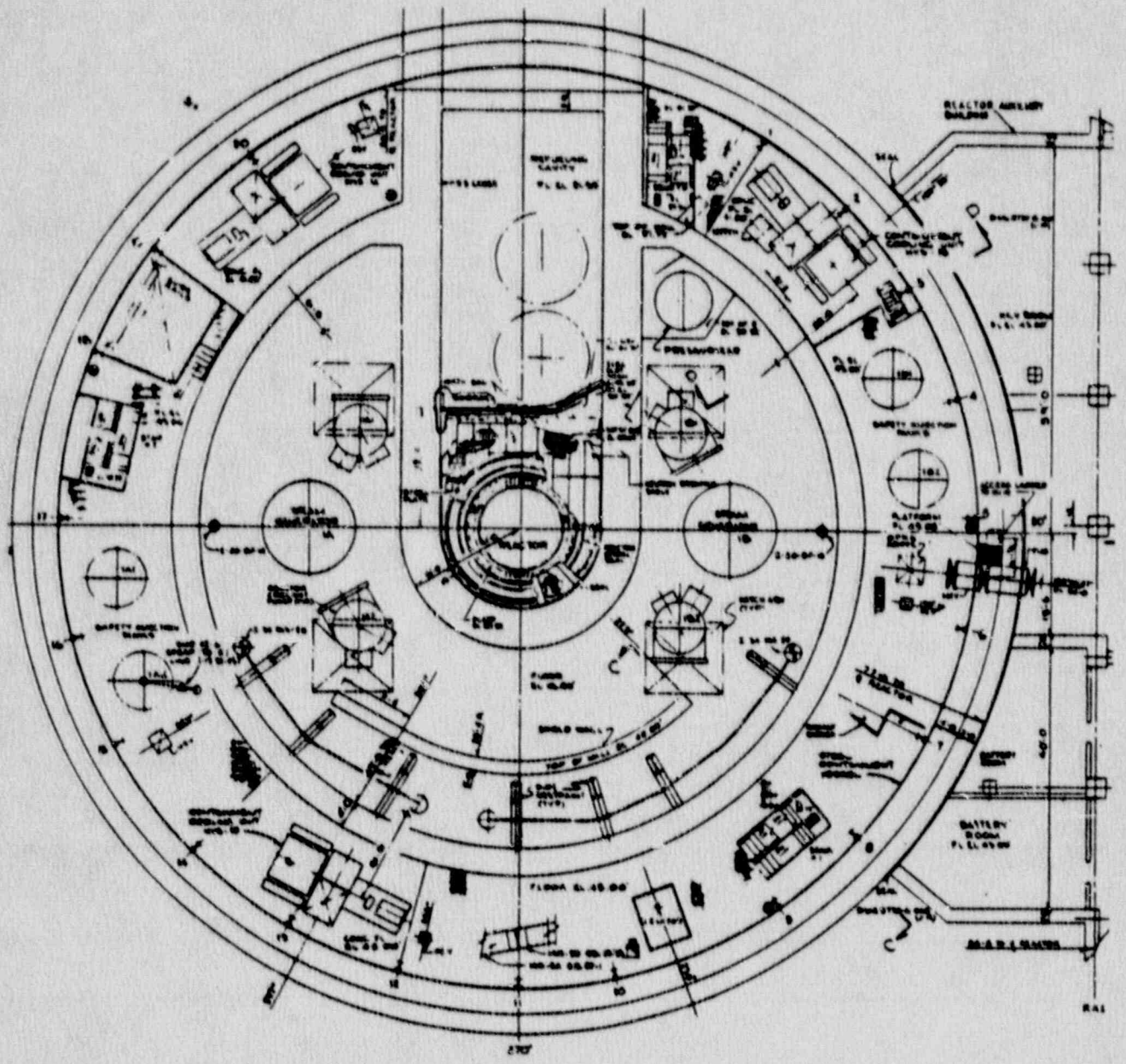


Figure 5.2-9. Plan View of the St. Lucie Large, Dry Containment (Steel Cylinder with Concrete Enclosure Building)

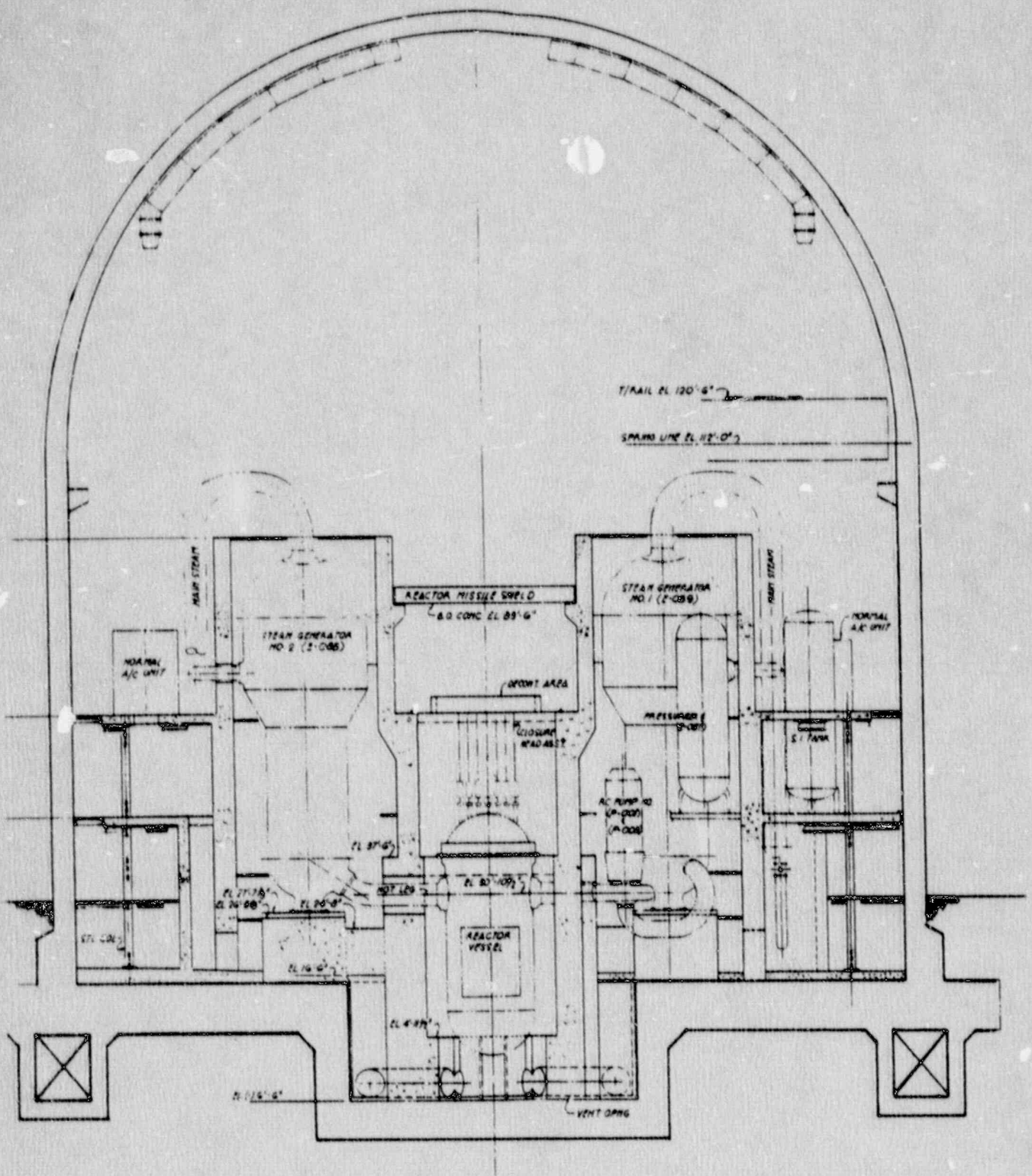


Figure 5.2-10. Section View of the San Onofre 2/3 Large, Dry Containment (3-D Post-Tensioned Concrete)

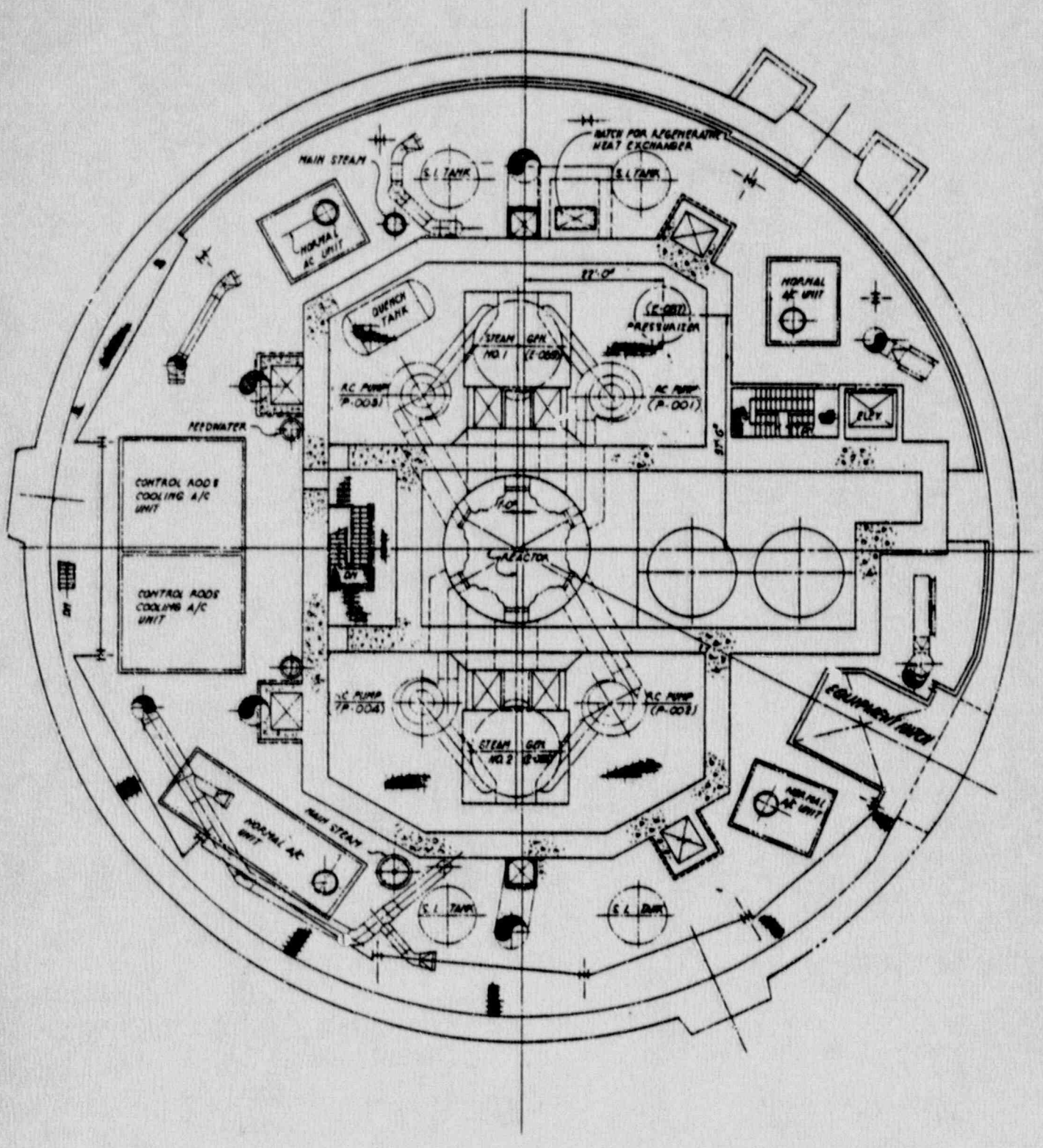


Figure 5.2-11. Plan View of the San Onofre 2/3 Large, Dry Containment (3-D Post-Tensioned Concrete)

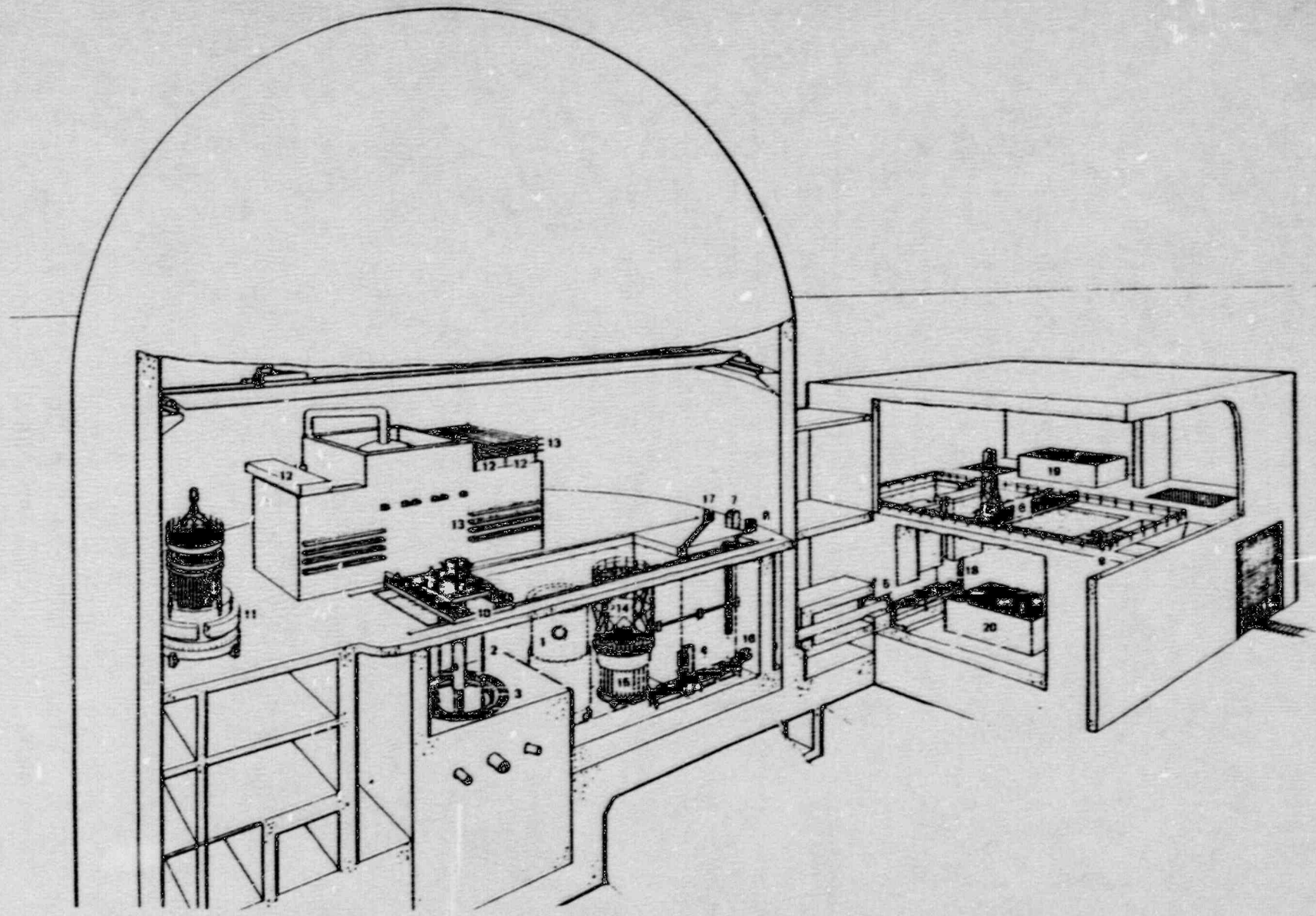


Figure 5.2-12. Cutaway View of the San Onofre 2/3 Containment and Fuel Storage Building Showing Refueling Facilities

5.3 2-Loop System 80 Combustion Engineering PWRs

Palo Verde 1, 2, and 3 are the only System 80 C-E PWRs built in the United States. The general arrangement of the System 80 2-loop PWR is shown in Figures 5.3-1 and 5.3-2.

5.3.1 Reactor Core and Fuel Assemblies

The Palo Verde core is rated at 3800 MWt and operates at a power density of 95.6 kW/liter. The core is comprised of 241 16 x 16 fuel assemblies, with an active core length of 150 inches. The general arrangement of this core is shown in Figure 5.3-3. The core support structure has been modified to allow the core to sit lower in the reactor vessel than in prior C-E plants.

Palo Verde is designed for "rodless" refueling in which all CEAs are removed after the boron concentration in the primary coolant has been increased to about 2150 ppm. All CEAs are removed with the upper guide structure, without unlatching the CEA extension shafts from the drive mechanisms.

Palo Verde also has bottom-mounted fixed and movable in-core instrumentation systems combined with a core monitoring computer system to provide on-line determination of core performance.

5.3.2 Reactivity Control System

Reactivity control is provided by new control element assemblies (CEAs) with 4, 8, or 12 control rods which are connected by a "spider" structure to the CEDM extension shaft. The 8 and 12 element CEAs are shown in Figure 5.3-4. The 4-rod CEAs are used for power shaping.

A revised upper guide structure with a barrel-calandria design isolates all control rods from the primary coolant cross-flow in the upper planum of the reactor vessel. The control rods move through the tubes of the calandria.

5.3.3 Reactor Coolant System

The Palo Verde System 80 RCS is comparable to the RCS of other 2-loop C-E PWRs with the following changes:

- Larger reactor vessel (diameter and length)
- Extended upper guide structure barrel-calandria
- Provisions for bottom entry in-core instrumentation
- Larger steam generators with economizer

The general arrangement of the reactor vessel and internals is shown in Figure 5.3-5. As shown in Figure 5.3-6, feedwater enters both the downcomer and the economizer of the System 80 steam generator. The steam generators in earlier C-E plants do not have an economizer. The System 80 pressurizer is shown in Figure 5.3-7.

5.3.4 Shutdown Cooling Systems

Shutdown cooling is provided by a multi-mode system that also performs the low-pressure safety injection (LPSI) function as part of the ECCS. The shutdown cooling heat exchangers can also be shared by the containment spray system.

5.3.5 Emergency Core Cooling Systems

The Palo Verde ECCS is comparable to the ECCS in the 2-loop pre-System 80 plants. The shutoff head of the HPSI pumps is less than RCS normal operating pressure, therefore, the RCS must depressurize before these pumps can provide any makeup. At full RCS pressure, makeup capacity is limited to that provided by three positive displacement charging pumps that are not part of the ECCS. The HPSI pumps can be aligned to take a suction on the containment sump.

5.3.6 Containment

The Palo Verde 1, 2, and 3 plants have a large, dry cylindrical containment constructed of three-dimension post-tensioned concrete with a steel liner. The general arrangement of the Palo Verde containment is shown in Figure 5.3-8.

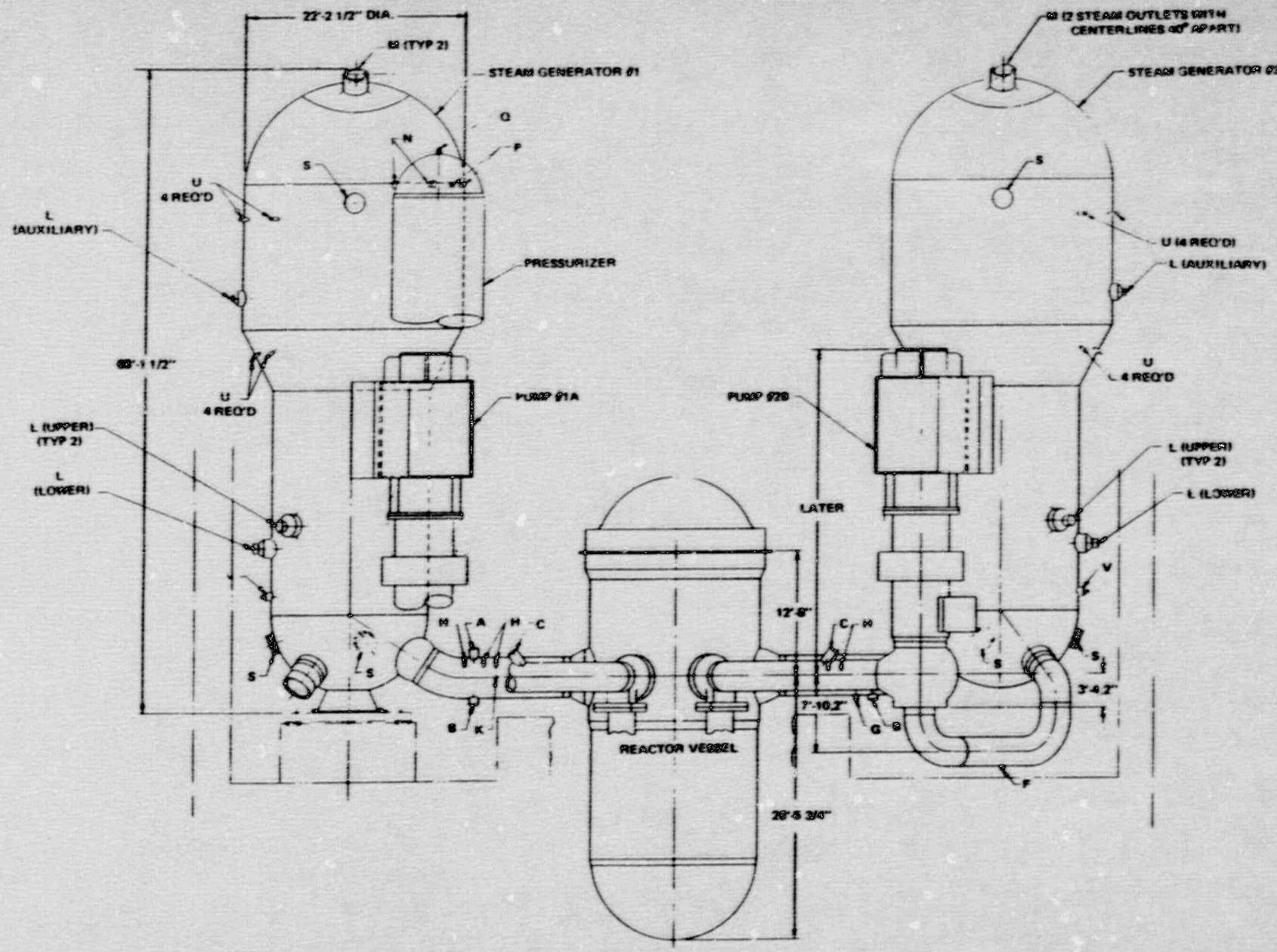


Figure 5.3-1. Combustion Engineering 2-Loop System 80 NSSS Side View

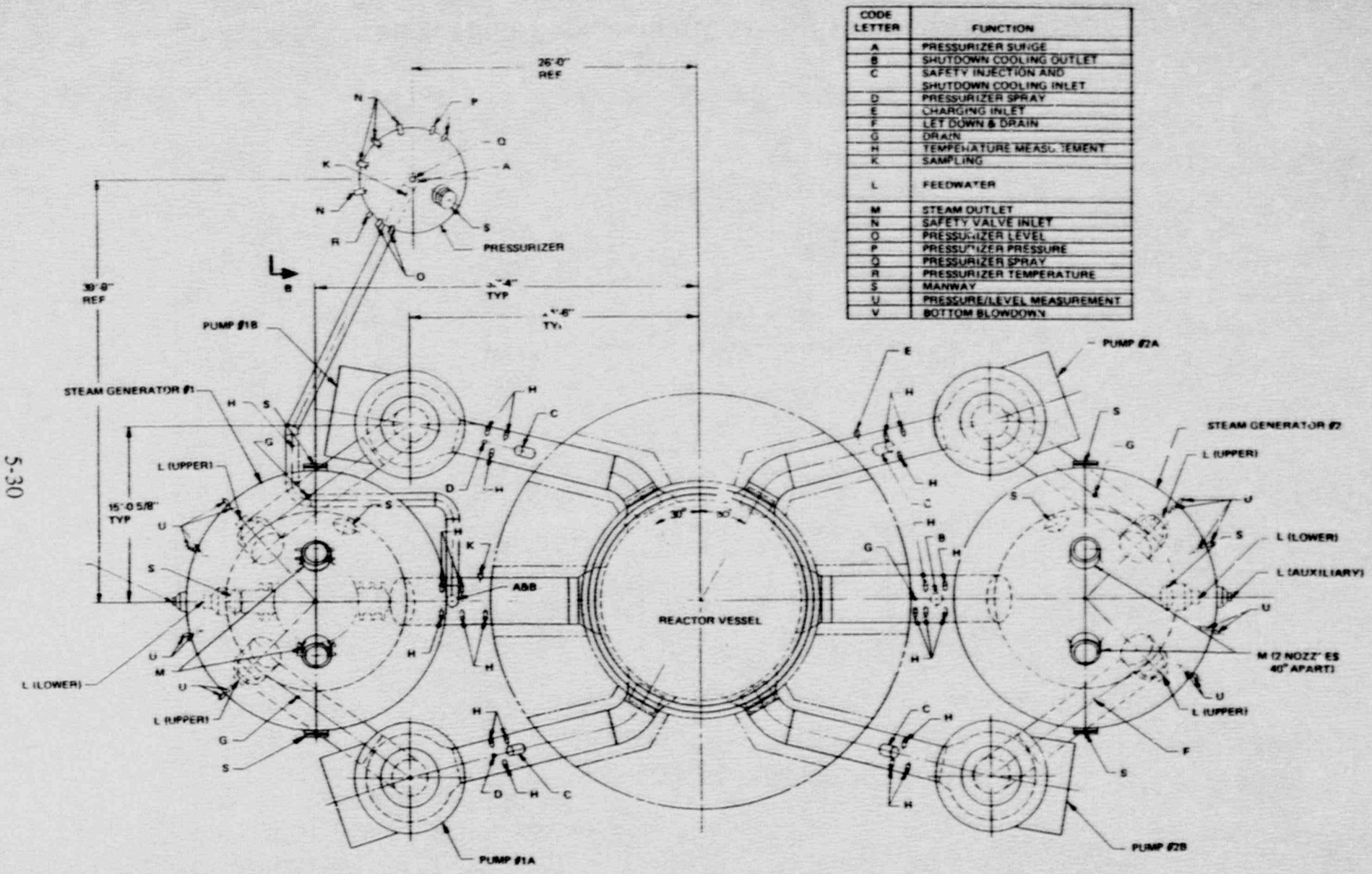


Figure 5.3-2. Combustion Engineering 2-Loop System 80 NSSS Plan View

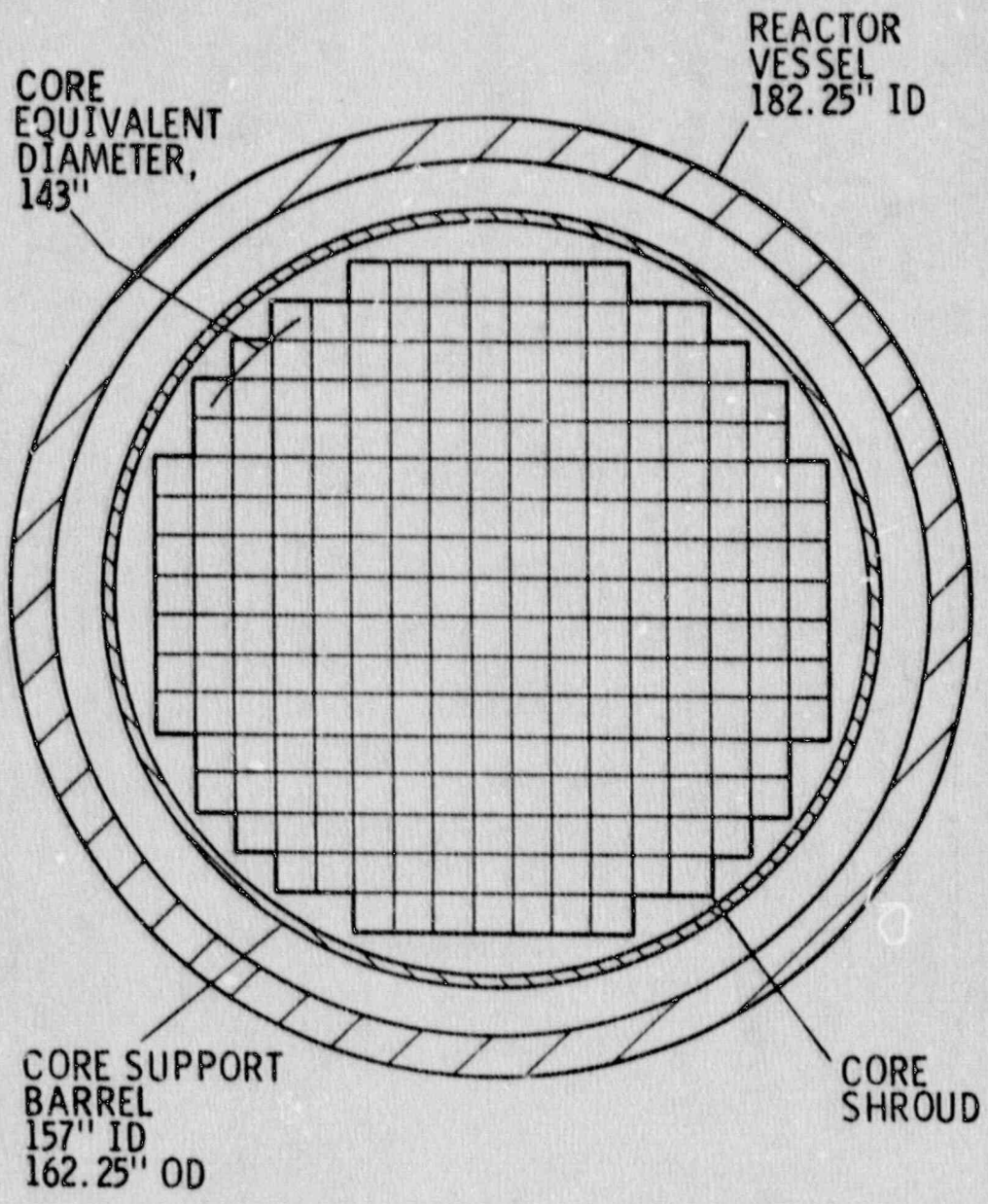


Figure 5.3-3. General Arrangement of the System 80 241 Fuel Assembly Core

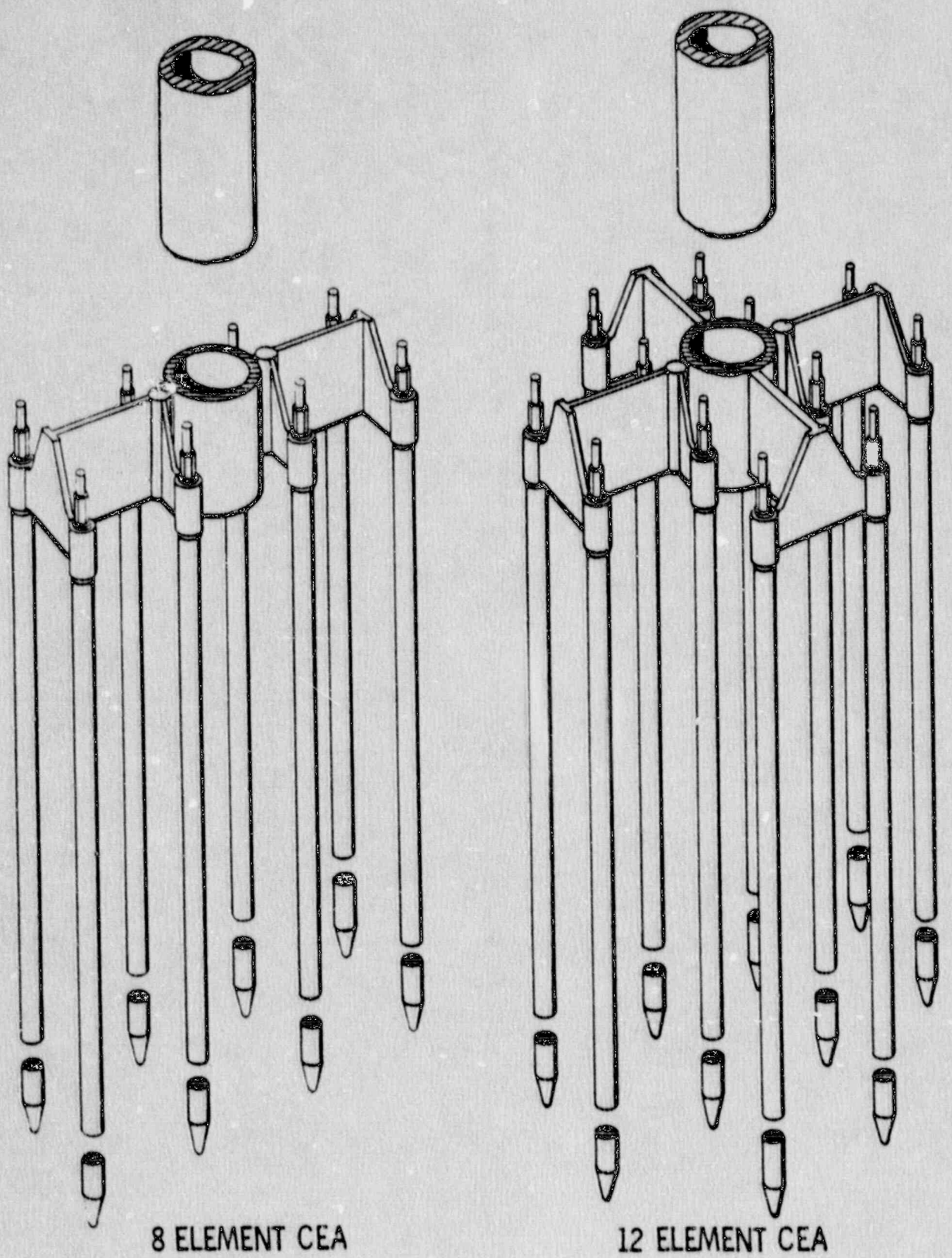


Figure 5.3-4. Combustion Engineering 8 and 12 Element System 80 CEAs

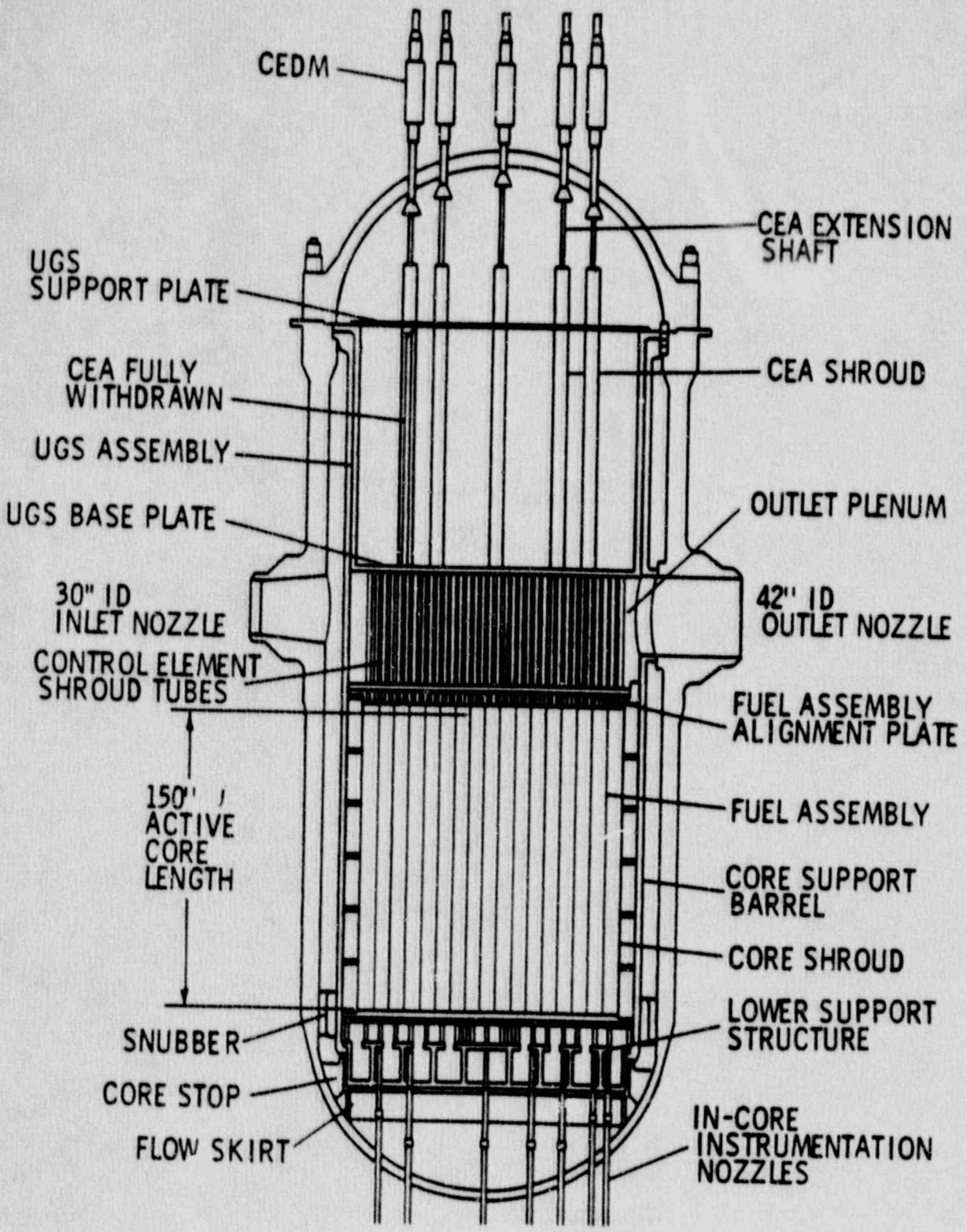


Figure 5.3-5. General Arrangement of the System 80 Reactor Vessel and Internals

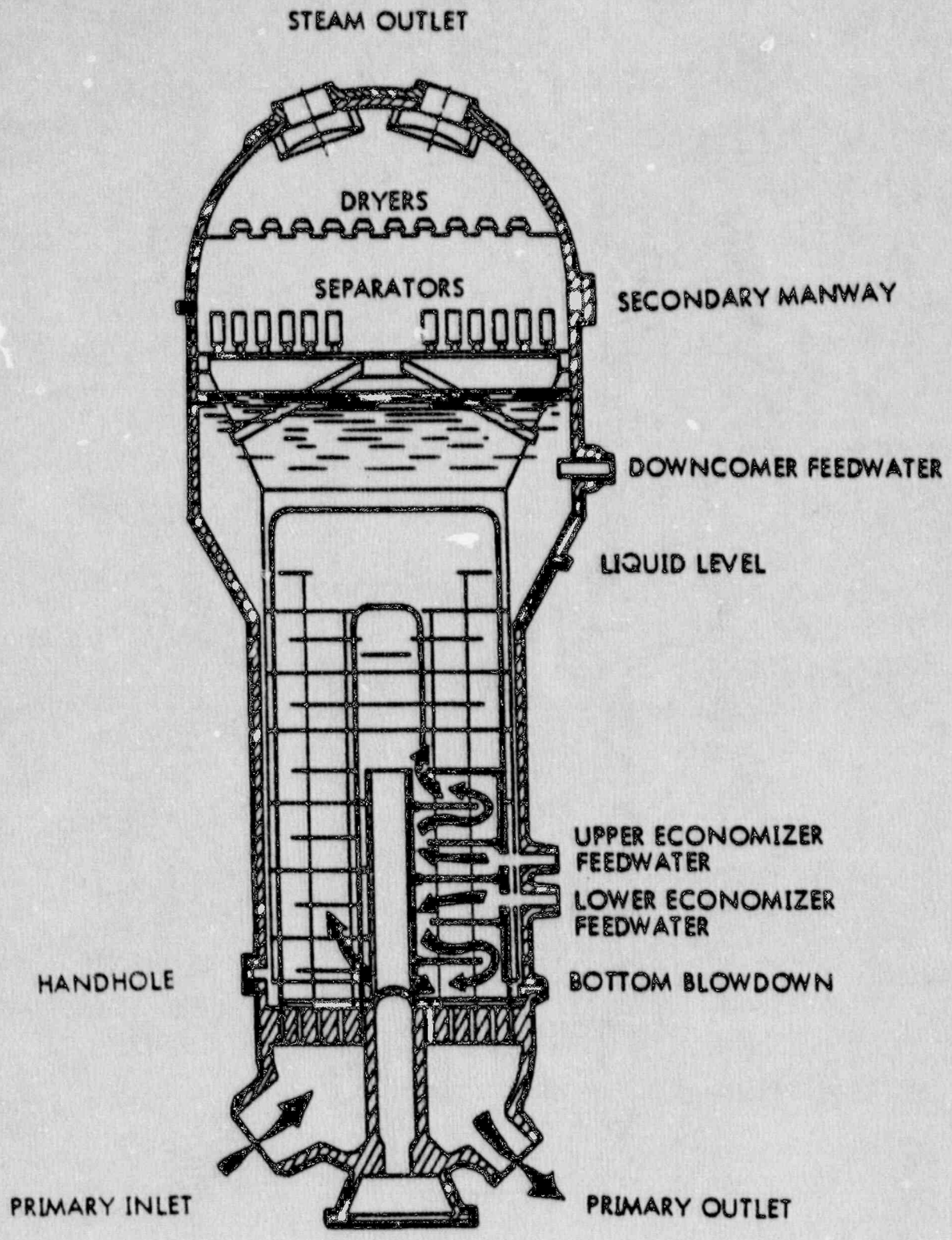


Figure 5.3-6. Combustion Engineering System 80 U-Tube Steam Generator

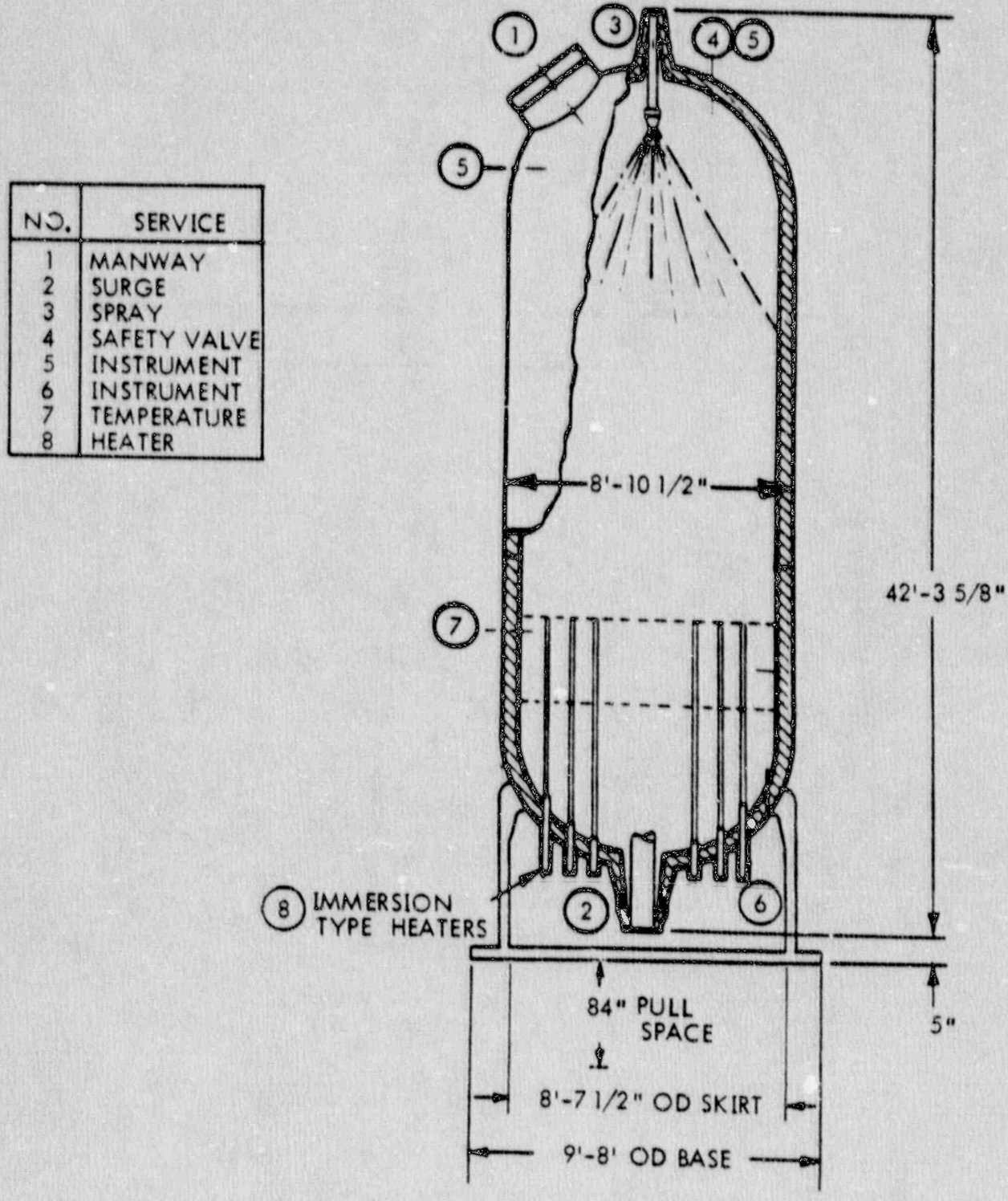


Figure 5.3-7. Combustion Engineering System 80 Pressurizer

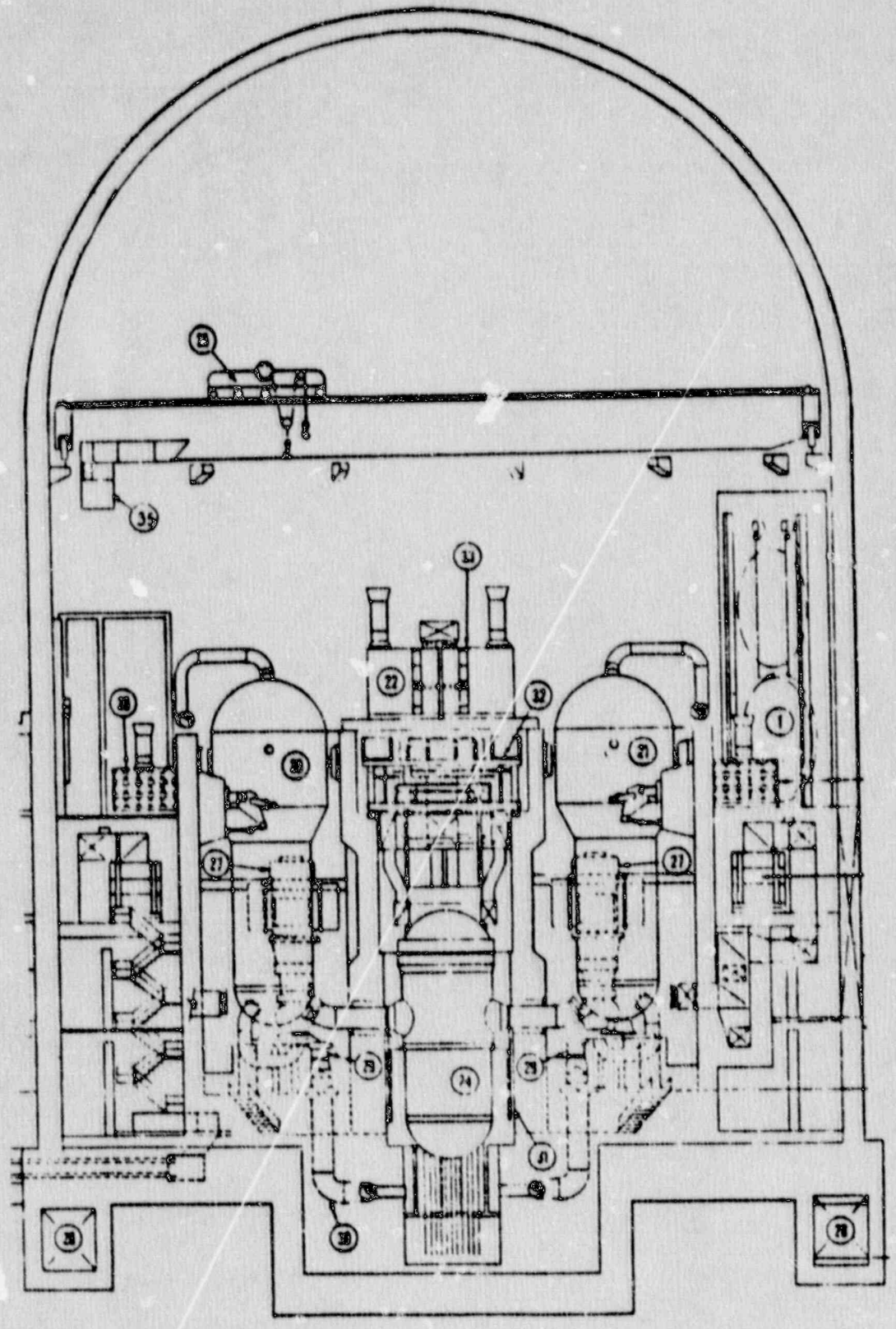


Figure 5.3-8. Section View of the Palo Verde Large, Dry Containment (3-D Post-Tensioned Concrete)

5.4 3-Loop Combustion Engineering PWR

Maine Yankee is the only 3-loop C-E PWR built in the United States.

5.4.1 Reactor Core and Fuel Assemblies

The Maine Yankee core is rated at 2440 MWt , with an approved "stretch" rating of 2630 MWt. The core is comprised of 217 14 x 14 fuel assemblies, with an active core length of 136.7 inches. Core average power density is 74.6 kW/liter.

5.4.2 Reactivity Control System

The Maine Yankee reactivity control system is comparable to the system in Combustion Engineering pre-System 80 2-loop plants.

5.4.3 Reactor Coolant System

In the Maine Yankee RCS, each primary loop is comprised of one hot leg that enters the bottom of a U-tube steam generator, and one cold leg that return flow to the reactor vessel. One reactor coolant pump is in each cold leg, for a total of three reactor coolant pumps. The pressurizer is connected to one of the hot legs. This arrangement is generally similar to that of a Westinghouse 3-loop plant.

The Maine Yankee reactor vessel is shown in Figure 5.4-1, the U-tube steam generator in Figure 5.4-2, and the pressurizer in Figure 5.4-3.

5.4.4 Shutdown Cooling Systems

Shutdown cooling is provided by a multi-mode system that also performs the low-pressure safety injection (LPSI) function as part of the ECCS. The shutdown cooling heat exchangers can also be shared by the containment spray system.

5.4.5 Emergency Core Cooling Systems

The Maine Yankee ECCS consists of high-pressure and low-pressure safety injection pump subsystems and three safety injection tanks (accumulators). All of these ECCS subsystems inject into the three RCS cold legs. There is no provision for injecting into the hot legs as in Westinghouse PWRs (i.e., during the latter phase of ECCS operation in the recirculation mode).

The Main Yankee ECCS is significantly different than all other C-E plants in that the HPSI pumps also serve as charging pumps and are capable of providing makeup to the RCS at the PORV setpoint pressure. The HPSI pumps at Maine Yankee cannot be aligned to take a suction on the containment sump. A recirculation flow path can be established with the LPSI and HPSI pumps operating in tandem as in Westinghouse PWRs. In this alignment, the LPSI pumps take a suction on the containment sump and deliver water to the suction of the HPSI pumps which complete the injection path into the RCS.

5.4.5 Containment

The Maine Yankee plant has a large, dry cylindrical containment constructed of reinforced concrete with a steel liner. The general arrangement of the Maine Yankee containment is shown in Figures 5.4-4 to 5.4-6.

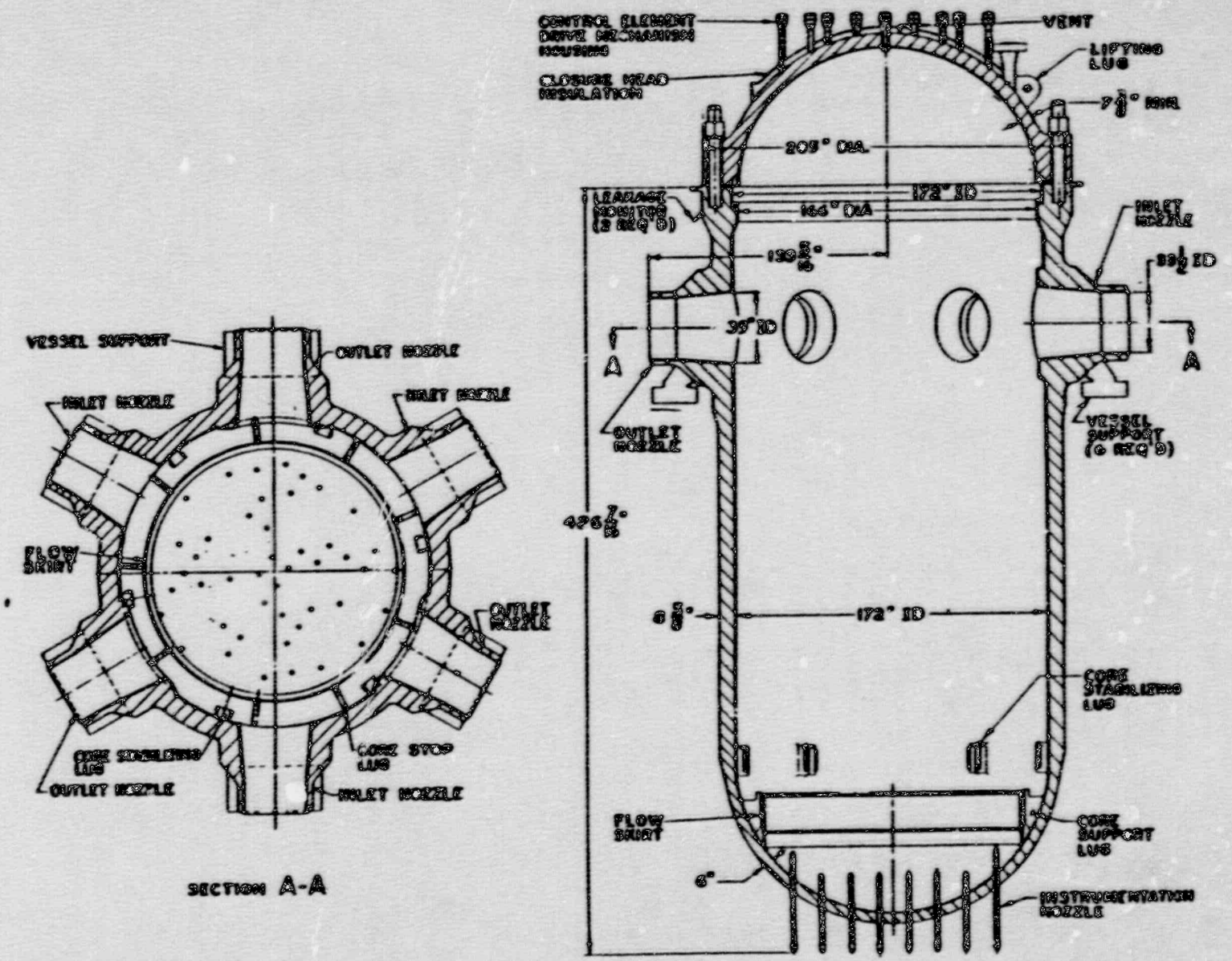


Figure 5.4-1. Maine Yankee Reactor Vessel

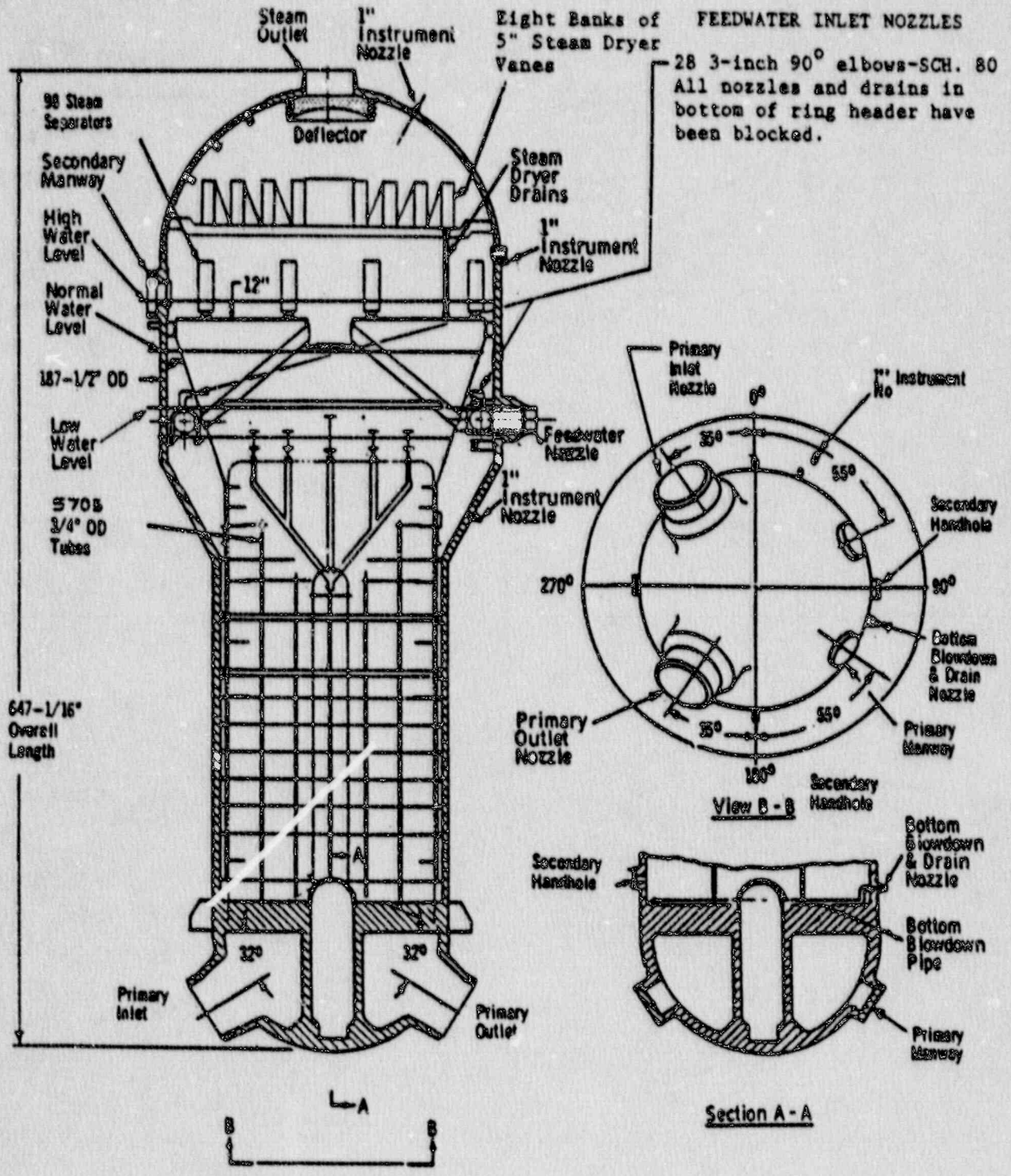


Figure 5.4-2. Maine Yankee U-Tube Steam Generator

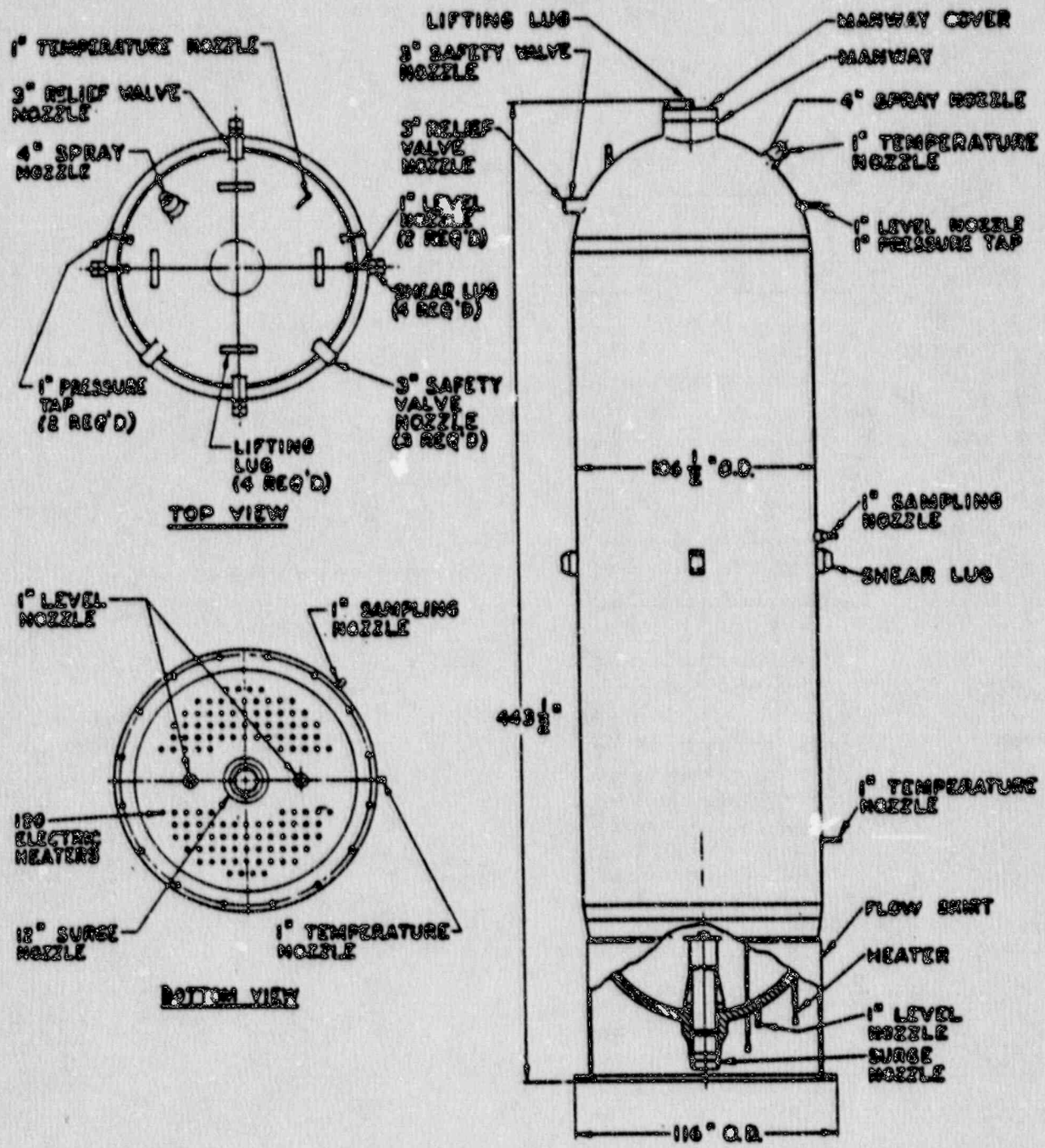


Figure 5.4-3. Maine Yankee Pressurizer

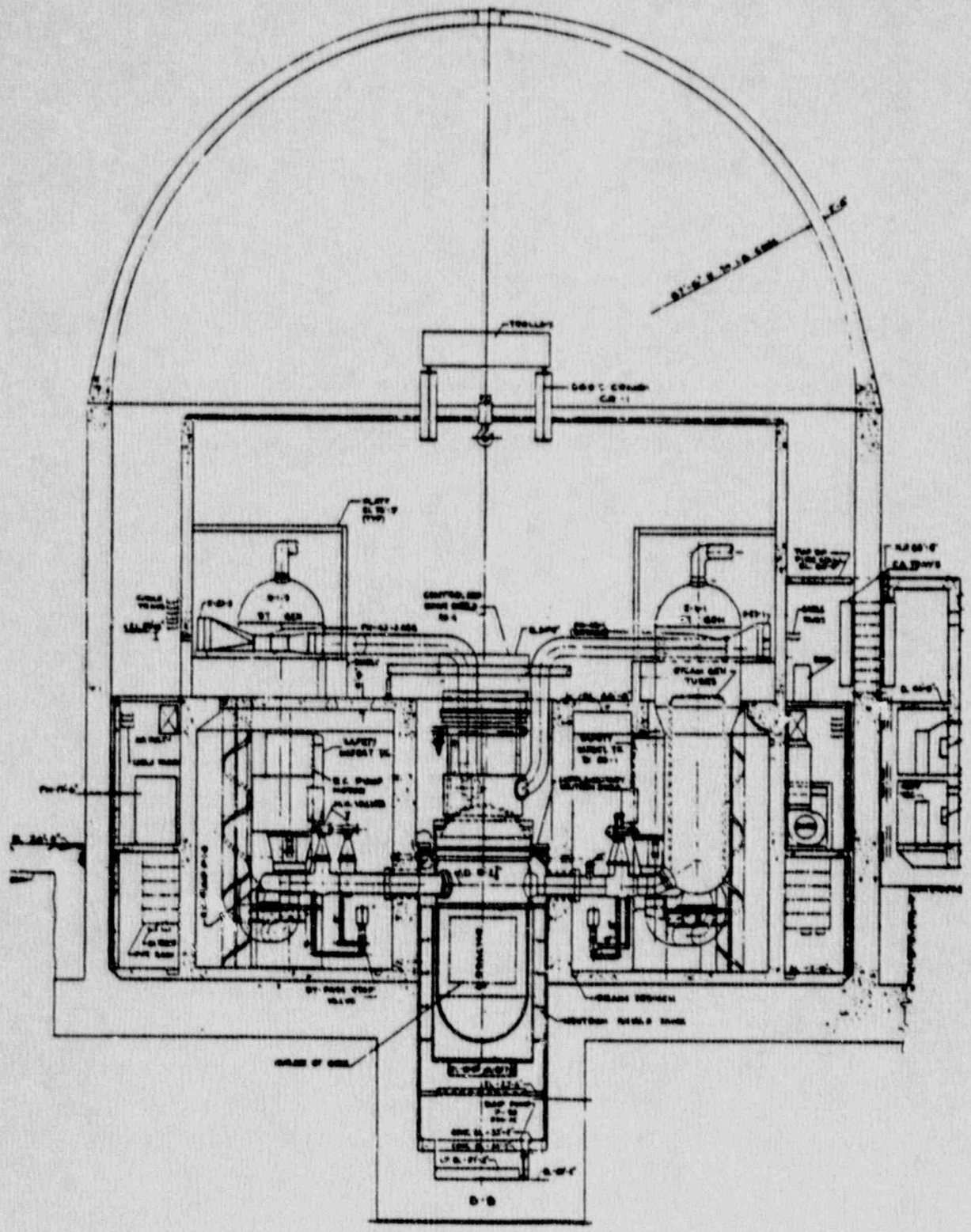


Figure 5.4-4. Section Views of the Maine Yankee Large, Dry Containment (Reinforced Concrete) (Sheet 1 of 2)

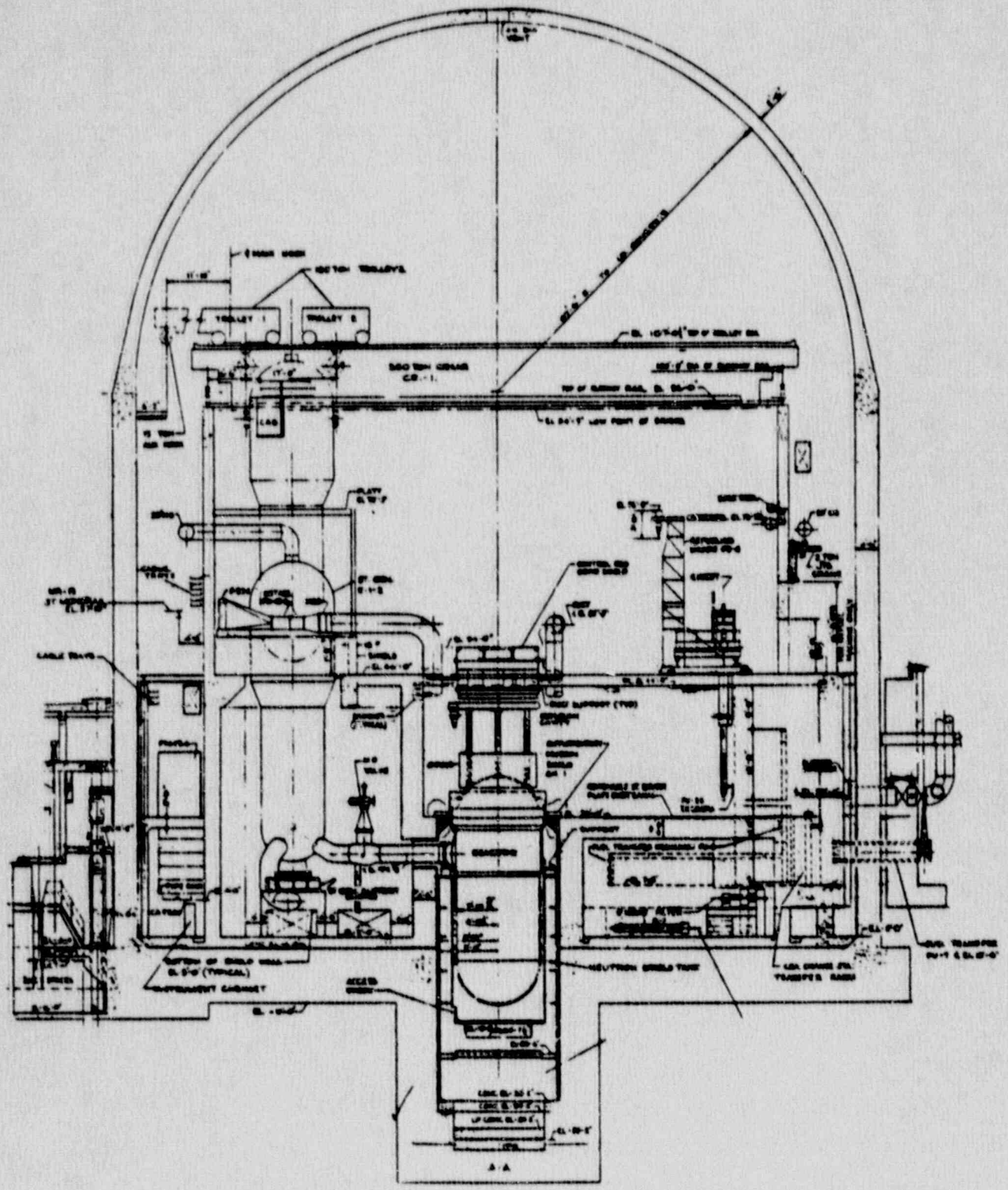


Figure 5.4-4. Section Views of the Maine Yankee Large, Dry Containment (Reinforced Concrete) (Sheet 2 of 2)

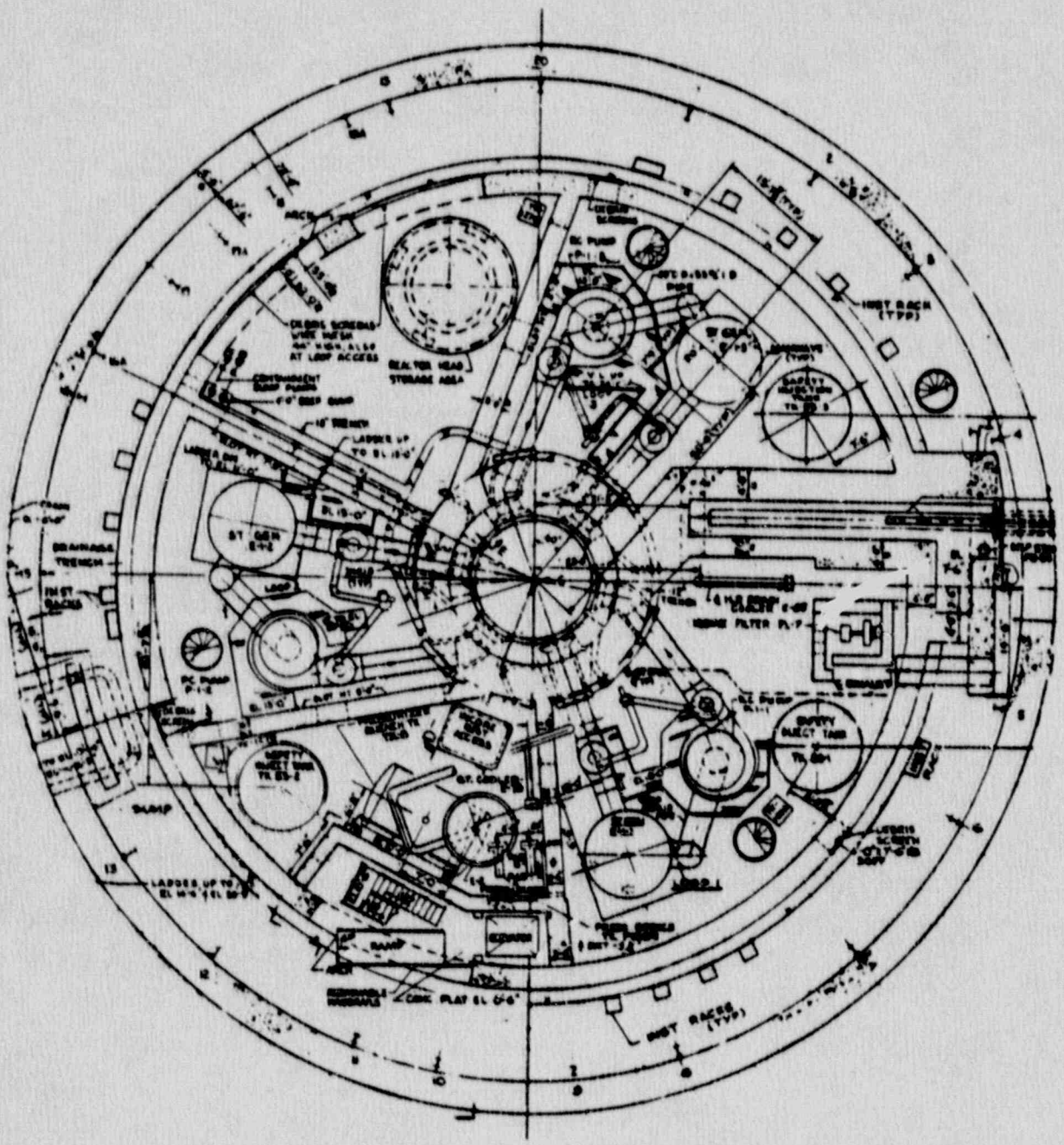


Figure 5.4-5. Plan View of the Maine Yankee Large, Dry Containment at the Level of the Primary Loops

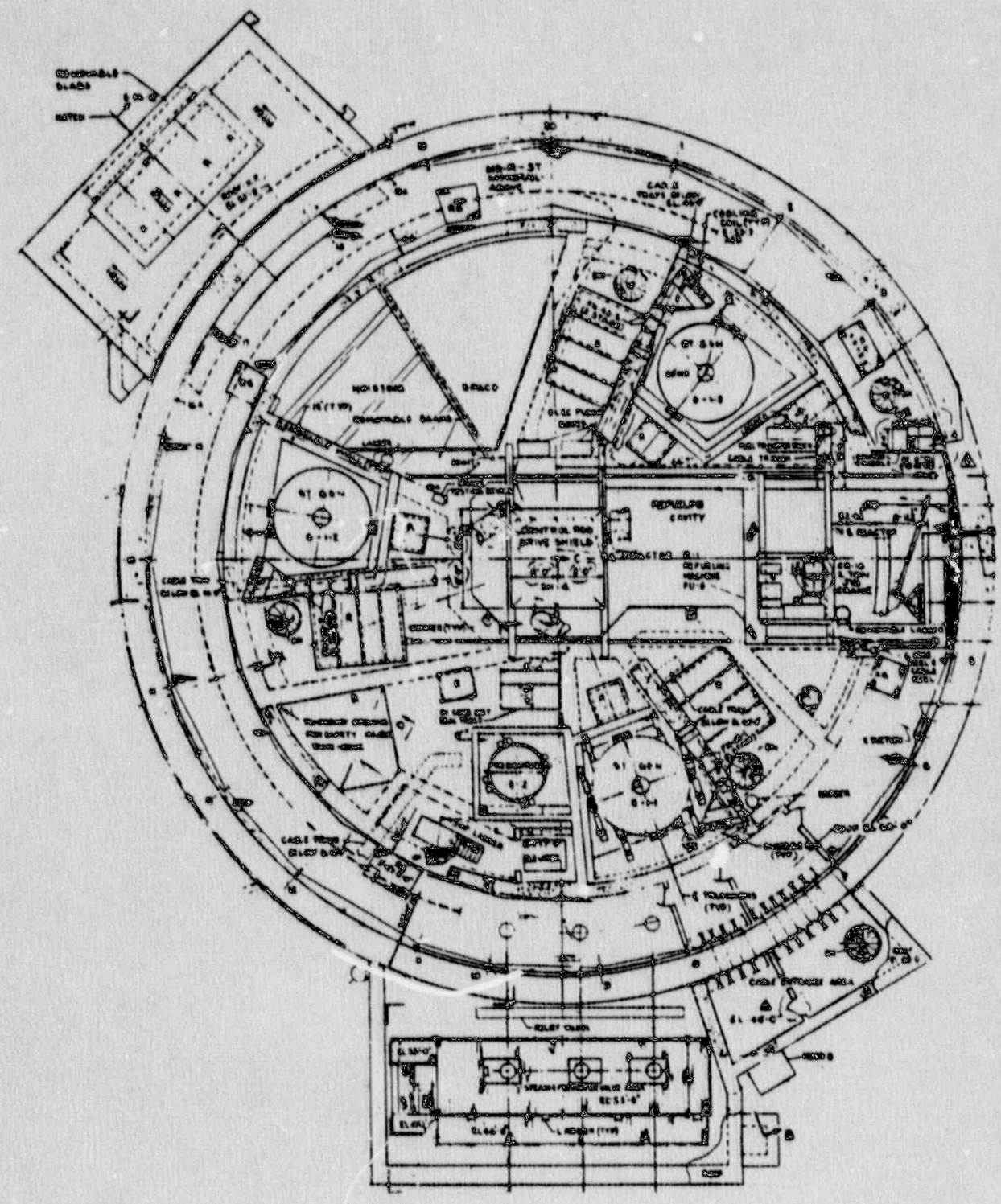


Figure 5.4-6. Plan View of the Maine Yankee Large, Dry Containment at the Operating Floor Level

5.5 Combustion Engineering PWR Comparative Data

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

- Table 5.5-1 Design Parameters for Representative Combustion Engineering PWRs
- Table 5.5-2 Comparison of Combustion Engineering PWR Vessel and Core Parameters
- Table 5.5-3 Combustion Engineering PWR System Comparison - RCS, AFW, Charging and HPSI
- Table 5.5-4 Comparison of Combustion Engineering PWR Pressurizer Relief Capacity
- Table 5.5-5 Comparison of Combustion Engineering PWR Containments
- Table 5.5-6 Comparison of Combustion engineering PWR Backup Electric Power Systems
- Table 5.5-7 Comparison of Combustion Engineering PWR Power Conversion Systems

Table 5.5-1. Design Parameters for Representative Combustion Engineering PWRs

REACTOR PLANT CHARACTERISTICS	C-E 2-LOOP PLANT SAN ONOFRE 2 AND 3	C-E 2-LOOP SYSTEM 80 PLANT PALO VERDE 1, 2, AND 3	C-E 3-LOOP PLANT MAINE YANKEE
Overall			
Number of loops	2 without isolation	2 without isolation	3 without isolation
Thermal capacity	3930 MWt	3800 MWt	2630 MWt
Electric capacity	1 x 1070 MWe / 1 x 1080 MWe	1221 MWe	810 MWe
Efficiency (net)	27.35%	32.13%	30.80%
Coolant pressure in primary circuit at exit from reactor	2233 psig	2235 psig	2235 psig
Coolant temperature at inlet	553°F	564.5°F	538.9°F
Coolant temperature at exit from reactor	611.2°F	621.2°F	592.8°F
Coolant flow rate through reactors (total)	148 E6 lb/hr	164 E6 lb/hr	122 E6 lb/hr
Core			
Height of active core	12.5 ft.	12.5 ft.	11.39 ft.
Equivalent diameter	11.33 ft.	11.97 ft.	11.33 ft.
Number of fuel assemblies	217	241	217
Number of control rods assemblies	83 / 8	76	85
Number of fuel elements in assembly	236	236	166
Diameter of fuel element	0.325 in.	0.382 in.	0.3795 in.
Area of heat transfer surface	62,000 sq.ft.	68,600 sq.ft.	47,000 sq.ft.
Mean specific heat flux	182,400 Btu/hr-sq.ft.	182,400 Btu/hr-sq.ft.	182,400 Btu/hr-sq.ft.
Number of fuel rods	49,580	58,876	36,352
Core loading	3 batch mixed central zone	3 batch mixed central zone	3 batch mixed central zone
Average burnup (first cycle)	12,731 MWd/MTu	Unk.	13,795 MWd/MTu
Fuel weight (as UO_2)	233,900 lbs.	257,100 lbs.	203,934 lbs.

Table 5.5-1. Design Parameters for Representative Combustion Engineering PWRs (Continued)

REACTOR PLANT CHARACTERISTICS	C-E 2-LOOP PLANT SAN ONOFRE 2 AND 3	C-E 2-LOOP SYSTEM 80 PLANT PALO VERDE 1, 2, AND 3	C-E 3-LOOP PLANT MAINE YANKEE
Reactor Vessel			
Vessel height	43.54 ft.	48 ft.	42.11 ft.
Inner diameter	14.33 ft.	15.19 ft.	14.33 ft.
Number of Openings for inlet and outlet nozzles	2 x 2	2 x 2	2 x 3
Steam Generator			
Number of units	2	2	3
Thermal power per unit	1965 MWt	1900 MWt	813.3 MWt
Shell side operating pressure (steam)	985 psig	1055 psig	885 psig
Tube side operating pressure	2235 psig	2235 psig	2235 psig
Tube side design flow	74 E6 lb/hr	82 E6 lb/hr	40.67 E6 lb/hr
Piping			
Hot leg inner dia.	42 in.	42 in.	33.5 in.
Cold leg inner dia.	30 in.	30 in.	33.5 in.
Between pump and steam generator	30 in.	30 in.	33.5 in.
Safety Valves			
First opening pressure	2485 psig	2485 psig	2485 psig
Capacity	Unk. each of 2	Unk. each of 4	228,100 lb/hr each of 3
			57 lb/hr/MWt each PORV of 2
Primary Coolant Pumps			
Number	4	4	3
Pump capacity	111,400 gpm	108,000 gpm	88,000 gpm
Coolant temperature	553 °F	564.5 °F	538.9 °F
Pressure rise	310 ft.head	363 ft.head	290 ft.head
Design pressure	2500 psia	2500 psia	2485 psig
Design temperature	650 °F	650 °F	650 °F
Motor rating (nameplate)	9,700 hp.	12,000 hp.	9,900 hp.

Table 5.5-2. Comparison of Combustion Engineering 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)
C-E	2-loop	Fort Calhoun 1	1420	140	106.45	128	80.40	133	14 x 14	45 F/4 P
C-E	2-loop	Palisades	2530	172	136.7	132	79.80	204	15 x 15	41 F/4 P
C-E	2-loop	Calvert Cliffs 1 & 2	2700	172	136	136.7	78.5 to 82.8	217	14 x 14	77 F/8 P
C-E	2-loop	Millstone 2	2560	172	136	136.7	78.50	217	14 x 14	73 F/8 P
C-E	2-loop	St. Lucie 1	2560	172	136	136.7	83.20	217	14 x 14	73 F/8 P
C-E	2-loop	St. Lucie 2	2560	172	136	150	83.20	217	16 x 16	73 F/8 P
C-E	2-loop	ANO-2	2815	171	123	150	96.60	177	16 x 16	73 F/8 P
C-E	2-loop	San Onofre 2 & 3	3390	172	136	150	94.90	217	16 x 16	83 F/8 P
C-E	2-loop	Waterford 3	3390	172	136	150	94.90	217	16 x 16	83 F/8 P
C-E	2-loop Sys 80	Palo Verde 1, 2 & 3	3800	182	143.6	150	95.60	241	16 x 16	76 F/13 P
C-E	3-loop	Maine Yankee	2630	174	136	136.7	74.60	217	14 x 14	77 F/8 P

Table 5.5-3. Combustion Engineering PWR System Comparison - RCS, AFW, Charging and HPSI

		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
ANO-2	C-E	2815	2	0/2	UT	1 1	M T	575 @ 1214 575 @ 1214	3	PD	44	44	3	Cent	300 @ 1214	0	
Calvert Cliffs 1 & 2	C-E	2700	2	2/2	UT	2	T	700 @ 1100	3	PD	44	44	3	Cent	345 @ 1084	0	
Fort Calhoun	C-E	1420	2	2/2	UT	1 1	M T	260 @ 1040 260 @ 1040	3	PD	40	40	3	Cent	150 @ 1214	0	
Maine Yankee	C-E	2630	3	2/3	UT	2 1	M T	500 @ 1100 500 @ 1100	3 1	Cent PD	150 @ 2650 30	150 30	Same as charging pumps				(f)
Millstone 2	C-E	2560	2	2/2	UT	2 1	M T	300 @ 1056 800 @ 1056	3	PD	44	44	3	Cent	315 @ 1084	0	
Palisades	C-E	2530	2	2/3	UT	1 1	M T	415 @ 1183 415 @ 1183	3	PD	40	40	3	Cent	300 @ 1084	0	(e)
Palo Verde 1, 2 & 3	C-E	3800	2	0/4	UT	1 1	M T	750 @ 1422 750 @ 1422	3	PD	44	N/A	2	Cent	814 @ 1233	0	
San Onofre 2 & 3	C-E	3390	2	0/2	UT	2 1	M T	860 @ 1232 860 @ 1232	3	PD	44	44	3	Cent	415 @ 1227	0	
St. Lucie 1 & 2	C-E	2560	2	2/3	UT	2 1	M T	250 @ 1200 500 @ 1200	3	PD	44	44	3	Cent	345 @ 1084	0	
Waterford 3	C-E	3390	2	0/2	UT	2 1	M T	350 @ 2673 700 @ 2671	3	PD	44 @ 2324	44	3	Cent	380 @ 1227	0	

Table 5.5-3. Combustion Engineering 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 OT = once-through

Notes:

(f) Maine Yankee has a variable capacity auxiliary charging pump (10 to 30 gpm) in addition to three main charging/HPI pumps.

(g) At Palisades, two charging pumps are fixed capacity and one is variable capacity. The turbine-driven pump at Palisades is not safety related.

Table 5.5-4. Comparison of Combustion Engineering 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)
ANO-2	C-E	1	None	- - -	- - -	2	280.6	2500
Calvert Cliffs 1 & 2	C-E	2	Dresser 315 33VX	56.7	2385	2	222.2	2485
Fort Calhoun	C-E	2	Dresser 315 33VS	69.7	2392	2	281.7	2485
Maine Yankee	C-E	2	Dresser 315 33VS-30	57	2385	3	228.1	2485
Millstone 2	C-E	2	Dresser 315 33VS-30	59.8	2380	2	231.3	2485
Palisades	C-E	2	Dresser 315 33VX	60.5	2385	3	272.7	2485
Palo Verde 1, 2 &3	C-E	0	None	- - -	- - -	4	Unk.	2485
San Onofre 2 & 3	C-E	0	None	- - -	- - -	2	Unk.	2485
St. Lucie 1 & 2	C-E	2	Dresser 315 33VS-30	59.8	2385	3	234.4	2485
Waterford 3	C-E	0	None	- - -	- - -	2	420	2485

* Klb/hr rating

Table 5.5-5. Comparison of Combustion Engineering 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?
ANO-2	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.78E+06	54	0.1	No
Calvert Cliffs 1 & 2	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.00E+06	50	0.33	No
Fort Calhoun 1	C-E	Gibbs & Hill	Dry	Concrete Cylinder w/ Steel Liner	2-D Diag. Prestressed	110	1.05E+06	60	0.2	No
Maine Yankee	C-E	Stone & Webster	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	135	1.86E+06	55	0.1	No
Millstone 2	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	1.92E+06	54	3	Yes
Palisades	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.60E+06	55	0.2	No
Palo Verde 1, 2 &3	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	146	2.60E+06	60	unk.	No
San Onofre 2 & 3	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	150	2.34E+06	60	0.3	No
St. Lucie 1 & 2	C-E	Ebasco	Dry	Steel Cylinder	- - -	140	2.50E+06	44	unk.	Yes
Waterford 3	C-E	Ebasco	Dry	Steel Cylinder	- - -	140	2.68E+06	44	0.5	Yes

Table 5.5-6. Comparison of Combustion Engineering 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
ANO-2	C-E	None	2	2850	Fairbanks-Morse	3	125	
Calvert Cliffs 1 & 2	C-E	3 *		2500	Fairbanks-Morse	4	125	*One diesel is swing between units
Maine Yankee	C-E	None	1 2	250 2500	unk. Gen. Motors	4	125	
Millstone 2	C-E	None	2	2750	Fairbanks-Morse	2	125	
Palisades	C-E	None	2	2500	Alco	2	125	
Palo Verde 1, 2 & 3	C-E	None	2	5500	unk.	4	125	
San Onofre 2 & 3	C-E	None	2	6000	DeLavale	4	125	
St. Lucie 1 & 2	C-E	None	2	3500	Gen. Motors	2	125	
Waterford 3	C-E	None	2	4400	unk.	3	125	

Table 5.5-7. Comparison of Combustion Engineering 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 CW Pumps	CW Pump Drive Type	Shutoff Head (psig)	Capacity (gpm)
ANO-2	C-E	Bechtel	Bechtel	858	32	Closed Loop	Nat. Cooling Tower	2	turbine	1,170	14,500
Calvert Cliffs 1 & 2	C-E	Bechtel	Bechtel	825	40	Once Through	Chesapeke Bay	2 (50%)	turbine	1,174	15,000
Fort Calhoun 1	C-E	Gibbs & Hill	Gibbs & Hill	478	unk.	Once Through	Missouri River	3	AC	1,027	8,000 @754 psig
Maine Yankee	C-E	Stone & Webster	Stone & Webster	810	50	Once Through	Montsweag Bay	2 (60%)	AC	1,543	14,000 @1031 psig
Millstone 2	C-E	Bechtel	Bechtel	857	43	Once Through	Niantic Bay	2 (55%)	turbine	910	15,000
Palisades	C-E	Bechtel	Bechtel	730	5	Closed Loop	Mech. Cooling Towers	2	turbine	1,144	13,500 @1144 psig
Palo Verde 1, 2 & 3	C-E	Bechtel	Bechtel	1333	55	Closed Loop	Mech. Cooling Towers	2 (65%)	turbine	unk.	unk.
San Onofre 2 & 3	C-E	Bechtel	Bechtel	1080	unk.	Once Through	Pacific Ocean	unk.	unk.	unk.	unk.
St. Lucie 1 & 2	C-E	Ebasco	Ebasco	839	5	Once Through	Atlantic Ocean	2	AC	unk.	unk.
Waterford 3	C-E	Ebasco	Ebasco	1075	63	Once Through	Mississippi River	2 (63%)	turbine	1,183	17,940 @ 932 psig

6. BABCOCK & WILCOX (B&W) PRESSURIZED WATER REACTORS (PWRs)

In the U.S., Babcock & Wilcox (B&W) has produced 10 PWRs, all with two primary loops and once-through steam generators (OTSG). There are three basic B&W PWR models that can be distinguished by the relative positioning of the reactor vessel and the OTSG, and by the number of fuel elements in the core. The terms "lowered loop" and "raised loop" define the relative positioning of the reactor vessel and the OTSG, and the reactor cores are comprised of either 177 or 205 fuel assemblies. As of March 1990, the numbers of units of each type are as follows:

- 177 "Lowered-loop" 11 units
- 177 "Raised-loop" 1 unit
- 205 "Raised Loop" 2 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 Babcock & Wilcox PWRs is presented in Section 6.1. The Babcock & Wilcox plants are described in Sections 6.2 to 6.4, and numerous detailed comparative tables are presented in Section 6.5. In Section 7, the comparative tables for the Babcock & Wilcox PWRs are compiled with similar tables for Westinghouse and Combustion Engineering PWRs.

6.1 Babcock & Wilcox PWR Overview

6.1.1 Primary System

In the B&W 2-loop RCS, each primary loop is comprised of one hot leg that enters the top of a once-through steam generator (OTSG), and two cold legs that exit the bottom of the OTSG and return flow to the reactor vessel. One reactor coolant pump is in each cold leg, for a total of four reactor coolant pumps. The pressurizer is connected to one of the hot legs.

The B&W RCS is designed to operate with a nearly constant average loop coolant temperature (T_{ave}). Under this control scheme shown in Figure 6.1-1, hot leg temperature (T_{hot}) increases and cold leg temperature (T_{cold}) decreases as reactor power increases. This is different than the constant T_{cold} control scheme used in Westinghouse and C-E PWRs.

The once-through steam generators produce superheated steam and do not require moisture separators or steam dryers. An OTSG has a relatively small secondary coolant inventory in comparison to the U-tube steam generators used in Westinghouse and Combustion Engineering PWRs. This results in a more rapid "dry-out" following loss of all feedwater to the once-through steam generators. Basic design characteristics of Babcock & Wilcox once-through steam generators are listed in Table 6.1-1.

6.1.2 Reactor Core and Fuel Assemblies

All B&W commercial PWRs are designed to operate with rod-type slightly-enriched fuel in a core comprised of 177 or 205 fuel assemblies. The general arrangement of a 177 fuel assembly core is shown in Figure 6.1-2 and the 205 fuel assembly core is shown in Figure 6.1-3. A comparison of basic B & W core parameters is presented in Section 6.5.

The fuel assemblies have square cross-sections in a 15 x 15 fuel rod array or a 17 x 17 array, with multi-finger control rods that insert into channels in the fuel assemblies. A brief description of the 15 x 15 and 17 x 17 fuel assembly design is provided below.

Table 6.1-1. General Characteristics of Babcock and Wilcox Steam Generators

Design Parameters	Oconee 1, 2 & 3	Davis Bessie	Bellefonte 1 & 2
Tube Side (primary)			
Design Pressure (psig)	2500	2500	2500
Norm. Operating Press. (psig)	unk.	unk.	2165
Design Temp. (°F)	650	650	670
Design Flow Rate (lb/hr)	6.57E+07	6.90E+07	unk.
Total Primary Side Vol. (ft3)	2030	2030	unk.
Shell Side (secondary)			
Design Pressure (psig)	1050	1050	1235
Full Power Pressure (psig)	910	910	1045
Design Temp. (°F)	600	600	630
Full Power Temp. (°F)	570	570	602
Steam Flow Rate (lb/hr) @ Full Power	5.60E+06	6.12E+06	6.05E+06
Total Secondary Side Vol. (ft3)	3412	3412	unk.

A. 15 x 15 Mark B Fuel Assemblies

The geometry of the 15 x 15 fuel assembly is shown in Figure 6.1-4. Of the 225 lattice positions in the 15 x 15 array of a Mark B fuel assembly, 208 positions are for fuel rods, 16 are guide tubes for the multi-finger control rod assemblies (CRAs), and axial power shaping rod assemblies (APSRAs), and one is for in-core instrumentation. Some Mark B fuel assemblies may have burnable poison rod assemblies (BPRAs) inserted in the 16 positions normally assigned for the control rods.

B. 17 x 17 Mark C Fuel Assembly

The geometry of the 17 x 17 fuel assembly is shown in Figure 6.1-5. Of the 289 lattice positions in the 17 x 17 array of a Mark C fuel assembly, 264 positions are for fuel rods, 24 are guide tubes for the multi-finger control rod assemblies (CRAs) and axial power shaping rod assemblies (APSRAs), and one is for in-core instrumentation. Some Mark C fuel assemblies may have burnable poison rod assemblies (BPRAs) inserted in the 24 positions normally assigned for the control rods. The principal advantages of the Mark C fuel assemblies over the Mark B are lower linear heat rates and lower fuel and peak clad temperatures.

6.1.3 Reactivity Control System

Core reactivity is controlled by full-length control rod assemblies (CRAs), axial power shaping rod assemblies (APSRAs), burnable poison rods (BPRs), and soluble boron in the coolant. Plants using the 15 x 15 fuel assemblies have 16 finger CRAs as shown in Figure 6.1-6 while the 17 x 17 fuel assemblies are designed for the 24 finger CRAs shown in Figure 6.1-7.

All B&W PWRs use roller-nut control rod drive mechanisms (CRDMs) to control the positioning of the CRAs and APSRAs. The B&W CRDM shown in Figure 6.1-8 consists of a motor tube, which houses a lead screw and its rotor assembly, and a snubber assembly. The motor tube is a closed pressure housing. An external motor stator surrounds the motor tube and a position indication assembly is arranged outside the upper extension of the motor tube. The extension shaft on each CRA and APSRA is a threaded leadscrew that extends into the CRDM and is engaged by rotating roller-nuts in the motor tube when the CRDM is energized. When the CRDM is de-energized, mechanical springs separate the roller-nut halves, disengaging the leadscrew and allowing the CRAs to fall into the core. A hydraulic snubber assembly within the upper housing decelerates the CRAs a short distance above the full-in position. The drives for the APSRAs do not disengage on loss of power. The CRDMs stators are water-cooled.

6.1.4 Emergency Core Cooling System

All B&W plants have high-pressure safety injection pumps that are capable of providing makeup to the RCS at the PORV setpoint pressure. The B&W ECCS provides injection paths into the RCS cold legs for the high-head injection pumps. Separate injection paths directly into the reactor vessel are provided for two core flood tanks (accumulators) and the two-train low-head injection pumps (see Figure 6.1-9).

6.1.3 Containment

All B&W PWRs have large, dry containments that have varied construction styles as shown in Figure 6.1-10. Descriptions of containments for B&W PWRs are included in the following sections.

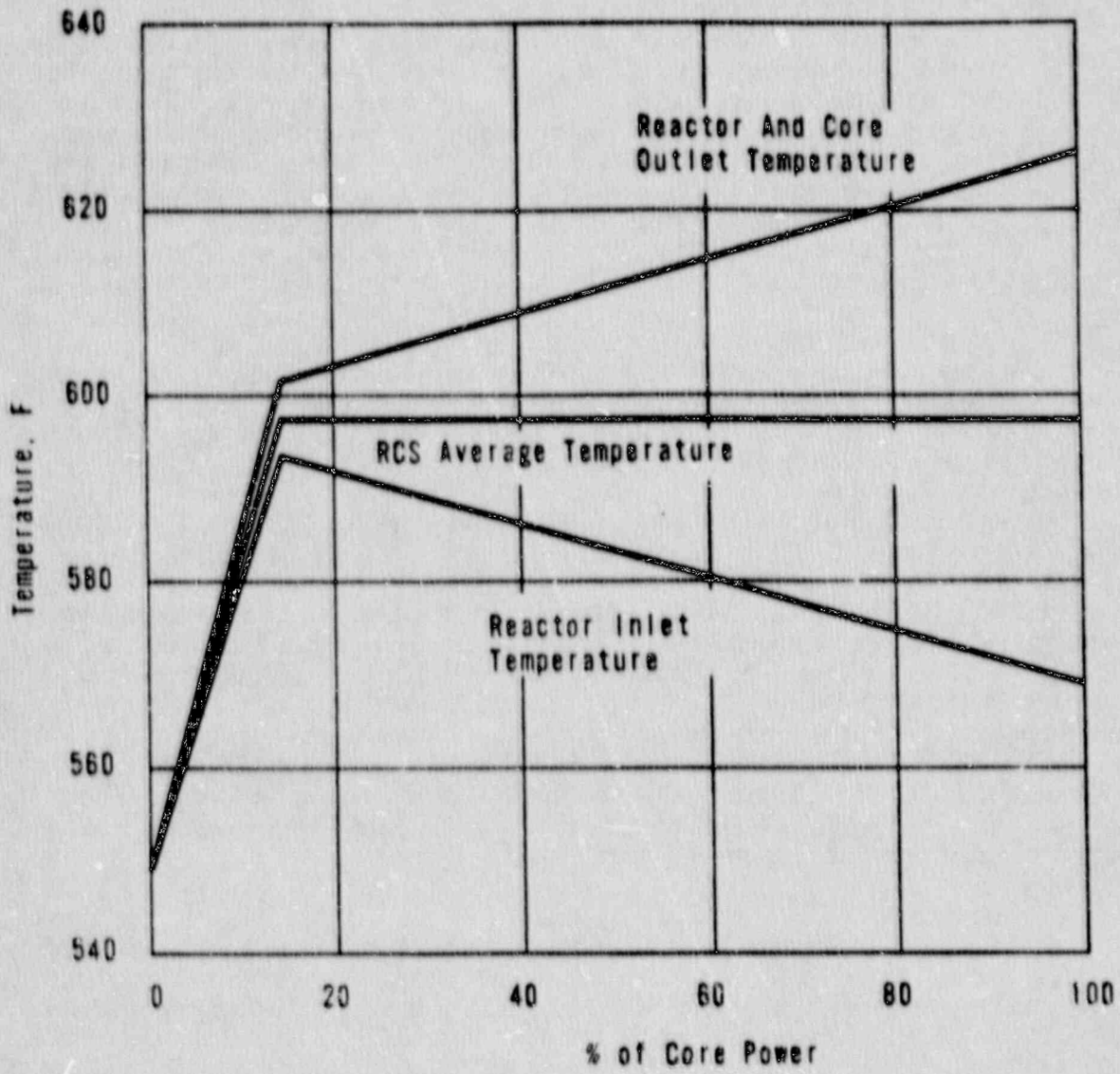


Figure 6.1-1. Typical Babcock & Wilcox RCS Temperature ontrol Program

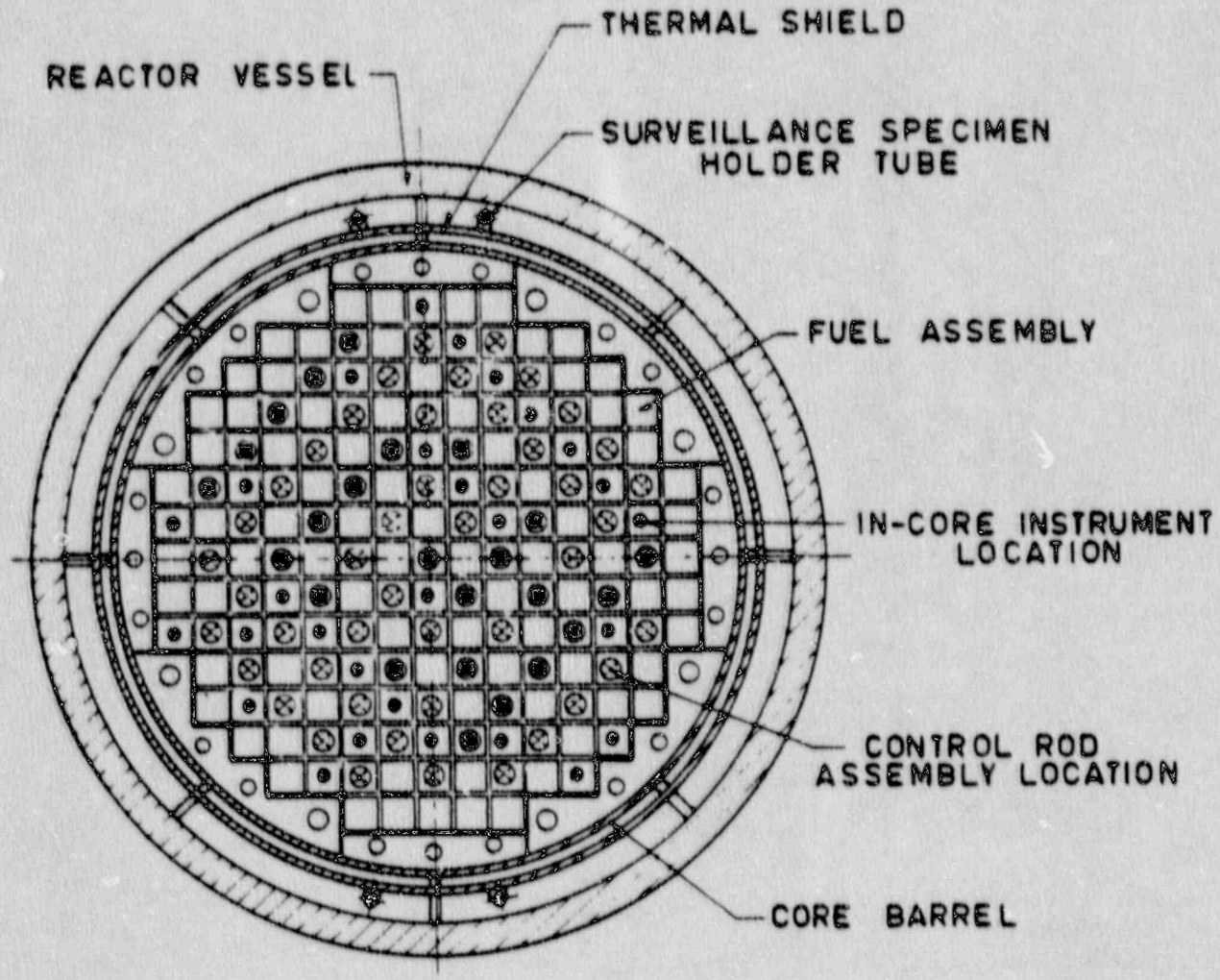


Figure 6.1-2. General Arrangement of a B & W 177 Fuel Assembly Core

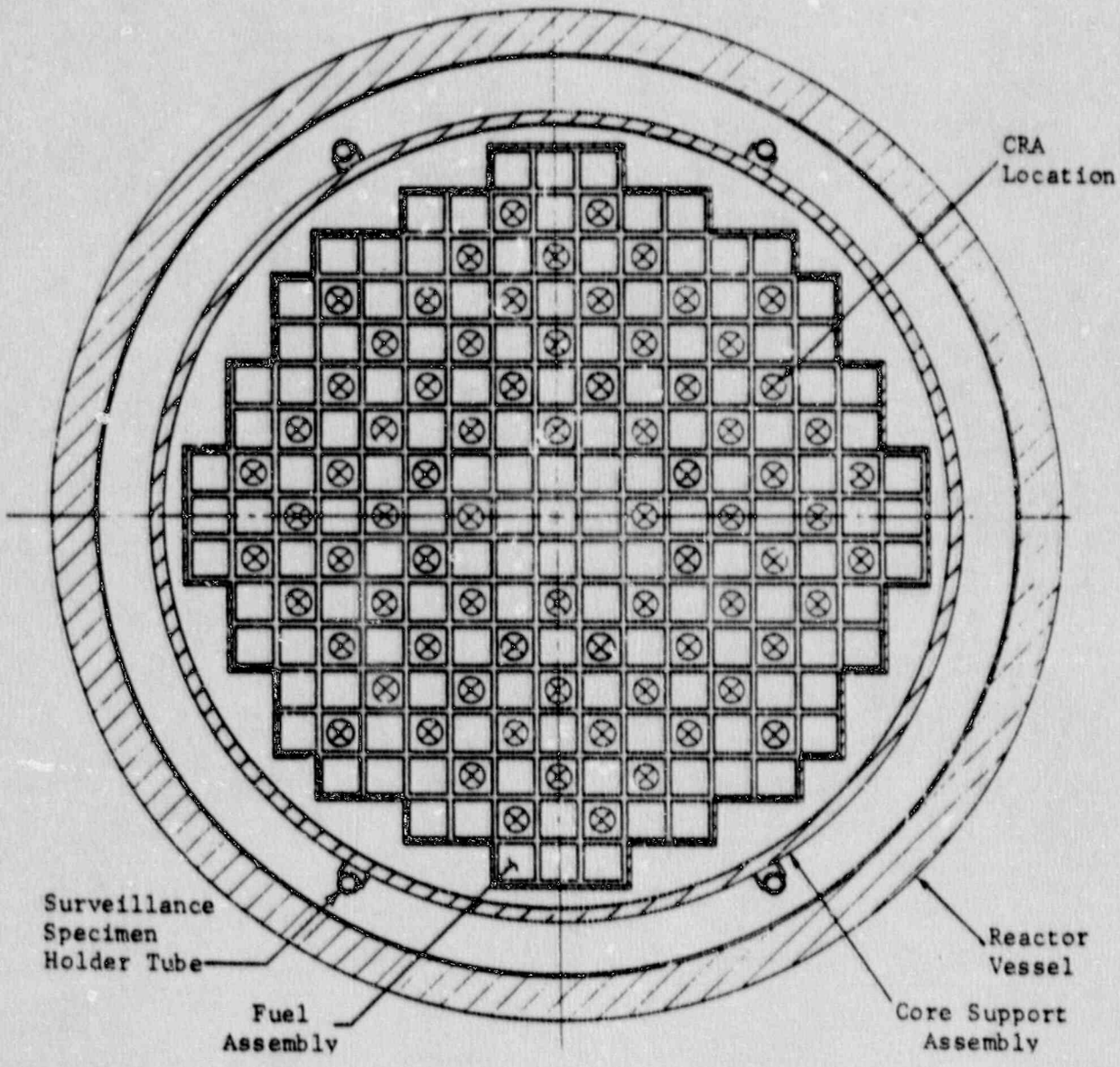


Figure 6.1-3. General Arrangement of a B & W 205 Fuel Assembly Core

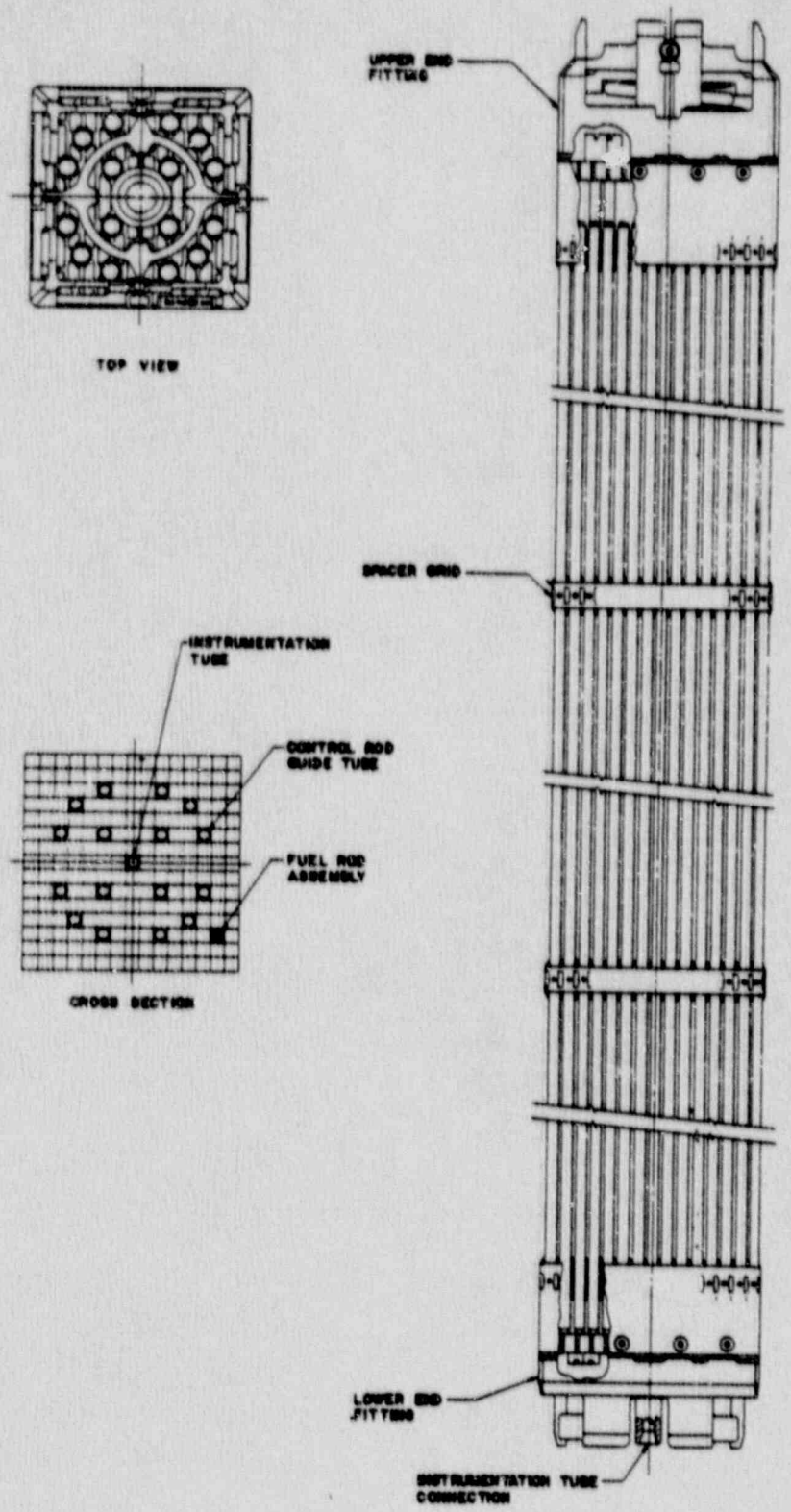


Figure 6.1-4. B & W 15 X 15 (Mark B) Fuel Assembly

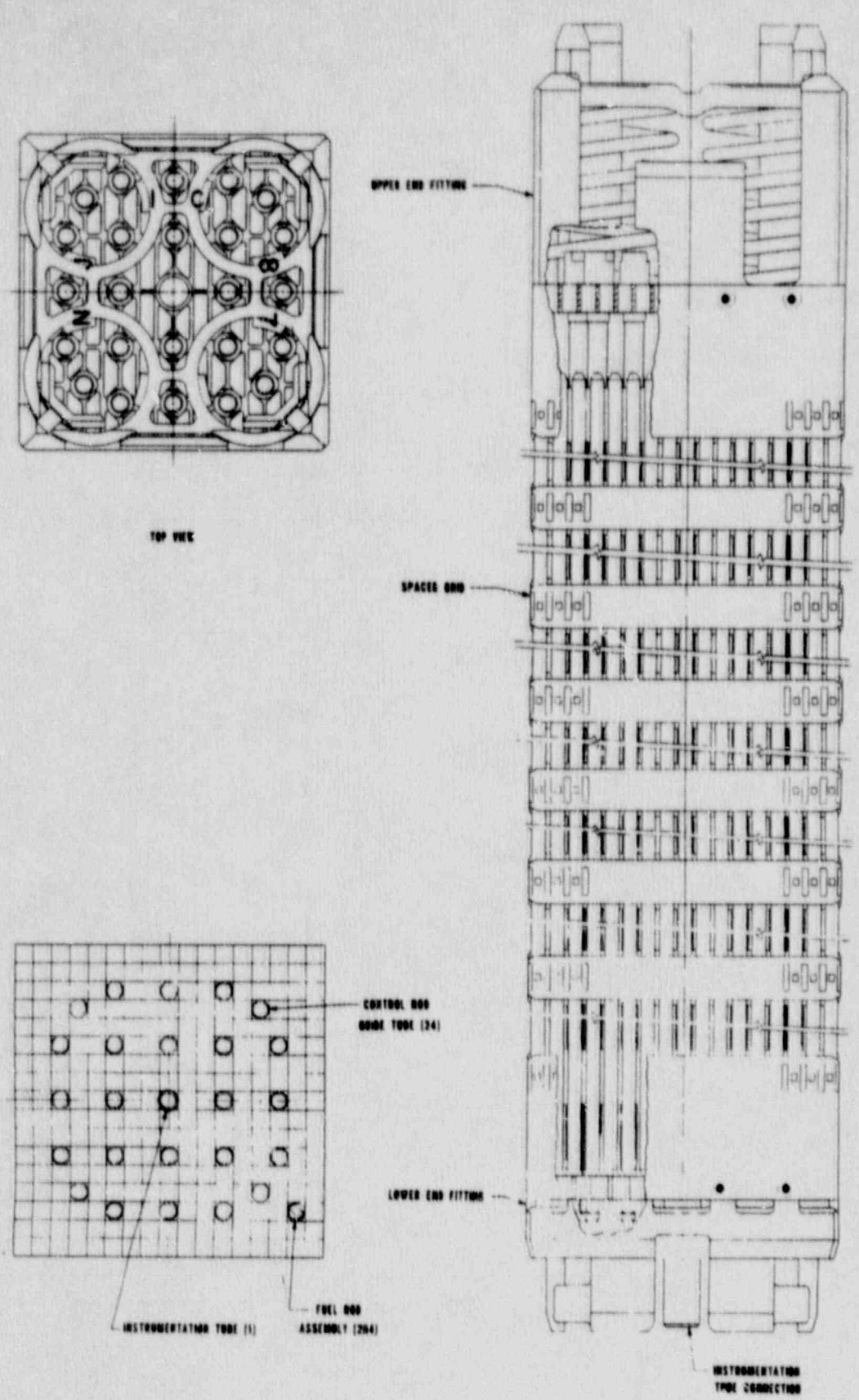


Figure 6.1-5. B & W 17 X 17 (Mark C) Fuel Assembly

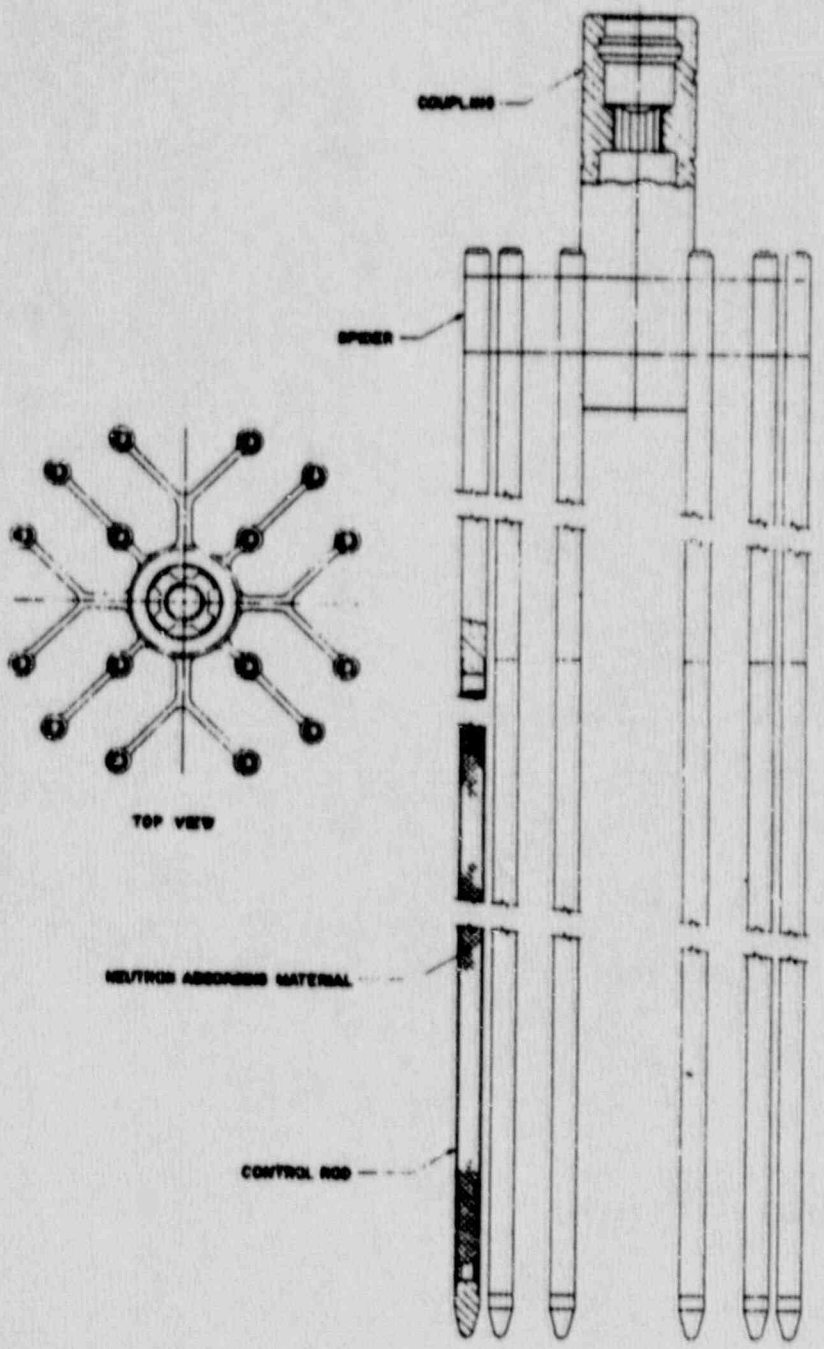


Figure 6.1-6. B & W 16 Element CRA

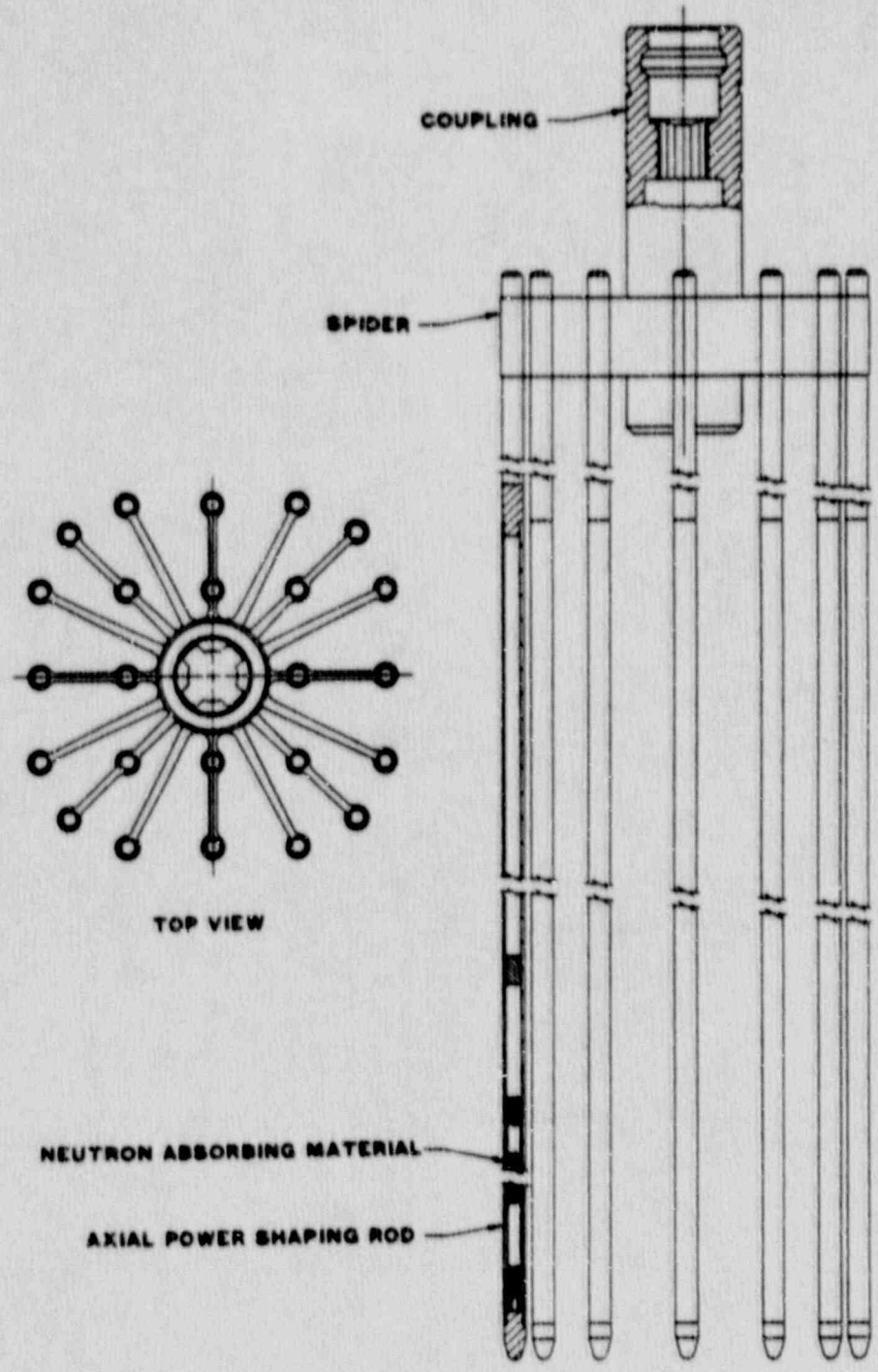


Figure 6.1-7. B & W 24 Element CRA

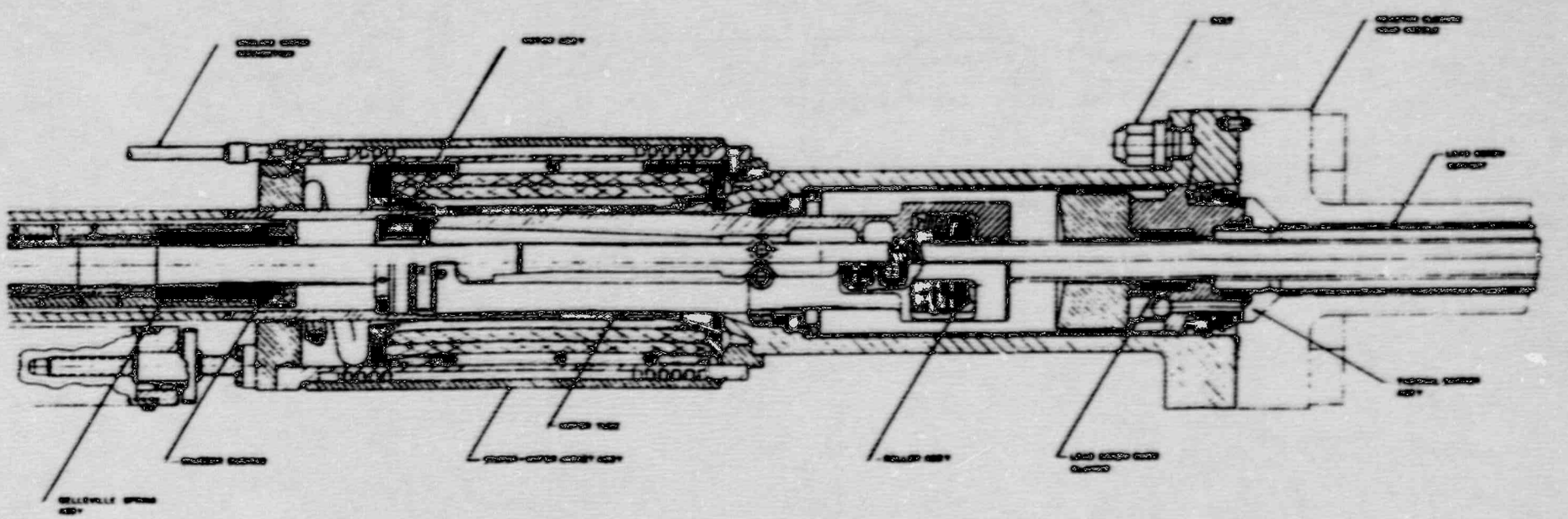


Figure 6.1-8. B & W Roller-Nut Control Rod Drive Mechanism

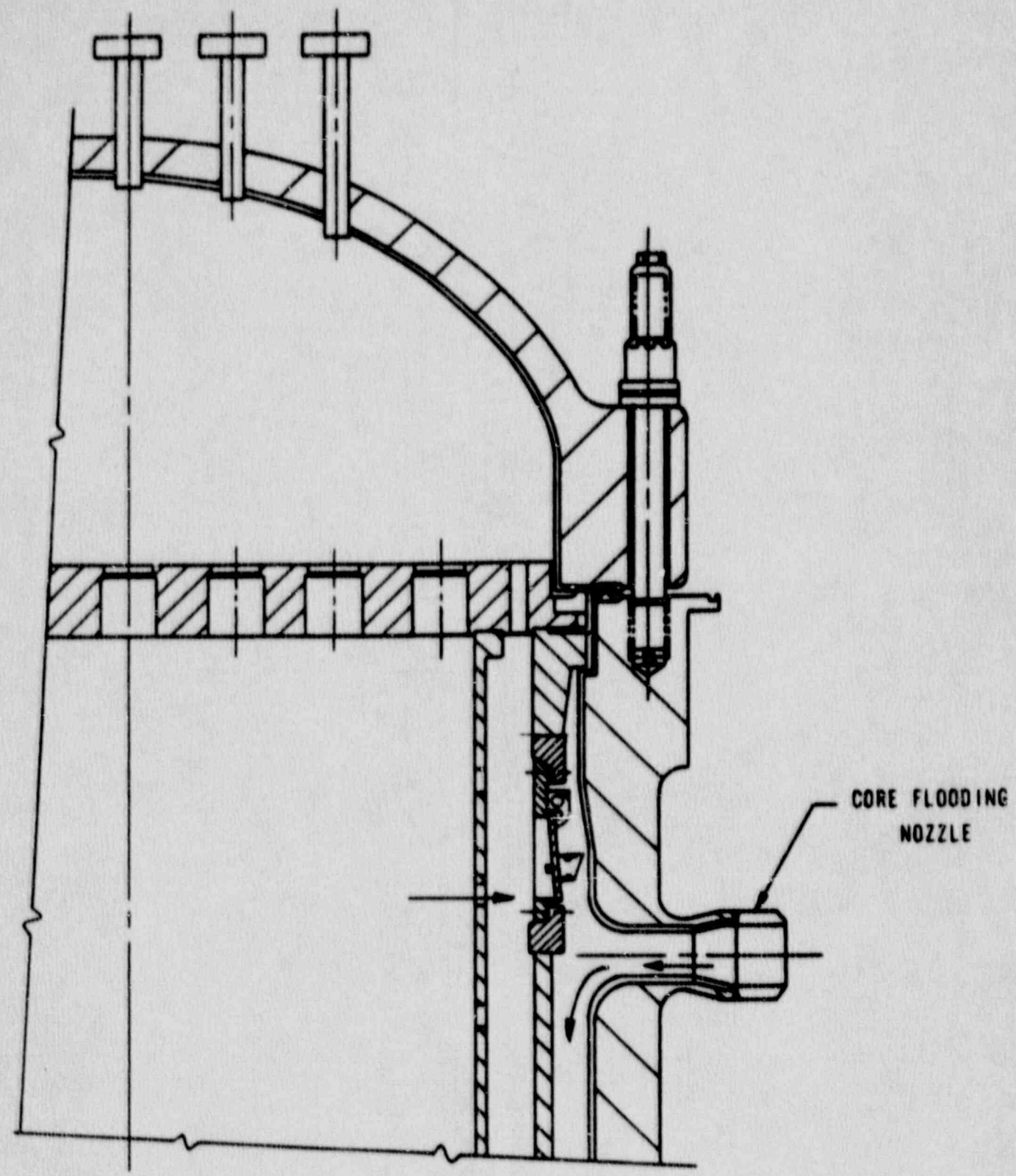


Figure 6.1-9. B & W Core Flooding Injection Path for Core Flood Tanks and Low-Pressure Injection System

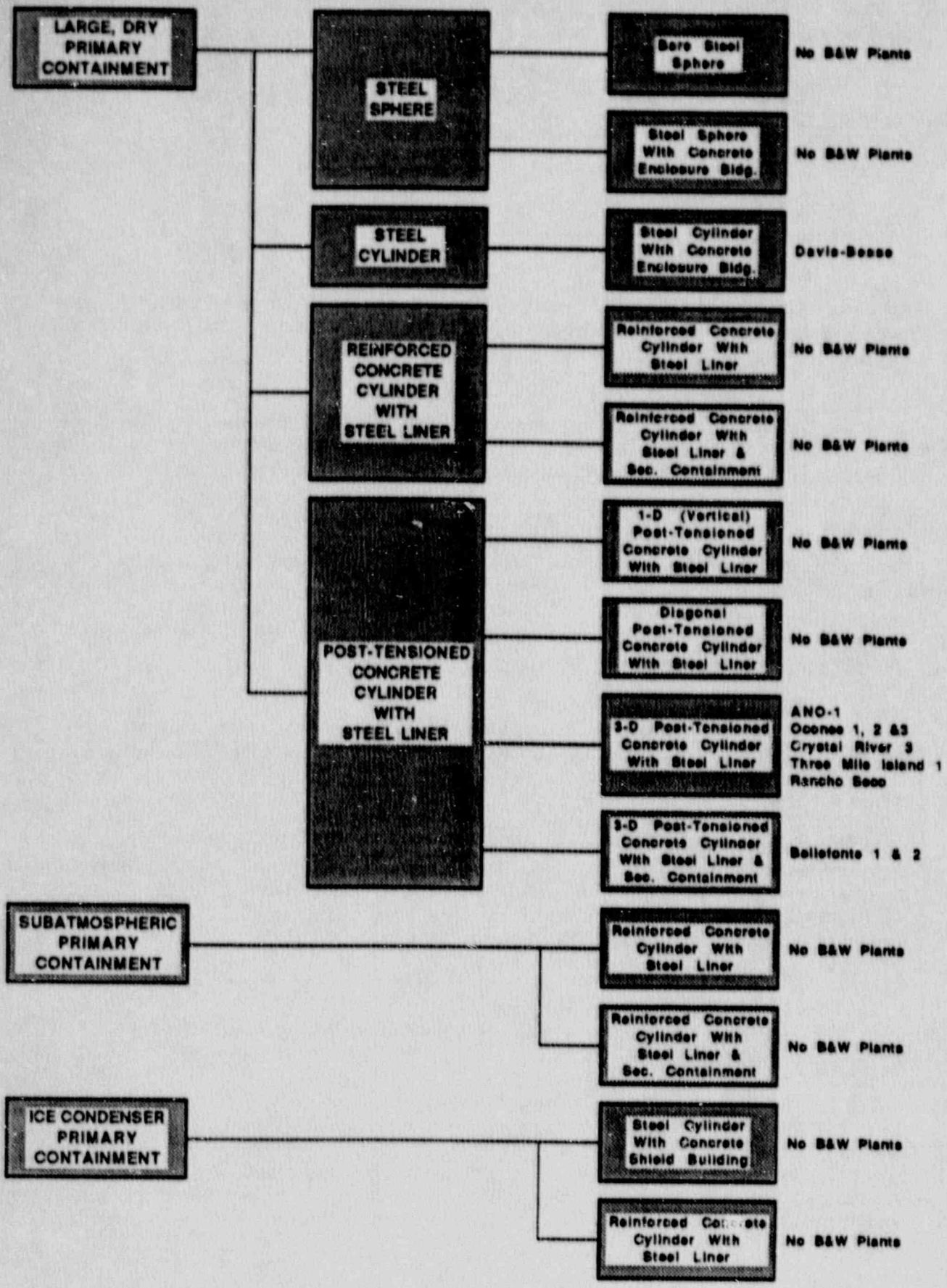


Figure 6.1-10. Distribution of Containment Types for Babcock & Wilcox Reactors

6.2 177 Lowered-Loop Babcock & Wilcox PWRs

Babcock & Wilcox lowered-loop PWR plants in the United States are the following:

- Arkansas Nuclear One-1
- Crystal River 3
- Oconee 1, 2, and 3
- Rancho Seco
- Three Mile Island 1

All of these plants had full power operating licenses as of 2/89. Three Mile Island 2 was another Babcock & Wilcox lowered-loop PWR. The general arrangement of the B & W 177 lowered-loop NSSS is shown in Figures 6.2-1 and 6.2-2.

6.2.1 Reactor Core and Fuel Assemblies

All lowered-loop plants have a 177 fuel assembly core that operates at an average power density in the range from 76 to 90 kW/liter and has a thermal power output of 2452 to 2772 MWt. This core is comprised of 15 x 15 Mark B fuel assemblies.

6.2.2 Reactivity Control System

Core reactivity is controlled by 61 full-length control rod assemblies (CRAs), 8 axial power shaping rod assemblies (APSRAs), burnable poison rods (BPRs), and soluble boron in the coolant. The neutron absorber in the CRAs and the APSRAs is Ag-In-Cd (silver-indium-cadmium), while the BPRs use B_4C (boron carbide). Each CRA and APSRA consists of 16 control rods that are joined at the top by a "spider" assembly that is attached to the CRDM extension shaft.

6.2.3 Reactor Coolant System

In the B&W 2-loop RCS, one RCS hot leg supplies each steam generator and two cold legs in each loop return flow to the reactor. A single reactor coolant pump is in each cold leg. The general arrangement of the reactor vessel and internal structures is shown in Figure 6.2-3. Reactor vessel inside diameter for these plants is 171 inches.

The B&W NSSS incorporates a once-through steam generator (OTSG) with an integral superheater. This system supplies dry steam with approximately 35°F superheat, therefore, no moisture separators are required. Reactor coolant flows downward through the steam generator. As seen in Figure 6.2-1, this results in a relatively long "candy cane" RCS hot leg. Flow paths inside the OTSG are shown in Figure 6.2-4. Note that the feedwater inlets are slightly above the midplane of the steam generator.

The pressurizer is shown in Figure 6.2-5.

6.2.4 Shutdown Cooling Systems

Shutdown cooling is performed by a multi-mode residual heat removal system that also performs the low-pressure safety injection (LPSI) function as part of the ECCS.

6.2.5 Emergency Core Cooling Systems

All lowered-loop plants have three high-pressure safety injection pumps that are capable of providing makeup to the RCS at the PORV setpoint pressure. The HPSI pumps inject into the four RCS cold legs. The LPSI pumps and the core flood tanks (safety injection accumulators) inject directly into the reactor vessel via core flood nozzles as described in Section 6.1.

6.2.6 Containment

All lowered-loop plants have three-dimension post-tensioned concrete containments with a steel liner. The general arrangement of the Three Mile Island 1 containment is shown in Figure 6.2-6 to 6.2-8.

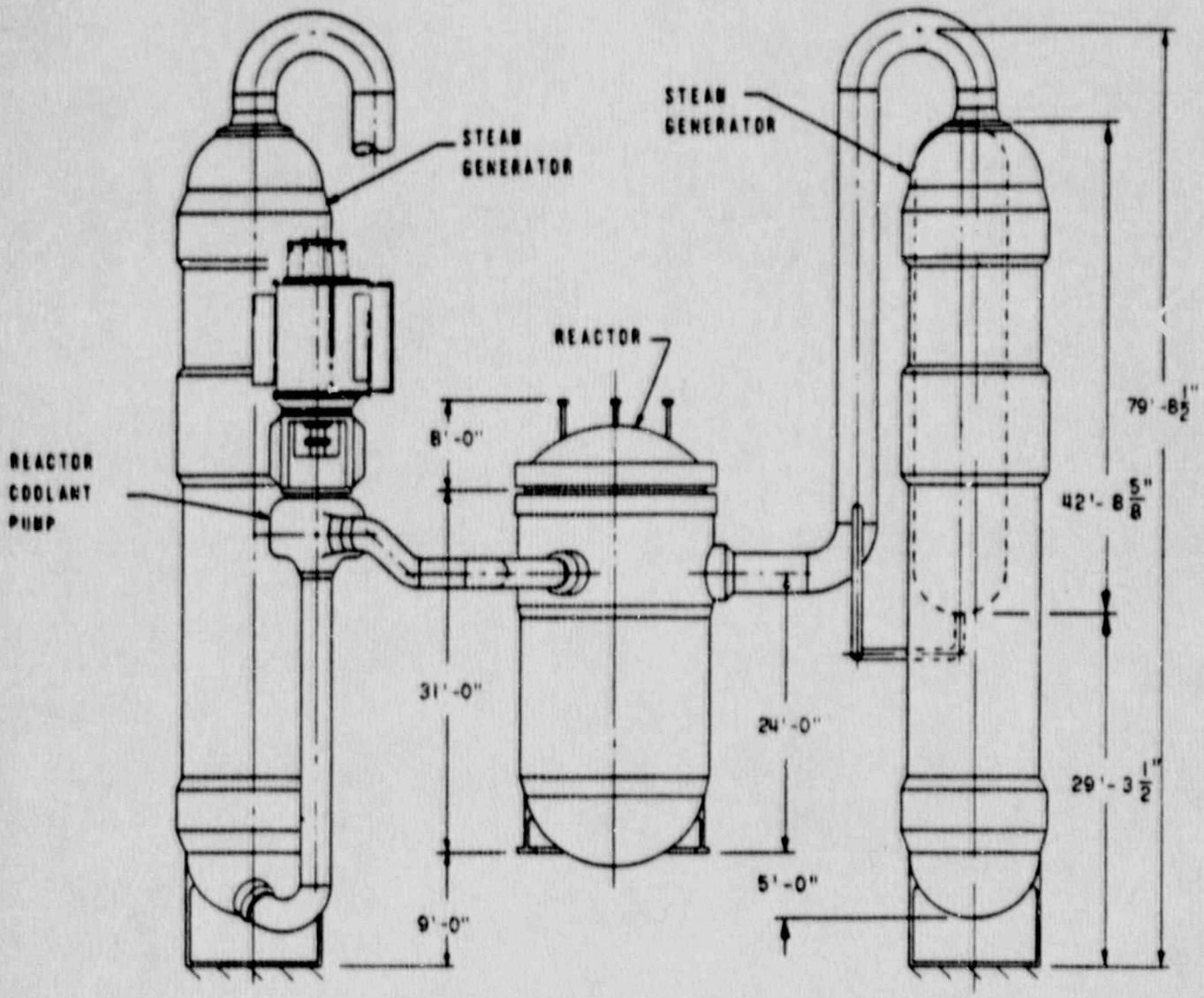


Figure 6.2-1. B & W 177 Lowered-Loop NSSS Side View

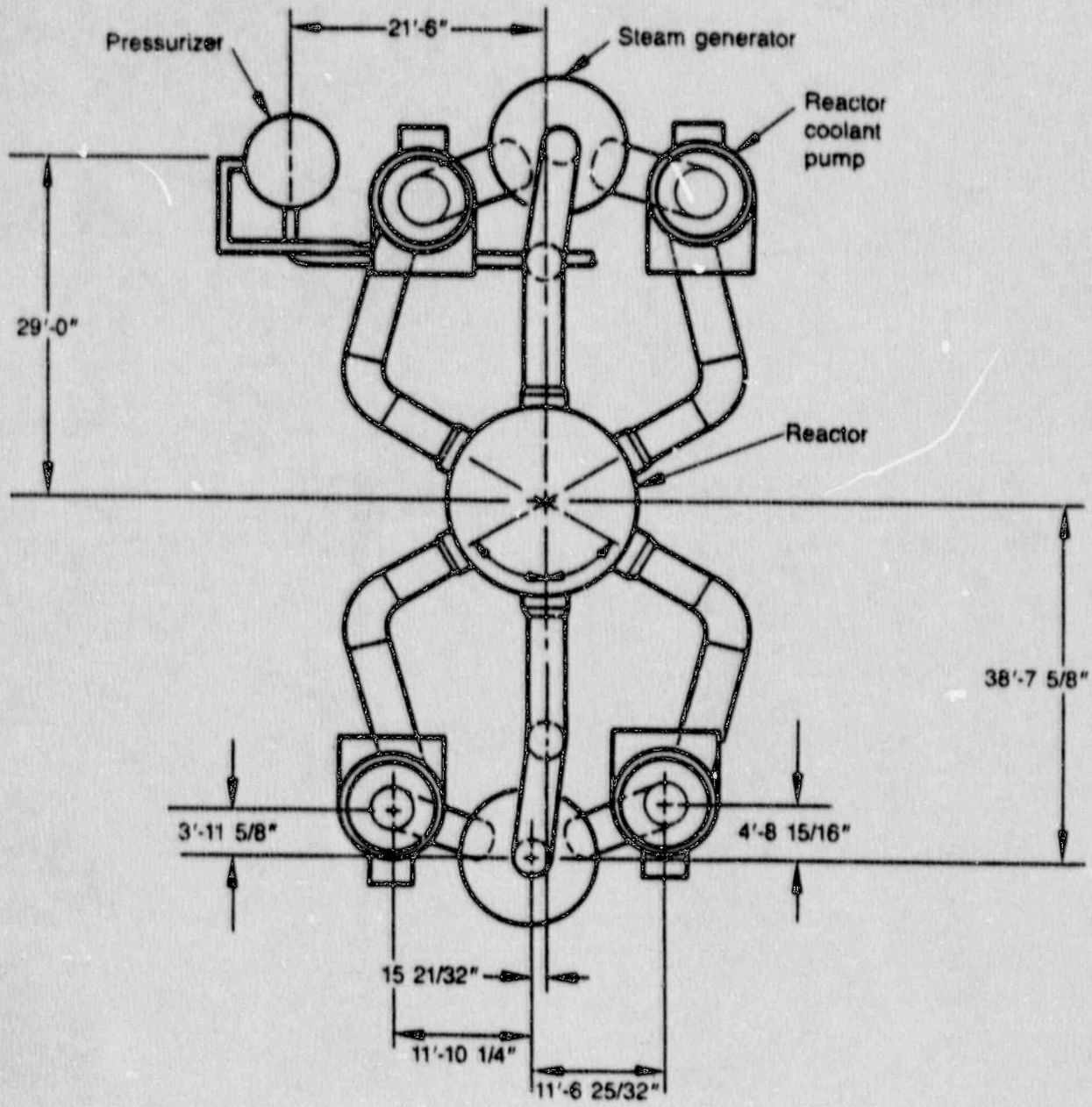


Figure 6.2-2. B & W 177 Lowered-Loop NSSS Plan View

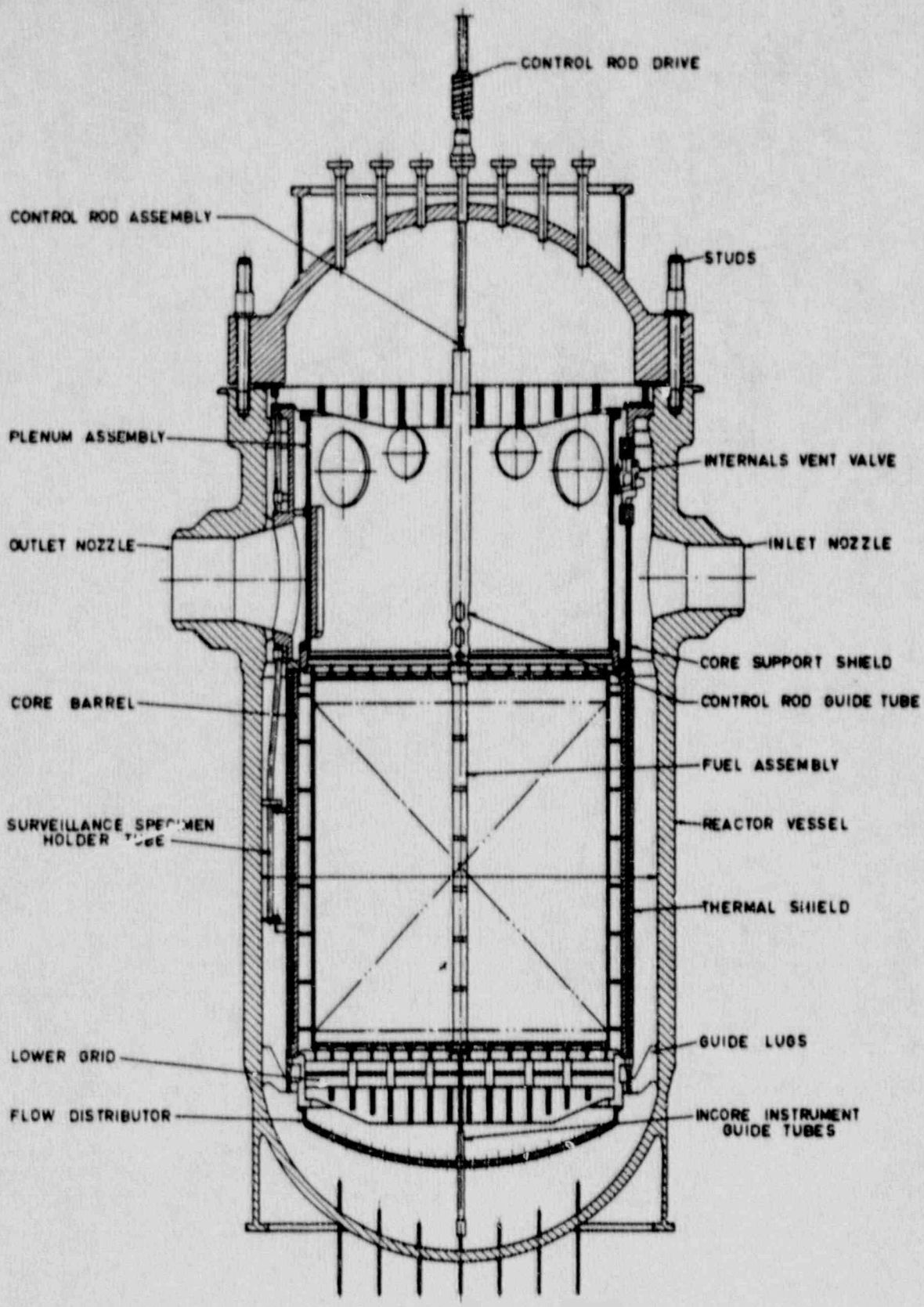


Figure 6.2-3. General Arrangement of the B & W 177 Plant Reactor Vessel and Internals

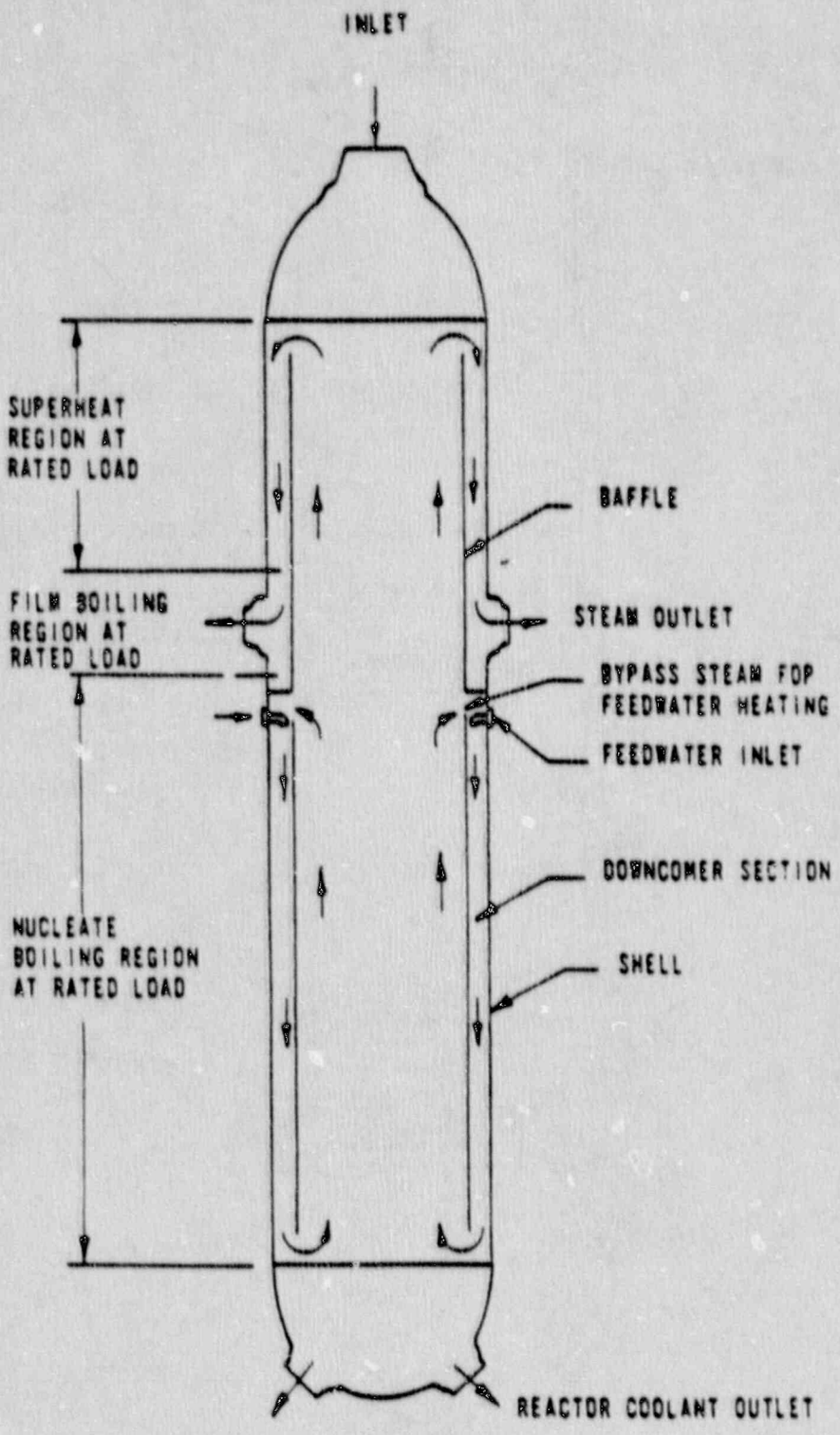


Figure 6.2-4. Flow Paths in the B & W 177 Plant Once-Through Steam Generator

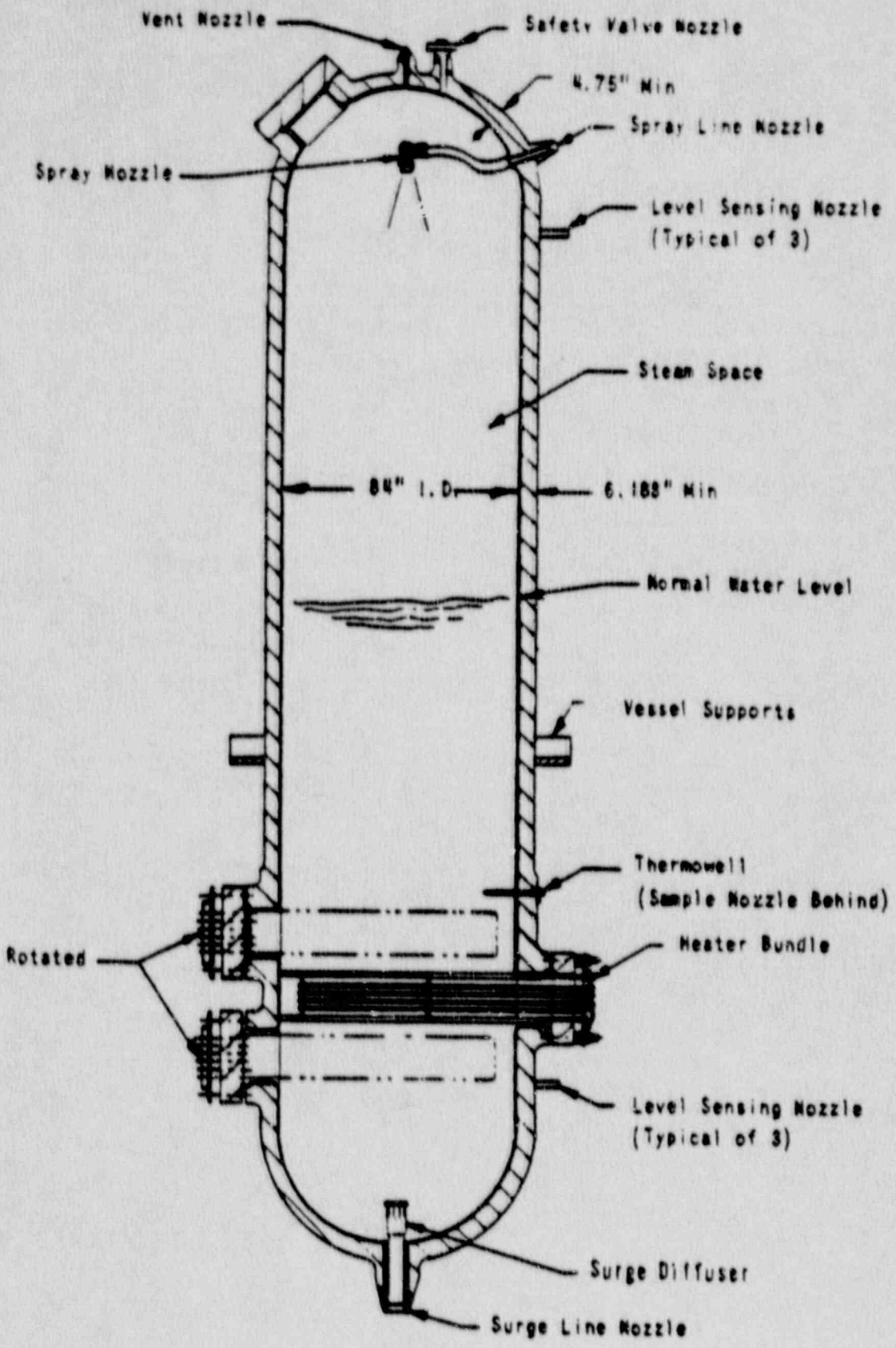


Figure 6.2-5. B & W 177 Plant Pressurizer

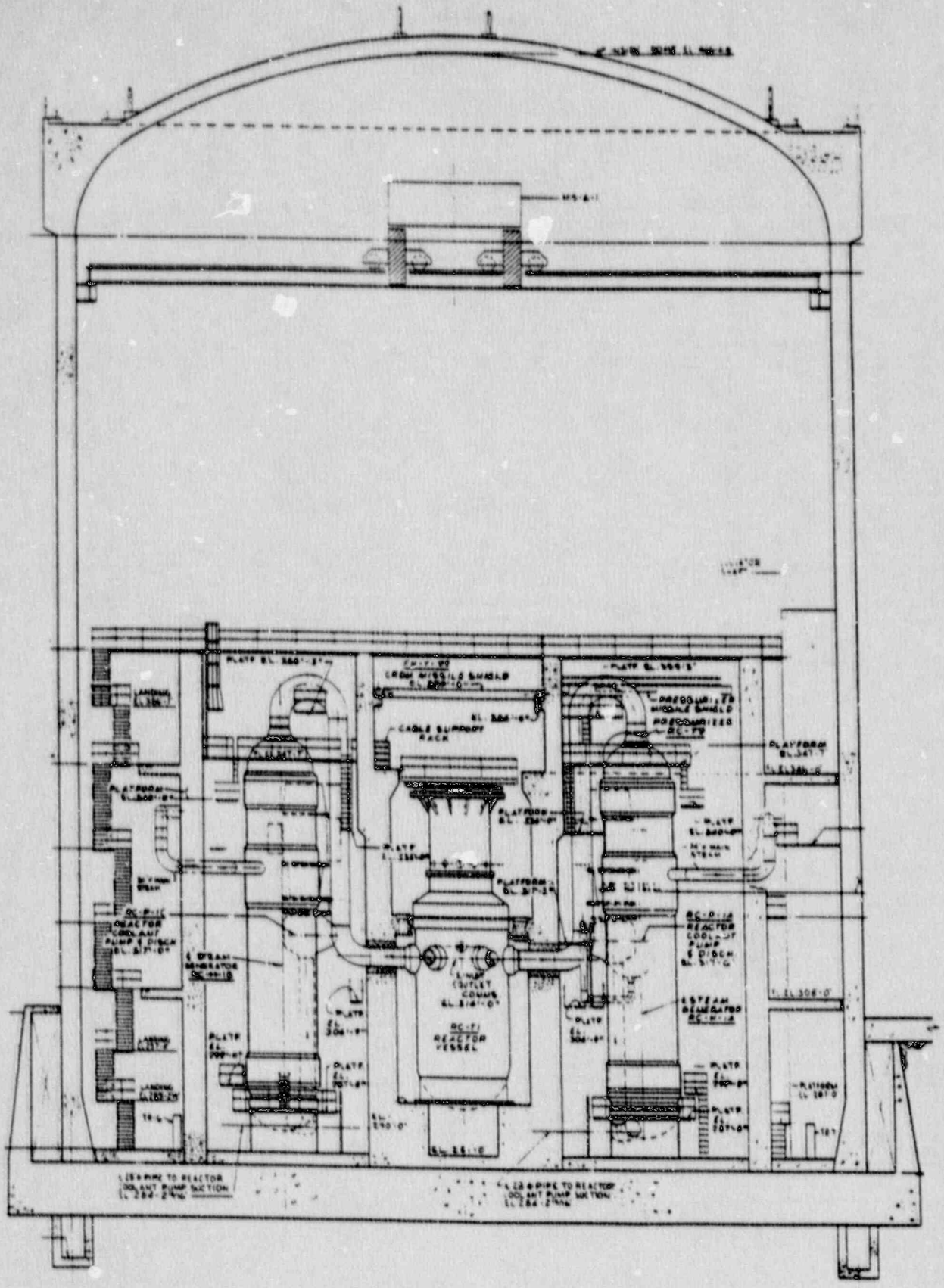


Figure 6.2-6. Section Views of the Three Mile Island 1 Large, Dry Containment (3-D Post-Tensioned Concrete) (Sheet 1 of 2)

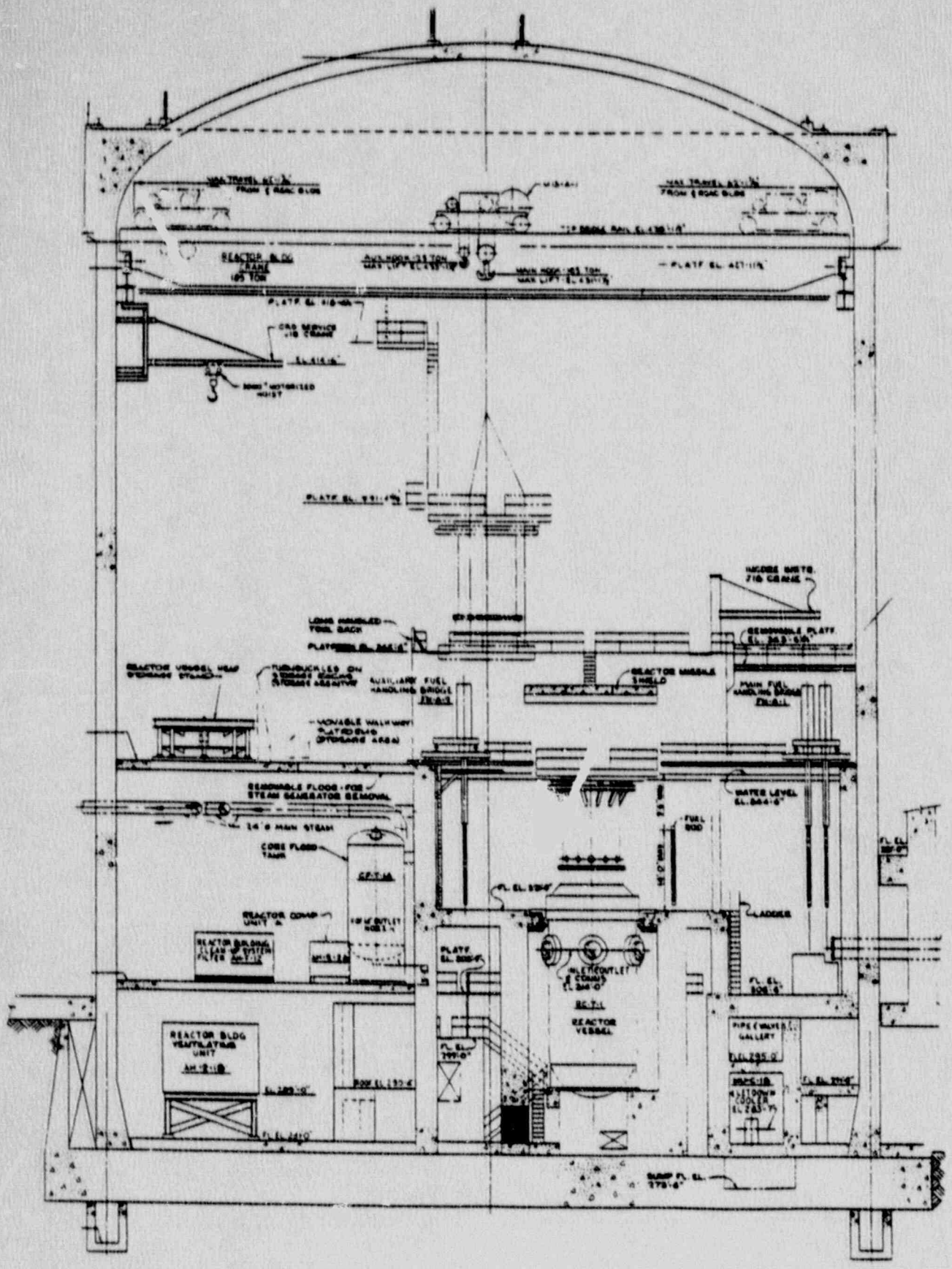


Figure 6.2-6. Section Views of the Three Mile Island 1 Large, Dry Containment (3-D Post-Tensioned Concrete) (Sheet 2 of 2)

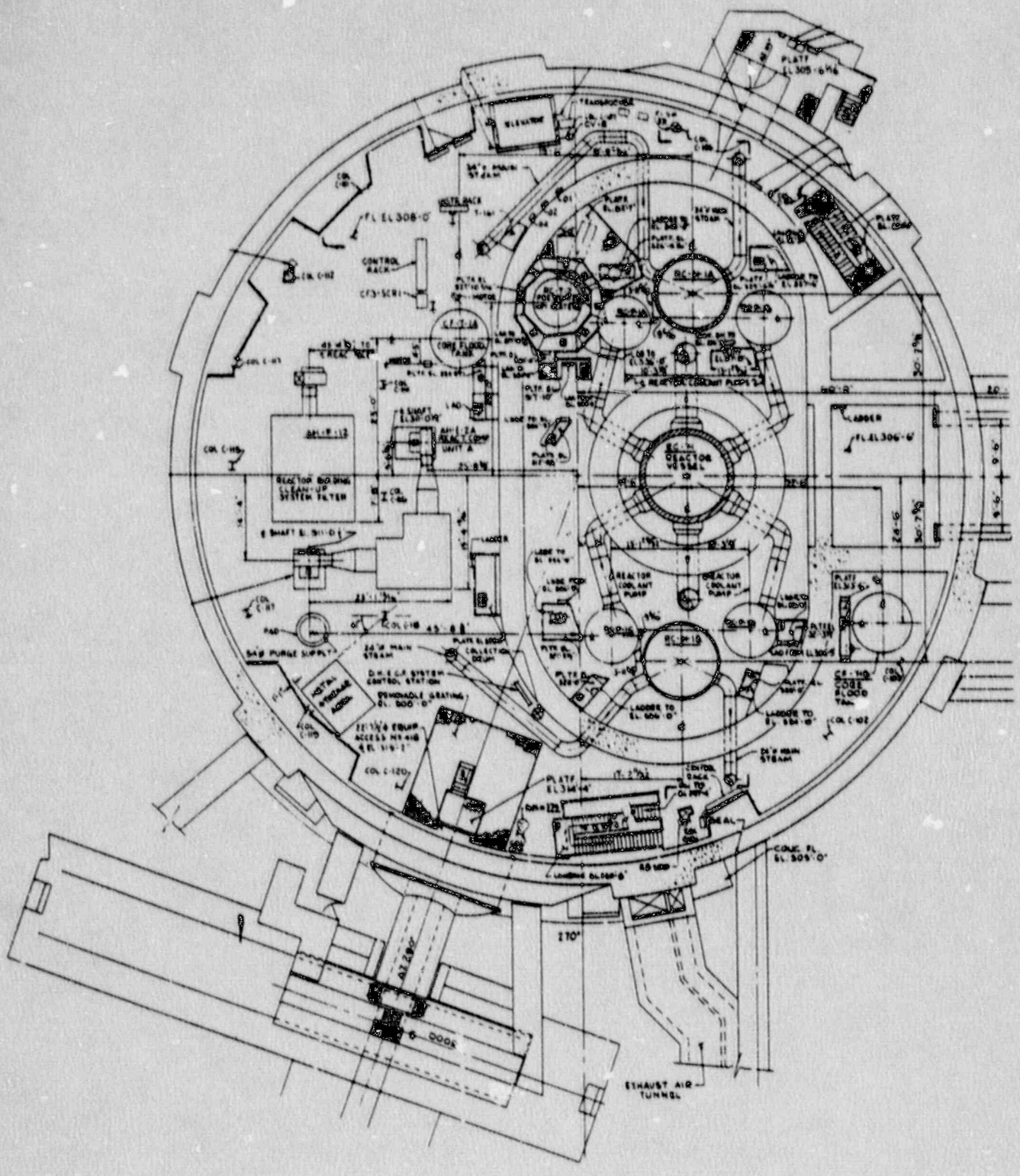


Figure 6.2-7. Plan View of the Three Mile Island 1 Large, Dry Containment at the Level of the Primary Loops

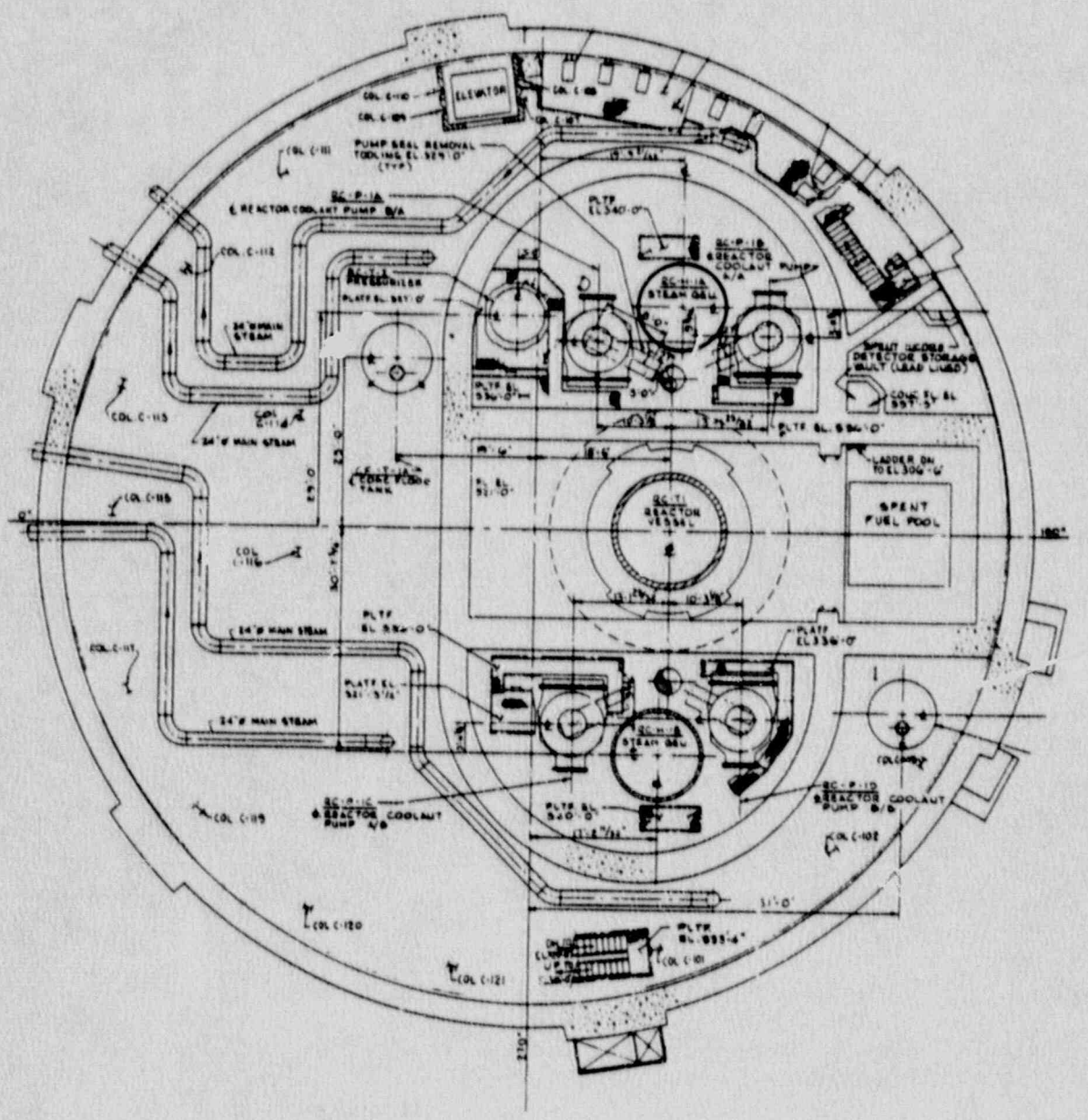


Figure 6.2-8. Plan View of the Three Mile Island 1 Large, Dry Containment at the Level of the Operating Floor

6.3 177 Raised-Loop Babcock & Wilcox PWRs

The Davis-Besse plant is the only Babcock & Wilcox 177 raised-loop PWR plant in the United States. Davis-Besse had a full power operating license as of 2/89. The general arrangement of the 177 raised-loop NSSS is shown in Figures 6.2-1 and 6.2-2.

6.3.1 Reactor Core and Fuel Assemblies

Davis-Besse has a 177 fuel assembly core that operates at an average power density of 90 to 91 kW/liter and has a thermal power output of 2772 MWt. This core is comprised of 15 x 15 Mark B fuel assemblies and is essentially the same as in the 177 lowered-loop plants.

6.3.2 Reactivity Control System

The reactivity control system for Davis-Besse is essentially the same as in the 177 lowered-loop plants, except that fewer full-length CRAs are used at Davis-Besse (i.e., 53 vs. 60 to 61 CRAs in the 177 lowered-loop plants).

6.3.3 Reactor Coolant System

The major difference between the 177 lowered-loop plants and Davis-Besse is the relative position of the reactor vessel and the steam generators. In the Davis-Besse plant, the raised position of the steam generator with respect to the reactor vessel is sufficient to promote natural circulation in the primary system even with low secondary coolant inventory in the steam generators.

The once-through steam generators for Davis-Besse are similar in design to those used in the lowered-loop plants.

6.3.4 Shutdown Cooling Systems

Shutdown cooling is performed by a multi-mode residual heat removal system that also performs the low-pressure safety injection function as part of the ECCS.

6.3.5 Emergency Core Cooling Systems

Unlike the 177 lowered-loop plants, Davis-Besse has separate systems for high-pressure injection and charging. The HPSI pumps have a shutoff head of 1625 psig. Makeup at high pressure is available from two centrifugal charging pumps rated at 150 gpm at 2514 psid (i.e. above the safety valve and PORV setpoints). The makeup system is not considered to be part of the ECCS and makeup capacity is somewhat lower at full RCS pressure than the HPSI pumps in the 177 lowered-loop plants (i.e. 150 gpm vs. 300 to 350 gpm).

The Davis-Besse LPSI system and core flood tanks are comparable to those in the 177 lowered-loop plants.

6.3.6 Containment

The Davis-Besse 177 raised-loop plant has a large, dry containment consisting of a cylindrical steel primary containment vessel inside a concrete shield building. The containment arrangement for Davis-Besse is illustrated in Figure 6.3-3.

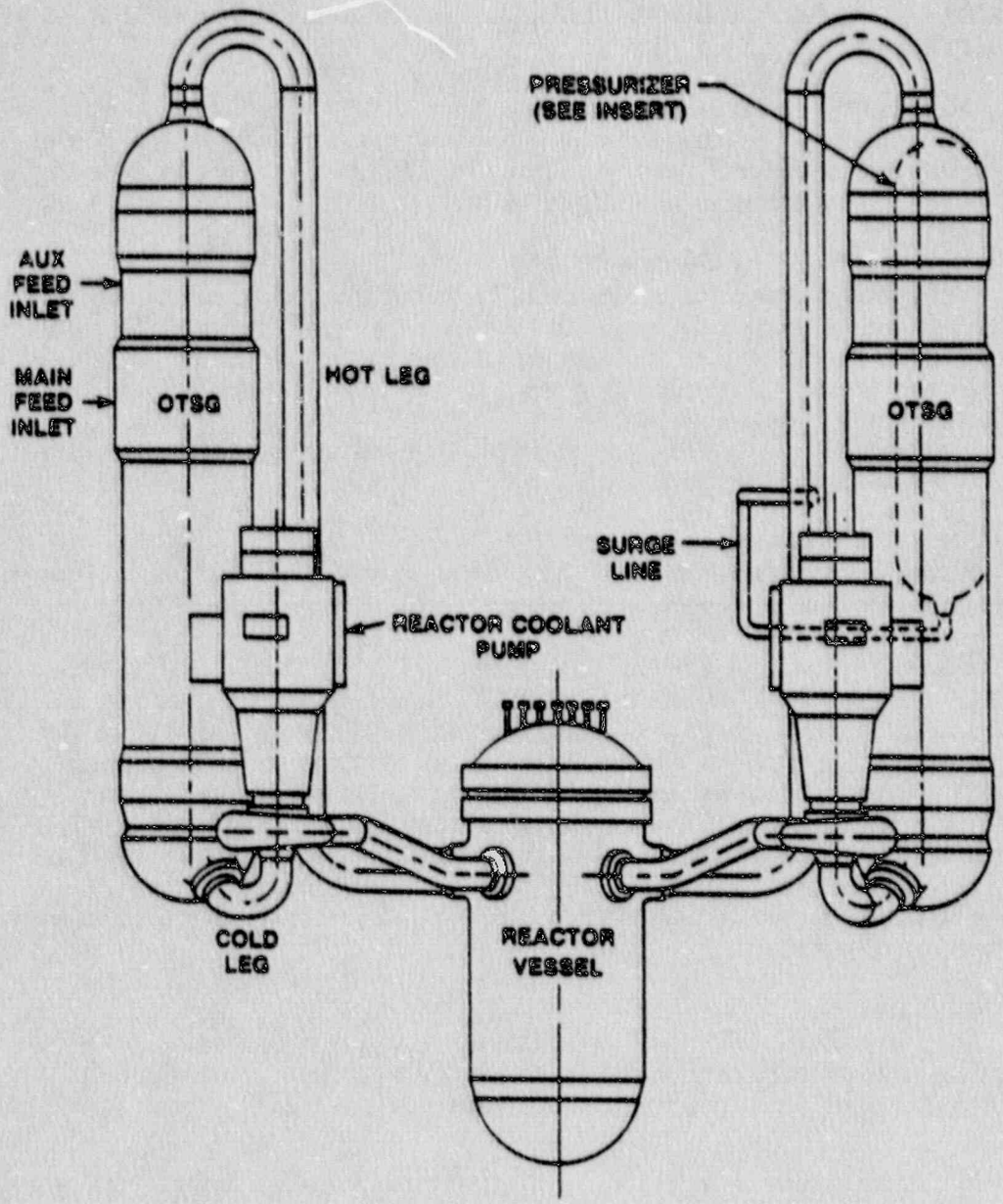


Figure 6.3-1. B & W 177 Raised-Loop NSSS Side View

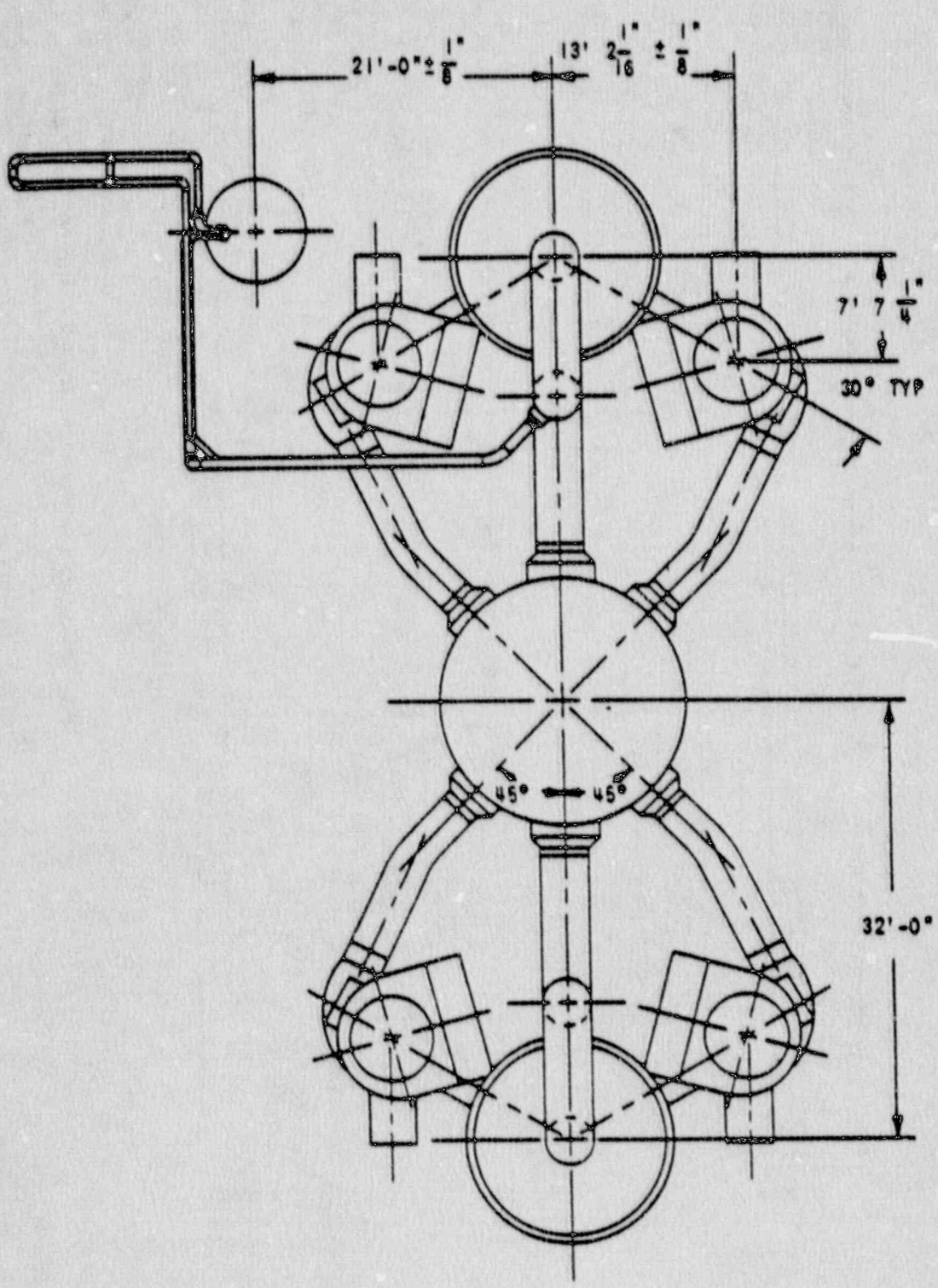


Figure 6.3-2. B & W 177 Raised-Loop NSSS Plan View

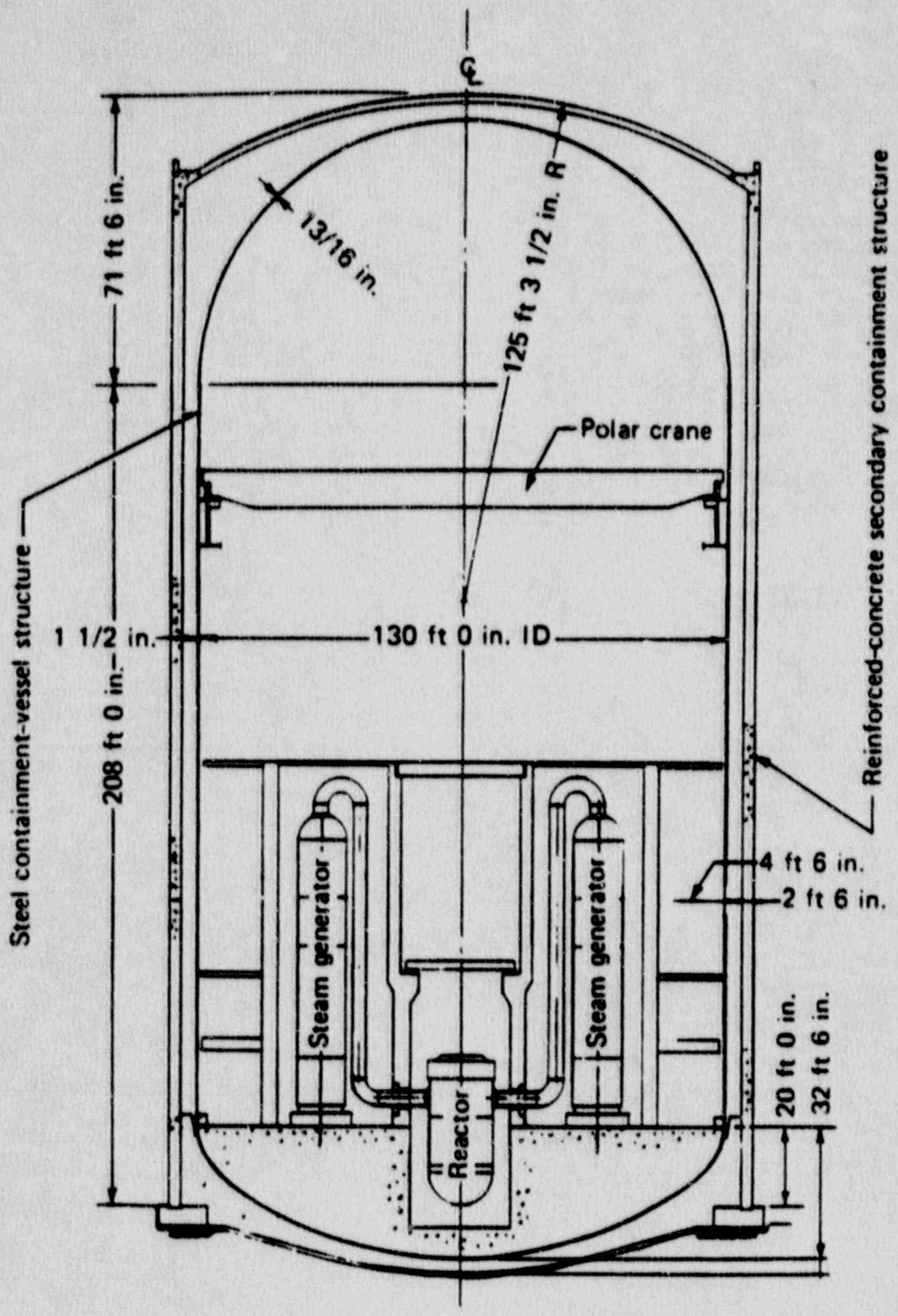


Figure 6.3-3. Section View of the Davis-Besse Large, Dry Containment (Steel Cylinder with Concrete Shield Building)

6.4 205 Raised-Loop Babcock & Wilcox PWRs

The Bellefonte 1 and 2 plants are the only Babcock & Wilcox 205 raised-loop PWR plants in the United States. Bellefonte 1 and 2 have construction permits, however, construction has been halted. The general arrangement of the B & W 205 raised-loop NSSS is shown in Figures 6.4-1 to 6.4-4.

6.4.1 Reactor Core and Fuel Assemblies

Bellefonte 1 and 2 have 205 fuel assembly cores that are designed to operate at an average power density of about 107 kW/liter with a thermal power output of 3620 MWt. This core is comprised of 17 x 17 Mark C fuel assemblies.

6.4.2 Reactivity Control System

Core reactivity is controlled by 64 full-length control rod assemblies (CRAs), 8 axial power shaping rod assemblies (APSRAs), 166 burnable poison rods (BPRs), and soluble boron in the coolant. The neutron absorber in the CRAs is B_4C (boron carbide), the APSRAs use Ag-In-Cd (silver-indium-cadmium), while the BPRs use B_4C (boron carbide) dispersed in alumina pellets. Each CRA and APSRA consists of 16 control rod fingers. Roller-nut rod drive mechanisms control the CRAs and APSRAs.

6.4.3 Reactor Coolant System

The basic appearance of the 205 raised-loop RCS is similar to that of Davis-Besse. The general arrangement of the reactor vessel and internal structures is shown in Figures 6.4-5. Reactor vessel inside diameter is 182 inches.

The once-through steam generator for a 205 raised-loop plant is shown in Figure 6.4-6. This steam generator differs in the following areas from the OTSGs in earlier plants:

- Greater heat transfer capability
- Feedwater inlets and steam outlets located lower on the steam generator vessel

The 205 plant steam generator is termed an "integral economizer" OTSG. The pressurizer for a 205 raised loop plant is shown in Figure 6.4-7.

6.4.4 Shutdown Cooling Systems

As in earlier B & W plants, shutdown cooling is performed by a multi-mode system that also performs the low-pressure safety-injection (LPSI) function as part of the ECCS.

6.4.5 Emergency Core Cooling Systems

The 205 raised-loop plant ECCS is comparable to the ECCS found in the 177 lowered-loop plants. The high-head injection pumps in the B & W 205 plant double as charging pumps during normal operation. They are, therefore, capable of adding water to the RCS against full system pressure. No positive displacement charging pumps are provided.

6.4.6 Containment

The Bellefonte plant has a large, dry containment constructed of three-dimension post-tensioned concrete with a steel liner and a secondary containment. The general arrangement of the primary containment is shown in Figures 6.4-8 and 6.4-9.

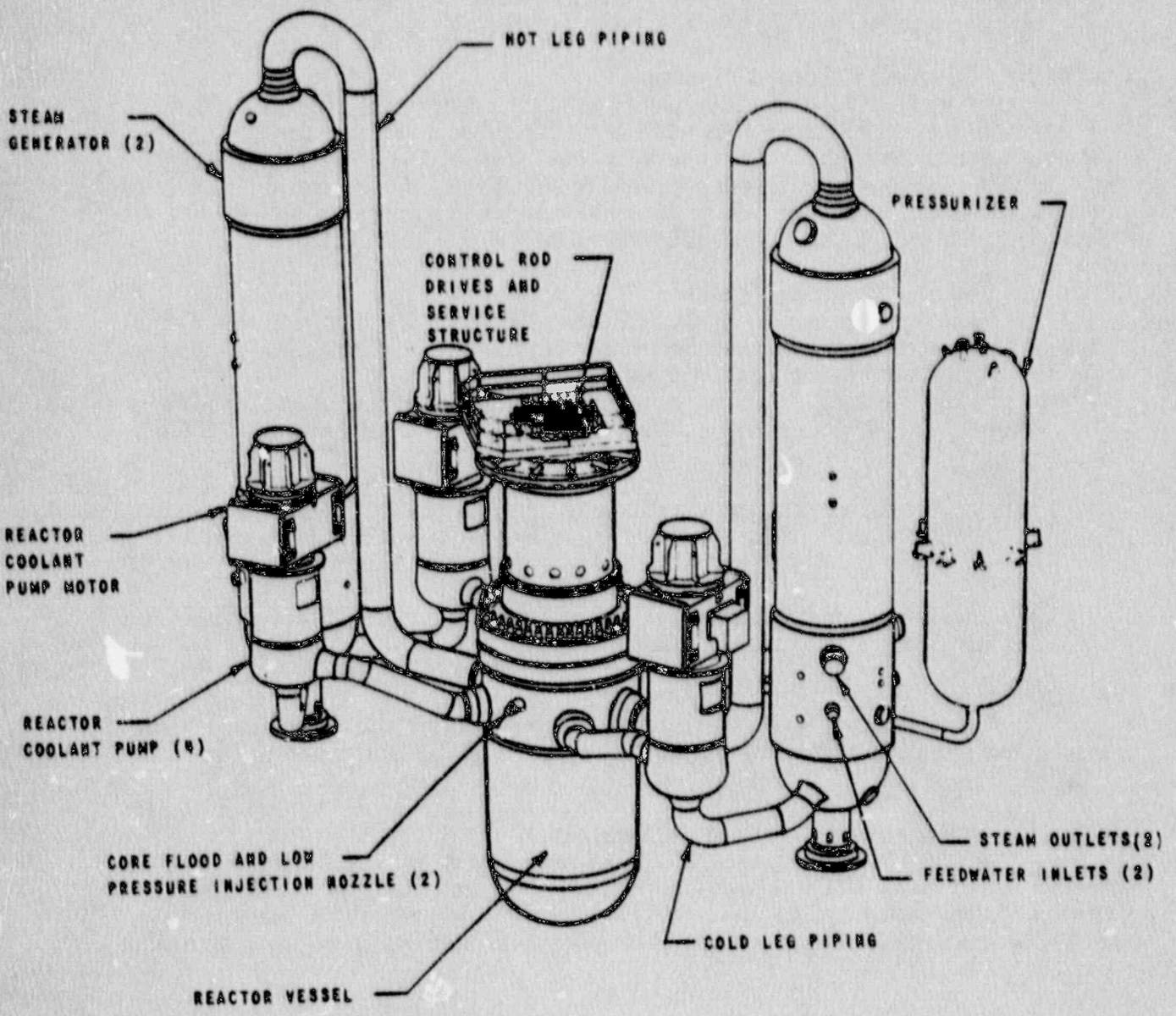


Figure 6.4-1. B & W 205 Raised-Loop NSSS

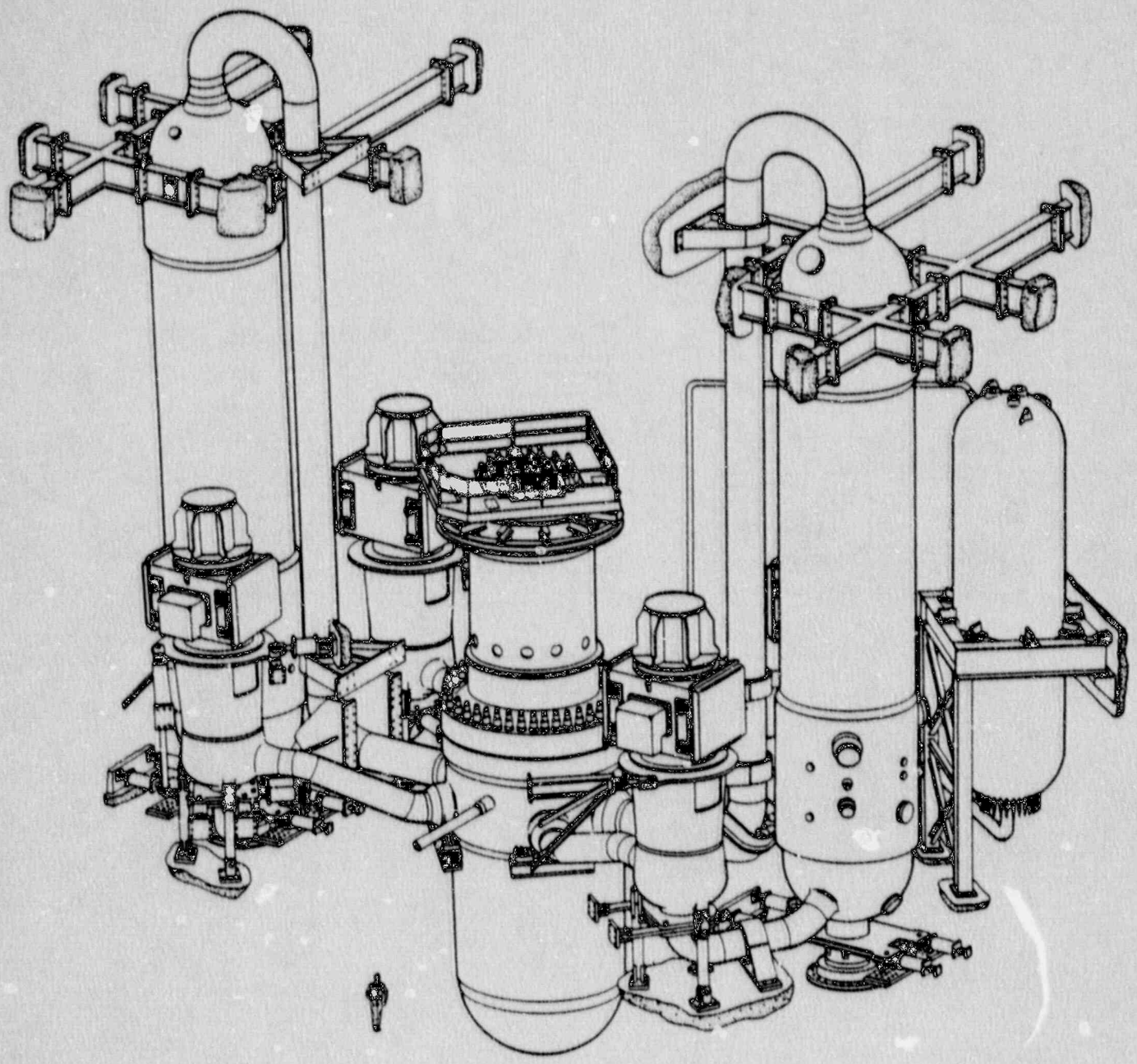


Figure 6.4-2. B & W 205 Raised-Loop NSSS Showing Component Supports

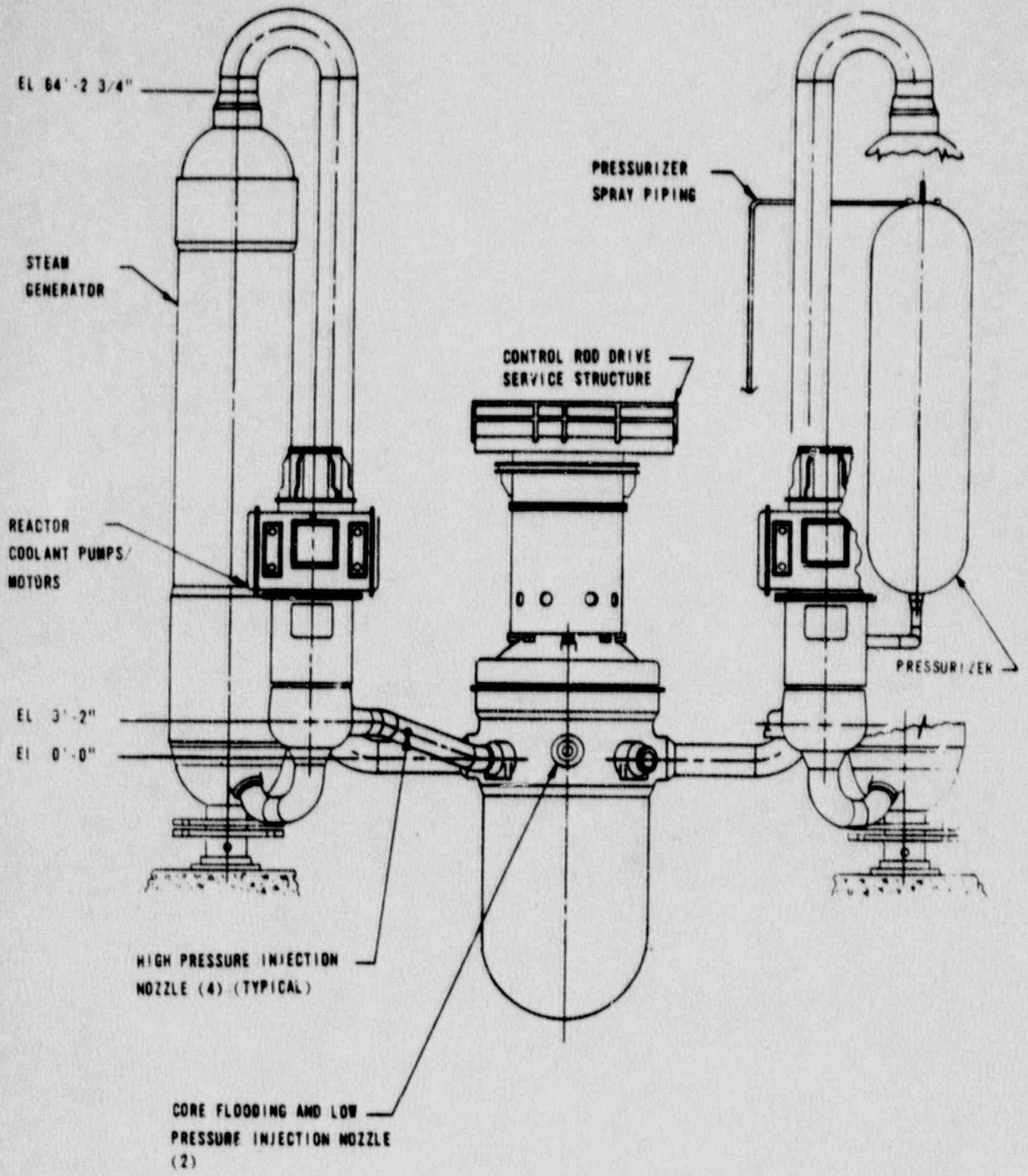


Figure 6.4-3. B & W 205 Raised-Loop NSSS Side View

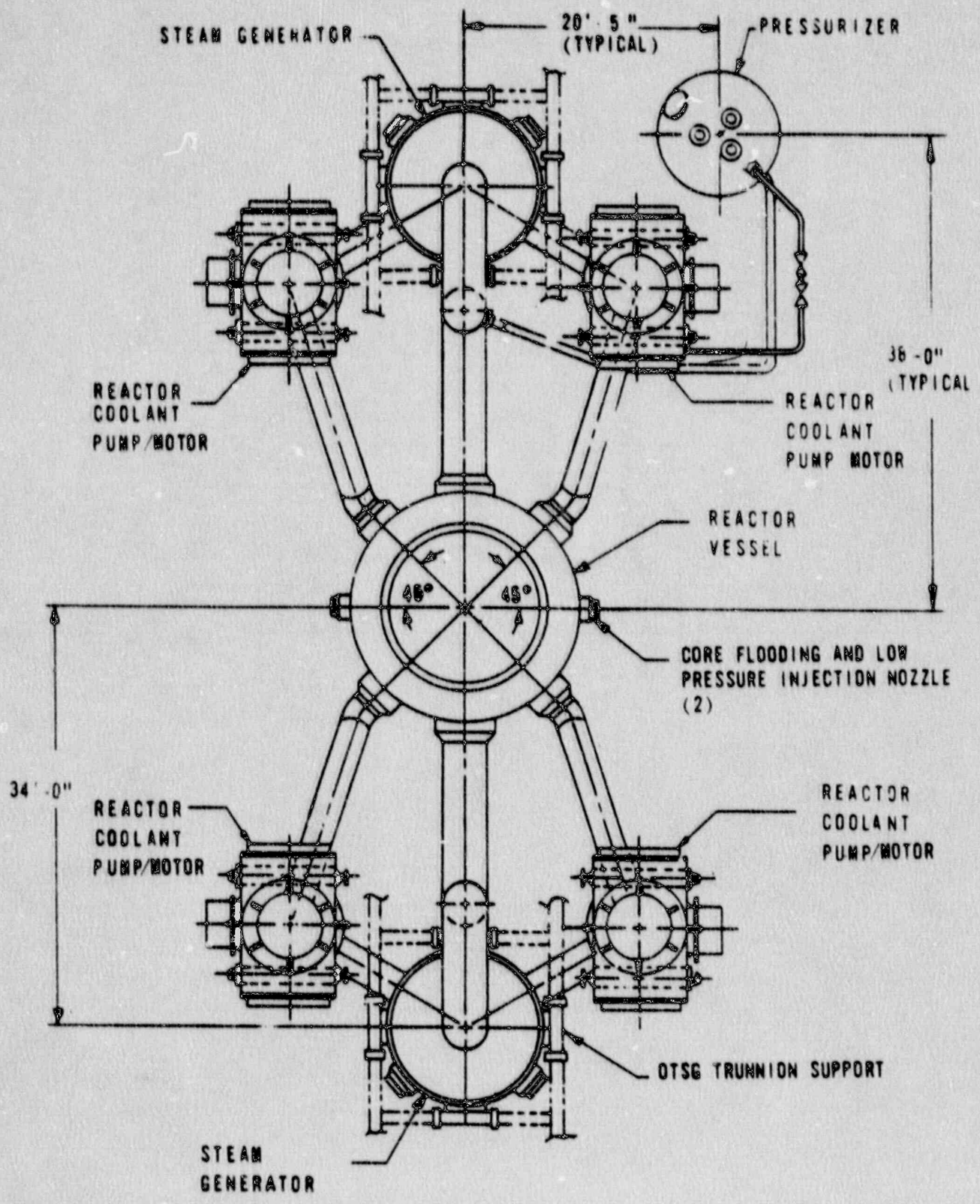


Figure 6.4-4. B & W 205 Raised-Loop NSSS Plan View

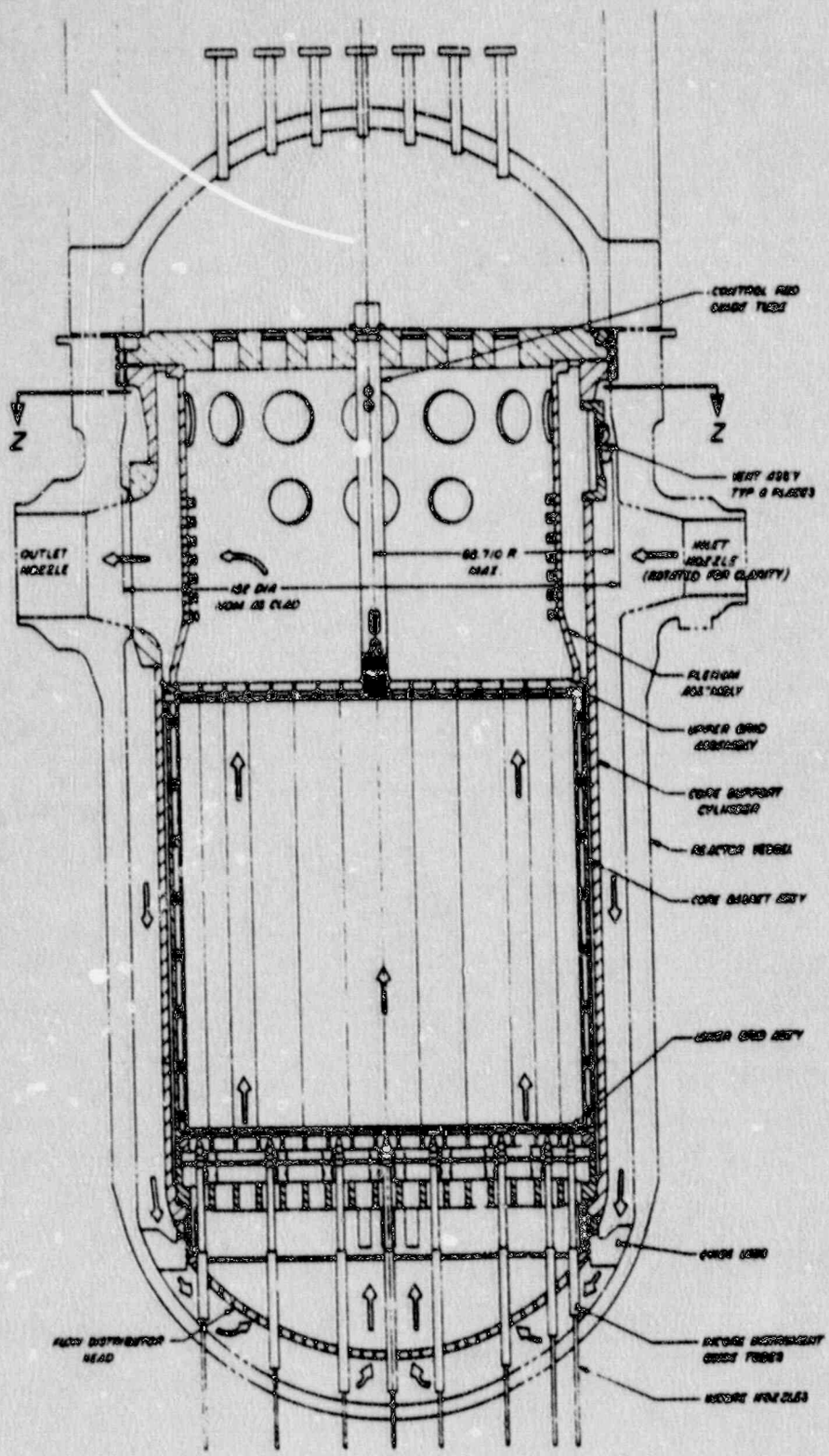


Figure 6.4-5. General Arrangement of the B & W 205 Plant Reactor Vessel and Internals

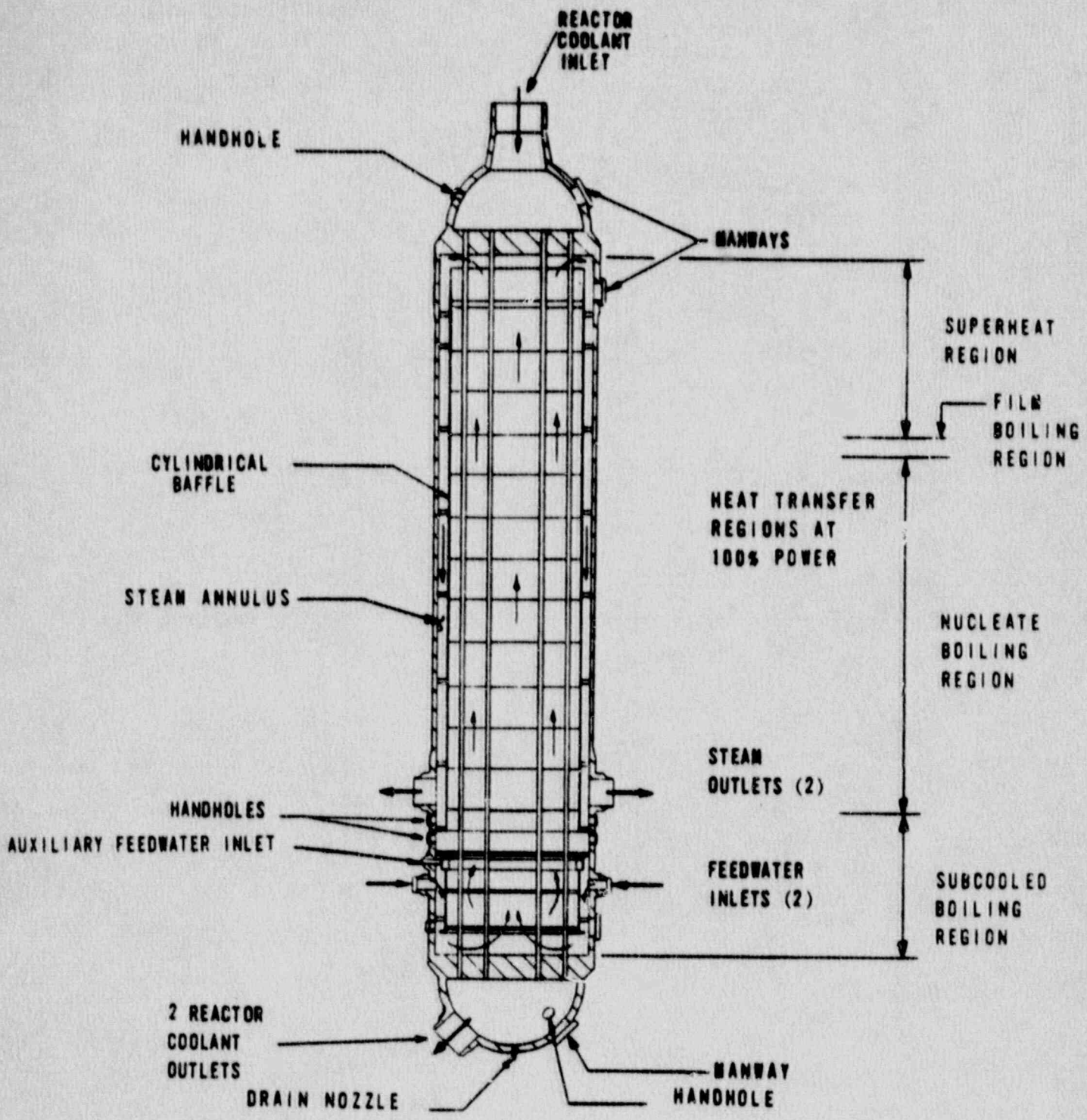


Figure 6.4-6. B & W 205 Plant Once-Through Steam Generator

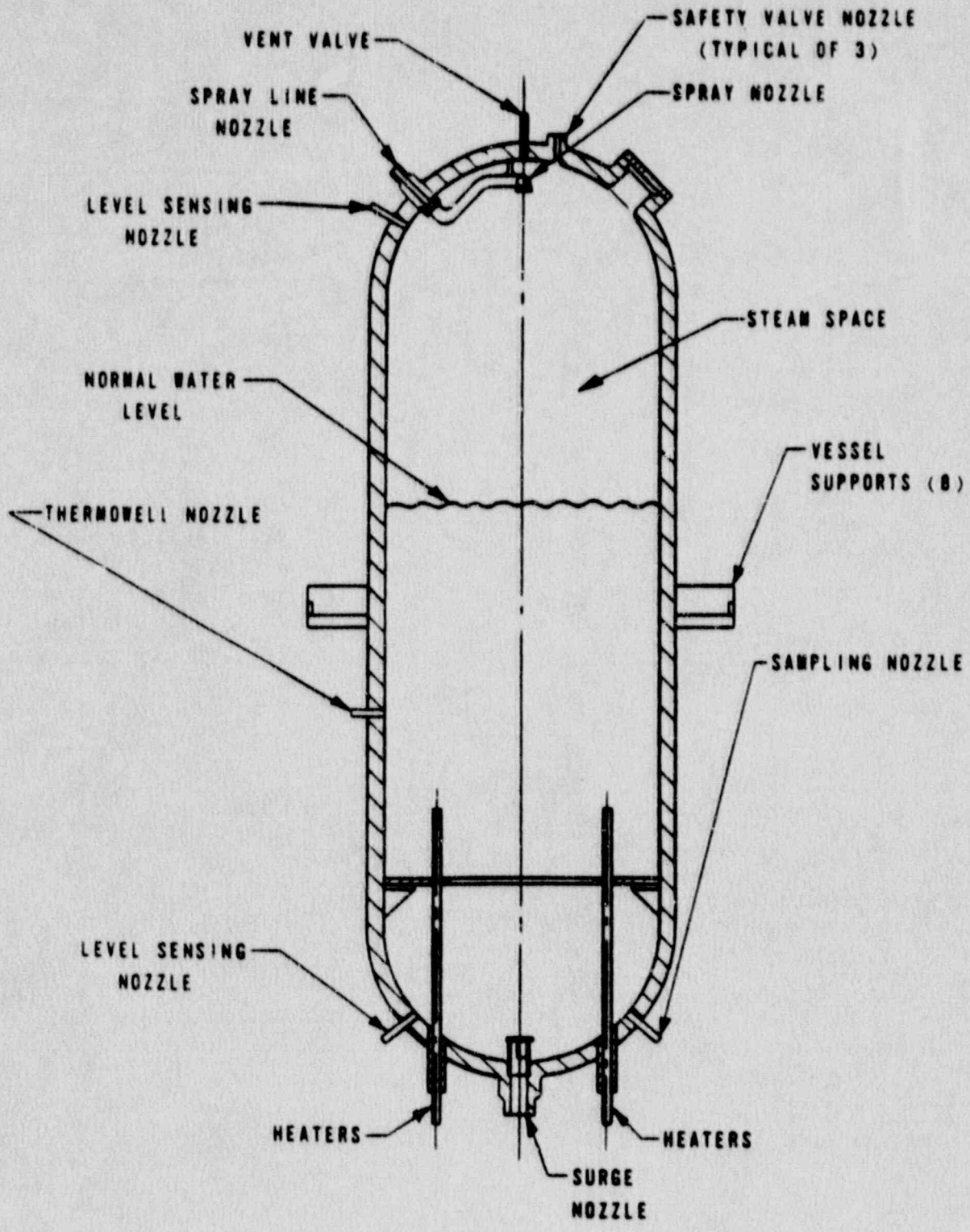


Figure 6.4-7. B & W 205 Plant Pressurizer

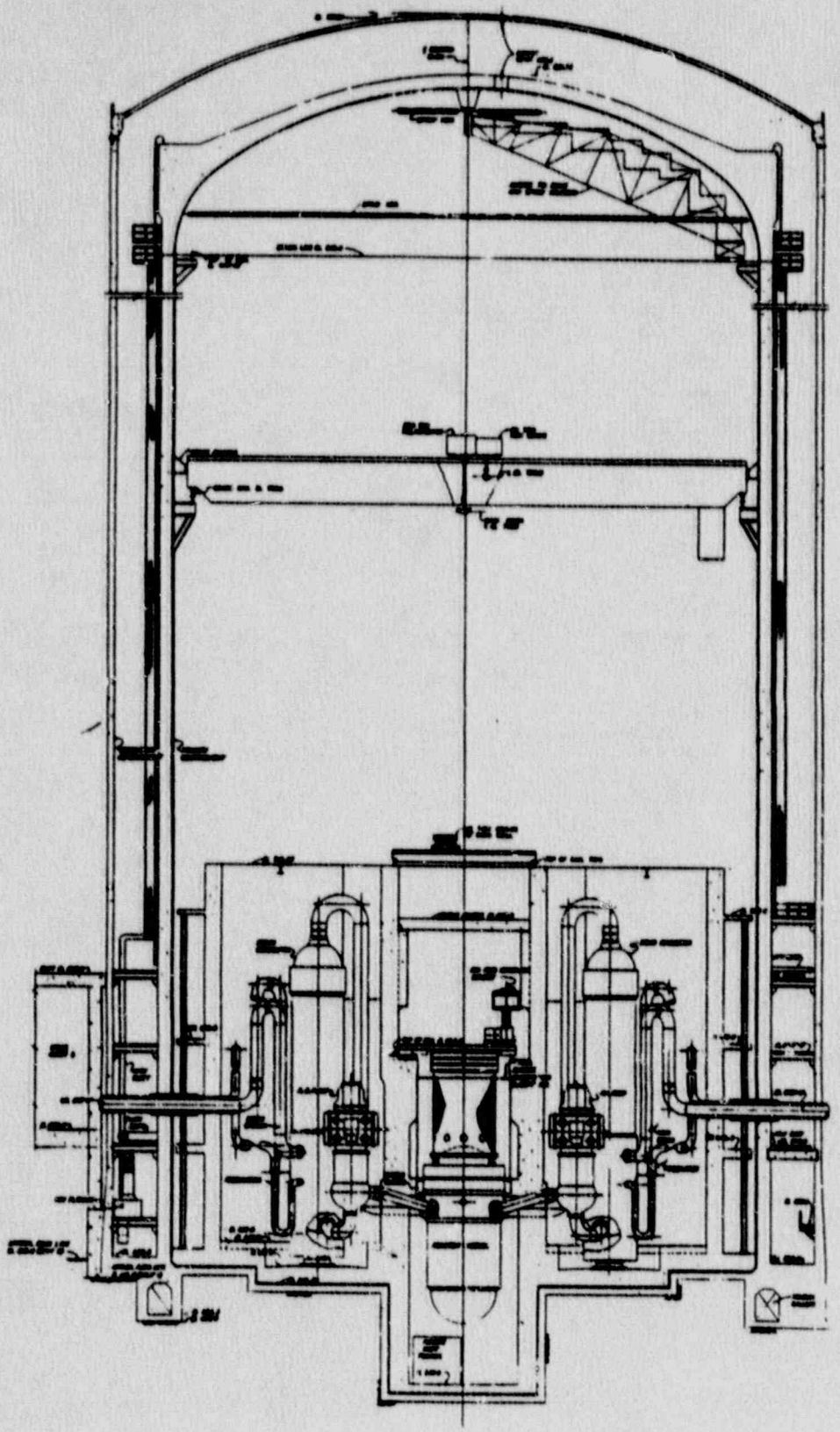


Figure 6.4-8. Section View of the Bellefonte Large, Dry Containment (3-D Post-Tensioned Concrete)

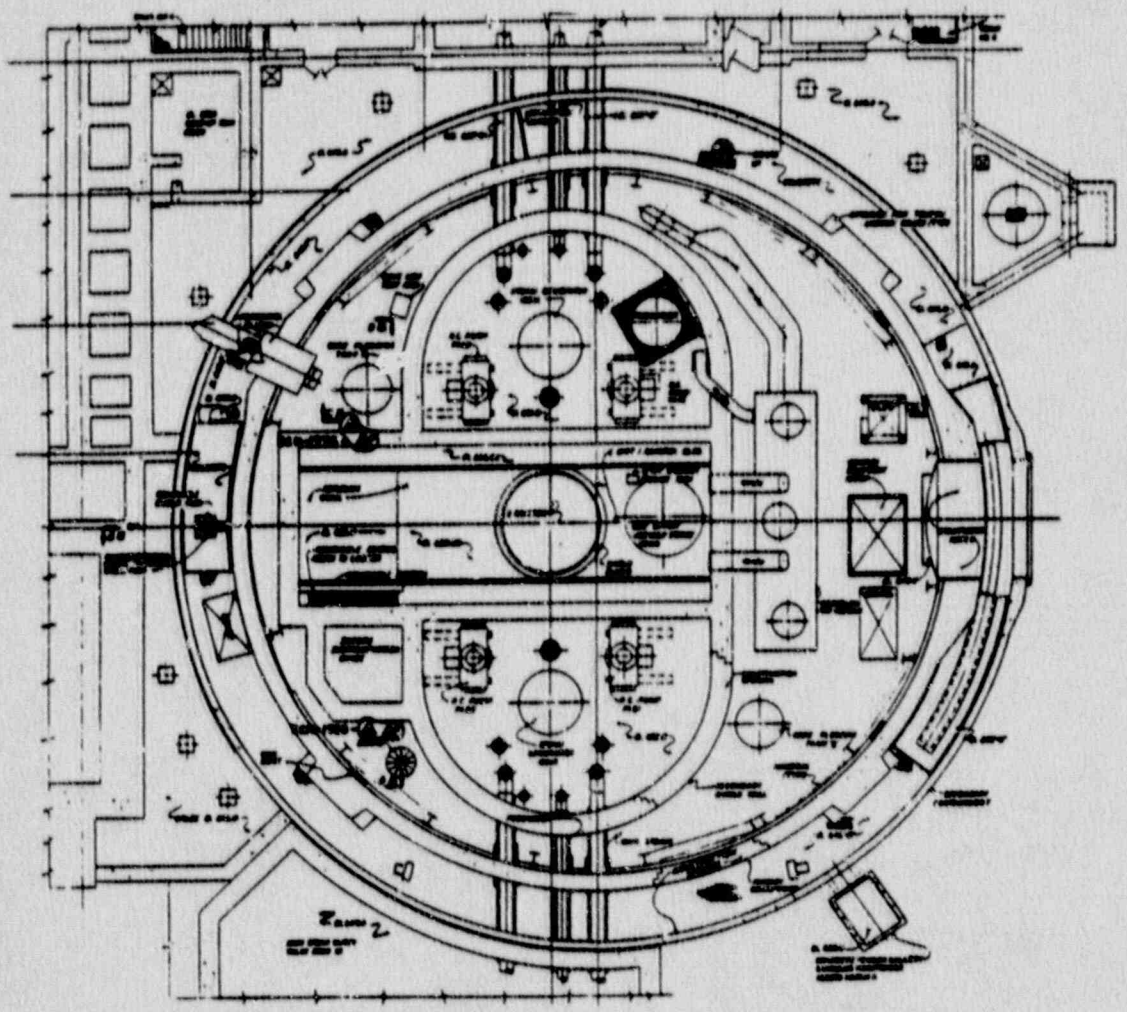


Figure 6.4-9. Plan View of the Bellefonte Large, Dry Containment (3-D Post-Tensioned Concrete)

6.5 Babcock & Wilcox PWR Comparative Data

This section contains the following tables which present comparative system data for Babcock and Wilcox PWRs:

- Table 6.5-1 Design Parameters for Representative Babcock and Wilcox PWRs
- Table 6.5-2 Comparison of Babcock & Wilcox PWR Vessel and Core Parameters
- Table 6.5-3 Babcock & Wilcox PWR System Comparison - RCS, AFW, Charging and HPSI
- Table 6.5-4 Comparison of Babcock & Wilcox PWR Pressurizer Relief Capacity
- Table 6.5-5 Comparison of Babcock & Wilcox PWR Containments
- Table 6.5-6 Comparison of Babcock & Wilcox PWR Backup Electric Power Systems
- Table 6.5-7 Comparison of Babcock & Wilcox PWR Power Conversion Systems

Table 6.5-1. Design Parameters for Representative Babcock & Wilcox PWRs

REACTOR PLANT CHARACTERISTICS	B&W LOWERED- LOOP PLANT TMI-1	B&W RAISED- LOOP PLANT DAVIS-BESSE
Overall		
Number of loops	2 without isolation	2 without isolation
Thermal capacity	2535 MWt	2772 MWt
Electric capacity	776 MWe	860 MWe
Efficiency (net)	30.61%	31.02%
Coolant pressure in primary circuit at exit from reactor	2185 psig	2185 psig
Coolant temperature at inlet	555.6°F	555.4 °F
Coolant temperature at exit from reactor	602.4°F	608.6 °F
Coolant flow rate through reactors (total)	Unk.	131.3E6 lb/hr
Core		
Height of active core	12 ft.	12 ft.
Equivalent diameter	10.74 ft.	10.74 ft.
Number of fuel assemblies	177	177
Number of control rods assemblies	61	53
Number of fuel elements in assembly	208	208
Diameter of fuel element	0.370 in.	0.370 in.
Area of heat transfer surface	49,734 sq.ft.	49,734 sq.ft.
Mean specific heat flux	182,400 Btu/hr-sq.ft.	182,400 Btu/hr-sq.ft.
Number of fuel rods	36,816	36,816
Core loading	3-region	3-region
Average burnup (first cycle)	14,400 MWd/MTu	14,605 MWd/MTu
Fuel weight (as U02)	93.1 metric tons	93.1 metric tons
Reactor Vessel		
Vessel height	40.73	40.73
Inner diameter	14.25	14.25
Number of Openings for Inlet and Outlet nozzles	2 x 2	2 x 2

Table 6.5-1. Design Parameters for Representative Babcock & Wilcox PWRs (Continued)

REACTOR PLANT CHARACTERISTICS	B&W LOWERED-LOOP PLANT TMI-1	B&W RAISED-LOOP PLANT DAVIS-BESSE
Steam Generator		
Number of units	2	2
Thermal power per unit	1292 MWt	1386 MWt
Shell side operating pressure (steam)	910 psig	910 psig
Tube side operating pressure	2185 psig	2185 psig
Tube side design flow	65.66 E6 lb/hr	65.66 E6 lb/hr
Piping		
Hot leg inner dia.	36 in.	36 in.
Cold leg inner dia.	28 in.	28 in.
Between pump and steam generator	Unk.	Unk.
Safety Valves		
First opening pressure	2435 psig	2435 psig
Capacity	243,000 lb/hr each of 2	242,000 lb/hr each of 2
	38.9 lb/hr/MWt each PORV of 1	40.4 lb/hr/MWt each PORV of 1
Primary Coolant Pumps		
Number	4	4
Pump capacity	88,000 gpm	88,000 gpm
Coolant temperature	557°F	557 °F
Pressure rise	355 ft.head	355 ft.head
Design pressure	2500 psig	2500 psig
Design temperature	650 °F	650 °F
Motor rating (nameplate)	9,000 hp.	9,000 hp.

Table 6.5-2. Comparison of Babcock & Wilcox 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)
B&W	177 L-loop	ANO-1	2568	171	128.9	144	76.00	177	15 x 15	61 F/8 P
B&W	177 L-loop	Crystal River 3	2452	171	128.9	144	79.60	177	15 x 15	60 F/8 P
B&W	177 L-loop	Oconee 1, 2 & 3	2568	171	128.9	144	83.40	177	15 x 15	61 F/8 P
B&W	177 L-loop	Rancho Seco	2772	171	128.9	144	90.00	177	15 x 15	61 F/8 P
B&W	177 L-loop	Three Mile Island 1	2535	171	128.9	144	82.30	177	15 x 15	61 F/8 P
B&W	177 R-loop	Davis-Besse	2772	171	128.9	144	90.60	177	15 x 15	53 F/8 P
B&W	205 R-loop	Bellefonte 1 & 2	3620	182	138.7	143	107.00	205	17 x 17	64 F/8 P

Table 6.5-3. Babcock & Wilcox PWR System Comparison - RCS, AFW, Charging and HPSI

		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
ANO-1	B&W	2568	2	1/2	OT	2 1	M T	780 @ 1127 780 @ 1127	3	Cent	400 @ 1600	270	Same as charging pumps				
Bellefonte 1 & 2	B&W	3820	2	1/2	OT	unk.	unk.	unk.	1	Cent	308 @ 3500	unk.	unk.	unk.	780	unk.	
Crystal River 3	B&W	2452	2	1/2	OT	1 1	M T	780 @ unk 740 @ unk	3	Cent	400 @ 1600	260	Same as charging pumps				
Davis Besse	B&W	2772	2	1/2	OT	2	T	1050 @ 1050	2	Cent	150 @ 2514	150	2	Cent	500 @ 1170	0	
Oconee 1, 2, & 3	B&W	2568	2	1/2	OT	1 1	M T	500 @ 1050 1080 @ 1050	3	Cent	450 @ 1700	270	Same as charging pumps				
Rancho Seco	B&W	2772	2	1/2	OT	2	M-T T	840 @ 1162 840 @ 1162	3	Cent	400 @ 1600	270	Same as charging pumps				(h)
TMI-1	B&W	2535	2	1/2	OT	1 1	M T	460 @ unk 920 @ unk	3	Cent	600 @ 1600	270	Same as charging pumps				

Table 6.5-3. Babcock and Wilcox 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 OT = once-through

Notes:

(h) Rancho Seco has one AFW pump with a tandem steam turbine/electric motor drive.

Table 6.5-4. Comparison of Babcock & Wilcox 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)
ANO-1	B&W	1	Dresser 315 33VX	38.9	2255	2	234	2500
Bellefonte 1 & 2	B&W	1	Unk.	160*	2295	2	500	2500
Crystal River 3	B&W	1	Dresser 315 33VX	40.8	2255	2	254	2500
Davis Besse	B&W	1	Crosby HPN-SN	40.4	2255	2	242	2435
Oconee 1, 2, & 3	B&W	1	Dresser 315 33VX	41.7	2255	2	254	2500
Rancho Seco	B&W	1	Dresser 315 33VX	40.4	2255	2	249	2500
TMI-1	B&W	1	Dresser 315 33VX	38.9	2255	2	243	2435

* Klb/hr rating

Table 6.5-5. Comparison of Babcock & Wilcox 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?
ANO-1	B&W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	2.09E+06	59	0.2	No
Bellefonte 1 & 2	B&W	TVA	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	135	3.80E+06	50	0.2	Yes
Crystal River 3	B&W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.00E+06	55	0.25	No
Davis Besse	B&W	Betchel	Dry	Steel Cylinder	- - -	130	2.87E+06	40	0.5	Yes
Oconee 1, 2, & 3	B&W	Duke / Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.91E+06	59	0.5	No
Rancho Seco	B&W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	1.98E+06	59	0.1	No
TMI-1	B&W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.00E+06	55	0.1	No

Table 6.5-6. Comparison of Babcock & Wilcox 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
ANO-1	B&W	None	2	2750	Gen. Motors	2	125	
Bellefonte 1 & 2	B&W	None	2	7000	DeLavalle	4	120	*Per 6.9 kV diesel generator
Crystal River 3	B&W	None	2	2750	Fairbanks-Morse	2	125/250	
Davis-Besse	B&W	None	2	2600	Gen. Motors	4	125/250	
Oconee 1, 2 & 3	B&W	2 *		87500*	unk.	4	125	* Keowee Dam hydroelectric generators
Rancho Seco	B&W	None	2 2	3500 2750	unk. Gen. Motors	4	125	
Three Mile Island 1	B&W	None	2	2600	Fairbanks-Morse	2	125/250	

Table 6.5-7. Comparison of Babcock & Wilcox 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)
ANO-1	B&W	Bechtel	Bechtel	836	15	Once Through	Water Discharge Tunnel	2	turbine	unk.	14,750 @ 824 psig
Bellefonte 1 & 2	B&W	TVA	TVA	1213	50	Closed Loop	Nat. Cooling Tower	2	turbine	1,161	16,585
Crystal River 3	B&W	Gilbert	Jones	821	15	Once Through	Gulf of Mexico	2	turbine	unk.	13,300 @988 psig
Davis Besse	B&W	Bechtel	Bechtel	860	25	Closed Loop	Nat. Cooling Tower	2	turbine	1,110	15,000 @932 psig
Oconee 1, 2, & 3	B&W	Bechtel/ Duke Power	Duke Power	846	25	Once Through	Lake Keowee	2 (75%)	turbine	unk.	25,000 @1060 psig
Rancho Seco	B&W	Bechtel	Bechtel	873	15	Closed Loop	Nat. Cooling Towers	2	turbine	1,387	unk.
TMI-1	B&W	Gilbert	UE&C	776	15	Closed Loop	Nat. Cooling Tower	2	turbine	unk.	unk.

7. COMPARATIVE PRESSURIZED WATER REACTOR (PWR) DATA

Comparisons of basic plant parameters and system design features of the Westinghouse, C-E, and B&W NSSS are presented in the following tables in this section:

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

Table 7-1. Comparison of PWR Vessel and Core Parameters

Reactor Plant Name	PWR Vendor	PWR Type	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)
ANO-1	B&W	177 L-loop	2568	171	128.9	144	76.00	177	15 x 15	61 F/8 P
ANO-2	C-E	2-loop	2815	171	123	150	96.60	177	16 x 16	73 F/8 P
Beaver Valley 1	W	3-loop	2660	157	119.5	143.7	100.00	157	17 x 17	48 F/5 P
Beaver Valley 2	W	3-loop	2660	157	119.5	143.7	100.00	157	17 x 17	48 F/5 P
Bellefonte 1 & 2	B&W	205 R-loop	3620	182	138.7	143	107.00	205	17 x 17	64 F/8 P
Braidwood 1 & 2	W	4-loop	3411	173	132.7	143.7	104.50	193	17 x 17	53 (total)
Byron 1 & 2	W	4-loop	3411	173	132.7	143.7	104.50	193	17 x 17	53 (total)
Callaway	W	4-loop	3425	173	132.7	143.7	109.20	193	17 x17	53 (total)
Calvert Cliffs 1 & 2	C-E	2-loop	2700	172	136	136.7	78.5 to 82.8	217	14 x 14	77 F/ 8 P
Catawba 1 & 2	W	4-loop	3411	173	132.7	143.7	103.5 to 44.6	193	17 x 17	53 (total)
Comanche Peak 1 & 2	W	4-loop	3425	167	132.7	143.7	103.3 to 104.5	193	17 x 17	53 (total)
Crystal River 3	B&W	177 L-loop	2452	171	128.9	144	79.60	177	15 x 15	60 F/8 P
D. C. Cook 1 & 2	W	4-loop	3250	173	132.7	143.7	98.0 to 103.8	193	17 x 17	53 (total)
Davis-Besse	B&W	177 R-loop	2772	171	128.9	144	90.60	177	15 x 15	53 F/8 P
Diablo Canyon 1 & 2	W	4-loop	3338	173	132.7	143.7	102.3 to 104.5	193	17 x 17	53 F/8 P
Farley 1 & 2	W	3-loop	2652	157	119.5	144	101.10	157	17 x 17	45 (total)
Fort Calhoun 1	C-E	2-loop	1420	140	106.45	128	80.40	133	14 x 14	45 F/4 P
Ginna	W	2-loop	1520	132	96.5	144	89.00	121	14 x 14	29 F/4 P
Haddam Neck	W	4-loop	1825	154	119.6	121.8	82.00	157	15 x 15	45 (total)
Indian Point 2	W	4-loop	2758	173	132.7	144	85.00	193	15 x 15	53 F/8 P

Table 7-1. Comparison of PWR Vessel and Core Parameters (Continued)

Reactor Plant Name	PWR Vendor	PWR Type	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)
Indian Point 3	W	4-loop	3025	173	132.7	144	92.70	193	15 x 15	53 F/8 P
Kewaunee	W	2-loop	1650	132	96.5	144	94.90	121	14 x 14	33 (total)
Maine Yankee	C-E	3-loop	2530	176	136	136.7	74.60	217	14 x 14	77 F/8 P
McGuire 1 & 2	W	4-loop	3425	173	132.7	143.7	103.50	193	17 x 17	53 (total)
Millstone 2	C-E	2-loop	2560	172	136	136.7	78.50	217	14 x 14	73 F/0 P
Millstone 3	W	4-loop	3411	173	132.7	143.7	104.50	193	17 x17	53 F/8 P
North Anna 1 & 2	W	3-loop	2775	157	119.5	144	108.70	157	17 x 17	48 (total)
Oconee 1, 2 & 3	B&W	177 L-loop	2568	171	128.9	144	83.40	177	15 x 15	61 F/8 P
Palisades	C-E	2-loop	2530	172	136.7	132	79.80	204	15 x 15	41 F/4 P
Palo Verde 1, 2 & 3	C-E	2-loop Sys 80	3800	182	143.6	150	95.60	241	16 x 16	76 F/13 P
Point Beach 1 & 2	W	2-loop	1518	132	96.5	144	87.00	121	14 x 14	37 (total)
Prairie Island 1 & 2	W	2-loop	1650	132	96.5	144	95.90	121	14 x 14	29 F/4 P
Rancho Seco	B&W	177 L-loop	2772	17.	128.9	144	90.00	177	15 x 15	61 F/8 P
Robinson 2	W	3-loop	2200	156	119.5	144	82.60	157	15 x 15	48 F/5 P
Salem 1 & 2	W	4-loop	3338	173	132.7	143.7	102.60	193	17 x 17	53 (total)
San Onofre 1	W	3-loop	1347	144	119.5	144	70.40	157	14 x 14	45 (total)
San Onofre 2 & 3	C-E	2-loop	3390	172	136	150	94.90	217	16 x 16	83 F/8 P
Seabrook 1	W	4-loop	3411	173	132.7	143.7	104.50	193	17 x 17	53 (total)
Sequoyah 1 & 2	W	4-loop	3411	173	132.7	143.7	103.50	193	17 x 17	53 F/8 P
Shearon Harris 1	W	3-loop	2785	157	119.5	144	105.00	157	17 x 17	52 (total)

Table 7-1. Comparison of PWR Vessel and Core Parameters (Continued)

Reactor Plant Name	PWR Vendor	PWR Type	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)
South Texas 1 & 2	W	4-loop	3817	173	132.7	168	105 (est.)	193	17 x 17	57 (total)
St. Lucie 1	C-E	2-loop	2560	172	136	136.7	83.20	217	14 x 14	73 F/8 P
St. Lucie 2	C-E	2-loop	2560	172	136	150	83.20	217	16 x 16	73 F/8 P
Summer	W	3-loop	2785	172	119.5	144	104.50	157	17 x 17	48 (total)
Surry 1 & 2	W	3-loop	2441	159	119.5	144	92.00	157	15 x 15	48 F/5 P
Three Mile Island 1	B&W	177 L-loop	2535	171	128.9	144	82.30	177	15 x 15	61 F/8 P
Trojan	W	4-loop	3411	173	132.7	143.7	105.50	193	17 x 17	53 F/8 P
Turkey Point 3 & 4	W	3-loop	2208	172	119.5	144	82.80	157	15 x 15	48 F/5 P
Vogtle 1 & 2	W	4-loop	3411	173	132.7	143.7	104.50	193	17 x 17	57 (total)
Waterford 3	C-E	2-loop	3390	172	136	150	94.90	217	16 x 16	83 F/8 P
Watts Bar 1 & 2	W	4-loop	3411	173	132.7	143.7	103.50	193	17 x 17	53 F/8 P
Wolf Creek	W	4-loop	3411	173	132.7	143.7	101.90	193	17 x 17	57 (total)
Yankee Rowe	W	4-loop	600	109	75.7	91.9	89.3 to 90.1	76	9 x (6 x 6)	24 (total)
Zion 1 & 2	W	4-loop	3250	173	132.7	144	100.00	193	15 x 15	57 (total)

Table 7-2. PWR System Comparison - RCS, AFW, Charging and HPSI

		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
ANO-1	B&W	2568	2	1/2	OT	2	M	780 @ 1127	3	Cent	400 @ 1600	270	Same as charging pumps				
						1	T	780 @ 1127									
ANO-2	C-E	2815	2	0/2	UT	1	M	575 @ 1214	3	PD	44	44	3	Cent	300 @ 1214	0	
						1	T	575 @ 1214									
Beaver Valley 1 & 2	W	2660	3	3/3	51	2	M	350 @ 1169	3	Cent	150 @ 2514	150	Same as cent. charging pumps				(a)
						1	T	700 @ 1169									
Bellefonte 1 & 2	B&W	3620	2	1/2	OT	unk.	unk.	unk.	1	Cent	306 @ 3500	unk.	unk.	unk.	780	unk.	
Braidwood 1 & 2	W	3411	4	2/3	51	1	M	890 @ 1452	2	Cent	150 @ 2526	150	2	Cent	400 @ 1106	0	(a)
						1	D	840 @ 1452	1	PD	98	98					
Byron 1 & 2	W	3411	4	2/3	51	1	M	890 @ 1452	2	Cent	150 @ 2526	150	2	Cent	400 @ 1106	0	(a)
						1	D	840 @ 1452	1	PD	98	98					
Callaway	W	3425	4	2/3	F	2	M	600 @ 1387	2	Cent	150	150	2	Cent	425 @ 1162	0	(a)
						1	T	1200 @ 1387	1	PD	98	98					
Calvert Cliffs 1 & 2	C-E	2700	2	2/2	UT	2	T	700 @ 1100	3	PD	44	44	3	Cent	345 @ 1084	0	
Catawba 1 & 2	W	3411	4	3/3	51	2	M	500 @ 1392	2	Cent	150 @ 2800	150	2	Cent	400 @ 1750	0	(a)
						1	T	100 @ 1395	1	PD	98	98					
Comanche Peak 1 & 2	W	3425	4	2/3	F	2	M	470 @ 1107	2	Cent	unk.	unk.	2	Cent	unk.	0	(a)
						1	T	900 @ 1107	1	PD							
Crystal River 3	B&W	2452	2	1/2	OT	1	M	740 @ unk	3	Cent	400 @ 1600	260	Same as charging pumps				
						1	T	740 @ unk									
D.C. Cook 1	W	3250	4	/3	51	1	M	450 @ 1177	2	Cent	150 @ 2800	150	2	Cent	400 @ 1700	0	(a) (b)
						1	T	900 @ 1177	1	PD	98	98					
D.C. Cook 2	W	3250	4	2/3	51F	1	M	450 @ 1177	2	Cent	150 @ 2800	150	2	Cent	400 @ 1700	0	(a) (b)
						1	T	900 @ 1177	1	PD	98	98					
Davis Besse	B&W	2772	2	1/2	OT	2	T	1050 @ 1050	2	Cent	150 @ 2514	150	2	Cent	500 @ 1170	0	
Diablo Canyon 1 & 2	W	3338	4	3/3	UT	2	M	440 @ 1300	2	Cent	150 @ 2514	[illegible]	2	Cent	425 @ 1084	0	
						1	T	880 @ unk.	1	PD	98 @ 2514						
Farley 1 & 2	W	2652	3	2/3	51	2	M	350 @ 1214	3	Cent	150 @ 2800	150	Same as cent. charging pumps				(a)
						1	M	700 @ 1214									
Fort Calhoun	C-E	1420	2	2/2	UT	1	M	260 @ 1040	3	PD	40	40	3	Cent	150 @ 1214	0	
						1	T	260 @ 1040									
Ginna	W	1520	2	2/2	44	2	M	200 @ 1344	3	PD	60	60	3	Cent	300 @ 1170	0	(a) (c)
						1	T	400 @ 1344									
						2	M	200 @ 1080									
Haddam Neck	W	1825	4	2/3	27	2	T	450 @ 1000	2	Cent	360 @ 2300	360	2	Cent	970 @ 1750	0	(a)
									1	PD	30	30					
Indian Point 2	W	2758	4	2/3	44F	2	M	400 @ 1350	3	PD	98	98	3	Cent	400 @ 1180		
						1	T	800 @ 1350									
Indian Point 3	W	3025	4	2/3	44F	2	M	400 @ 1350	3	PD	98	98	3	Cent	400 @ 1080	0	(a)
						1	T	800 @ 1350									

Table 7-2. 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
Kewaunee	W	1650	2	2/2	51	2	M	240 @ 1235	3	PD	60	60	2	Cent	700 @ 1082	0	(a)
						1	T	240 @ 1235									
Maine Yankee	C-E	2630	3	2/3	UT	2	M	500 @ 1100	3	Cent	150 @ 2850	150	Same as charging pumps				(f)
						1	T	500 @ 1100	1	PD	30	30					
McGuire 1 & 2	W	3425	4	3/3	51	2	M	450 @ 1655	2	Cent	150 @ 2514	150	2	Cent	400 @ 1106	0	(a)
						1	T	900 @ 1730	1	PD	55	55					
Millstone 2	C-E	2560	2	2/2	UT	2	M	300 @ 1056	3	PD	44	44	3	Cent	315 @ 1084	0	
						1	T	600 @ 1056									
Millstone 3	W	3411	4	2/3	51	2	M	575 @ 1290	3	Cent	150 @ 2800	150	2	Cent	425 @ 1500	0	(a)
						1	T	1150 @ 1290									
North Anna 1 & 2	W	2775	3	2/3	51F	2	M	350 @ 1214	3	Cent	150 @ 2500	150	Same as cent. charging pumps				(a)
						1	T	700 @ 1214									
Oconee 1, 2, & 3	B&W	2568	2	1/2	OT	1	M	500 @ 1050	3	Cent	450 @ 1700	270	Same as charging pumps				
						1	T	1080 @ 1050									
Palisades	C-E	2530	2	2/3	UT	1	M	415 @ 1183	3	PD	40	40	3	Cent	300 @ 1084	0	(g)
						1	T	415 @ 1183									
Palo Verde 1, 2 & 3	C-E	3800	2	0/4	UT	1	M	750 @ 1422	3	PD	44	N/A	2	Cent	814 @ 1233	0	
						1	T	750 @ 1422									
Point Beach 1 & 2	W	1518	2	2/2	44F	1	M	200 @ 1192	3	PD	60.5	60.5	2	Cent	700 @ 1750	0	(a) (b)
						1	T	400 @ 1192									
Prairie Island 1 & 2	W	1650	2	2/2	51	1	M	200 @ 1200	3	PD	60.5	60.5	2	Cent	700 @ 1082	0	(a) (d)
						1	T	200 @ 1200									
Rancho Seco	B&W	2772	2	1/2	OT	2	M-T	840 @ 1162	3	Cent	400 @ 1600	270	Same as charging pumps				(h)
							T	840 @ 1162									
Robinson	W	2200	3	2/3	44F	2	M	300 @ 1300	3	PD	77	77	3	Cent	375 @ 1750	0	(a)
						1	T	600 @ 1300									
Salem 1 & 2	W	3338	4	2/3	51	2	M	440 @ 1300	2	Cent	150 @ 2800	150	2	Cent	unk.	0	(a)
						1	T	880 @ 1550	1	PD	98	98					
San Onofre 1	W	1347	3	2/2	27	1	M	235 @ 1035	2	Cent	0 @ 2400	0	Same as cent. charging pumps				(a)
						1	T	300 @ 1110									
San Onofre 2 & 3	C-E	3390	2	0/2	UT	2	M	860 @ 1232	3	PD	44	44	3	Cent	415 @ 1227	0	
						1	T	860 @ 1232									
Seabrook	W	3411	4	2/3	F	1	M	710 @ 1322	2	Cent	150 @ 2800	150	2	Cent	425 @ 1750	0	(a)
						1	T	710 @ 1322	1	PD	98	98					
Sequoyah 1 & 2	W	3411	4	2/3	51	2	M	440 @ 1257	2	Cent	150 @ 2514	150	2	Cent	425 @ 1084	0	
						1	T	880 @ 1257	1	PD	55	55					
Shearon Harris 1	W	2785	3	3/3	UT	2	M	400 @ 1265	3	Cent	150 @ 2514	150	Same as cent. charging pumps				
						1	T	900 @ 1265									
South Texas 1 & 2	W	3817	4	2/3	F	3	M	540 @ 1435	2	Cent	160 @ 2513	160	3	Cent	800 @ 1235	0	
						1	T	540 @ 1435	1	PD	35	35					
St. Lucie 1 & 2	C-E	2560	2	2/3	UT	2	M	250 @ 1200	3	PD	44	44	3	Cent	345 @ 1084	0	
						1	T	500 @ 1200									

Table 7-2. 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
Summer	W	2785	3	unk.	UT	2	M	700 @ 1211	3	Cent	150 @ unk	150	Same as cent. charging pumps				
						1	T	1010 @ 1211									
Surry 1 & 2	W	2441	3	2/3	51F	2	M	350 @ 1183	3	Cent	150 @ 2485	150	Same as cent. charging pumps				(a)
						1	T	700 @ 1183									
TMI-1	B&W	2535	2	1/2	OT	1	M	460 @ unk	3	Cent	400 @ 1600	270	Same as charging pumps				
						1	T	920 @ unk									
Trojan	W	3411	4	2/3	51	1	T	960 @ 1474	2	Cent	150 @ 2800	150	2	Cent	425 @ 1700	0	(a)
						1	D	960 @ 1474	1	PD	98	98					
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	M	630 @ 1517	2	Cent	150 @ 2514	150	2	Cent	425 @ 1162	0	(a)
						1	T	1117 @ 1517	1	PD	98 @ unk.	98					
Waterford 3	C-E	3390	2	0/2	UT	2	M	350 @ 2673	3	PD	44 @ 2324	44	3	Cent	380 @ 1227	0	
						1	T	700 @ 2673									
Watts Bar 1 & 2	W	3411	4	2/3	51	2	M	470 @ 1600	2	Cent	150 @ 2514	150	2	Cent	unk.	0	(a)
						1	T	940 @ 1600	1	PD	98 @ 3200	98					
Wolf Creek	W	3411	4	2/3	F	2	M	600 @ 1387	2	Cent	150 @ 2514	150	2	Cent	425 @ 1161	0	(c)
						1	T	1200 @ 1387	1	PD	98 @ 2514	98					
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	M	450 @ 1343	2	Cent	150 @ 2800	150	2	Cent	400 @ 1084	0	(a)
						1	T	900 @ 1343	1	PD	98	98					

Table 7-2. 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 two 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.
(f) Maine Yankee has a variable capacity auxiliary charging pump (10 to 30 gpm) in addition to three main charging/HPI pumps.
(g) At Palisades, two charging pumps are fixed capacity and one is variable capacity. The turbine-driven pump at Palisades is not safety related.
(h) Rancho Seco has one AFW pump with a tandem steam turbine/electric motor drive.

Table 7-3. Comparison of 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)
ANO-1	B&W	1	Dresser 315 33VX	38.9	2255	2	234	2500
ANO-2	C-E	1	None	- - -	- - -	2	280.6	2500
Beaver Valley 1 & 2	W	3	Masoneilan 38-20771	79.9	2335	3	345	2485
Bellefonte 1 & 2	B&W	1	Unk.	160*	2295	2	500	2500
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
Calvert Cliffs 1 & 2	C-E	2	Dresser 315 33VX	56.7	2385	2	222.2	2485
Catawba 1 & 2	W	3	Unk.	210*	2485	3	420	Unk.
Comanche Peak 1 & 2	W	2	Unk.	210*	2185	3	420	2485
Crystal River 3	B&W	1	Dresser 315 33VX	40.8	2255	2	254	2500
D.C. Cook 1	W	3	Masoneilan 38-20721	64.6	2335	3	129.2	2485
D.C. Cook 2	W	2	38-20721	61.8	2335	3	123.5	2485
Davis Besse	B&W	1	Crosby HPN-SN	40.4	2255	2	242	2435
Diablo Canyon 1 & 2	W	3	Unk.	Unk.	Unk.	3	Unk.	Unk.

Table 7-3. Comparison of 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)
Farley 1 & 2	W	2	Copes-Vulcan D-100-160	79.2	2335	3	130.1	2485
Fort Calhoun	C-E	2	Dresser 315 33VS	69.7	2392	2	281.7	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
Maine Yankee	C-E	2	Dresser 315 33VS-30	57	2385	3	228.1	2485
McGuire 1 & 2	W	3	Unk.	210*	2335	3	420	2485
Millstone 2	C-E	2	Dresser 315 33VS-30	59.8	2380	2	231.3	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
Oconee 1, 2, & 3	B&W	1	Dresser 315 33VX	41.7	2255	2	254	2500
Palisades	C-E	2	Dresser 315 33VX	60.5	2385	3	272.7	2485
Palo Verde 1, 2 &3	C-E	0	None	- - -	- - -	4	Unk.	2485

Table 7-3. Comparison of 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)
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
Rancho Seco	B&W	1	Dresser 315 33VX	40.4	2255	2	249	2500
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
San Onofre 2 & 3	C-E	0	None	- - -	- - -	2	Unk.	2485
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
St. Lucie 1 & 2	C-E	2	Dresser 315 33VS-30	59.8	2385	3	234.4	2485
Summer	W	Unk.	Unk.	Unk.	Unk.	Unk.	Unk.	Unk.
Surry 1 & 2	W	2	Copes-Vulcan IA58RGP	86	2335	3	120.5	2360
TMI-1	B&W	1	Dresser 315 33VX	38.9	2255	2	243	2435

Table 7-3. Comparison of 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)
Trojan	W	2	Copes-Vulcan D-100-160	61.6	2350	3	123.1	2485
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
Waterford 3	C-E	0	None	- - -	- - -	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 7-4. Comparison of 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?
ANO-1	B&W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	2.09E+06	59	0.2	No
ANO-2	C-E	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.78E+06	54	0.1	No
Beaver Valley 1 & 2	W	Stone & Webster	Sub Atm.	Concrete Cylinder w/ Steel Liner	Reinforced	126	1.80E+06	54	0.1	No
Bellefonte 1 & 2	B&W	TVA	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	135	3.80E+06	50	0.2	Yes
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
Calliaway	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	140	2.50E+06	60	0.1	No
Calvert Cliffs 1 & 2	C-E	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.00E+06	50	0.33	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
Crystal River 3	B&W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.00E+06	55	0.25	No
D.C. Cook 1 & 2	W	AEP	Ice Cond.	Concrete Cylinder w/ Steel Liner	Reinforced	115	unk.	12	0.25	No
Davis Besse	B&W	Betchel	Dry	Steel Cylinder	- - -	130	2.87E+06	40	0.5	Yes
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
Fort Calhoun 1	C-E	Gibbs & Hill	Dry	Concrete Cylinder w/ Steel Liner	2-D Diag. Prestressed	110	1.05E+06	60	0.2	No
Ginna	W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	1-D Vert. Prestressed	105	9.97E+05	60	0.1	No

Table 7-4. Comparison of 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?
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
Maine Yankee	C-E	Stone & Webster	Dry	Concrete Cylinder w/ Steel Liner	Reinforced	135	1.86E+06	55	0.1	No
McGuire 1 & 2	W	Duke Power	Ice Cond.	Concrete Cylinder w/ Steel Liner	Reinforced	115	unk.	28	0.2	No
Millstone 2	C-E	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	1.92E+06	54	3	Yes
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	125	unk.	45	0.1	No
Oconee 1, 2, & 3	B&W	Duke / Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.91E+06	59	0.5	No
Palisades	C-E	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.60E+06	55	0.2	No
Palo Verde 1, [illegible] &3	C-E	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	146	2.60E+05	60	unk.	No
Point Beach 1 & 2	W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	105	unk.	60	0.4	No
Prairie Island 1 & 2	W	Pioneer	Dry	Steel Cylinder	- - -	105	unk.	41	0.5	Yes
Rancho Seco	B&W	Betchel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	1.98E+06	59	0.1	No
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

Table 7-4. Comparison of 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?
San Onofre 1	W	Bechtel	Dry	Steel Sphere	- - -	140	1.44E+06	47	0.5	Yes
San Onofre 2 & 3	C-E	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	150	2.34E+06	60	0.3	No
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
St. Lucie 1 & 2	C-E	Ebasco	Dry	Steel Cylinder	- - -	140	2.50E+06	44	unk.	Yes
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
TMI-1	B&W	Gilbert	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	130	2.00E+06	55	0.1	No
Trojan	W	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	124	2.00E+06	60	0.2	No
Turkey Point 3 & 4	W	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	116	1.55E+06	59	0.25	No
Vogtle 1 & 2	W	Bechtel	Dry	Concrete Cylinder w/ Steel Liner	3-D Prestressed	140	2.70E+06	52	0.1	No
Waterford 3	C-E	Ebasco	Dry	Steel Cylinder	- - -	140	2.68E+06	44	0.5	Yes
Watts Bar 1 & 2	W	TVA	Ice Cond.	Steel Cylinder	- - -	115	unk.	15	0.5	Yes
Wolf Creek	W	Bechtel	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 7-5. Comparison of 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
ANO-1	B&W	None	2	2750	Gen. Motors	2	125	
ANO-2	C-E	None	2	2850	Fairbanks-Morse	3	125	
Beaver Valley 1 & 2	W	None	2	2600	Gen. Motors	5 * 4 **	125	*Unit 1 **Unit 2 (2 are diesel batts.)
Bellefonte 1 & 2	B&W	None	2	7000	DeLavalle	4	120	*Per 6.9 kV diesel generator
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	
Calvert Cliffs 1 & 2	C-E	3 *		2500	Fairbanks-Morse	4	125	*One diesel is swing between units
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	
Crystal River 3	B&W	None	2	2750	Fairbanks-Morse	2	125/250	
D.C. Cook 1 & 2	W	None	2	3500* 3600**	Worthington Worthington	4	250	*Unit 1 **Unit 2
Davis-Besse	B&W	None	2	2600	Gen. Motors	4	125/250	
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	

Table 7-5. Comparison of 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
Fort Calhoun 1	C-E	None	2	2500	Gen. Motors	2	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	
Maine Yankee	C-E	None	1 2	250 2500	unk. Gen. Motors	4	125	
McGuire 1 & 2	W	None	2	4000	unk.	4	125	
Millstone 2	C-E	None	2	2750	Fairbanks-Morse	2	125	
Millstone 3	W	None	2	4986	Fairbanks-Morse	4	125	
North Anna 1 & 2	W	None	2	2750	Fairbanks-Morse	8	125	
Oconee 1, 2 & 3	B&W	2 *		87500*	unk.	4	125	* Keowee Dam hydroelectric generators
Palisades	C-E	None	2	2500	Alco	2	125	
Palo Verde 1, 2 & 3	C-E	None	2	5500	unk.	4	125	
Point Beach 1 & 2	W	2		2850	Gen. Motors	2	125	

Table 7-5. Comparison of 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	
Rancho Seco	B&W	None	2 2	3500 2750	unk. Gen. Motors	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	
San Onofre 2 & 3	C-E	None	2	6000	DeLavals	4	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.	
St. Lucie 1 & 2	C-E	None	2	3500	Gen. Motors	2	125	
Summer	W	None	2	4250	Fairbanks-Morse	unk.	unk.	
Surry 1 & 2	W	1	1	2850	Gen. Motors	4	125	
Three Mile Island 1	B&W	None	2	2600	Fairbanks-Morse	2	125/250	
Trojan	W	None	2	4418	Gen. Motors	2 1	125 250	

Table 7-5. Comparison of 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
Turkey Point 3 & 4	W	2		2500	Schoonmaker GM	4	125	
Vogtle 1 & 2	W	None	2	7000	unk.	4	125	
Waterford 3	C-E	None	2	4400	unk.	3	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 7-6. Comparison of 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)
ANO-1	B&W	Bechtel	Bechtel	836	15	Once Through	Water Discharge Tunnel	2	turbine	unk.	14,750 @ 824 psig
ANO-2	C-E	Bechtel	Bechtel	858	32	Closed Loop	Nat. Cooling Tower	2	turbine	1,170	14,500
Beaver Valley 1 & 2	W	Stone & Webster	Duquesne Light Co.	833	85	Closed Loop	Nat. Cooling Tower	2	AC	unk.	unk.
Bellefonte 1 & 2	B&W	TVA	TVA	1213	50	Closed Loop	Nat. Cooling Tower	2	turbine	1,161	16,595
Braidwood 1 & 2	W	Sargent & Lundy	Commonwealth Edison	1120	40	Once Through	[illegible]	2 (50%) 1 (50%)	turbine AC	unk. unk.	unk. unk.
Byron 1 & 2	W	Sargent & Lundy	Commonwealth Edison	1120	40	Closed Loop	Nat. Cooling [illegible]	2 (50%) 1 (50%)	turbine AC	unk. unk.	unk. unk.
Calloway	W	Bechtel	Daniel	1120	40	Closed Loop	Nat. Cooling Tower	2 (67%)	turbine	unk.	unk.
Calvert Cliffs 1 & 2	C-E	Bechtel	Bechtel	825	40	Once Through	Chesapeake Bay	2 (50%)	turbine	1,174	15,000
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 Reservoir	2 (50%)	turbine	986	19,900
Crystal River 3	B&W	Gilbert	Jones	821	15	Once Through	Gulf of Mexico	2	turbine	unk.	13,300 @988 psig
D.C. Cook 1 & 2	W	AEP	AEP	1060	85	Once Through	Lake Michigan	2 (70%)	turbine	1,138	16,750
Davis Besse	B&W	Bechtel	Bechtel	860	25	Closed Loop	Nat. Cooling Tower	2	turbine	1,110	15,000 @932 psig
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	1,476	15,000 @5250 RPM
Fort Calhoun 1	C-E	Gibbs & Hill	Gibbs & Hill	478	unk.	Once Through	Missouri River	3	AC	1,027	8,000 @754 psig
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	9,800
Indian Point 2	W	UE&C	Westinghouse	849	40	Once Through	Hudson River	2	turbine	unk.	1,530 @970 psig
Indian Point 3	W	UE&C	Westinghouse	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
Maine Yankee	C-E	Stone & Webster	Stone & Webster	810	50	Once Through	Montsweag Bay	2 (60%)	AC	1,543	14,000 @1031 psig

Table 7-6. Comparison of 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)
McGuire 1 & 2	W	Duke Power	Duke Power	1129	100	Once Through	Lake Norman	2	turbine	unk.	18,000
Millstone 2	C-E	Bechtel	Bechtel	857	43	Once Through	Niantic Bay	2 (55%)	turbine	910	15,000
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
Oconee 1, 2, & 3	B&W	Bechtel/ Duke Power	Duke Power	846	25	Once Through	Lake Keowee	2 (75%)	turbine	unk.	25,000 @1060 psig
Palisades	C-E	Bechtel	Bechtel	730	5	Closed Loop	Mech. Cooling Towers	2	turbine	1,144	13,500 @1144 psig
Palo Verde 1, 2 & 3	C-E	Bechtel	Bechtel	1333	55	Closed Loop	Mech. Cooling Towers	3 (65%)	turbine	unk.	unk.
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.
Rancho Seco	B&W	Bechtel	Bechtel	873	15	Closed Loop	Nat. Cooling Towers	2	turbine	1,387	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
San Onofre 2 & 3	C-E	Bechtel	Bechtel	1080	unk.	Once Through	Pacific Ocean	unk.	unk.	unk.	unk.
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.
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.
St. Lucie 1 & 2	C-E	Ebasco	Ebasco	839	5	Once Through	Atlantic Ocean	2	AC	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
TMI-1	B&W	Gilbert	UE&C	776	15	Closed Loop	Nat. Cooling Tower	2	turbine	unk.	unk.

Table 7-6. Comparison of 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)
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
Waterford 3	C-E	Ebasco	Ebasco	1075	63	Once Through	Mississippi River	2 (63%)	turbine	1,183	17,940 @ 932 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
[illegible]on 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.

8. BOILING WATER REACTOR (BWR) 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 BWR systems is provided, focusing on basic system functions, design, operation and interfaces. In Section 9, more detailed descriptive information is presented on the different BWR models currently operating in the U.S., and comparative data summaries for BWR systems are found in Section 10. 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.

8.1 Introduction to the Boiling Water Reactor

General Electric is the only supplier of Boiling Water Reactors (BWR) Nuclear Steam Supply Systems (NSSS) currently operating in the United States. These plants represent six General Electric "product lines", BWR/1 to BWR/6, all of which utilize a direct steam cycle, and in this respect, differ from the Pressurized Water Reactors (PWRs) which are supplied by other NSSS vendors. In the BWR direct cycle, steam is generated directly in the reactor core, passed through a steam separator and dryer assembly inside the reactor vessel, and then routed to the turbine generator to produce electricity. After passing through the turbine, the primary steam is condensed and returned to the reactor vessel by the condensate and feedwater systems.

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 BWR do not cause undue risk to the health and safety of the public. The spectrum of "generic" BWR systems is listed in Table 8.1-1. As indicated in this table, some systems are normally supplied by the Nuclear Steam 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.

8.2 BWR Primary System (the "Nuclear Boiler")

The "nuclear boiler" consists of the reactor vessel, the reactor core and internal structures, and two to five external recirculation loops containing the reactor coolant pumps that drive the flow in the reactor vessel. BWRs have been built in a variety of reactor vessel sizes ranging from an inside diameter of 106 inches to 251 inches. The vessel sizes applicable to the six BWR product lines are summarized in Table 8.2-1 and a more detailed listing is provided in Section 10.

All BWR plants except the earliest natural circulation BWR/1s have external recirculation pump loops to drive primary coolant flow through the reactor vessel. In BWR/2 plants, recirculation flow was driven entirely by motor-driven pumps in the external recirculation loops. As shown in Figure 8.2-1, the recirculation water inlets enter the bottom of the reactor vessel, below the level of the core.

Recirculation flow in BWR/3 and later plants is driven by the combined action of the motor-driven pumps in two external recirculation loops and jet pumps around the core barrel. The jet pumps provide approximately two-thirds of the recirculation flow rate. A representative jet pump BWR (i.e. BWR/3 to BWR/6) nuclear boiler is shown in Figures 8.2-2 and 8.2-3. Details of the jet pump installation are shown in Figures 8.2-4 and 8.2-5. Under LOCA conditions, the jet pump structures form standpipes that prevent breaks in the recirculation loops from uncovering the core (see Figure 8.2-6). High performance jet pumps were introduced in the BWR/6 product line, permitting a smaller annulus between the reactor vessel and the core shroud for recirculation flow.

Reactor power control is affected by the influence of variable recirculation flowrate on the rate of boiling and hence the void fraction in the core region. In BWR/2 to BWR/4 plants, recirculation flow is controlled by variable speed motor-generator sets which drive variable speed reactor recirculation pumps. This control arrangement is shown in Figure 8.2-7. The response speed of this means of recirculation flow control is limited by the inertia of the primary pumps and the variable-speed motor generator units that provide pump power. The recirculation system for BWR/5 and BWR/6 plants was simplified and made more responsive to fast load transients by replacing the variable speed pump feature with constant speed reactor recirculation pumps and recirculation flow control valves. This control arrangement is shown in Figure 8.2-8.

Among the operating U.S. BWR plants, only Big Rock Point has an external steam drum for steam separation. All BWR/2 to BWR/6 plants have steam separators and dryers that are internal to the reactor vessel, in an arrangement similar to that shown in Figures 8.2-1 and 8.2-2. Steam separators and dryers are illustrated in Figures 8.2-9 to 8.2-10.

Water level in the reactor vessel typically is measured by Yarway instruments that consist of a reference leg, a variable leg, and a differential pressure cell as shown in Figure 8.2-11. This instrument measures water level in the annulus around the reactor core. As shown in this figure, the reference leg and the variable leg are contained in a reflective metallic insulating shell and are connected to a differential pressure cell located outside of the drywell. Reactor steam condenses in the condensing chamber and excess condensate flows back to the reactor vessel through the variable leg. This flow path keeps the variable leg temperature near reactor coolant operating temperature, and the reference leg is heated due to its proximity to the variable leg. The result is partial temperature compensation for the reactor vessel water level measurement.

Examples of reactor vessel water level instrument ranges are shown in Figure 8.2-12. The 0" to +60" range is the "normal range", and the -155" to +60" range is the "wide range". The +200" to -100" instrument is principally of interest during accident conditions, and typically is referred to as the "fuel zone" range. The 0" to +400" instrument is principally used to monitor vessel flood-up prior to refueling. Note that all instruments do not have the same reference "0".

8.3 Reactor Core and Fuel Assemblies

A BWR generates heat in a low-enrichment, light-water cooled and moderated core. A comparison of basic BWR core parameters is presented in Table 8.3-1, and a more complete listing is included in Section 10.

All BWR fuel assemblies consist of a square array of fuel and burnable poison rods surrounded by a solid "fuel channel". The functions of the fuel channel are to position the rod bundle, provide guide surfaces for the control rods that are inserted between the fuel assemblies, and make possible flow orificing for the individual assemblies. The design of the fuel assembly with the fuel channel surrounding the array of fuel rods is important to the performance of the BWR Emergency Core Cooling Systems (ECCS). The channel limits coolant flow through the core to flow in the axial direction. By preventing radial flow, the channel helps prevent ECCS coolant from bypassing hot spots in the core.

Among the operating BWRs, Big Rock Point has a unique 12 x 12 fuel assembly design. BWR/2 to BWR/5 plants were introduced with 7 x 7 fuel assemblies, and the 8 x 8 fuel assemblies were introduced in the BWR/6. A typical BWR fuel assembly for a BWR/2 to BWR/6 plant is shown in Figure 8.3-1. BWR fuel assemblies for these plants have common external characteristics, but differ in the array of fuel rods that fit within the fuel channel. Three basic variations currently are in use; the 7 x 7, 8 x 8, and 9 x 9 fuel assemblies. The 7 x 7 assemblies have a higher liner heat generation rate and are more limited by fuel pellet-clad interactions (PCI) than the other two assembly designs. Use of the 7 x 7 fuel assemblies is being phased out. A brief description of each type of fuel assembly is provided below.

8.3.1 12 x 12 (Big Rock Point) Fuel Assembly

The 12 x 12 fuel assembly is a square array of 144 rods surrounded by a fuel channel as shown in Figure 8.3-1. As in all BWR/2 to BWR/6 plants to follow, four fuel assemblies are arranged around a single cruciform control rod. In the 12 x 12 fuel assembly, 132 of the rods are fuel rods and the remaining 12 are "special" rods used to minimize the local heat generation hot spot factor. The physical arrangement of this fuel assembly with the fuel channel removed is shown in Figure 8.3-2.

8.3.2 7 x 7 Fuel Assembly

The 7 x 7 fuel assembly was the original standard fuel assembly for BWR/2 to BWR/5 plants. The basic geometry of this fuel assembly is shown in Figure 8.3-3. All 49 of the rods were fuel rods, with an average enrichment of 2.12% U-235. Four fuel assemblies are arranged around a single cruciform control rod. Plants with these fuel assemblies operate at power densities from 36 kW/liter (BWR/2s) to 51 kW/liter (BWR/4s and BWR/5s). Linear heat generation rate is about 18.5 kW/ft. Many of the plants that originally used the 7 x 7 fuel assemblies have been retrofit with 8 x 8 and/or 9 x 9 fuel assemblies.

8.3.3 8 x 8 Fuel Assembly

The 8 x 8 fuel assembly was initially introduced in 1972 in the BWR/6 product line. It offers operating and safety advantages over the earlier 7 x 7 array, and has been backfit into many earlier BWR product lines (BWR/2 through BWR/5 series). The basic arrangement of four fuel assemblies surrounding a cruciform control rod remains unchanged, and the 8 x 8 assembly fits within the same dimensional envelope as the 7 x 7 assembly. As shown in Figure 8.3-4, the 8 x 8 assembly design consists of 63 fuel rods and one water rod. Average enrichment of a core with 8 x 8 fuel assemblies is in the range from 2.43 to 2.83% U-235, somewhat higher than for a core with 7 x 7 assemblies. The 8 x 8 assembly incorporate thicker fuel channels and thinner control blades than the 7 x 7 design.

The change to 8 x 8 assemblies provides multiple benefits. First, the 64-rod bundle provides a 13 percent increase in heat transfer surface over the 7 x 7 assembly.

With the 8 x 8 assemblies, core power density has been increased to 56 kW/liter from the 51 kW/liter available from the 7 x 7 assembly. Although core power density has increased with the introduction of the 8 x 8 assembly, the individual duty on the fuel rods was significantly decreased from a peak linear heat generation rate of 18.5 kW/ft to 13.4 kW/ft. The lower peak linear heat generation rate provides significantly lower fuel operating temperature and reduces the rate and extent of fuel pellet expansion, thereby reducing operational limitations from PCI and permitting longer fuel life and an increase in the energy output from the fuel before the useful life of the cladding is reached. Average fuel burnup is on the order of 30 GWd/tU.

The reduced linear heat generation rate results in lower clad stresses during power transients, thus enabling faster load following and scram recovery. In addition, clad temperatures are significantly lower during a postulated design basis LOCA. As a result, it is claimed that significant metal-water reactions are precluded and post-LOCA hydrogen generation is reduced in comparison to the 7 x 7 assemblies. This is an advantage that reduces the demands on containment combustible gas control systems and the need for containment inerting.

8.3.4 9 x 9 Fuel Assembly

The 9 x 9 fuel assembly is designed to be dimensionally and hydraulically compatible with 8 x 8 assemblies. The 9 x 9 fuel assembly offers a lower linear heat generation rate than 8 x 8 assemblies, and thereby further reduces the operational limitations associated with pellet-clad interaction (PCI). Average enrichment for the 9 x 9 fuel assemblies designed by Exxon Nuclear is 3.15% U-235. It is estimated that the 9 x 9 assembly design allows an average fuel burnup on the order of 36 GWd/tU to be achieved (about 20 percent higher than for 8 x 8 fuel assemblies).

8.4 Reactivity Control Systems

BWR reactivity control is by three independent systems; (a) the Reactor Recirculation Flow Control System, (b) the Control Rod Drive Hydraulic System (CRDHS), and (c) the Standby Liquid Control System (SLCS). The Reactor Recirculation Flow Control System operates in conjunction with the main turbine pressure regulator controls to provide automatic load following. Reactor power control is effected by the influence of variable recirculation flowrate on the rate of boiling and hence the void fraction in the core region. The negative moderator void coefficient that is characteristic of the BWR tends to reduce power as the rate of boiling increases. In early BWR plants (i.e., BWR/2 to BWR/4), recirculation flow is controlled by variable speed motor-generator sets which drive variable speed reactor recirculation pumps. In BWR/5 and BWR/6 plants, this control function is accomplished with control valves in the recirculation loops. In some BWRs, the recirculation pumps are tripped in response to an Anticipated Transient Without Scram (ATWS) in order to cause increased boiling in the core and a rapid insertion of negative reactivity due to the higher void fraction in the core.

The CRDHS provides control for both long- and short-term reactivity changes (e.g., startup, shutdown, operational transients, fuel burnup, and xenon effects) and is used for rapid shutdown (e.g., reactor trip or scram). All BWRs have bottom-entry control rods that are individually controlled by hydraulic control units located outside of the drywell. A typical BWR control rod is shown in Figure 8.4-1 and the rod drive is shown in Figure 8.4-2. A hydraulic control unit (HCU) is illustrated in Figure 8.4-3 and associated controls for a typical BWR are shown in Figure 8.4-4. Each control rod is driven by a double-acting hydraulic piston, and normal rod control requires coordinated action of the directional control valves (see Figures 8.4-2 and 8.4-4) to properly align high-pressure hydraulic fluid (water) to one side of the piston and open an exhaust path from the other side of the piston.

Reactor scram is accomplished by opening the scram inlet and outlet valves in all hydraulic control units (see Figure 8.4-4). The scram inlet valves align the high-

pressure scram accumulators to the "insert" side of the control rod pistons and the scram outlet valves align the opposite side of the control rod pistons to the low-pressure scram discharge volume. The result is rapid insertion of all control rods into the reactor core. Scram is implemented by deenergizing both scram pilot valves in each hydraulic control unit. Alternatively, scram can be implemented by energizing either of the two backup scram pilot valves that are in the air supply path to all hydraulic control units.

In some BWRs, an additional set of pilot valves has been installed in the scram valve air supply. The function of these valves is comparable to the backup scram pilot valves, however, they are actuated in response to an ATWS by the Alternate Rod Insertion (ARI) system which has an actuation logic that is independent and diverse from the RPS. The ARI system has been added to improve the reliability of the scram system and decrease the likelihood of an ATWS.

The Standby Liquid Control System (SLCS), consists of redundant trains of high-pressure, low-capacity pumps that can inject a concentrated boron solution into the reactor vessel below the core. Although not intended to provide rapid reactor shutdown, the SLCS provides a redundant, independent, and diverse means to render the reactor subcritical and maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner.

8.5 Heat Transfer Systems for Power Operation

When the reactor is operating at power, the normal heat removal path is by means of two fluid system loops as illustrated in Figure 8.5-1. The primary heat transfer loop consists of the RCS and the Steam and Power Conversion System. This is a closed, two-phase loop in which steam is generated in the reactor core and supplied at about 1000 psig to the turbine-generator which extracts power from the steam to generate electricity. The main turbine exhausts to the main condenser which operates under partial vacuum conditions (20 inches mercury vacuum). Heat is transferred in the main condenser to a secondary circulating water cooling loop, and the condensed steam is returned to the reactor vessel by the main condensate and feedwater systems which together increase the coolant pressure back up to 1000 psig.

The secondary cooling loop in a BWR 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.

8.6 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 reactor vessel. 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 Figure 8.5-1) 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, high-pressure shutdown cooling is provided by isolation condensers (early BWRs) or by the Reactor

Core Isolation Cooling (RCIC) System (later BWRs). No BWRs have both isolation condensers and an RCIC system.

An isolation condenser is a simple condensing heat exchanger that, on the primary side, receives steam from the reactor vessel and returns the condensate to the reactor vessel via a closed-loop, single-phase, natural circulation heat transport loop as shown in Figure 8.6-1. The secondary water is boiled in the isolation condenser and the secondary steam is vented to the atmosphere to complete the heat transfer path to the ultimate heat sink. The secondary-side water volume is sufficient for 30 to 60 minutes of cooling. Makeup of secondary water is provided from the fire water system or other suitable water source. No heat is dumped to the suppression pool during isolation condenser operation.

The RCIC system has a single steam turbine-driven pump that is supplied with steam from one of the main steam lines and exhausts to the suppression pool. The system initially operates on an open cycle, taking water from the condensate storage tank (CST), as illustrated in Figure 8.6-2, and injecting into the reactor vessel via a main feedwater line (BWR/3 and BWR/4 plants) or through a reactor vessel head spray nozzle (BWR/5 and BWR/6 plants). When the CST is empty, the RCIC pump suction is aligned to the suppression pool and the suction line from the CST is isolated. Heat from the primary system is transferred to the suppression pool by the continuous or intermittent operation of the primary safety/relief valves. Suppression pool cooling must be initiated to complete the heat transfer path to the ultimate heat sink. RCIC operation must be terminated at primary steam pressures less than about 50 psig.

Some later BWRs (some BWR/4s, all BWR/5s and BWR/6s) have the capability of operating the Residual Heat Removal (RHR) heat exchangers as steam condensers and using the RCIC pump to return the condensate to the reactor vessel. In contrast to the simple RCIC system described previously, this operating mode enables a closed-loop heat transfer path to be established rapidly, thereby minimizing the amount of heat dumped to the suppression pool via the primary safety/relief valves. Heat removed by the RHR heat exchangers is transferred to the ultimate heat sink by the RHR (High-Pressure) Service Water System as illustrated in Figure 8.6-3. The RCIC turbine still exhausts to the suppression pool.

8.7 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 125 psig by Steam and Power Conversion System, the isolation condensers, or the RCIC system. Early BWRs have dedicated RHR systems that are separate from the low-pressure ECCS subsystems. Later BWRs have multi-mode RHR systems that perform the shutdown cooling function as well as a variety of ECCS and containment cooling functions.

The RHR shutdown cooling suction line is opened to align the suction of the RHR pumps to a reactor recirculation loop, on the suction side of an idle recirculation pump. Flow is established through the RHR heat exchangers and the primary coolant is then returned to the reactor vessel via a recirculation line (on the discharge of an idle recirculation pump) or a main feedwater line (later model BWRs only). The RHR heat exchangers transfer heat to the RHR (High-Pressure) Service Water System as illustrated in Figure 8.7-1 (for a multi-mode RHR system).

The RHR (High-Pressure) Service Water System is a single phase, moderate-pressure system that is dedicated to providing cooling water for the RHR heat exchangers. In later BWRs (BWR/5s and BWR/6s) RHR cooling is supplied by an Essential Service Water System that also provides cooling for other safety-related components. In either case, the service water systems 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).

Due to the relatively high discharge pressure of the RHR (High-Pressure) Service Water pumps (about 300 psid), the service water system can be used in an emergency to provide BWR core or containment flooding. This capability is implemented by opening the cross-tie between the service water system and the RHR return line to the RCS. In a multi-mode RHR system, this return line branches to the reactor vessel, the suppression pool, and the drywell.

8.8 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

BWR plants provide for RCS overpressure protection, pressure relief, and RCS depressurization in various ways. In the earliest plants (BWR/1 and some BWR/2 plants) overpressure protection was provided by safety valves (see Figure 8.8-1) that mechanically open on high pressure in the RCS. In these plants, the ADS function is accomplished by "Electromatic" type power-actuated relief valves (PARVs) that are controlled by a solenoid pilot valve (see Figure 8.8-2).

In the BWR/3 product line, the trend was away from PARVs to safety/relief valves that were capable of both lifting mechanically on high pressure and being actuated at lower pressures to perform either the pressure relief or the ADS function. In the BWR/4 product line, the trend was away from safety valves to safety/relief valves for accomplishing the overpressure protection function. Later BWR/4, and all BWR/5 and BWR/6 plants only use safety/relief valves to accomplish the overpressure protection, pressure relief, and ADS functions. The Target-Rock three-stage safety/relief valve shown in Figure 8.8-3 is representative of the safety/relief valves used in BWR/3 to BWR/5 plants. In the BWR/6 product line, the Crosby safety/relief valve was introduced. This valve, shown in Figure 8.8-4, is actuated by an external pneumatic piston.

8.9 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) injects makeup water into the reactor vessel 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. The emergency core cooling function is accomplished by an integrated network of systems that provide core cooling and coolant inventory control following transients and LOCAs.

8.9.1 High-Pressure ECCS Subsystems

The BWR ECCS typically includes a high-pressure injection or core spray subsystem, one or more low-pressure systems for coolant injection and/or core spray, and an automatic depressurization system (ADS). The high-pressure injection or core spray subsystem is intended to respond to small LOCAs, while the low-pressure subsystems are intended to respond to large LOCAs which rapidly depressurize the RCS. The ADS is used to depressurize the RCS when the high-pressure ECCS subsystem fails to perform adequately, thereby allowing the low-pressure ECCS subsystems to provide core cooling.

In the earliest BWRs, the high-pressure ECCS subsystem is the Feedwater Coolant Injection (FWCI) System. This subsystem uses the main condensate and feedwater systems described in Section 8.5 to provide RCS makeup from the main condenser hotwell and/or other water sources. Heat is dumped from the RCS to the suppression pool via the primary safety valves.

For BWR/3 and BWR/4 plants, the high-pressure ECCS subsystem is the High-Pressure Coolant Injection (HPCI) System which has a single steam turbine-driven pump that is supplied with steam from one of the main steam lines and exhausts to the suppression pool. The system initially operates on an open cycle, taking water from the condensate storage tank (CST), as illustrated in Figure 8.9-1, and injecting into the reactor vessel via a main feedwater line. When the CST is empty, the HPCI pump suction is

aligned to the suppression pool and the suction line from the CST is isolated. Heat from the primary system is transferred to the suppression pool by the continuous or intermittent operation of the primary safety/relief valves. Suppression pool cooling must be initiated to complete the heat transfer path to the ultimate heat sink. The HPCI system provides primary system makeup at RCS pressures from 1150 to 150 psig. Operation must be terminated at primary steam pressures less than about 150 psig due to poor steam conditions for the two-stage HPCI turbine drive.

For BWR/5 and BWR/6 plants, the high-pressure ECCS subsystem is the High-Pressure Core Spray (HPCS) System which has a single motor-driven pump that initially operates on an open cycle, taking water from the condensate storage tank (CST), as illustrated in Figure 8.9-2, and injecting into the reactor vessel via a spray sparger located above the reactor core. When the CST is empty, the HPCS pump suction is aligned to the suppression pool and the suction line from the CST is isolated. Heat from the primary system is transferred to the suppression pool by the continuous or intermittent operation of the primary safety/relief valves. Suppression pool cooling must be initiated to complete the heat transfer path to the ultimate heat sink. The HPCS pump can operate over the full range of RCS pressure.

8.9 Low-Pressure ECCS Subsystems

All BWR plants have low-pressure makeup systems that are intended to respond to large LOCAs and rapidly pressurize the RCS. All BWRs except Dresden 1 also have the capability to depressurize the RCS by opening primary relief valves which discharge to the suppression pool. This depressurization capability is used to enable low pressure makeup systems to provide for core cooling when the high-pressure heat removal and injection systems have failed. All BWR/1 to BWR/6 plants have a low pressure core spray system that injects via spargers located above the reactor Core. BWR/3 and later plants also have a Low-Pressure Coolant Injection (LPCI) system that provides a core flooding capability.

All BWRs have a low-pressure Core Spray System that uses motor-driven pumps to deliver water from the suppression pool to the reactor vessel via one or more spray spargers located above the reactor core, as shown in Figure 8.9-3. In BWR/2 to BWR/4 plants, the Core Spray System consists of two 100 percent capacity trains with two separate spargers as shown in Figure 8.9-4. In BWR/5 and BWR/6 plants there is a single-train Low-Pressure Core Spray (LPCS) system that is supplemented by the single-train HPCS system to provide a redundant low pressure core spray capability. In these plants, the LPCS and HPCS systems inject into independent spargers.

BWR/1 and BWR/2 plants do not have a Low-Pressure Coolant Injection (LPCI) system. This ECCS subsystem was introduced with the BWR/3 product line as a dedicated injection system (i.e., not integrated into a multi-mode RHR system) that delivered water from the suppression pool to the reactor vessel via a recirculation loop. In later BWR/3 plants and all BWR/4 to BWR/6 plants, the LPCI system is an operating mode of the RHR system shown in Figure 8.7-1. Earlier LPCI systems inject via a recirculation loop, while LPCI systems in BWR/5 and BWR/6 plants inject directly into the reactor vessel.

8.10 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. All operating U.S. BWR plants, except the Big Rock Point BWR/1 plant, have a pressure suppression primary containment consisting of a "drywell" containing the reactor vessel and primary system components, and a suppression pool or "wetwell" containing a large volume of water to condense steam discharged from the primary system and to serve as a short-term heat sink. A variety of pressure suppression primary containment designs are used in BWR plants (pre-Mark I, Mark I, II, and III) however all are functionally similar and all have a

secondary containment. No operating U.S. BWR plants use the pre-Mark I pressure suppression containment. With the exception of Big Rock Point, all U.S. BWRs have a secondary containment which is formed by parts of the reactor building surrounding the primary containment structure. The distributions of BWRs containment type and construction are shown in Figures 8.10-1 and 8.10-2, respectively. A general comparison of the Mark I, II, and III pressure suppression containments is presented in Figure 8.10-3 and in Table 8.10-1. More detailed comparative data are included in Section 10.

During normal operation, the BWR primary containment is closed, with cooling provided by a recirculating ventilation system. The plant areas comprising the secondary containment are continuously ventilated and are exhausted through an elevated stack. Following an accident, the containment isolation system causes isolation valves to close in certain lines that penetrate the primary containment boundary. The secondary containment exhaust is directed to the Standby Gas Treatment System for processing prior to release through the elevated stack. Containment heat removal is accomplished by the dedicated suppression pool cooling system (BWR/2) or by a multi-mode RHR system operating in the suppression pool cooling mode (BWR/3 to BWR/6 plants). Post-LOCA combustible gas concentration in the primary containment is controlled by inerting in the Mark I containment and by hydrogen recombiners and igniters in the Mark II and III containments.

8.10.1 Large, Dry Containment

The Big Rock Point BWR/1 plant is housed in a large, dry, spherical, steel containment as shown in Figure 8.10-4. The Dresden 1 BWR/1 had a similar containment. These containments have a sump for collection of primary coolant lost during a LOCA. The core and containment spray systems are aligned to take a suction on the sump during the recirculation phase of LOCA response. This type of containment is functionally similar to large, dry, steel containments used in early PWR plants.

8.10.2 Pre-Mark I Pressure Suppression Containment

The Humboldt Bay BWR/1 plant was the first U.S. BWR with a pressure suppression containment (a pre-Mark I containment). The pre-Mark I pressure suppression containment consisted of two chambers; the cylindrical, reinforced concrete drywell containing the reactor vessel and primary system components, and the annular, cylindrical suppression chamber (wetwell) which contains a large quantity of water used for pressure suppression and as a heat sink. As shown in Figure 8.10-5, the drywell vents into the suppression chamber by means of large diameter horizontal vent headers connected to vertical standpipes that discharge steam below the surface of the suppression pool.

8.10.3 Mark I Pressure Suppression Containment

The Mark I containment shown in Figure 8.10-6 is the predominant type of BWR containment. The Mark I containment is divided into two chambers: (a) the drywell which houses the reactor vessel and primary system components, and (b) a torus-shaped suppression chamber (wetwell) which contains a large quantity of water used for pressure suppression and as a heat sink. All Mark I primary containments are of free-standing steel construction as shown in Figure 8.10-1 except for the Brunswick 1 and 2 plants. The Brunswick Mark I containment is a steel-lined reinforced concrete structure. Details of the reactor drywell are shown in Figures 8.10-8.

If a pipe rupture occurs inside containment, the drywell becomes pressurized by steam blowing down from the primary coolant system. The steam enters the downcomers and is routed to the torus where it is discharged beneath the water surface of the pressure suppression pool. The wetwell is designed to condense the steam discharged to the torus and thus mitigate the post-accident containment pressure transient. The wetwell water also provides a source of water for the ECCS systems. The design pressure for a Mark I containment typically is in the range from 56 to 62 psig. In the Oyster Creek and Nine Mile

Point 1 plants, the drywell design pressure is 62 psig, while the torus design pressure is 35 psig. In all other Mark I plants, the drywell and torus have the same design pressure. In order to improve the ability of Mark I plants to cope with severe accidents, the NRC has required the installation of "hard" vents (i.e. pressure relief devices with relatively thick-walled discharge piping rather than thin-walled ducting) to protect against overpressurization of the primary containment. This vent line is routed via an air cleanup system to the plant stack.

The Mark I containments have the smallest free air volume of the three BWR pressure suppression containment designs, particularly in the drywell. They have approximately one-sixth the free air volume of a large dry PWR containment. The small containment volume and the original use of 7 x 7 fuel assemblies in plants with Mark I containments results in a high post-LOCA containment pressure and hydrogen concentration. BWR plants with Mark I containment operate with a nitrogen inerted drywell atmosphere to prevent a potentially flammable post-LOCA containment environment. With the introduction of the 8 x 8 and 9 x 9 fuel assembly designs, BWR utility owners have been able to demonstrate reduced post-LOCA hydrogen generation. The drywell inerting system operates in three modes: (a) high flow rate during inerting prior to reactor startup, (b) low flow rate to maintain drywell oxygen concentration less than 4 percent during normal operation, and (c) off during refueling or maintenance in the drywell. In the high flow rate mode, nitrogen is supplied at up to 200,000 scfh to establish inert atmosphere conditions in the drywell, while about 5,000 scf/day is supplied in the low flow rate mode to maintain inert conditions. The drywell purge system reestablishes normal drywell atmosphere conditions prior to refueling or as required prior to inspection and maintenance in the drywell.

The Mark I containment was found to be difficult to construct. In addition, its small size resulted in congested spaces that left little room for auxiliary systems or for maintenance. Backfitting requirements such as improved pipe restraints and snubbers further complicated component accessibility and maintainability within a Mark I containment.

The secondary containment areas of the Mark I containment include the ECCS and RHR pump rooms, reactor water cleanup system equipment rooms, steam tunnel, and the fuel handling and storage area on the upper level of the reactor building. The Standby Gas Treatment System (SGTS) ventilates the secondary containment areas and exhausts to an elevated release point. A representative SGTS is shown in Figure 8.10-9.

8.10.4 Mark II Pressure Suppression Containment

Mark II containments are used for some BWR/4 plants (Limerick 1 and 2, Shoreham, and Susquehanna 1 and 2) and for all BWR/5 plants (LaSalle 1 and 2, Nine Mile Point 2, and WNP-2). This containment design retained the basic pressure suppression function of the Mark I containment, but rearranged the drywell and the suppression chamber into an "over/under" configuration as illustrated in Figure 8.10-10. The suppression chamber is below the drywell with vertical downcomers connecting the two volumes. Steam released into the drywell during a LOCA is directed into the downcomers and is discharged below the water level in the suppression chamber. This simpler vent configuration results in lower head loss in the downcomers, thereby contributing to a lower containment design pressure than in the Mark I design (i.e. typically 45 psig vs. 62 psig).

The Mark II primary containment design has been implemented using the following three different construction techniques:

- Steel, free-standing	WNP-2
- Reinforced concrete with steel liner	Nine Mile Point 2, Shoreham, Limerick 1 and 2, Susquehanna 1 and 2
- Post-tensioned concrete with steel liner	LaSalle 1 and 2

The relatively simpler shape of the Mark II containment was expected to be easier to construct than a Mark I containment, however, the conical shape of the upper portion of the Mark II drywell was still difficult to construct. Better space utilization in the Mark II plants resulted in a smaller reactor building than in Mark I plants. The shape and greater volume of the drywell allowed better access to main steam, feedwater, recirculation and ECCS components in the drywell. The use of the motor-driven HPCS eliminated the HPCI steam lines and the below-grade HPCI pump room that was an appendage off the side of the Mark I reactor building.

The Mark II drywell is inerted, and the inerting and purge systems are comparable to those found in Mark I containments.

The secondary containment areas of the Mark II containment include the ECCS and RHR pump rooms, reactor water cleanup system equipment rooms, steam tunnel, and the fuel handling and storage area on the upper level of the reactor building. The Standby Gas Treatment System (SGTS) ventilates the secondary containment areas and exhausts to an elevated release point.

8.10.5 Mark III Pressure Suppression Containment

The Mark III containment consists of a drywell and pressure suppression chamber inside a primary containment shell that is surrounded by an enclosure building and various equipment rooms that function as part of the secondary containment boundary. The basic arrangement of a Mark III containment is shown in Figure 8.10-11. The primary and secondary containment boundaries are identified in Figure 8.10-12.

The Mark III primary containment design has been implemented using the following two different construction techniques:

- Steel , free-standing	Perry 1 and 2, River Bend 1
- Reinforced concrete with steel liner	Clinton 1, Grand Gulf 1

In all cases, the drywell is a reinforced concrete structure. The lower portion of the River Bend steel primary containment (i.e. the suppression pool) is reinforced with stiffener rings. In the Perry Mark III containment, the steel primary containment is backed by concrete in the suppression pool area.

In comparison to the Mark II containment, the Mark III containment is a simpler structural design with the reactor placed lower in the containment, resulting in improved seismic response and a shorter construction schedule. The downcomers in the Mark II containment have been replaced by a weir wall with horizontal vents to separate the drywell from the suppression pool. The volume and arrangement of the Mark III containment provides lower containment design pressure and improved accessibility of the nuclear boiler for in-service inspection.

The Mark III containment does not require inerting. The lower peak clad temperatures predicted for the BWR/6 with 8 x 8 fuel elements limits the combustible gas generation. Flammability control is achieved through direct control of hydrogen concentration (e.g., hydrogen recombiners).

The fuel pool has been moved out of the reactor building and placed in a separate fuel handling building. The fuel pool was on the fuel operating floor level in Mark I and II reactor buildings.

8.11 Component Cooling Systems

The function of the PWR Component Cooling Water System (CCWS) described in Section 3 typically is performed in a BWR by the Reactor Building Closed Cooling Water (RBCCW) System and the Turbine Building Closed Cooling Water (TBCCW) System. The RBCCW system serves safety and non-safety heat loads, while the TBCCW serves only non-safety heat loads. These are closed-loop, low-temperature, low-pressure, single-phase cooling system that provides cooling for a wide range of components.

As illustrated in Figure 8.11-1, the RBCCW system 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 used 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 RBCCW cooling for a variety of systems including the reactor water cleanup system and 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 RBCCW 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 RBCCW and the area(s) being cooled.

The RBCCW system rejects heat to the ultimate heat sink through the cooling loop indicated as the Service Water System in Figure 8.11-1. This service water system may operate on an open, closed, or combined cycle as described previously.

8.12 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 the Reactor Protection System (RPS) and a set of actuation systems that, as a group, perform the function of the PWR Engineered Safety Feature Actuation System (ESFAS) described in Section 3. As described previously, the function of the RPS is to initiate a reactor scram when needed. The actuation logic for the various BWR safety systems provides for automatic actuation based on the detection of abnormal conditions in the reactor plant. As appropriate, the actuation systems can actuate systems necessary for adequate RCS coolant inventory control and/or core cooling, containment isolation and cooling, radioactive release control, emergency power, and component cooling.

As illustrated in Figure 8.12-1, the actuation systems include provisions for manual actuation at the system level (typically from the control room) or at the actuation-train level (typically from the output logic cabinets). A manual trip from the control room actuates all components that would be actuated by an automatic actuation signal. A manual

trip from the output logic cabinets actuates only the components that are controlled by the respective actuation train.

The relationship between the automatic actuation systems and other means of actuation is also shown in Figure 8.12-1. Individual remote-manual component controls, which do not use any part of the automatic actuation logic, are generally 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.

8.13 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. In BWR/2 to BWR/4 plants, there typically are two AC power divisions, A and B, while in BWR/5 and BWR/6 plants there are three AC power divisions, A, B, and C. The typical distribution of ECCS loads among the AC divisions is listed in Table 8.13-1.

Most DC loads are supplied from 125 VDC panels, although many BWR plants have 250 VDC distribution systems to support DC-powered motor-operated valves or other relatively large DC-powered components (HPCI and RCIC steam line isolation valves, HPCI lube oil pump). As summarized in Table 8.13-2, BWR/2 to BWR/4 plants typically have two 125/250 VDC divisions, while BWR/5 and BWR/6 plants have three divisions. 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.

Representative AC and DC power distribution systems for a BWR/4 plant (Cooper) are shown in Figures 8.13-1 and 8.13-2. Similar diagrams for a BWR/6 plant (River Bend) are shown in Figures 8.13-3 and 8.13-4. The 250 VDC batteries and buses in Figure 8.13-2 are needed to support the RCIC and HPCI systems in the BWR/4 plant. The third AC division in Figure 8.13-3 is needed to support the motor-driven HPCS pump in BWR/5 and BWR/6 plants.

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 8.13-5 and 8.13-6. As shown in Figure 8.13-6, 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 8.1-1. Summary of BWR Systems

BWR System	NSSS Scope	BOP Scope
Reactor	X	
Reactivity Control System	X	
Reactor Recirculation System	X	
Residual Heat Removal System	X	
Reactor Water Cleanup System	X	
Nuclear Leak Detection System	X	
Containment		X
Reactor Trip System	X	
Standby Liquid Control System	X	
Control Rod Drive Hydraulic System	X	
Reactor Core Isolation Cooling System	X	
Emergency Core Cooling System	X	
Standby Gas Treatment System		X
Onsite Electric Power System		X
Offsite Electric Power System		X
Main Steam System		X
Turbine Generator and Auxiliaries		X
Main Condenser & Evacuation System		X
Condensate & Feedwater System		X
Condensate Cleanup System		X
Circulating Water System		X
New & Spent Fuel Storage	X	
Fuel Handling System	X	
Habitability System		X
Engineered Safety Feature Actuation Sys.	X	X
Safety Related Display Instrumentation	X	X
Service Water System		X
Closed Cooling Water System		X
Ultimate Heat Sink		X
Compressed Air System		X
Process Sampling System		X
HVAC System		X
Fire Protection System		X
Diesel Generator & Auxiliary System		X
Radioactive Gaseous Waste System	X	
Radioactive Liquid Waste System		X
Radioactive Solid Waste System		X
Radiation Protection System		X

Table 8.2-1. Summary of BWR Vessel Sizes

BWR Product Line	Reactor Vessel Inside Diameter (inches)	Applicable Plants
BWR/1	106	Big Rock Point
	121	Humboldt Bay
	146	Dresden 1
BWR/2	213	Nine Mile Point 1, Oyster Creek
BWR/3	206	Monticello
	224	Millstone 1, Pilgrim 1
	251	Dresden 2 & 3, Quad Cities 1 & 2
BWR/4	183	Duane Arnold
	206	Vermont Yankee
	218	Brunswick 1 & 2, Cooper, Fitzpatrick, Hatch 1 & 2, Shoreham
	251	Browns Ferry 1, 2, & 3, Fermi 2, Hope Creek 1 & 2, Limerick 1 & 2, Peach Bottom 2 & 3, Susquehanna 1 & 2
BWR/5	251	LaSalle 1 & 2, Nine Mile Point 2, WNP-2
BWR/6	218	Clinton 1, River Bend 1
	238	Perry 1 & 2
	251	Grand Gulf 1

Table 8.3-1. Comparison of BWR Core Parameters

BWR Product Line	Representative Plant in Group	BWR Group Thermal Power (MWt)	Number of Fuel Assemblies	Number of Control Rods	Core Equivalent Diameter (in)	Core Active Height (in)	Core Average Power Density (kW/liter)
BWR/1	Humboldt Bay	230	172	40	77.5	79	40.3
	Big Rock Point	240	84	24	62.5	70	46
	Dresden 1	626	452	80	128.9	106	31.2
BWR/2	Nine Mile Point 1	1850	532	129	167.2	144	40.6
	Oyster Creek	1930	560	137	170.5	144	40.6
BWR/3	Monticello	1600	484	121	149	144	41.1
	Millstone 1	2000	580	145	163.1	144	40.8
	Quad Cities 1 and 2	2500	724	177	182.2	144	40.6
BWR/4	Vermont Yankee	1600 to 1700	368	89	129.9	144	50.9
	Cooper	2380	548	137	158.5	150	51.8
	Brunswick 1 and 2	2430 to 2500	560	137	160.2	150	50.5
	Peach Bottom 2 & 3	3300	764	185	187.1	150	49.2
BWR/5	Nine Mile Point 2	3300	764	185	187.1	150	49.2
BWR/6	Clinton	2900	624	145	169.1	150	52.4
	Perry 1 and 2	3600	748	177	185.2	150	54
	Grand Gulf	3850	800	193	191.5	150	54.1

Table 8.10-1. Comparison of BWR Pressure Suppression Containment Design Parameters

Parameter	Mark I	Mark II	Mark III
BWR product line	BWR/2 to BWR/4	Late-BWR/4 and BWR/5	BWR/6
Reactor power level (MWt)	1850 to 3300	2436 to 3323	2894 to 3833
Drywell shape	"Light bulb"	Frustrum of a cone	Cylinder
Typical Drywell Construction	Free-standing steel***	Reinforced concrete	Reinforced concrete
Drywell air volume (ft^3)	131,000 to 176,000	200,000 to 303,000	250,000 to 278,000
Drywell design pressure (psig)	56 to 62	45 to 53	15
Drywell design temperature (°F)	281 to 340	281 to 340	330
Drywell inerting	Yes	Yes	No
Suppression chamber (wetwell)	Torus below and around the drywell	Annular cylinder below the drywell	Annular cylinder around the drywell
Typical suppression chamber construction	Free-standing steel	Reinforced or post-tensioned concrete with steel liner, or free-standing steel	Free-standing steel, or reinforced concrete with steel liner
Suppression chamber water volume (ft^3)	78,000 to 122,000	81,000 to 155,000	120,000 to 136,000
Suppression chamber air volume (ft^3)	95,000 to 134,000	134,000 to 192,000	713,000 to 1,264,000
Suppression chamber design pressure (psig)*	56 to 62	45 to 53	15
Suppression chamber design temperature (°F)**	281 to 310	212 to 281	185

Table 8.10-1. Comparison of BWR Pressure Suppression Containment Design Parameters (Continued)

Parameter	Mark I	Mark II	Mark III
Configuration of vents between the drywell and the suppression chamber	Diagonal large-diameter pipes ending in "ram's head" distribution manifold and vertical piping venting below the water level of the pool	Vertical standpipes submerged about 6 feet below water level of the suppression pool	Horizontal vent openings below the water level of suppression pool
Typical vent area (ft^3)	238	219	480

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* All containments for BWR/3 to BWR/6 plants have the same drywell and torus design pressure. The BWR/2 Mark I containment has a 62 psig drywell design pressure and a 35 psig torus design pressure.

** All Mark I BWRs except Nine Mile Point 1 (205°F), Oyster Creek (150°F), Brunswick and Fitzpatrick (both 220°F).

*** All except Brunswick which is reinforced concrete.

Table 8.13-1. Comparison of BWR AC Load Distribution

BWR Product Line	AC Division A	AC Division B	AC Division C
BWR/2	Core Spray A/C	Core Spray B/D	-------
	Containment Spray A	Containment Spray B	
	Emerg. Service Water A	Emerg. Service Water B	
BWR/3 to BWR/4	LPCI (RHR) A/C	LPCI (RHR) B/D	-------
	Core Spray A	Core Spray B	
	RHR Service Water A	RHR Service Water B	
BWR/5 and BWR/6	LPCI (RHR) A	LPCI (RHR) B	HPCS
	LPCS	LPCI C	
	Emerg. Service Water A	Emerg. Service Water B	Emerg. Service Water C

Table 8.13-2. Comparison of BWR DC Load Distribution

BWR Product Line	DC Division A	DC Division B	DC Division C
BWR/2	Isolation Condenser valve power	Isolation Condenser valve power	-------
	ADS A control	ADS B control	
	Core Spray A/C control	Core Spray B/D control	
	Containment Spray A control	Containment Spray B control	
	Emerg. Service Water A control	Emerg. Service Water B control	
	Diesel A control	Diesel B control	
BWR/3 to BWR/4	RCIC power and control	HPCI power and control	-------
	ADS A control	ADS B control	
	LPCI (RHR) A/C control	LPCI (RHR) B/D control	
	Core Spray A control	Core Spray B control	
	RHR Service Water A control	RHR Service Water B control	
	Diesel A control	Diesel B control	
BWR/5 and BWR/6	RCIC power and control		HPCS control
	ADS A control	ADS B control	
	LPCI (RHR) A control	LPCI (RHR) B control	
	LPCS control	LPCI C control	
	Emerg. Service Water A control	Emerg. Service Water B control	Emerg. Service Water C control
	Diesel A control	Diesel B control	Diesel C control

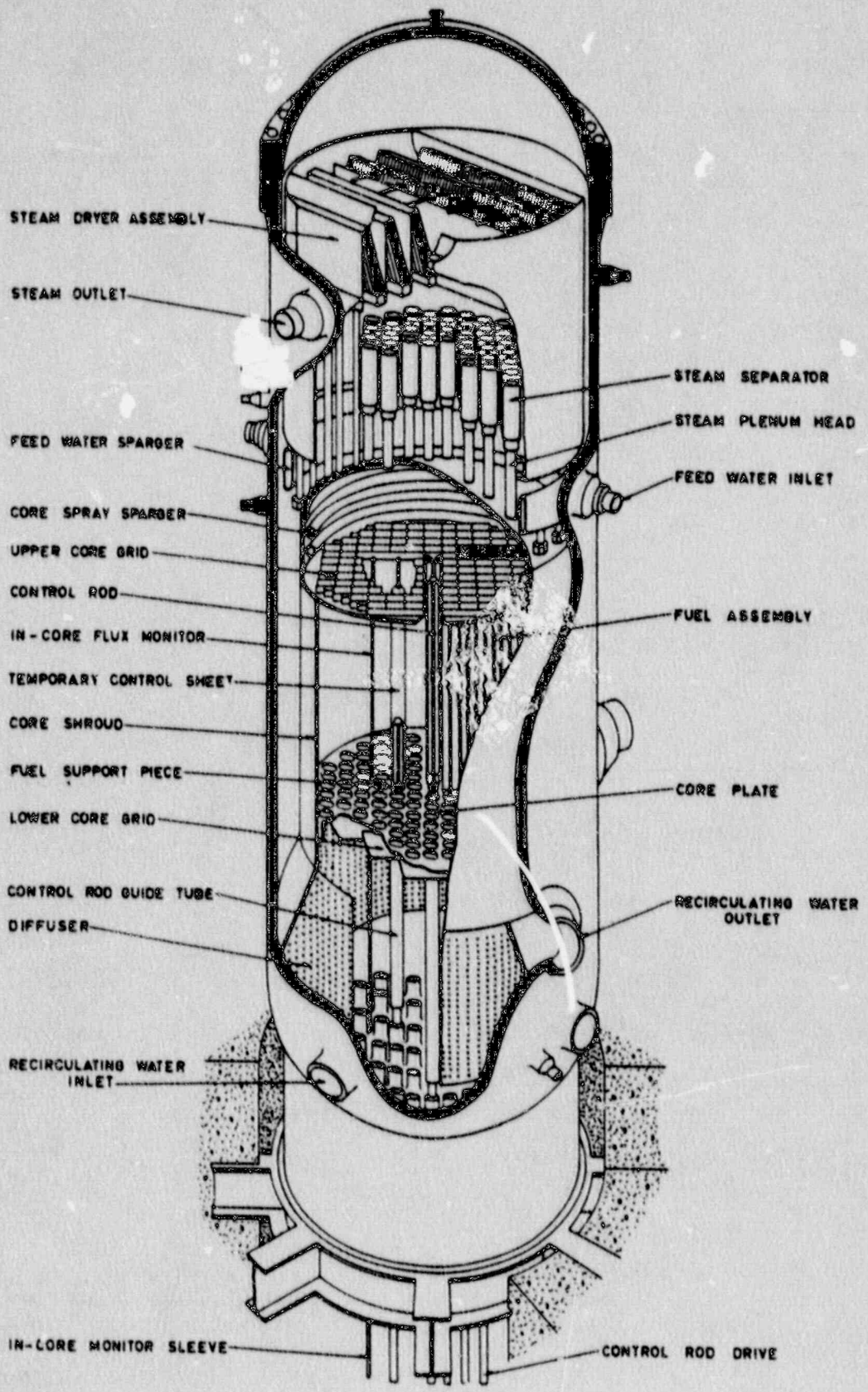


Figure 8.2-1. BWR/2 Non-Jet Pump BWR Nuclear Boiler

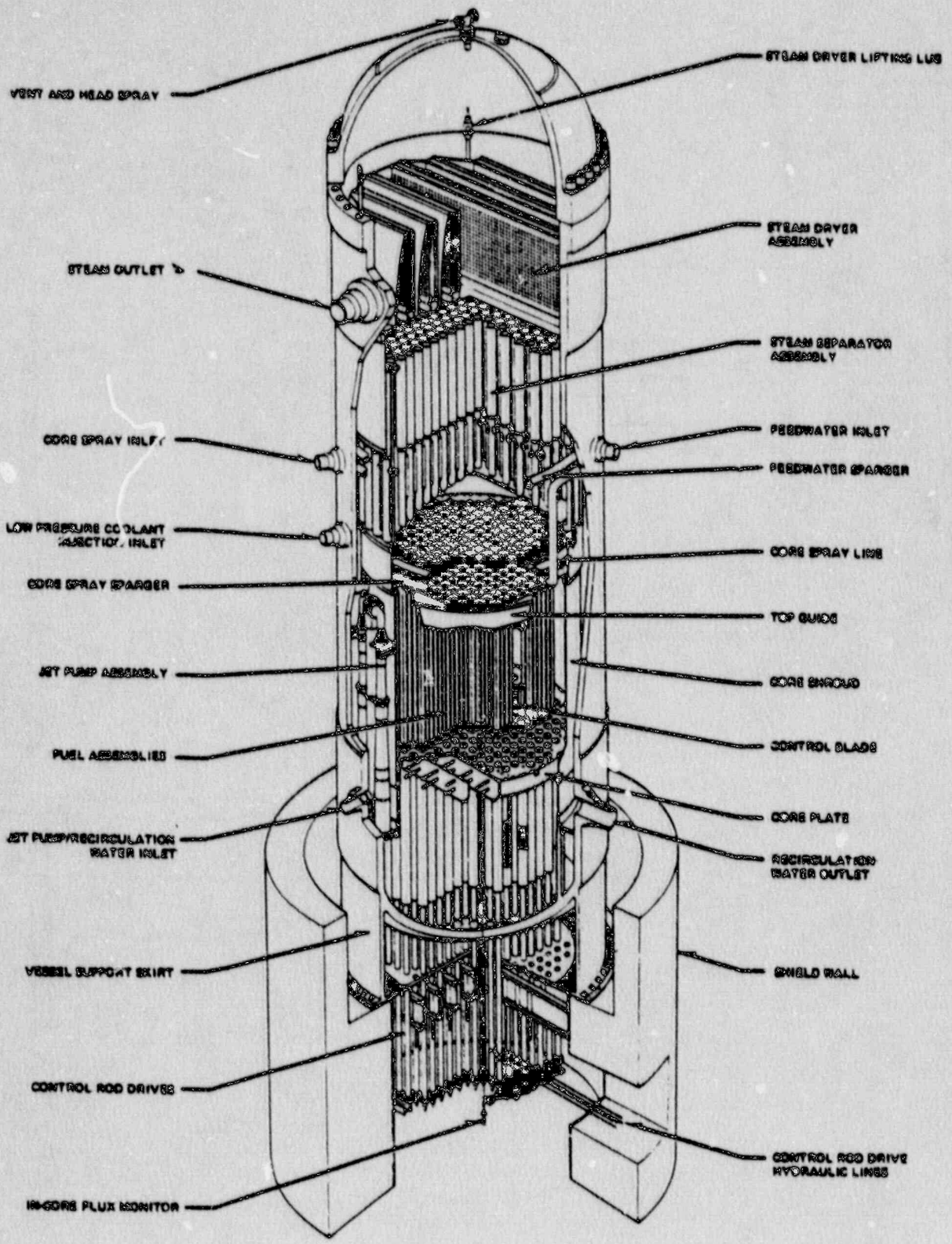


Figure 8.2-2. Representative BWR/3 to BWR/6 Jet Pump Nuclear Boiler

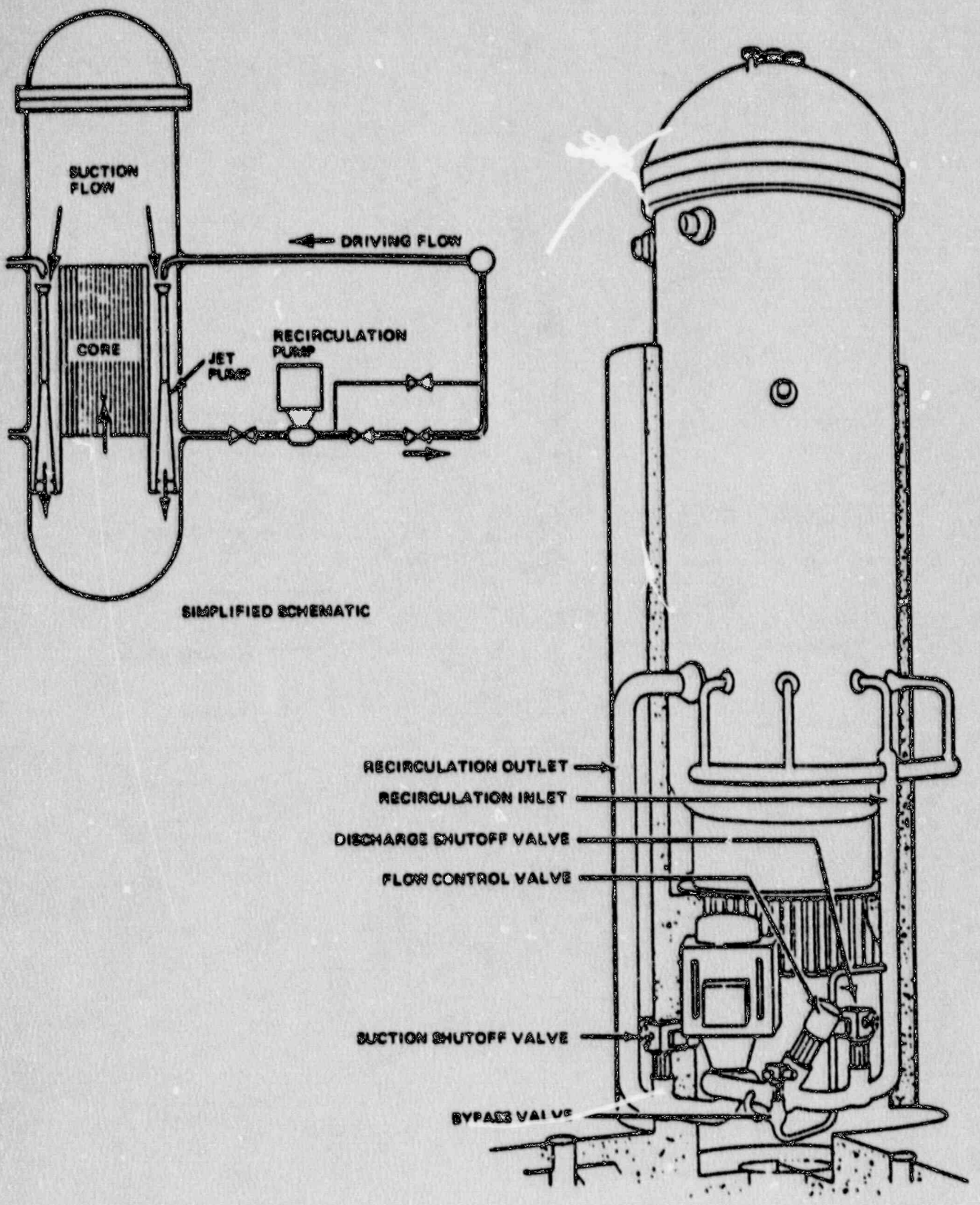


Figure 8.2-3. Jet Pump Recirculation System Elevation and Isometric (Representative of BWR/5 and BWR/6 Plants)

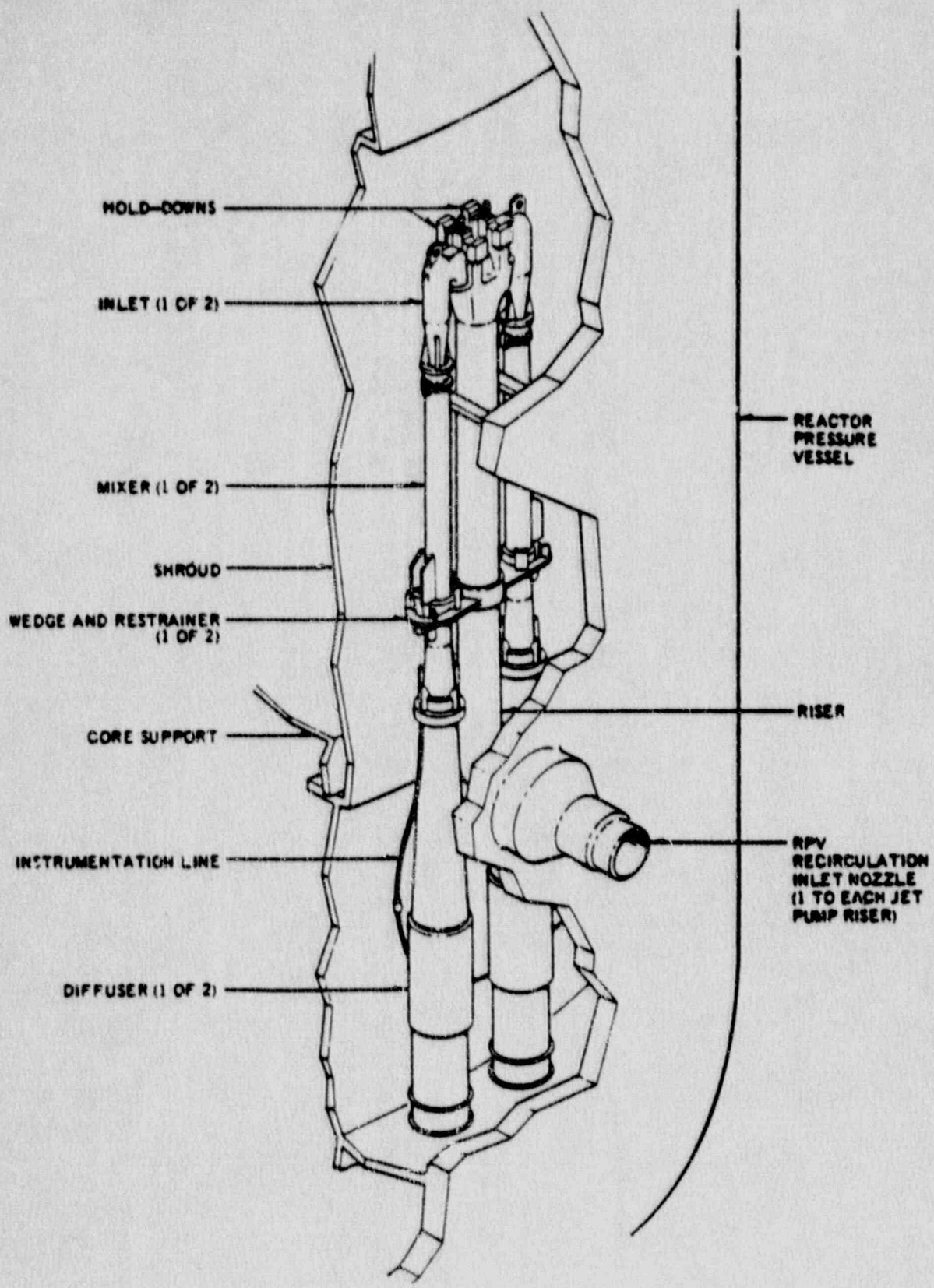


Figure 8.2-4. Typical Jet Pump Installation

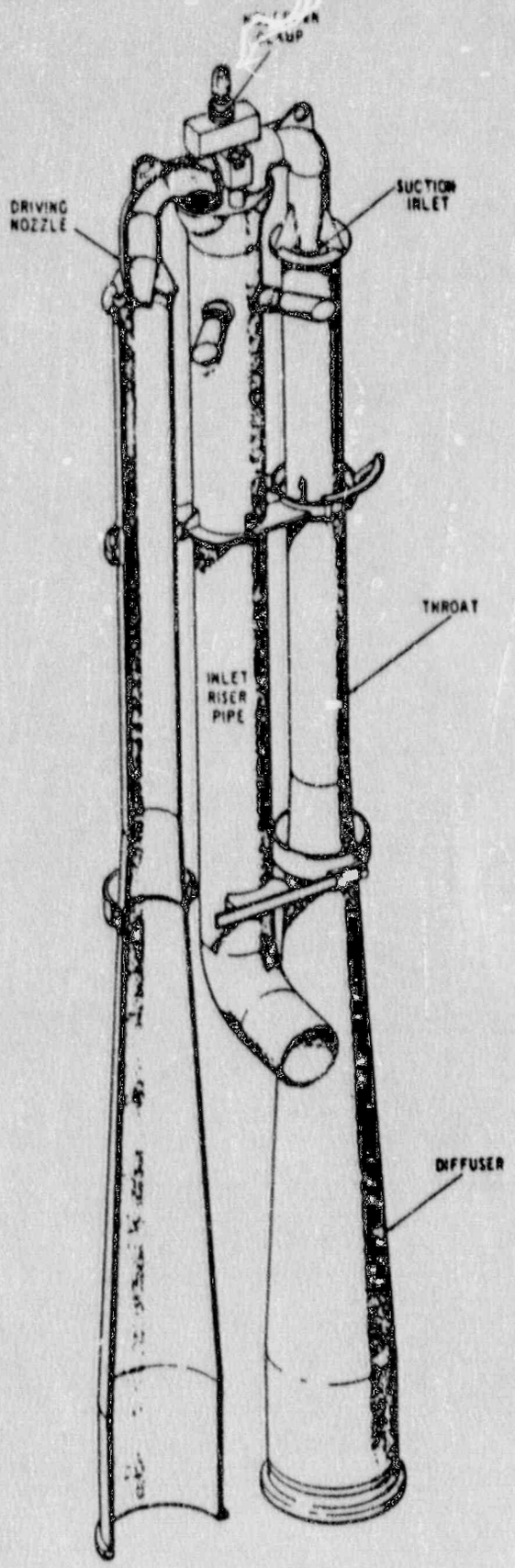


Figure 8.2-5. Jet Pump Details

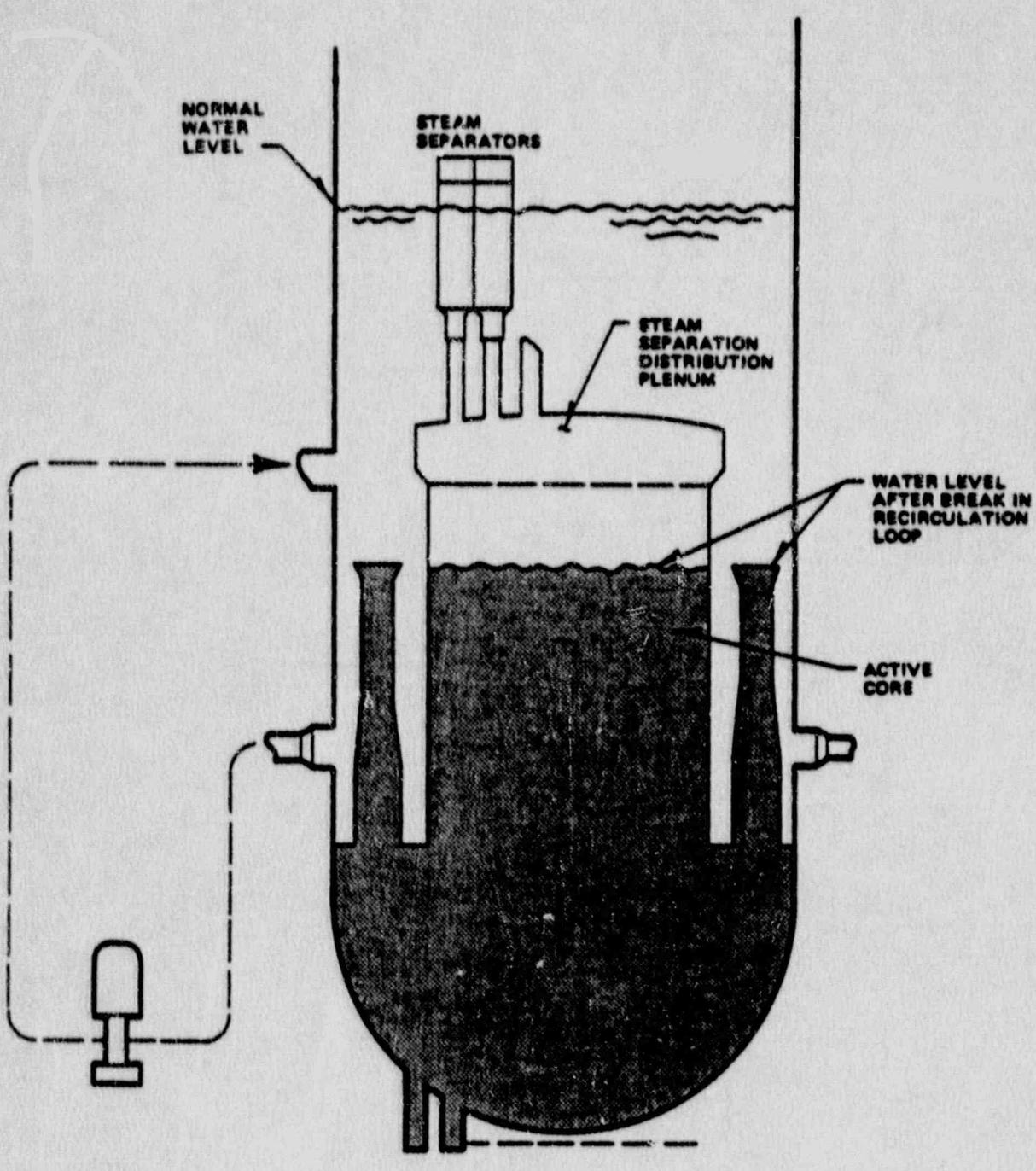


Figure 8.2-6. Core Flooding Capability with Jet Pump Recirculation System

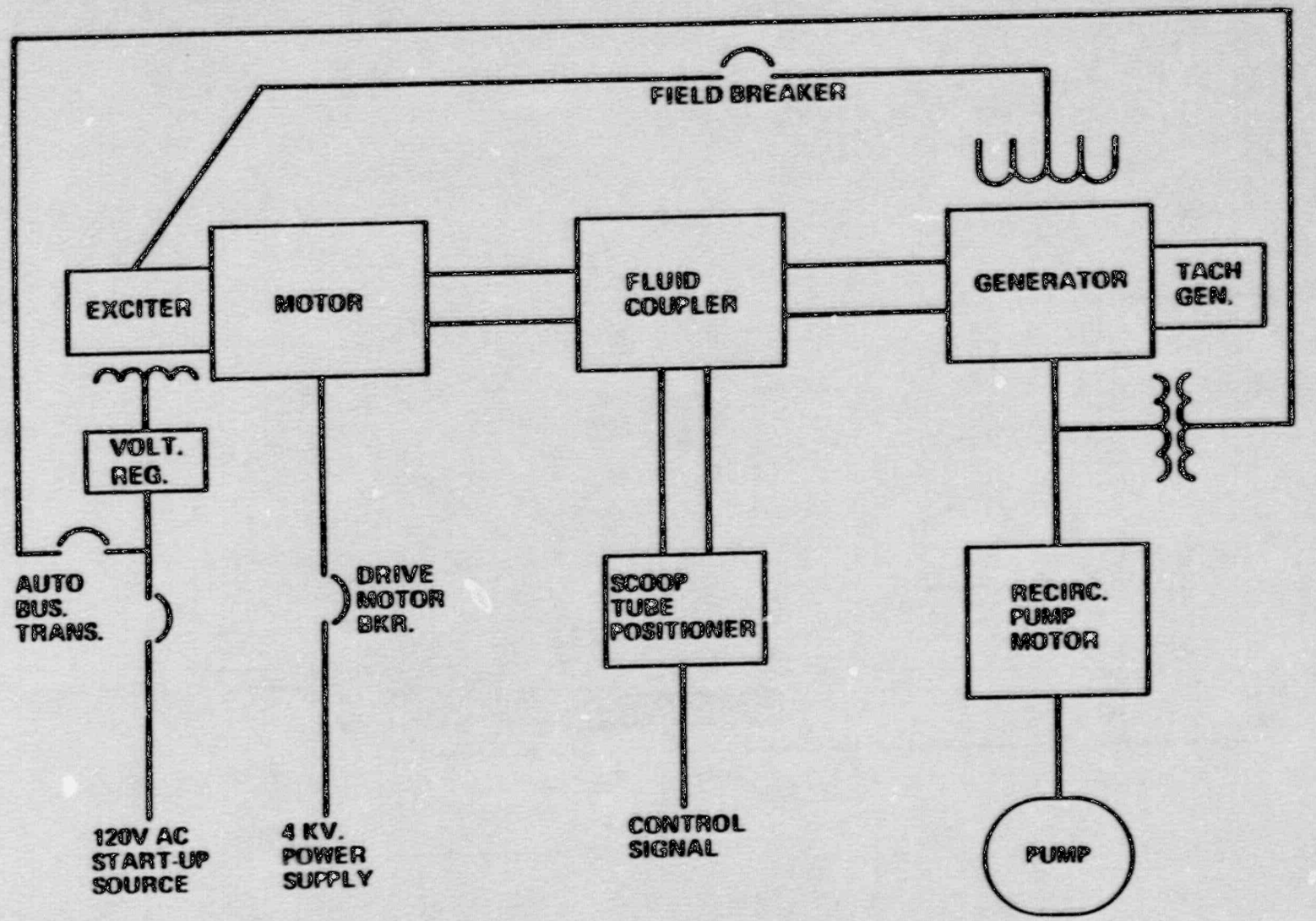


Figure 8.2-7. Representative BWR/2 to BWR/4 Recirculation Pump Variable Speed Control System

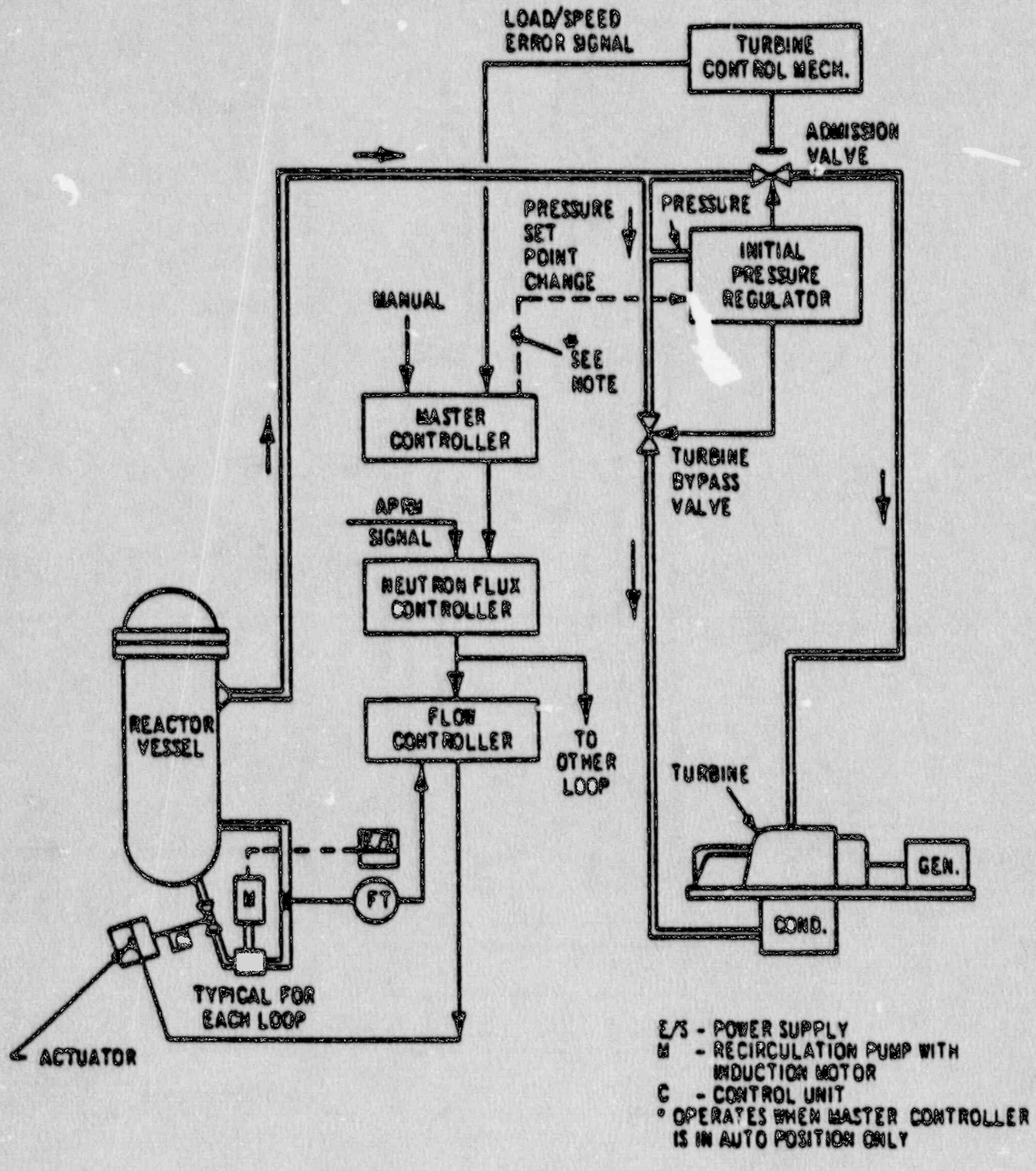


Figure 8.2-8. Representative BWR/5 and BWR/6 Recirculation Flow Control System

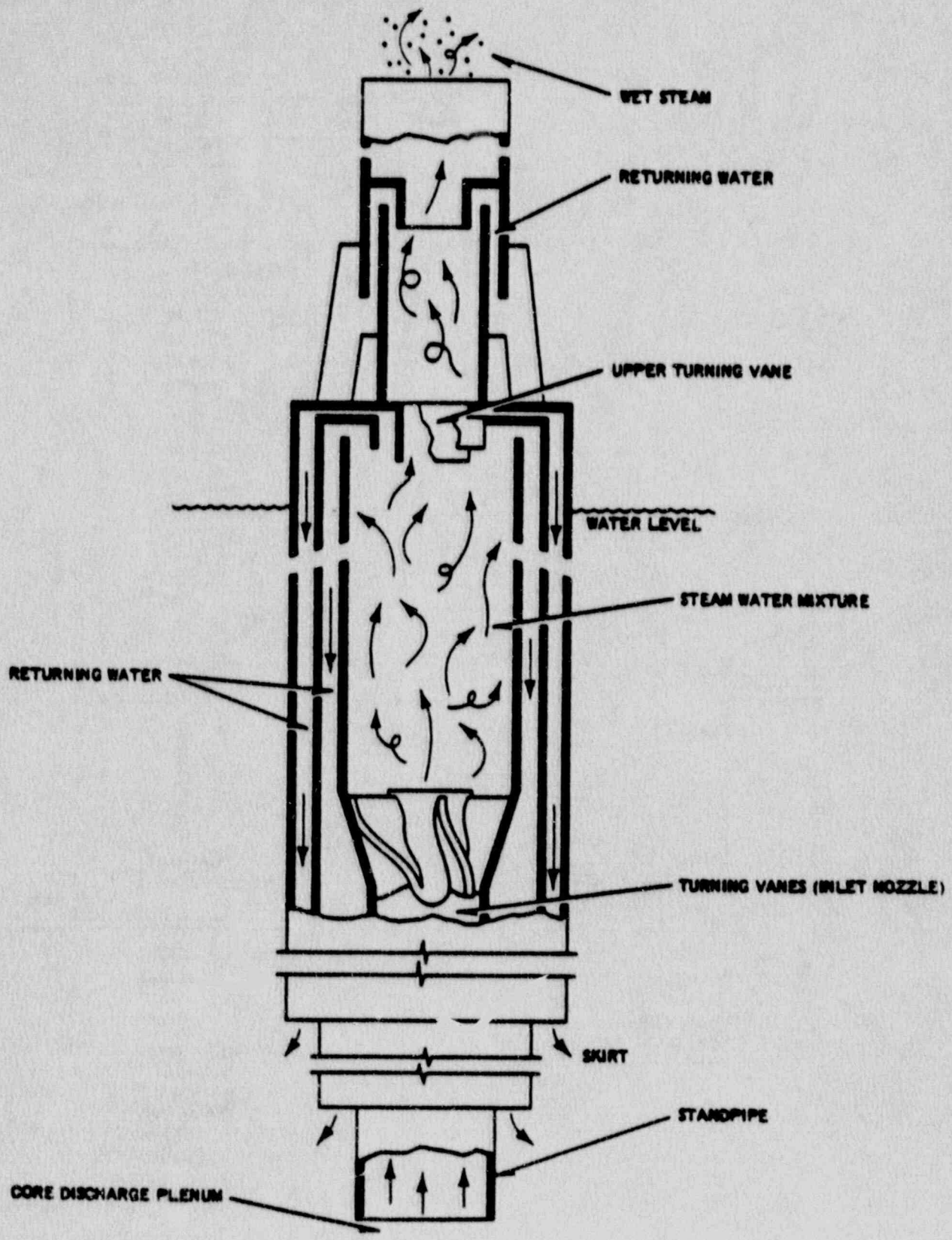


Figure 8.2-9. Steam Separator

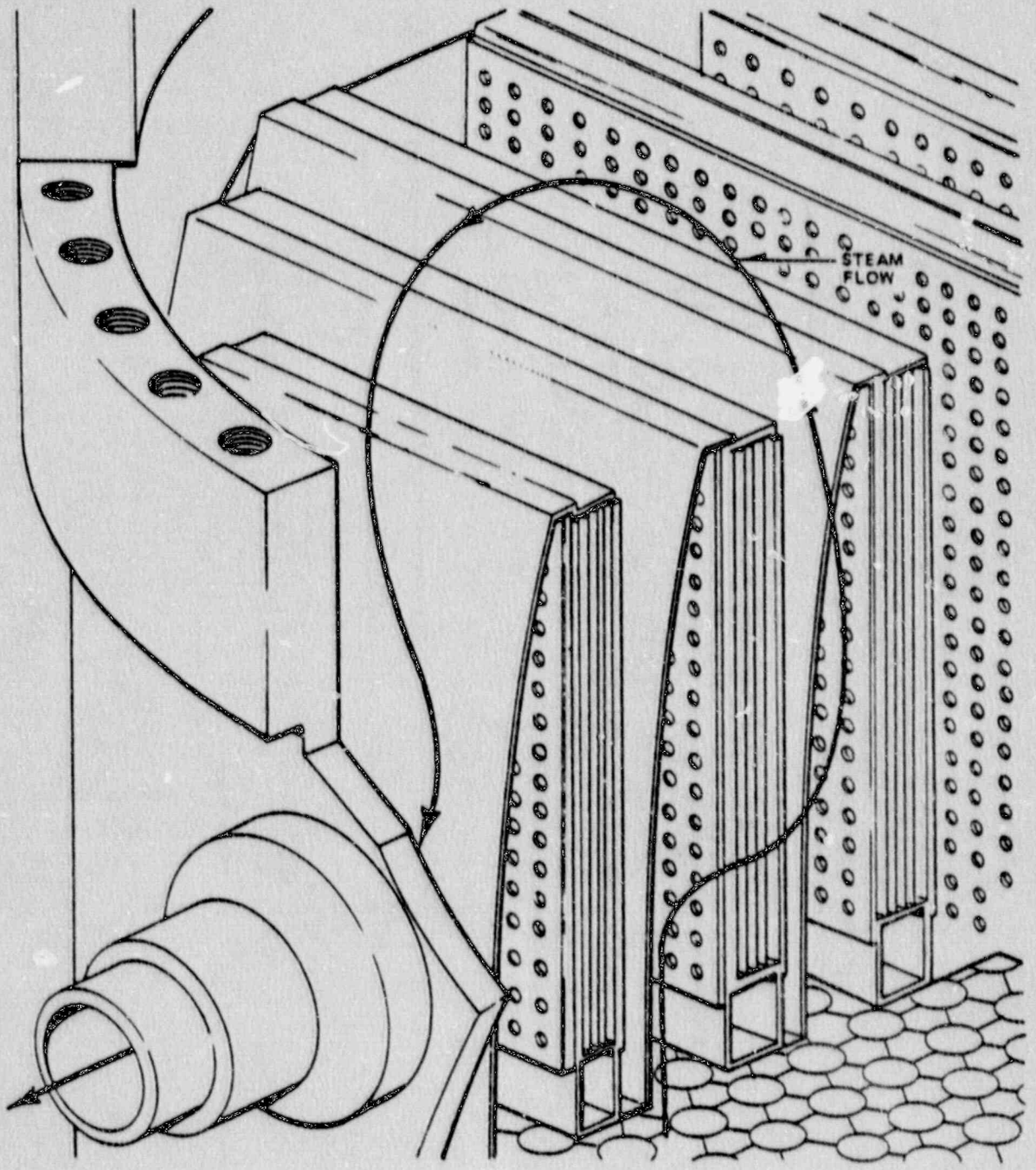


Figure 8.2-10. Steam Dryer

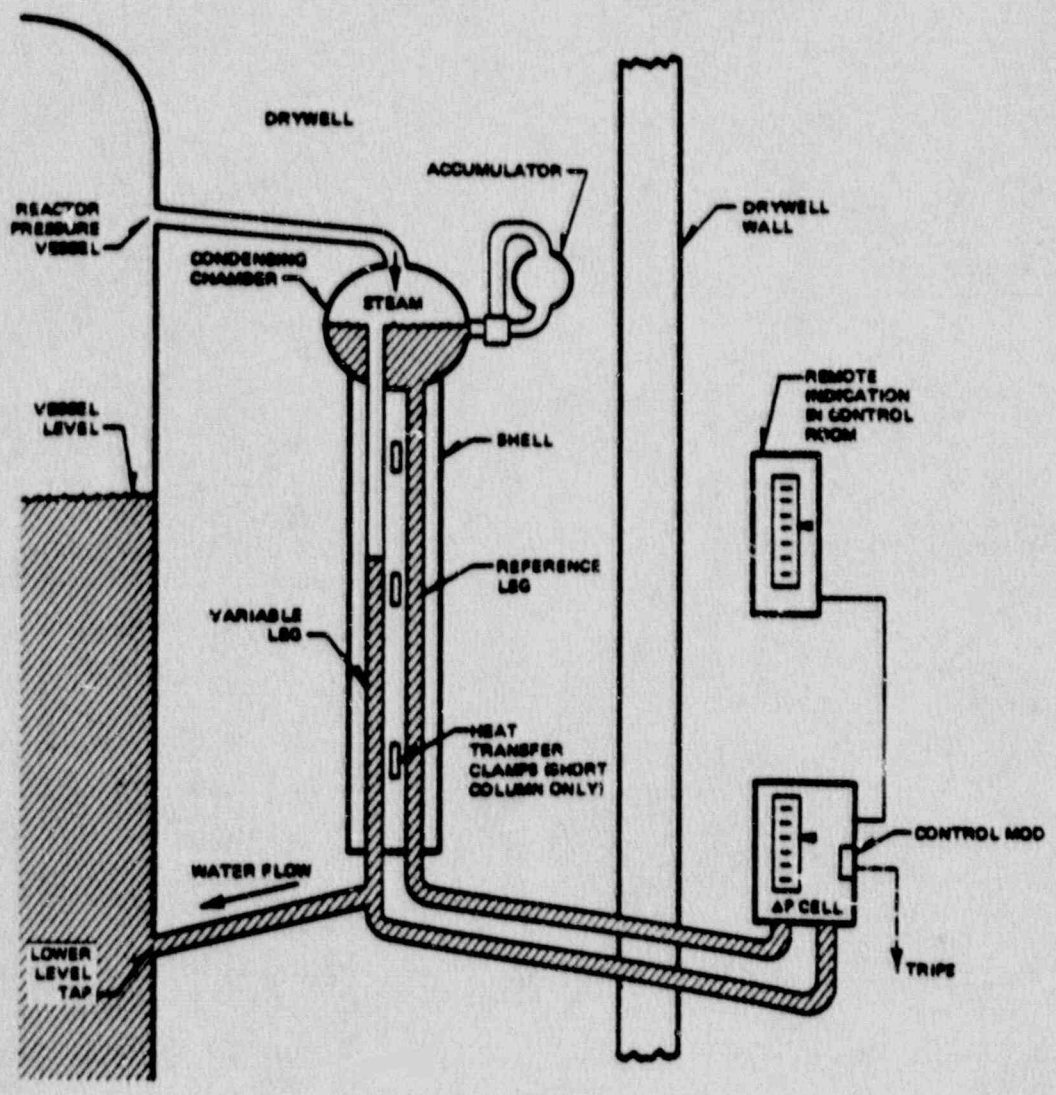


Figure 8.2-11. Simplified Diagram of a Yarway Reactor Vessel Water Level Measuring Instrument

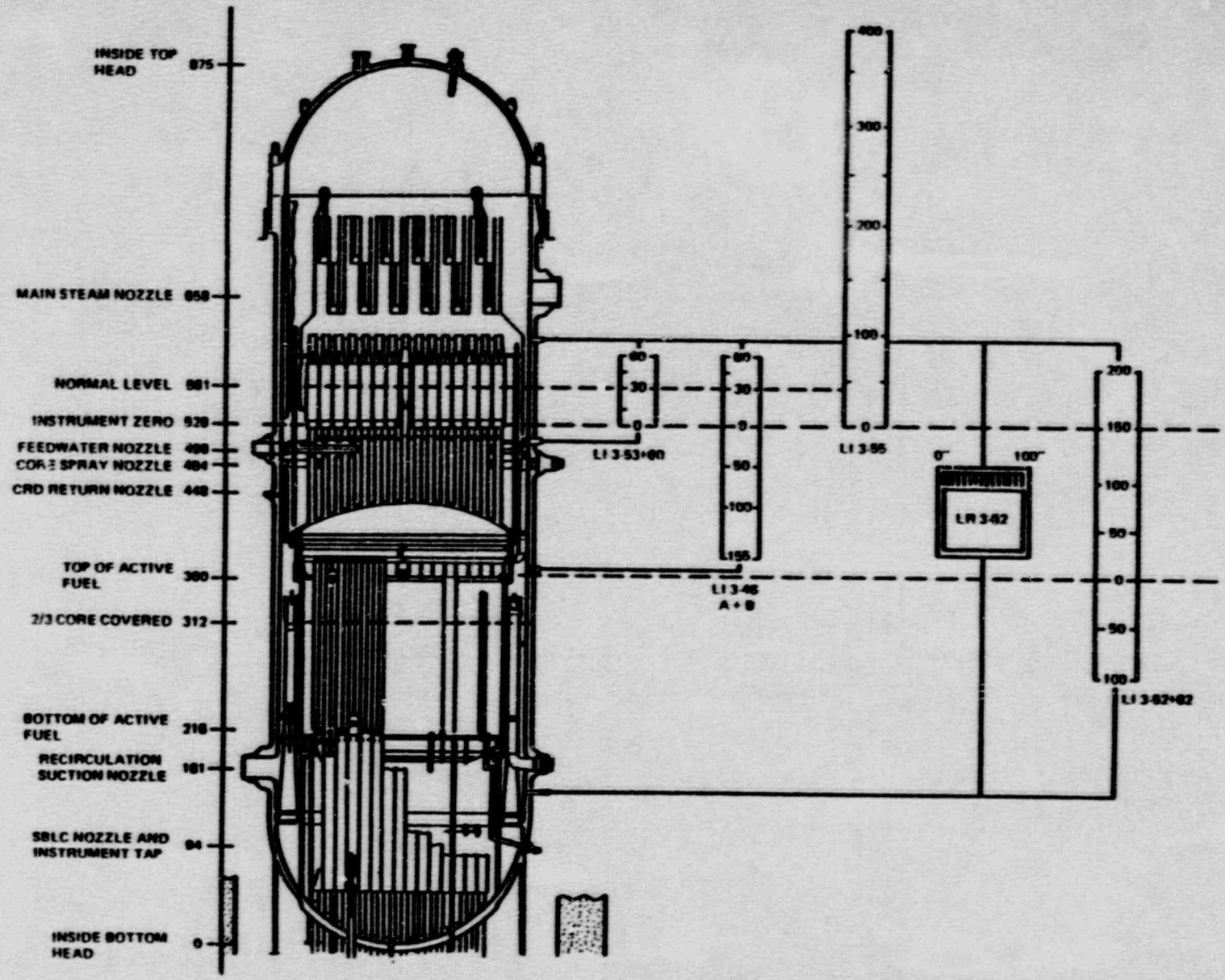


Figure 8.2-12. Example of BWR Vessel Water Level Instrument Ranges

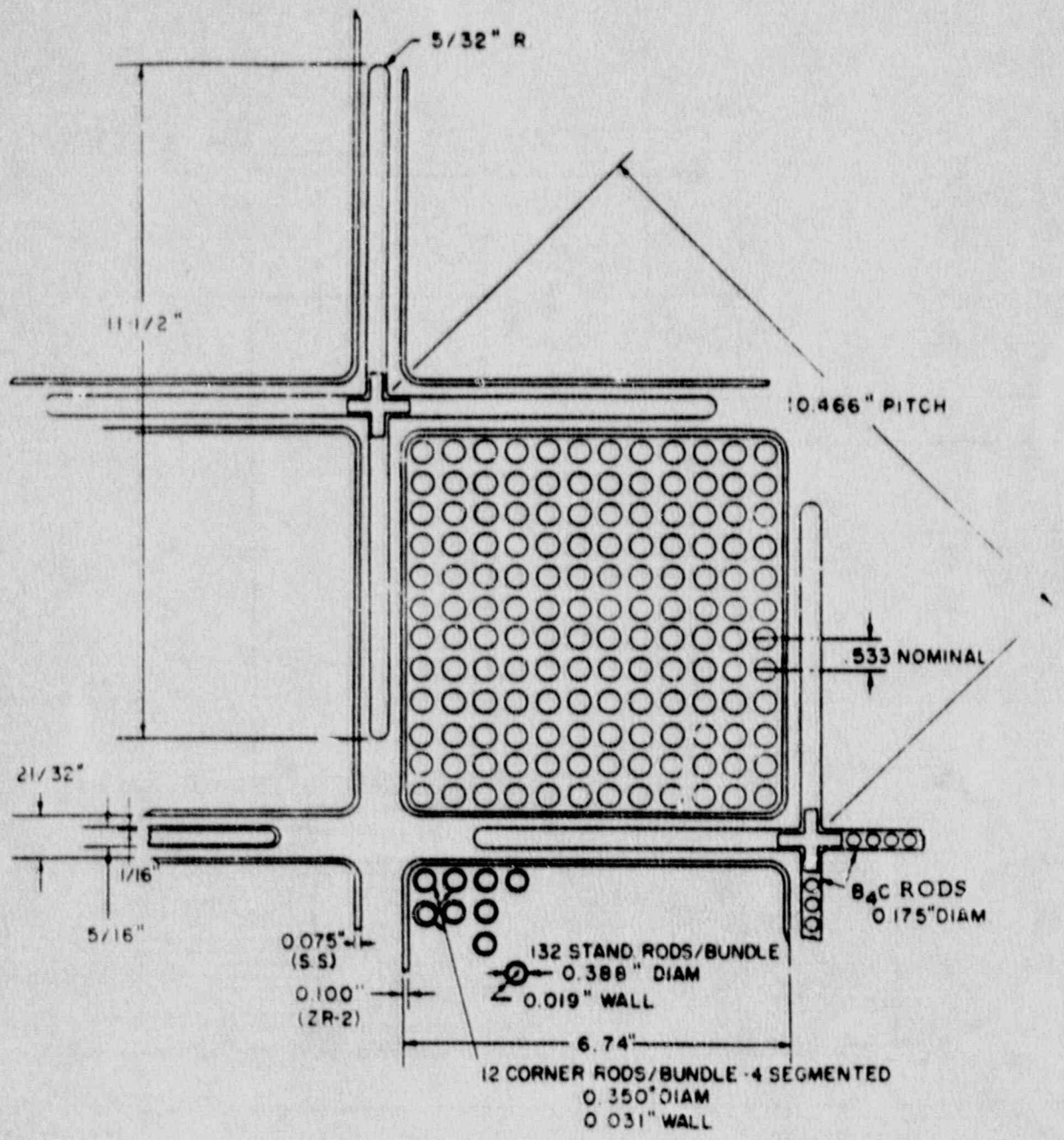


Figure 8.3-1. Big Rock Point Fuel Assembly and Control Rod Geometry

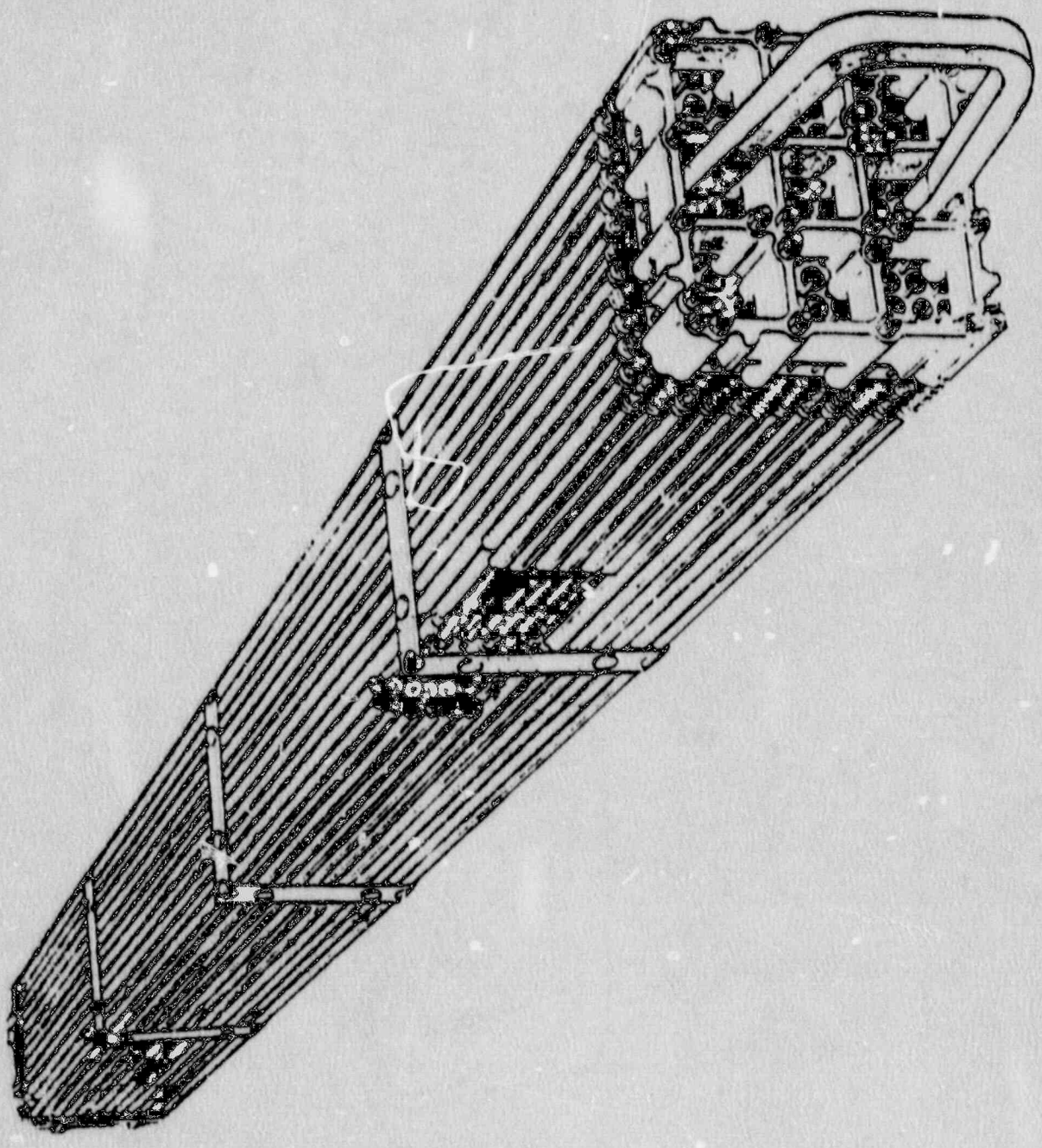


Figure 8.3-2. Details of Big Rock Point (BWR/1) Fuel Assembly (Fuel Channel Removed)

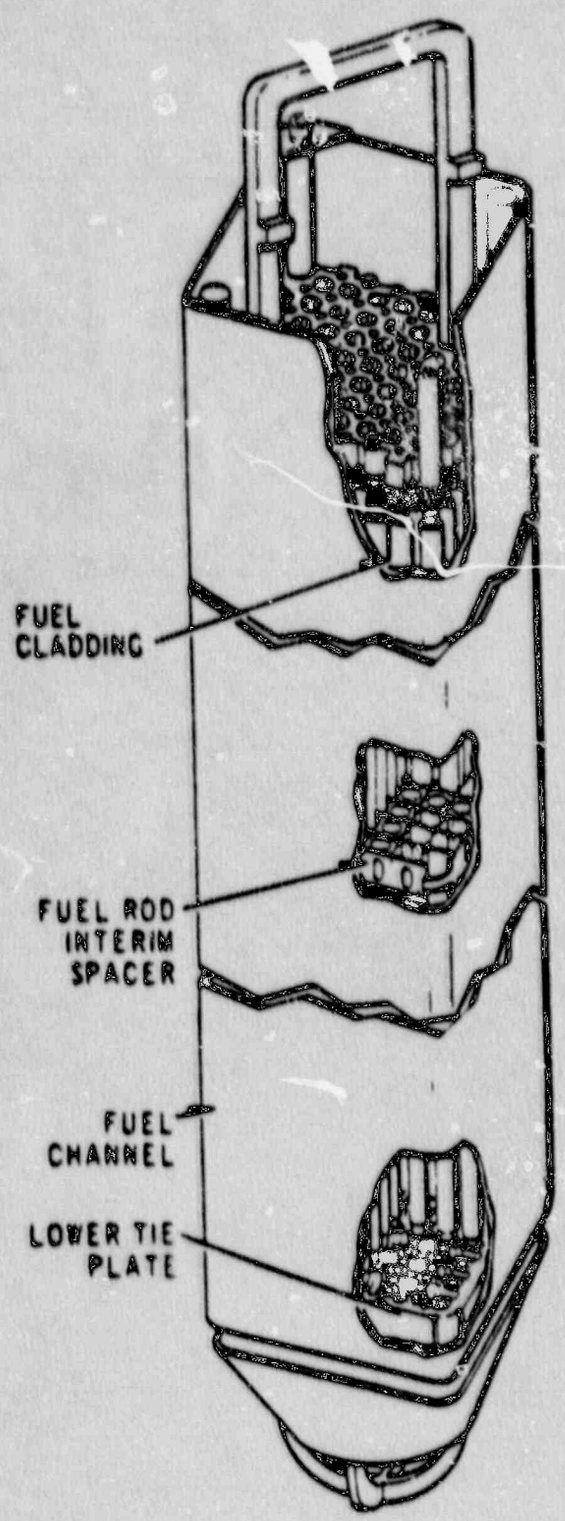
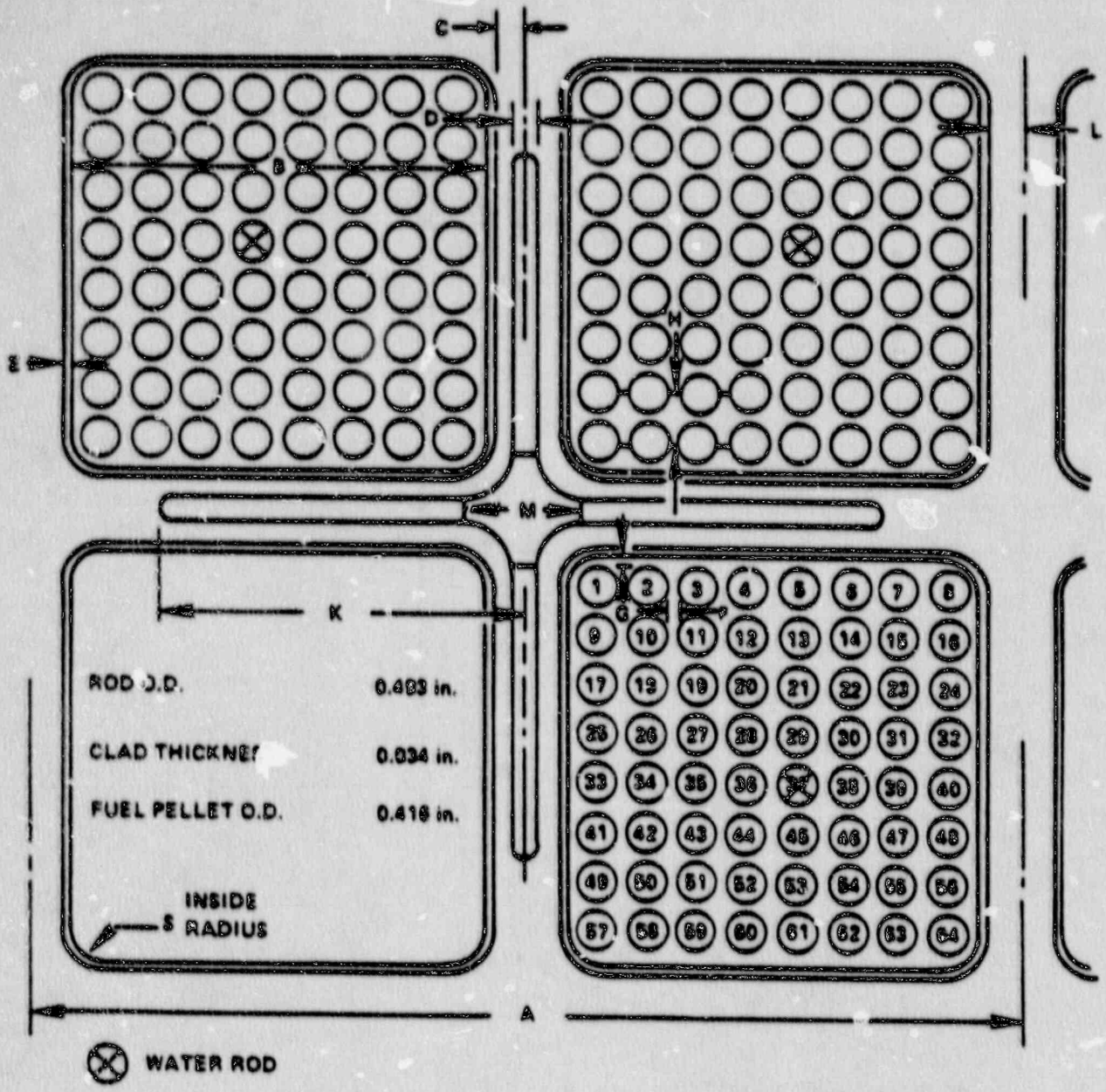


Figure 8.3-3. General Arrangement of a 7 X 7 Fuel Assembly



DIM. IDENTIFICATION	A	B	C	D	E	F	G	H	I	J
DIM. INCHES	12.0	5.278	0.241	0.350	0.000	0.147	0.1525	0.640		

DIM.IDENTIFICATION	K	L	M	N	O	P	Q	R	S	
DIM. INCHES	4.875	0.241	1.953						0.400	

Figure 8.3-4. 8 X 8 Fuel Assembly and Control Rod Geometry

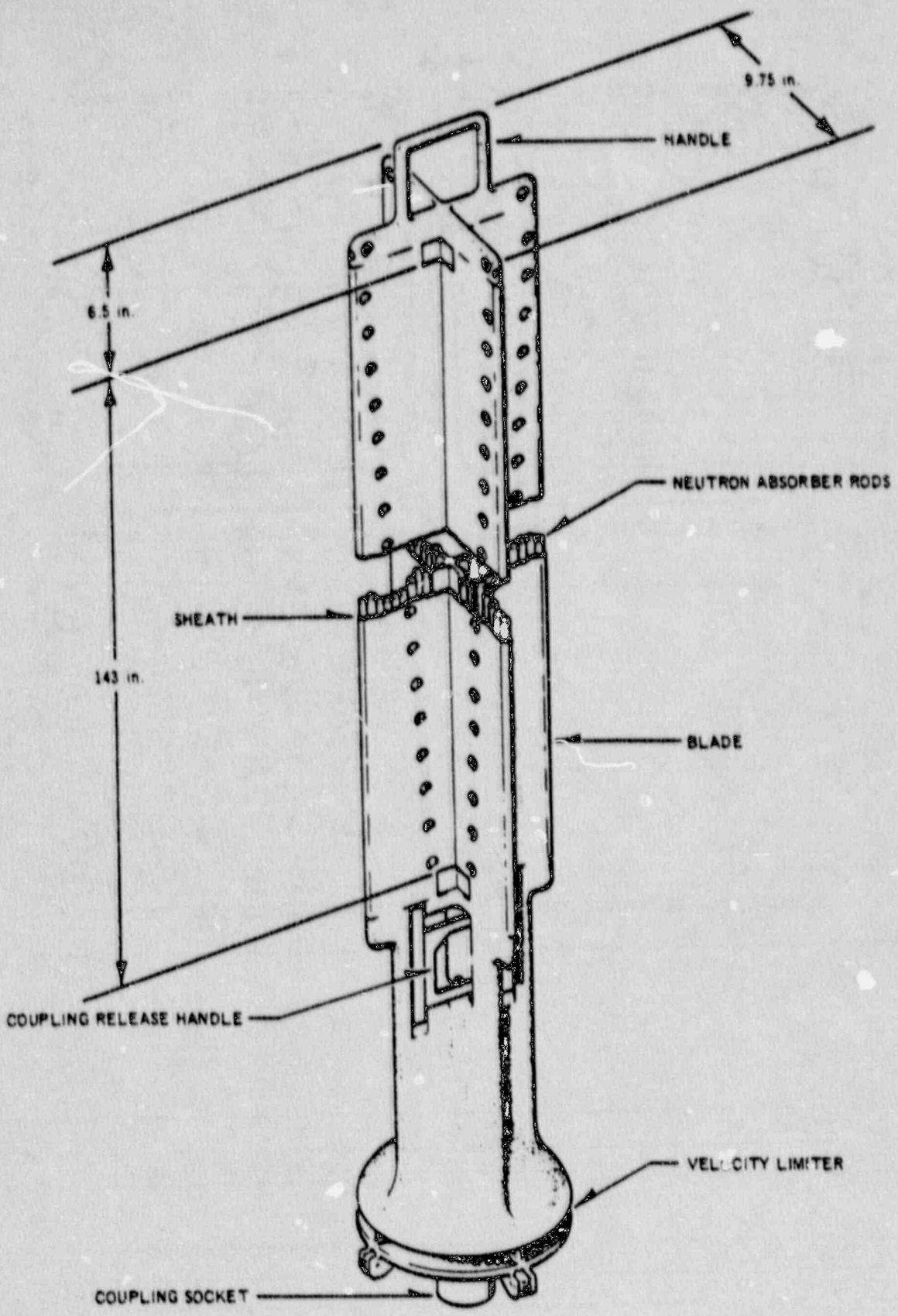


Figure 8.4-1. BWR Control Rod Assembly

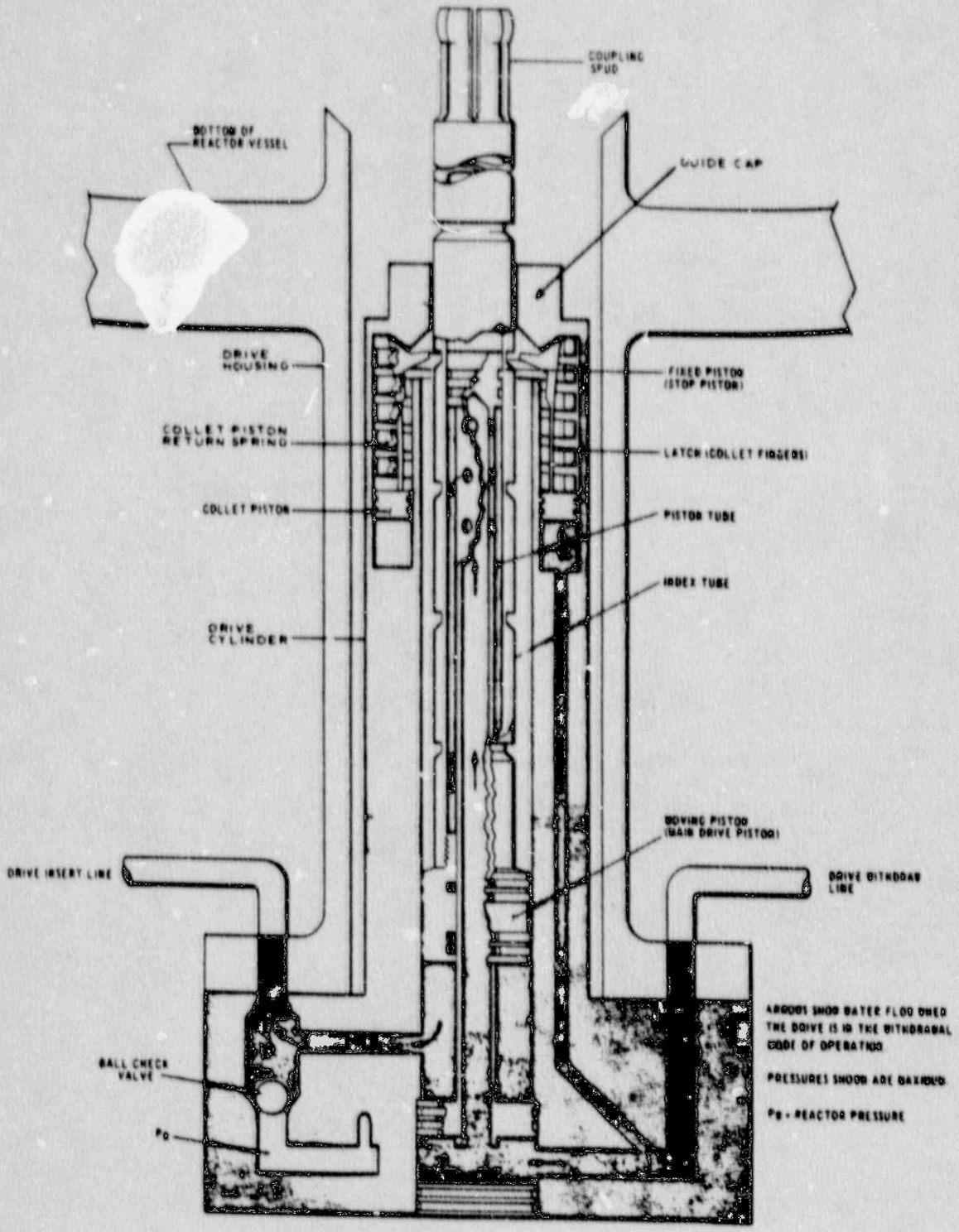


Figure 8.4-2. BWR Hydraulic Control Rod Drive

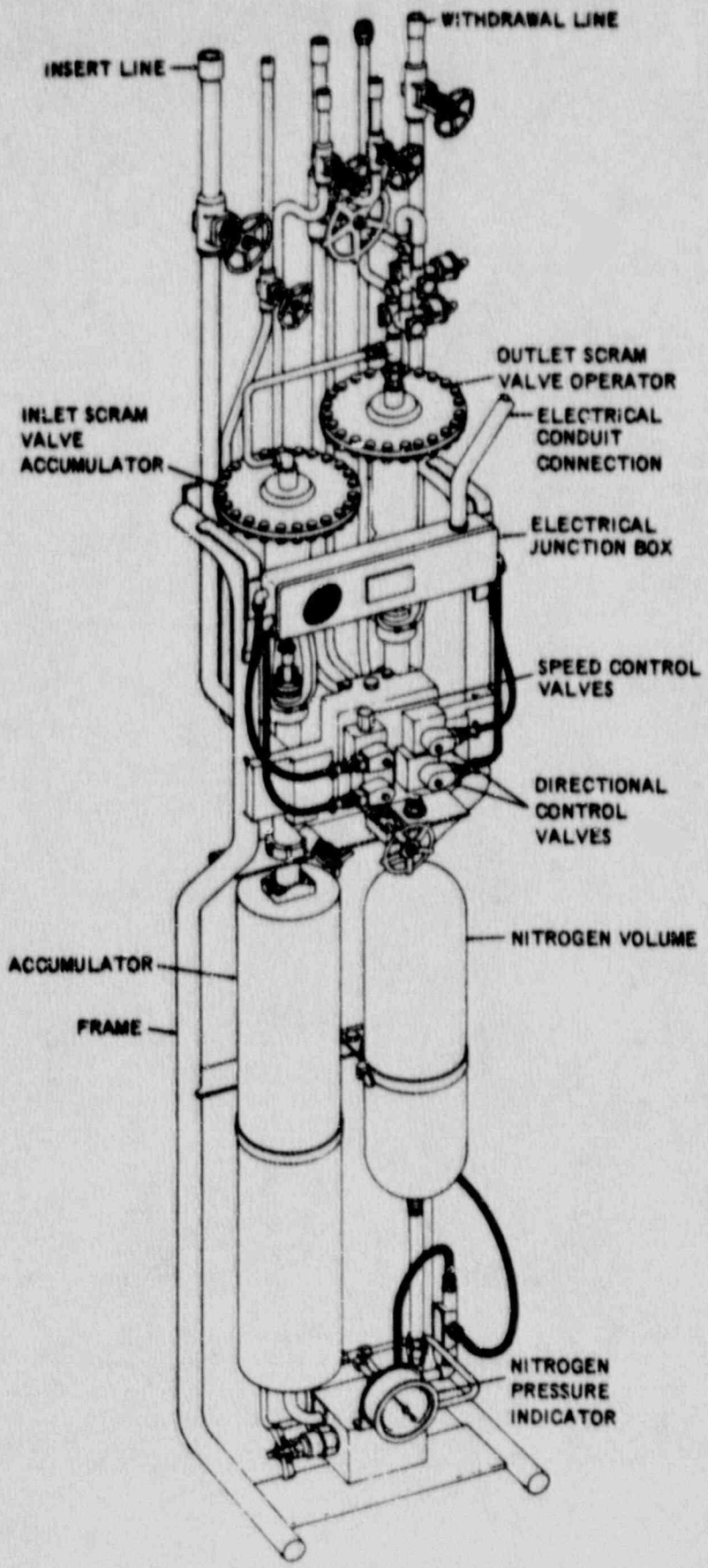


Figure 8.4-3. Control Rod Drive Hydraulic Control Unit

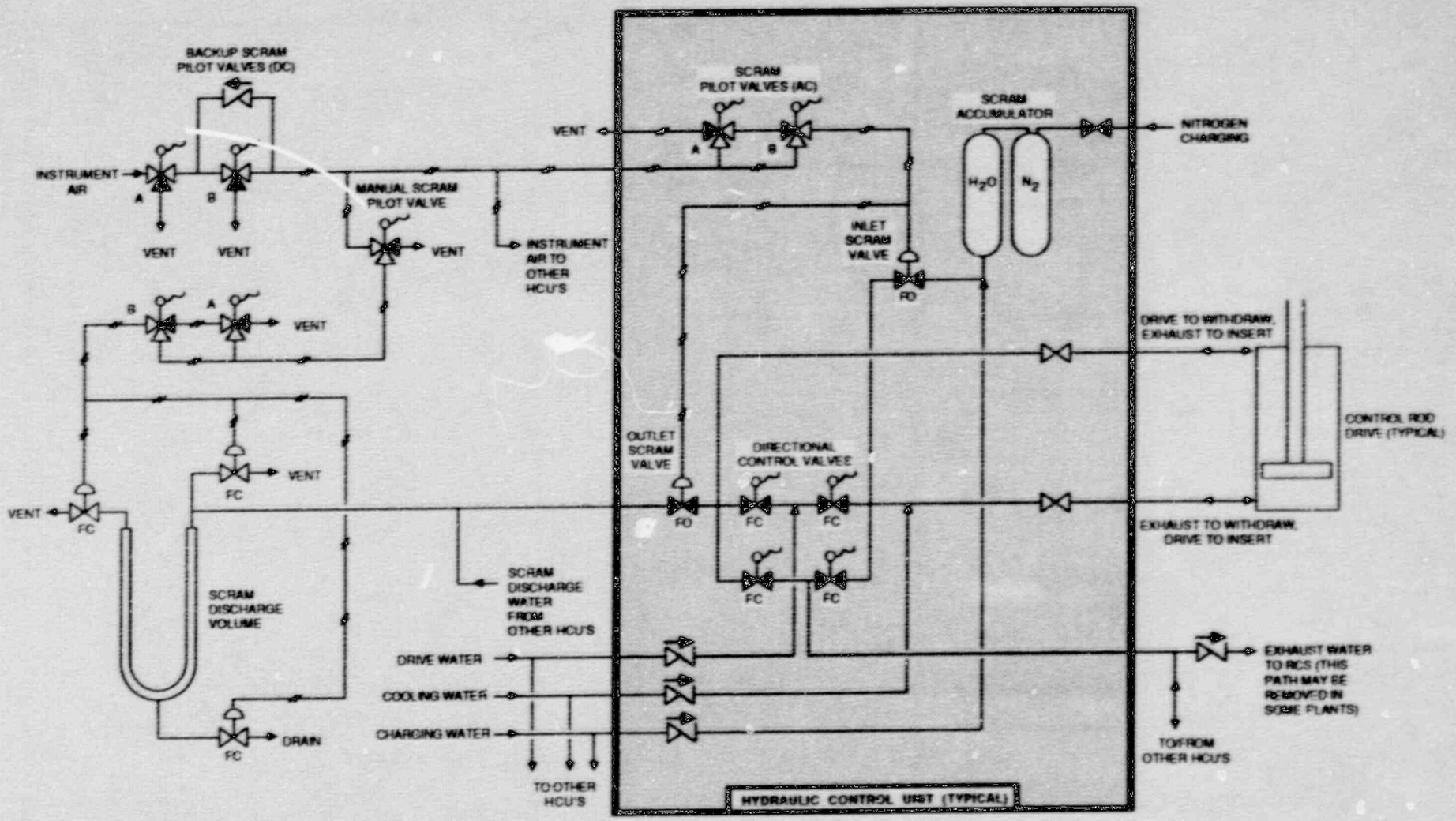


Figure 8.4-4. Simplified Diagram of Portions of the Control Rod Drive Hydraulic System That Are Related to the Scram Function

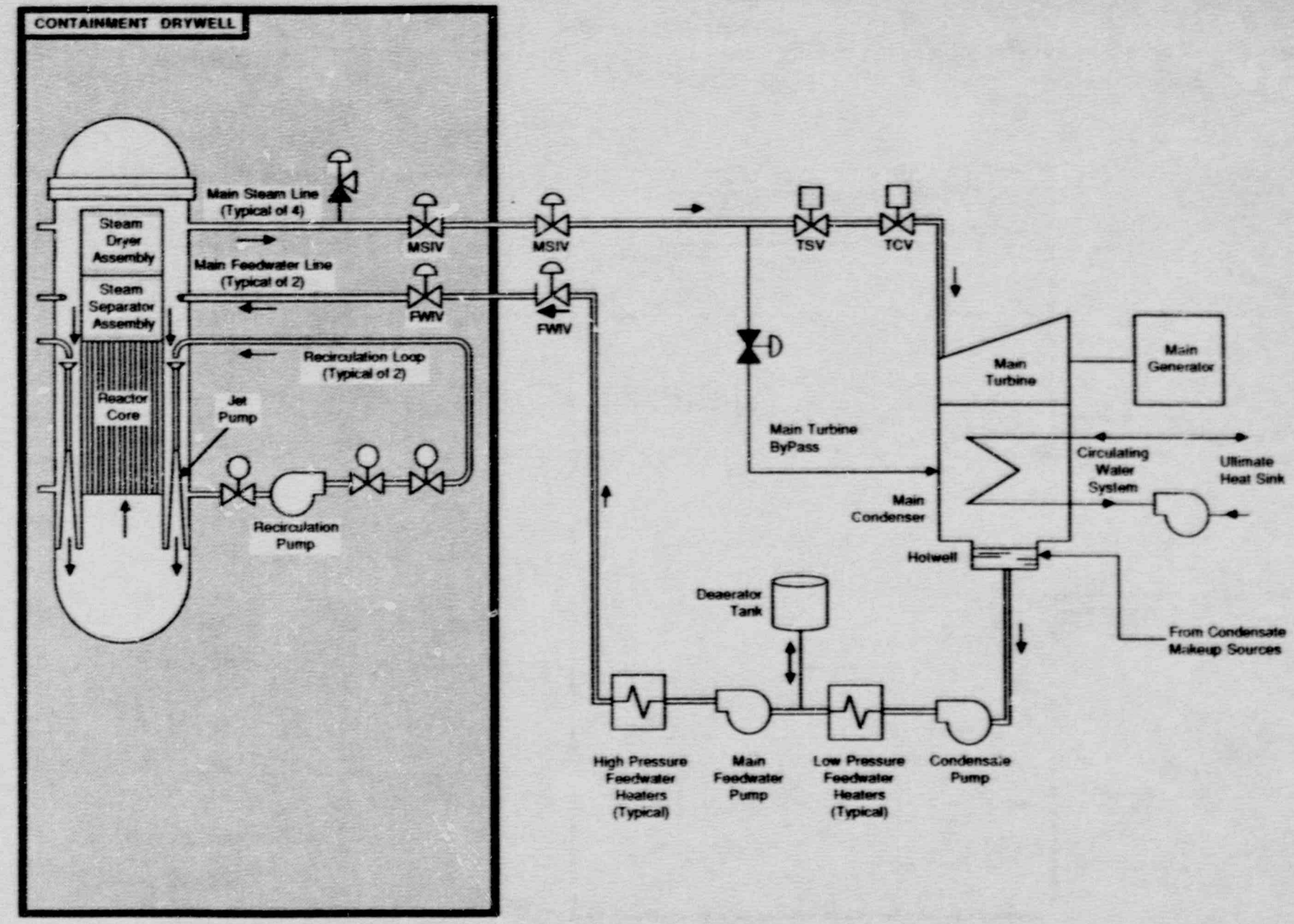


Figure 8.5-1. Normal BWR Heat Transport Paths During Power Operation

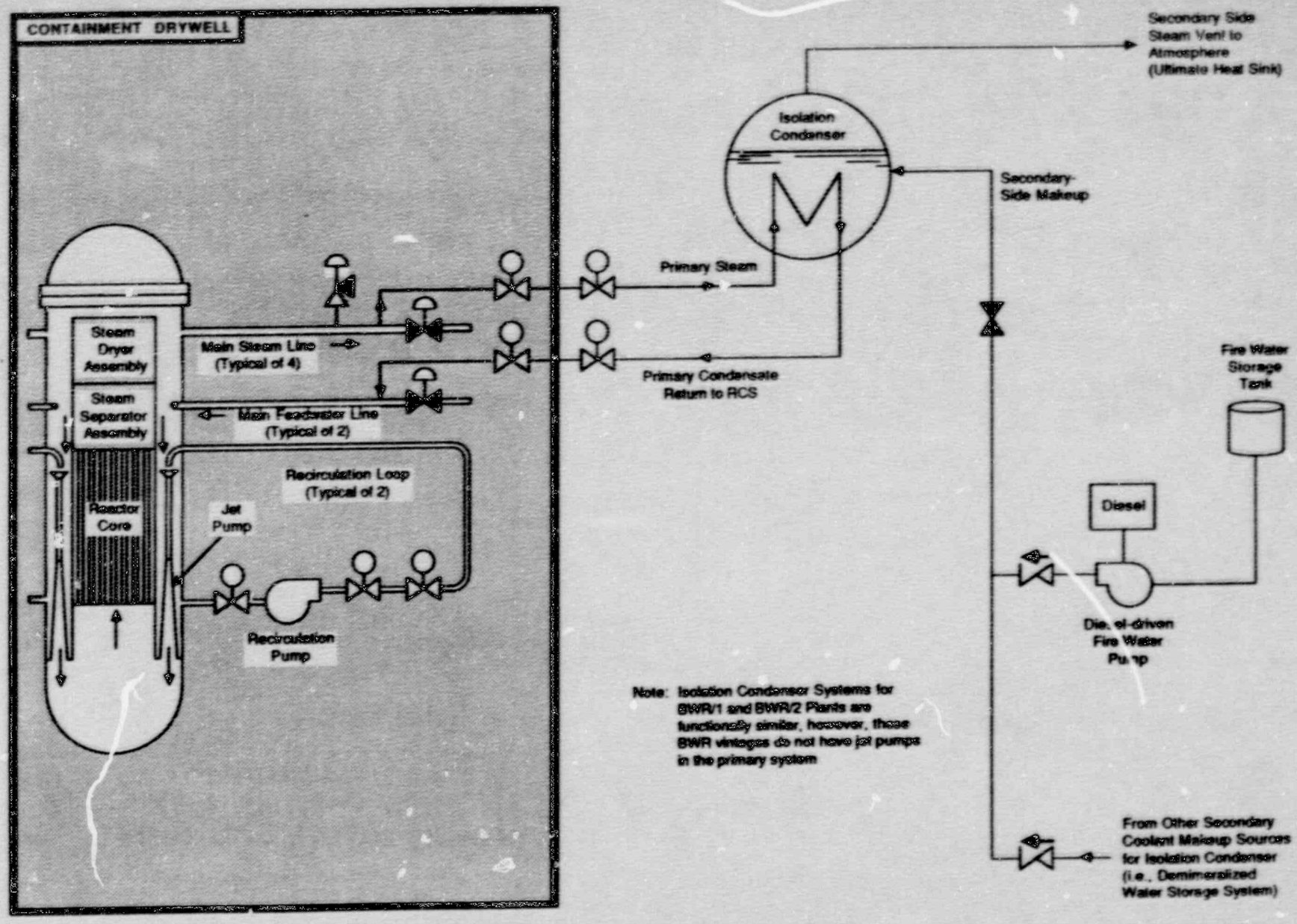


Figure 8.6-1. BWR/3 Isolation Condenser Heat Transport Paths

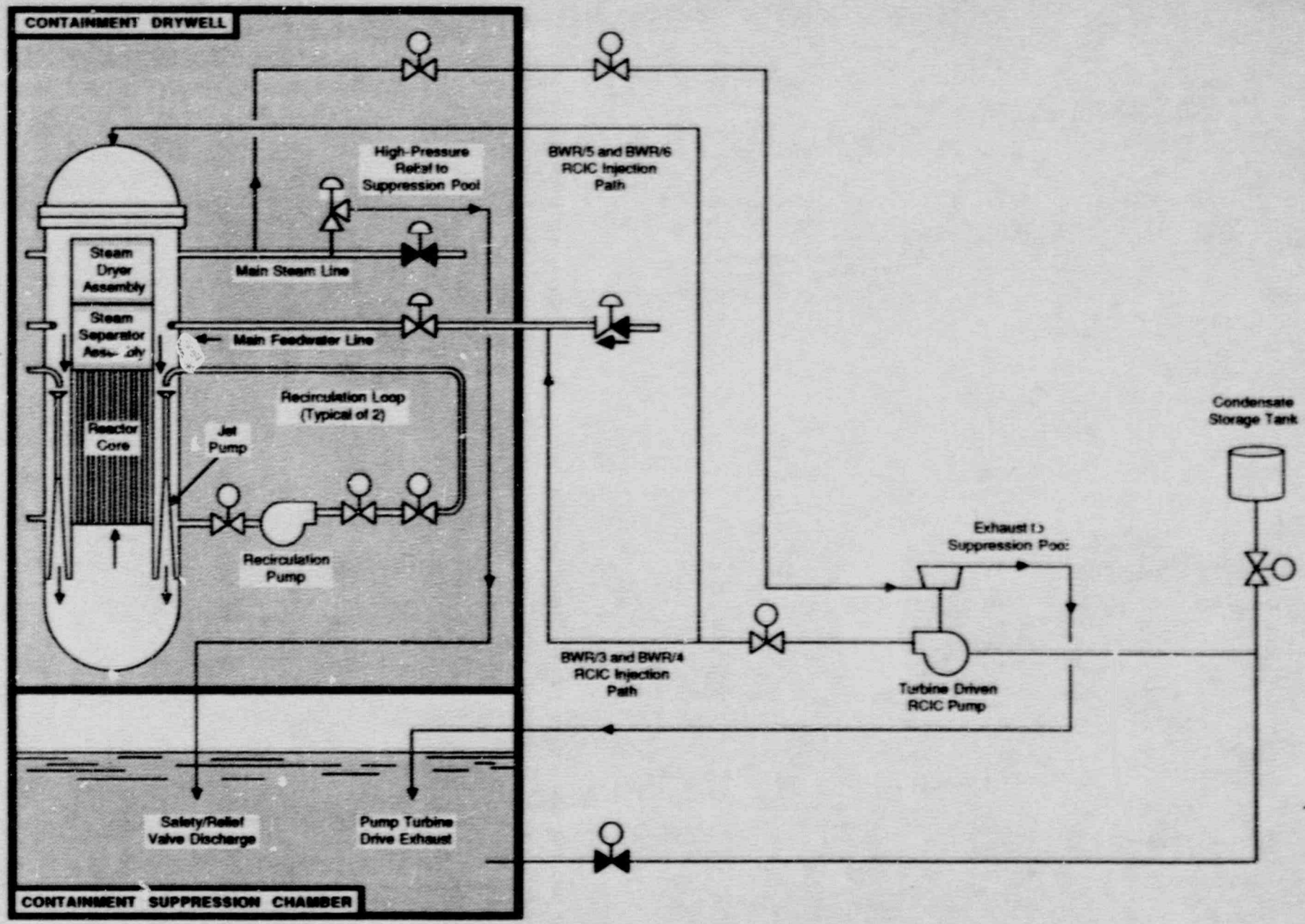


Figure 8.6-2. BWR/4 to BWR/6 Reactor Core Isolation Cooling System Heat Transport Paths

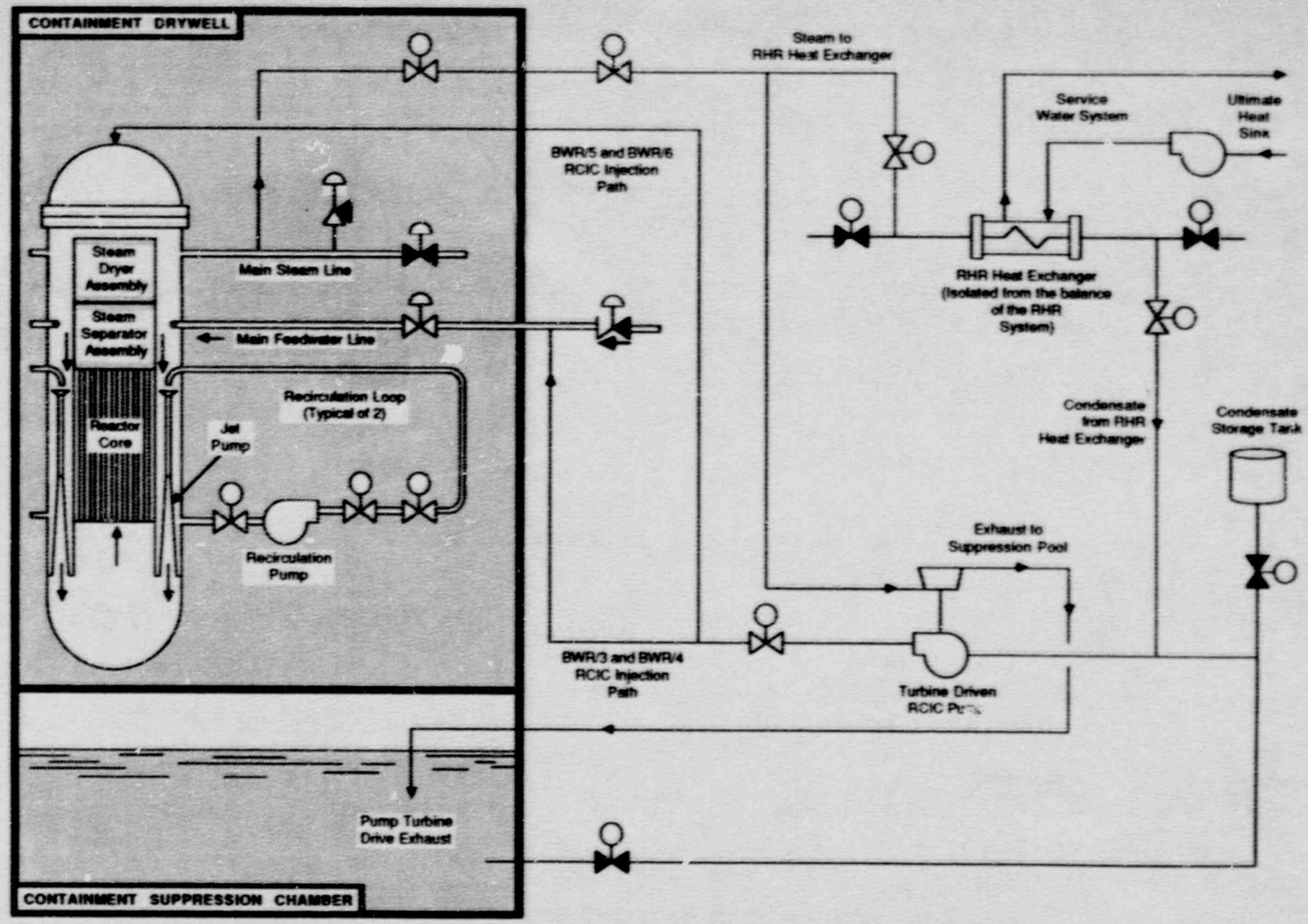


Figure 8.6-3. BWR/4 to BWR/6 Steam Condensing Reactor Core Isolation Cooling System Heat Transport Paths

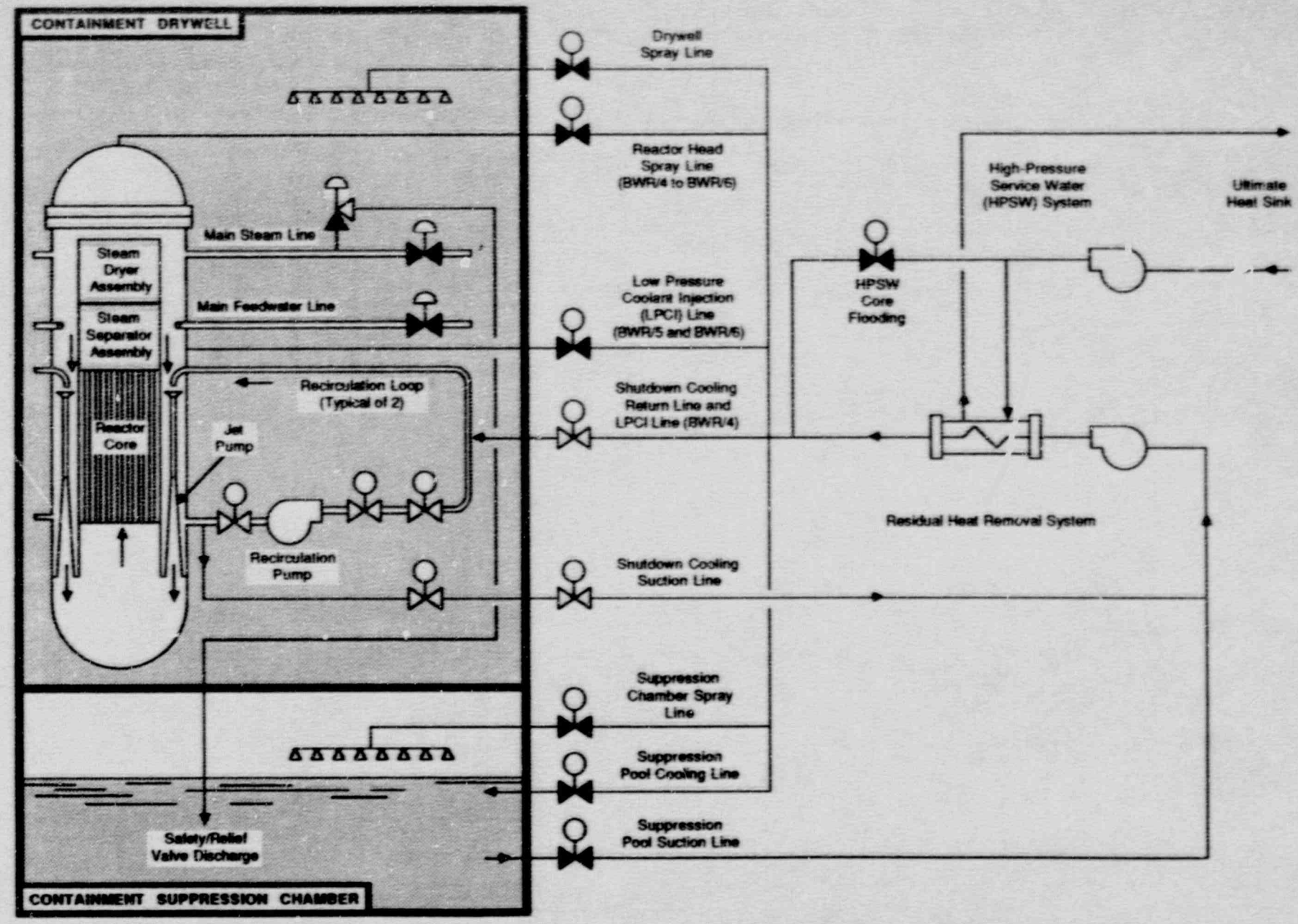


Figure 8.7-1. BWR/4, BWR/5 and BWR/6 Heat Transport Paths Using Multi-Mode RHR System in the Shutdown Cooling Mode

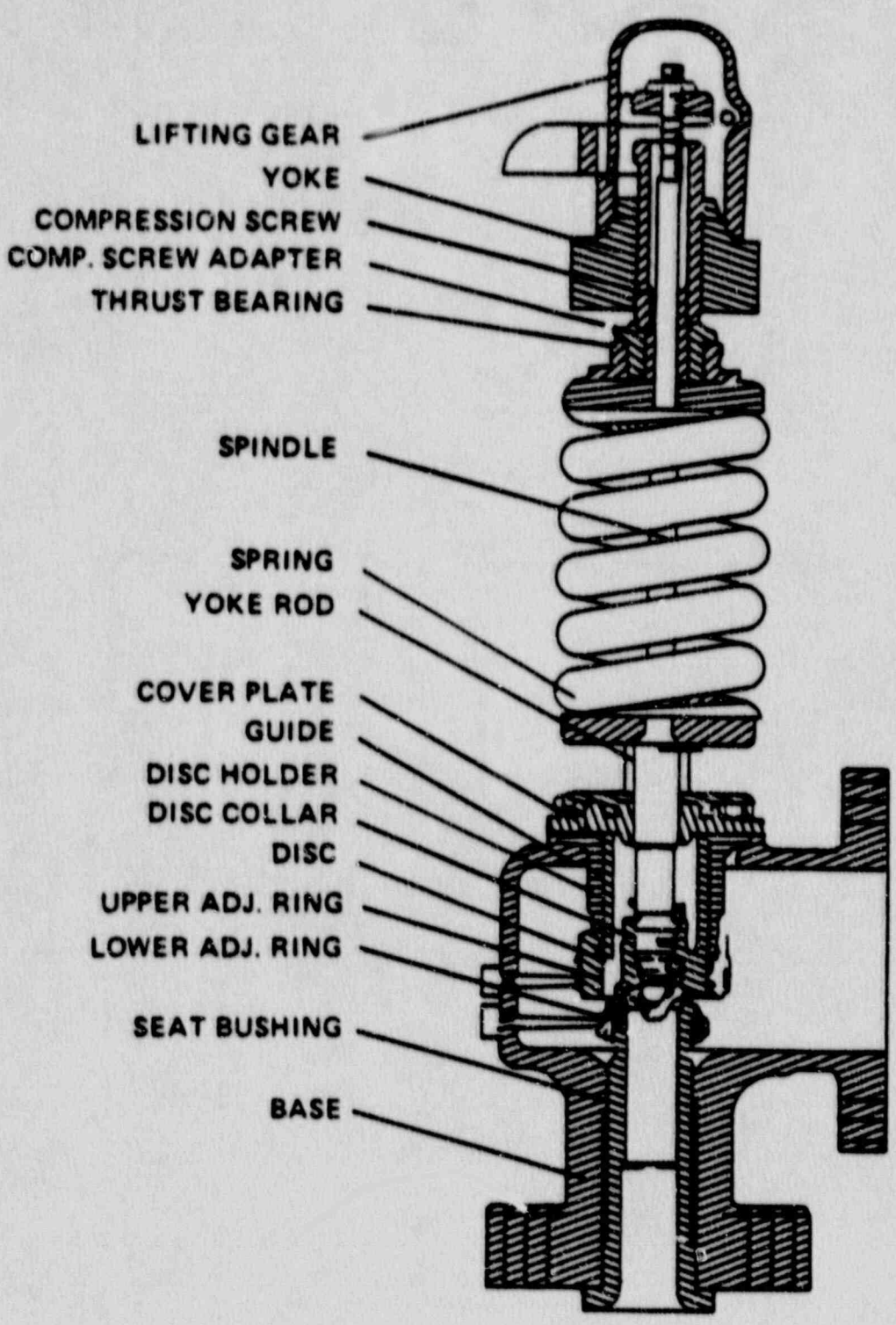


Figure 8.8-1. Typical BWR Safety Valve

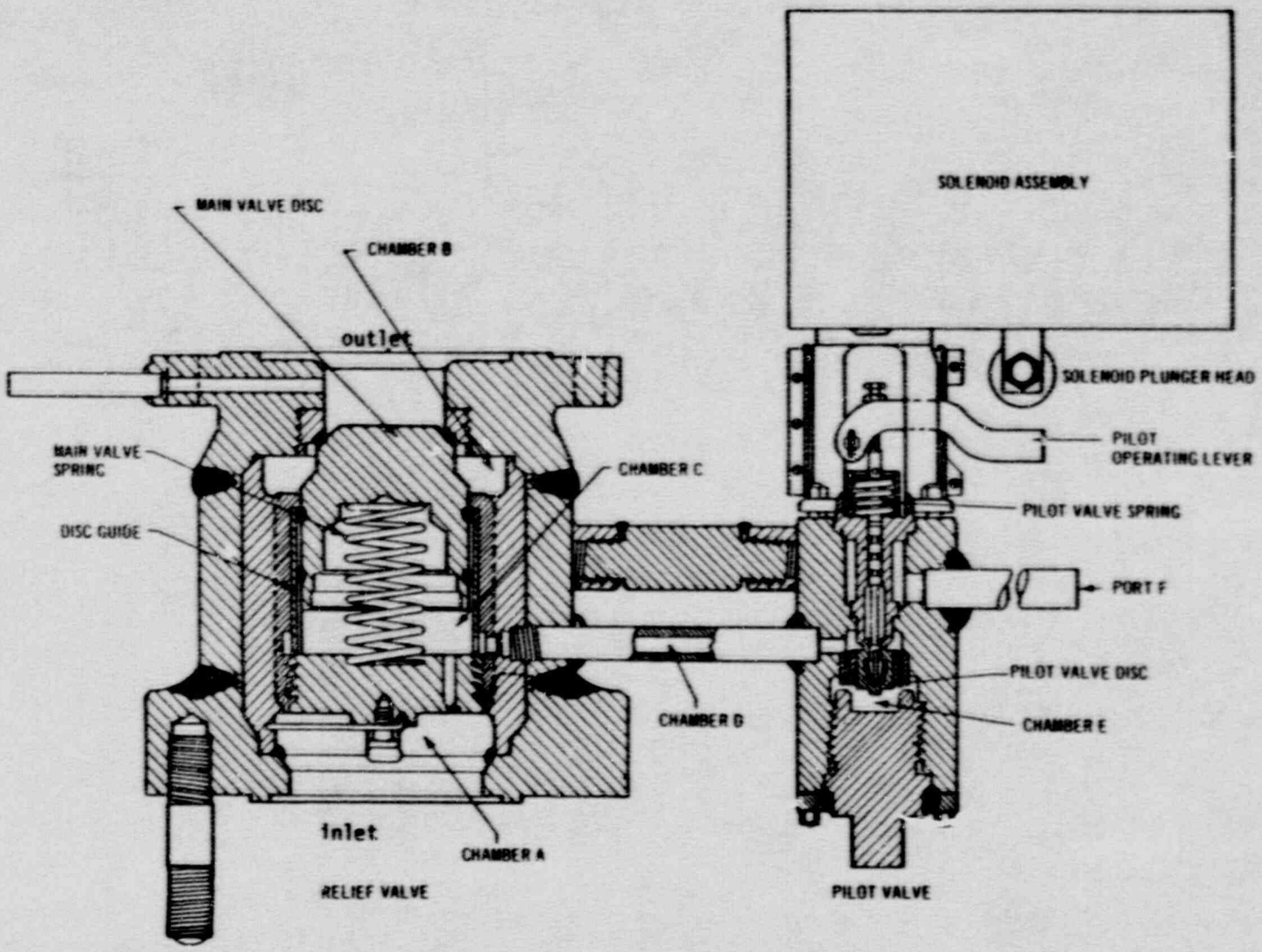


Figure 8.8-2. BWR/2 Power-Actuated Relief Valve (PARV)

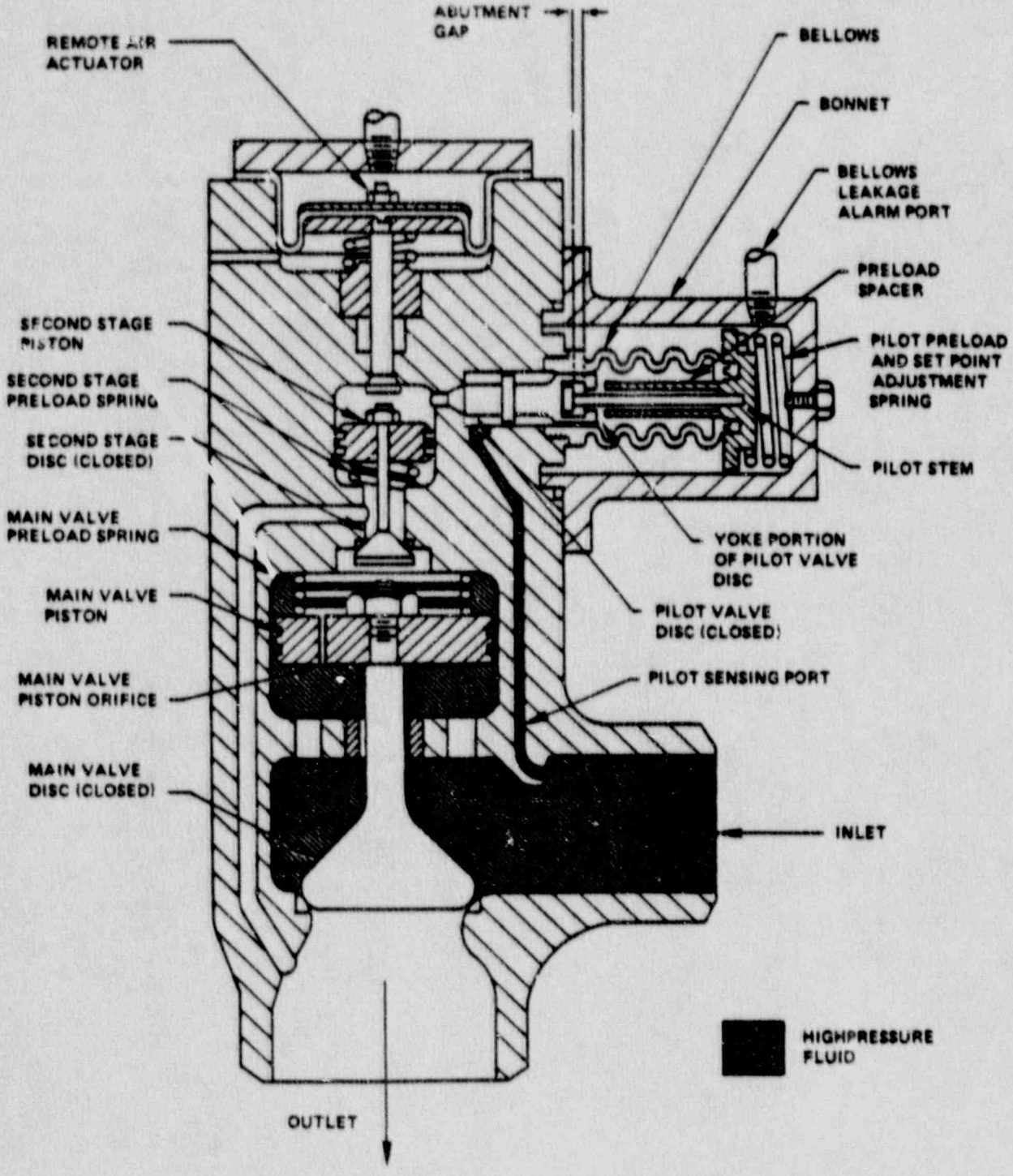


Figure 8.8-3. BWR Three-Stage Pilot-Operated Safety/Relief Valve

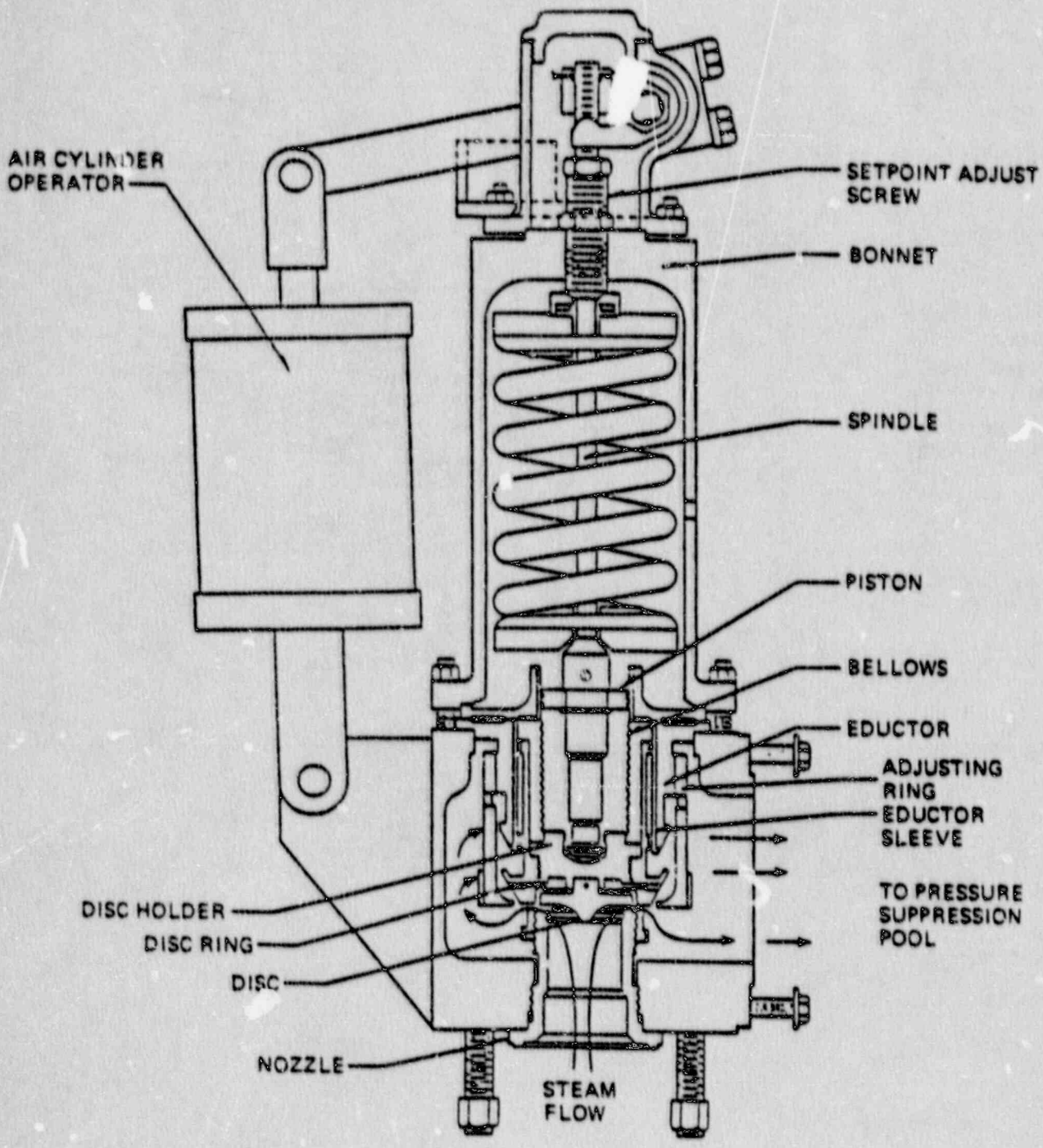


Figure 8.8-4. BWR Crosby Safety/Relief Valve

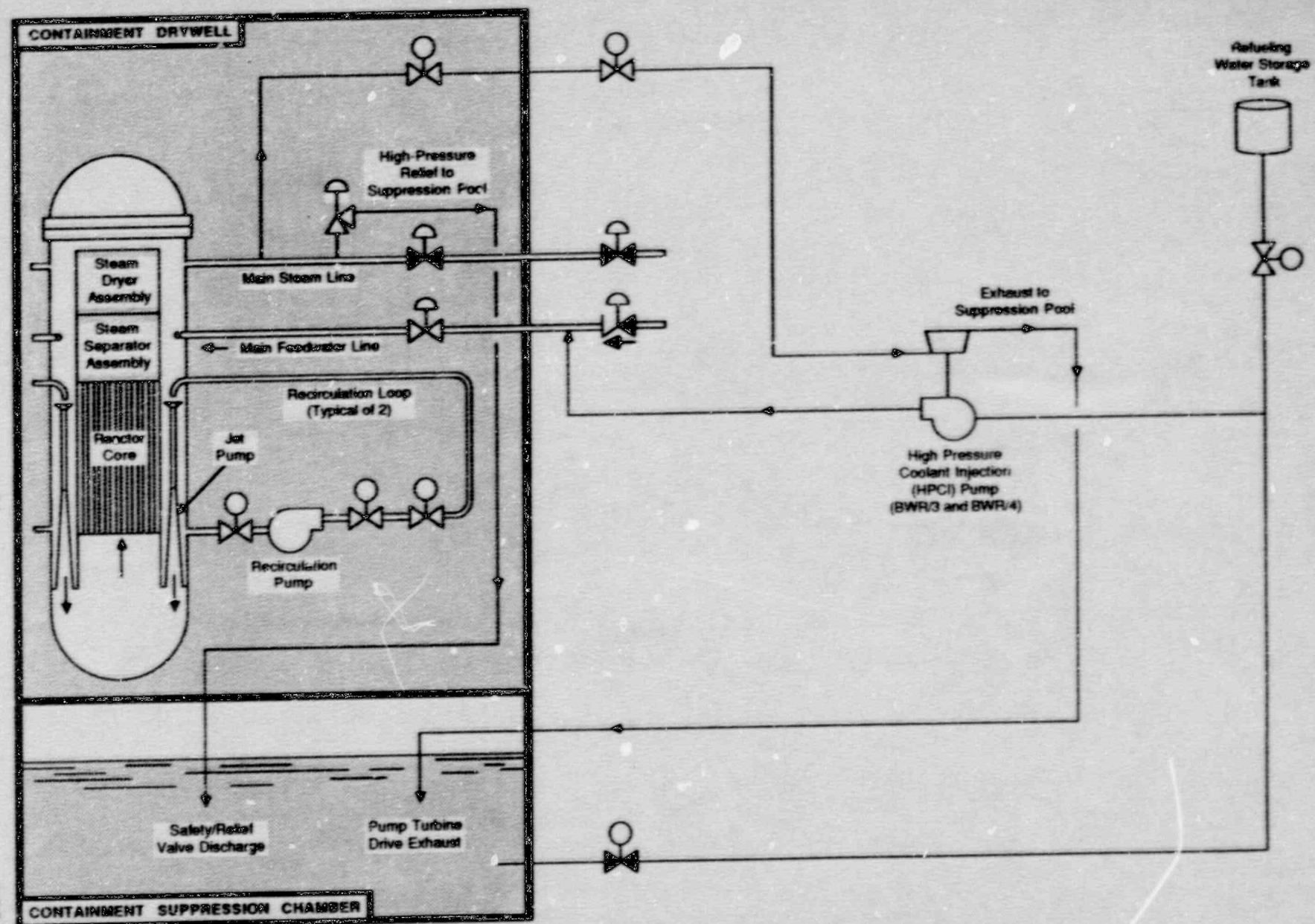


Figure 8.9-1. BWR/3 to BWR/4 High Pressure Coolant Injection (HPCI) Coolant Injection and Heat Transport Paths

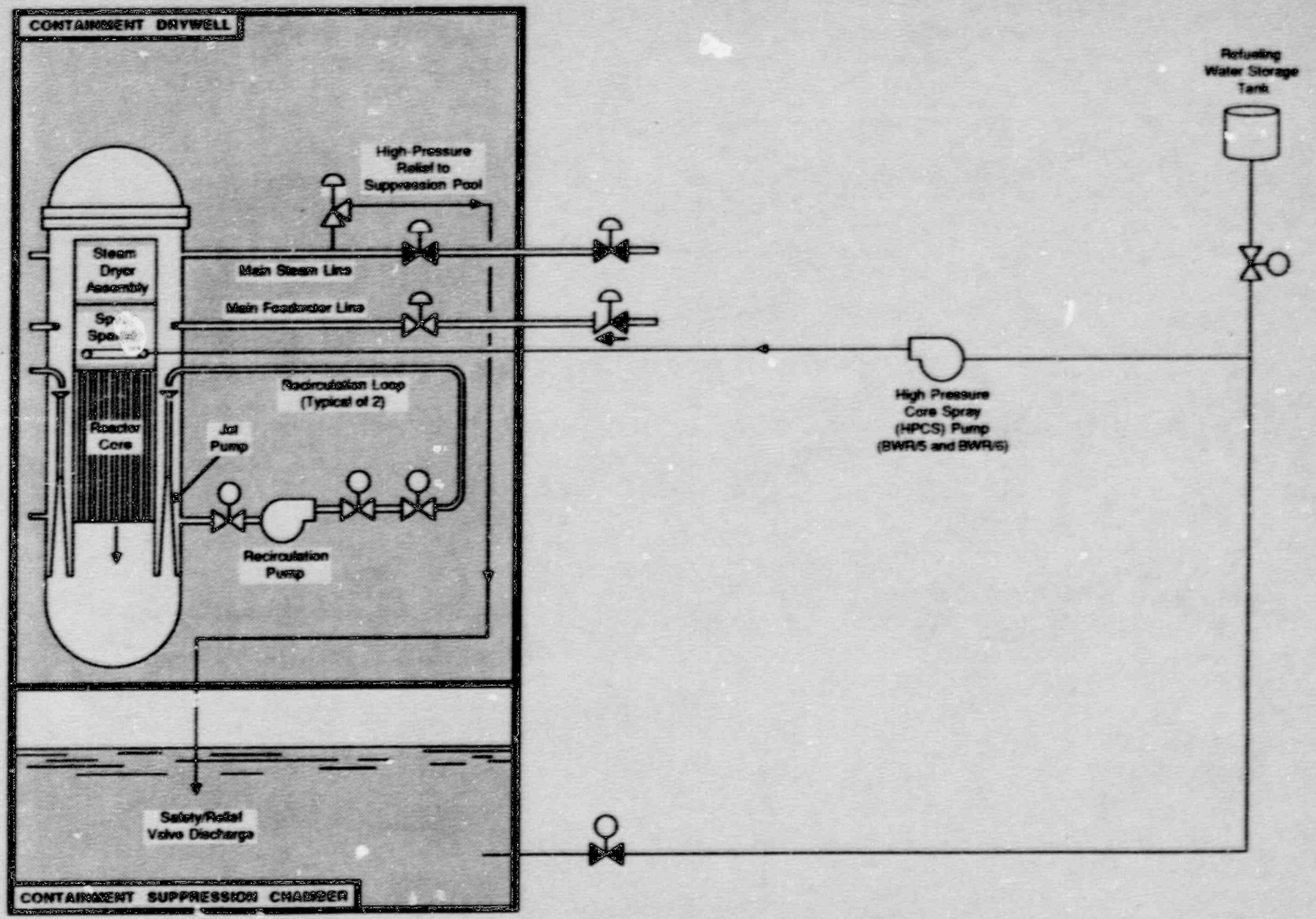


Figure 8.9-2. BWR/5 to BWR/6 High Pressure Coolant Injection (HPCS) Coolant Injection and Heat Transport Paths

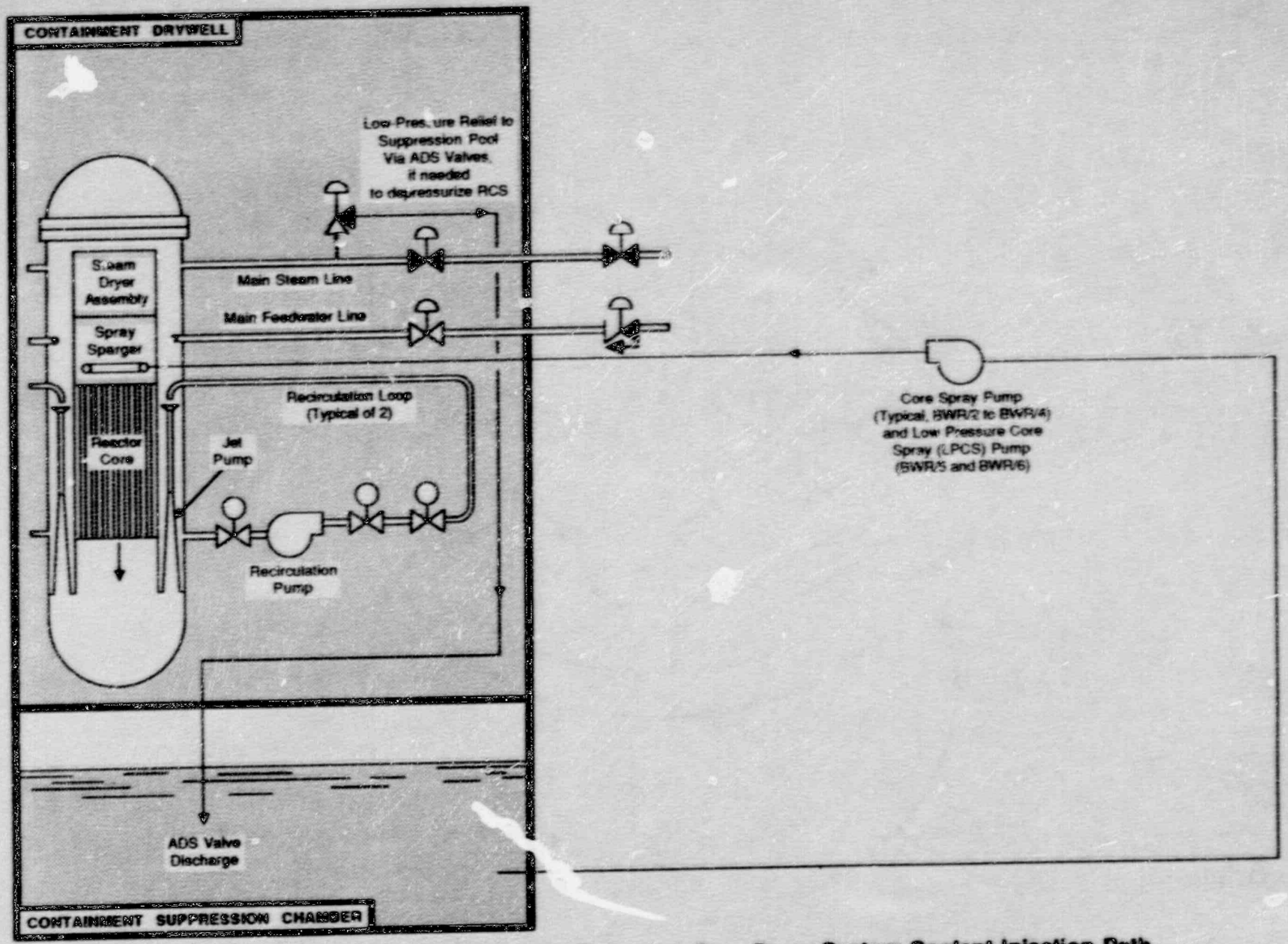


Figure 8.9-3. BWR/2 to BWR/6 (Low Pressure) Core Spray System Coolant Injection Path

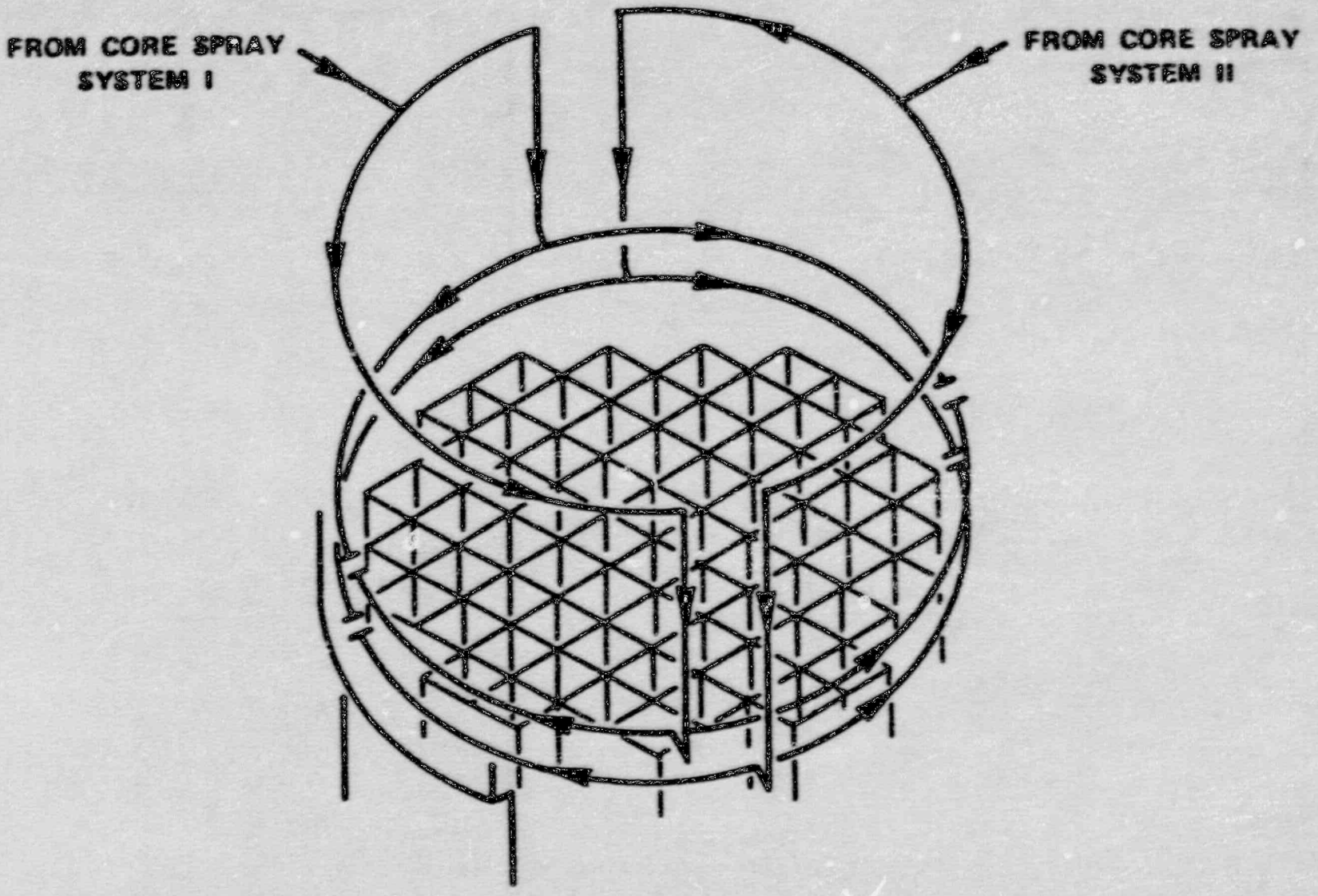


Figure 8.9-4. BWR Core Spray System Sparger Location Above the Reactor Core

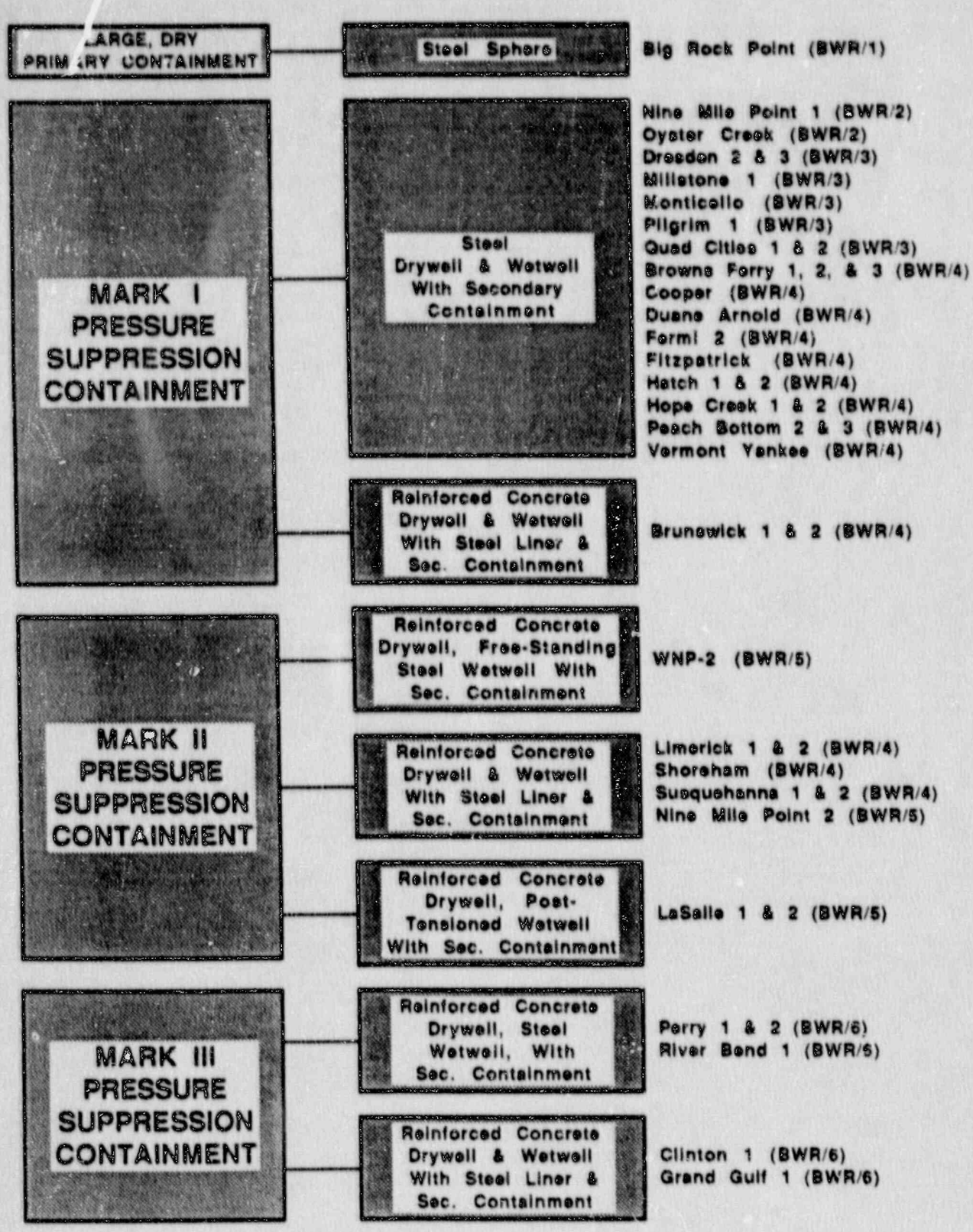


Figure 8.10-1. Distribution of BWR Containments by Model

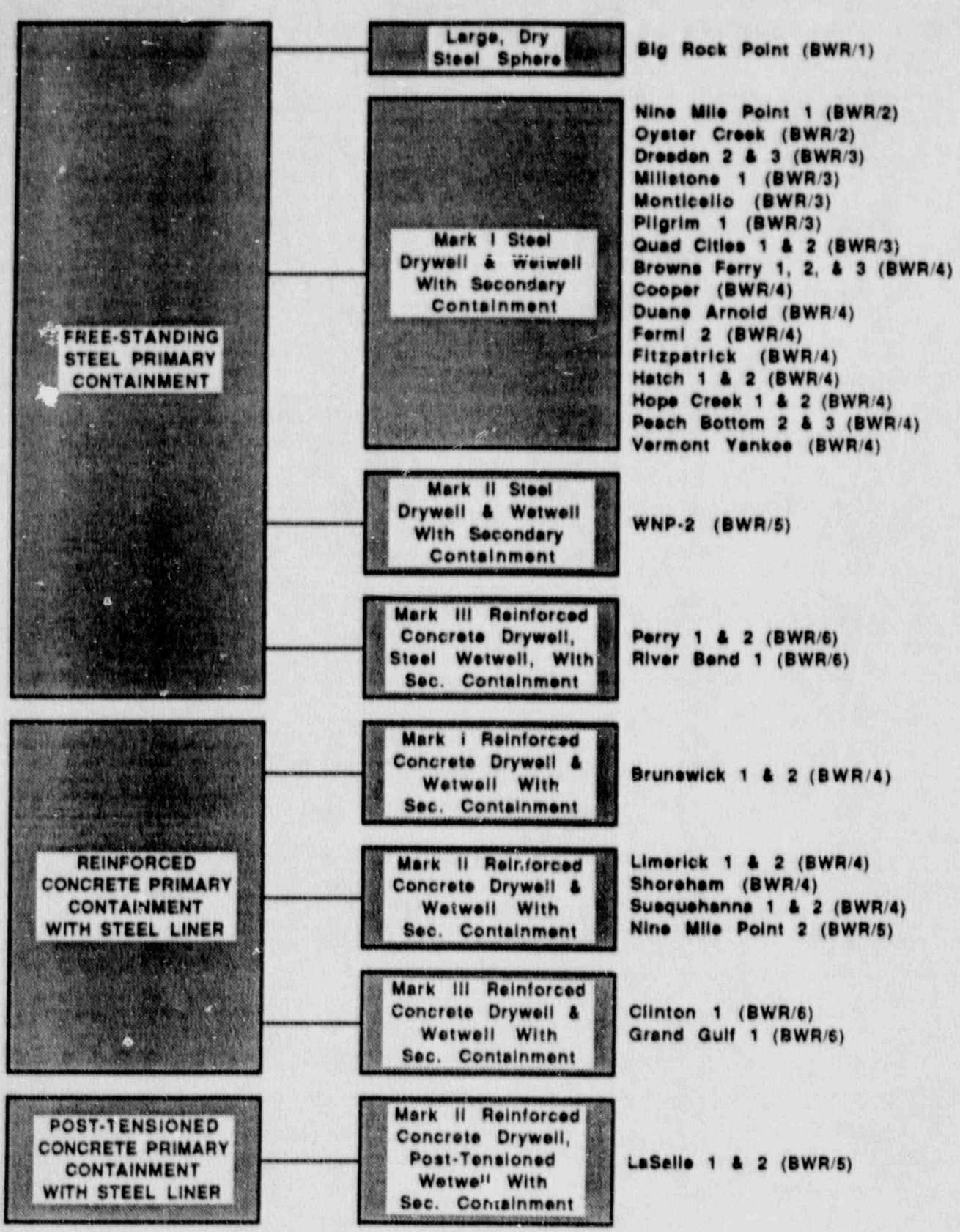


Figure 8.10-2. Distribution of BWR Containments by Construction Type

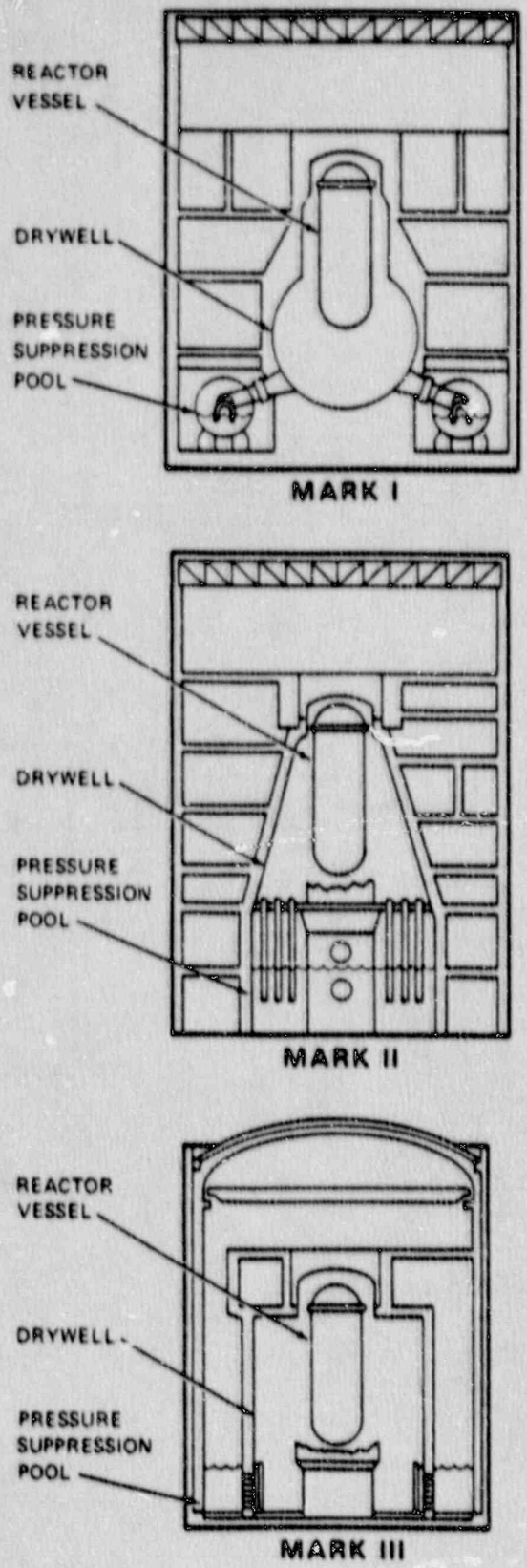


Figure 8.10-3. BWR Pressure Suppression Containments

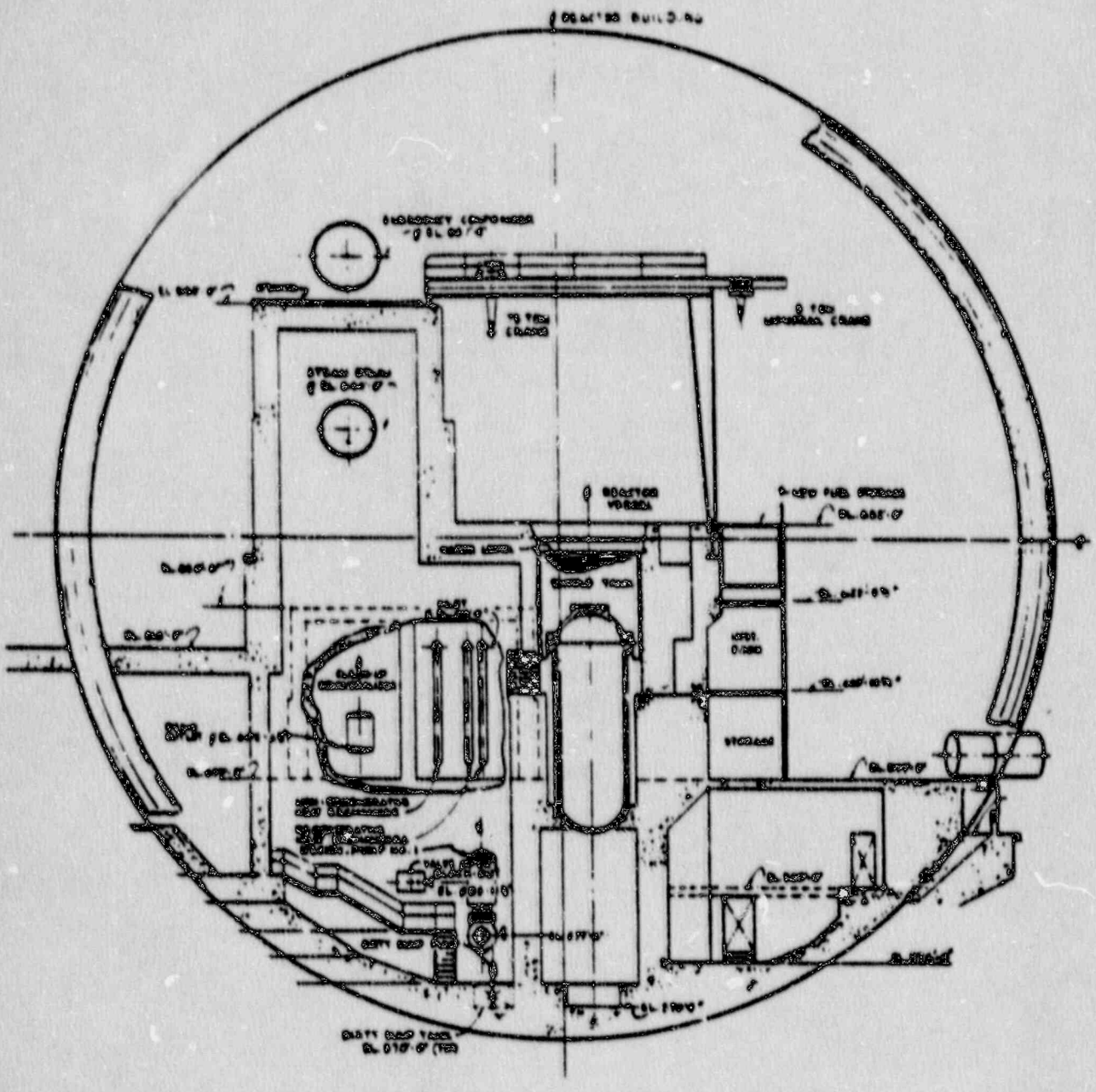


Figure 8.10-4. Big Rock Point Large, Dry Containment (Steel Sphere)

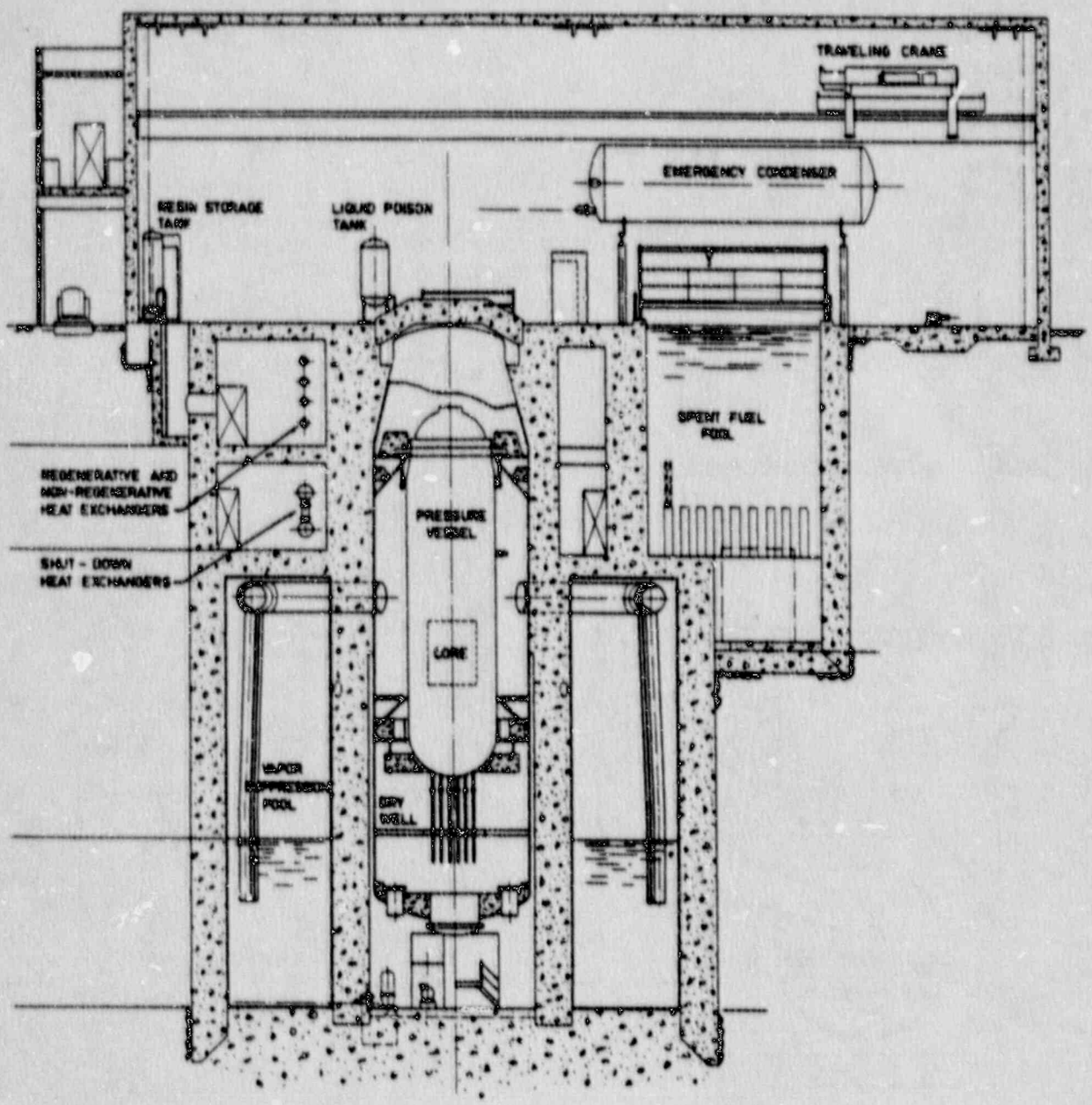


Figure 8.10-5. Humboldt Bay Pre-Mark I Pressure Suppression Containment

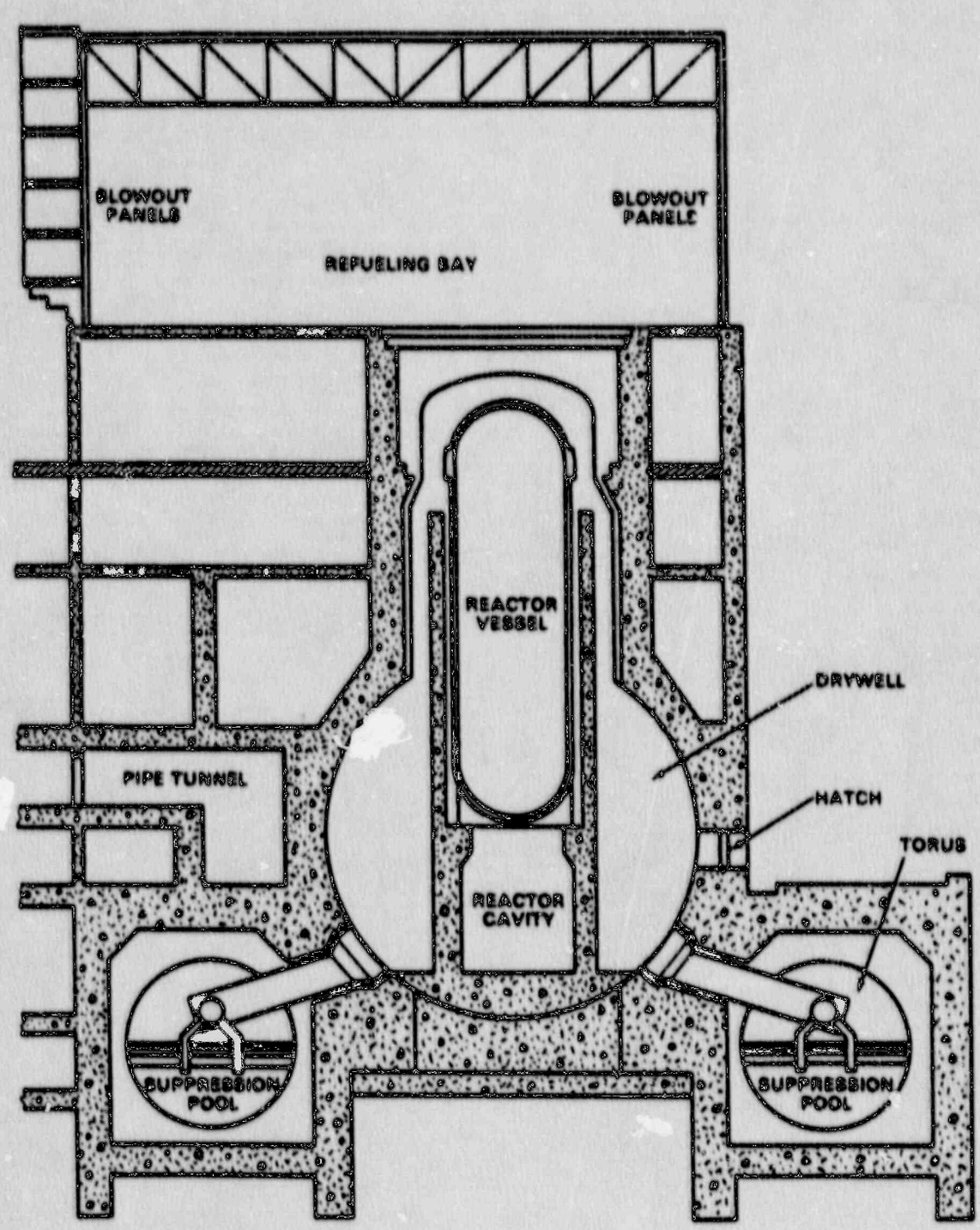


Figure 8.10-6. General Arrangement of a BWR Mark I Containment

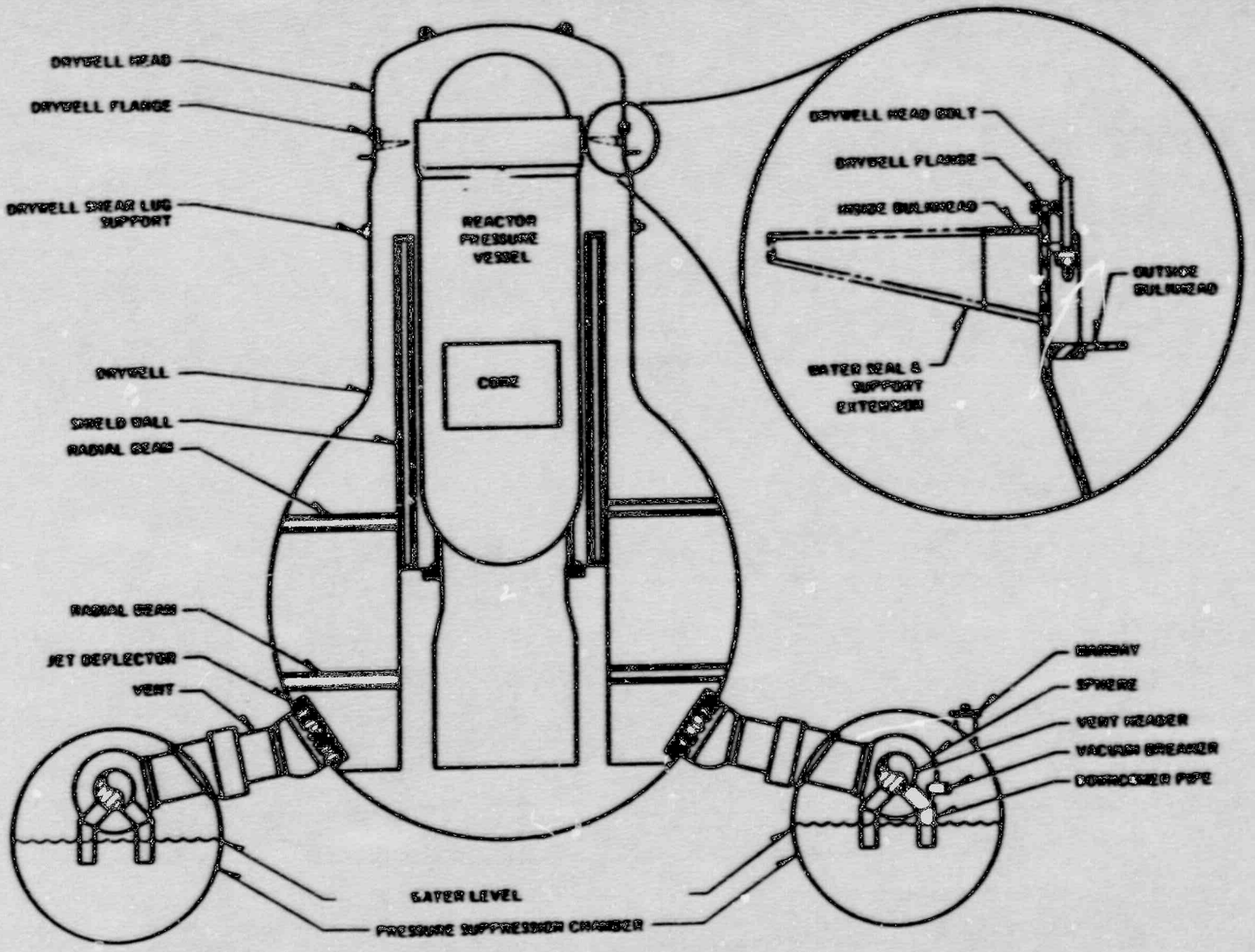


Figure 8.10-7. BWR Mark I Free-Standing Steel Primary Containment

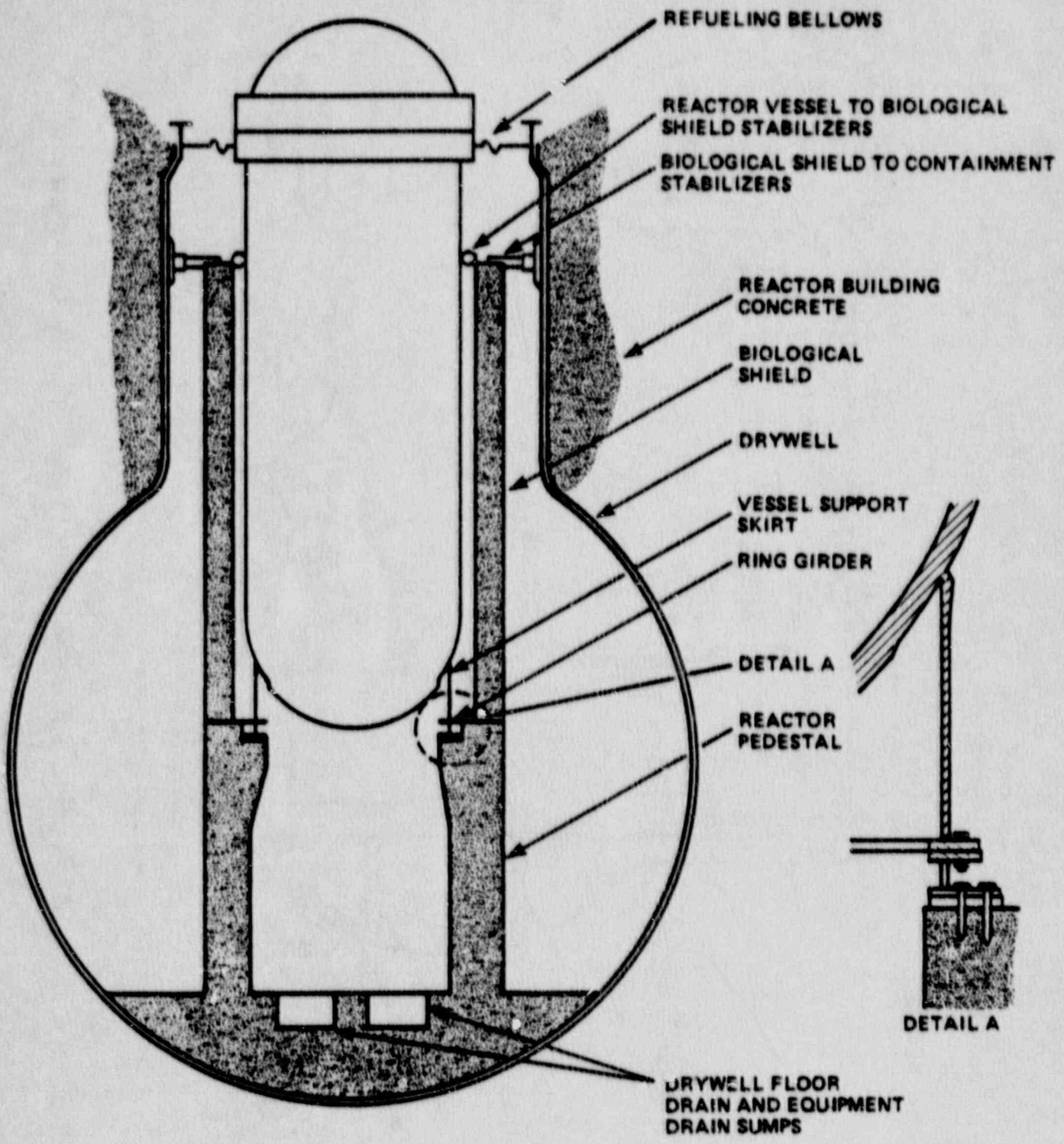


Figure 8.10-8. Details of the Mark I Containment Drywell

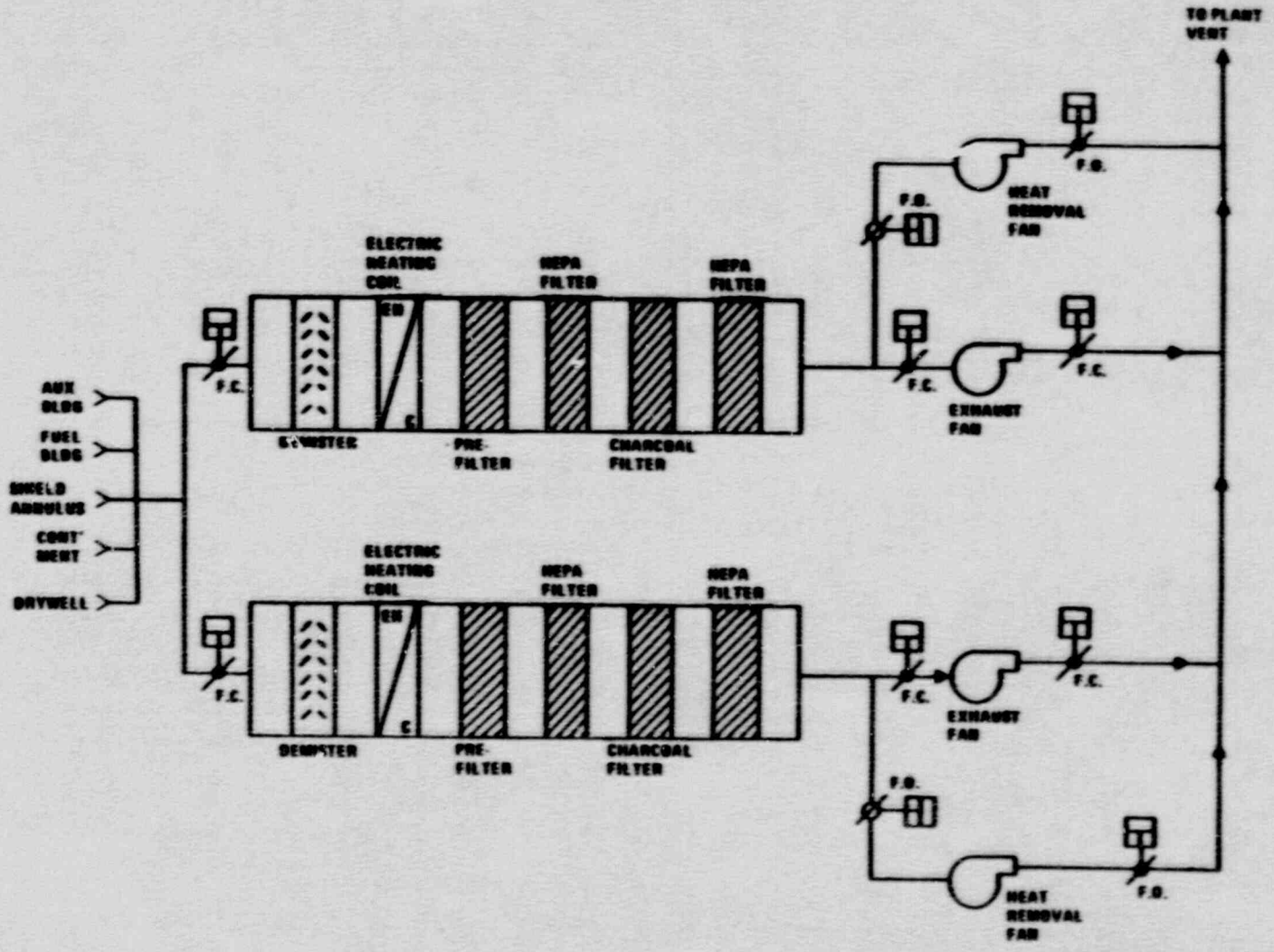


Figure 8.10-9. Typical BWR Standby Gas Treatment System

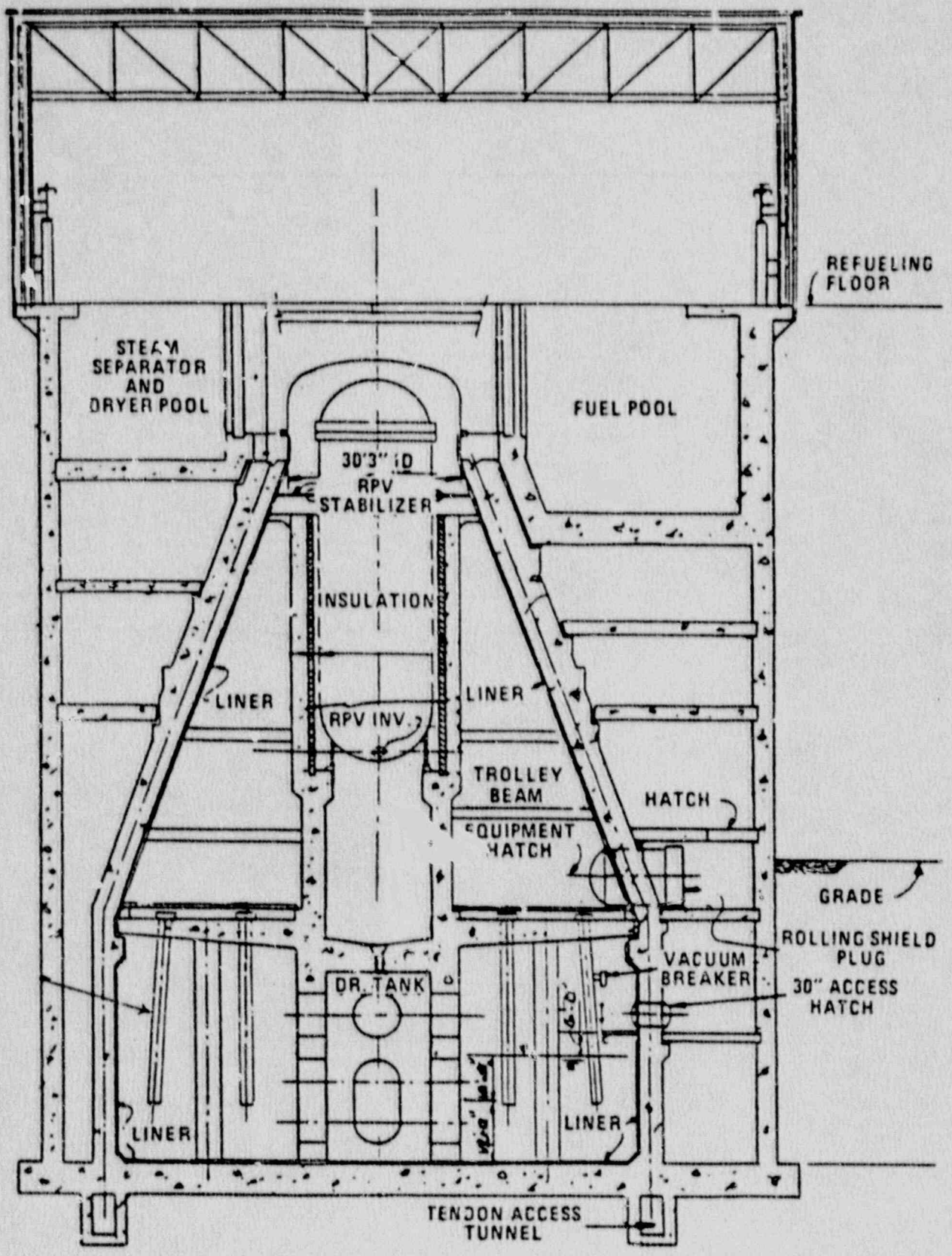


Figure 8.10-10. General Arrangement of a BWR Mark II Containment

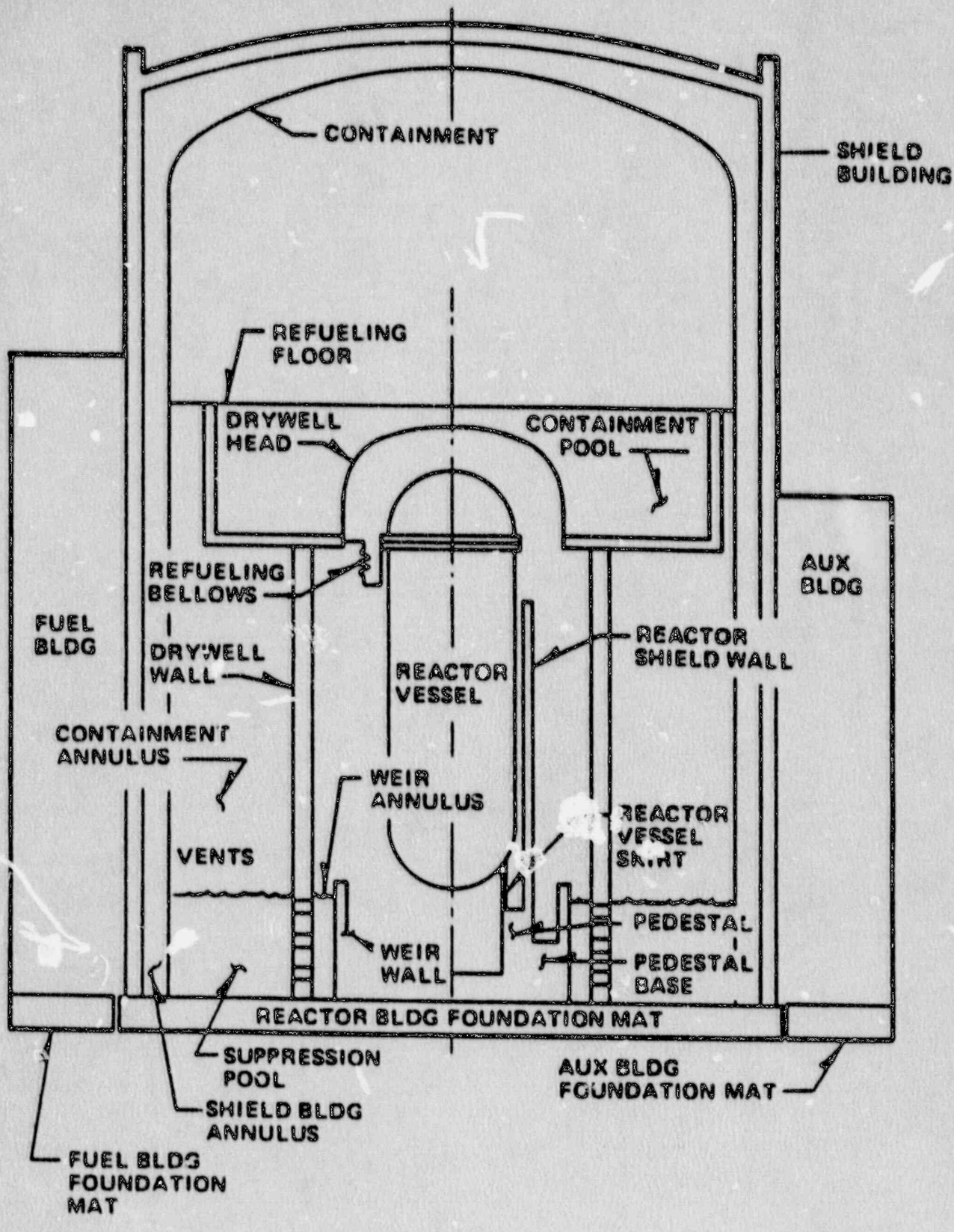


Figure 8.10-11. General Arrangement of a BWR Mark III Containment

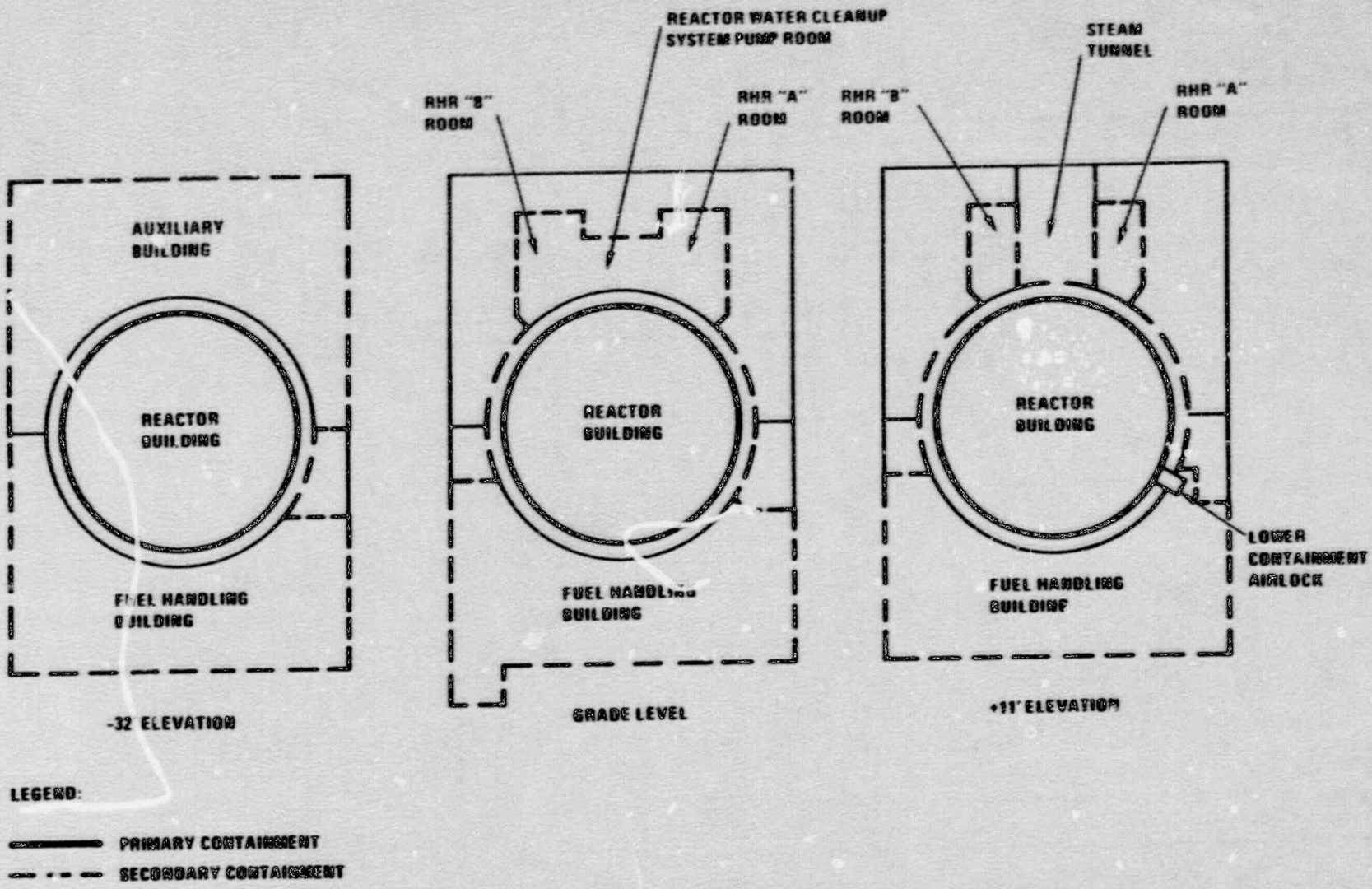


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Figure 8.10-12. Definition of Primary and Secondary Containment Boundaries in a Mark III Containment (Sheet 1 of 2)

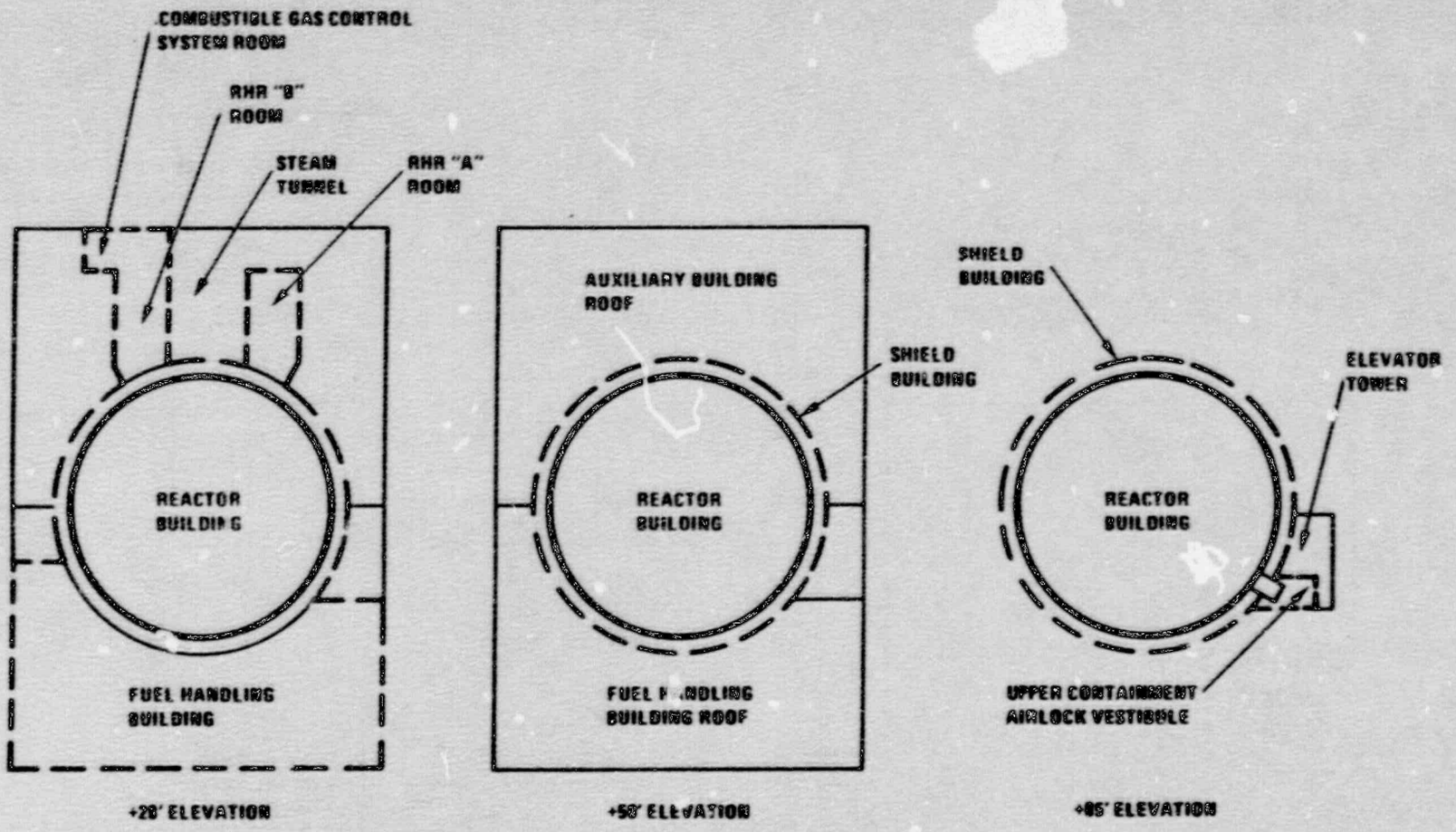


Figure 8.10-12. Definition of Primary and Secondary Containment Boundaries in a Mark III Containment (Sheet 2 of 2)

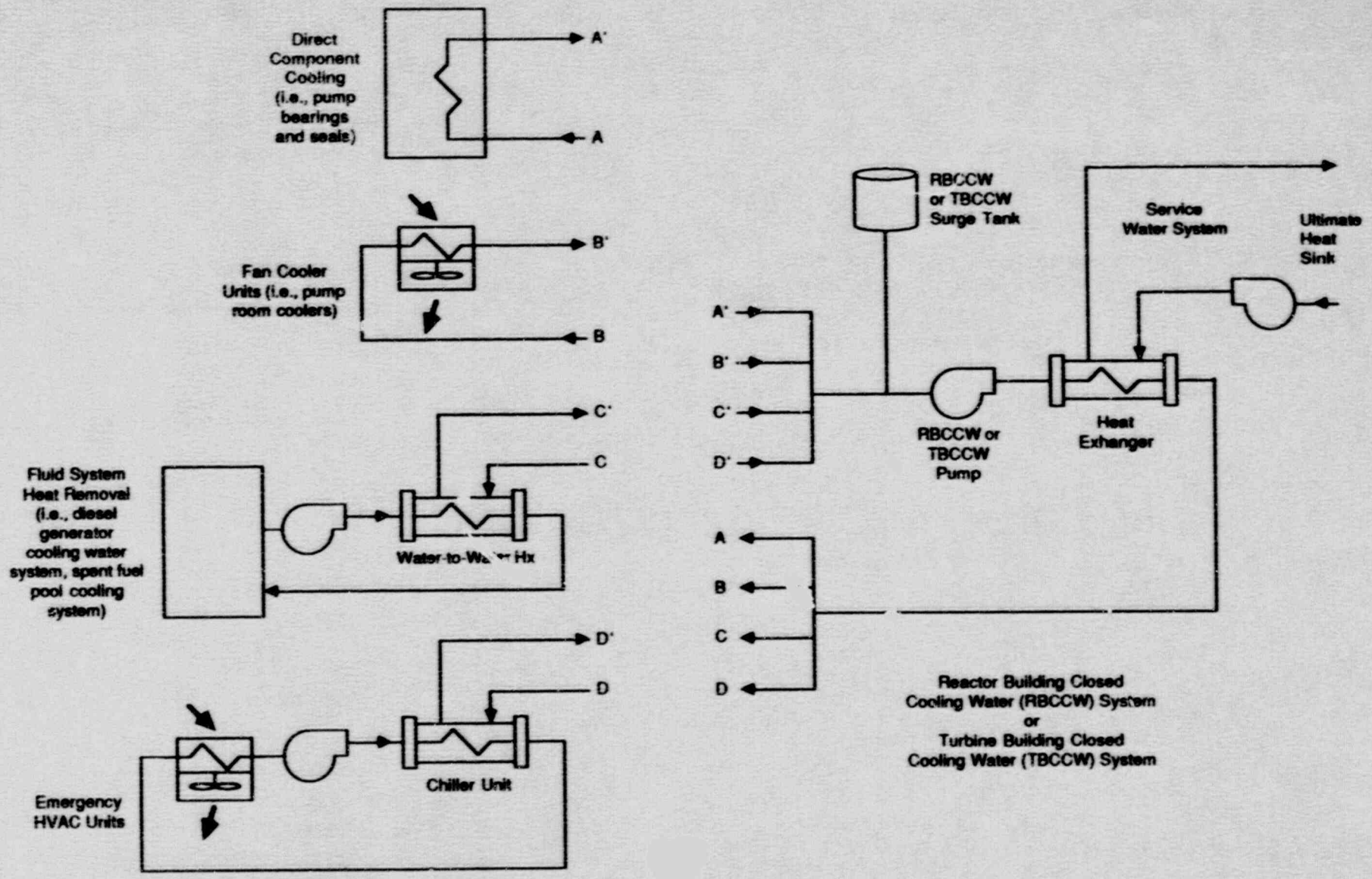
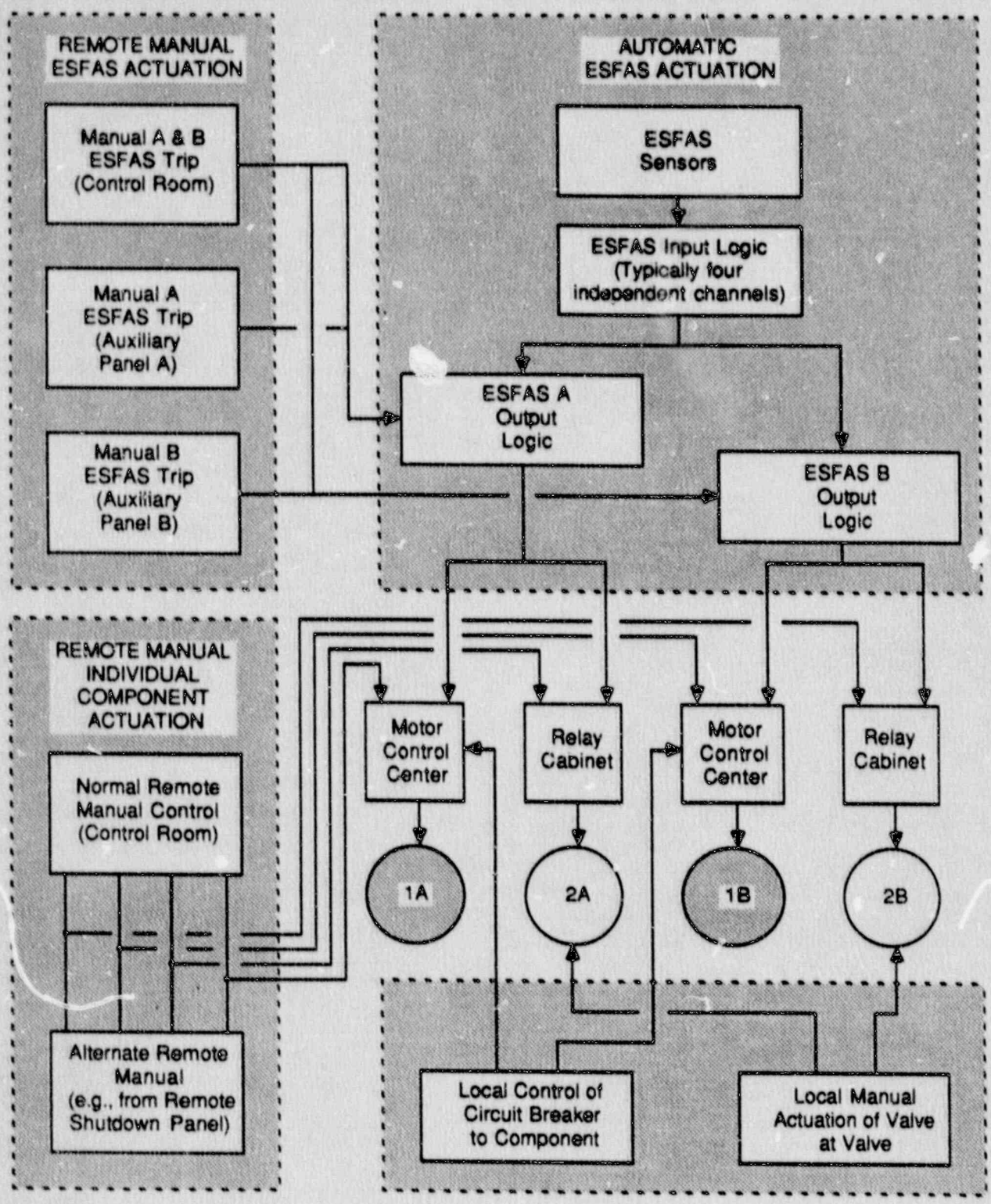


Figure 8.11-1. Examples of Component Cooling System Heat Transport Paths



Note: Components 1A and 1B are typical of motor-driven pumps and motor-operated valves.
Components 2A and 2B are typical of pneumatically or hydraulically-operated valves.

Figure 8.12-1. Example of Actuation System Interfaces

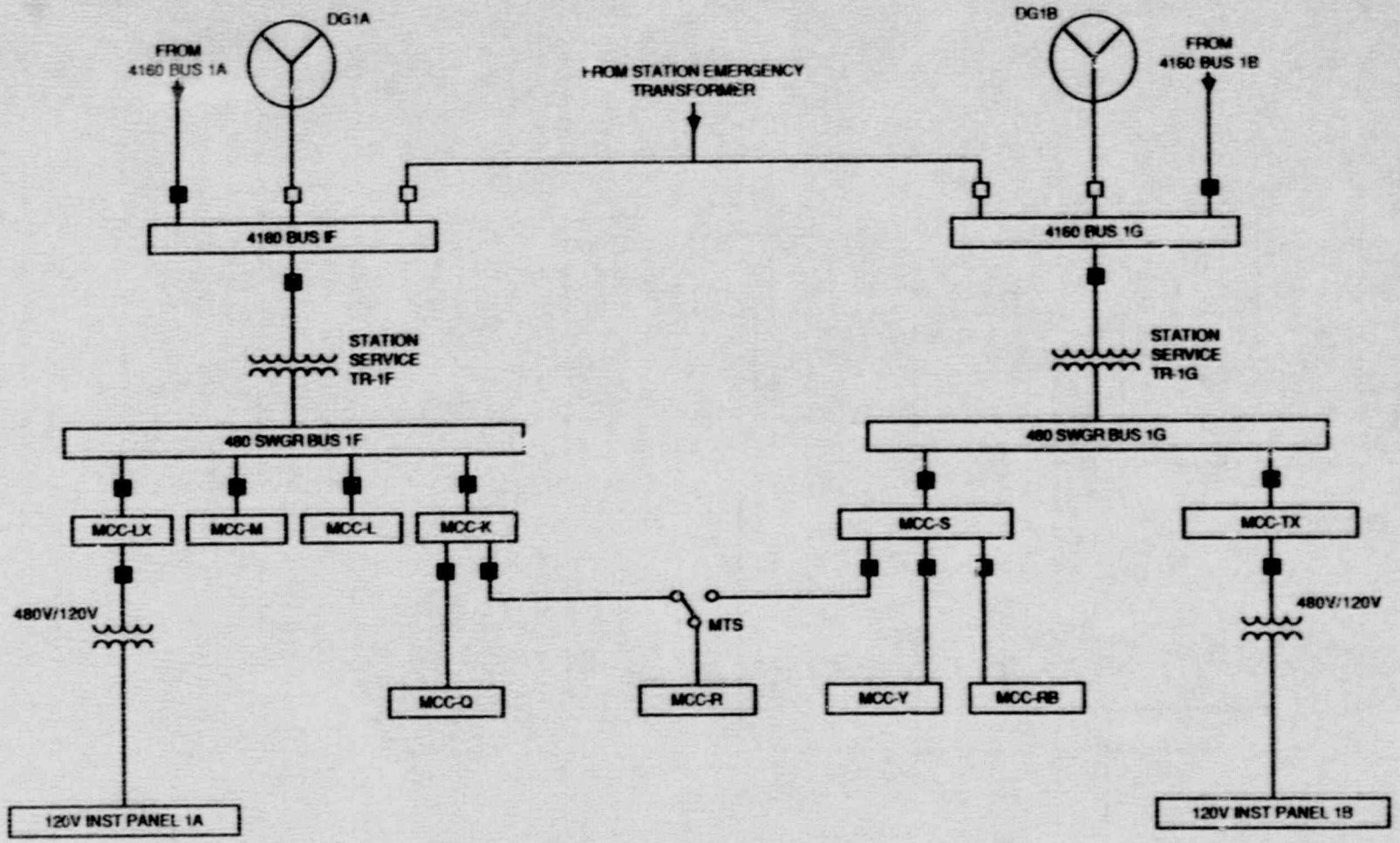


Figure 8.13-1. Representative BWR/4 4160, 480 and 120 VAC Electric Power Distribution System (Cooper)

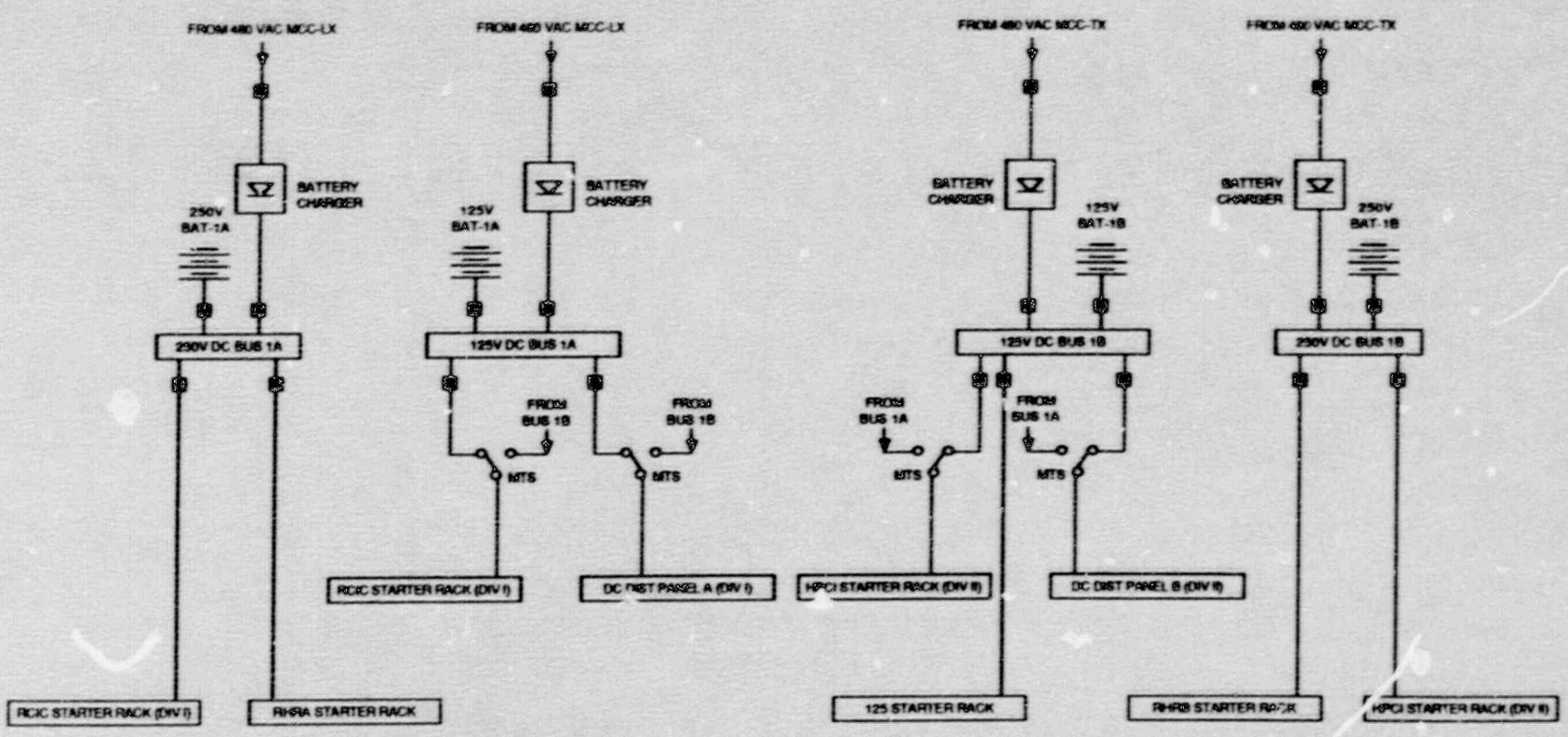


Figure 8.13-2. Representative BWR/4 125 and 250 VDC Electric Power Distribution System (Cooper)

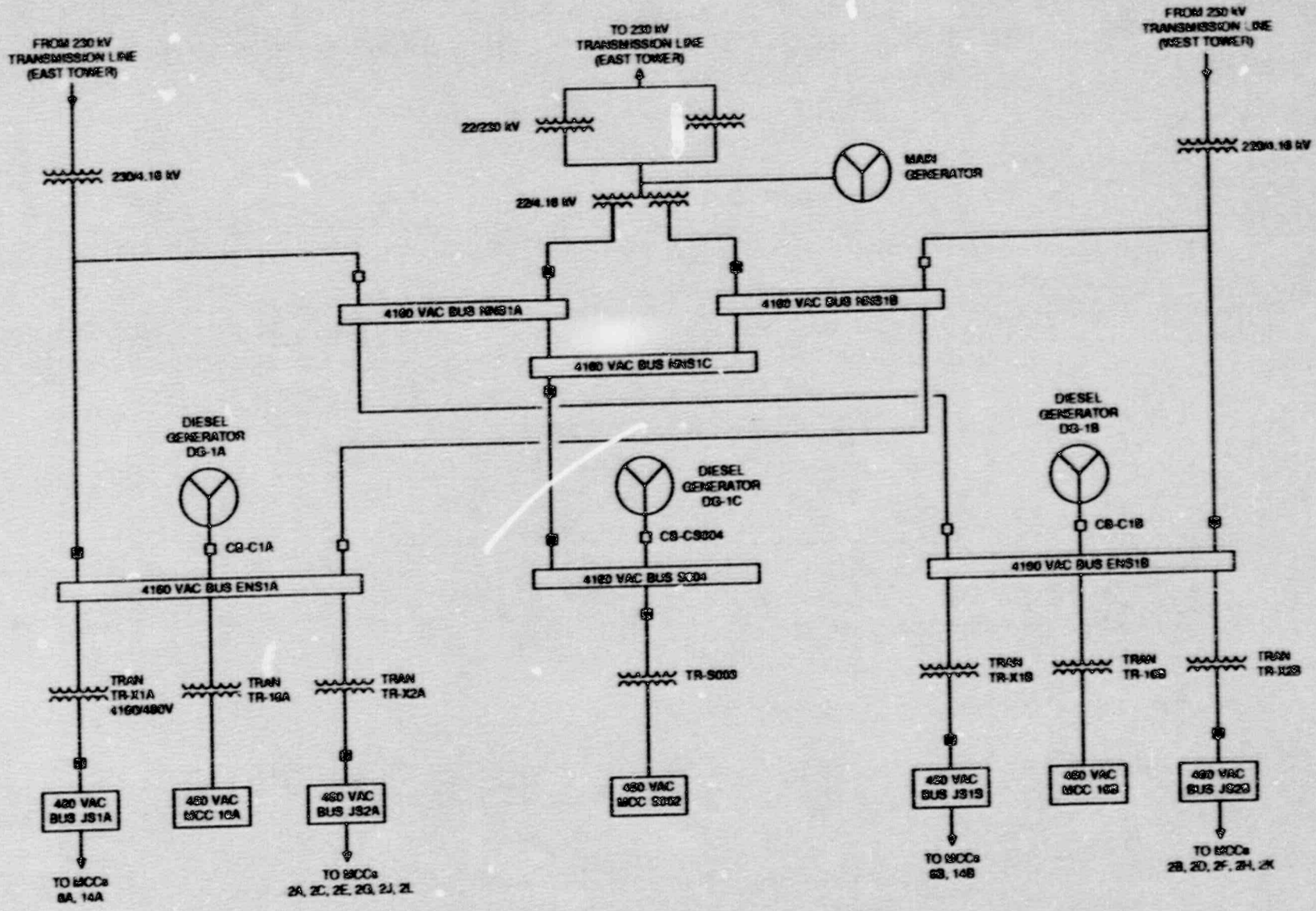


Figure 8.13-3. Representative BWR/6 4160 and 480 VAC Electric Power Distribution System (Liver Bend)

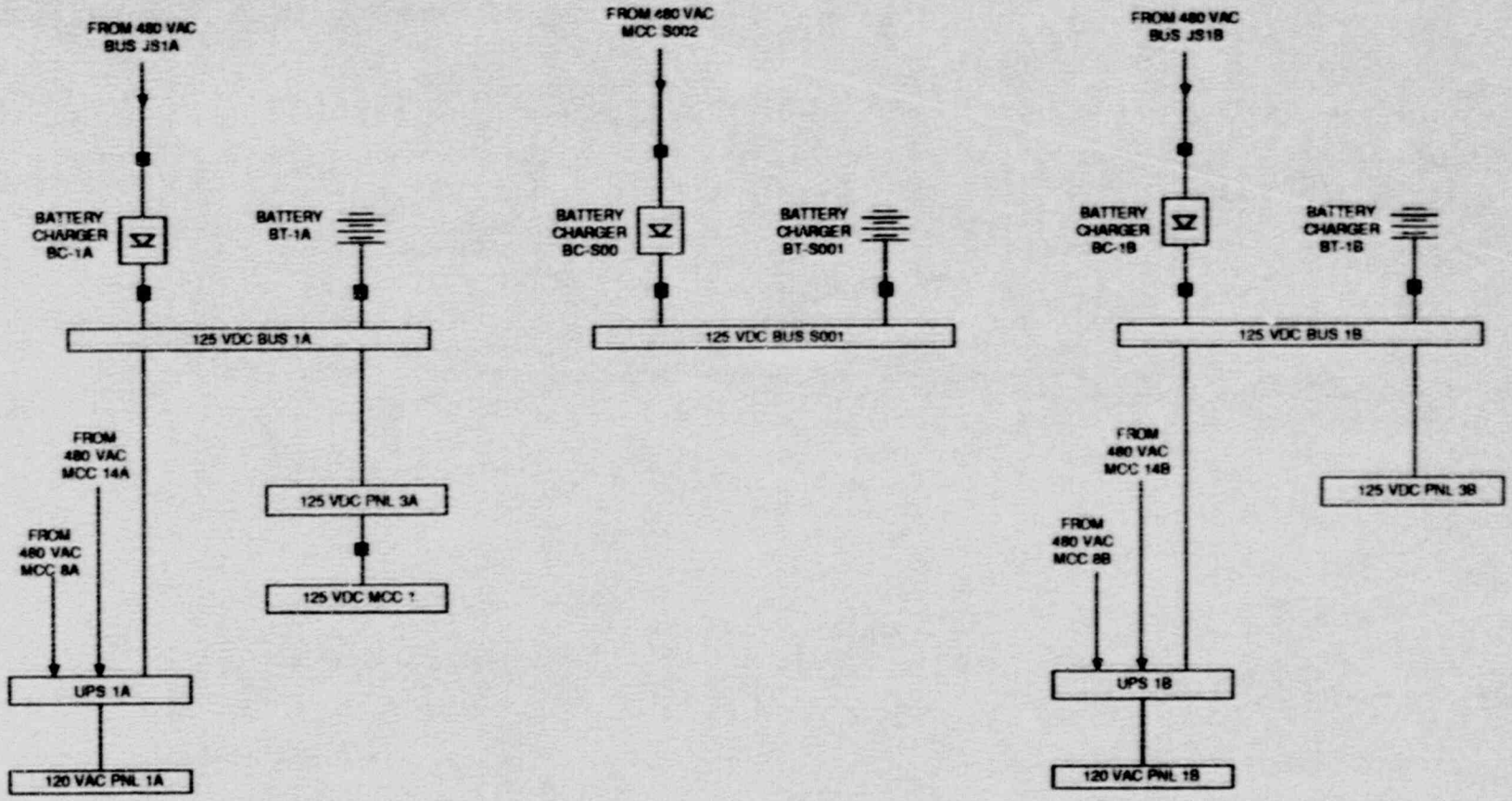


Figure 8.13-4. Representative BWR/6 125 VDC and 120 VAC Electric Power Distribution System (River Bend)

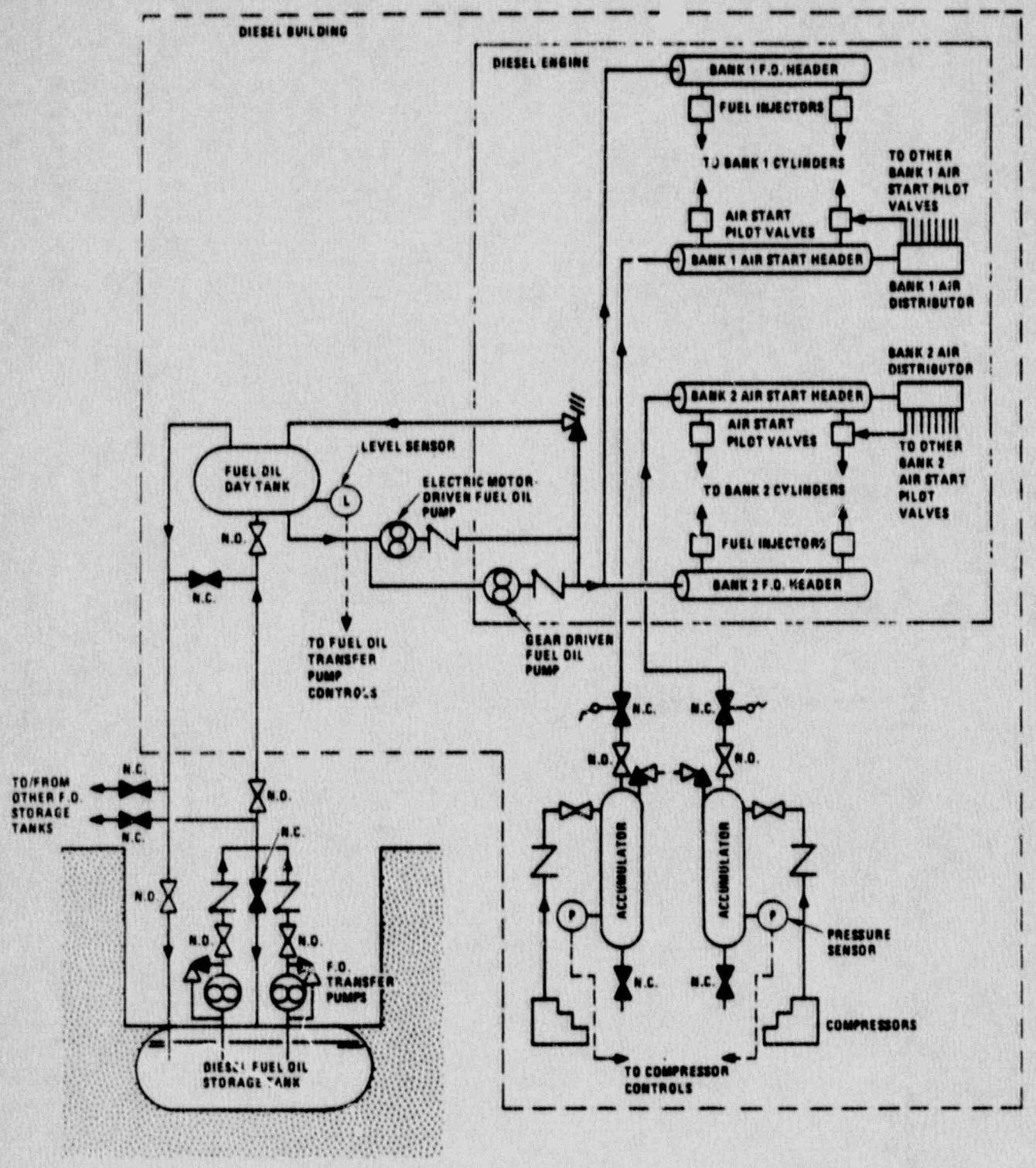


Figure 8.13-5. Diesel Fuel Oil and Air Starting Systems

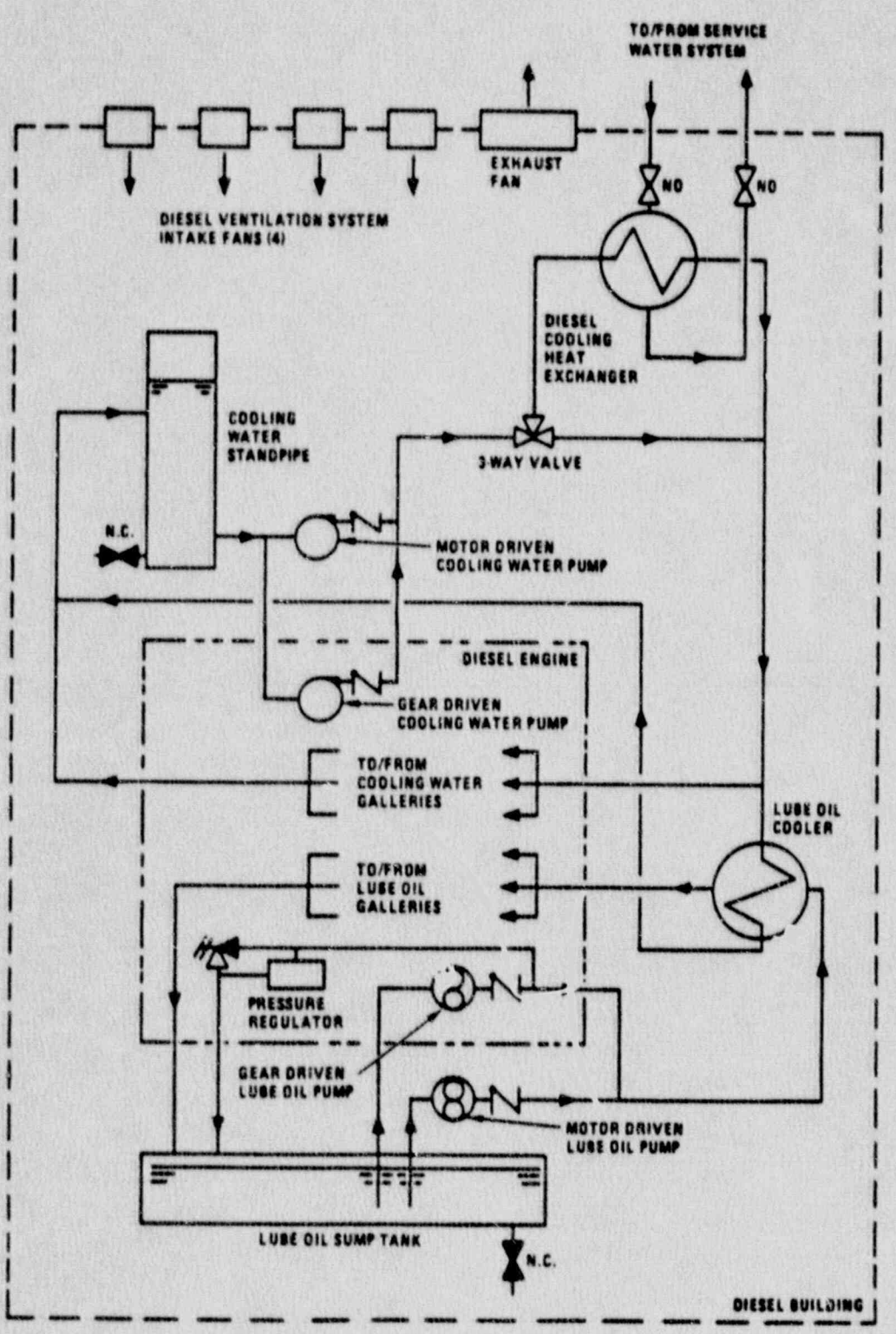


Figure 8.13-6. Diesel Lubrication, Cooling and Room Ventilation Systems

9. GENERAL ELECTRIC (GE) BOILING WATER REACTORS (BWRs)

There are six General Electric BWR "product lines" designated BWR/1 to BWR/6. A comparison of the basic features of these six product lines is found in Table 9-1. As of March 1990, BWRs from all six product lines are operating in the United States. Basic features of General Electric BWRs in each product line are described in detail in this section.

9.1 BWR/1

The three BWR/1 plants in the United States are the following:

- Dresden 1
- Humboldt Bay
- Big Rock Point

Of these reactor plants, only Big Rock Point has a full power operating license as of 2/89. The other two plants were formerly licensed to operate but have been permanently shutdown; Humboldt Bay in 1976, and Dresden 1 in 1978. International reactors included in the BWR/1 vintage include Dodeward (Netherlands), KRB (Federal Republic of Germany), Tarapur (India), and Garigliano (Italy). The descriptions in this section will focus on Big Rock Point.

9.1.1 Reactor Core and Reactivity Control Systems

The Big Rock Point core is rated at 240 MWt, has a power density of 46 kW/liter, and is comprised of 84 fuel assemblies. A power density of about 60 kW/liter was achieved in a special test with a reduced size core.

9.1.2 Reactor Coolant System, Including the Reactor Recirculation System

A variety of reactor coolant system designs are included among the exploratory BWRs classed as BWR/1s. Included are the following:

- Natural circulation dual-cycle plants with core power densities typically less than 40 kW/liter (i.e., Dresden 1)
- Natural circulation direct-cycle plants with power densities less than 25 kW/liter (i.e., Humboldt Bay)
- A forced circulation direct-cycle plant with a power density of 46 kW/liter (i.e., Big Rock Point)

All BWR/1 plants are "non-jet pump" plants. Big Rock Point was the only BWR/1 plant with forced recirculation flow by means of two external recirculation pump loops that took a suction on the steam drum and discharged to the lower plenum of the reactor vessel as shown in Figure 9.1-1. Details of the Big Rock Point reactor vessel in Figure 9.1-2 show that there are no steam spearators or dryers inside the reactor vessel and the recirculation line inlet to the reactor vessel is below the level of the core. Use of the external recirculation loops allowed Big Rock Point to operate at higher core power densities than could be achieved with the natural circulation plants.

9.1.3 Reactor Vessel Level Instrumentation

Reactor vessel water level measurements in BWR/1 to BWR/3, and most BWR/4 plants are made by temperature compensated differential pressure instruments that have a heated reference column. This type of instrument is described in Section 8.

9.1.4 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

Primary system overpressure protection at Big Rock Point is provided by six safety valves located on the steam drum. A four-train Reactor Depressurization System (RDS) connected to the steam drum performs the function of the ADS in later-model BWRs. The RDS uses solenoid-operated pressure relief valves.

9.1.5 Shutdown Cooling Systems

The shutdown cooling capability at Big Rock Point is comprised of an isolation condenser system for high-pressure decay heat removal and a low-pressure shutdown cooling system. The isolation condenser receives and condenses primary steam from the steam drum, and returns the condensate to the steam drum. The dedicated shutdown cooling (RHR) system takes a suction on the reactor vessel and returns coolant directly to the reactor vessel.

9.1.6 Emergency Core Cooling Systems

The emergency core cooling function at Big Rock Point is accomplished by the following:

- RCS heat removal at high pressure by the isolation condenser system (which provides no primary makeup capability).
- Reactor Depressurization System (RDS) consisting of four trains of power-operated relief valves to depressurize the RCS.
- The fire system pumps to accomplish the emergency coolant injection function
- The core spray system to accomplish the emergency coolant recirculation function (i.e. recirculating water from the containment sump back to the reactor vessel).

9.1.7 Containment

The Big Rock Point primary system is housed in a dry, spherical, steel containment. Dresden 1 had a similar containment, while Humboldt Bay was the first U.S. BWR with a pressure suppression containment (a pre-Mark I containment). The Big Rock Point containment is shown in Figures 9.1-3 and 9.1-5. The locations of the steam drum and the emergency (isolation) condenser can be seen in Figure 9.1-3. Note also that the spent fuel storage pool is inside containment.

Table 9.1-1. General Electric BWR Comparison

Product Line Designation	Year of Introduction	Representative U.S. Plants and General Characteristics

BWR/1	1955	*Dresden 1, Humboldt Bay, Big Rock Point*
		- First commercial BWRs, several different designs included in this class
		  - Dual-cycle natural circulation BWRs with power density up to 40 kW/liter and maximum power output of 250 MWe
		  - Direct-cycle natural circulation BWRs with power density of 40 kW/liter
		  - Forced circulation direct-cycle BWR with power density of 46 kW/liter and maximum power of 240 MWt
		- First use of steam separators inside the reactor vessel (KRB)
		- Some plants with dry containment, others with initial version of pressure suppression containment

BWR/2	1963	*Oyster Creek, Nine Mile Point 1* - First large commercial direct-cycle BWRs - Forced primary recirculation by external pump loops only - Recirculation flow control varying pump speed - Increased maximum power output to 1850 to 1930 MWt (500 to 640 MWe) - Power density of 36 kW/liter initially, but now operating at power densities up to 40.6 kW/liter - Isolation condensers and dedicated RHR system for decay heat removal - ECCS consisting of FWCI, ADS, and Core Spray - First use of Mark I pressure suppression containment
BWR/3	1965	*Monticello, Quad Cities* - Primary recirculation by external pump loops plus jet pumps in the core annulus - Increased maximum power output to 2500 MWt (810 MWe) - Power density increased to 41 kW/liter - RCIC and multi-mode RHR system for decay heat removal - Improved ECCS with HPCI, ADS, Core Spray, and LPCI (RHR operating mode) - Mark I pressure suppression containment

Table 9.1-1. General Electric BWR Comparison (continued)

Product Line Designation	Year of Introduction	Representative U.S. Plants and General Characteristics
BWR/4	1966	*Browns Ferry 1, 2, & 3, Peach Bottom 2 & 3* - Increased maximum power output to 3300 MWt (1100 MWe) - Power density increased to 51 kW/liter - Linear heat generation rate 18.5 kW/ft - Added steam-condensing mode of RCIC/RHR system operation - ECCS same as BWR/3 - Mark I pressure suppression containment for most plants in this class
BWR/5	1969	*LaSalle, Nine Mile Point 2, WNP-2* - Core design unchanged from late BWR/4s - Improved ECCS; HPCS, ADS, LPCS, and LPCI (RHR operating mode) - Control valves used for recirculation flow control - Mark II pressure suppression containment - Containment inerting not required
BWR/6	1972	*Clinton, Grand Gulf, Perry, River Bend* - Smaller diameter jet pumps allows larger core in same size reactor vessel - Increased maximum power output to 3800 MWt (1300 MWe) - First use of 8 x 8 fuel assembly - Increased power density to 54 kW/liter - Decreased linear heat generation rate to 13.4 kW/ft - Mark III pressure suppression containment - Containment inerting not required - Improved steam separators and dryers - Improved control and instrumentation systems incorporating digital technology

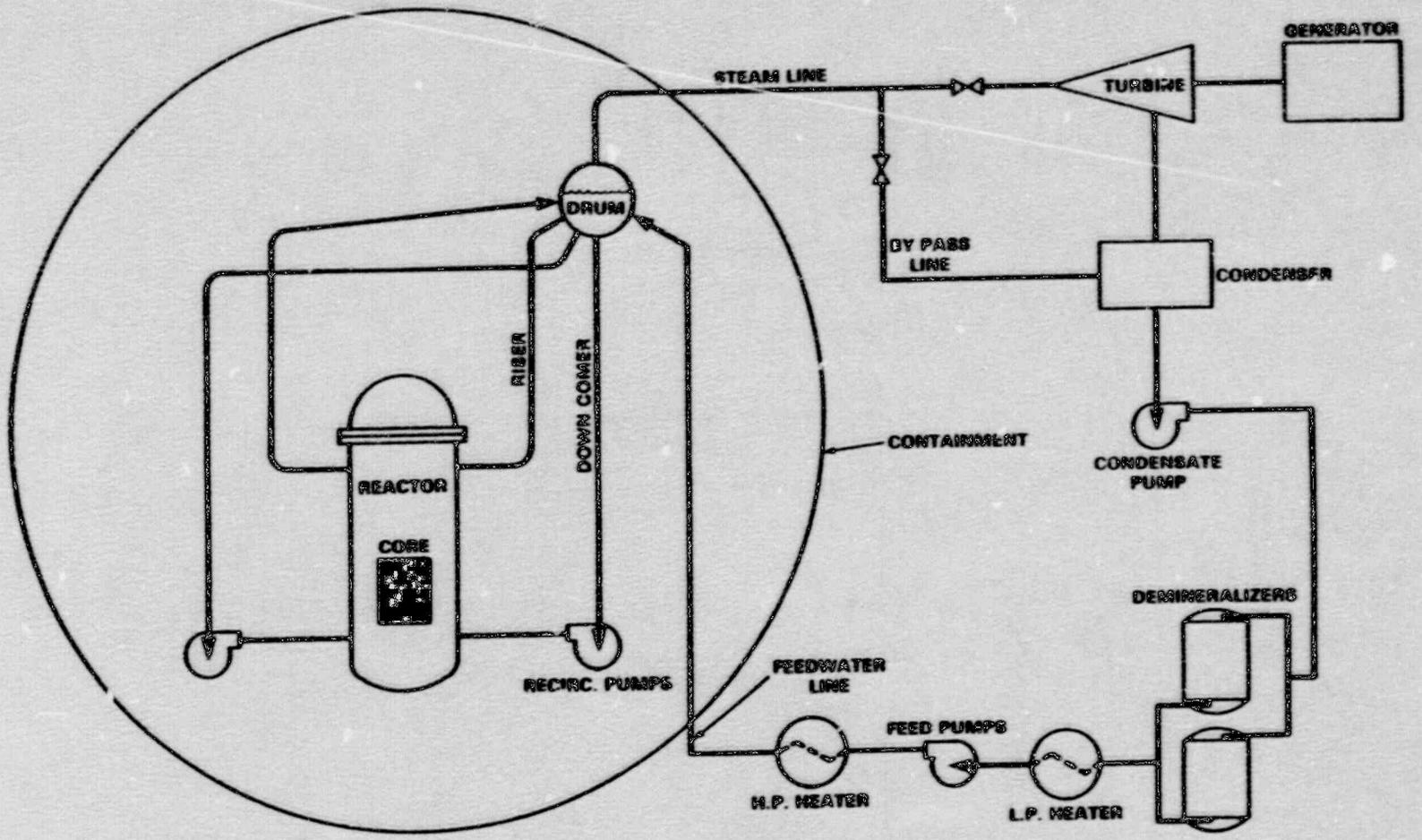


Figure 9.1-1. Simplified Diagram of the Big Rock Point Power Conversion System

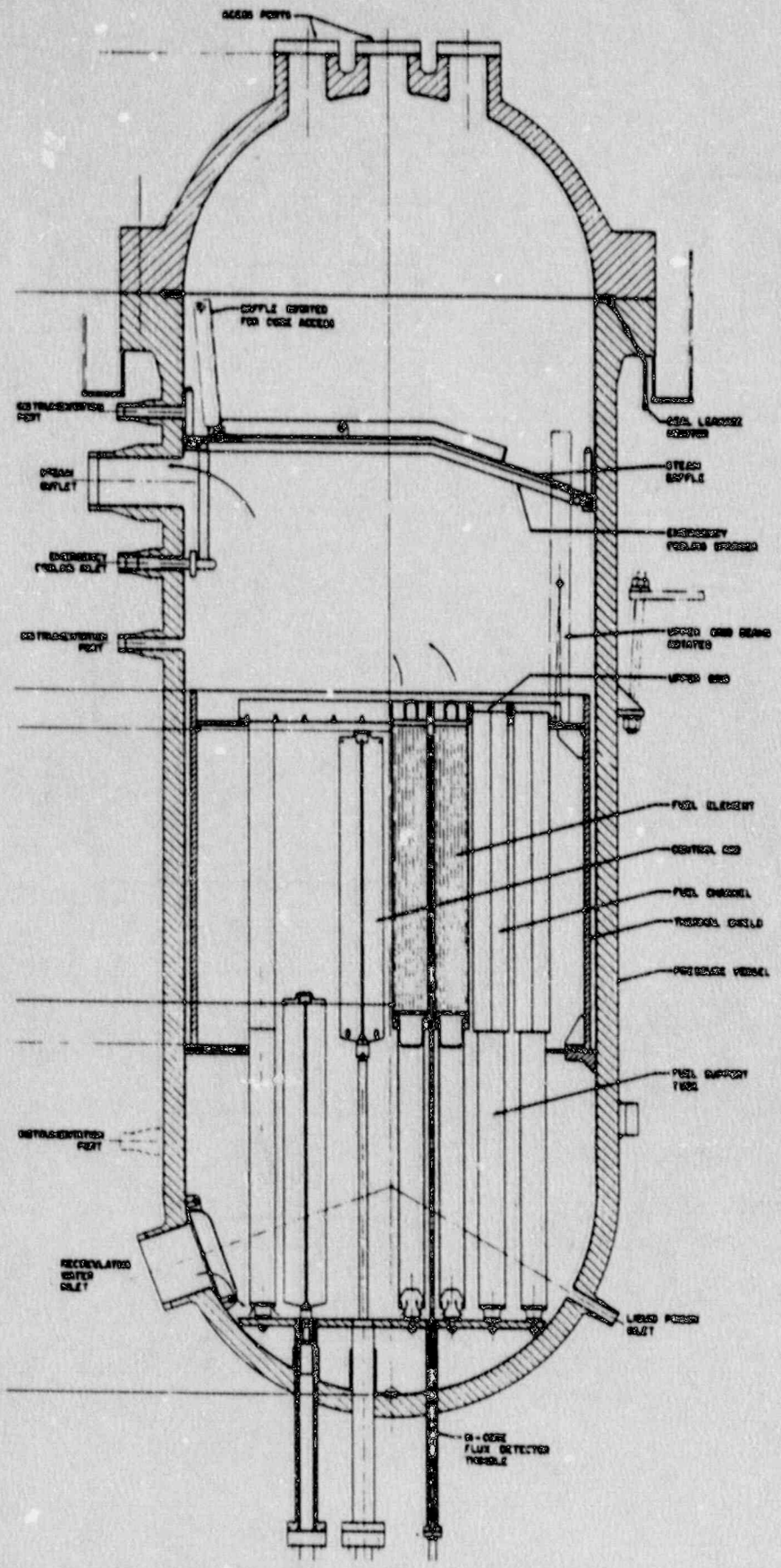


Figure 9.1-2. Big Rock Point Reactor Vessel and Internals

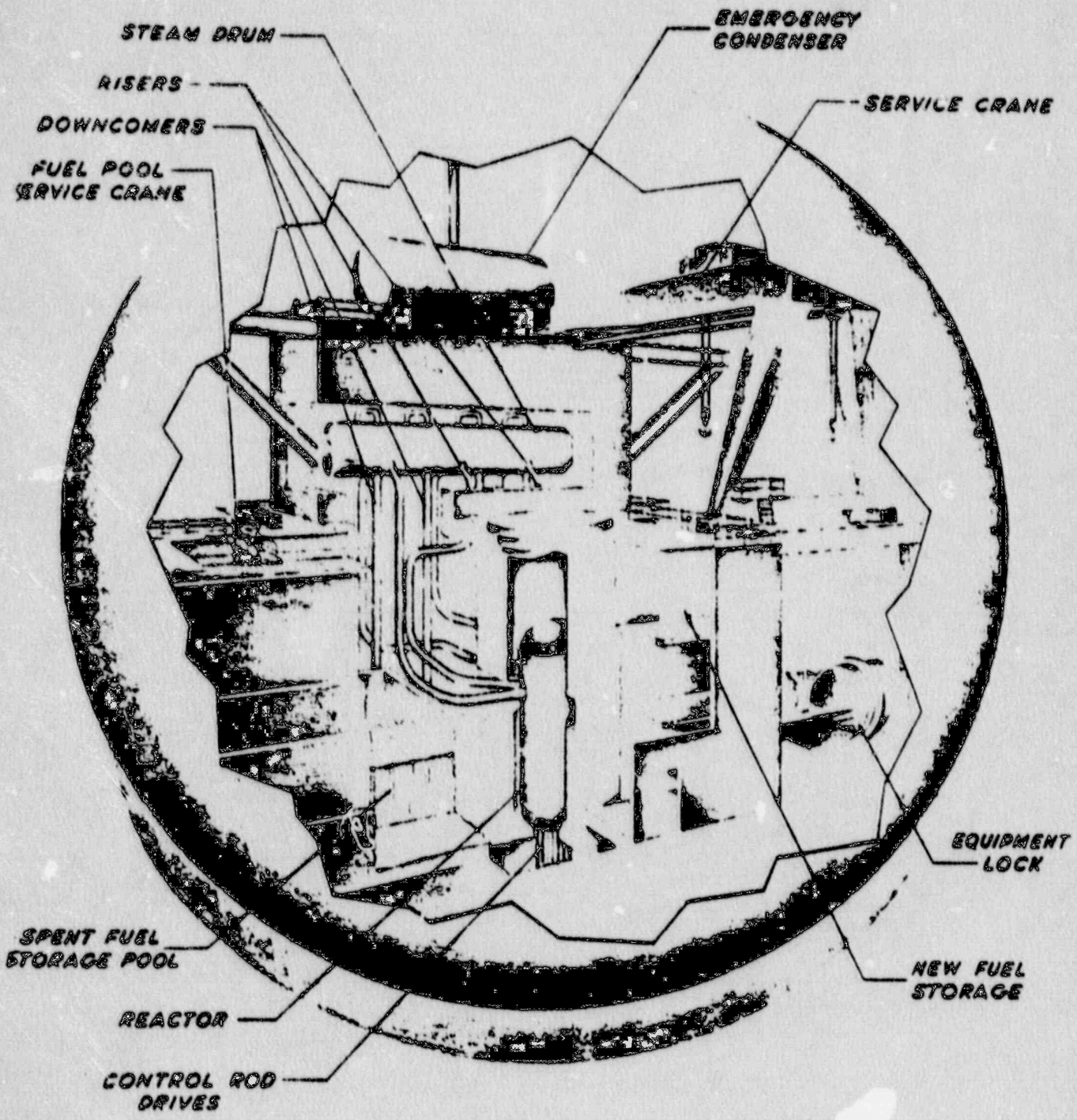


Figure 9.1-3. Cutaway View of the Big Rock Point Large, Dry Containment

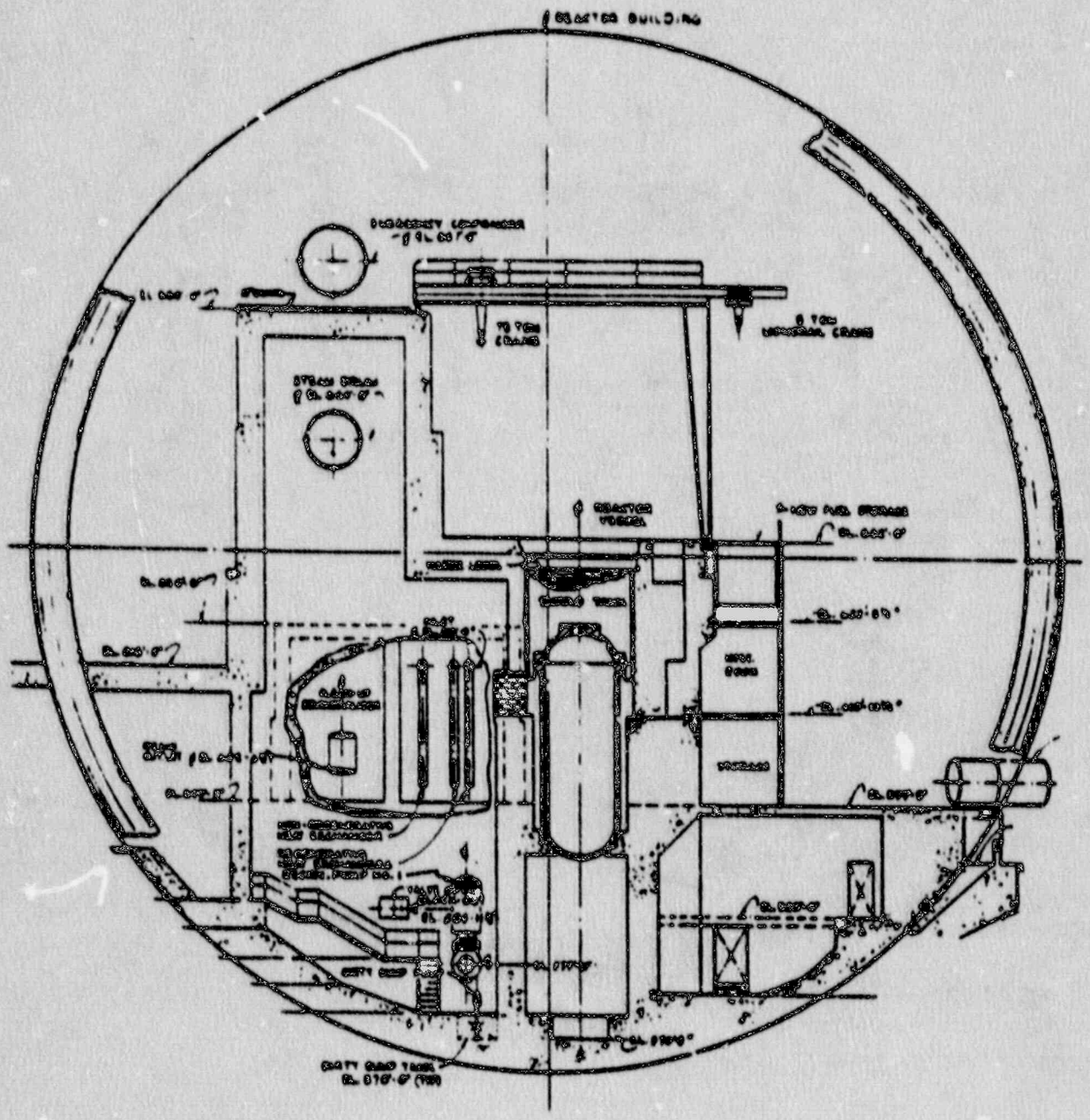


Figure 9.1-4. Section View of the Big Rock Point Large, Dry Containment

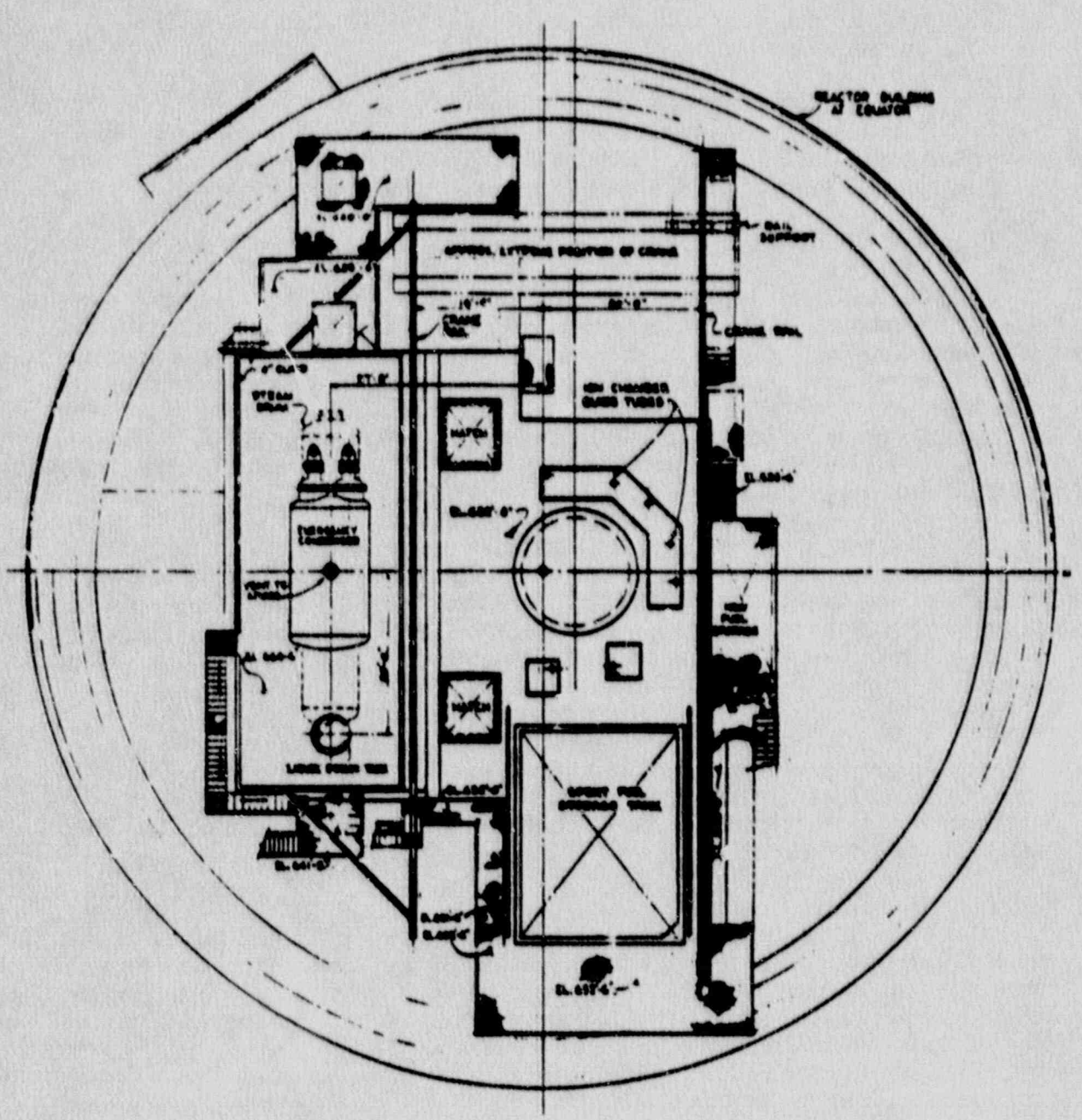


Figure 9.1-5. Plan View of the Big Rock Point Large, Dry Containment

9.2 BWR/2

The two BWR/2 plants in the United States are the following:

- Nine Mile Point 1
- Oyster Creek

Nine Mile Point 1 had a full power operating license as of 2/89 while Oyster Creek had a provisional operating license.

9.2.1 Reactor Core and Reactivity Control Systems

BWR/2 reactor cores are rated in the 1850 to 1930 MWt range, operate with a power density of 36 kw/liter, and are comprised of 532 (Nine Mile Point 1) to 560 (Oyster Creek) fuel assemblies. The BWR/2 was introduced with 7 x 7 fuel assemblies.

9.2.2 Reactor Coolant System, Including the Reactor Recirculation System

BWR/2 plants are "non-jet pump" plants. Five external recirculation loops provide for forced circulation of primary coolant. The basic arrangement of the BWR/2 nuclear boiler is shown in Figure 9.2-1. Note the location of the recirculating water outlet line (to the recirculation pumps) and the recirculating water inlet line (from the recirculation pumps). Note also the "skirt" that separates the bottom of the downcomer annulus from the lower plenum of the reactor vessel. In the jet pump BWRs (i.e., BWR/3 to BWR/6) the jet pumps discharge through this "skirt", while in BWR/2 plants, the recirculation loop piping enters below the "skirt".

9.2.3 Reactor Vessel Level Instrumentation

Reactor vessel water level measurements in BWR/1 to BWR/3, and most BWR/4 plants are made by temperature compensated differential pressure instruments that have a heated reference column. Operation of this type of instrument is described in Section 8. Typical reactor vessel water level instrumentation ranges for BWR/2 plants are shown in Figure 9.2-2.

9.2.4 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

BWR/2 plants provide RCS overpressure protection with 16 safety valves. The ADS function is accomplished with 5 to 6 "Electromatic" type power-actuated relief valves (PARVs) on the main steam lines.

9.2.5 Shutdown Cooling Systems

Initially, shutdown cooling is accomplished by using the main turbine bypass system to direct steam to the main condenser, and returning coolant to the reactor with the condensate and feedwater systems. Heat is transferred to the ultimate heat sink by the circulating water system. Alternatively, the isolation condenser can be aligned to create a high-pressure, natural circulation heat transfer path with primary steam rising to the isolation condenser and primary condensate returning to the reactor vessel. Heat is transferred to the ultimate heat sink by boiling secondary coolant in the isolation condenser and venting the secondary steam to the atmosphere. Oyster Creek has two isolation condensers located below the refueling floor level of the reactor building. Nine Mile Point 1 has four isolation condensers located on the refueling floor level.

Shutdown cooling is performed directly by a dedicated RHR system after the RCS has been cooled and depressurized to less than about 125 psig. The RHR pumps are aligned to take a suction on a recirculation loop (suction side of a recirculation pump) and return the coolant to the recirculation loop (discharge side of a recirculation pump). Heat is transferred to the ultimate heat sink by the RHR service water system.

9.2.6 Emergency Core Cooling Systems

The ECCS network for a BWR/2 plant consists of the following:

- The single-train Feedwater Coolant Injection (FWCI) system which injects water from condensate storage sources into the reactor vessel via a feedwater line.
- Automatic Depressurization System (ADS) which uses power-actuated relief valves to depressurize the RCS
- The two-train Core Spray (CS) system which injects water from the suppression pool directly into the reactor vessel, above the core.

Note the the BWR/2 ECCS does not include a Low Pressure Coolant Inejction (LPCI) system as found in BWR/3 to BWR/6 plants.

The FWCI system utilizes a motor-driven pump to provide makeup at high to moderate RCS pressure. The backup to the FWCI system for mitigating small breaks is the Automatic Depressurization System (ADS) and the low-pressure injection systems. The ADS uses several safety/relief valves to depressurize the reactor vessel to below the shutoff head of pumps in the Core Spray and LPCI systems.

The CS system injects water directly into the reactor vessel, via spargers located above the reactor core. The AC power lineup for the BWR/2 ECCS includes two electrical divisions with one core spray train powered from each division. The FWCI pump is powered from the non-Class 1E portion of the electric power system.

9.2.7 Containment (Mark I)

The Mark I containment was first introduced with the BWR/2 reactors. The general configuration of the Oyster Creek Mark I containment is shown in Figures 9.2-3. The cutaway view of the Nine Mile 1 reactor building in Figure 9.2-4 shows the layout of the refueling floor and the four isolation condensers in that plant. The Mark I containments for the BWR/2 plants are the only Mark I containments with different design pressures for the drywell and the torus. In both BWR/2 plants the drywell has a 62 psig design pressure and the torus has a 35 psig design pressure.

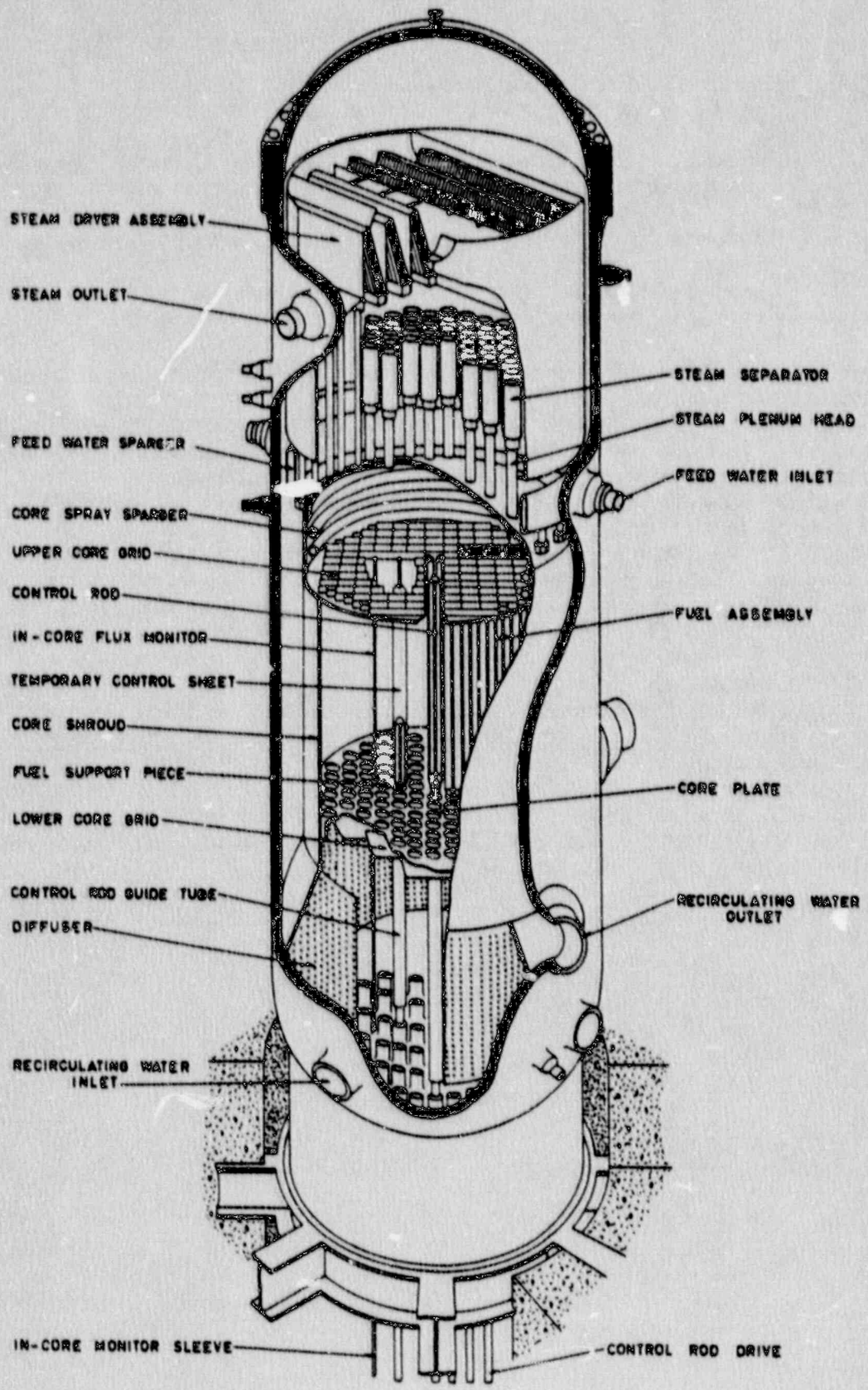


Figure 9.2-1. Typical BWR/2 Nuclear Boiler

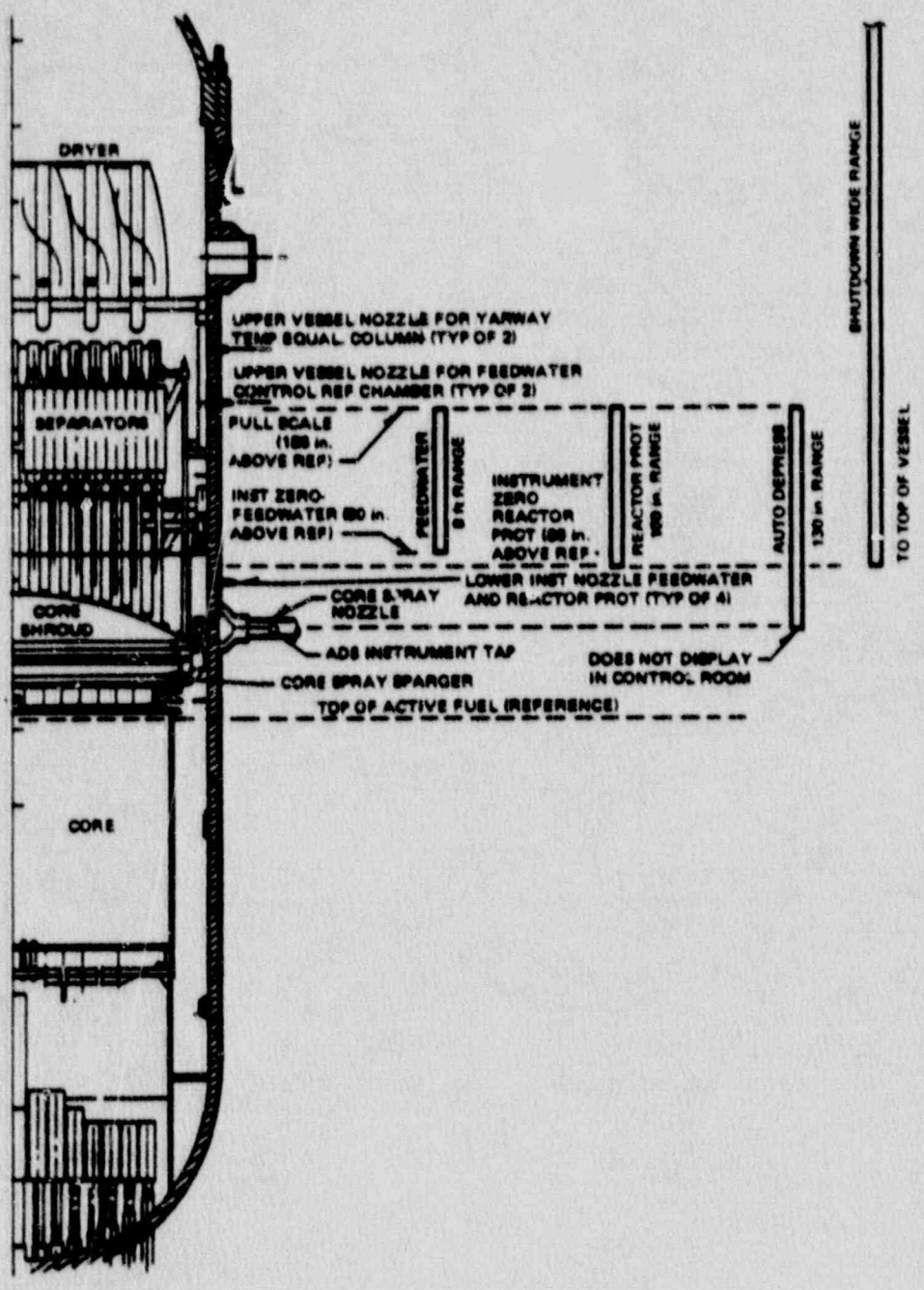


Figure 9.2-2. Typical BWR/2 Vessel Water Level Instrument Ranges

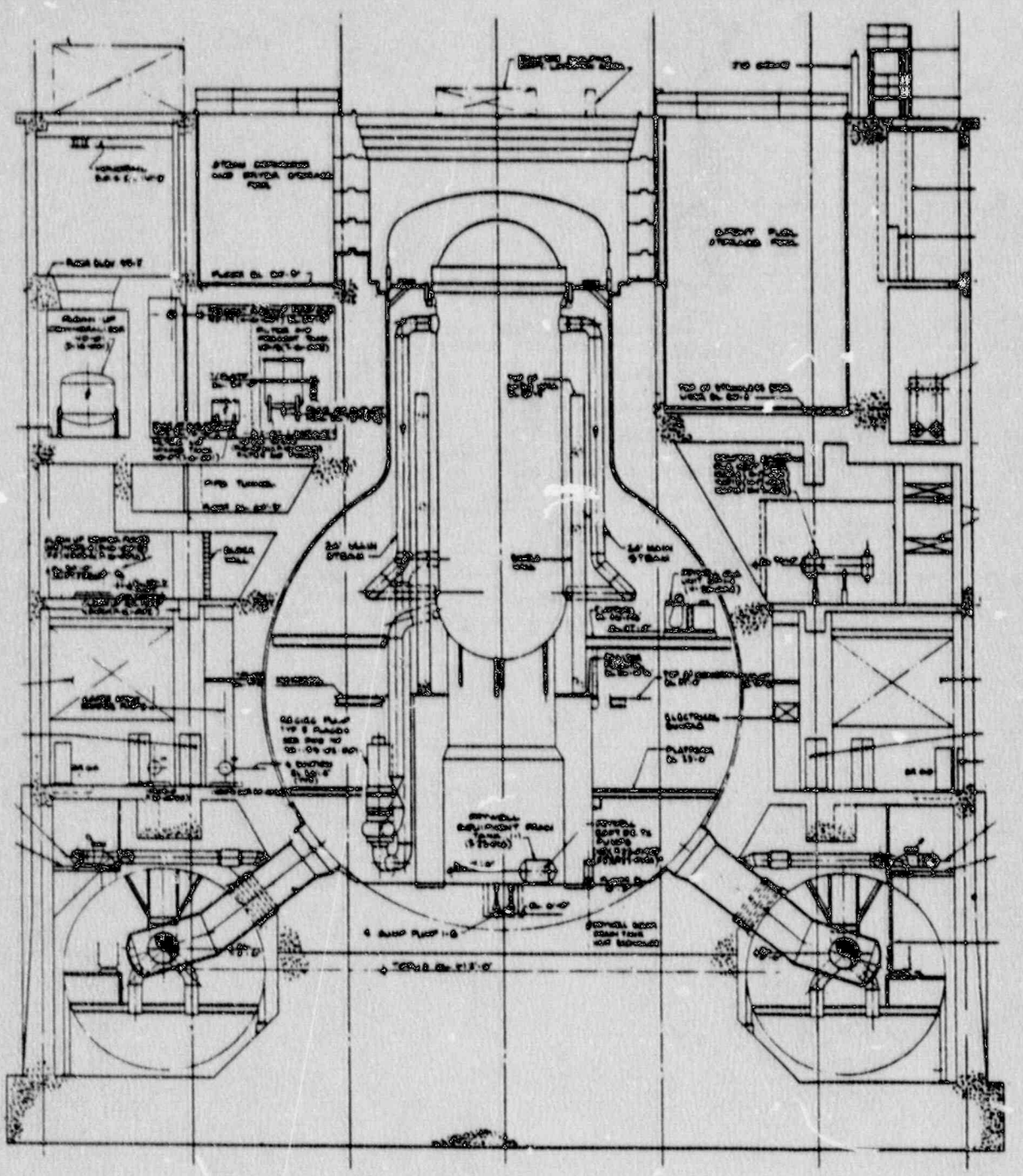


Figure 9.2-3. General Arrangement of the Oyster Creek Mark I Containment (Sheet 1 of 2)

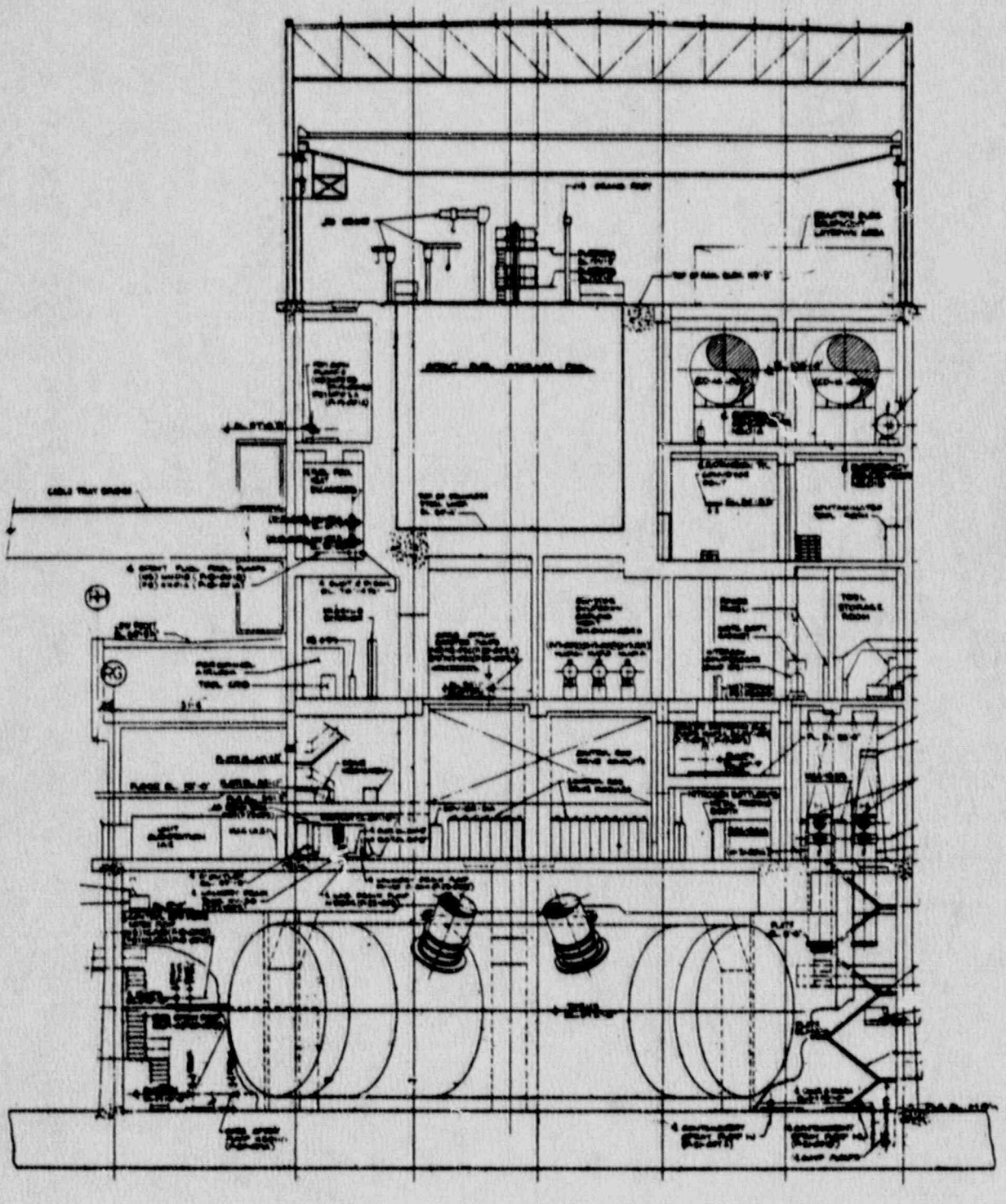


Figure 9.2-3. General Arrangement of the Oyster Creek Mark I Containment (Sheet 2 of 2)

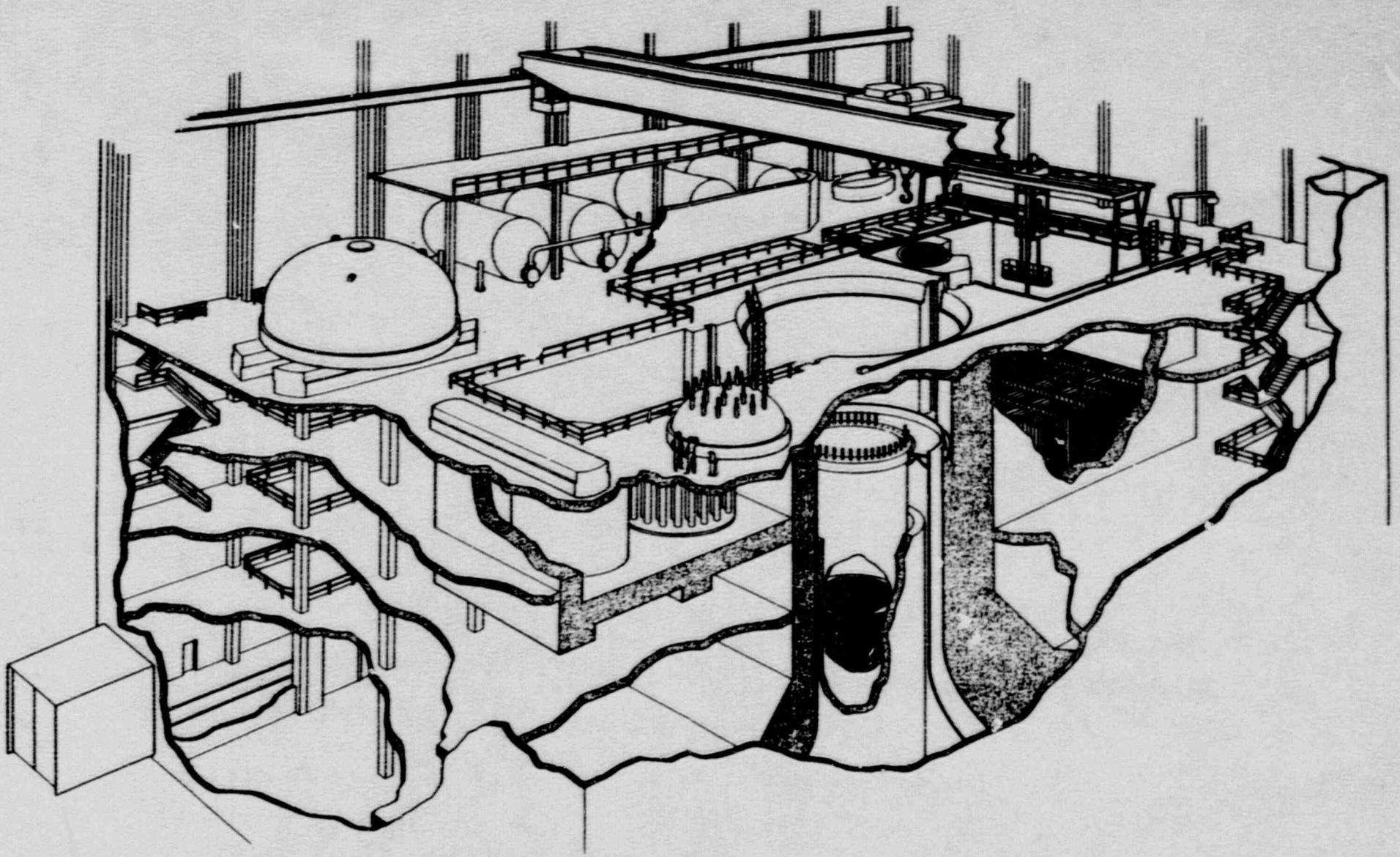


Figure 9.2-4. Cutaway View of the Nine Mile Point 1 Reactor Building

9.3 BWR/3

The seven BWR/3 plants in the United States are the following:

- Dresden 2 and 3
- Millstone 1
- Monticello
- Pilgrim 1
- Quad Cities 1 and 2

All of these reactor plants had full power operating licenses as of 2/89 except Dresden 2 which had a provisional operating license.

9.3.1 Reactor Core and Reactivity Control Systems

Plants classed as BWR/3s can be grouped into the following categories based on core thermal power output:

- 1600	Monticello
- 2000 MWt	Millstone 1 and Pilgrim 1
- 2500 MWt	Dresden 2 and 3 and Quad Cities 1 and 2

The smallest BWR/3 core has 484 fuel assemblies, the intermediate size cores have 580 fuel assemblies, and the largest cores have 724 fuel assemblies. The BWR/3 was introduced with 7 x 7 fuel assemblies, however many plants have changed to 8 x 8 fuel assemblies during successive refueling cycles. Maximum core power density was 40 kw/liter with 7 x 7 fuel assemblies.

9.3.2 Reactor Coolant System, Including the Reactor Recirculation System

A typical BWR/3 nuclear boiler is shown in Figure 9.3-1. BWR/3 plants were the first BWRs to use jet pumps to aid in driving recirculation flow. The use of jet pumps reduced the number of external recirculation pump loops from five in BWR/2 plants to two in BWR/3 to BWR/6 plants. A total of 20 jet pumps are arranged in the annulus between the reactor vessel and the core shroud and are driven by flow from the two external recirculation loops. The use of jet pumps eliminates recirculation line penetrations below the level of the reactor core, thereby improving the capability to maintain adequate core cooling following a LOCA. At full power approximately one-third of jet pump flow is driven flow and two-thirds is suction flow. Recirculation flow control is achieved by varying the speed of the pumps in the two external recirculation loops.

In BWR/3 plants, the discharge manifolds of the two external recirculation loops can be cross-connected. This feature allows one recirculation pump to drive all 20 jet pumps at reduced flow.

Natural circulation can be achieved in the primary system when the recirculation pumps are shutdown. A BWR can operate at approximately 10 percent power in the natural circulation mode.

9.3.3 Reactor Vessel Level Instrumentation

Reactor vessel water level measurements in BWR/1 to BWR/3, and most BWR/4 plants are made by temperature compensated differential pressure instruments that have a heated reference column. Operation of this type of instrument is described in Section 8. Typical reactor vessel water level instrumentation ranges for BWR/3 and BWR/4 plants are shown in Figure 9.3-2.

9.3.4 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

BWR/3 plants provide RCS overpressure protection with safety valves and one or more Target Rock type safety/relief valves. Some BWR/3s also have "Electromatic" type power-actuated relief valves that are used for the ADS.

9.3.5 Shutdown Cooling Systems

Initially, shutdown cooling is accomplished by using the main turbine bypass system to direct steam to the main condenser, and returning coolant to the reactor with the condensate and feedwater systems. Heat is transferred to the ultimate heat sink by the circulating water system.

Early BWR/3 plants have isolation condensers (Dresden 2 and 3 and Millstone 1) that can provide the same high-pressure shutdown cooling capability as found in BWR/2 plants. Later BWR/3 plants have a Reactor Core Isolation Cooling (RCIC) system which uses a steam turbine-driven pump to supply makeup to the RCS at high pressure from the refueling water storage tank or from the suppression pool. During RCIC system operation, primary steam and decay heat are transferred via the safety/relief valves to the suppression pool which acts as a heat sink. The RCIC steam turbine also exhausts to the suppression pool. Suppression pool cooling is provided by the RHR system.

Shutdown cooling is performed directly by the RHR system after the RCS has been cooled and depressurized to less than about 125 psig. In this operating mode, the RHR pumps are aligned to take a suction on a recirculation loop (suction side of a recirculation pump) and return the coolant to the recirculation loop (discharge side of a recirculation pump). Heat is transferred to the ultimate heat sink by the RHR service water system. Unlike BWR/2 plants, the RHR system in most BWR/3 plants is a multi-mode system that also performs the ECCS functions of low-pressure coolant injection (LPCI) and containment cooling (drywell and torus spray and suppression pool cooling). Millstone 1 and Dresden 2 and 3 are the only BWR/3 plants to have dedicated RHR systems.

In the early BWR/3s (i.e. Dresden 2 and 3), reactor vessel head spray is provided by the Control Rod Drive Hydraulic System (CRDHS) pumps. When head spray is required during cooldown, one of two CRDHS pumps is aligned for this function. In later BWR plants, the head spray function is accomplished by the RHR system.

9.3.6 Emergency Core Cooling Systems

The ECCS network for a BWR/3 plant consists of the following:

- The single-train High-Pressure Coolant Injection (HPCI) system which injects water from the refueling water storage tank or the suppression pool into the reactor vessel via a feedwater line.
- Automatic Depressurization System (ADS) which uses about half of the SRV's (typically uses 6 of the 11) to depressurize the RCS
- The two-train Core Spray (CS) system which injects water from the suppression pool directly into the reactor vessel, above the core.
- The two-train Low-Pressure Coolant Injection (LPCI) System which injects water from the suppression pool directly into the downcomer region of the reactor vessel. The LPCI system is an operating mode of the RHR system.

The HPCI system utilizes a steam turbine-driven pump to provide makeup at high to moderate RCS pressure (i.e. from 1150 to about 150 psig). The HPCI pump is automatically tripped when turbine steam inlet pressure drops below 100 psig, therefore, the HPCI system cannot be used for mitigating large LOCAs that depressurize the RCS. In early BWR/3 plants (i.e. Dresden 2 and 3), the HPCI steam line is connected directly to the reactor vessel. In later BWR/3s, the HPCI steam line is connected to one of the main steam

lines. The backup to the HPCI system for mitigating small breaks is the Automatic Depressurization System (ADS) and the low-pressure injection systems. The ADS uses several safety/relief valves to depressurize the reactor vessel to below the shutoff head of the pumps in the Core Spray and LPCI systems (i.e. typically 200 to 300 psig).

The CS system injects water directly into the reactor vessel, via spargers located above the reactor core.

The LPCI system is an operating mode of the Residual Heat Removal (RHR) system and injects water into the reactor vessel via the the recirculation lines. A "loop select" logic is used in some LPCI systems to detect the occurrence of a LOCA in a recirculation loop and direct the discharges of both LPCI trains to the remaining intact recirculation loop. It has been determined that the loop selection mechanism allows a single active failure (the LPCI injection valve in the path to the intact recirculation loop) to disable the entire LPCI system, placing dependence on the Core Spray system to accomplish the low pressure core cooling function. As noted above, the LPCI system is an operating mode of the RHR system which also can be aligned for suppression pool cooling, suppression chamber spray, or drywell spray.

The AC power lineup for the BWR/3 ECCS includes two electrical divisions with one LPCS and two LPCI pumps powered from each division. The HPCI pump is powered by a steam turbine.

9.3.7 Containment (Mark I)

All BWR/3 plants have a Mark I containment that is functionally similar to the Mark I containment found in BWR/2 plants. Other sources of steam discharged to the torus include the HPCI and RCIC pump turbines which are not found in BWR/2 plants.

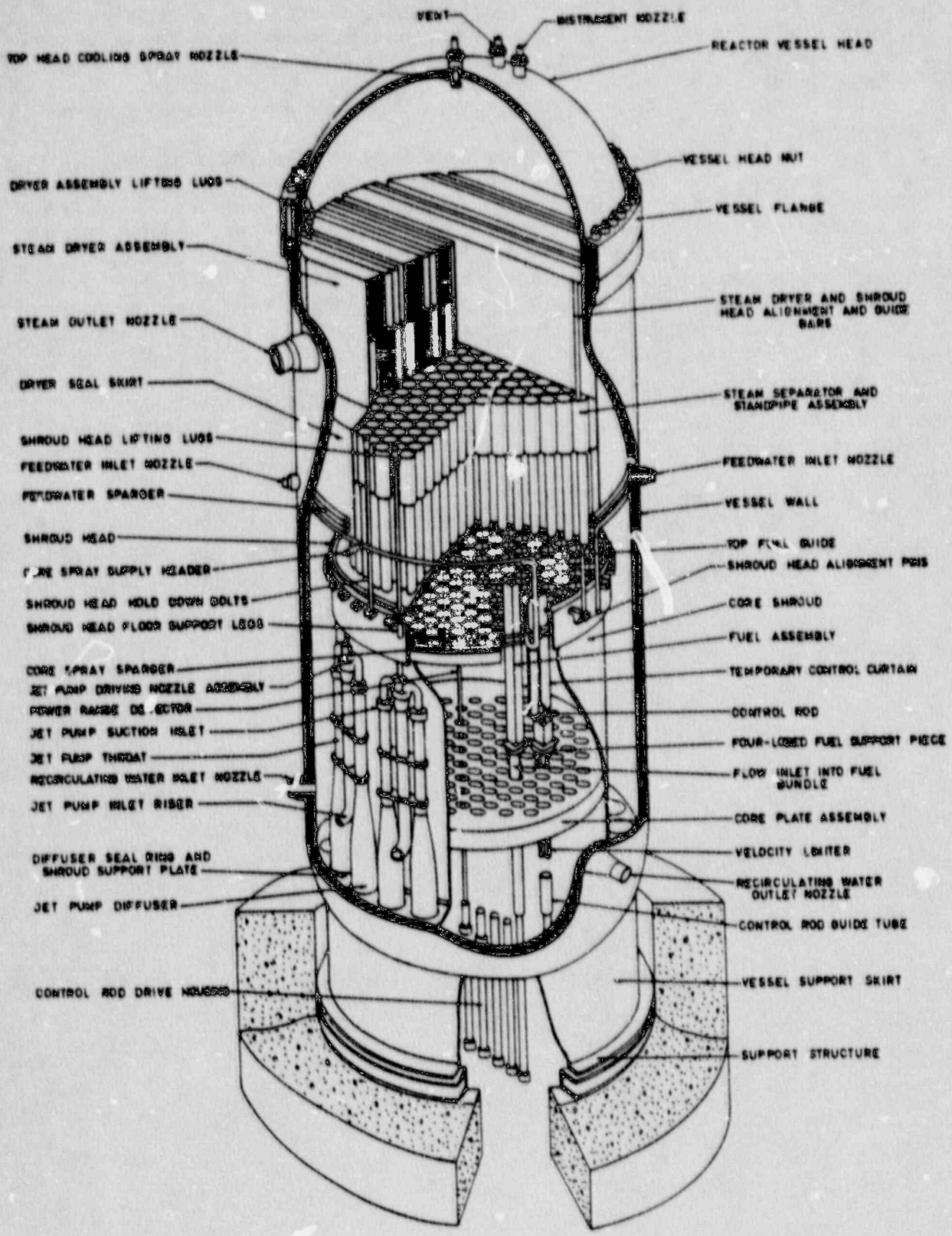
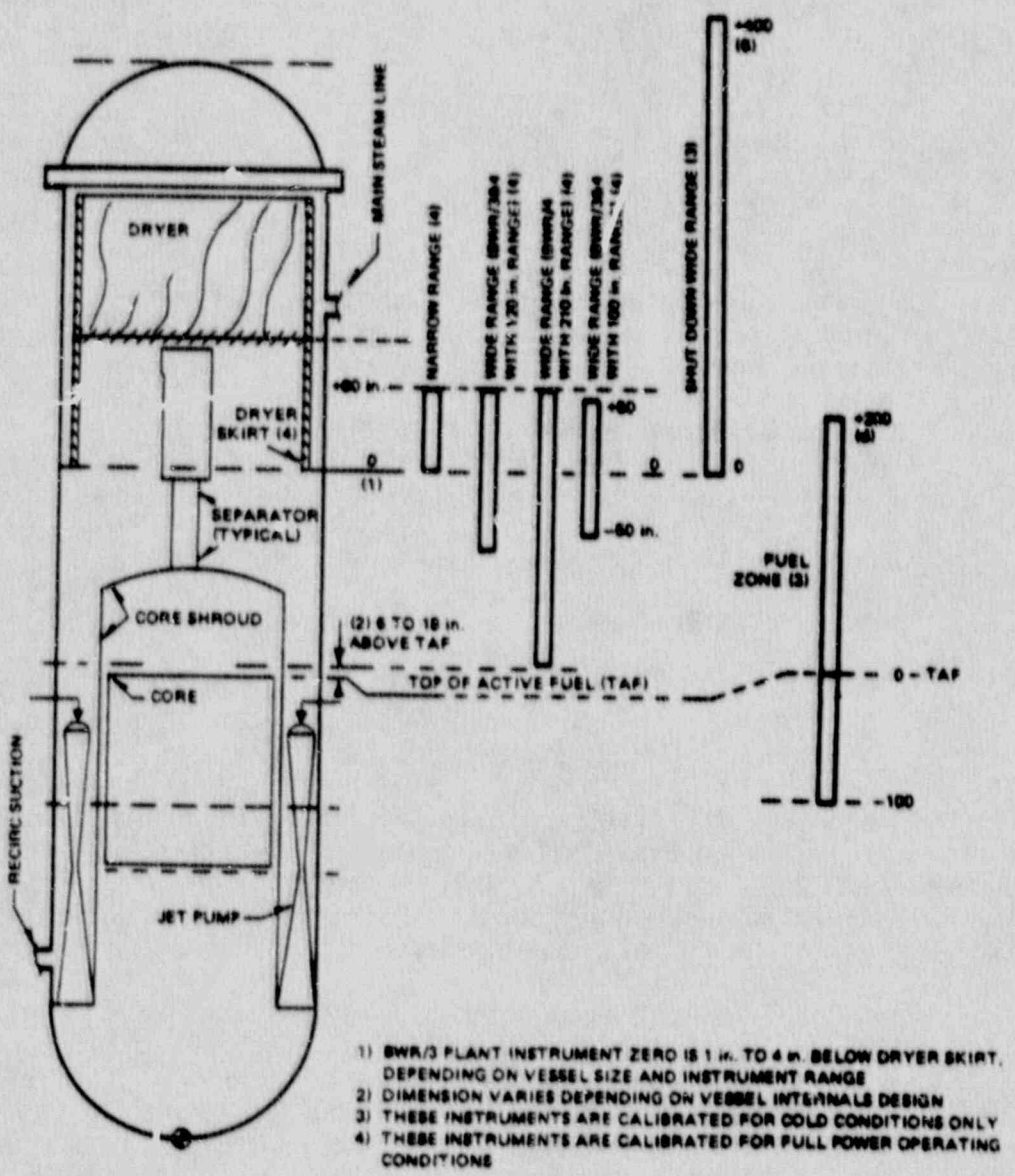


Figure 9.3-1. Typical BWR/3 Nuclear Boiler



1) BWR/3 PLANT INSTRUMENT ZERO IS 1 in. TO 4 in. BELOW DRYER SKIRT, DEPENDING ON VESSEL SIZE AND INSTRUMENT RANGE
2) DIMENSION VARIES DEPENDING ON VESSEL INTERNALS DESIGN
3) THESE INSTRUMENTS ARE CALIBRATED FOR COLD CONDITIONS ONLY
4) THESE INSTRUMENTS ARE CALIBRATED FOR FULL POWER OPERATING CONDITIONS

Figure 9.3-2. Typical BWR/3 Vessel Water Level Instrument Ranges

9.4 BWR/4

The twenty BWR/4 plants in the United States are the following:

- Browns Ferry 1, 2, and 3
- Brunswick 1 and 2
- Cooper
- Duane Arnold
- Fermi 2
- Fitzpatrick
- Hatch 1 and 2
- Hope Creek 1
- Limerick 1 and 2
- Peach Bottom 2 and 3
- Shoreham
- Susquehanna 1 and 2
- Vermont Yankee

All of these reactor plants except Limerick 2 and Shoreham had full power operating licenses as of 2/89. Limerick 2 entered commercial operation in early 1990. Shoreham had a low power operating license

9.4.1 Reactor Core and Reactivity Control Systems

Plants classed as BWR/4s can be grouped into the following categories based on core thermal power output:

- 1600 to 1700 MWt	Duane Arnold and Vermont Yankee
- 2380 MWt	Cooper
- 2430 to 2500 MWt	Brunswick 1 and 2, Fitzpatrick, Hatch 1 and 2, and Shoreham
- 3300 MWt	Browns Ferry 1, 2, and 3, Fermi 2, Hope Creek, Limerick 1 and 2, Peach Bottom 2 and 3, and Susquehanna 1 and 2

The smallest BWR/4 cores typically have 368 fuel assemblies, the intermediate size cores have 548 to 560 fuel assemblies, and the largest cores have 764 fuel assemblies. The BWR/4 was introduced with 7 x 7 fuel assemblies, however, many plants have changed to 8 x 8 fuel assemblies during successive refueling cycles. Maximum core power density was 51 kw/liter with 7 x 7 fuel assemblies.

9.4.2 Reactor Coolant System, Including the Reactor Recirculation System

A typical BWR/4 nuclear boiler is shown in Figure 9.4-1. As in BWR/2 and BWR/3 plants, recirculation flow rate in a BWR/4 plant is controlled by variable speed motor-generator sets which drive variable speed reactor recirculation pumps. The cross-connection between the recirculation loop discharge manifolds found in BWR/3 plants is deleted in the BWR/4 and later plants.

9.4.3 Reactor Vessel Level Instrumentation

Reactor vessel water level measurements in BWR/1 to BWR/3, and most BWR/4 plants are made by temperature compensated differential pressure instruments that have a heated reference column. Operation of this type of instrument is described in Section 8. Reactor vessel water level measurements in some BWR/4 plants are made by differential pressure instruments that have non-heated reference column. The reference column is maintained full of condensate by a condensing chamber at the top of the column

which is connected to the reactor vessel steam dome by a small diameter instrument line. Excess condensate drains back into the reactor vessel by way of the instrument line. Typical reactor vessel water level instrumentation ranges for BWR/3 and BWR/4 plants are shown in Figure 9.3-2.

9.4.4 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

All BWR/4 plants are equipped with Target Rock type safety/relief valves (SRVs) that operate mechanically for the overpressure protection function and can be power-operated for relief at lower pressures and for depressurization of the primary system. About half of the BWR/4 plants also have safety valves to accomplish the overpressure protection function.

9.4.5 Shutdown Cooling Systems

Initially, shutdown cooling is accomplished by using the main turbine bypass system to direct steam to the main condenser, and returning coolant to the reactor with the condensate and feedwater systems. Heat is transferred to the ultimate heat sink by the circulating water system.

Alternatively in most BWR/4 plants, the RCIC and RHR systems can be used in the steam-condensing mode of operation to provide a closed-loop high-pressure shutdown cooling capability. In this operating mode, high-pressure steam from the RCS passes through pressure-reducing valves to the RHR heat exchangers where the steam is condensed and heat is transferred to the ultimate heat sink by the RHR service water system. The primary condensate is directed to the suction of the RCIC pump which returns the coolant to the RCS. About 1-1/2 hours after reactor shutdown, one RHR heat exchanger operating as a steam condenser has sufficient capacity to handle the decay heat load. The advantage of this closed-loop core cooling scheme is the greatly reduced amount of heat dumped to the suppression pool. In some plants, the steam-condensing mode of RCIC and RHR system operation has been permanently disabled.

Shutdown cooling is performed directly by the RHR system after the RCS has been cooled and depressurized to less than about 125 psig. In this operating mode, the RHR pumps are aligned to take a suction on a recirculation loop (suction side of a recirculation pump) and return the coolant to the recirculation loop (discharge side of a recirculation pump). Heat is transferred to the ultimate heat sink by the RHR service water system.

9.4.6 Emergency Core Cooling Systems

The ECCS network for a BWR/4 plant consists of the following:

- The single-train High-Pressure Coolant Injection (HPCI) system which injects water from the refueling water storage tank or the suppression pool into the reactor vessel via a feedwater line.
- Automatic Depressurization System (ADS) which uses about half of the SRVs (typically uses 6 of the 11) to depressurize the RCS
- The two-train Core Spray (CS) system which injects water from the suppression pool directly into the reactor vessel, via spray spargers located above the core.
- The two-train Low-Pressure Coolant Injection (LPCI) System which injects water from the suppression pool directly into the downcomer region of the reactor vessel. The LPCI system is an operating mode of the RHR system.

The BWR/4 ECCS is comparable to the BWR/3 ECCS described in Section 9.3, however, there are differences in the reactor vessel level setpoints which are used to actuate the

various ECCS subsystems. Note that a few BWR/4 plants have 2 x 50% capacity core spray pumps per train rather than the more common 1 x 100% capacity pump per train.

9.4.7 Containment (Mark I or Mark II)

Most BWR/4 plants have Mark I containments that are comparable to those found in BWR/2 and BWR/3 plants. The general arrangement of the Fitzpatrick Mark I containment is shown in Figure 9.4-2. Note that the Brunswick Mark I containment is of unique construction. Rather than having a free-standing steel drywell and wetwell as in all other Mark I containments, the Brunswick Mark I containment is constructed of reinforced concrete with a steel liner. Mark II containments are used for Limerick 1 and 2, Susquehanna 1 and 2, and Shoreham. All of these Mark II containments have a reinforced concrete drywell and wetwell with a steel liner. The general arrangement of the Limerick Mark II containment is shown in Figure 9.4-3. The Mark I and II containment are described in Section 8.

Most BWR/4 plants have eight groups of containment isolation valves (up from five in BWR/3 plants).

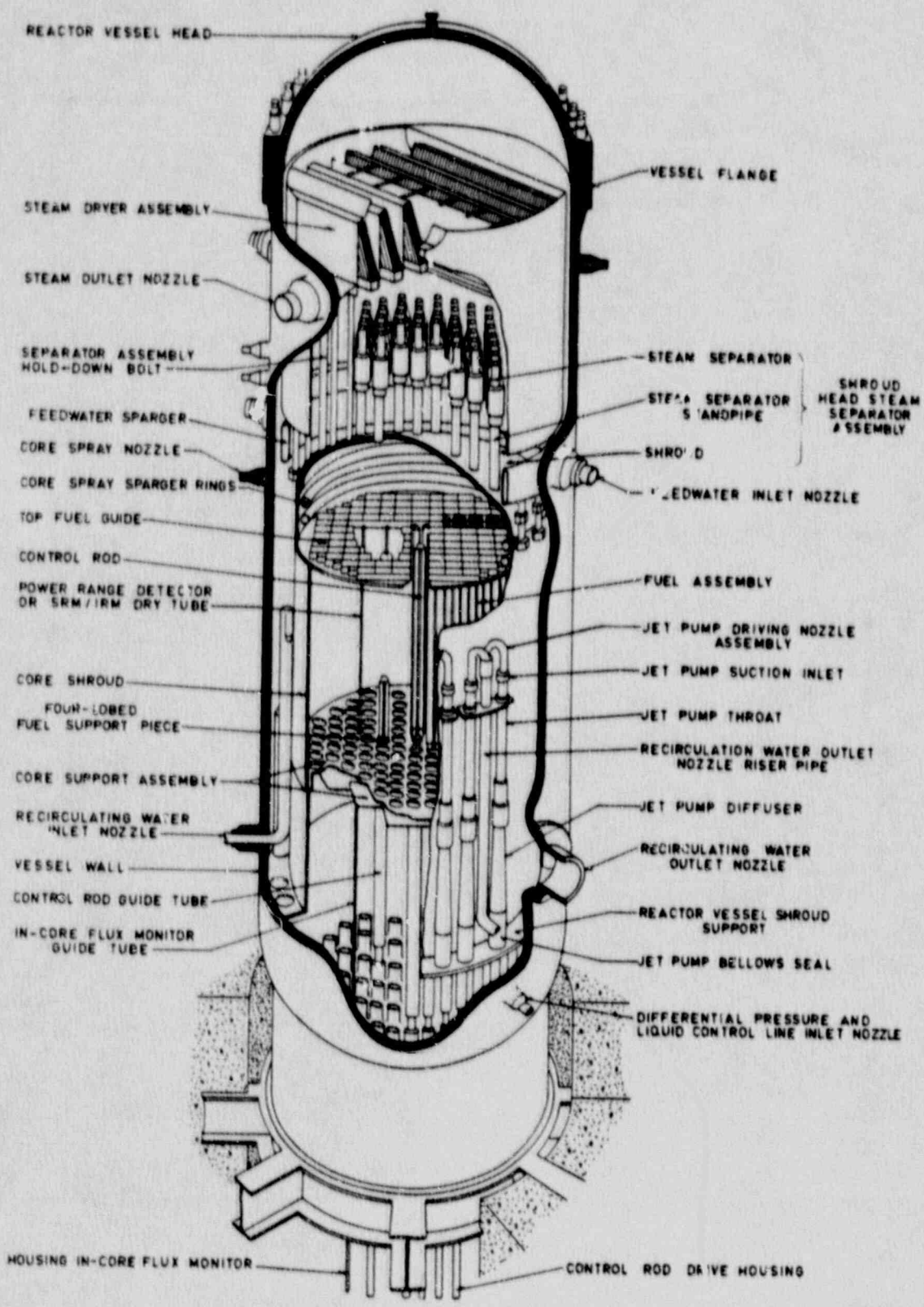


Figure 9.4-1. Typical BWR/4 Nuclear Boiler

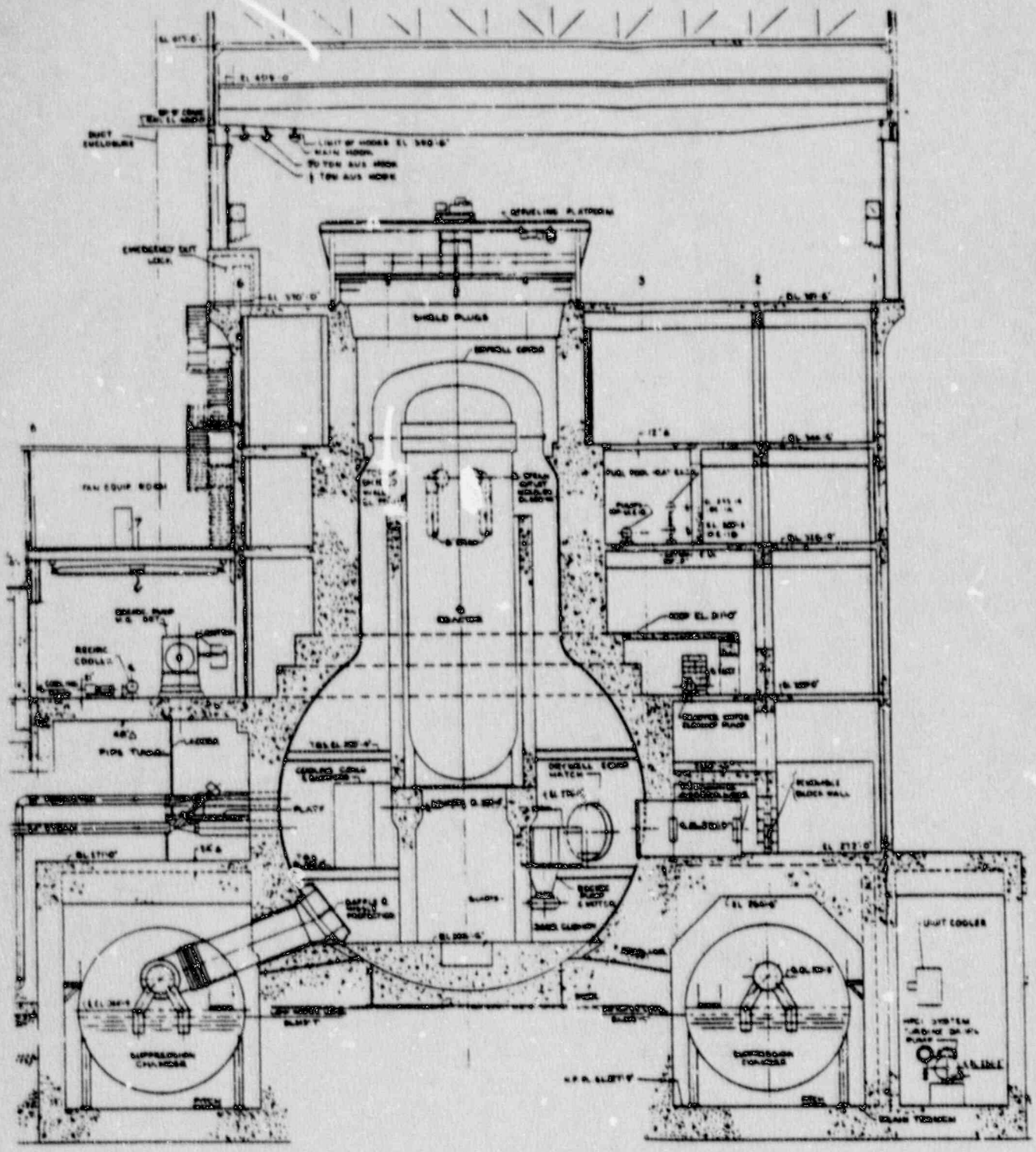


Figure 9.4-2. General Arrangement of the Fitzpatrick Mark I Containment

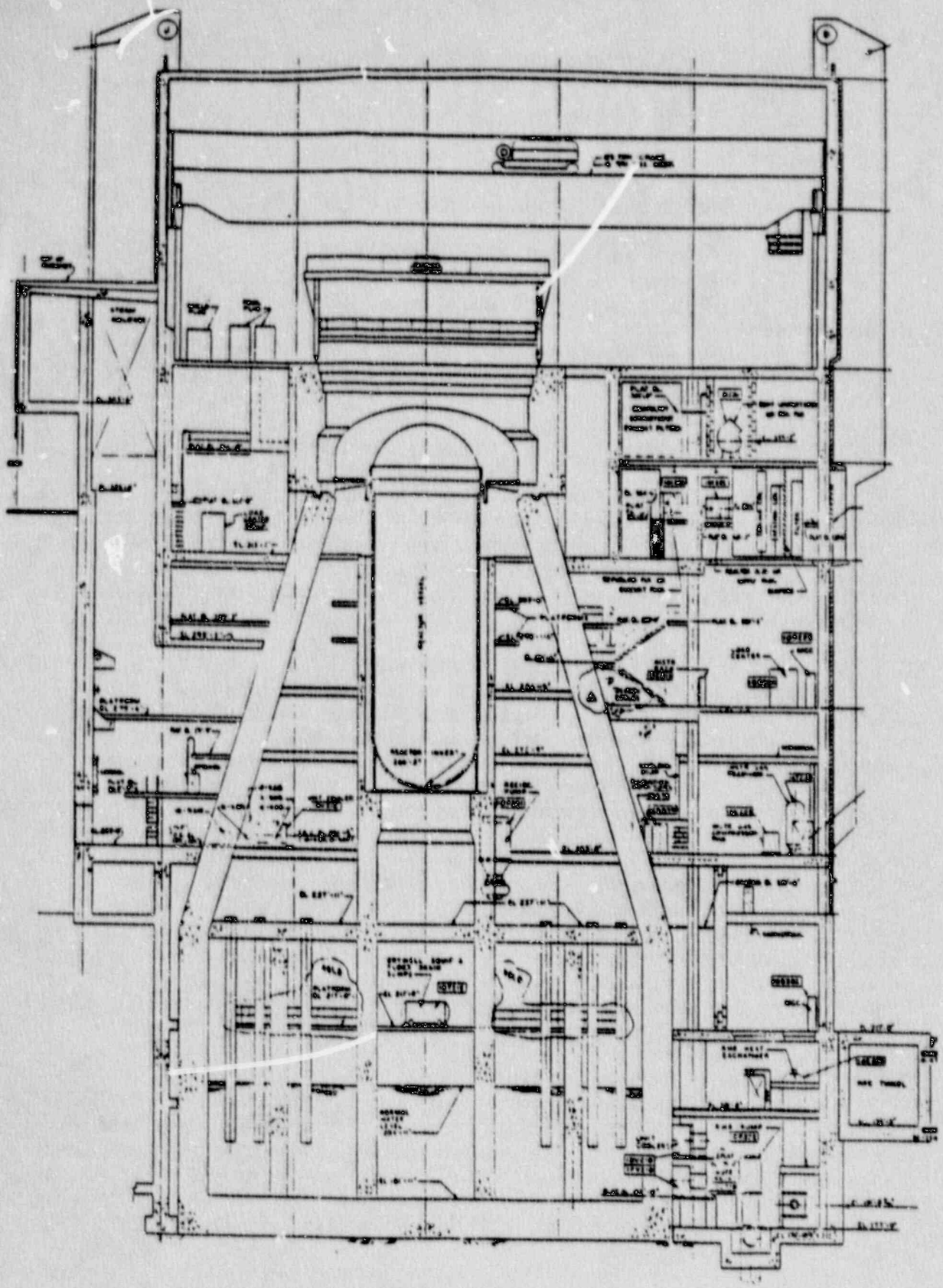


Figure 9.4-3. General Arrangement of the Limerick Mark II Containment

9.5 BWR/5

The five BWR/5 plants in the United States are the following:

- LaSalle 1 and 2
- Nine Mile Point 2
- WNP-2

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

9.5.1 Reactor Core and Reactivity Control Systems

The BWR/5 nuclear boiler is comparable to the BWR/4 nuclear boiler shown in Figure 9.4-1. The BWR/5 reactor core is rated at about 3300 MWt with a power density of 51 kW/liter and is comprised of 764 fuel assemblies. The core and reactivity control system design is essentially unchanged from the later BWR/4 design. The BWR/5 was introduced with 7 x 7 fuel elements, but has been changed to 8 x 8 fuel assemblies during successive refueling cycles.

9.5.2 Reactor Coolant System, Including the Reactor Recirculation System

The recirculation system for BWR/5 was simplified by replacing the variable speed recirculation pump feature found in BWR/2 to BWR/4 plants with constant speed reactor recirculation pumps and recirculation flow control valves. The revised system provides improved plant response to load transients (i.e., step increase or decrease in load demand). The main turbine control also was modified to match the faster response of the BWR/5 recirculation flow control system.

9.5.3 Reactor Vessel Level Instrumentation

Reactor vessel water level measurements in all BWR/5 plants are made by differential pressure instruments that have a non-heated reference column. The reference column is maintained full of condensate by a condensing chamber at the top of the column which is connected to the reactor vessel steam dome by a small diameter instrument line. Excess condensate drains back into the reactor vessel by way of the instrument line. Typical reactor vessel water level instrumentation ranges for a BWR/5 are shown in Figure 9.5-1.

9.5.4 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

BWR/5 plants are equipped only with safety/relief valves (SRVs) that operate mechanically for the overpressure protection function and can be power-operated for relief at lower pressures and for depressurization of the primary system. Typically, 18 SRVs are connected to the four main steam lines and discharge to the suppression pool. The SRVs typically are Target Rock three-stage valves which are described in Section 8.

9.5.5 Shutdown Cooling Systems

The BWR/5 shutdown cooling capability is comparable to the BWR/4 shutdown cooling capability described in Section 9.4. The BWR/5 RCIC and RHR systems are designed for operating in the steam-condensing mode to provide a closed-loop high-pressure shutdown cooling capability. A safety-related service water system completes the heat transfer path from the RHR heat exchangers and other safety-related heat loads to the ultimate heat sink. There is no separate high-pressure RHR service water system as in earlier BWR plants.

9.5.6 Emergency Core Cooling Systems

The ECCS network for a BWR/5 plant consists of the following:

- The single-train High-Pressure Core Spray (HPCS) system which injects water from the refueling water storage tank or the suppression pool directly into the reactor vessel, via a spray sparger located above the core.
- Automatic Depressurization System (ADS) which uses less than half of the SRVs (typically uses 7 of the 18) to depressurize the RCS
- The single-train Low-Pressure Core Spray (LPCS) system which injects water from the suppression pool directly into the reactor vessel, via a spray sparger located above the core.
- The three-train Low-Pressure Coolant Injection (LPCI) System which injects water from the suppression pool directly into the core bypass region inside the core shroud. The LPCI system is an operating mode of the RHR system. Note that RHR trains A and B have multiple modes of operation. Train C operates only in the LPCI mode.

The HPCS system replaces the HPCI system found in BWR/3 and BWR/4 plants. The HPCS system uses an electric motor-driven pump to inject directly into the reactor vessel through a spray header above the core and is designed to work over the full range of RCS operating pressure that may be encountered following a LOCA. The HPCS pump is powered from a dedicated diesel generator which functions as a constant-torque drive. Except at the extremes of the operating range, HPCS flow is inversely proportional to reactor vessel pressure. In contrast, the HPCI system in BWR/3 and BWR/4 plants uses a steam turbine-driven pump to inject water into the reactor vessel via a feedwater line under high to moderate pressure conditions. The HPCI steam turbine is automatically tripped under low steam pressure conditions.

The BWR/5 LPCS system is a single-train system. In earlier BWRs, the Core Spray System typically was a two-train system. As discussed above, the HPCS system is capable of operating over the full range of RCS operating pressure, therefore, the HPCS performs the function of the redundant core spray train in earlier BWRs.

The BWR/5 LPCI system injects directly into the reactor vessel (into the core bypass region inside the core shroud) rather than into a recirculation loop as is done in BWR/2 to BWR/4 plants. This design change reduces LPCI head loss and power requirements in comparison to earlier BWRs. As noted above, the LPCI system is an operating mode of the RHR system which also can be aligned for reactor vessel head spray, suppression pool cooling, suppression chamber spray, drywell spray, or high-pressure decay heat removal (i.e. steam-condensing operation) in conjunction with the RCIC system.

As in most other BWRs, the service water system is capable of being connected to an RHR injection line to perform the low-pressure core or containment flooding function.

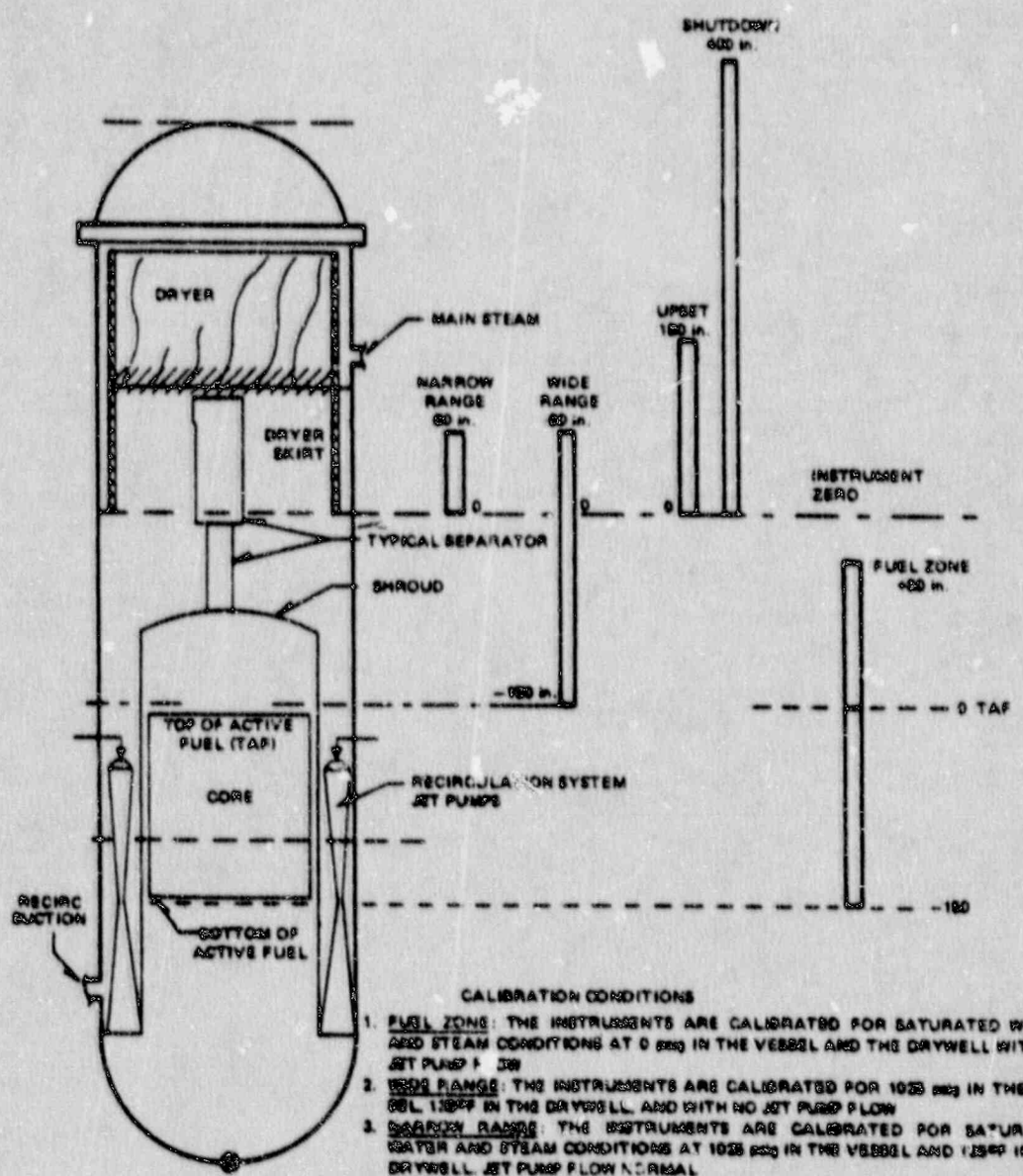
The AC power lineup for the BWR/5 ECCS includes three electrical divisions with one LPCS and one LPCI pump on division A, two LPCI pumps in division B, and the HPCS pump on division C.

9.5.7 Containment (Mark II)

All BWR/5 plants have Mark II containments, although construction techniques vary significantly, as shown below:

Containment Construction	Applicable Plant
Reinforced concrete drywell, free-standing steel wetwell	WNP-2
Reinforced concrete drywell and wetwell with a steel liner	Nine Mile Point 2
Reinforced concrete drywell, post-tensioned concrete wetwell with a steel liner	LaSalle 1 and 2

All Mark II containments also have a secondary containment, as described in Section 8.



CALIBRATION CONDITIONS

1. FUEL ZONE: THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER AND STEAM CONDITIONS AT 0 psig IN THE VESSEL AND THE DRYWELL WITH NO JET PUMP FLOW.
2. WIDE RANGE: THE INSTRUMENTS ARE CALIBRATED FOR 1035 psig IN THE VESSEL, 135°F IN THE DRYWELL, AND WITH NO JET PUMP FLOW.
3. NARROW RANGE: THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER AND STEAM CONDITIONS AT 1035 psig IN THE VESSEL AND 135°F IN THE DRYWELL. JET PUMP FLOW NORMAL.
4. UPSET RANGE: THE INSTRUMENT IS CALIBRATED FOR SATURATED WATER AND STEAM CONDITIONS AT 1035 psig IN THE VESSEL AND 135°F IN THE DRYWELL. JET PUMP FLOW NORMAL.
5. SHUTDOWN: THE INSTRUMENT IS CALIBRATED FOR 130°F WATER AT 0 psig IN VESSEL AND 65°F IN THE DRYWELL. NO JET PUMP FLOW.

Figure 9.5-1. Typical BWR/5 Vessel Water Level Instrument Ranges

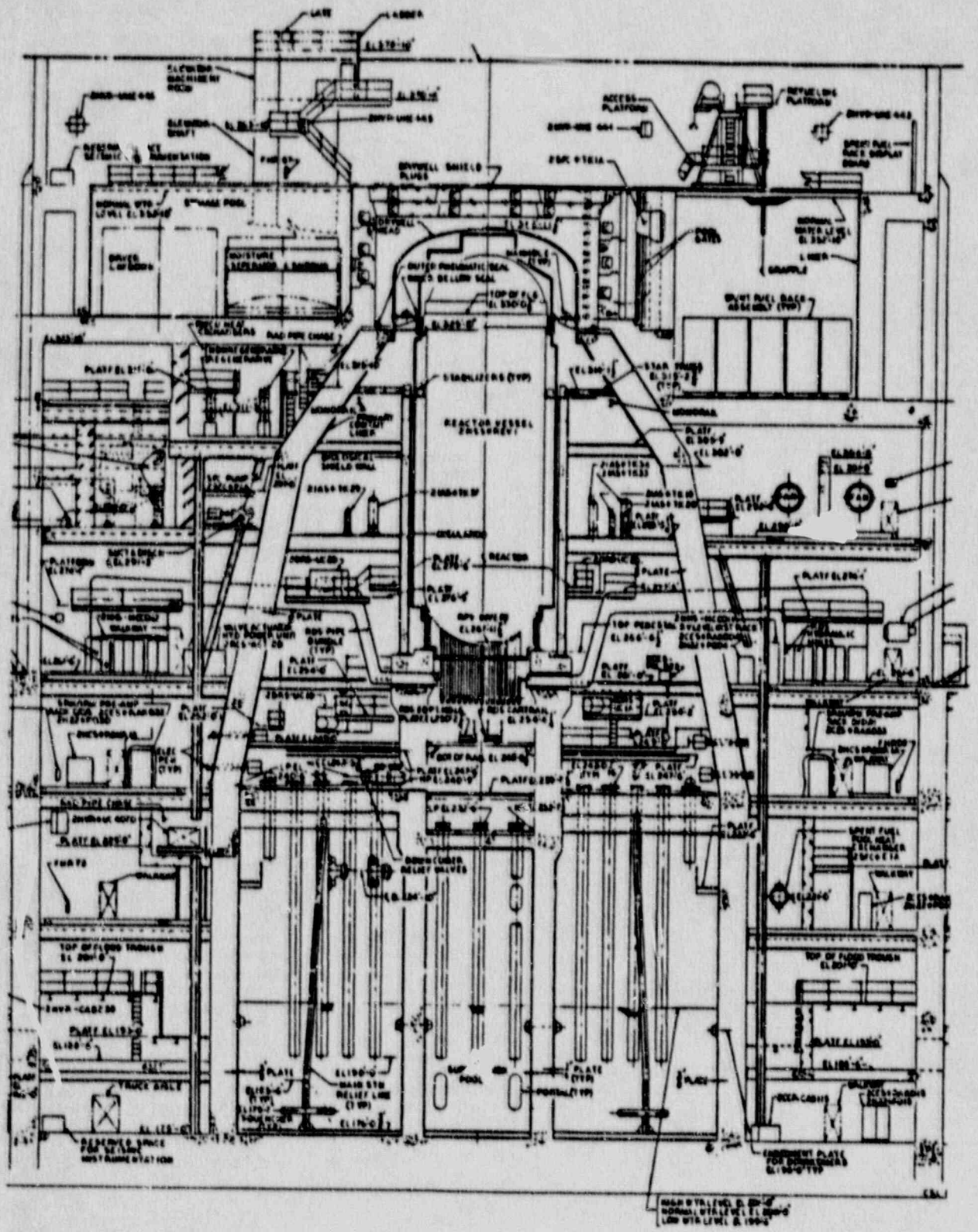


Figure 9.5-2. General Arrangement of the Nine Mile Point 2 Mark II Containment

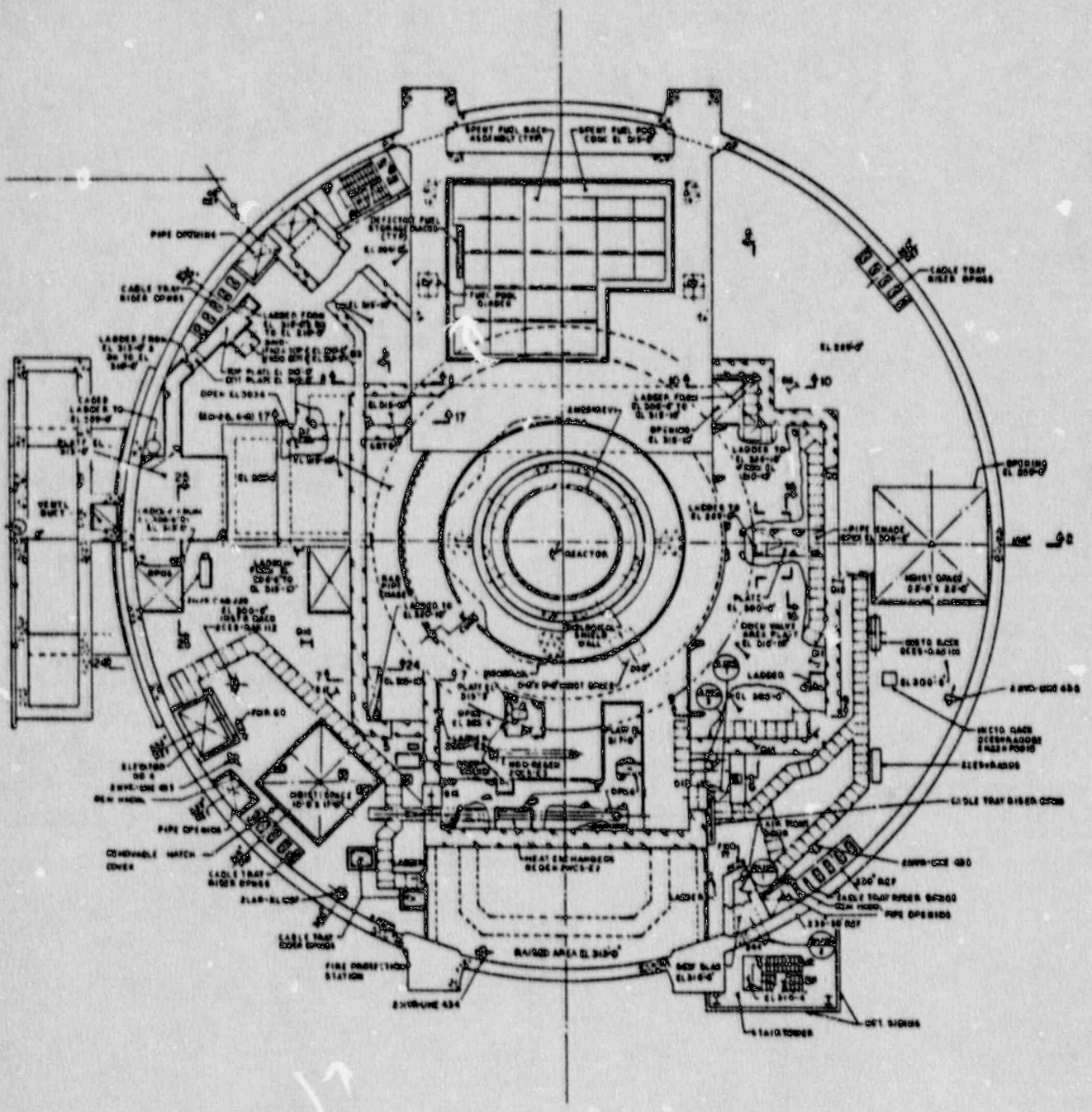


Figure 9.5-3. Plan View of the Nine Mile Point 2 Mark II Containment at the Elevation of the Reactor Core

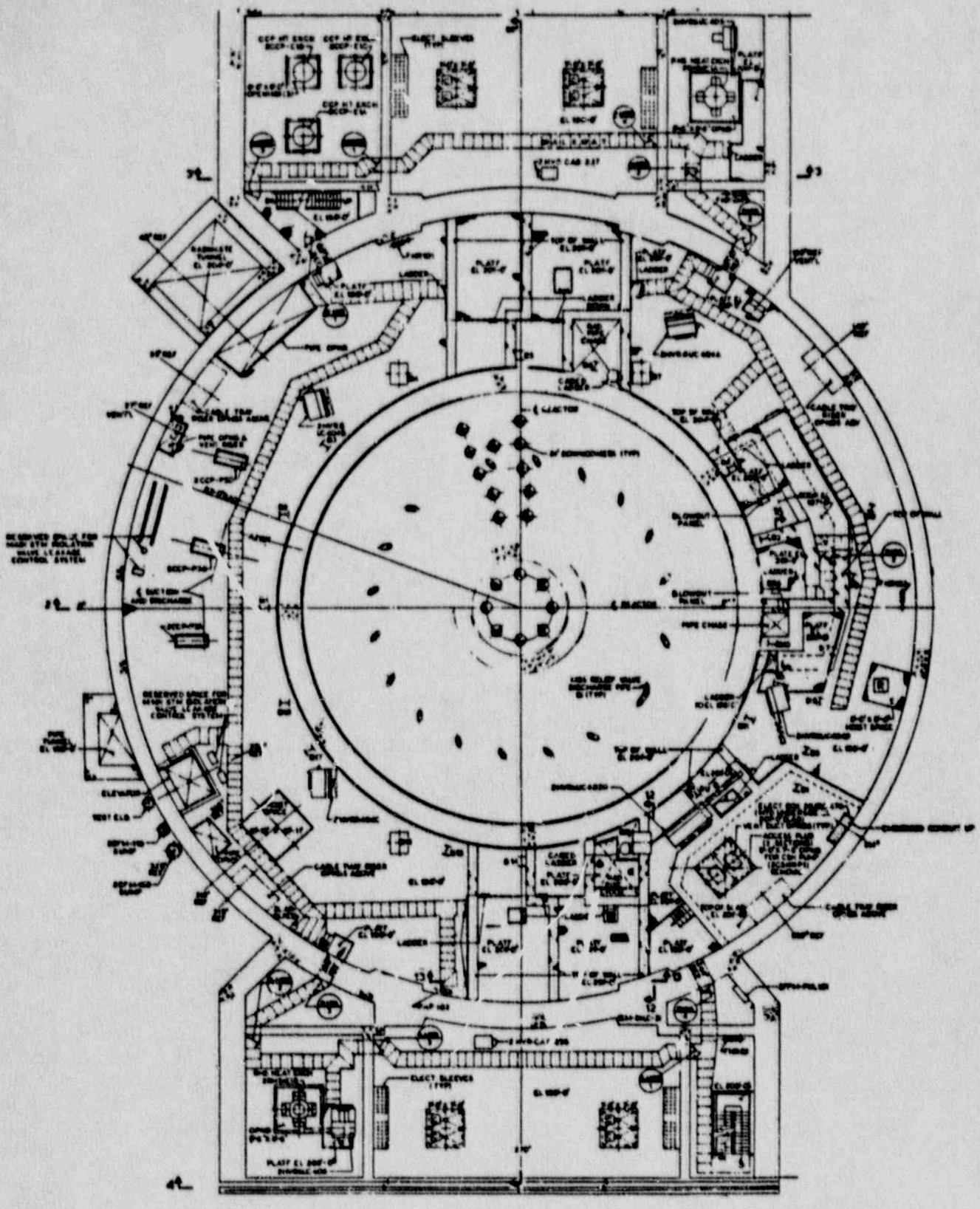


Figure 9.5-4. Plan View of the Nine Mile Point 2 Mark II Containment at the Level of the Suppression Pool

9.6 BWR/6

The six BWR/6 plants in the United States are the following:

- Clinton 1
- Grand Gulf 1 and 2
- Perry 1 and 2
- River Bend 1

All of these plants except Grand Gulf 2 and Perry 2 had full power operating licenses as of 2/89. Construction on Grand Gulf 2 and Perry 2 has been halted. There are two standard BWR/6 designs; one with a 238 inch diameter reactor vessel and one with a 251 inch reactor vessel. Clinton and River Bend use the smaller vessel, while Grand Gulf and Perry use the larger vessel.

9.6.1 Reactor Core and Reactivity Control Systems

Plants classed as BWR/6s can be grouped into the following categories based on core thermal power output:

- 2900 MWt	Clinton and River Bend
- 3600	Perry
- 3850 MWt	Grand Gulf

The lowest power level plants have cores comprised of 624 fuel assemblies, the intermediate power level plant (Perry) has a 748 fuel assembly core, and the core in the highest power level plant (Grand Gulf) consists of 800 fuel assemblies. The use of smaller diameter jet pumps allowed BWR/6 core diameter to be increased without increasing the size of the reactor vessel.

The BWR/6 introduced the use of the 8 x 8 fuel assembly which allowed core power density to be increased from 51 to 56 kw/liter while decreasing the linear heat generation rate from 18.5 to 13.4 kW/ft. The higher power density of the BWR/6 core allows the core to be smaller than a comparably rated BWR/4 or BWR/5 core. In the Clinton and River Bend plants, this enables the use of a smaller reactor vessel.

9.6.2 Reactor Coolant System, Including the Reactor Recirculation System

A typical BWR/6 nuclear boiler is shown in Figure 9.6-1. The recirculation system for BWR/6 uses constant speed reactor recirculation pumps and recirculation flow control valves as found in BWR/5 plants. The BWR/6 recirculation system uses an improved jet pump design that permits a smaller annulus between the reactor vessel and the core shroud for recirculation flow. This allows an increase in core diameter for the same size reactor vessel.

9.6.3 Reactor Vessel Level Instrumentation

Reactor vessel water level instrumentation in BWR/6 plants is comparable to that used in BWR/5 plants. Reactor vessel water level trip and alarm elevations associated with a BWR/6 are shown in Figure 9.6-2.

9.6.4 RCS Overpressure Protection System and Automatic Depressurization System (ADS)

BWR/6 plants are equipped only with safety/relief valves (SRVs) that operate mechanically for the overpressure protection function and can be power-operated for relief at lower pressures and for depressurization of the primary system. Typically, 16 to 19 Crosby SRVs are connected to the four main steam lines and discharge to the suppression

pool. Operation of this type of valve in the relief mode is controlled by an external pneumatic piston.

9.6.5 Shutdown Cooling Systems

The BWR/6 shutdown cooling capability, comprising the single-train RCIC system and the two-train RHR system, is comparable to that found in BWR/5 plants. A change made in the RHR system was to return coolant to the RCS via a feedwater line rather than a recirculation line during operation on the shutdown cooling mode.

9.6.6 Emergency Core Cooling Systems

The ECCS network for a BWR/6 plant consists of the following:

- The single-train High-Pressure Core Spray (HPCS) system which injects water from the refueling water storage tank or the suppression pool directly into the reactor vessel, via a spray sparger located above the core.
- Automatic Depressurization System (ADS) which typically uses less than half of the SRVs to depressurize the RCS
- The single-train Low-Pressure Core Spray (LPCS) system which injects water from the suppression pool directly into the reactor vessel, via a spray sparger located above the core.
- The three-train Low-Pressure Coolant Injection (LPCI) System which injects water from the suppression pool directly into the core bypass region inside the core shroud. The LPCI system is an operating mode of the RHR system. Note that RHR trains A and B have multiple modes of operation. Train C operates only in the LPCI mode.

The BWR/6 ECCS is comparable to the BWR/5 ECCS described in Section 9.5.

9.6.7 Containment

All BWR/6 plants have Mark III containments, although the construction techniques vary, as shown below:

Containment Construction	Applicable Plants
Reinforced concrete drywell, steel wetwell	Perry and River Bend
Reinforced concrete drywell and wetwell with steel liner	Clinton and Grand Gulf

All Mark III containments also have a secondary containment, as described in Section 8.

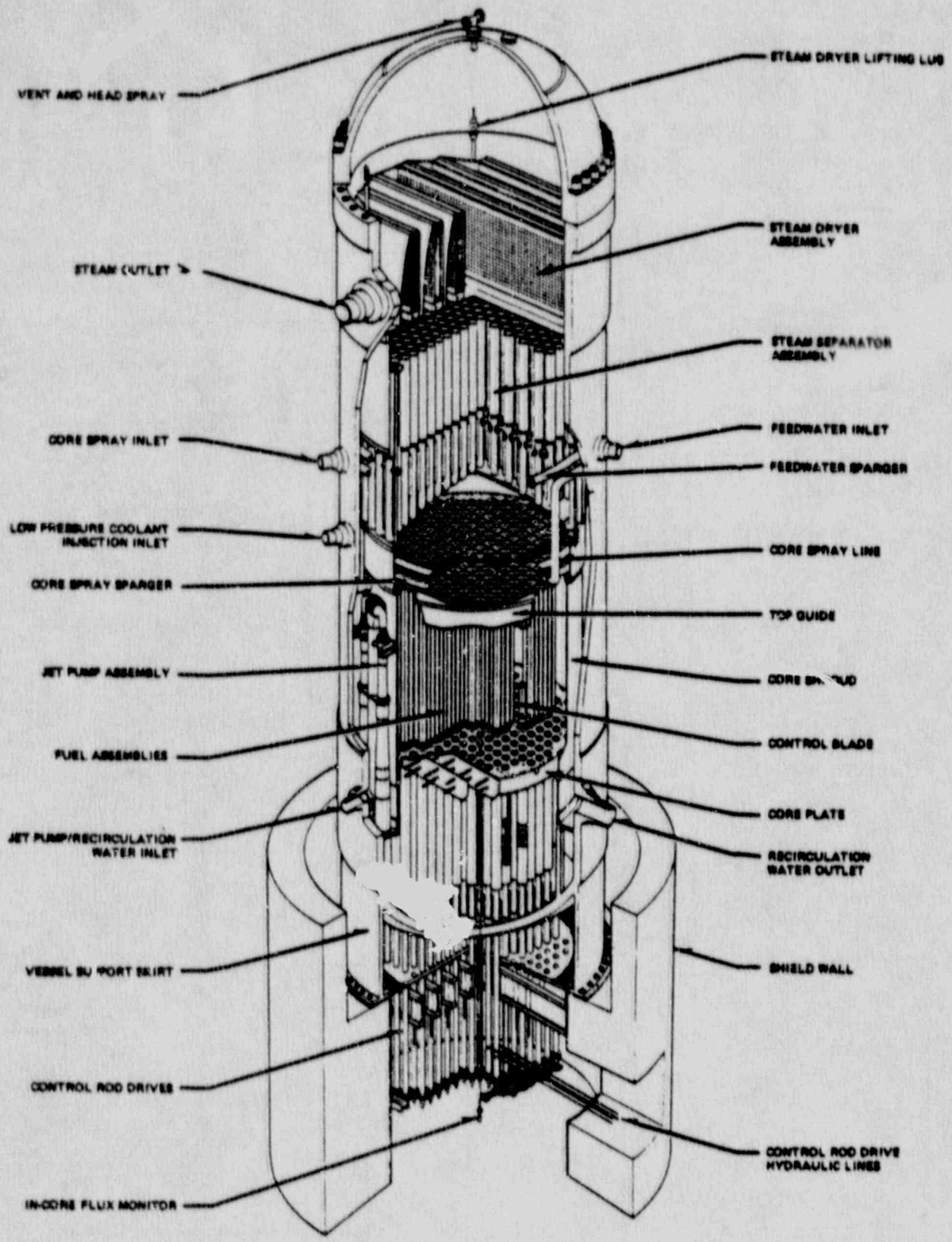


Figure 9.6-1. Typical BWR/6 Nuclear Boiler

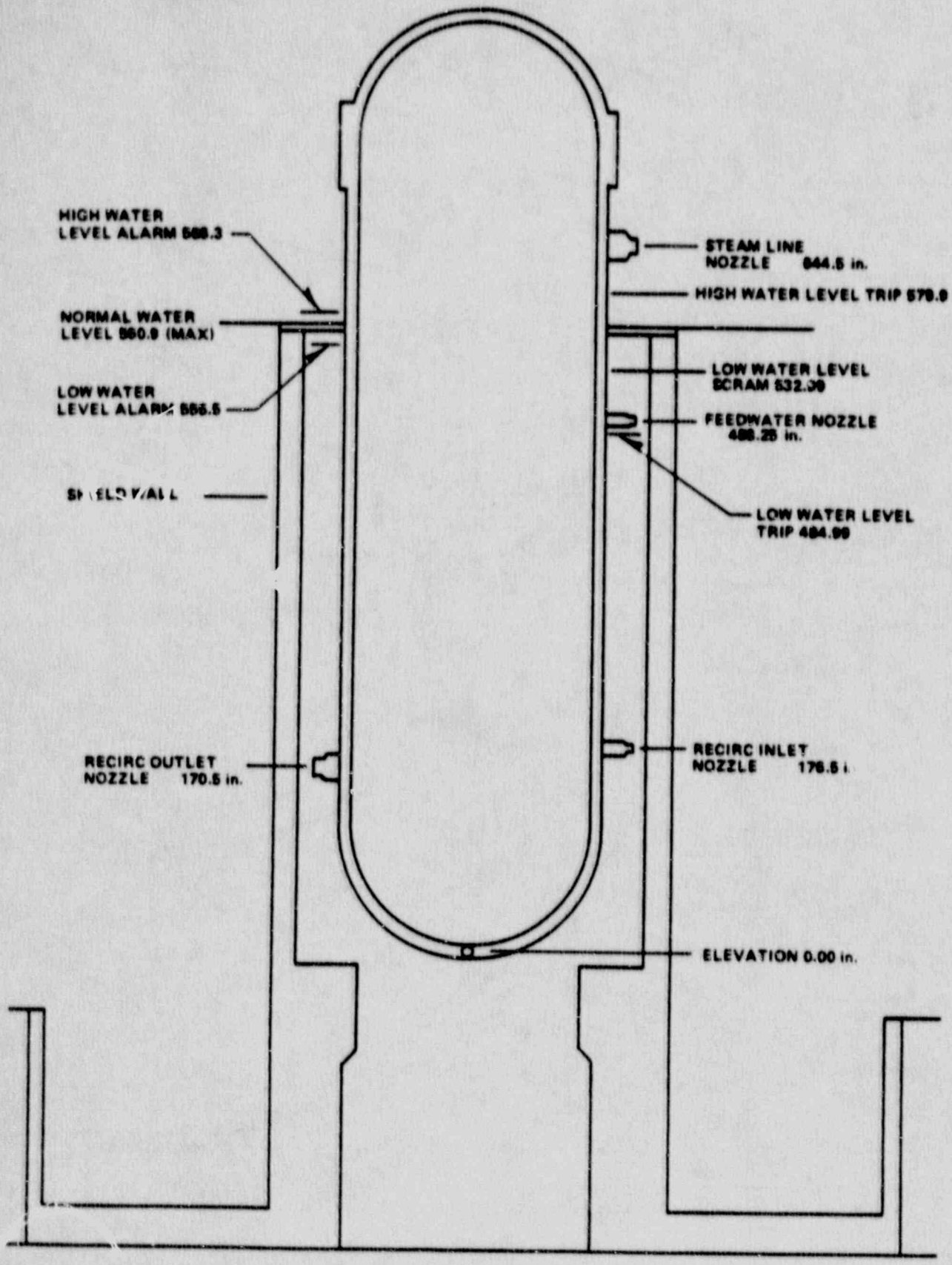


Figure 9.6-2. Water Level Trip and Alarm Elevations Associated with a BWR/6

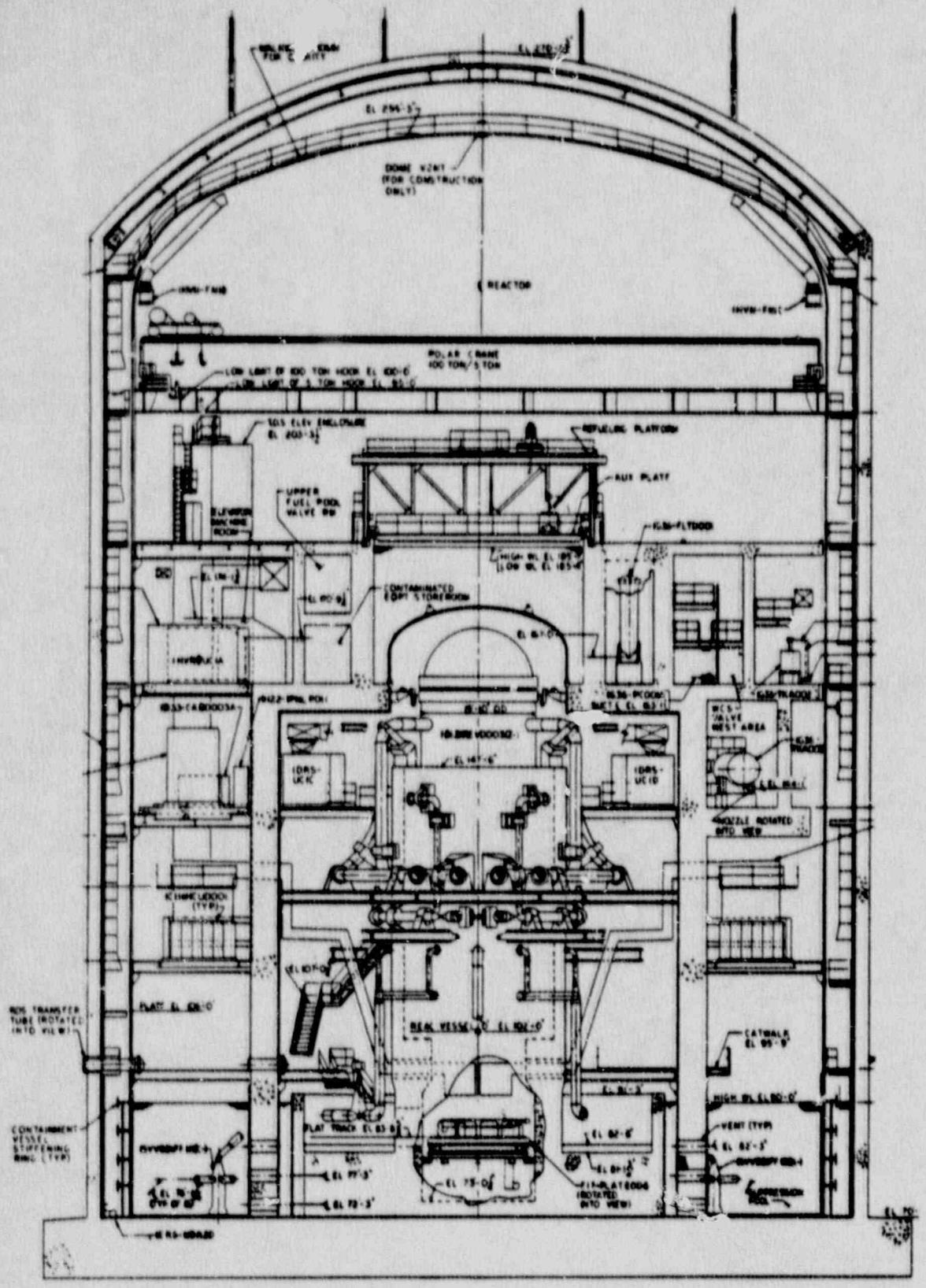


Figure 9.6-3. General Arrangement of the River Bend Mark III Containment

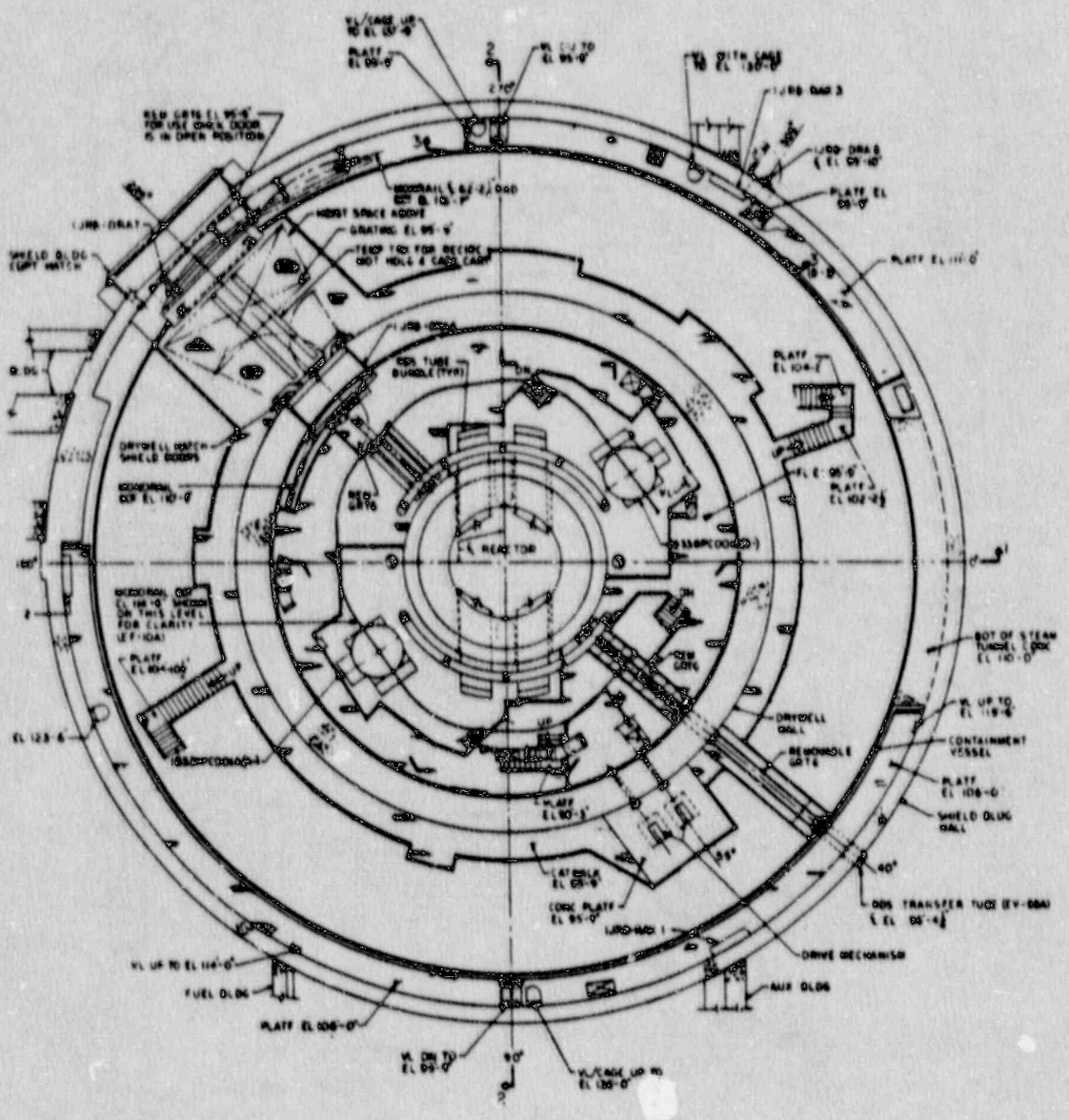


Figure 9.6-4. Plan View of the River Bend Mark III Containment at the Elevation of the Suppression Pool

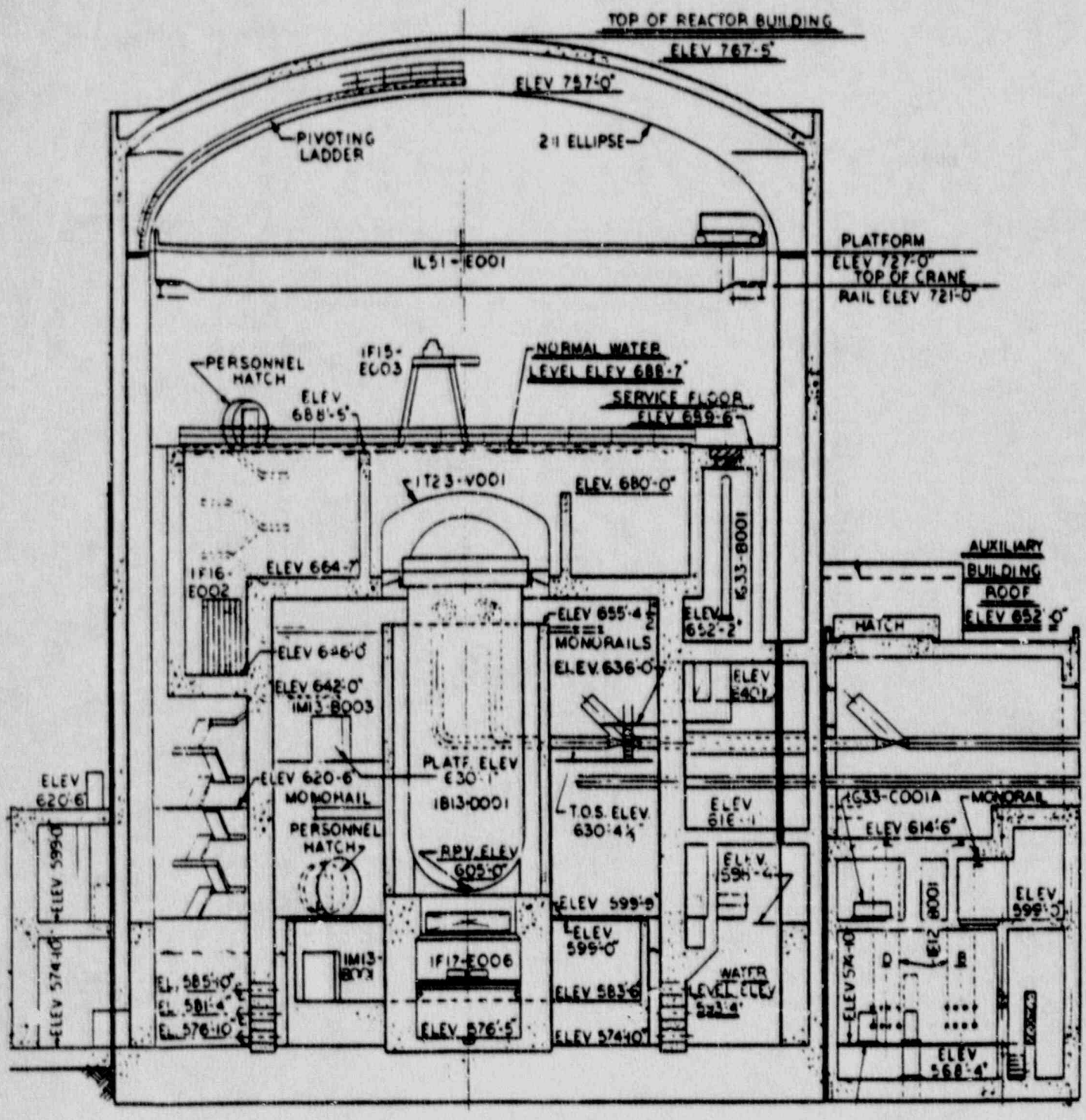


Figure 9.6-5. General Arrangement of the Perry Mark III Containment

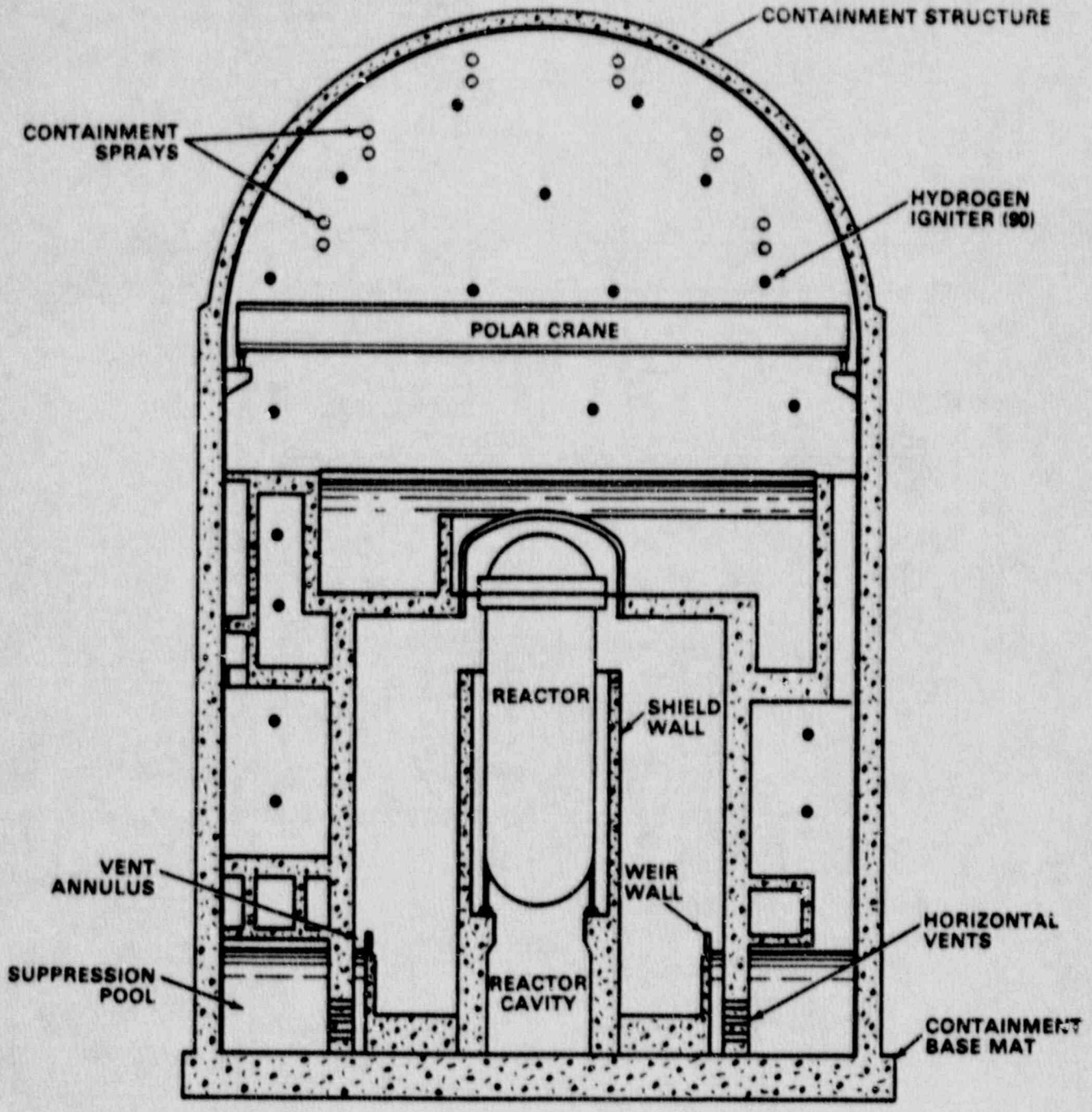


Figure 9.6-6. General Arrangement of the Grand Gulf Mark III Containment

10. COMPARATIVE BOILING WATER REACTOR (BWR) DATA

Comparisons of basic plant parameters and system design features of General Electric BWR/1 to BWR/6 plants are presented in the following tables in this section:

- Table 10-1 Design Parameters for Representative General Electric PWRs
- Table 10-2 Comparison of General electric BWR Vessel and Core Parameters
- Table 10-3 General Electric Boiling Water Reactor Plant System Comparison
- Table 10-4 Comparison of General Electric BWR Recirculation and Shutdown Cooling
- Table 10-5 Comparison of General Electric BWR Primary System Relief Capacity
- Table 10-6 Comparison of General Electric BWR Emergency Coolant Injection and Spray Systems
- Table 10-7 Comparison of General Electric BWR Containments
- Table 10-8 Comparison of General Electric BWR Backup Electric Power Systems
- Table 10-9 Comparison of General Electric BWR Power Conversion Systems

Table 10.1. Design Parameters for Representative General Electric BWRs

REACTOR PLANT CHARACTERISTICS	BWR/1 BIG ROCK POINT	BWR/2 OYSTER CREEK	BWR/3 MONTICELLO	BWR/4 PEACH BOTTOM 251-764	BWR/5 LA SALLE 251-764	BWR/6 GRAND GULF 251-784
Thermal and Hydraulic Design						
Rated Power (MWt)	157	1930	1670	3293	3323	3833
Design Power (MWt) (ECCS Design Basis)	270	1967	1670	3440	3489	4025
Steam Flow Rate (lb/hr)	Unk.	7,260,000	6,770,000	13,380,000	14,560,000	16,488,000
Core Coolant Flow Rate (lb/hr)	11,600,000	61,000,000	57,600,000	102,500,000	108,500,000	113,500,000
Feedwater Flow Rate (lb/hr)	Unk.	7,300,000	6,770,000	13,380,000	14,560,000	16,488,000
Feedwater Temperature (degrees Fahrenheit)	346	311.9	376.3	376.1	420	420
System Pressure, Nominal in Steam Dome (psia)	1035	1035	1015	1020	1020	1040
Average Power Density (kW/liter)	46	40.6	41.1	49.2	48.7	54.1
Maximum Thermal Output (kw/ft)	38	Unk.	17.5	18.5	18.5	13.40
Maximum Heat Flux (Btu/hr-ft2)	110,000	392,700	405,000	428,400	354,300	354,300
Maximum UO2 Temperature (degrees Fahrenheit)	3600	4175	2750	4493	4380	3337
Coolant Enthalpy at Core Inlet (Btu/lb)	Unk.	517.3	522.9	527.6	527.6	528.1
Core Maximum Exit Voids Within Assemblies (%)	Unk.	75	76	76	75	76
Equivalent Core Diameter (inches)	62.5	170.5	149	187.1	187.1	191.5

Table [illegible] Design Parameters for Representative General Electric BWRs (Continued)

REACTOR PLANT CHARACTERISTICS	BWR/1 BIG ROCK POINT	BWR/2 OYSTER CREEK	BWR/3 MONTICELLO	BWR/4 PEACH BOTTOM 251-764	BWR/5 LA SALLE 251-764	BWR/6 GRAND GULF 251-784
Fuel Assemblies / Control Rods						
Fuel Pellet Diameter	0.345	0.4195	0.488	0.488	0.488	0.41
Bundle Arrangement	12 x 12 (original)	7 x 7 (original)	7 x 7 (original)	7 x 7 (original)	7 x 7 (original)	8 x 8 (original)
Fuel Rods Per Fuel Assembly	132	49	49	49	49	62
Number of Fuel Assemblies	84	560	484	764	764	784
Number of Control Rods	24	137	121	185	185	193
Reactor Coolant Recirculation Design						
Recirculation Flow Control Mode	Variable Speed Recirculation Pumps	Variable Speed Recirculation Pumps	Variable Speed Recirculation Pumps	Variable Speed Recirculation Pumps	Recirculation Flow Control Valves	Recirculation Flow Control Valves
Coolant / Steam Pipe Diameter (inches)	Unk./ Unk.	26 / Unk.	28 / 18	28 / 28	24 / 26	24 / 28
Number of Recirculation Loops	2	5	2	2	2	2
Number of Recirculation Pumps	2 Centrifugal Pumps	5 Centrifugal Pumps	20 Jet Pumps	20 Jet Pumps	20 Jet Pumps	24 Jet Pumps
Inlet Pressure (psig)	Unk.	Unk.	1148	1148	1250	1250
Outlet Pressure (psig)	Unk.	Unk.	1248	1326	1430	1625
Design Temperature (Degrees Fahrenheit)	Unk.	Unk.	562	562	575	575
Pump Maximum Flow Rate (Gpm)	1,500	32,000	32,500	45,200	47,372	44,900
Pressure Vessel						
Inside Diameter (ft.)	8.83	17.75	17.17	20.92	20.92	20.92
Inside Height (ft.)	28.42	63.83	63.17	72.92	72.92	73.00
Wall Thickness of Cylinder (in.)	5.25	7.125	5.187	6.3125	6.75	6.14
Design Pressure (psig)	1700	1250	1250	1250	1250	1250

Table 10-1. Design Parameters for Representative General Electric BWRs (Continued)

REACTOR PLANT CHARACTERISTICS	BWR/1 BIG ROCK POINT	BWR/2 OYSTER CREEK	BWR/3 MONTICELLO	BWR/4 PEACH BOTTOM 251-764	BWR/5 LA SALLE 251-764	BWR/6 GRAND GULF 251-784
Core Standby Cooling Subsystems						
High Pressure Cooling / Injection Systems	Isolation Condenser	Isolation Condenser FWCI	1x RCIC-Turbine 1x HPCI-Turbine	1x RCIC-Turbine 1x HPCI-Turbine	1x RCIC-Turbine 1x HPCS-Motor	1x RCIC-Turbine 1x HPCS-Motor
RCS Depressurization System	ADS	ADS	ADS	ADS	ADS	ADS
Low Pressure Core Spray Systems	Low Pressure Recirc. via Spray Sys.	None	4 x RHR/LPCI	4 x RHR/LPCI	3x RHR/LPCI	3 x RHR/LPCI
Low Pressure Injection Systems	Fire Water System	2 x LPCS	2 x LPCS	2 x LPCS	1 x LPCS	1 x LPCS
Containment						
Type	Large Dry Primary Containment	Mark I Pressure Suppression Primary Containment Plus Secondary Containment	Mark I Pressure Suppression Primary Containment Plus Secondary Containment	Mark I Pressure Suppression Primary Containment Plus Secondary Containment	Mark II Pressure Suppression Primary Containment Plus Secondary Containment	Mark III Pressure Suppression Primary Containment Plus Secondary Containment
Inerting	Not Required	Required	Required	Required	Required	Not Required
Drywell - Construction	Not Applicable	Steel Vessel, Light Bulb Shaped	Steel Vessel, Light Bulb Shaped	Steel Vessel, Light Bulb Shaped	Concrete Frustrum of Cone	Concrete Cylinder
Drywell Design Temperature (°F)	235	175/281	281	281	340	330
Drywell Internal Design Puressure (psig)	27	62/35	56	56	45	30
Drywell Free Air Volume (ft3)	1,150,000	180,000	134,200	175,800	221,500	270,000
Suppression Chamber - Construction	Not Applicable	Steel Vessel, Torus Shaped	Steel Vessel, Torus Shaped	Steel Vessel, Torus Shaped	Concrete Vessel, Cylinder Under Drywell	Steel Vessel, Cylinder Around Drywell
Suppression Chamber Design Temp. (°F)	Not Applicable	150	281	281	275	185
Suppression Chamber Design Press. (psig)	Not Applicable	35	56	52	45	15
Suppression Chamber Water Volume (ft3)	Not Applicable	87,900	77,970	123,000	109,100	136,000
Suppression Chamber Air Volume (ft3)	Not Applicable	128,000	108,250	127,700	166,400	1,264,000
Design Leak Rate (vol%/day)	0.5	0.5	0.5	0.5	0.5	0.35

Table 10-2. Comparison of General Electric BWR Vessel and Core Parameters

BWR 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
BWR/1	Big Rock Point	240	106	62.5	70	46	84	12 x 12	24
BWR/2	Nine Mile Point 1	1850	213	167.2	144	40.6	532	8 x 8	129
BWR/2	Oyster Creek	1930	213	170.5	144	40.6	560	8 x 8	137
BWR/3	Monticello	1670	206	149	144	41.1	484	8 x 8	121
BWR/3	Millstone 1	2011	224	163.1	144	40.8	580	8 x 8	145
BWR/3	Pilgrim 1	1998	224	163.1	144	40.2	580	8 x 8	145
BWR/3	Dresden 2 and 3	2527	251	182.2	144	41.1	724	9 x 9	177
BWR/3	Quad Cities 1 and 2	2511	251	182.2	144	40.6	724	8 x 8	177
BWR/4	Duane Arnold	1658	183	129.9	144	50.9	368	8 x 8	89
BWR/4	Vermont Yankee	1593	206	131.5	144	50.9	368	8 x 8	89
BWR/4	Cooper	2381	218	158.5	150	51.8	548	8 x 8	137
BWR/4	Brunswick 1 and 2	2436	218	160.2	150	50.5	560	8 x 8	137
BWR/4	Fitzpatrick	2436	218	160.2	150	50.1	560	8 x 8	137
BWR/4	Hatch 1 and 2	2436	218	160.2	150	49.2	560	8 x 8	137
BWR/4	Shorsham	2436	218	160.2	150	49.2	560	8 x 8	137
BWR/4	Browns Ferry 1, 2 & 3	3293	251	187.1	150	50	764	8 x 8	185
BWR/4	Fermi 2	3293	251	187.1	150	48.7	764	8 x 8	185
BWR/4	Hope Creek 1 and 2	3293	251	187.1	150	48.7	764	8 x 8	185
BWR/4	Limerick 1 and 2	3293	251	187.1	150	50.8	764	8 x 8	185

Table 10-2. Comparison of General Electric BWR Vessel and Core Parameters (Continued)

BWR 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
BWR/4	Peach Bottom 2 & 3	3293	251	187.1	150	49.2	764	8 x 8	185
BWR/4	Susquehanna 1 and 2	3292	251	187.1	150	48.7	764	8 x 8	185
BWR/5	LaSalle 1 and 2	3293	251	187.1	150	48.7	764	8 x 8	185
BWR/5	Nine Mile Point 2	3323	251	187.1	150	49.2	764	8 x 8	185
BWR/5	WNP-2	3323	251	187.1	150	49.2	764	8 x 8	185
BWR/6	Clinton	2894	218	169.1	150	52.4	624	8 x 8	145
BWR/6	River Bend 1	2894	218	169.1	150	52.4	624	8 x 8	145
BWR/6	Perry 1 and 2	3578	238	185.2	150	54	748	8 x 8	177
BWR/6	Grand Gulf	3833	251	191.5	150	54.1	800	8 x 8	193

Table 10-3. General Electric Boiling Water Reactor Plant System Comparison

				Systems Capable of Core Coolant Injection at High Pressure						Systems Capable of Core Coolant Injection at Low Pressure					Systems for Shutdown Cooling Only		Systems for Containment & Supp. Pool Cooling		RCS Pressure Relief Valves (Overpressure Protection and/or Depressurization)			
Plant	BWR Type	Core MWt	Cont. Type	FWCI	HPCI	HPCS	RCIC	RCIC-SC	CRDHS	LPCS	LPCI	RHR-M1	RHR-M2	OTHER	ISO COND	RHR-S	CS-S	CS-M	SV	SRV	PARV	ADS
Big Rock Point	1	240	Dry											(C)				(D)				
Oyster Creek	2	1930	MK-I																			
Nine Mile Point 1	2	1850	MK-I																			
Millstone 1	3	2011	MK-I															(E)				
Dresden 2 & 3	3	2527	MK-I															(E)				
Pilgrim 1	3	1998	MK-I															(F)				
Monticello	3	1670	MK-I															(F)				
Quad Cities 1 & 2	3	2511	MK-I															(F)				
Browns Ferry 1, 2 & 3	4	3293	MK-I															(F)				
Vermont Yankee	4	1593	MK-I															(F)				
Peach Bottom 2 & 3	4	3293	MK-I															(F)				
Hatch 1	4	2436	MK-I															(F)				
Hatch 2	4	2436	MK-I															(F)				
Cooper	4	2381	MK-I															(F)				
Duane Arnold	4	1658	MK-I															(F)				
Fitzpatrick	4	2436	MK-I															(F)				
Brunswick 1 & 2	4	2436	MK-I															(F)				
Fermi 2	4	3293	MK-I				(H)					(H)						(F)				
Hope Creek 1	4	3293	MK-I															(F)				
Limerick 1 & 2	4	3293	MK-II															(F)				
Shoreham	4	2436	MK-II															(F)				
Susquehanna 1 & 2	4	3292	MK-II															(F)				
WNP-2	5	3323	MK-II							(G)								(F)				
LaSalle 1 & 2	5	3293	MK-II															(F)				
Nine Mile Point 2	5	3323	MK-II							(G)								(F)				
Grand Gulf 1 & 2	6	3833	MK-III							(G)								(F)				
Perry 1 & 2	6	3579	MK-III							(G)								(F)				
Clinton 1	6	2894	MK-III							(G)								(F)				
River Bend 1	6	2894	MK-III							(G)								(F)				

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Table 10-3. General Electric Boiling Water Reactor Plant System Comparison (continued).

Notes:

(A) Data compiled from a variety of sources including NUREG-0626, NEDO-24708, NUREG/CR-2585, NUREG/CR-3713 and ORNL-NSIC-55.

(B) System abbreviations are defined as follows:

FWCI	=	feedwater coolant injection system
HPCI	=	high pressure coolant injection system
HPCS	=	high pressure core spray system
RCIC	=	reactor core isolation cooling system (single mode system)
RCIC-SC	=	similar to RCIC plus steam condensing operation with RHR-M2
CRDHS	=	control rod drive hydraulic system
LPCS	=	low pressure core spray system
LPCI	=	low pressure coolant injection system (single mode system)
RHR-M1	=	multi-mode RHR system performing LPCI, shutdown cooling, suppression pool cooling and containment spray functions
RHR-M2	=	same as RHR-M1 plus steam condensing operation -SC
ISO COND	=	isolation condenser system for shutdown cooling at h re
RHR-S	=	single-mode residual heat removal system for shutdc g at low pressure
CS-M	=	containment spray, which is an operating mode of sor r multi-mode system
CS-S	=	single-mode containment spray system
SV	=	safety valves which open mechanically on high system pressure
SRV	=	safety/relief valves which will open mechanically on high system pressure or can be opened electrically at lower pressures
PARV	=	power actuated (Electromatic type) relief valves which require electrical power for actuation at any system pressure
ADS	=	automatic depressurization system which utilizes some or all SRVs and PARVs

(C) Fire protection system can be aligned for low pressure coolant injection

(D) Containment/suppression pool cooling is an operating mode of the LPCS system

(E) Containment/suppression pool cooling is an operating mode of the LPCI system

(F) Containment/suppression pool cooling is an operating mode multi-mode RHR system (RHR-M1 or RHR-M2)

(G) Only a single 100% capacity LPCS pump. Other plants typically have two 100% capacity LPCS trains

(H) Piping for RCIC/RHR steam condensing operation removed

Table 10-4. Comparison of General Electric BWR Recirculation and Shutdown Cooling

			Recirculation System		RHR System					RHR Service Water System		Core Isolation Cooling System	
BWR Reactor Plant	BWR Type	Power MWt	# Jet Pumps	Flow Rate/Pump (gpm)	# Loops	# Pumps	Flow Rate/Pump (gpm @ psid)	# HX	Max. Capacity per HX (Btu/hr)	# Pumps	Flow Rate/Pump (gpm @ psid)	Type	Capacity
Big Rock Point	1	240	2 (cent.)	18000	2	2	500 @ 75	2	unk.	2		Isolation Condenser	1 x 32E+6 Btu/hr
Nine Mile Point 1	2	1850	5 (cent.)	36000	1/3	3	unk.	3	12.5E+6	4	2500 @ ?	Isolation Condenser	4 x 95E+6 Btu/hr
Oyster Creek 1	2	1930	5 (cent.)	36000	1/3	3	unk.	3	11E+6	2	3400 @ ?	Isolation Condenser	2 x 205E+6 Btu/hr
Dresden 2 & 3	3	2527	20	45200	2/3	3	6750 @ ?	3	27E+6	4	3500 @ ?	Isolation Condenser	1 x 252.5E+6 Btu/hr
Millstone 1	3	2011	20	45200	2	2	2600 @ ?	2	40E+6	2	2600 @ ?	Isolation Condenser	1 x 206E+6 Btu/hr
Monticello	3	1670	20	42500	2	4	4000 @ 20	2	24E+5	4	3500 @ ?	RCIC	400 gpm @ 1120 psid
Pilgrim 1	3	1998	20	45200	2	4	4500 @ 20	2	64E+6	6	1700 @ 43	RCIC	400 gpm @ 1120 psid
Quad Cities 1 & 2	3	2511	20	45000	2	4	4830 @ 20	2	unk.	4	7000 @ 325	RCIC	400 gpm @ 1120 psid
Browns Ferry 1, 2 & 3	4	3293	20	45200	2	4	10000 @ 20	4	70E+6	8	4500 @ ?	RCIC	616 gpm @ 1120 psid
Brunswick 1 & 2	4	2436	20	45200	2	4	7700 @ 20	2	90.8E+6	4	4000 @ 107	RCIC	400 gpm @ 1214 psid
Cooper	4	2381	20	45200	2	4	7700 @ 20	3	70E+6	4	4000 @ 347	RCIC	400 gpm @ 1500 psid
Duane Arnold	4	1593	18	27100	2	4	4800 @ 20	2	35E+6	4	2400 @ 292	RCIC	416 gpm @ 1120 psid
Fermi 2	4	3293	20	45200	2	4	1000 @ 20	2	41.6E+6	4	4500 @ 80	RCIC	616 gpm @ 1500 psid
Fitzpatrick	4	2436	20	45200	2	4	7710 @ 20	2	70E+6	4	4000 @ 116	RCIC	600 gpm @ 1214 psid
Hatch 1	4	2436	20	45200	2	4	7700 @ ?	2	32E+6	4	4000 @ 414	RCIC	400 gpm @ 1120 psid
Hatch 2	4	2436	20	45200	2	4	7700 @ ?	2	32E+6	4	4000 @ 414	RCIC	416 gpm @ 1136 psid
Hope Creek 1	4	3293	20	45200	2	4	10000 @ 20	2	unk.	4	18500 @ 99	RCIC	616 gpm @ 1500 psid
Limerick 1 & 2	4	3293	20	45200	2	4	10000 @ 20	2	unk.	3	12000 @ ?	RCIC	616 gpm @ 1120 psid
Peach Bottom 2 & 3	4	3293	20	45200	2	4	10000 @ 20	4	70E+6	3	14000 @ ?	RCIC	616 gpm @ 1120 psid
Shoreham	4	2436	20	45200	2	4	7700 @ 20	2	70.9E+6	4	8600 @ 64	RCIC	400 gpm @ 1120 psid
Susquehanna 1 & 2	4	3293	20	45200	2	4	10000 @ 20	2	unk.	2	9000 @ ?	RCIC	600 gpm @ 1500 psid
Vermont Yankee	4	1593	20	32500	2	4	7200 @ 20	2	57.5E+6	4	2700 @ ?	RCIC	400 gpm @ 1120 psid

Table 10-4. Comparison of General Electric BWR Recirculation and Shutdown Cooling (Continued)

Plant Name	BWR Type	Power MWt	Recirculation System		RHR System					RHR Service Water System		Core Isolation Cooling System	
			# Jet Pumps	Flow Rate/Pump (gpm)	# Loops	# Pumps	Flow Rate/Pump (gpm @ psid)	# HX	Max. Capacity per Hx (Btu/hr)	# Pumps	Flow Rate/Pump (gpm @ psid)	Type	Capacity
LaSalle 1 & 2	5	3293	20	45200	2	2	7450 @ ?	2	unk.	4	7400 @ ?	RCIC	625 gpm @ 1120 psid
Nine Mile Point 2	5	3323	20	47200	2	2	7450 @ 20	2	unk.	6	10000 @ 80	RCIC	625 gpm @ 1500 psid
[illegible] 2	5	3323	20	45200	2	2	7087 @ 20	2	unk.	3	10500 @ 217	RCIC	600 gpm @ 1120 psid
Clinton 1	6	2894	20	32500	2	2	5050 @ 20	2	116.7E+6	2 1	16500 @ ? 1100 @ ?	RCIC	600 gpm @ 1177 psid
Grand Gulf 1 & 2	6	3833	24	44800	2	2	7450 @ 20	2x2	184.7E+6 per 2 Hx set	2 1	12000 @ ? 1300 @ ?	RCIC	600 gpm @ 1200 psid
Perry 1 & 2	6	3579	20	42000	2	2	7100 @ ?	2x2	unk.	2 1	11500 @ 69 930 @ 69	RCIC	700 gpm @ 1177 psid
River Bend 1	6	2894	20	32500	2	2	5120 @ 20	2x2	unk.	4	7800 @ 74	RCIC	600 gpm @ 1170 psid

Note: Maximum RHR heat exchanger capacity applies during post-accident cooling.

Table 16-5. Comparison of General Electric BWR Primary System Relief Capacity

BWR Reactor Plant	BWR Type	# of Safety Valves	# Safety Relief Valves	# Powered Relief Valves	Lowest Setpoint (psig)		Capacity (Klb/hr)		Notes
					Safety	Relief	Safety	Relief	
Big Rock Point	1	6	0	4	1585	1535 to 1585	240	518	Relief valves used by reactor depressurization sys. only
Nine Mile Point 1	2	16	0	6	1218 to 1254	1090 to 100	633 to 651	600	Electromatic relief valves
Oyster Creek 1	2	16	0	5	1212 to 1239	1050	634	600	
Dresden 2 & 3	3	8	1	4	1115 to 1260	1115 to 1135	622 to 655	540 to 622	Electromatic relief valves
Millstone 1	3	0	6	0	1095 to 1125	1095 to 1125	800	800	
Monticello	3	2	6	0	unk.	1108	unk.	840	
Pilgrim 1	3	2	3	0	1240	1095	640	800	
Quad Cities 1 & 2	3	8	1	4	1124 to 1260	1080 to 1124	598 to 644	558 to 598	Electromatic relief valves Target Rock SRV
Browns Ferry 1, 2, & 3	4	0	13	0	1080 to 1100	1030 to 1100	850 to 867	850 to 867	
Brunswick 1 & 2	4	0	11	0	1105 to 1125	1105 to 1125	830 to 845	830 to 845	
Cooper	4	3	8	0	1080 to 1240	1080 to 1100	642 to 870	870	
Duane Arnold	4	2	6	0	1240	1080	642	830	
Fermi 2	4	0	15	0	1110 to 1130	1110 to 1130	884 to 900	884 to 900	
Fitzpatrick	4	0	11	0	1090 to 1140	1090 to 1140	798 to 855	798 to 855	
Hatch 1	4	0	11	0	1080 to 1100	1080 to 1100	870	870	
Hatch 2	4	2	9	0	1080 to 1100	1080 to 1100	870	870	
Hope Creek 1 & 2	4	0	14	0	1108 to 1130	1108 to 1130	884 to 901	884 to 901	
Limerick 1 & 2	4	0	14	0	1130 to 1150	1130 to 1150	901.5 to 917	901.5 to 917	

Table 16-5. Comparison of General Electric BWR Primary System Relief Capacity (Continued)

BWR Reactor Plant	BWR Type	# of Safety Valves	# Safety Relief Valves	# Powered Relief Valves	Lowest Setpoint (psig)		Capacity (Klb/hr)		Notes
					Safety	Relief	Safety	Relief	
Peach Bottom 2 & 3	4	2	11	0	1230	1105	925	819	
Shoreham	4	0	11	0	1115	1115	889	889	
Susquehanna 1 & 2	4	0	16	0	1146	1146	845	845	
Vermont Yankee	4	2	4	0	1240	1080	925	800	
LaSalle 1 & 2	5	0	17	0	1076	1076	850	850	
Nine Mile Point 2	5	0	18	0	1148 to 1205	1148 to 1205	882 to 925	882 to 925	
WNP-2	5	0	18	0	1177 to 1217	1177 to 1217	864 to 906	864 to 906	
Clinton 1	6	0	16	0	1103 to 1123	1103 to 1123	895 to 913	895 to 913	
Grand Gulf 1 & 2	6	0	20	0	1165 to 1190	1165 to 1190	895 to 913	895 to 913	
Perry 1 & 2	6	0	19	0	1165 to 1190	1165 to 1190	895 to 913	895 to 913	
River Bend 1	6	0	16	0	1123	1103	913	895	

Note:
1. Safety valves open mechanically at a predetermined high pressure setpoint.
2. Power-actuated relief valves require power for operation.
3. Safety/relief valves open mechanically at a pre-determined high pressure setpoint and can be power actuated to open at a lower pressure.

Table 10-6. Comparison of General Electric BWR Emergency Coolant Injection and Spray Systems

				High Pressure Injection System		Low Pressure Injection System			Low Pressure Core Spray System		
BWR Reactor Plant	BWR Type	Containment Type	Power MWt	# Pumps	Flow Rate / Pump (gpm @ psid)	# Loops	# of pumps	Flow Rate/pump (gpm @ psid)	# Loops	# Pumps	Flow Rate/pump (gpm @ psid)
Big Rock Point	1	Dry	240	None	-----	None	---	-----	2	2	400 (fire pumps)
Oyster Creek 1	2	Mk I	1930	None	-----	2	4	3700 @ 195	2	4	3400 @ 285
Nine Mile Point 1	2	Mk I	1850	2 X FWCI	3800 @ 1000	2	4	3400 @ 302	2	4	3400 @ 113
Millstone 1	3	Mk I	2011	3 X FWCI	8000 @ 100 to 1125	2	4	5000 @ 0	2	2	3600 @ 90
Dresden 2 & 3	3	Mk I	2527	1 X HPCI	5600 @ 165 to 1135	2	4	5350 @ 0	2	2	4500 @ 90
Monticello	3	Mk I	1670	1 X HPCI	3000 @ 135 to 1135	2	4	4000 @ 20	2	2	3020 @ 307
Pilgrim 1	3	Mk I	1998	1 X HPCI	4250 @ 135 to 1135	2	4	4800 @ 20	2	2	3020 @ 307
Quad Cities 1 & 2	3	Mk I	2511	1 X HPCI	5600 @ 150 to 1125	2	4	4830 @ 20	2	2	4500 @ 90
Browns Ferry 1, 2 & 3	4	Mk I	3293	1 X HPCI	5000 @ 1120	2	4	10000 @ 20	2	2	3125 @ 122
Brunswick 1 & 2	4	Mk I	2436	1 X HPCI	4250 @ 150 to 1120	2	4	7700 @ 20	2	2	4700 @ 113
Cooper	4	Mk I	2381	1 X HPCI	4250 @ 150 to 1140	2	4	7700 @ 20	2	2	4500 @ 113
Duane Arnold	4	Mk I	1593	1 X HPCI	3000 @ 150 to 1120	2	4	4800 @ 20	2	2	3020 @ 113
Fermi 2	4	Mk I	3293	1 X HPCI	5000 @ 1120	2	4	10000 @ 20	2	2	6250 @ 122
Fitzpatrick	4	Mk I	2436	1 X HPCI	4250 @ 150 to 1120	2	4	7710 @ 20	2	2	4825 @ 113
Hatch 1 & 2	4	Mk I	2436	1 X HPCI	4250 @ 150 to 1120	2	4	7700 @ 20	2	2	4825 @ 120
Hope Creek 1 & 2	4	Mk I	3293	1 X HPCI	500 @ 1140	2	4	10000 @ 20	2	2	6250 @ 122
Limerick 1 & 2	4	Mk II	3293	1 X HPCI	5600 @ ?	2	4	10000 @ 20	2	4	6350 @ 105
Peach Bottom 2 & 3	4	Mk I	3293	1 X HPCI	5000 @ 1120	2	4	10000 @ 20	2	4	3125 @ 122
Shoreham	4	Mk II	2436	1 X HPCI	4250 @ 150 to 11[illegible]	2	4	7700 @ 20	2	2	4725 @ 123
Susquehanna 1 & 2	4	Mk II	3293	1 X HPCI	5000 @ 150 to 1120	2	4	10000 @ 20	2	4	3125 @ 122
Vermont Yankee	4	Mk I	1593	1 X HPCI	4250 @ 1120	2	4	7200 @ 20	2	2	3000 @ 120

Table 10-6. Comparison of General Electric BWR Emergency Coolant Injection and Spray Systems (Continued)

BWR Reactor Plant	BWR Type	Containment Type	Power MWt	High Pressure Injection System		Low Pressure Injection System			Low Pressure Core Spray System		
				# Pumps	Flow Rate / Pump (gpm @ psid)	# Loops	# of pumps	Flow Rate/pump (gpm @ psid)	# Loops	# Pumps	Flow Rate/pump (gpm @ psid)
LaSalle 1 & 2	5	Mk II	3293	1 X HPCS	1650 @ 1110	3	3	7450 @ 20	1	1	6250 @ 122
Nine Mile Point 2	5	Mk II	3323	1 X HPCS	1650 @ 1110	3	3	7450 @ 20	1	1	6250 @ 122
WNP-2	5	Mk II	3323	1 X HPCS	1650 @ 1110	3	3	7087 @ 20	1	1	6250 @ 122
Clinton 1	6	Mk III	2894	1 X HPCS	1400 @ 1147	3	3	5050 @ 20	1	1	5010 @ 119
Grand Gulf 1 & 2	6	Mk III	3833	1 X HPCS	1500 @ 1130	3	3	7450 @ 20	1	1	7000 @ 122
Perry 1 & 2	6	Mk III	3579	1 X HPCS	1485 @ 1130	3	3	7100 @ ?	1	1	6110 @ ?
River Bend 1	6	Mk III	2894	1 X HPCS	1400 @ 1147	3	3	5010 @ 20	1	1	5010 @ 119

Table 10-7. Comparison of General Electric BWR Containments

Plant Name	BWR Type	Primary Cont. Type	Drywell Material & Construction	Drywell Free Volume (ft3)	Drywell Design Temp (deg F)	Wetwell Material & Construction	Wetwell Min. Free Vol. (ft3)	Wetwell Min. Water Vol. (ft3)	Wetwell Design Temp (deg F)	Design Pressure (psig)	Design Leak Rate % vol/day
Big Rock Point	1	Dry	Steel Sphere	1,150,000	235	Not Applicable	Not Applicable	Not Applicable	Not Applicable	27	0.5
Nine Mile Point 1	2	Mk I	Steel	180,000	310	Steel	120,000	69,000	205	62 Drywell 35 Torus	0.5
Oyster Creek	2	Mk I	Steel	180,000	281	Steel	128,000	77,970	150	62 Drywell 35 Torus	0.5
Dresden 2 & 3	3	Mk I	Steel	158,236	281	Steel	117,245	112,203	281	62	0.5
Millstone 1	3	Mk I	Steel	146,900	281	Steel	190,900	83,500	281	62	0.5
Monticello	3	Mk I	Steel	134,200	281	Steel	108,250	77,970	281	56	0.5
Pilgrim 1	3	Mk I	Steel	147,000	281	Steel	110,000	84,000	281	56	0.5
Quad Cities 1 & 2	3	Mk I	Steel	158,236	281	Steel	117,000	115,600	281	56	0.5
Browns Ferry 1, 2 & 3	4	Mk I	Steel	159,000	281	Steel	119,000	85,000	281	56	0.5
Brunswick 1 & 2	4	Mk I	Reinf. Conc.	164,100	300	Reinf. Conc W/ Steel liner	124,000	87,600	220	62	0.5
Cooper	4	Mk I	Steel	145,430	281	Steel	109,810	87,660	281	56	0.5
Duane Arnold	4	Mk I	Steel	130,930	281	Steel	94,630	61,500	281	56	0.5
Fermi 2	4	Mk I	Steel	163,730	340	Steel	127,760	121,080	281	56	0.5
Fitzpatrick	4	Mk I	Steel	150,000	309	Steel	114,000	105,000	220	56	0.5
Hatch 1 & 2	4	Mk I	Steel	146,240	281	Steel	110,950	87,660	281	56	1.2
Hope Creek 1 & 2	4	Mk I	Steel	169,000	340	Steel	133,500	122,000	300	62	0.5
Limerick 1 & 2	4	Mk II	Reinf. Conc.	243,580	281	Reinf. Conc. w/ Steel liner	147,670	134,600	281	56	0.5
Peach Bottom 2 & 3	4	Mk I	Steel	175,800	281	Steel	127,700	123,000	281	56	0.5
Shoreham	4	Mk II	Reinf. Conc.	192,500	296	Reinf. Conc. w/ Steel liner	134,000	81,350	225	48	0.5
Susquehanna 1 & 2	4	Mk II	Reinf. Conc.	239,600	340	Reinf. Conc. w/ Steel liner	148,590	131,550	220	53	0.5
Vermont Yankee	4	Mk I	Steel	134,200	281	Steel	108,250	77,970	281	56	0.5

Table 10-7. Comparison of General Electric BWR Containments (Continued)

Plant Name	BWR Type	Primary Cont. Type	Drywell Material & Construction	Drywell Free Volume (ft3)	Drywell Design Temp (deg F)	Wetwell Material & Construction	Wetwell Min. Free Vol. (ft3)	Wetwell Min. Water Vol. (ft3)	Wetwell Design Temp (deg F)	Design Pressure (psig)	Design Leak Rate % vol/day
LaSalle 1 & 2	5	Mk II	Reinf. Conc.	209,300	340	Post-tens. Conc. w/ Steel liner	164,500	124,000	275	45	0.5
Nine Mile Point 2	5	Mk II	Reinf. Conc.	303,418	293	Reinf. Conc. w/ Steel liner	192,028	154,794	212	45	1.1
WNP-2	5	Mk II	Reinf. Conc.	200,540	340	Free-Standing Steel	144,184	112,197	270	45	0.5
Clinton 1	6	Mk III	Reinf. Conc.	250,000	330	Reinf. Conc W/ Steel liner	1,075,300	135,700	185	15	0.65
Grand Gulf 1 & 2	6	Mk III	Reinf. Conc.	270,000	330	Reinf. Conc. w/ Steel liner	1,264,000	136,000	185	15	0.35
Perry 1 & 2	6	Mk III	Reinf. Conc	278,000	330	Steel	802,000	120,000	185	15	0.2
River Bend 1	6	Mk III	Reinf. Conc.	251,000	330	Steel	713,000	127,930	185	15	0.26

Table 10-8. Comparison of General Electric BWR Backup Electric Power Systems

Reactor Plant	BWR Type	Shared Diesels per Plant	Dedicated Diesels per Unit	Continuous rating (kW)	Diesel Manufacturer	# of Batteries per Plant	Voltage	Notes
Big Rock Point	1	None	1	200	Caterpiller	1	125	
Nine Mile Point 1	2	None	2	2560	Gen. Motors	2	125	
Oyster Creek 1	2	None	2	2500	Gen.Motors	2	125	
Dresden 2 & 3	3	1	1	2850	Gen. Motors	2 2	125 250	
Millstone 1	3	None	1 1 *	3000 12000*	Fairbanks-Morse unk.	2	125	* Gas Turbine Generator
Monticello	3	None	2	2500	Gen. Motors	2 1	125 250	
Pilgrim 1	3	None	2	2600	Alco	2 1	125 250	
Quad Cities 1 & 2	3	1	1	2850	Gen. Motors	2 2	125 250	
Browns Ferry 1, 2 & 3	4	4 *	4	2600	Gen. Motors	3 * *	250	*4 DG's for units 1 & 2. 4 DG's for unit 3. **3 x 250 VDC divisions supply all 3 units
Brunswick 1 & 2	4	None	2	3500	Nordberg	4	125	
Cooper	4	None	2	4000	Cooper-Bessemer	2 2	125 250	
Duane Arnold	4	None	2	2850	Fairbanks-Morse	2 1	125 250	
Fermi 2	4	None	4	2850	Fairbanks-Morse	2	130/260	
Fitzpatrick	4	None	4 *	2600	Bruce GM	2	125	* Paralleled in sets of 2 generators supplying one AC division
Hatch 1 & 2	4	1	2	2850	Fairbanks-Morse	4	125/250	
Hope Creek 1 & 2	4	None	4	4430	Fairbanks-Morse	6 2	125 250*	*1 for HPCI / 1 for RCIC
Limerick 1 & 2	4	None	4	2850	Colt-Pielstick	2 2	125 250	

Table 10-8. Comparison of General Electric BWR Backup Electric Power Systems (Continued)

Reactor Plant	BWR Type	Shared Diesels per Plant	Dedicated Diesels per Unit	Continuous reting (kW)	Diesel Manufacturer	# of Batteries per Plant	Voltage	Notes
Peach Bottom 2 & 3	4	4		2600	Fairbanks-Morse	8 4	125 125/250	4 diesel generators supply 4 AC divisions in each unit.
Shoreham	4	None	3	2750	Gen. Motors	unk.	unk.	
Susquehanna 1 & 2	4	4		2950	unk.	8 4	125 250	4 diesel generators supply 4 AC divisions in each unit.
Vermont Yankee	4	None	2	3000	Gen. Motors	2	125	
LaSalle 1 & 2	5	1	2 1 *	2600 unk.	unk.	6	250 125	*HPCS diesel
Nine Mile Point 2	5	None	2 1 *	4400 2657	Cooper Gen. Motors	3	125	*HPCS diesel
WNP-2	5	None	2 1 *	4400 2600	Gen. Motors unk.	2 1	125 250	*HPCS diesel
Clinton 1	6	None	2 1 *	3500 2200	unk. unk.	4	125	*HPCS diesel
Grand Gulf 1 & 2	6	None	2 1 *	7000 3300	DeLavale Gen. Motors	4	125	*HPCS diesel
Perry 1 & 2	6	None	2 1 *	7000 2560	unk.	3	125	*HPCS diesel
River Bend 1	6	None	2 1 *	3500 2850	unk.	3	125	*HPCS diesel

Table 10-9. Comparison of General Electric BWR Power Conversion Systems

Plant Name	BWR Type	Architect/ Engineer	Constructor	Turbine Gen. Cap. (MWe)	Turbine Bypass Capability (%)	Condenser Cooling Type	Normal Heat Sink	# Main FW Pumps	FW Pump Drive Type
Big Rock Point	1	Bechtel	Bechtel	69	unk.	Once Through	Lake Michigan	2	AC
Nine Mile Point 1	2	Niagra Mohawk Power Co.	Stone & Webster	610	40	Once Through	Lake Ontario	1 2	turbine AC
Oyster Creek	2	Burns & Roe GE	Burns & Roe	620	40	Once Through	Barnegat Bay	3	AC
Dresden 2 & 3	3	Sargent & Lundy	UE & C	773	40	Once Through	Cor[illegible] Lake	3	AC
Millstone 1	3	Ebasco	Ebasco	654	105	Once Through	Niantic Bay	3	AC
Monticello	3	Bechtel	Bechtel	536	15	Closed Loop	Mech. Cooling Towers	2	AC
Pilgrim 1	3	Bechtel	Bechtel	670	25	Once Through	Cape Cod Bay	3	AC
Quad Cities	3	Sargent & Lundy	UE &C	769	40	Once Through	Mississippi River	3	AC
Browns Ferry 1, 2 & 3	4	TVA	TVA	1065	25	Combined Cycle	Mech. Cooling Towers & Tennessee River	3	turbine
Brunswick 1 & 2	4	UE & C	Brown & Root	849	25	Once Through	Cape Fear River	2	turbine
Cooper	4	Burns & Roe	Burns & Roe	764	25	Once Through	Missouri River	2	turbine
Duane Arnold	4	Bechtel	Bechtel	565	25	Closed Loop	Mech. Cooling Towers	2	AC
Fermi 2	4	Detroit Edison	Daniel	1073	25	Closed Loop	Nat. Cooling Towers	2	turbine
Fitzpatrick	4	Stone & Webster	Stone & Webster	778	25	Once Through	Lake Ontario	2	turbine
Hatch 1 & 2	4	SCS/Bechtel	Georgia Power Co.	768	25	Closed Loop	Mech. Cooling Towers	2	turbine
Hope Creek 1 & 2	4	Bechtel	Bechtel	1067	unk.	Closed Loop	Nat. Cooling Towers	unk.	unk.
Limerick 1 & 2	4	Bechtel	Bechtel	1055	25	Closed Loop	Nat. Cooling Towers	3	turbine
Peach Bottom 2 & 3	4	Bechtel	Bechtel	1051	25	Combined Cycle	Mech Cooling Towers & Susquehana River	3	turbine
Shoreham	4	Stone & Webster	Long Island Lighting Co.	809	25	Once Through	Long Island Sound	3	turbine
Susquehanna 1 & 2	4	Bechtel	Bechtel	1032	unk.	Closed Loop	Mech. Cooling Towers	unk.	unk.

Table 10-9. Comparison of General Electric BWR Power Conversion Systems (Continued)

Plant Name	BWR Type	Architect/ Engineer	Constructor	Turbine Gen. Cap. (MWe)	Turbine Bypass Capability (%)	Condenser Cooling Type	Normal Heat Sink	# Main FW Pumps	FW Pump Drive Type
Vermont Yankee	4	Ebasco	Ebasco	504	105	Combined Cycle	Lake Monticello & Mechanical Cooling Tower	3	AC
LaSalle 1 & 2	5	Sargent & Lundy	Commonwealth Edison	1036	25	Closed Loop	Cooling Lake	2 1	turbine AC
Nine Mile Point 2	5	Stone & Webster	Stone & Webster	1080	25	Once Through	Lake Ontario	3	AC
WNP-2	5	Burns & Roe	Bechtel	1095	25	Closed Loop	Mech. Cooling Towers	2	turbine
Clinton 1	6	Sargent & Lundy	Baldwin	930	35	Once Through	Lake Clinton (man-made)	2 1	turbine AC
Grand Gulf 1 & 2	6	Bechtel	Bechtel	1142	35	Closed Loop	Nat. Cooling Towers	2	turbine
Perry 1 & 2	6	Gilbert	Cleveland Electric Co.	1205	35	Closed Loop	Nat.Cooling Towers	3	turbine
River Bend 1	6	Stone & Webster	Stone & Webster	936	10	Closed Loop	Mech. Cooling Towers	3	AC

APPENDIX A

BIBLIOGRAPHY OF GENERAL PWR AND BWR REFERENCES

TABLE OF CONTENTS

Section		Page
A1.	Introduction	A1
A2.	Light Water Reactors (LWRs), General	A1
A3.	Pressurized Water Reactors (PWRs), General	A4
A4.	Westinghouse PWRs	A6
	A4.1 Westinghouse PWRs, General	A6
	A4.2 Westinghouse 2-Loop PWRs	A8
	A4.3 Westinghouse 3-Loop PWRs	A8
	A4.4 Westinghouse 4-Loop PWRs	A9
A5.	Combustion Engineering PWRs	A9
A6.	Babcock & Wilcox PWRs	A11
A7.	General Electric Boiling Water Reactors (BWRs)	A13

A1. INTRODUCTION

This appendix is a bibliography of general references for PWR and BWR plants. The references are organized in the following categories:

- Light Water Reactors (LWRs), general
- Pressurized Water Reactors (PWRs), general
- Westinghouse PWRs
  - Westinghouse PWRs, general
  - Westinghouse 2-Loop PWRs
  - Westinghouse 3-Loop PWRs
  - Westinghouse 4-Loop PWRs
- Combustion Engineering PWRs
- Babcock & Wilcox PWRs
- General Electric Boiling Water Reactors (BWRs)

Three other bibliographies provide additional references as follows:

- Appendix B: Bibliography of plant-specific references
- Appendix C: Bibliography of nuclear power plant system references
- Appendix D: Bibliography of nuclear power plant component references

A2. LIGHT-WATER REACTORS (LWRs), GENERAL

1 TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," USNRC, March 23, 1962 .

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APPENDIX B

BIBLIOGRAPHY OF PLANT-SPECIFIC REFERENCES FOR U.S. COMMERCIAL NUCLEAR POWER PLANTS

TABLE OF CONTENTS

Section		Page
B1.	Introduction	B1
B2.	Arkansas Nuclear One - Unit 1	B1
B3.	Arkansas Nuclear One - Unit 2	B2
B4.	Beaver Valley 1	B2
B5.	Beaver Valley 2	B2
B6.	Bellefonte 1 and 2	B3
B7.	Big Rock Point	B3
B8.	Braidwood 1 and 2	B3
B9.	Browns Ferry 1	B3
B10.	Browns Ferry 2	B5
B11.	Browns Ferry 3	B5
B12.	Brunswick 1 and 2	B5
B13.	Byron 1 and 2	B5
B14.	Callaway	B6
B15.	Calvert Cliffs 1 and 2	B6
B16.	Catawba 1 and 2	B7
B17.	Clinton 1	B7
B18.	Comanche Peak 1 and 2	B7
B19.	Cooper	B8
B20.	Crystal River 3	B8
B21.	D. C. Cook 1 and 2	B8
B22.	Davis-Besse	B9
B23.	Diablo Canyon 1 and 2	B9
B24.	Dresden 2 and 3	B10
B25.	Duane Arnold	B11

TABLE OF CONTENTS
(Continued)

Section		Page
B26.	Farley 1 and 2	B11
B27.	Fermi 2	B11
B28.	Fitzpatrick	B12
B29.	Fort Calhoun	B12
B30.	Ginna	B12
B31.	Grand Gulf 1 and 2	B13
B32.	Haddam Neck	B13
B33.	Hatch 1 and 2	B14
B34.	Hope Creek 1	B14
B35.	Indian Point 2	B14
B36.	Indian Point 3	B15
B37.	Kewaunee	B17
B38.	LaSalle 1 and 2	B17
B39.	Limerick 1 and 2	B17
B40.	Maine Yankee	B18
B41.	McGuire 1 and 2	B18
B42.	Millstone 1	B19
B43.	Millstone 2	B19
B44.	Millstone 3	B19
B45.	Monticello	B20
B46.	Nine Mile Point 1	B20
B47.	Nine Mile Point 2	B20
B48.	North Anna 1 and 2	B21
B49.	Oconee 1, 2, and 3	B21

TABLE OF CONTENTS
(Continued)

Section		Page
B50.	Oyster Creek	B23
B51.	Palisades	B23
B52.	Palo Verde 1, 2 and 3	B24
B53.	Peach Bottom 2 and 3	B24
B54.	Perry 1 and 2	B24
B55.	Pilgrim 1	B25
B56.	Point Beach 1 and 2	B25
B57.	Prairie Island 1 and 2	B25
B58.	Quad Cities 1 and 2	B26
B59.	Rancho Seco	B26
B60.	River Bend 1	B27
B61.	Robinson 2	B27
B62.	Salem 1 and 2	B28
B63.	San Onofre 1	B29
B64.	San Onofre 2 and 3	B29
B65.	Seabrook 1	B30
B66.	Sequoyah 1 and 2	B30
B67.	Shearon Harris	B31
B68.	Shoreham	B32
B69.	South Texas 1 and 2	B32
B70.	St. Lucie 1 and 2	B32
B71.	Surry 1 and 2	B33
B72.	Susquehanna 1 and 2	B34
B73.	Three Mile Island 1	B34

TABLE OF CONTENTS
(Continued)

Section		Page
B74.	Trojan	B35
B75.	Turkey Point 3 and 4	B35
B76.	Vermont Yankee	B36
B77.	Virgil C. Summer	B36
B78.	Vogtle 1 and 2	B36
B79.	Waterford 3	B37
B80.	Watts Bar 1 and 2	B37
B81.	WNP-2	B38
B82.	Wolf Creek	B38
B83.	Yankee-Rowe	B38
B84.	Zion 1 and 2	B39

B1. INTRODUCTION

This appendix is a bibliography of plant-specific references for PWR and BWR plants. The references are grouped by plant name, as listed in the Table of Contents. Three other bibliographies provide additional references as follows:

- Appendix A: Bibliography of general PWR and BWR references
- Appendix C: Bibliography of nuclear power plant systems references
- Appendix D: Bibliography of nuclear power plant component references

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APPENDIX C

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TABLE OF CONTENTS

Section			Page
C1.	Introduction		C1
C2.	Reactor Plant Systems, General		C1
C3.	Reactor and Reactivity Control Systems		C1
	C3.1	Reactor Core and Fuel	C1
	C3.2	LWR Reactivity Control Systems, General	C2
	C3.3	PWR Reactivity Control Systems	C2
	C3.4	BWR Reactivity Control Systems	C3
C4.	Reactor Coolant System and Connected Systems		C4
	C4.1	LWR Reactor Coolant System, General	C4
	C4.2	PWR Reactor Coolant System	C5
	C4.3	BWR Reactor Coolant System	C5
	C4.4	Shutdown Cooling Systems	C6
C5.	Engineered Safety Feature (ESF) Core Cooling Systems		C7
	C5.1	Emergency Core Cooling Systems (ECCS)	C7
	C5.2	PWR Auxiliary Feedwater Systems	C10
C6.	Containment Systems		C11
	C6.1	Containment, General	C11
	C6.2	Steel Containment	C12
	C6.3	Concrete Containment	C13
	C6.4	PWR Large Dry Containment	C14
	C6.5	PWR Ice Condenser Containment	C14
	C6.6	BWR Pressure Suppression Containment	C15
		C6.6.1 BWR Containment, General	C15
		C6.6.2 Mark I Containment	C16
		C6.6.3 Mark II Containment	C16
		C6.6.4 Mark III Containment	C17
	C6.7	Containment Penetrations and Isolation Systems	C17
	C6.8	Containment Atmosphere Cleanup Systems	C19
	C6.9	Containment Hydrogen Control Systems	C20
	C6.10	Other Containment Auxiliary Systems	C22
C7.	Instrumentation and Control Systems		C22
	C7.1	Instrumentation and Control Systems, General	C22
	C7.2.	Safety System (RPS and ESFAS)	C25
	C7.3	Control Systems	C26
	C7.4	Annunciator and Display Systems	C27
C8.	Electric Power Systems		C27
	C8.1	Electric Power Systems, General	C27
	C8.2	AC Electric Power Systems	C29
	C8.3	DC Electric Power Systems	C30

TABLE OF CONTENTS
(Continued)

Section			Page
C9.	Auxiliary Systems		C30
	C9.1	Fuel Storage and Handling Systems	C30
	C9.2	Service Water and Ultimate Heat Sink Systems	C31
	C9.3	Component Cooling Water Systems	C33
	C9.4	Instrument and Service Air or Gas Systems	C33
	C9.5	Heating, Ventilating, and Air-Conditioning (HVAC) Systems and Air Cleaning Systems	C33
	C9.6	Fire Protection Systems	C35
C10.	Steam and Power Conversion System		C36
	C10.1	Secondary Coolant System, General	C36
	C10.2	Main Steam System	C37
	C10.3	Condensate System	C37
	C10.4	Main Feedwater System	C38
	C10.5	Circulating Water (Main Condenser Cooling Water) System	C38

C1. INTRODUCTION

This appendix is a bibliography of nuclear power plant system references. The references are organized in the following general categories:

- Reactor plant systems, general
- Reactor and reactivity control systems
- Reactor coolant system and connected systems
- Engineered safety feature (ESF) core cooling systems
- Containment systems
- Instrumentation and control systems
- Electric power systems
- Auxiliary systems
- Steam and power conversion system

The complete organization of this bibliography is listed in the Table of Contents. Three other bibliographies provide additional references as follows:

- Appendix A: Bibliography of general PWR and BWR references
- Appendix B: Bibliography of plant-specific references
- Appendix D: Bibliography of nuclear power plant component references

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APPENDIX D

BIBLIOGRAPHY OF NUCLEAR POWER PLANT COMPONENT REFERENCES

TABLE OF CONTENTS

Section			Page
D1.	Introduction		D1
D2.	Components, General		D1
D3.	Fluid System Components		D2
	D3.1	Pumps	D2
		D3.1.1 Pumps, General	D2
		D3.1.2 Reactor Coolant Pumps	D3
		D3.1.3 Other Pumps	D5
	D3.2	Valves	D5
		D3.2.1 Valves, General	D5
		D3.2.2 Motor-Operated Valves	D7
		D3.2.3 Pneumatic Valves	D7
		D3.2.4 Hydraulic Valves	D8
		D3.2.5 Solenoid Valves	D8
		D3.2.6 Pressure Relief Valves	D8
		D3.2.7 Check Valves	D10
	D3.3	Reactor Pressure Vessels	D10
	D3.4	Steam Generators	D14
		D3.4.1 Steam Generators, General	D14
		D3.4.2 U-Tube Steam Generators	D16
		D3.4.3 Once-Through Steam Generators	D17
	D3.5	Other Pressure Vessels	D17
	D3.6	Tanks (Low Pressure and Atmospheric)	D17
	D3.7	Heat Exchangers (Water-to-Water)	D17
	D3.8	Main Condensers	D18
	D3.9	Cooling Towers	D18
	D3.10	Piping	D18
	D3.11	Other Fluid System Components	D25
D4.	Instrumentation and Control System Components		D25
	D4.1	Sensors and Transmitters	D25
	D4.2	Relays and Switches	D26
	D4.3	Terminal Blocks	D28
	D4.4	Isolation Devices	D28
D5.	Electric Power System Components		D28
	D5.1	AC or DC Power Distribution Equipment	D28
	D5.2	Circuit Breakers	D29
	D5.3	Station Batteries	D30
	D5.4	Battery Chargers	D30
	D5.5	Inverters	D31
	D5.6	Electrical Cables and Cable Trays	D31
D6.	Prime Movers		D32
	D6.1	Electric Motor Drives	D32
	D6.2	Diesel Generators	D33
	D6.3	Steam Turbine Generators	D35
	D6.4	Gas Turbines	D35

TABLE OF CONTENTS
(Continued)

Section			Page
D7.	Heating, Ventilating, and Air-Conditioning (HVAC) and Air Cleanup System Components		D35
	D7.1	Ventilation Fans	D35
	D7.2	Dampers	D35
	D7.3	Filters / Adsorber Banks	D36
	D7.4	Ducting	D36
D8.	Instrument and Service Air or Gas System Components		D36
D9.	Mechanical Components		D37
	D9.1	Airlocks	D37
	D9.2	Cranes and Hoists	D37
	D9.3	Component Supports and Restraints	D37
	D9.4	Thermal Insulation	D38
	D9.5	Fasteners	D39

D1. INTRODUCTION

This appendix is a bibliography of nuclear power plant component references. The references are organized in the following general categories:

- Components, general
- Fluid system components
- Instrumentation and control system components
- Electric power system components
- Prime movers (ie. diesel generators, electric motors, steam turbines)
- Heating, ventilating and air-conditioning (HVAC) and air cleanup system
- Mechanical components (airlocks, cranes, component supports and restraints, thermal insulation, fasteners)

The complete organization of this bibliography is listed in the Table of Contents. Three other bibliographies provide additional references as follows:

- Appendix A: Bibliography of general PWR and BWR references
- Appendix B: Bibliography of plant-specific references
- Appendix C: Bibliography of nuclear power plant component references

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