C. Ryder

Cornell University

Sibley School of
Mechanical and Aerospace Engineering
Upson and Grumman Halls
Ithaca, New York 14853

June 15, 1984

Mr. Mel Silberberg
Accident Source Term Program
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Mel:

Some time ago Irv Spiewak, partly at my urging, requested information along the following lines. I repeat the request and would appreciate having the information sent to me directly just as soon as possible.

We are interested, in general, in determining how well the various calculations can keep track of the water in the system. In order to do so, we must know <u>where</u> the initial inventories are and <u>how large</u> they are. Accordingly, the following would be most helpful.

A simple block diagram (per reactor type) is needed that will show:

1. Volume of Water
2. Mass of Water
3. Pressure of Water
4. Normal Temperature of Water

in each of the reservoirs, including the reactor vessel, that are connected to the reactor vessel. The paths of connection should be indicated on the diagram (i.e. via pumps or gravity or whatever).

Thus, we are interested in knowing how much water may be available through one or more of the ECCS systems. We wish to know at what pressures they operate. We want to know how much water can be made available by sprays, supression pools, ice baths or whatever other engineered safety system may be in place.

In addition, it would be nice to know how water is distributed between the reactor vessel, the pressurizer (for PWRs) the coolant loops and the steam generators (for PWRs) under normal operating conditions. It might even be helpful to know how large the steam (vapor) bubble is in the pressurizer under normal operating conditions.

In other words, we would much appreciate having a concise, but complete, aqueous description of the plants (2 PWR and 2 BWR) that we have been looking at. Then, should anyone ever ask, "where's the water?", we will be able to answer without a moment's hesitation.

Sincerely,

Peter

P. L. Auer
Professor

cc: R. Wilson - Harvard
I. Spiewak - Oak Ridge

/gf

#5

Battelle
Columbus Laboratories
505 King Avenue
Columbus, Ohio 43201
Telephone (614) 424-6424
Telex 24-5454

July 2, 1984

Dr. Peter Auer
Upson Hall
Cornell University
Ithaca, New York 14853

Dear Dr. Auer:

Enclosed are the four additional figures that we discussed. If we can be of further help, let me know.

Sincerely,

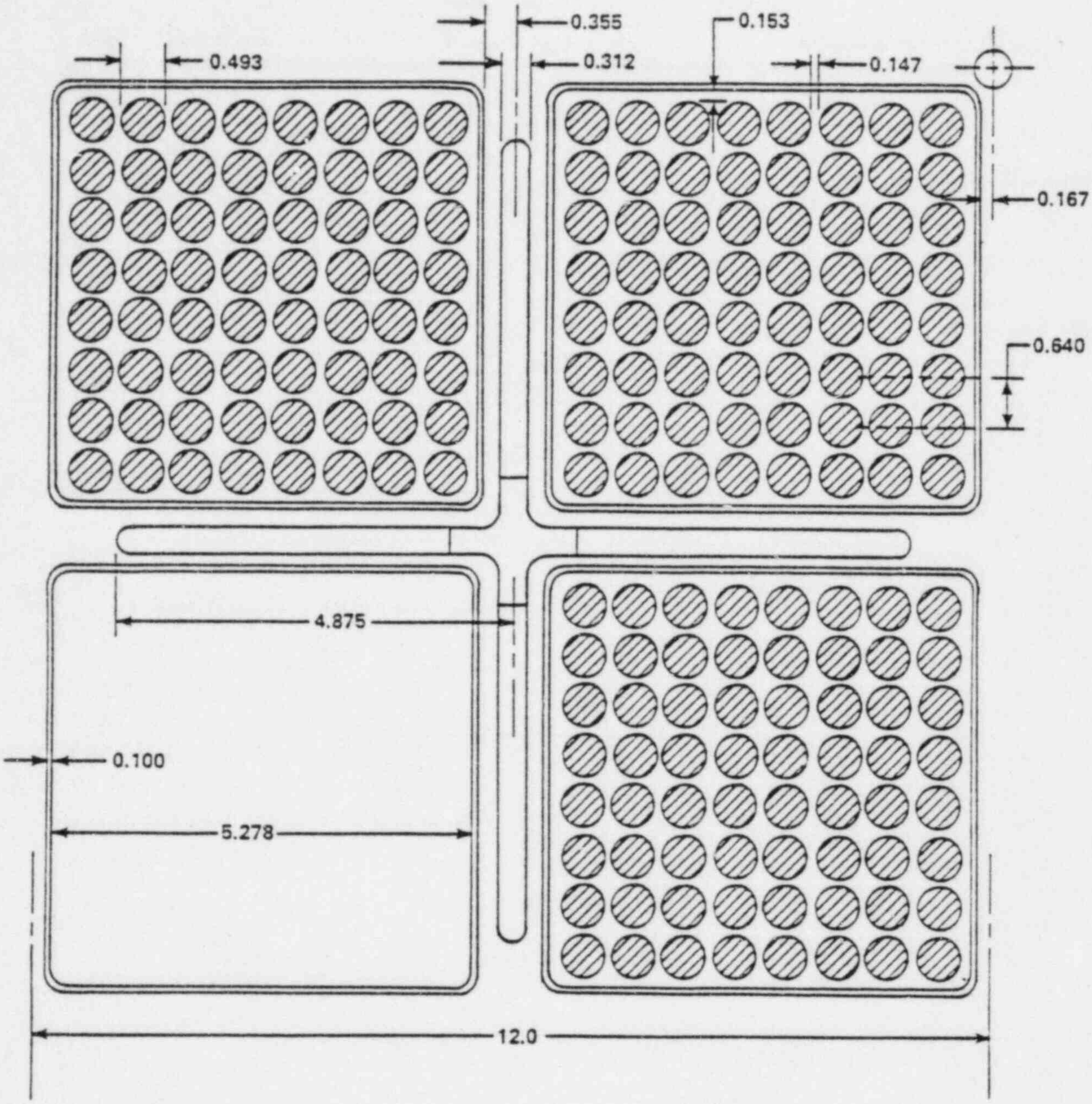
Richard S. Denning
Senior Research Leader
Nuclear Technology & Physical
Sciences Department

RSD:erc

Enc.

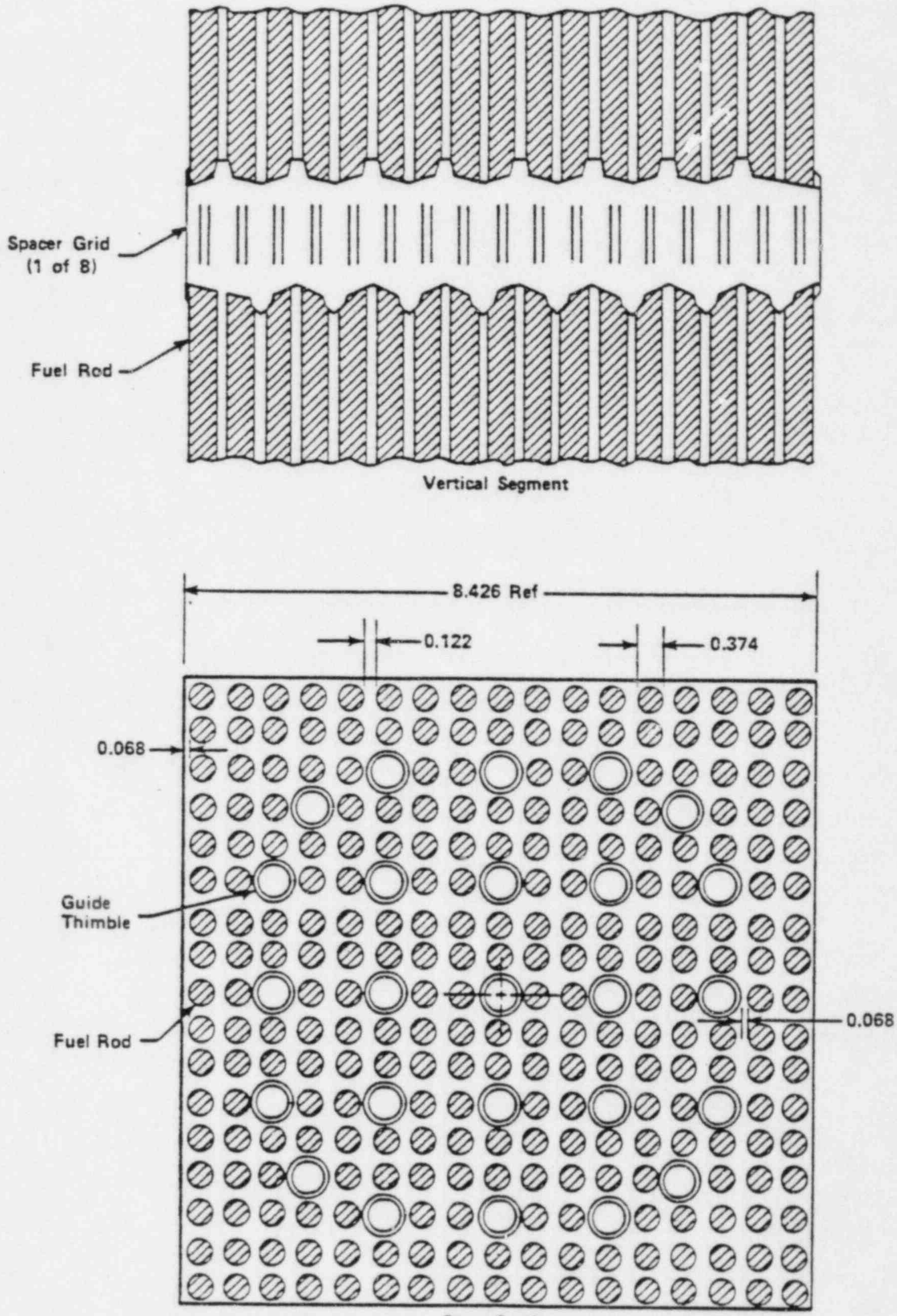
xc: [redacted]
JA Gieseke, BCL
P. Cybulskis, BCL
A. Wolford, BCL

164



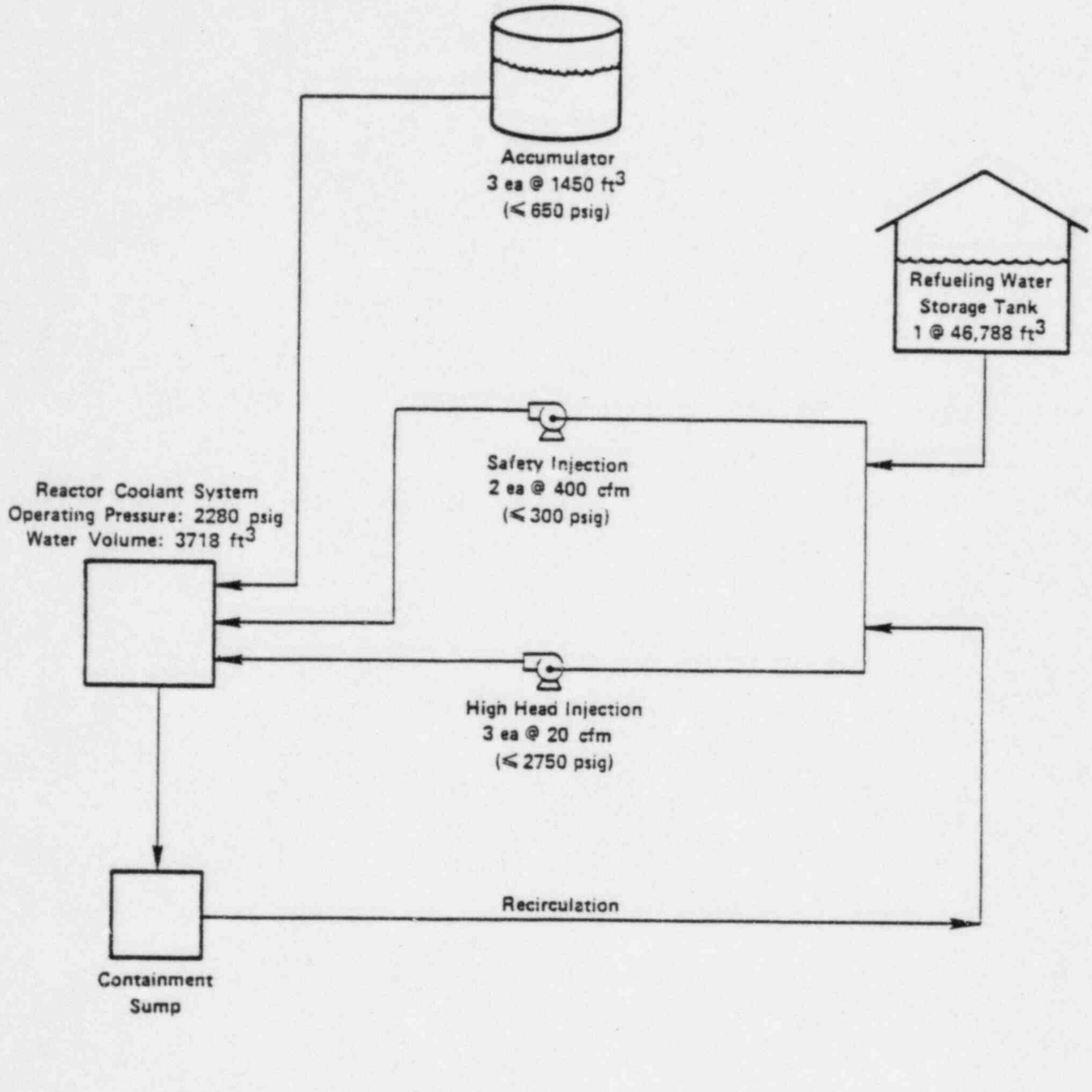
(dimensions in inches)

FOUR BWR FUEL BUNDLES WITH CONTROL BLADE

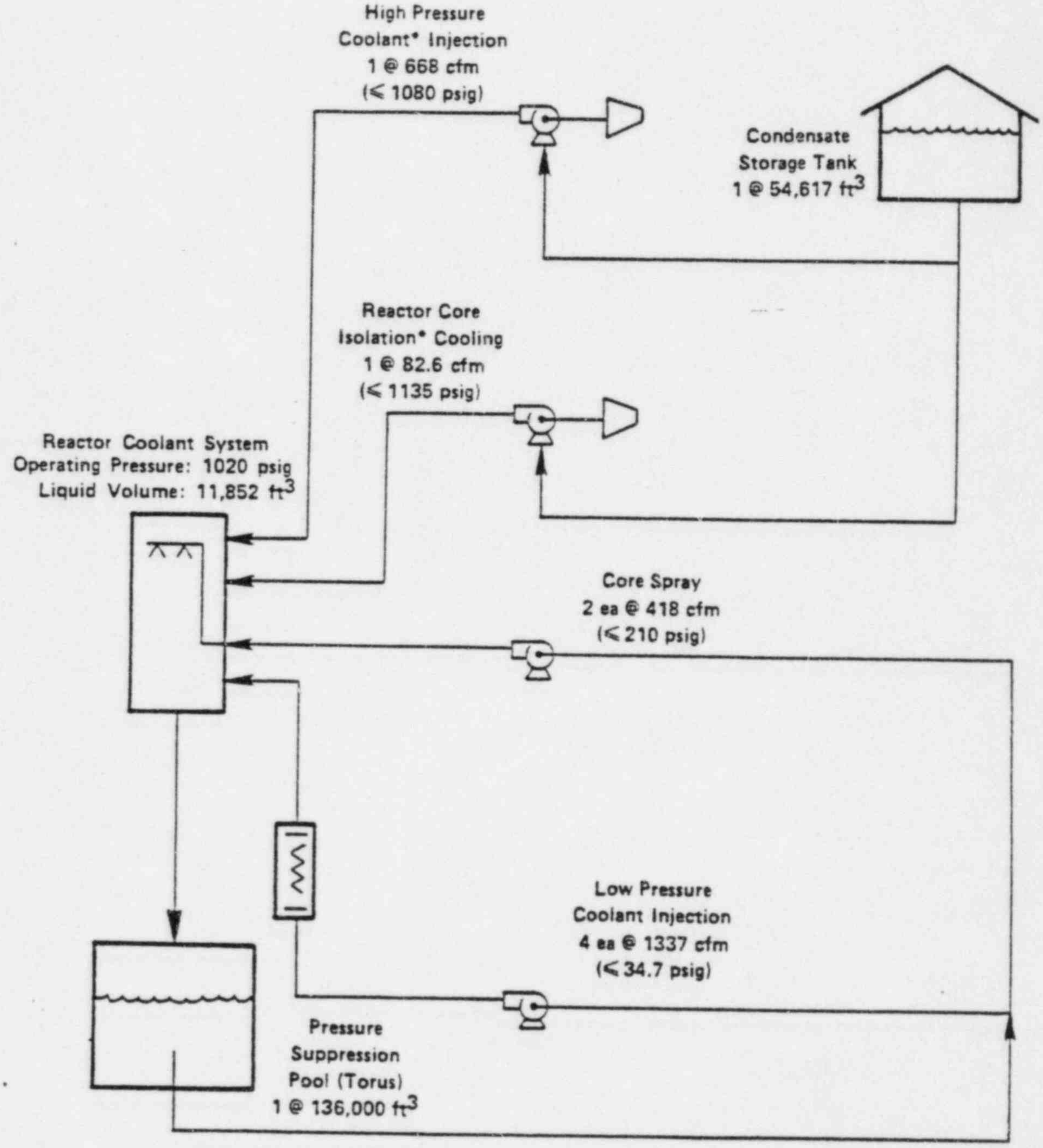


(dimension in inches)

PWR 17 x 17 FUEL BUNDLE



PWR EMERGENCY CORE COOLING SYSTEM PARAMETERS (SURRY)



*Pumps are turbine-driven by steam generated by decay heat.

BWR EMERGENCY CORE COOLING SYSTEM PARAMETERS (PEACH BOTTOM)

Approved before final typing by
P. Cybulskis and J. A. Gieseke.

Battelle
Columbus Laboratories
505 King Avenue
Columbus, Ohio 43201
Telephone (614) 424-6424
Telex 24-5454

June 15, 1984

Dr. Peter Auer
Cornell University
Ithica, New York 14853

Dear Dr. Auer:

I was asked to prepare a response to your letter to Mel Silberberg of May 26, 1984, in which you requested illustrations and geometric detail for PWR and BWR systems. Since design drawings and safety analysis report information are usually provided in English units, we have chosen to use them in our response. The data that we have provided to you come from safety analysis reports and supplementary information submitted to the NRC during plant licensing (provided by Walter Pasedag of the NRC). None of the data that I have included is proprietary, although we have used proprietary data to perform the analysis in "Radionuclide Release Under Specific LWR Accident Conditions", BMI-2104. You may use the data presented here as you wish. Data for the Surry plant are presented as representative of the PWR design and for the Peach Bottom plant as representative of the BWR design. With regards to the reactor cooling system geometries, the differences within the PWR plant class and within the BWR plant class are relatively minor. The containment layout can, of course, differ substantially.

In particular, we have attempted to explicitly address the following information needs which were requested in your letter:

- System layout
- Normal operation coolant flow paths
- Major component descriptions
- Pressure vessel internals
- Fuel assembly data and core loading arrangement
- Emergency core cooling system diagrams.

We have tried to order the figures in logical sequence. Table 1 identifies the figure number with its information content. We have also presented the volumetric data and the water inventory under normal operating conditions as requested.

If after you have reviewed this information you need some addition data or if some discussions would be helpful, please feel free to call me at (614) 424-7510.

Sincerely,

Richard S. Denning
Senior Research Leader
Nuclear Technology & Physical
Sciences Department

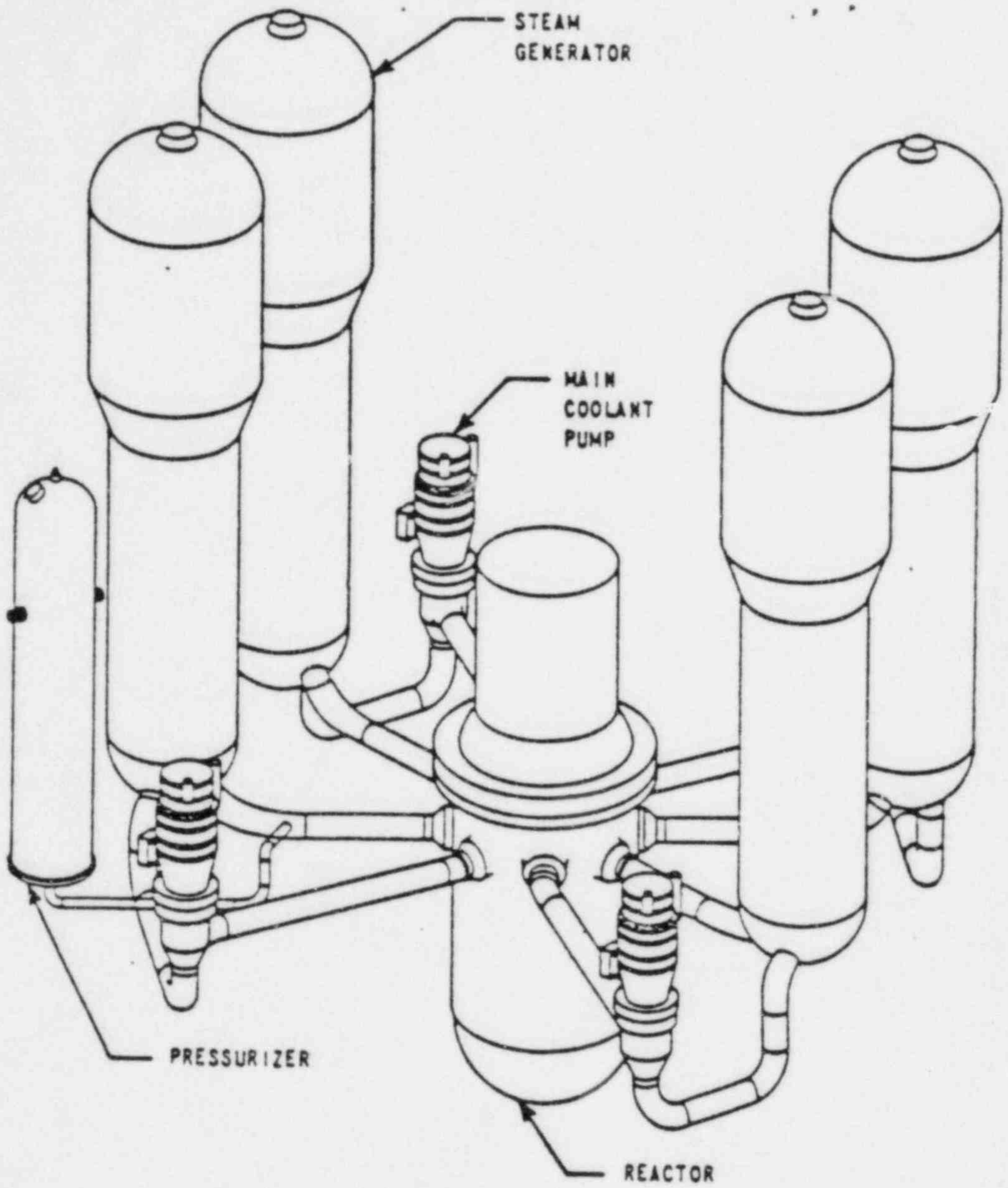
RSD:erc

Enc.

TABLE 1. INFORMATION/DRAWING REFERENCE

		PWR (Surry)	BWR (Peach Bottom)
1.	SYSTEM LAYOUT	P1.1 Pictoral	B1.1 Pictoral
2.	NORMAL COOLANT FLOWPATH	P2.1, P2.2 Schematic	B2.1 Cartoon/Schematic B2.2, B2.3 Inter Vessel Flo
3.	MAJOR COMPONENTS	P3.1, P3.2 Reactor Pressure Vessel, CX. P3.3 Steam Generator (SG) P3.4 SG, Dimension P3.5 Pressurizer, Dimension	B3.1. Reactor Vessel Pictoral B3.2 Reactor Vessel CX. B3.3 Reactor Vessel Elevati
4.	PRESSURE VESSEL INTERNALS	P4.1 Lower Internals, Pictoral P4.2 Lower Internals, Dimension P4.3, P4.4 Upper Core Support Assembly (Plenum)	B4.1 Steam Separator B4.2 Steam Dryer B4.3 Jet Pump
5.	FUEL ASSEMBLY & CORE LOADING	P5.1, P5.2 ~~Fuel Assembly~~ P5.3 Assembly + Pin Data P5.4, P5.5 Core CX.	B5.1 Isometric B5.2 CX. & Dimension B5.3, B5.4 Assembly & Pin Data
6.	EMERGENCY CORE COOLING SYSTEMS (ECCS)	P6.1 ECCS Flow P6.2 Accumulator Flow P6.3 High Pressure Injection, Flow P6.4 Safety Injection, Flow	B6.1 ECCS Flow B6.2 High Pressure Injection System, Flow B6.3 Core Spray System, Flow B6.4 Residual Heat Removal System, Flow
7.	CONTAINMENT		B7.1 Primary Cont. B7.2 Drywell Cutaway
8.	COOLANT INVENTORY & SELECT COMPONENT VOLUMES	P8.1 Tabulation P8.2 Press. Vessel - Breakdown	B8.1 Tabulation B8.2, B8.3 Pressure Vessel Breakdo

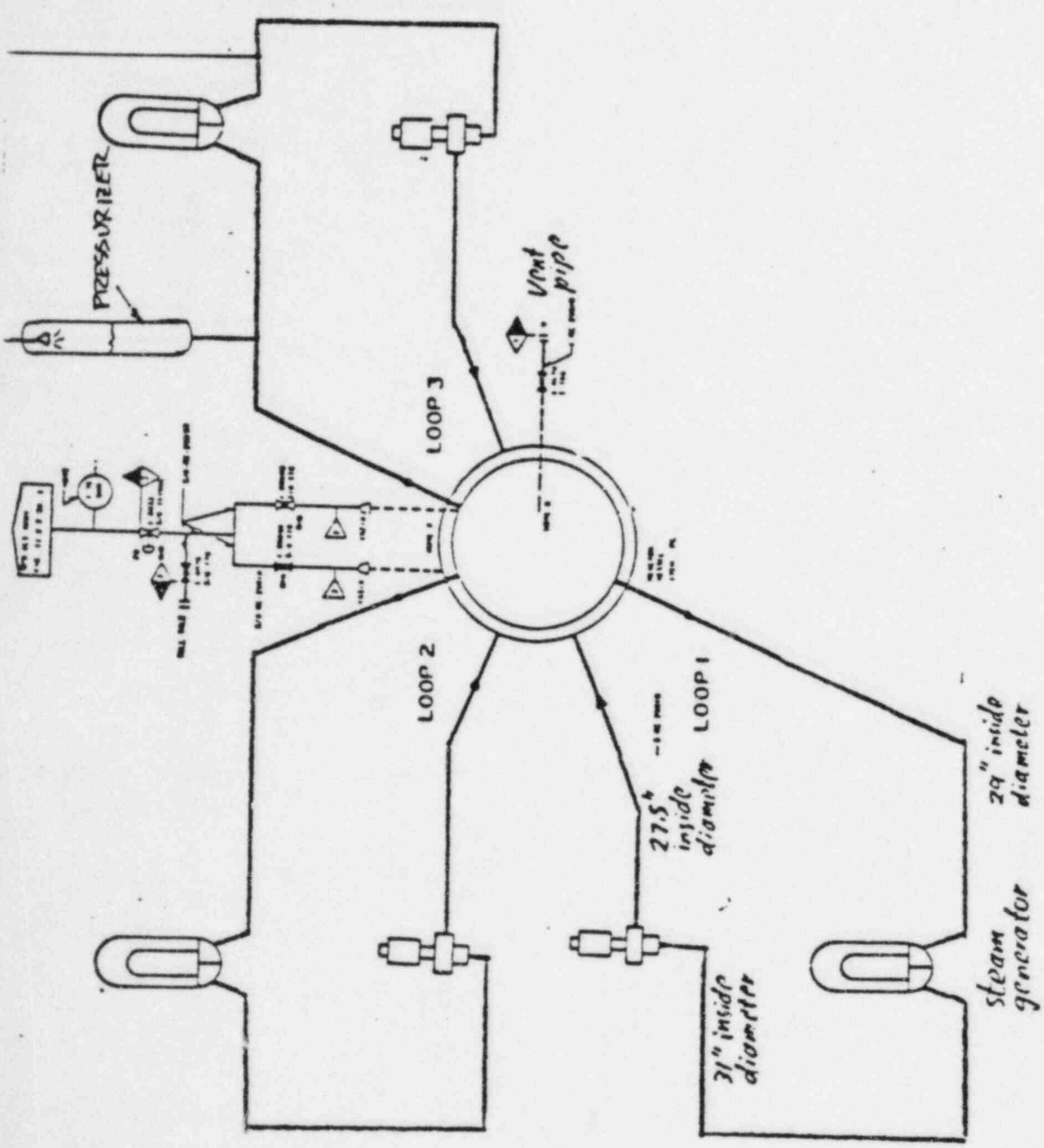
SURRY 1 PWR
DATA



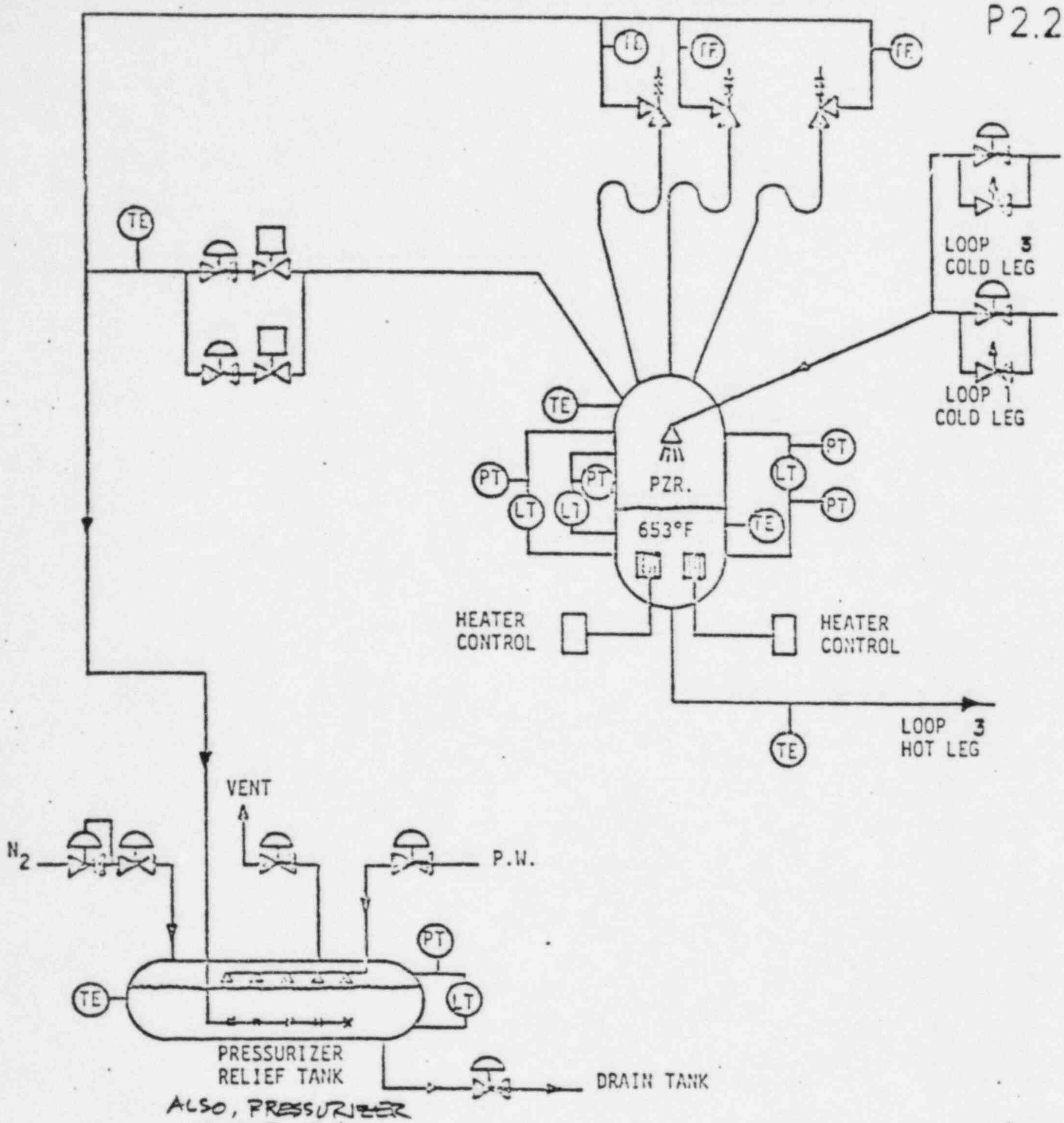
Simplified Diagram of Four-Loop Nuclear Steam
Supply System

Note Surry 1 is a 3 loop system,
Above is for illustration

Simplified
-Flow

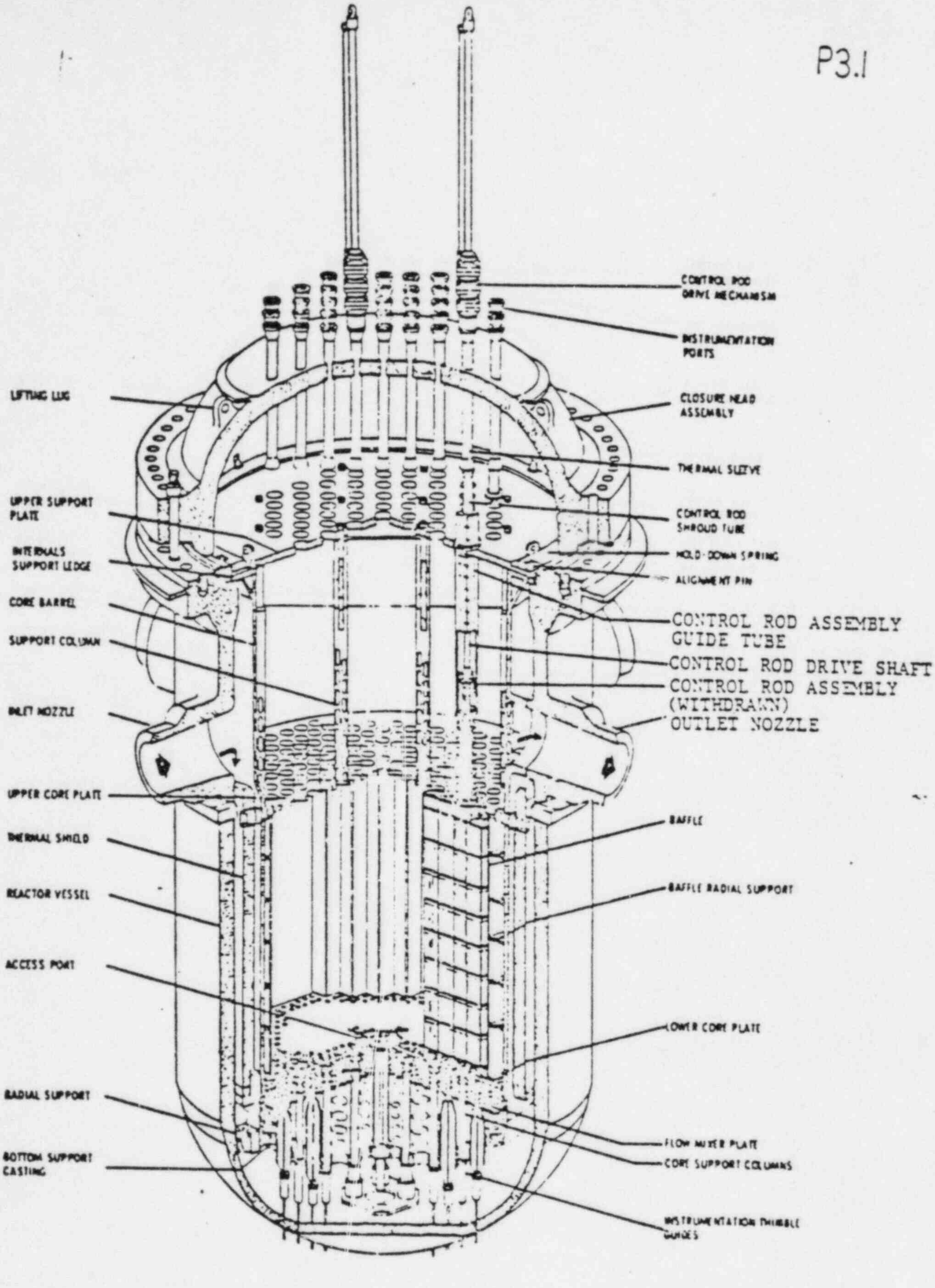


SIMPLIFIED FLOW SCHEMATIC,
PRIMARY SYSTEM



PRESSURIZER AND ASSOCIATED PIPING

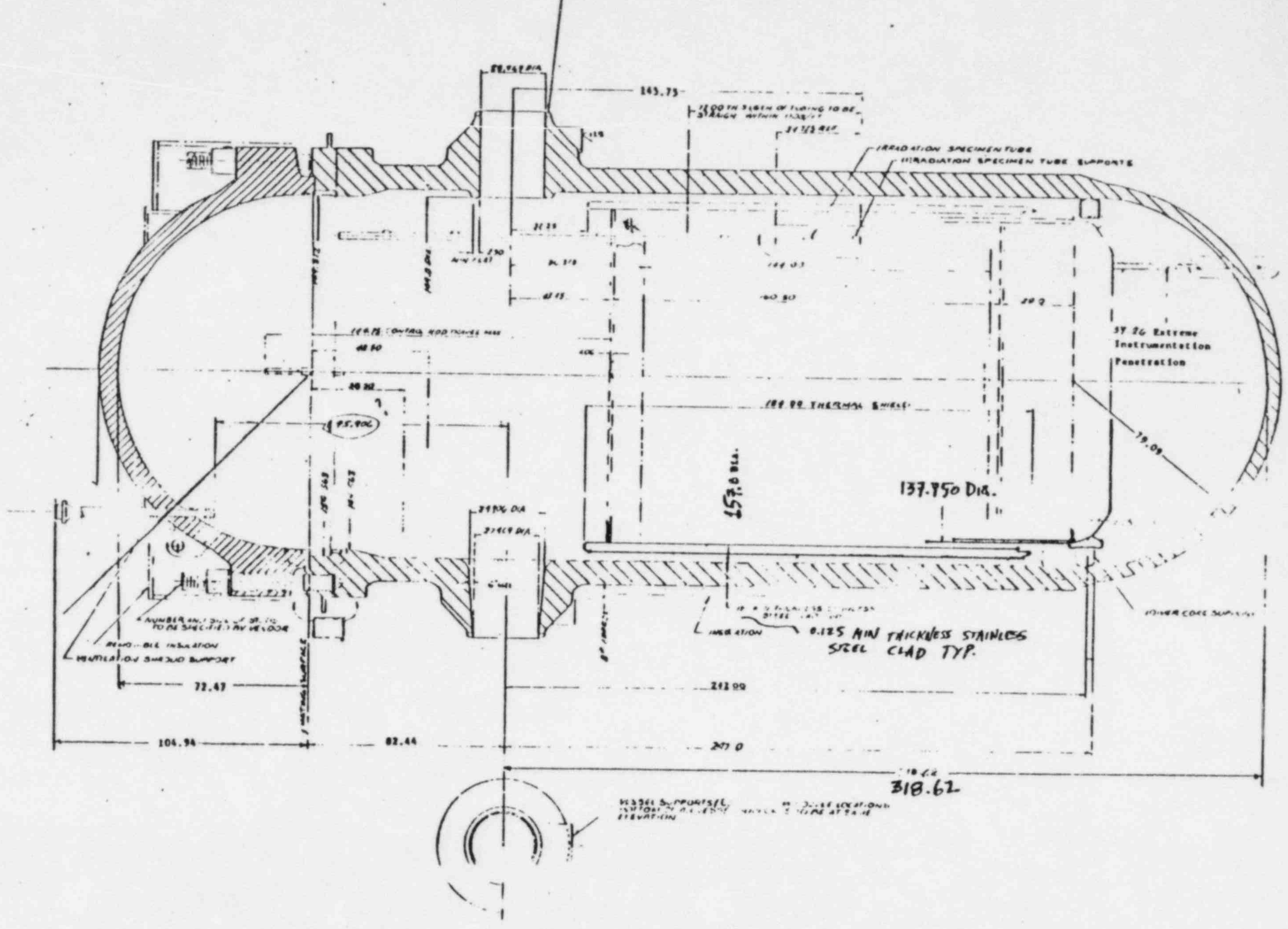
P3.1



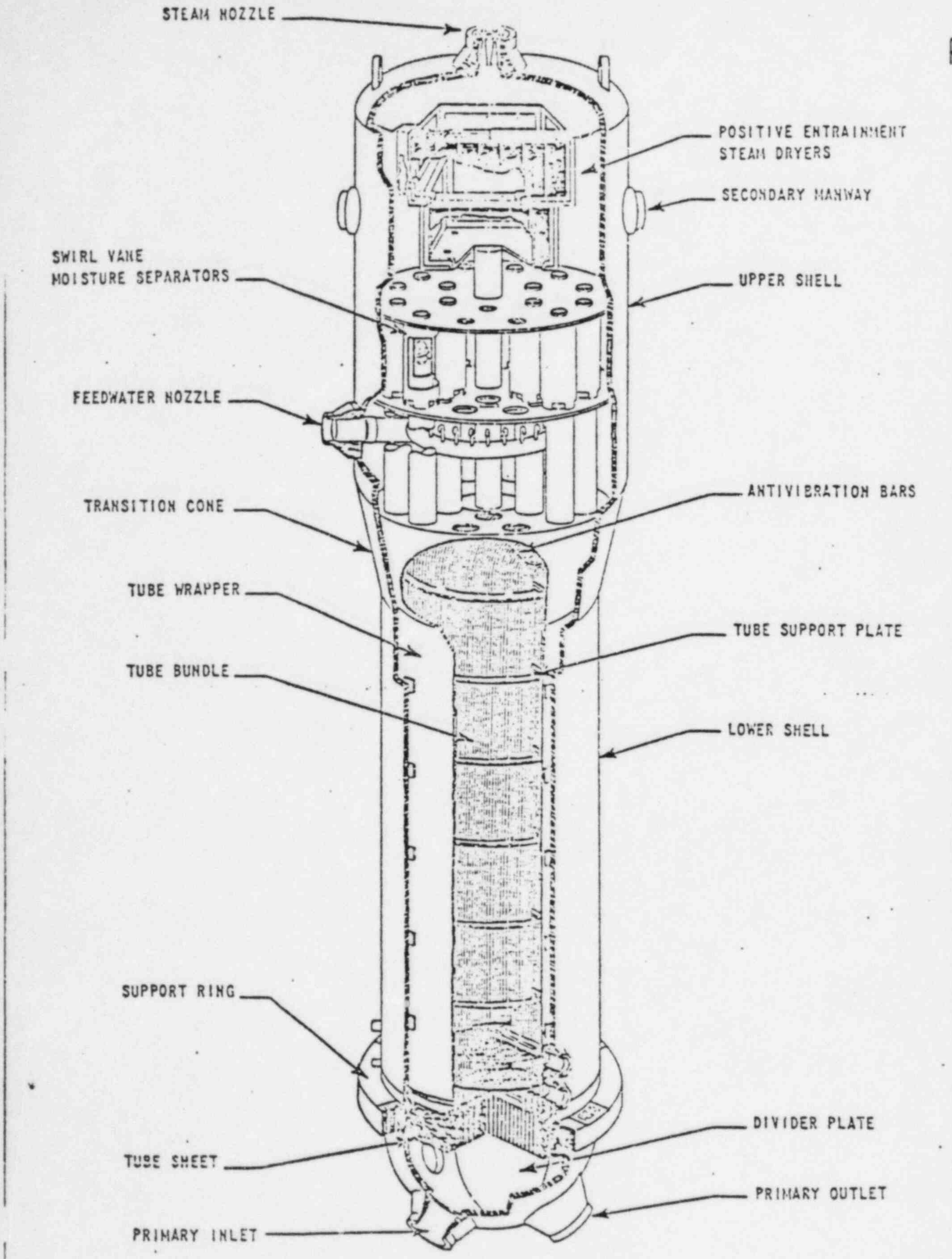
TYPICAL REACTOR VESSEL AND INTERNALS

PICTORAL

P3.2

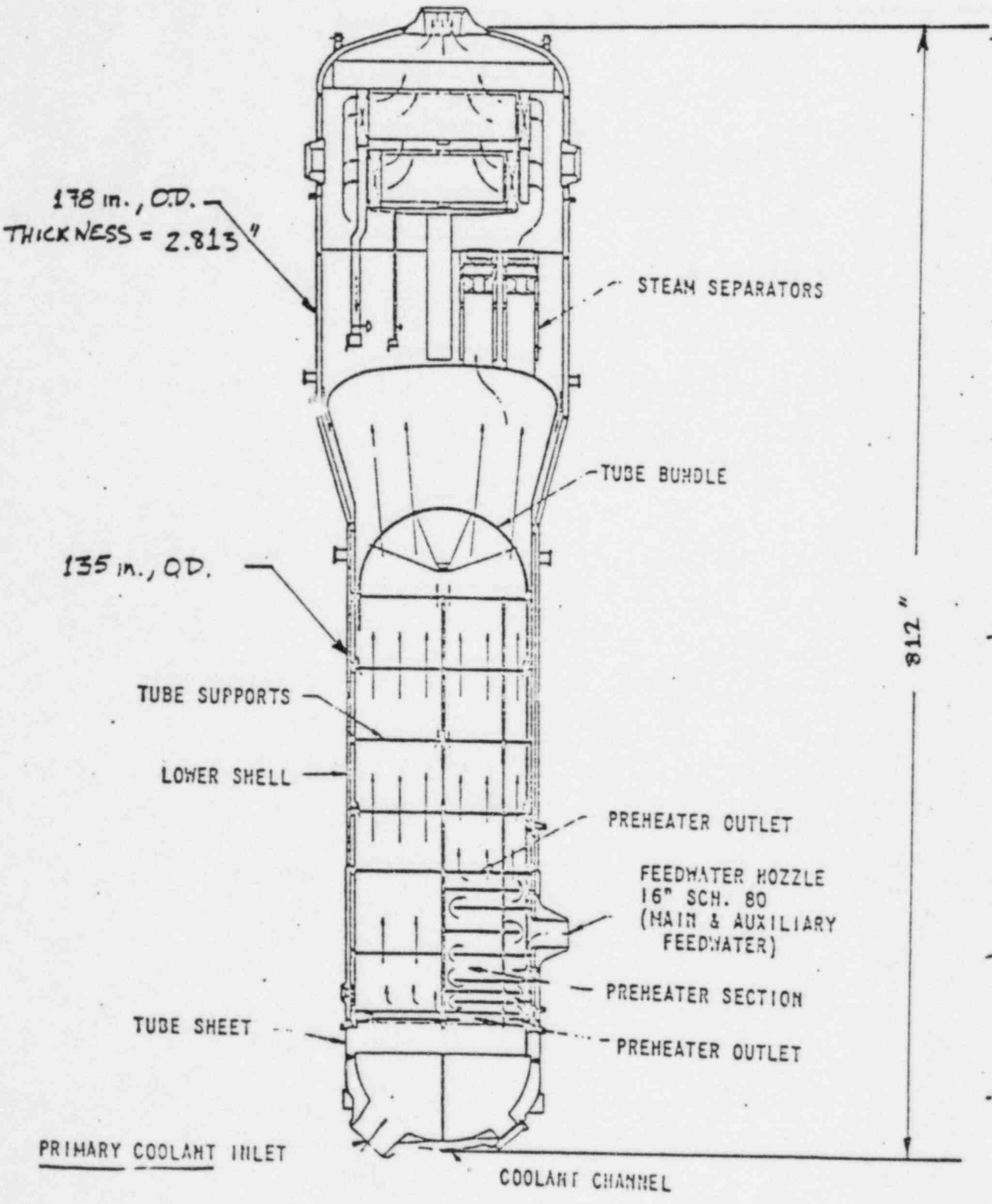


REACTOR VESSEL

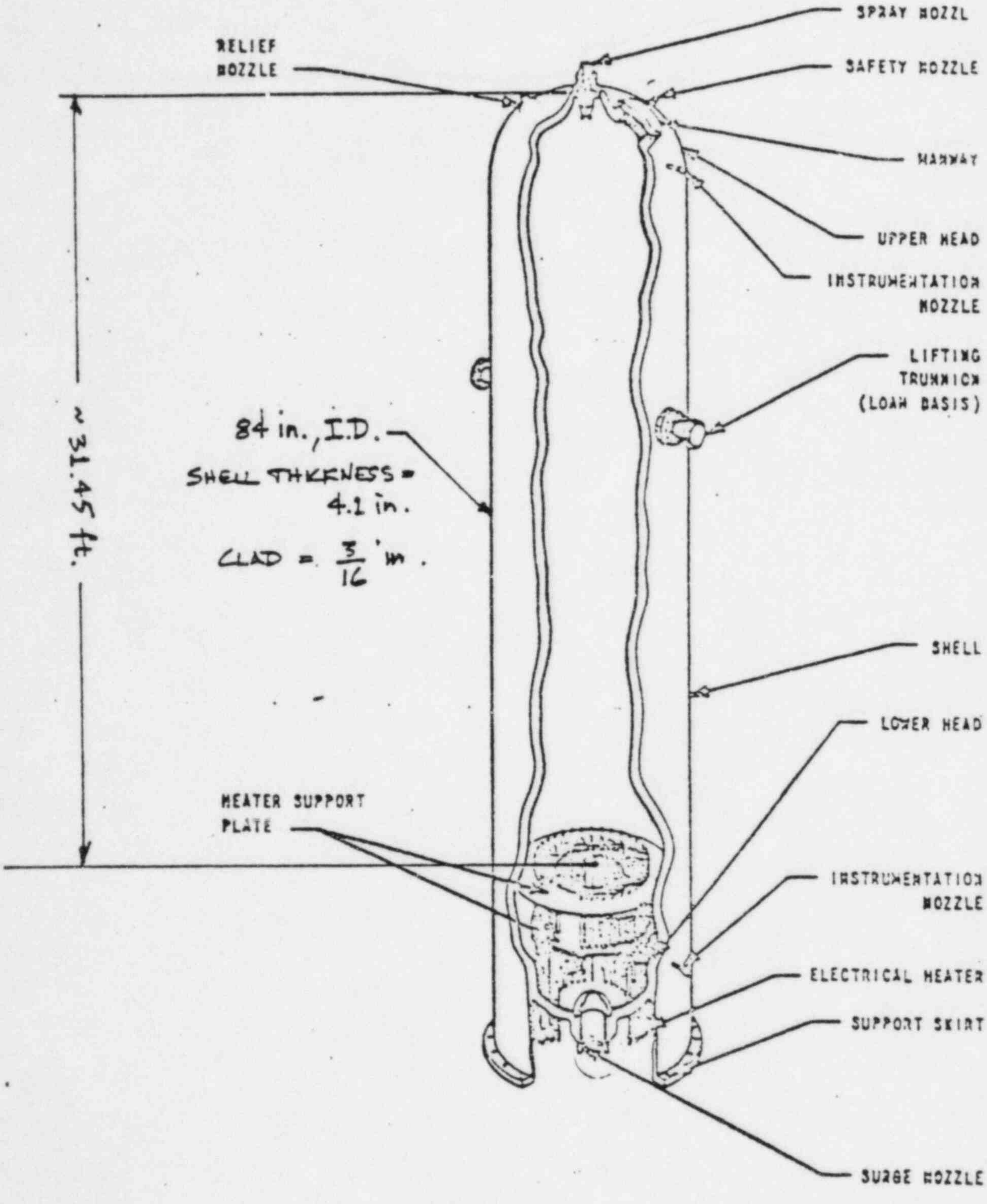


Westinghouse STEAM GENERATOR

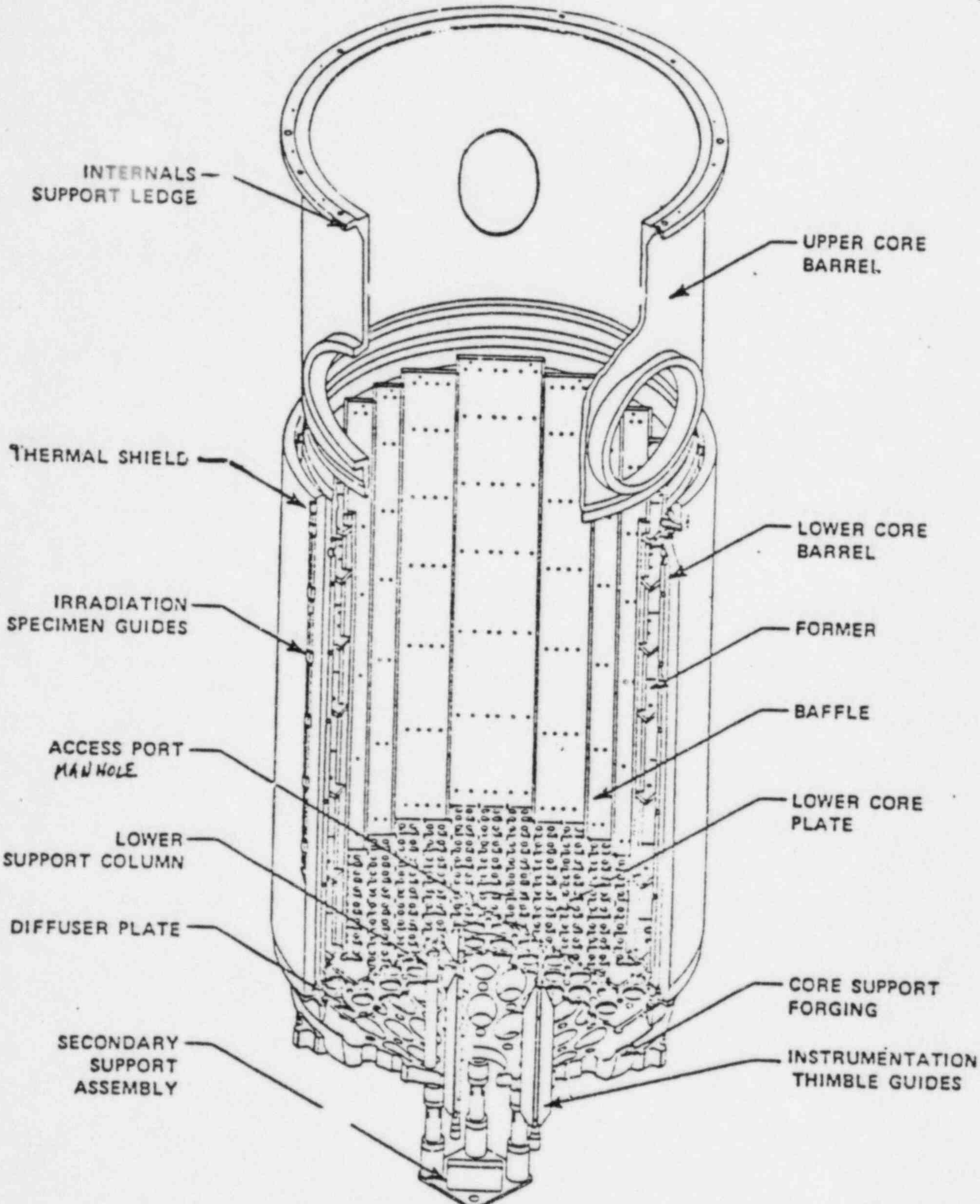
MB 3593



- PREHEAT STEAM GENERATOR FLOW PATH

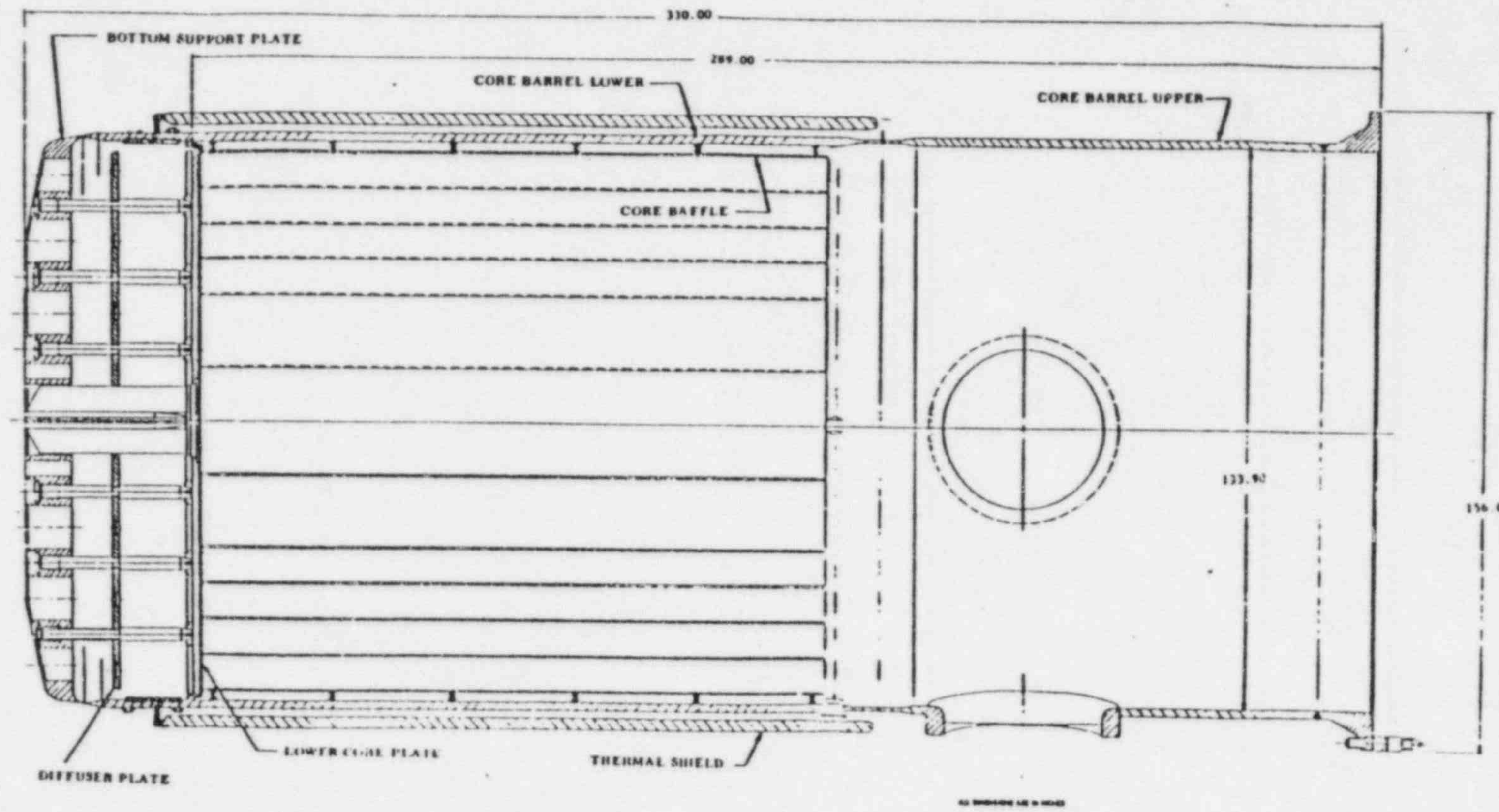


WESTINGHOUSE PRESSURIZER



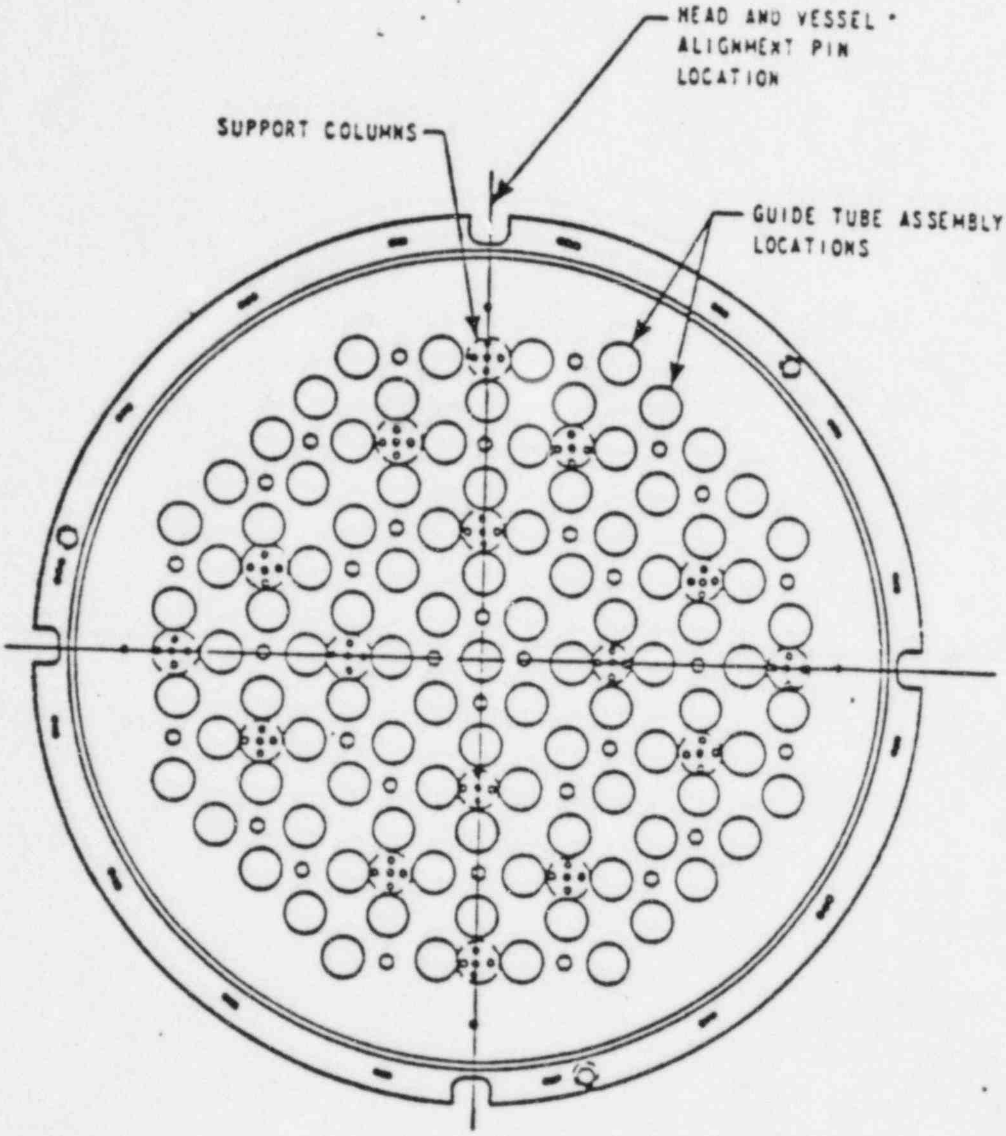
Lower Core Support Structure

FIGURE 3.5-5
12/1/69

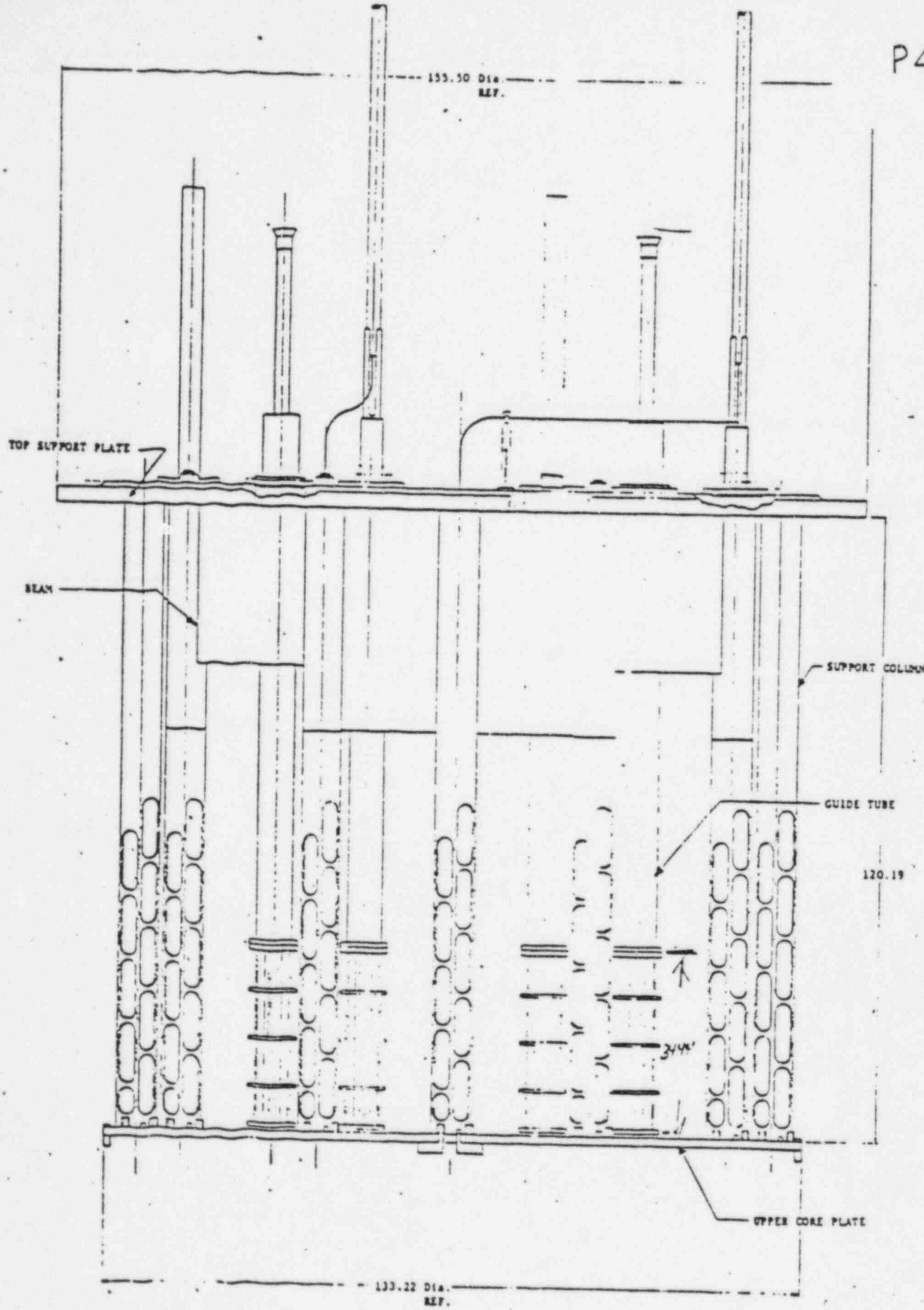


LOWER CORE S

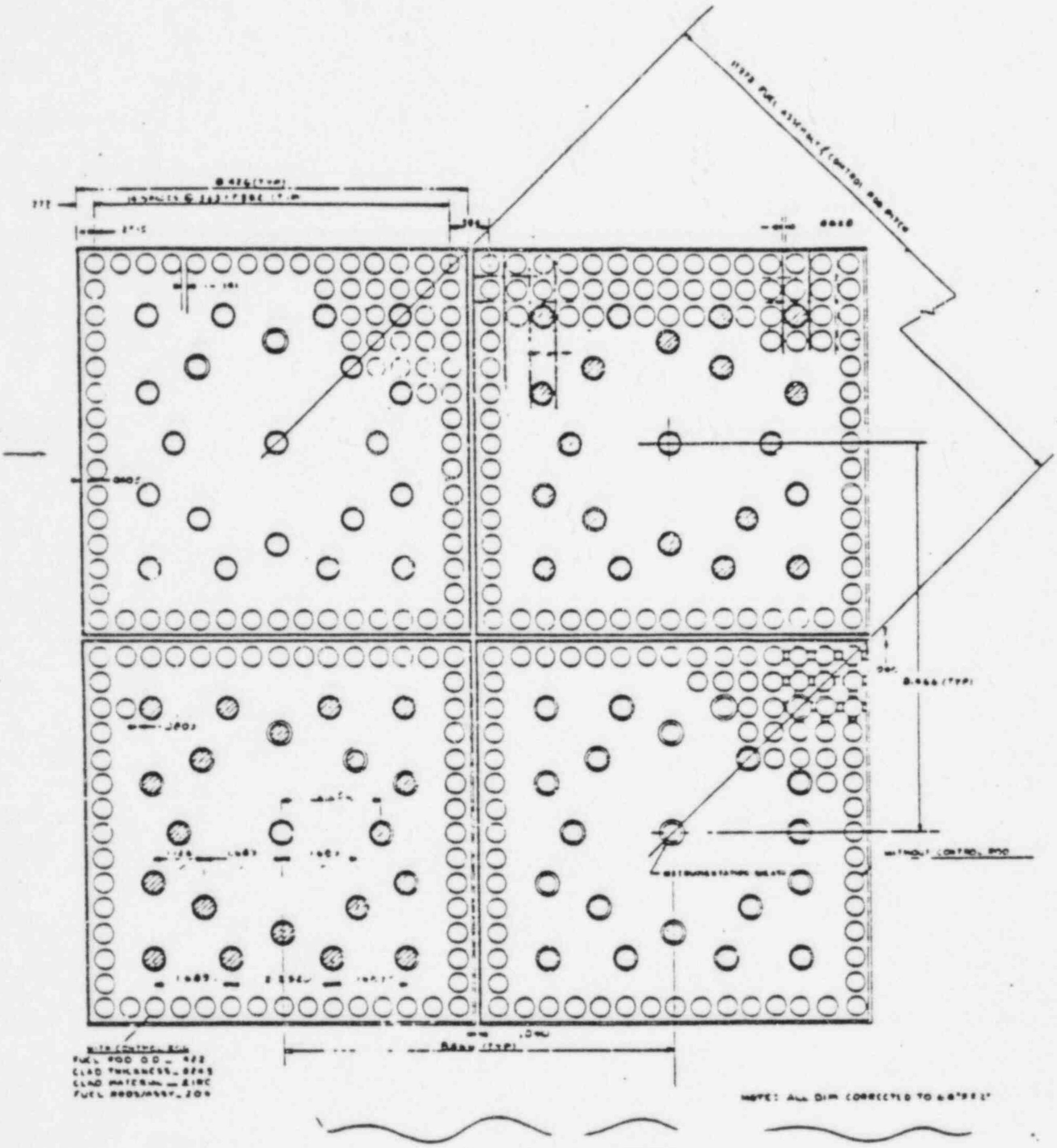
P4.2



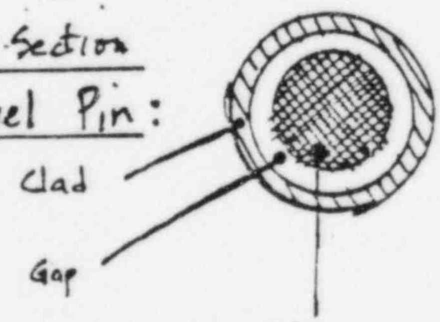
Plan View of Upper Core Support Structure



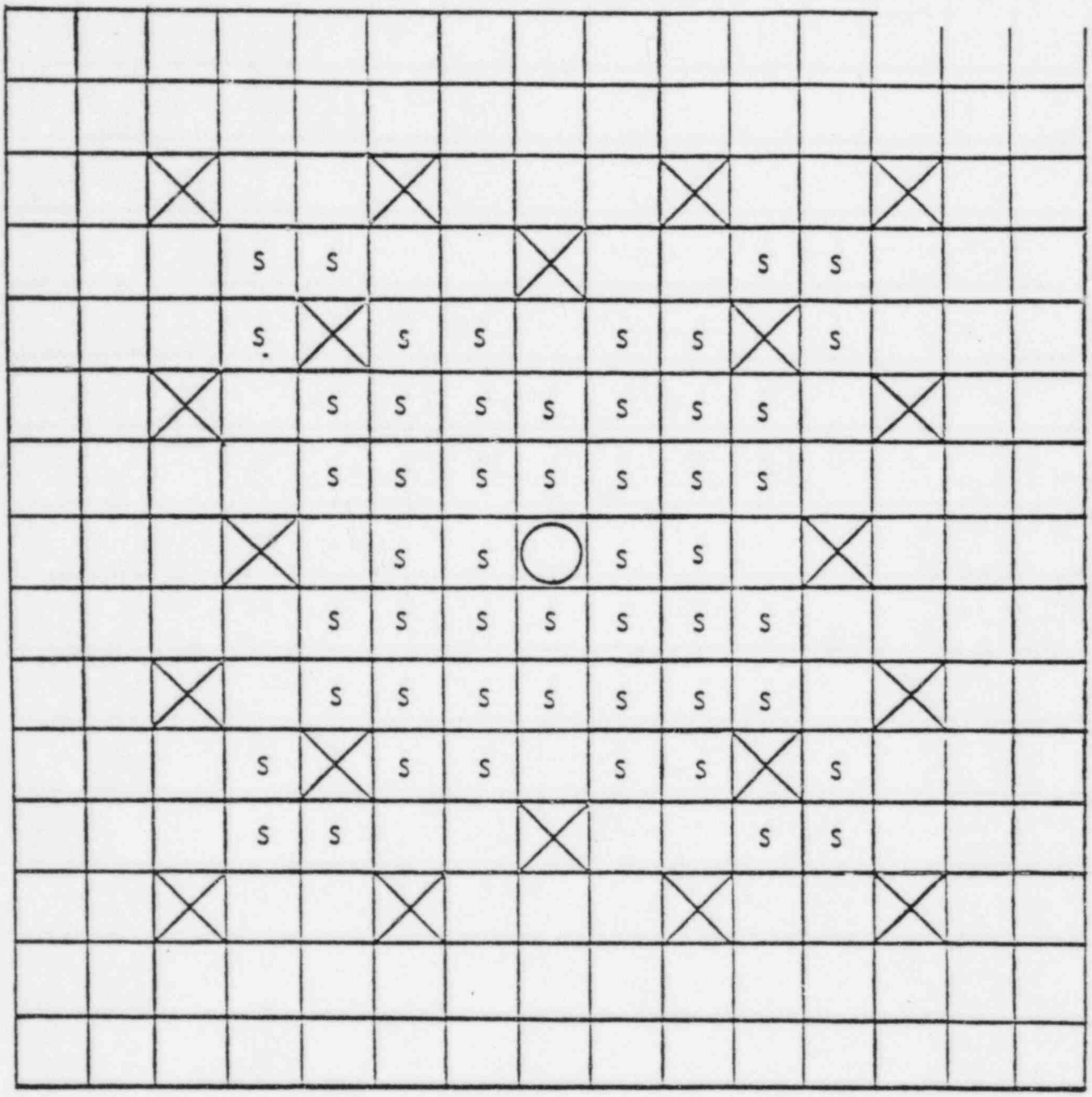
UPPER CORE SUPPORT ASSEMBLY



Radial Cross-Section through Fuel Pin:



FUEL ASSEMBLY AND CONTROL ROD ASSEMBLY CROSS SECTION



- Thimble Locations

S - Special (Removable) Fuel Rods

- Regular Fuel Rods

- Instrumentation Tube

LOCATION OF REMOVABLE RODS
WITHIN AN ASSEMBLY

FUEL/COOLANT CHANNEL DATA

(Note: Dimensions in inches, square inches)

CORE

Equivalent Diameter (Approximated as a Cylinder)	119.7
Active Height	144
Cross Sectional Area	11275

ASSEMBLIES

Number	157
Rod Array	[illegible]5
Rods per Assembly	
Fuel	204
Control	20
Instrumentation	1

RODS

Number	32028
* OD	0.422
* Clad Thickness	0.0243
* Radial Gap	0.00375
Rod Pitch (Center-to-Center, non diagonal adjacent rods)	0.563

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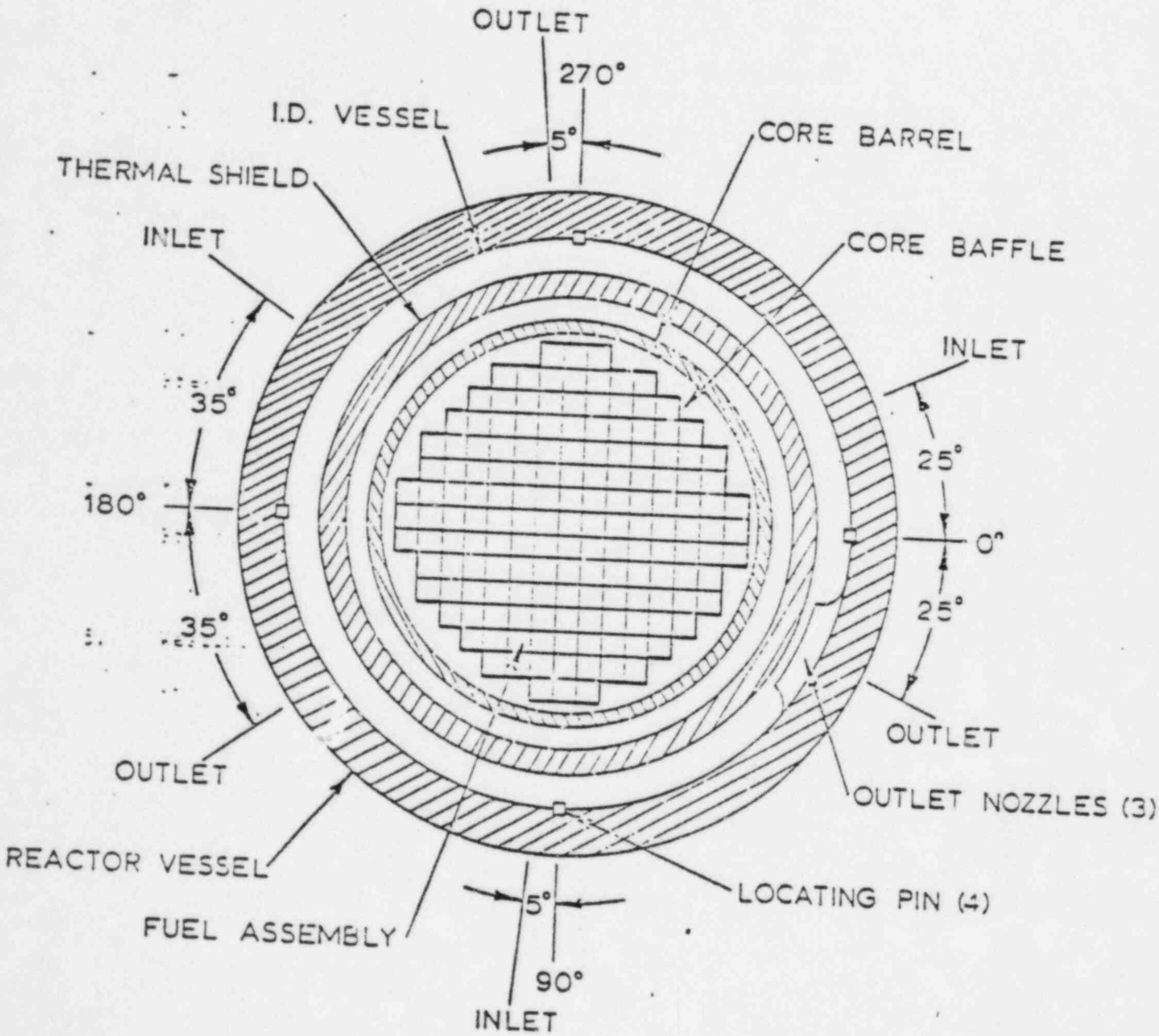
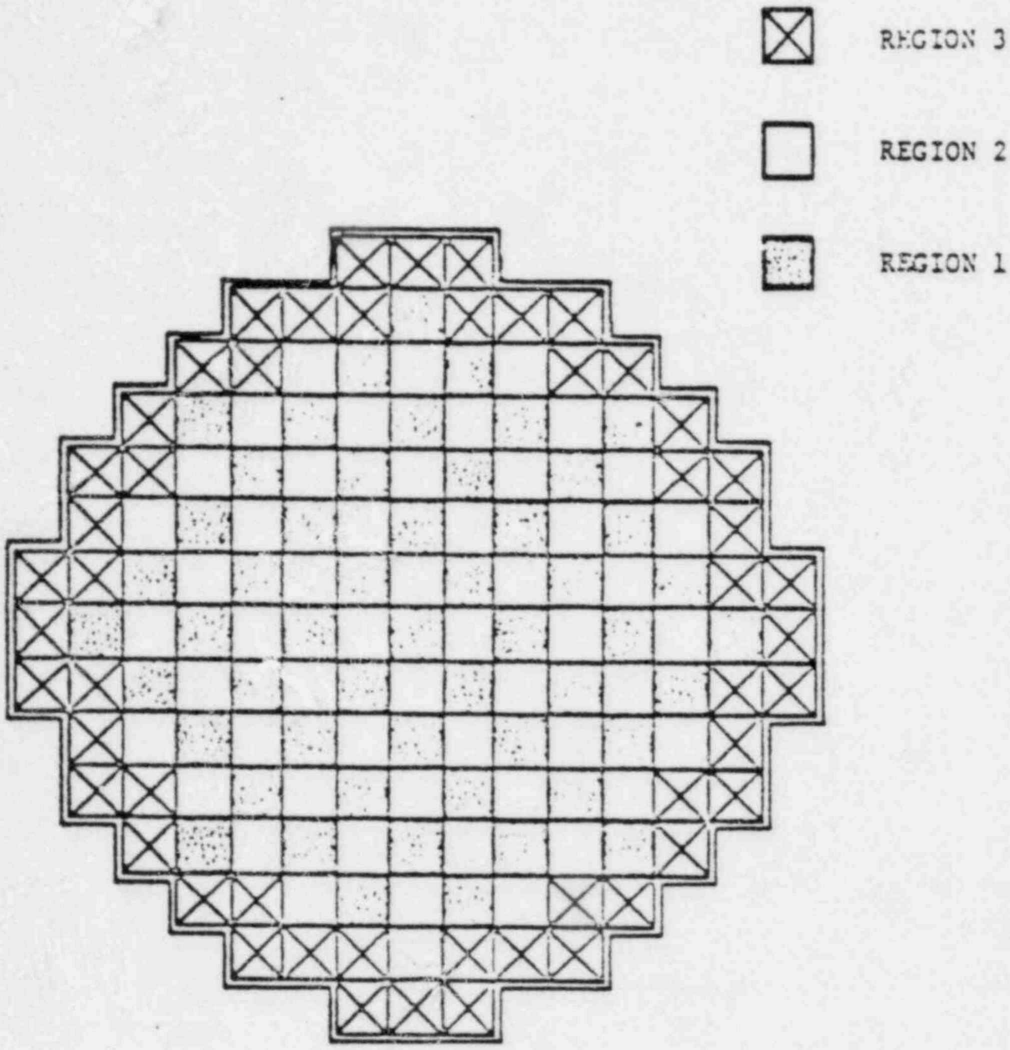
* Consult Sketch on P5.1

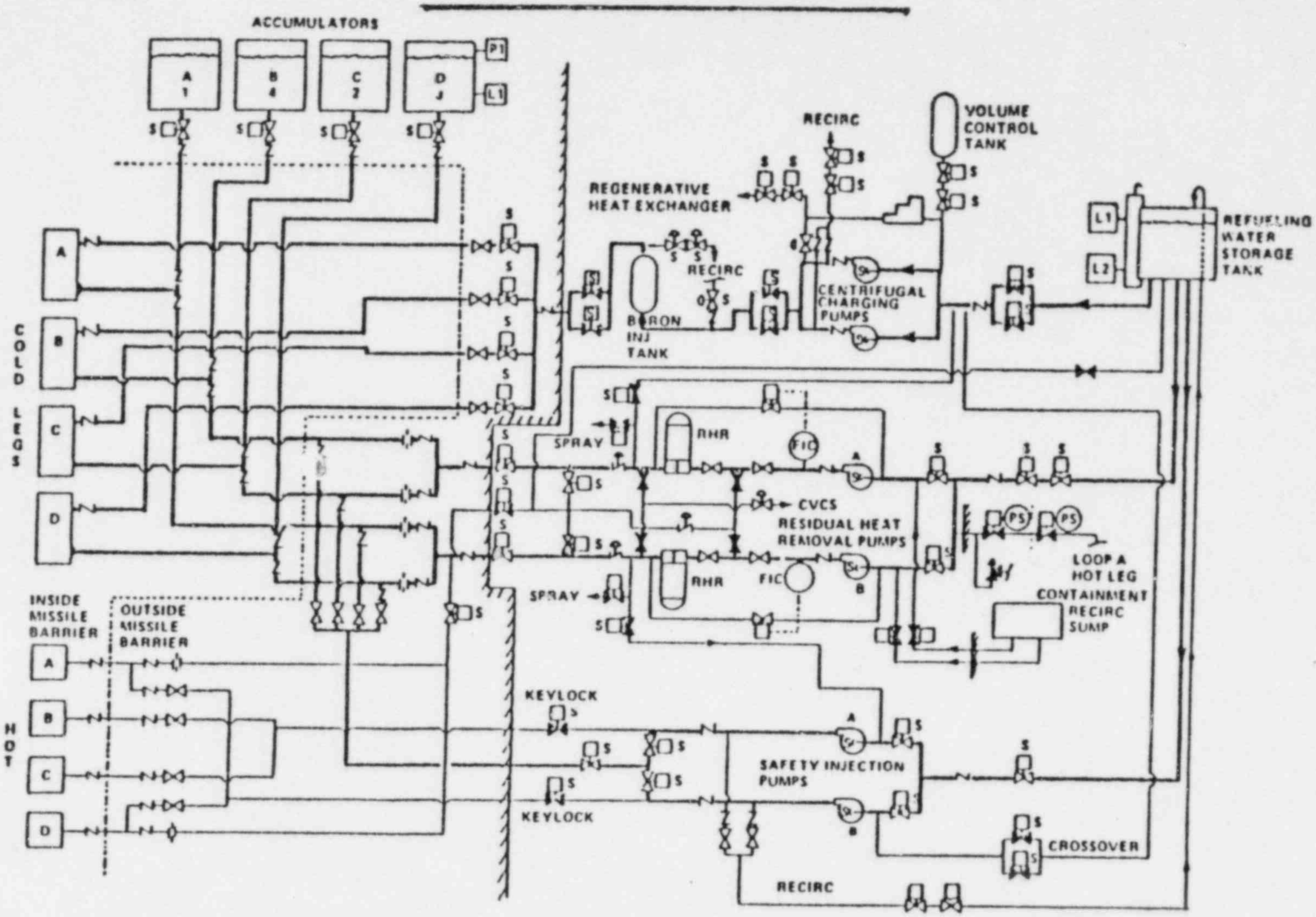


CORE CROSS SECTION



CORE LOADING ARRANGEMENTS

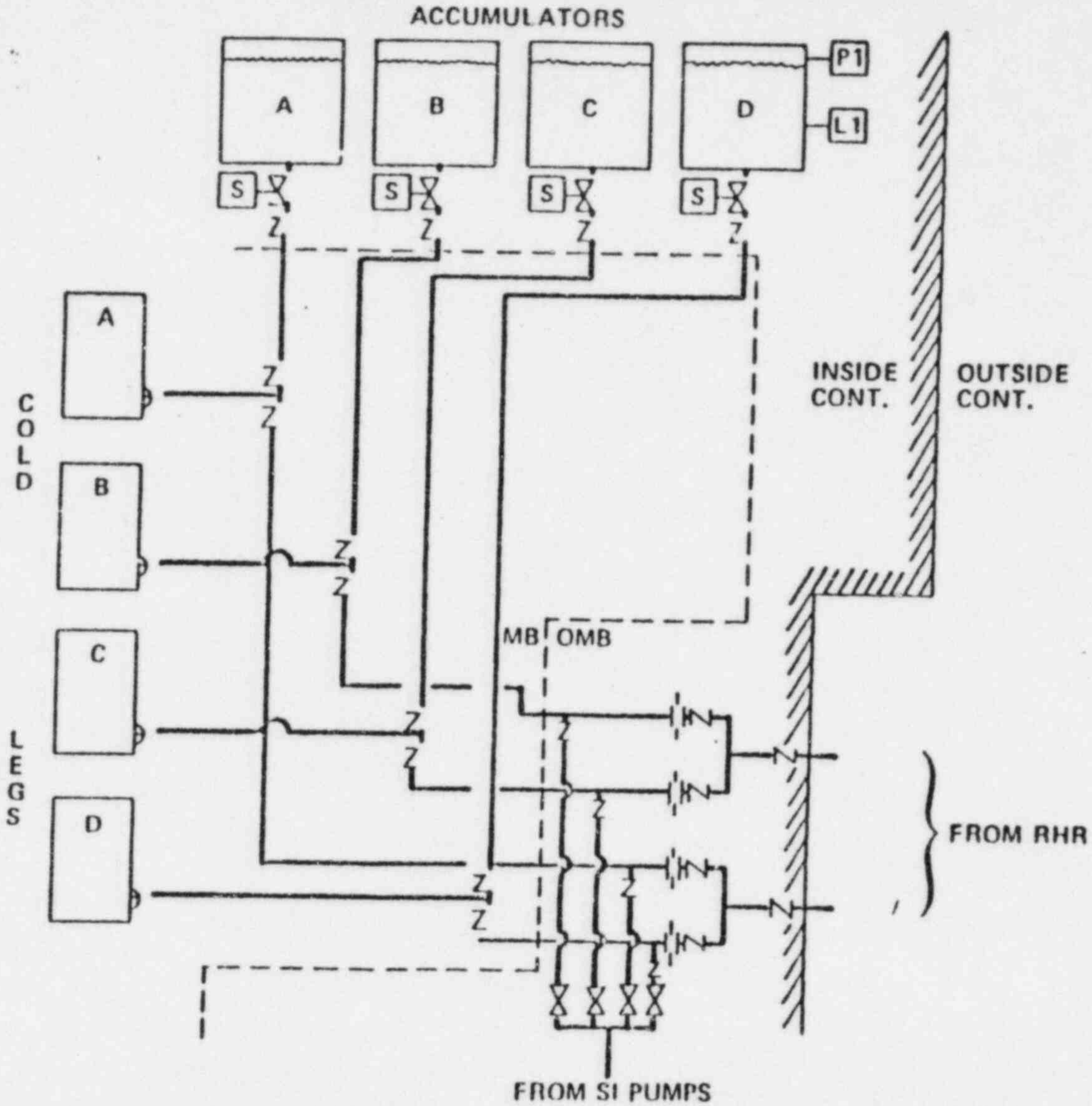
EMERGENCY CORE COOLING SYSTEM



NOTE: ACTUALLY THERE ARE ONLY THREE(3) LOOPS; COLD LEGS AND ACCUMULATORS, ETC.

P6.1

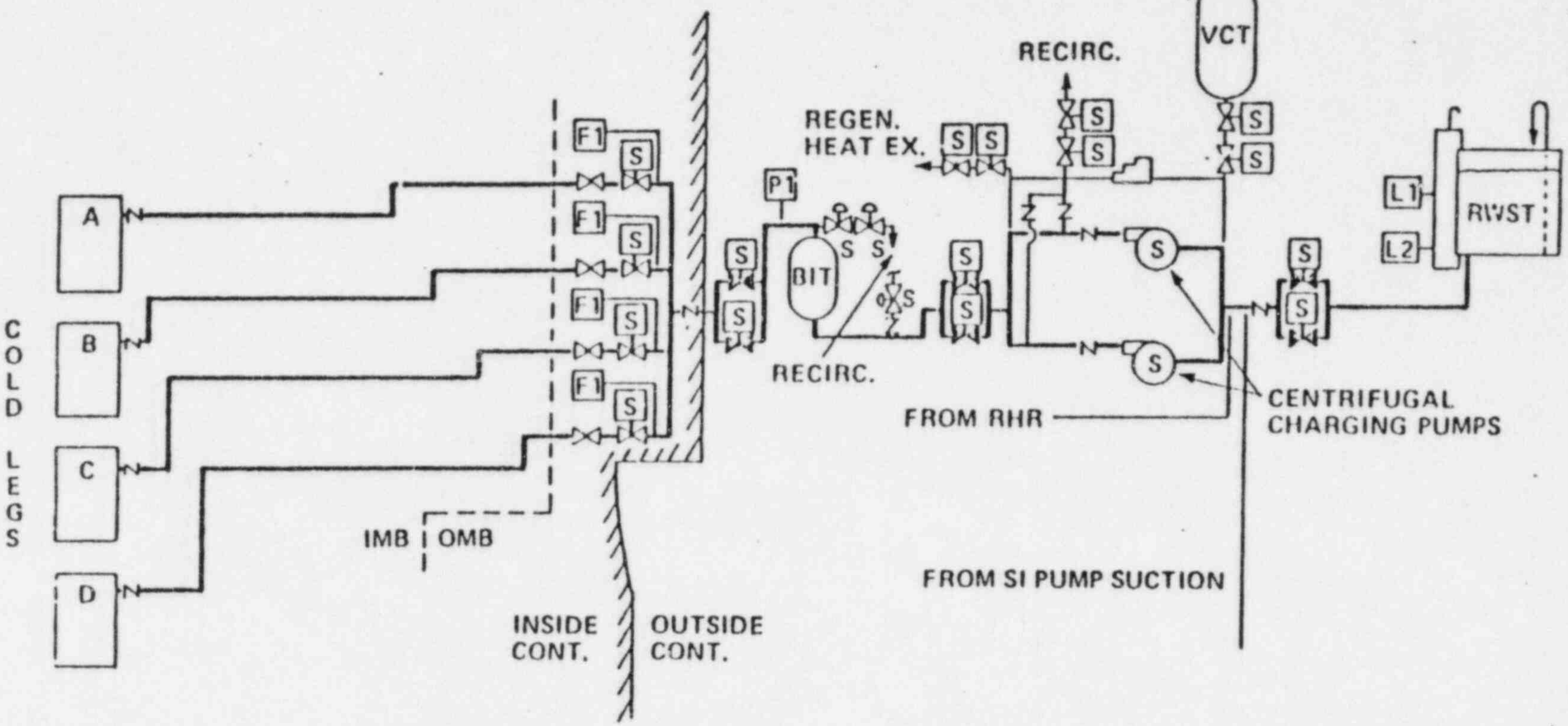
ACCUMULATOR SYSTEM



ACTUALLY: 3 LOOPS ONLY

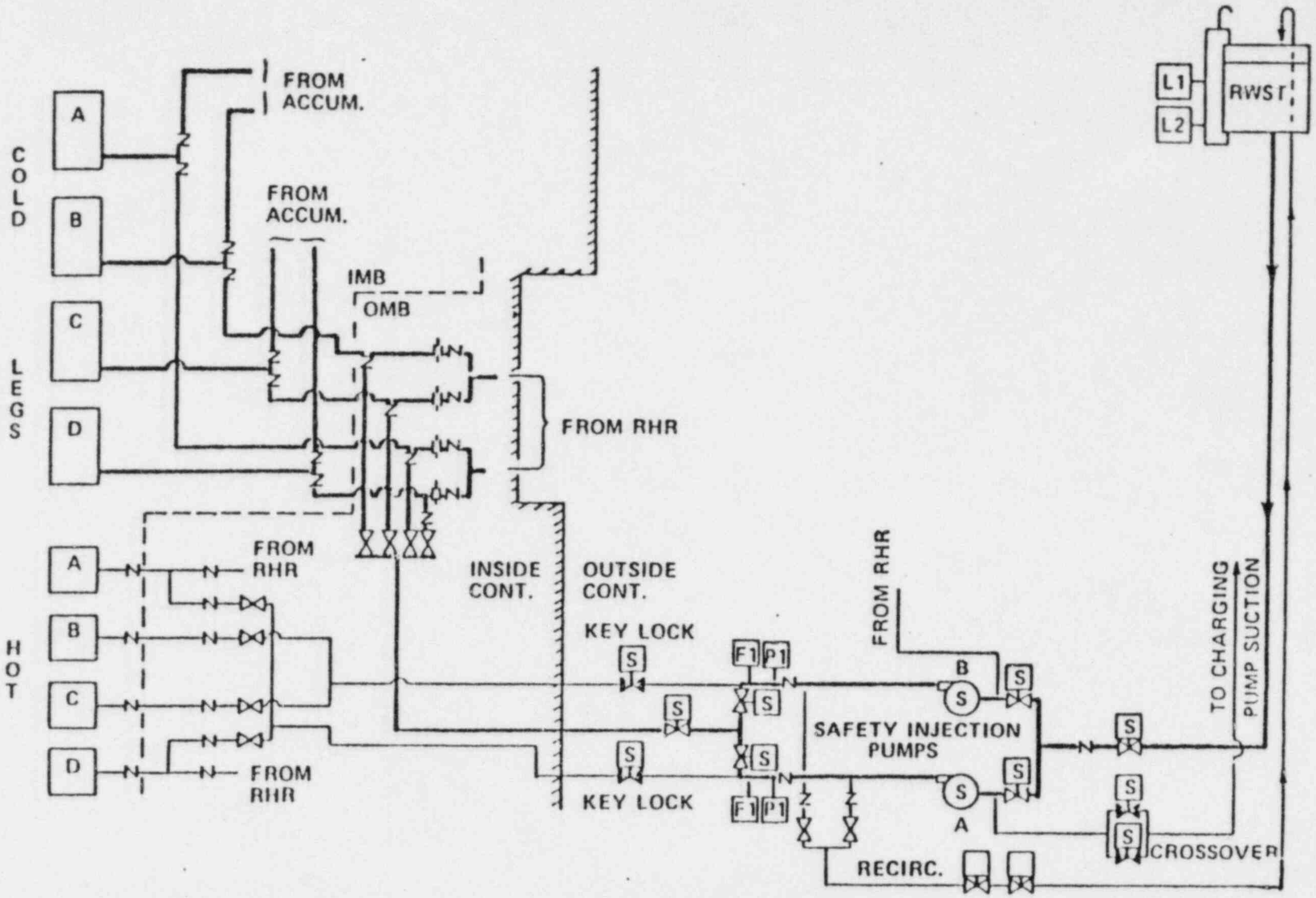
P6.2

HIGH HEAD INJECTION SYSTEM



ACTUALLY 3 LOOPS ONLY

SAFETY INJECTION PUMP SYSTEM



ACTUALLY 3 LOOPS ONLY

P6.4

COOLANT INVENTORIES AND SELECT COMPONENT VOLUMES FOR THE SURRY 1 PWR

P8.1

	VOLUME (ft^3)	MASS (lb_m)
* REACTOR VESSEL Total Water, Internals in place	3718	130 518
PIPING 3 loops	1041	36 544
STEAM GENERATOR (@ Hot Full Power)	1077	37 807
Primary Water	1689	59 287
Secondary Water	3870	29 271
Secondary Steam	↓	↓
PRESSURIZER (@ Hot Full Power, T=653°F)		
Water	780	27 387
Steam	520	3933
PRESSURIZER RELIEF TANK		
Total Volume	1300	— NA —
Standby Water (T=120°F)	900	55 542
ACCUMULATORS (3)		
Total	4350	— NA —
Water	2775	171 032
REFUELLING WATER STORAGE TANK (T=45°F)	46 788	2 920 599

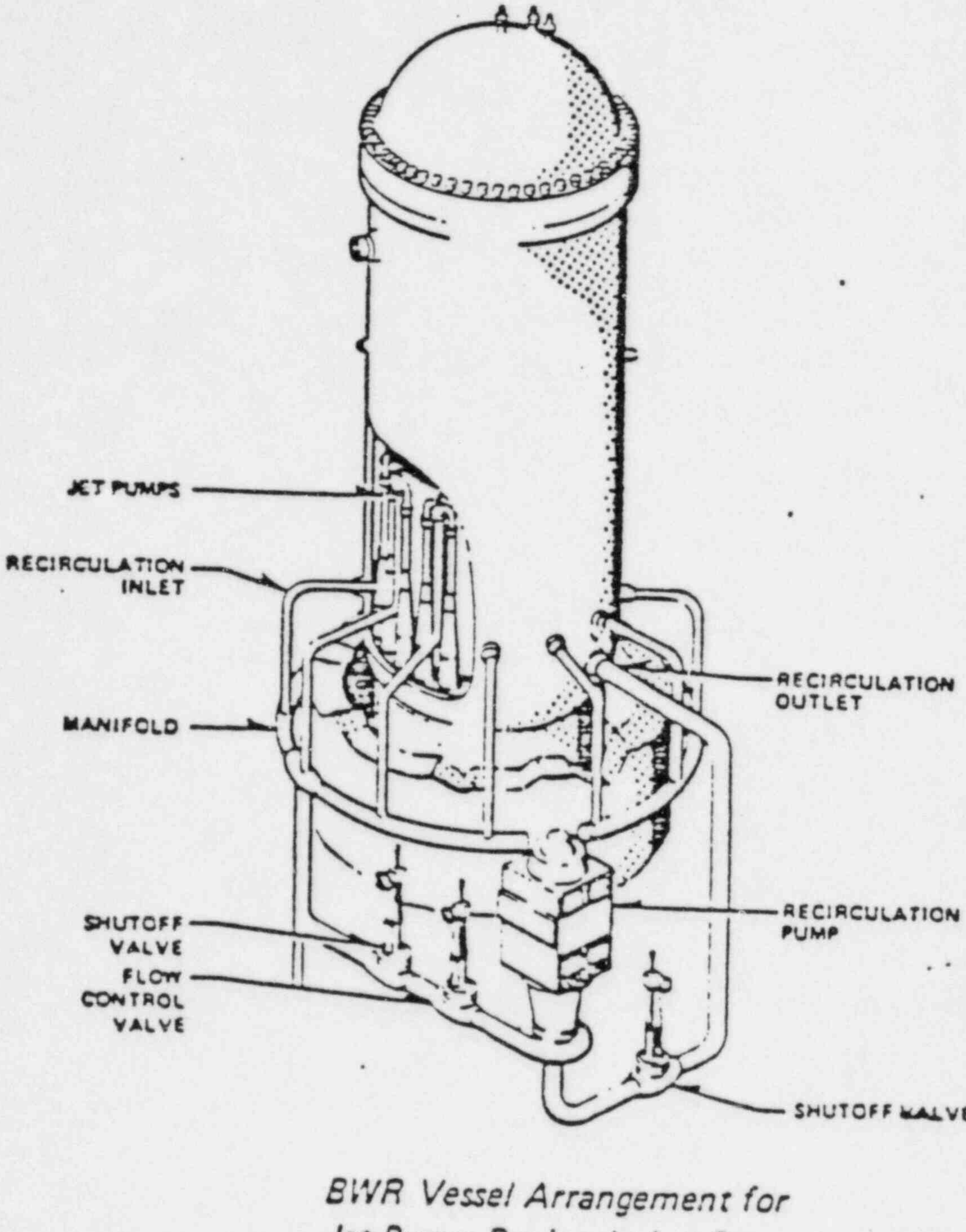
* SEE P8.2 FOR BREAKDOWN

WATER VOLUME
DISTRIBUTION WITHIN THE VESSEL

	VOLUME (ft^3)
BELOW CORE BOTTOM	910
CORE REGION	960
. RE TOP TO PIPING (NOZZLE CENTERS)	429
PIPING TO FLANGE	965
TOP HEAD	454
TOTAL	3718

PEACH BOTTOM 2 BWR

DATA



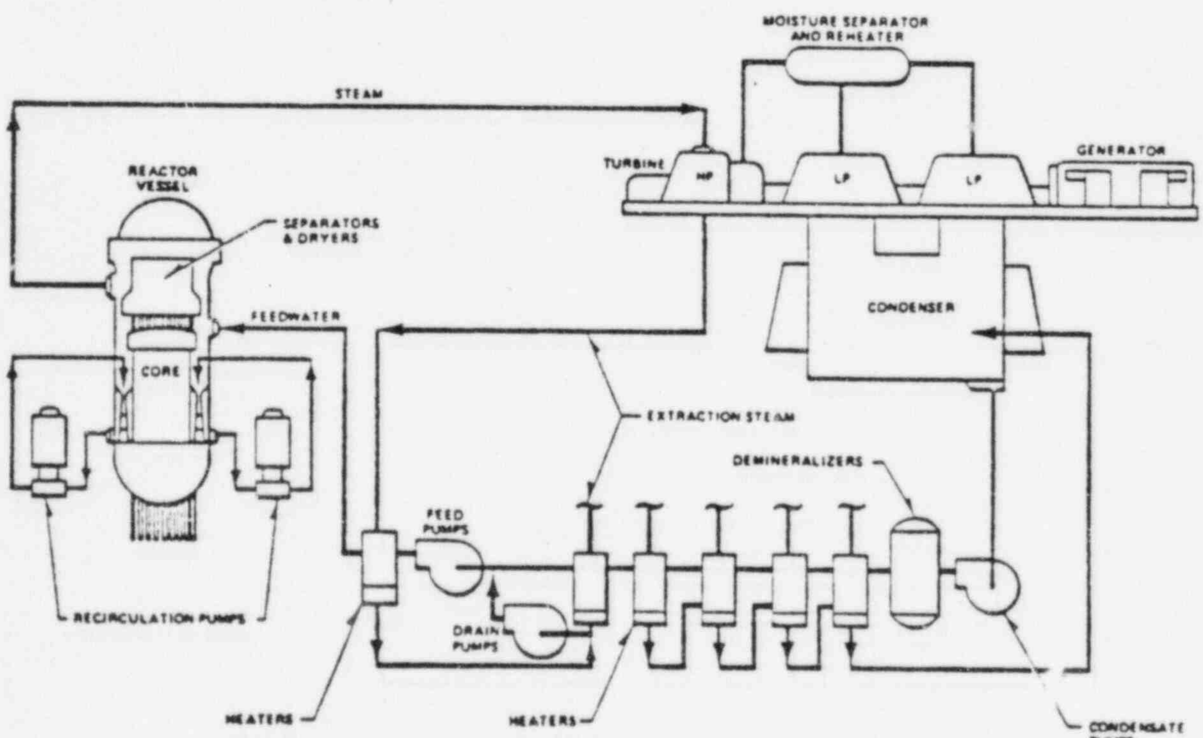
BWR Vessel Arrangement for Jet Pump Recirculation System

SUMMARY DESCRIPTION

The direct cycle boiling water reactor nuclear system (Figure 1-1) is a steam generating system consisting of a nuclear core and an internal structure assembled within a pressure vessel, auxiliary systems to accommodate the operational and safeguard requirements of the nuclear reactor, and necessary controls and instrumentation. Water is circulated through the reactor core, producing saturated steam which is separated from recirculation water, dried in the top of the vessel, and directed to the steam turbine-generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization.

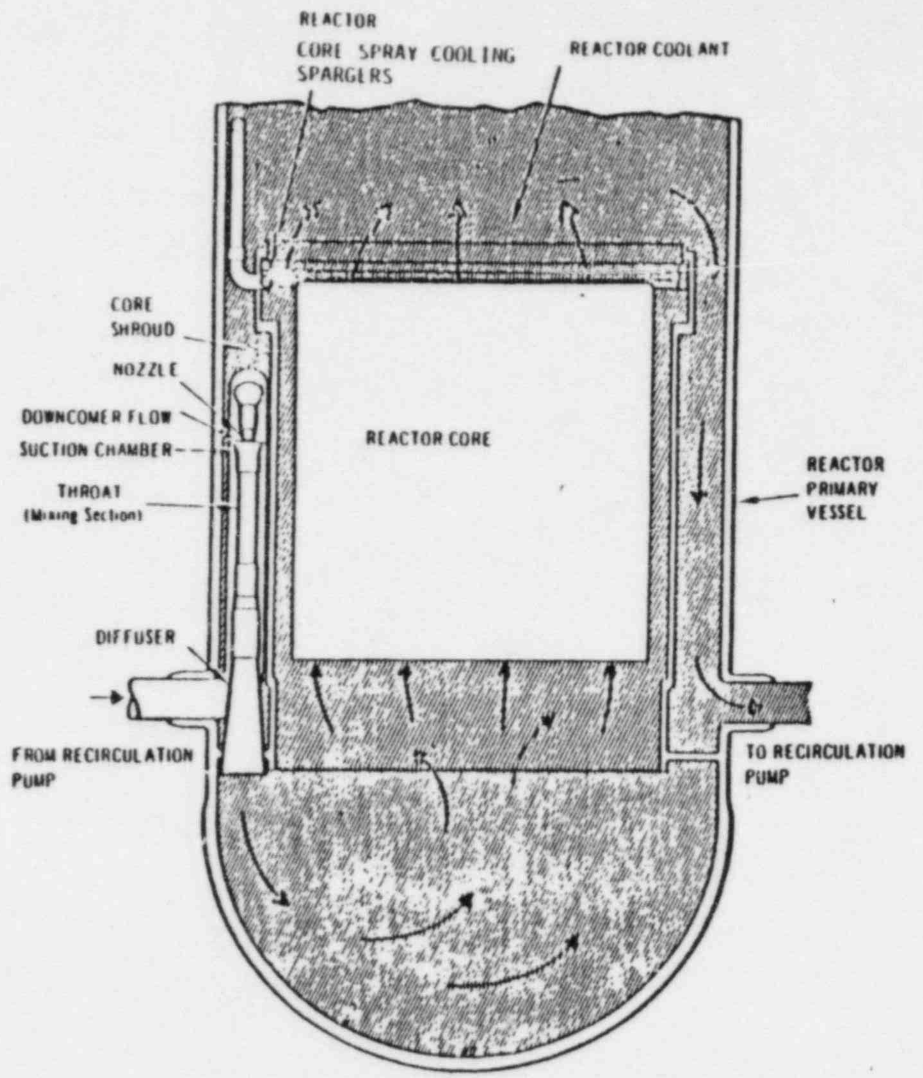
The steam from a boiling water reactor is, of course, radioactive. The radioactivity is primarily N-16, a very short-lived isotope (7 seconds half-life) so that the radioactivity of the steam system exists only during power generation. Extensive generating experience has fully demonstrated that shutdown maintenance on a BWR turbine, condensate, and feedwater components can be performed essentially as at a fossil-fuel plant. Carryover of long-lived radioactive particles from the primary system to the turbine/feedwater system is virtually nonexistent. More than 300 billion kWh of successful operating experience and numerous refueling and maintenance outages in plants using this direct-cycle approach support the soundness of General Electric's choice of the BWR.

The reactor core, the source of nuclear heat, consists of fuel assemblies and control rods contained within the reactor vessel and cooled by the recirculating water system. A 1220-MWe BWR/6 core consists of 748 fuel assemblies and 177 control rods, forming a core array about 16 feet in diameter and 14 feet high. (See Section 3 for detailed description of the BWR core.) The power level is maintained or adjusted by positioning control rods up and down within the core. The BWR core power level is further adjustable by changing the recirculation flow rate without changing control rod position. This unique BWR feature helps achieve the superior load-following capability of the BWR.



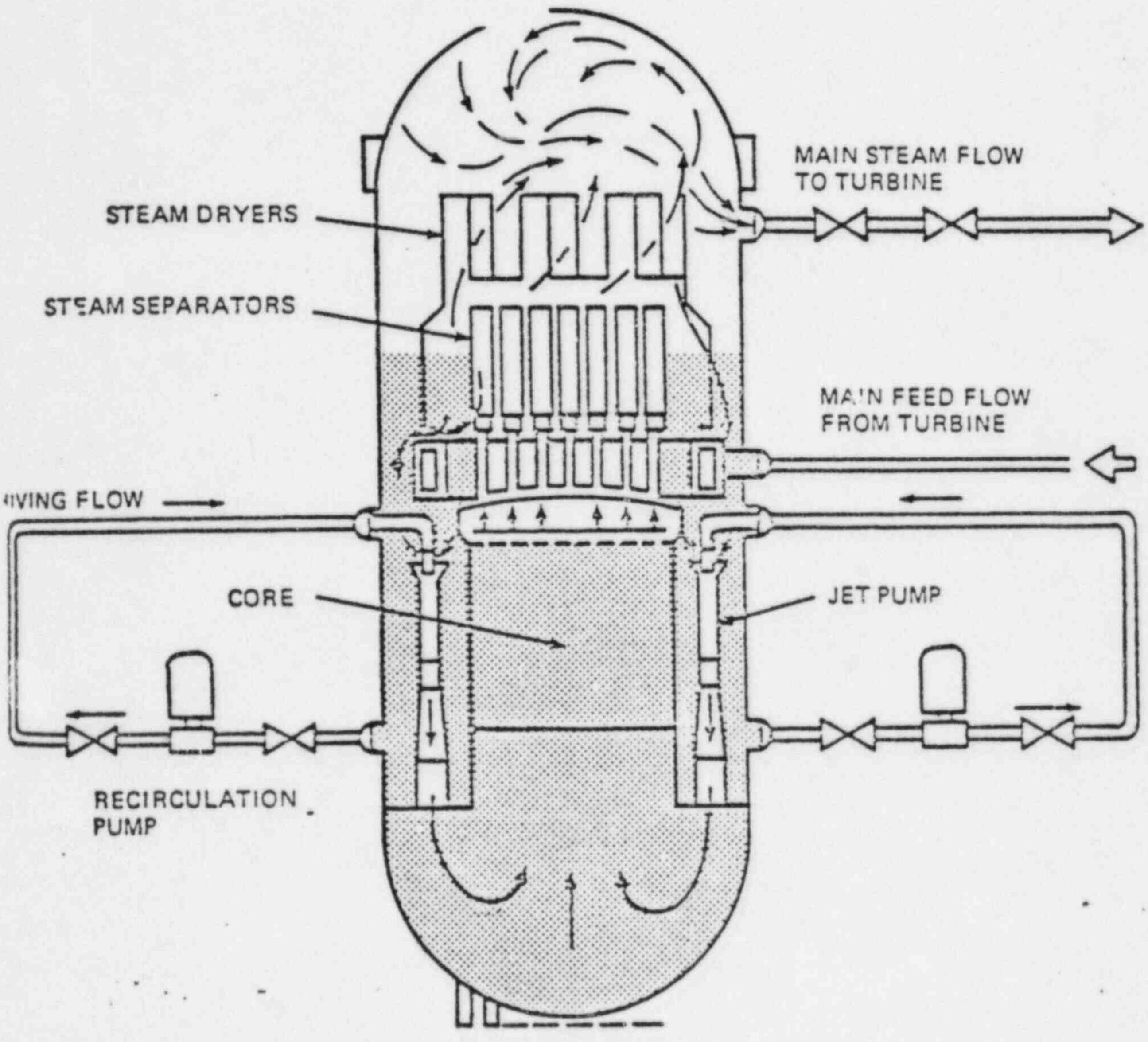
Direct-Cycle Reactor System

Schematic

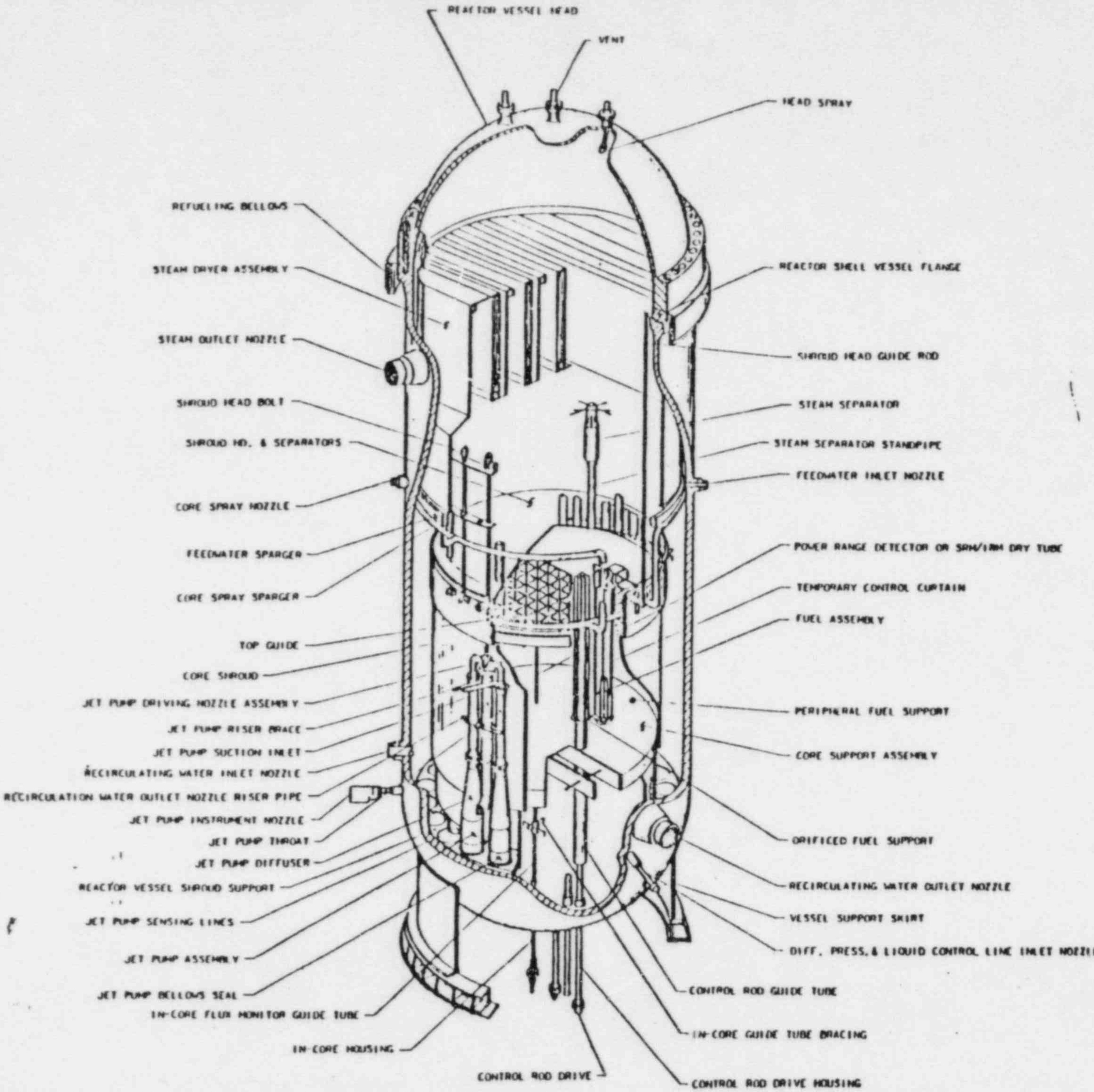


PHILADEL
PEACH BOTT
FINAL S
REACTOR V

B2.2



Reactor Vessel Internal Flow Paths



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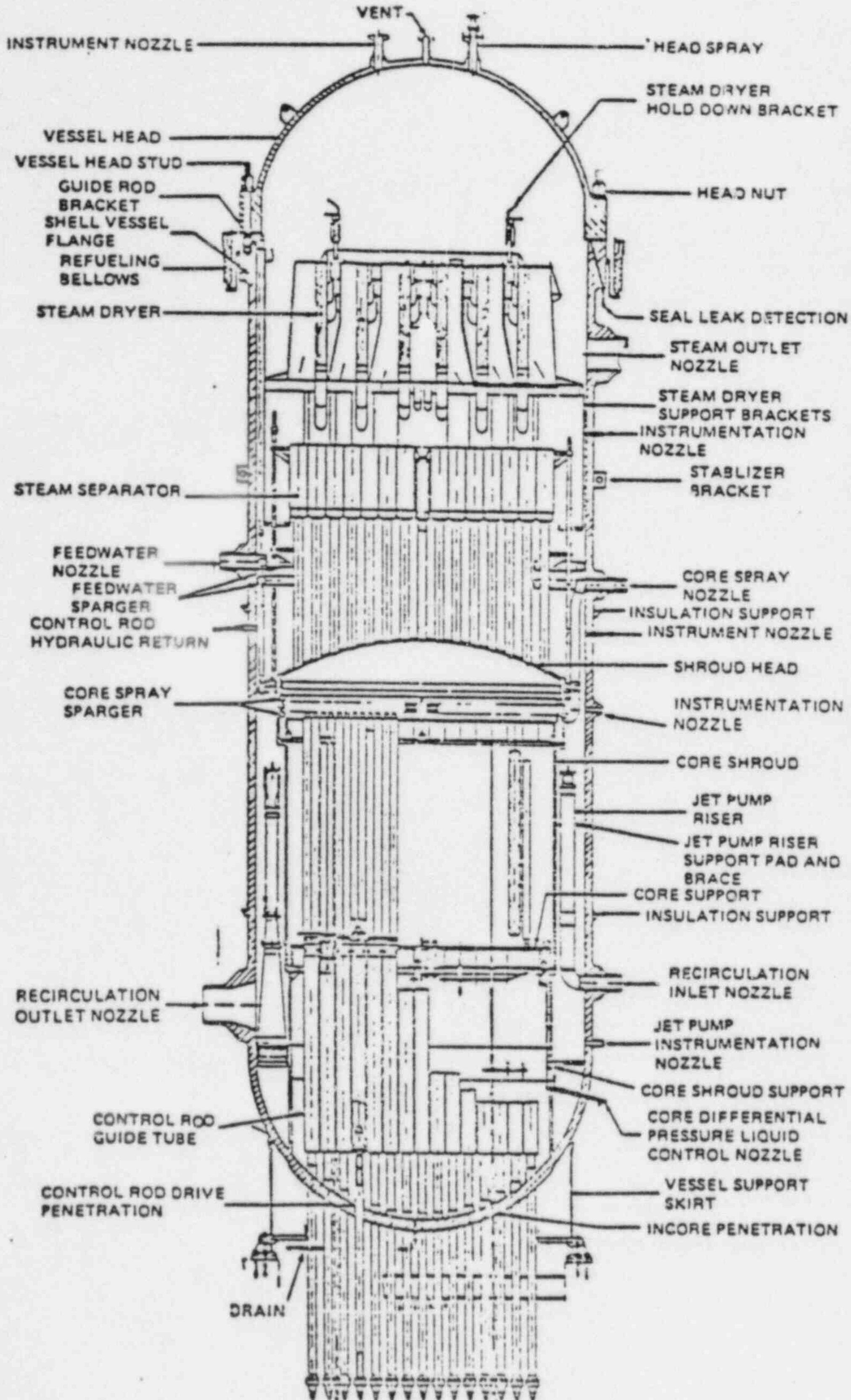
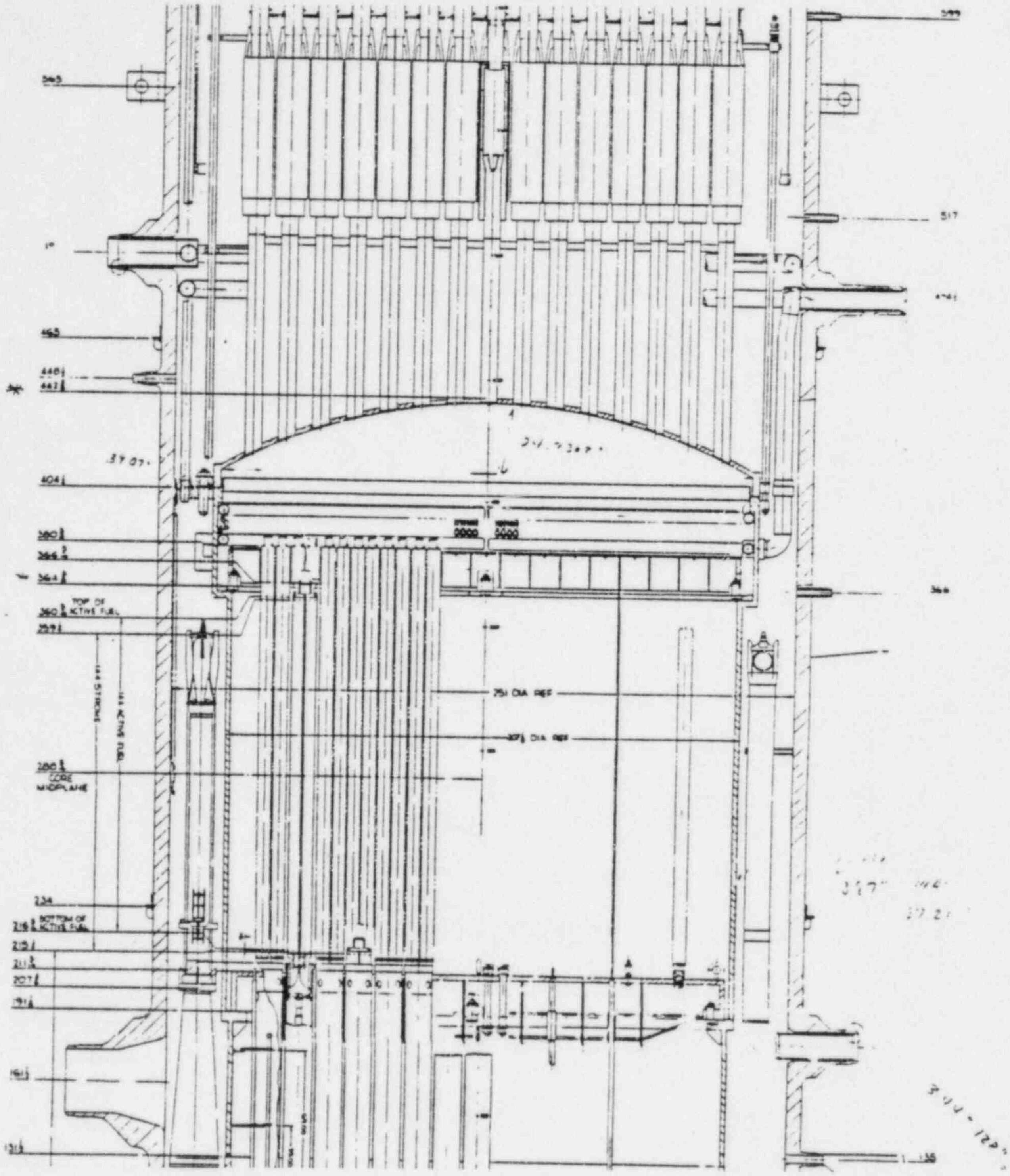
B3.1

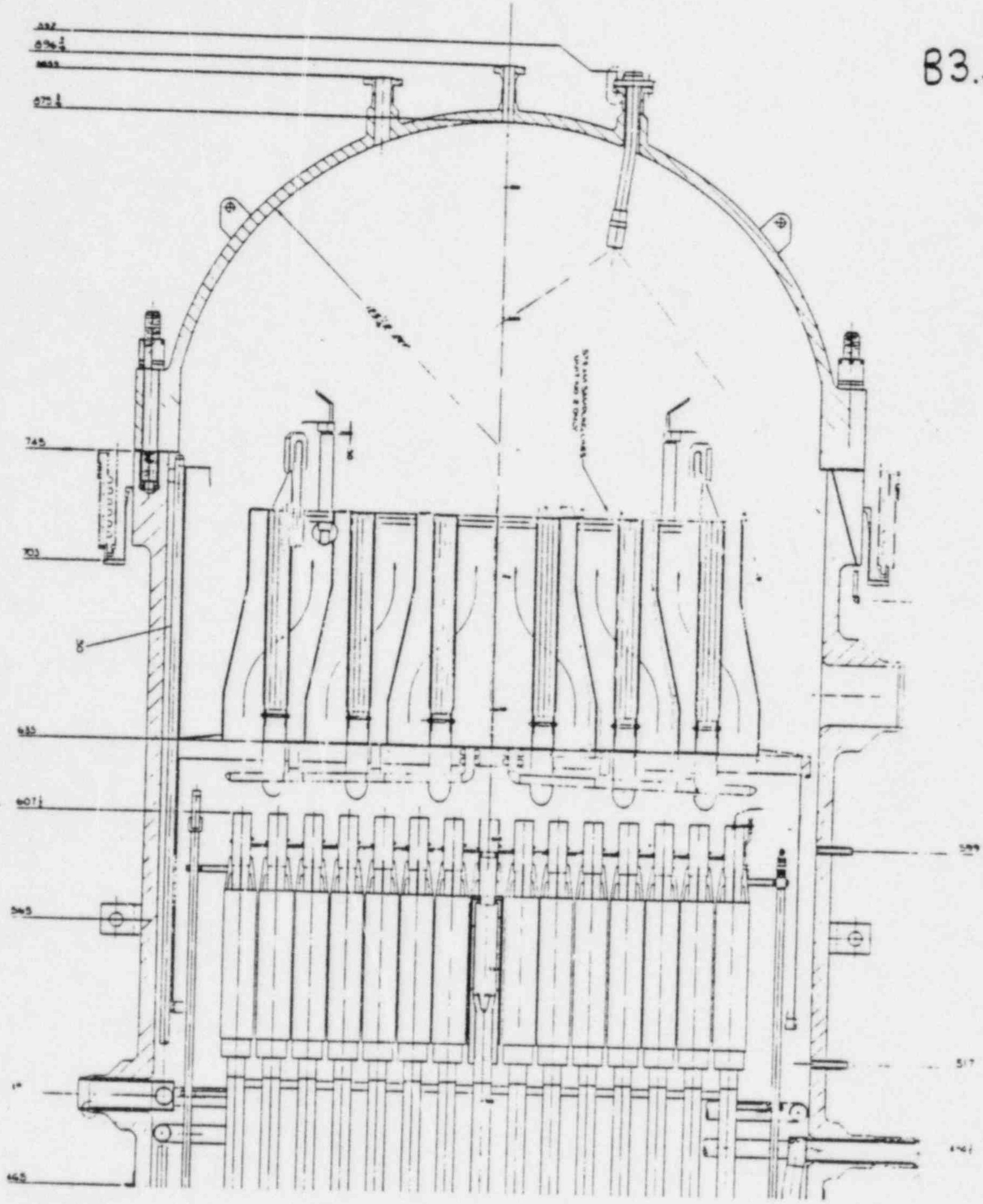


Reactor Vessel

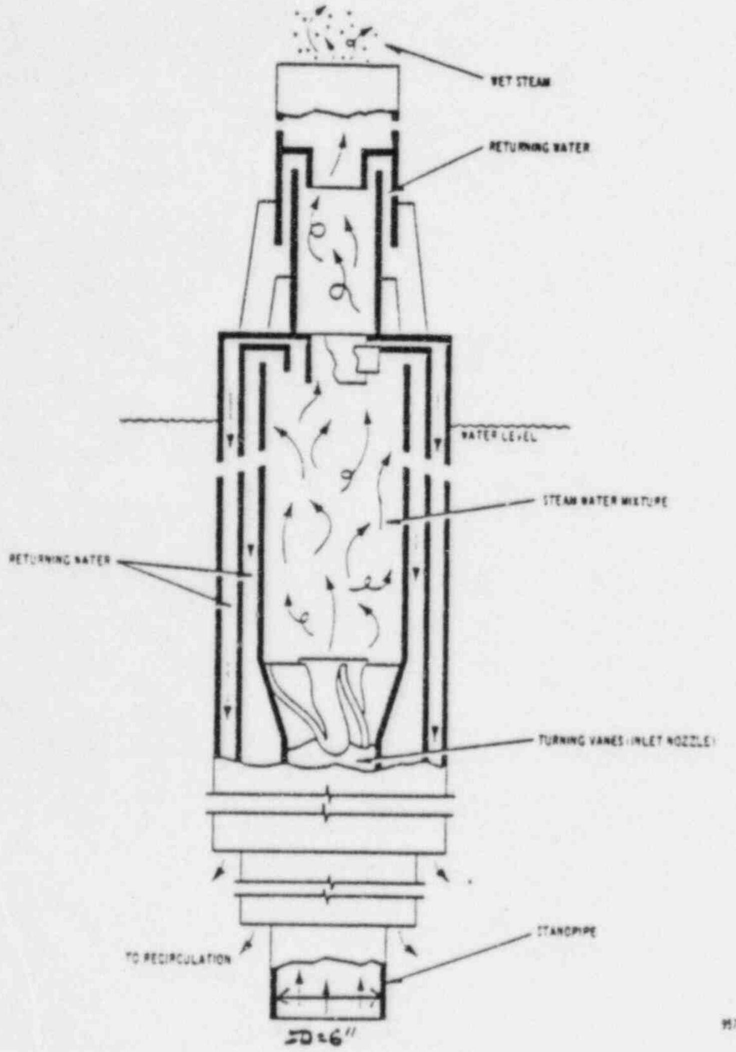


TOP OF ACTIVE FUEL
CORE MIDPLANE
BOTTOM OF ACTIVE FUEL
ACTIVE FUEL
251 DIA REF
234
517
599

B3.3



B 4.1



95761

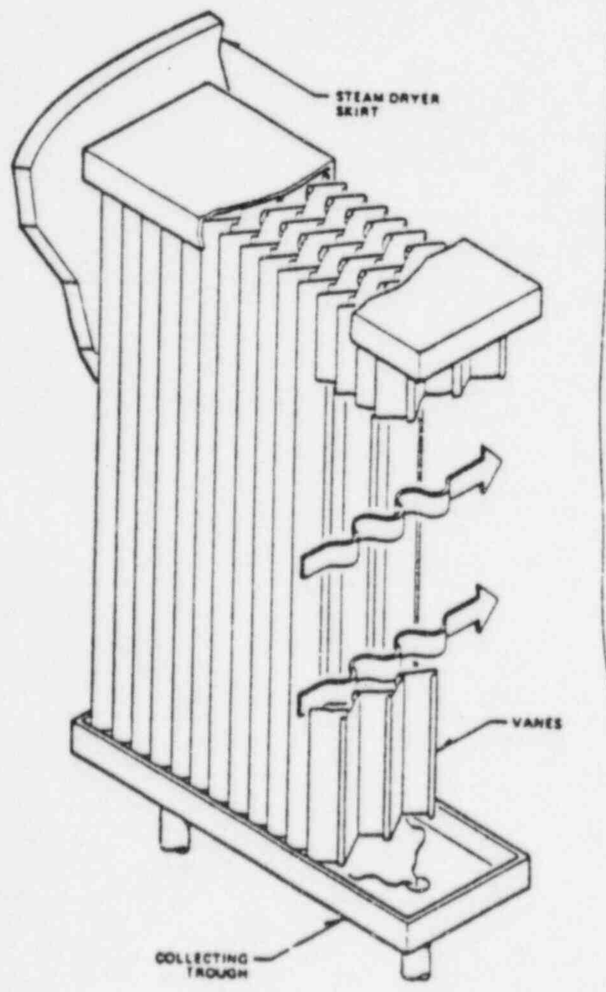
PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3
FINAL SAFETY ANALYSIS REPORT

STEAM SEPARATOR

CONTROL ROD DRIVE HOUSING SUPPORT

The control rod drive housing support is an engineered safeguard designed to prevent a nuclear transient in the highly unlikely event that there is a control rod drive housing failure. This device consists of a beam structure, from which housing supports are suspended, located below the reactor vessel. The supports restrict vertical outward movement of the control rod drive or housing in the event of failure and are easily removable for drive maintenance.

The control rod drive housings are also restrained with respect to lateral seismic loadings by sets of adjustable spacer blocks (which are bolted to every other control drive housing flange) and by a beam attached to the reactor support pedestal.



Steam Dryer

REACTOR WATER RECIRCULATION

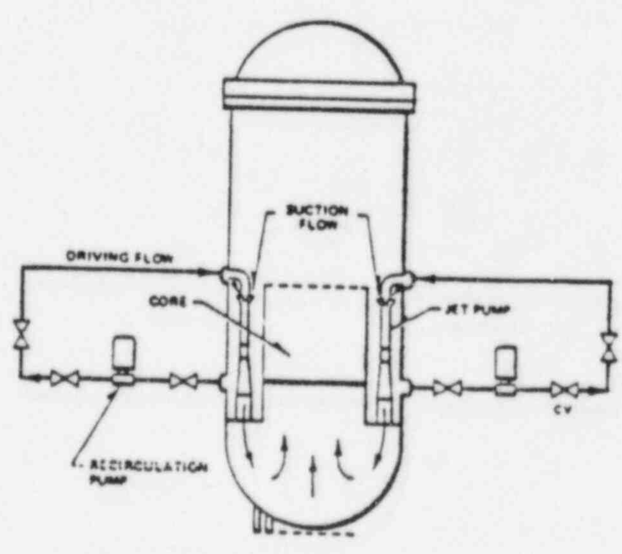
The function of the reactor water recirculation system is to circulate the required coolant through the reactor core. The system consists of two loops external to the reactor vessel each containing a pump with a directly coupled water-cooled (air-water) motor, a flow control valve, and two shutoff valves.

High performance jet pumps located within the reactor vessel are used in the BWR recirculation system. The jet pumps, which have no moving parts, provide a continuous internal circulation path for a major portion of the core coolant flow. The system was first incorporated in the Dresden 2 plant design and since has been a standard part of the General Electric boiling water reactor.

Flow Path

The jet pump recirculation system (Figure 2-4) provides forced circulation flow through BWR cores.

The recirculation pumps take suction from the downward flow in the annulus between the core shroud and the vessel wall. Approximately one-third of the core flow is taken from the vessel through the two recirculation nozzles. There, it is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump nozzle into the initial stage of the jet pump throat where, due to a momentum exchange process, it induces surrounding water in the downcomer region to be drawn into the jet pump throat where these two flows mix and then diffuse in the diffuser, to be finally discharged into the lower core plenum.



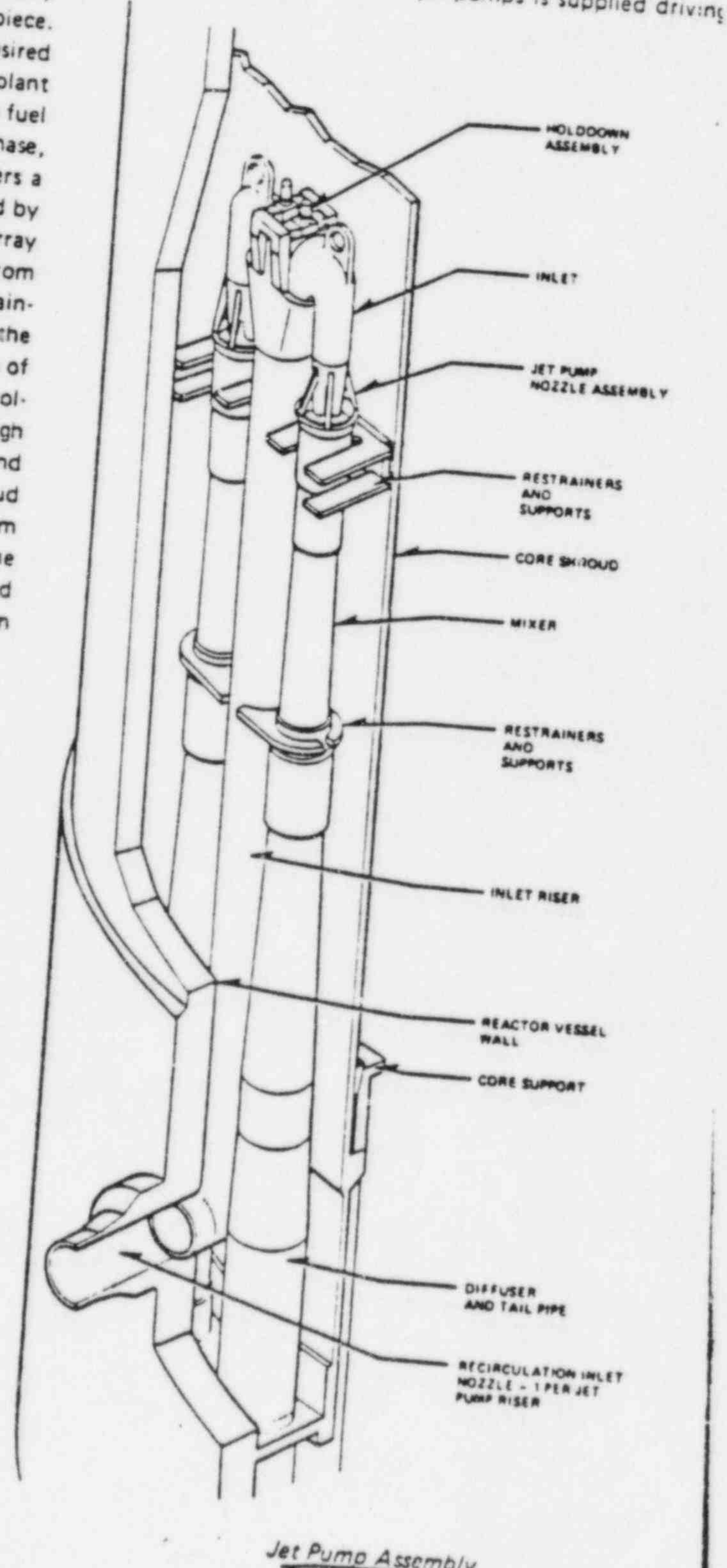
Jet Pump Recirculation System

The jet pump diffusers are welded into openings in the core shroud support shelf which forms a barrier between the lower plenum and the suction side of the jet pump. The flow of water turns upward, where it flows between the control rod drive guide tubes and enters into the fuel support where the flow is individually directed to each fuel bundle through the nose piece. Orifices in each fuel support piece provide the desired flow distribution among the fuel assemblies. The coolant water passes along the individual fuel rods inside the fuel channel where it is heated and becomes a two-phase, steam-water mixture. The steam-water mixture enters a plenum located directly above the core and bounded by the separator dome which opens to the separator array of fixed steam separators. The steam is separated from the water and passes through a dryer where any remaining water is removed. The saturated steam leaves the vessel through steam line nozzles located near the top of the vessel body and is piped to the turbine. Water collected in the support tray of the dryer is routed through drain lines, joins the water leaving the separators, and flows downward in the annulus between the core shroud and the vessel wall. Feedwater is added to the system through spargers located above the annulus and joins the downward flow of water. A portion of this downward flow enters the jet pumps and the remainder exits from the vessel as recirculation flow.

B4.3

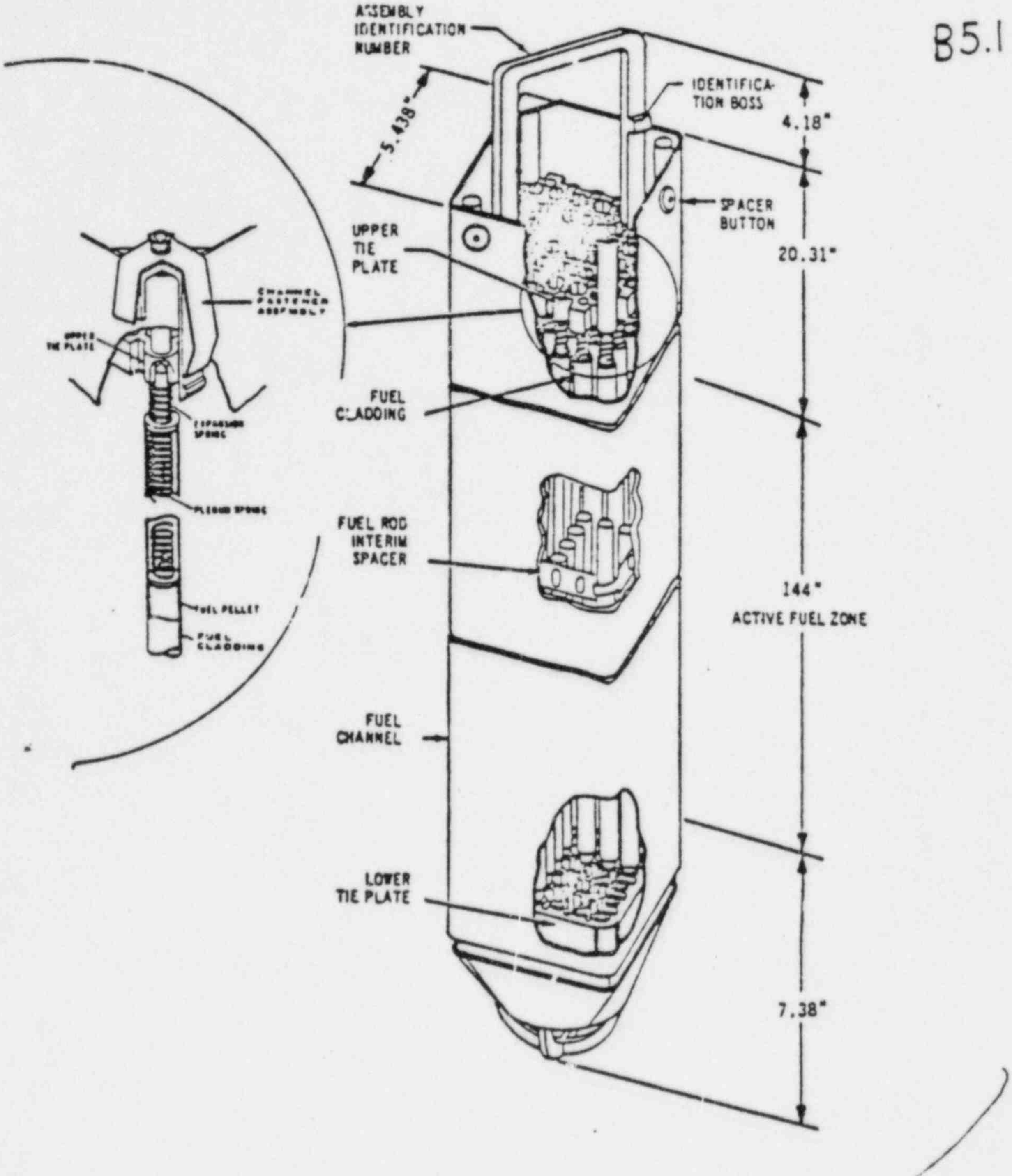
Jet Pump Applications in BWRs

The jet pumps are located in the annular regio[n] between the core shroud and the vessel inner wal[l] (Figure 2-5). Each pair of jet pumps is supplied driving



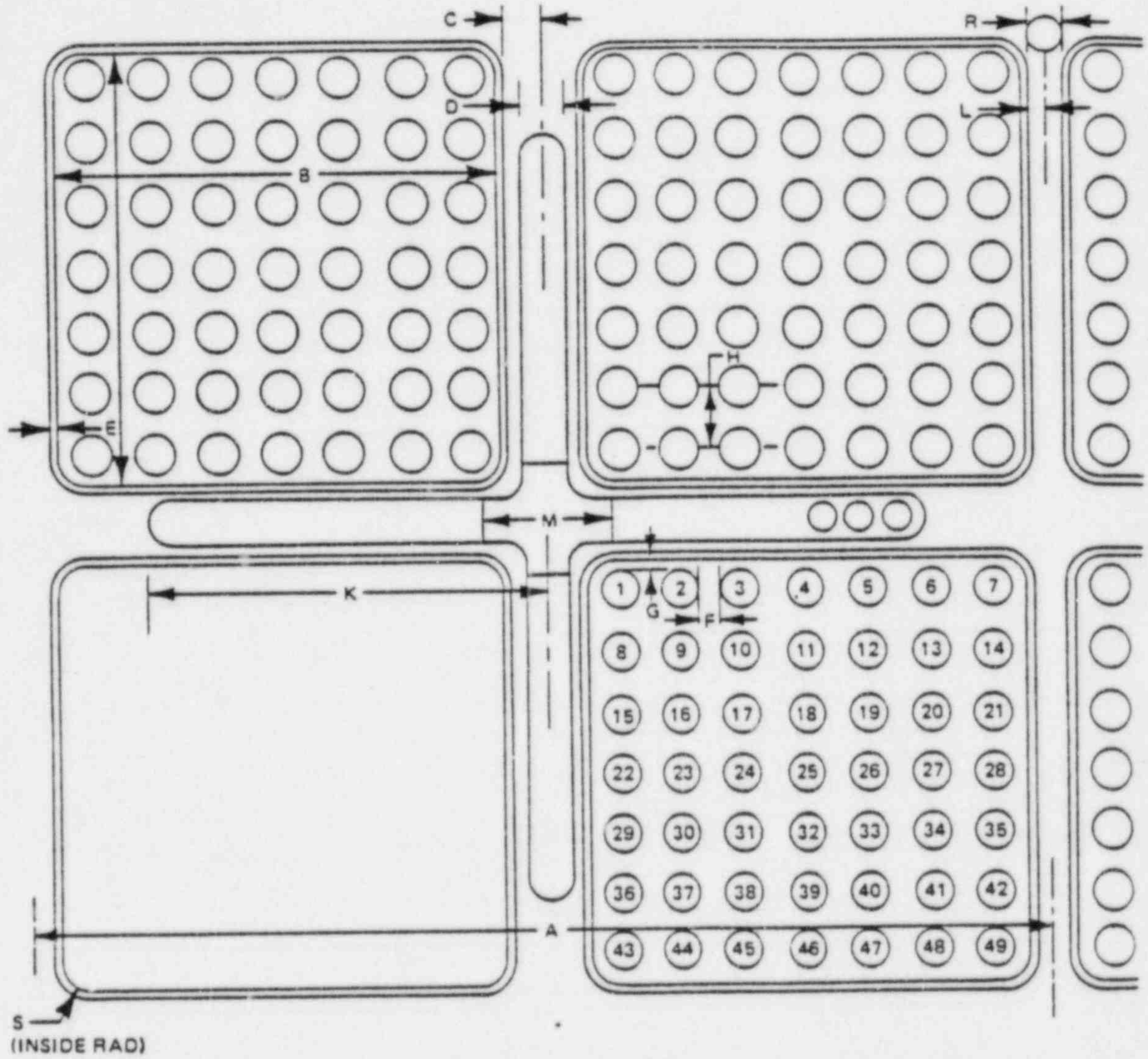
Jet Pump Assembly

B5.1



PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3
FINAL SAFETY ANALYSIS REPORT

FUEL ASSEMBLY - ISOMETRIC



DIM. IDENTIFICATION	A	B	C	D	E	F	G	H	I	J
DIM. INCHES	12.0	5.278	0.375		0.080	0.175	0.1435	0.738		

DIM. IDENTIFICATION	K	L	M	N	O	P	Q	R	S	
DIM. INCHES		0.187							0.380	

Figure 10. Initial Fuel Assembly Lattice

B5.3

INITIAL FUEL DESCRIPTION

	Type 1	Type 2	Type 3
Fuel Assembly			
Number of Fuel Assemblies per Batch ..	168	263	333
Fuel Rod Array	7 x 7	7 x 7	7 x 7
Fuel Rod Pitch, in	0.738	0.738	0.738
Peripheral-Rod-to-Channel Spacing, in.	0.1435	0.1435	0.1435
1/2 Width of Wide Water Gap, in	0.375	0.375	0.375
1/2 Width of Narrow Water Gap, in	0.188	0.188	0.188
Cladding Length, in	160	160	160
Bundle Average Enrichment (wt % U-235 in Total U)	1.10	2.50	2.50
Control Augmentation			
Type	NONE	Fuel Rods Containing Gd_2O_3	Fuel Rods Containing Gd_2O_3
Number		4	5
Control Length, in		144(3), 60(1)	144(3), 108(1), 36(1
Control Material		3.0 wt % Gd_2O_3	3.0 wt % Gd_2O_3(3) 4.0 wt % Gd_2O_3(2)
Locations		In Fuel Lattice	In Fuel Lattice
Weight of U per Fuel Assembly			
lb	432.3	412.4	412.1
kg	196.1	187.1	186.9
Channel			
Outside Dimensions, in	5.438 x 5.438	5.438 x 5.438	5.438 x 5.438
Thickness, in	0.080	0.080	0.080
Inside Corner Radius, in	0.38	0.38	0.38
Material	Zr-4	Zr-4	Zr-4
Water-UO_2 Volume Ratio (cold)	2.43	2.53	2.53

B5.4

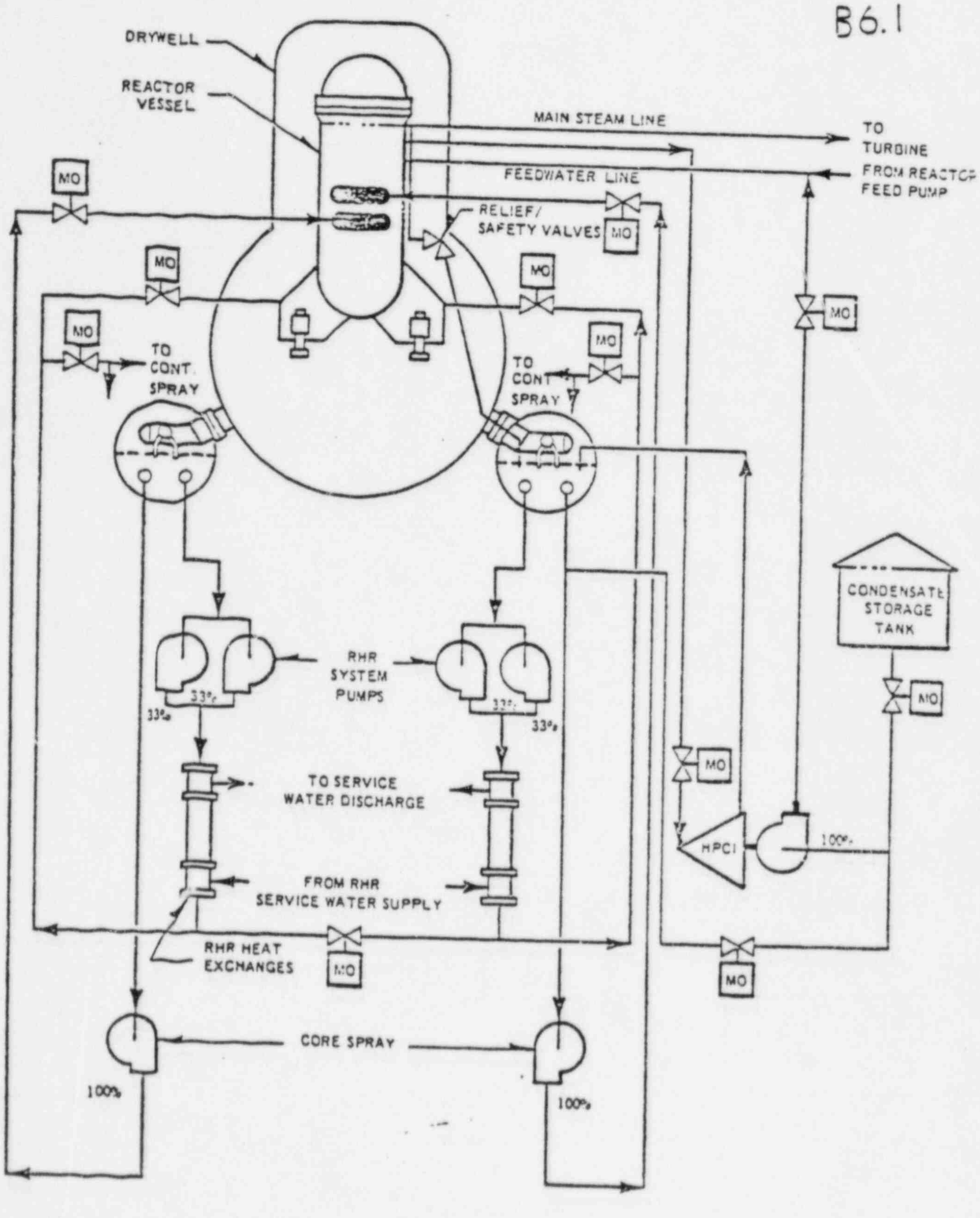
FUEL ASSEMBLY DATA

Initial Load

Assembly Type	1	2	3
No. of Assemblies, Initial Core	168	263	333
No. of Assemblies, Cycle 2	0	261	315
Geometry	7 x 7	7 x 7	7 x 7
Assembly Pitch, in	6.0	6.0	6.0
Fuel Rod Pitch	0.738	0.738	0.738
Fuel Rods per Assembly	49	49	49
Instrument Rods per Assembly	0	0	0
Water Rods per Assembly	0	0	0
Burnable Poison Positions	0	4	5
No. of Spacer Grids	7	7	7
Inconel per Grid, lb	0.102	0.102	0.102
Zr-4 per Grid, lb	0.537	0.537	0.537
Spacer Width, in	1.625	1.625	1.625
Assembly Average Fuel Composition			
Gd_2O_3, gm	0	441	547
UO_2, kg	222.44	212.21	212.06
Total Fuel, kg	222.44	212.65	212.61

*60 Assemolies channeled with 0.100 inch thick channels, 8 with 0.120 inch thick channels.

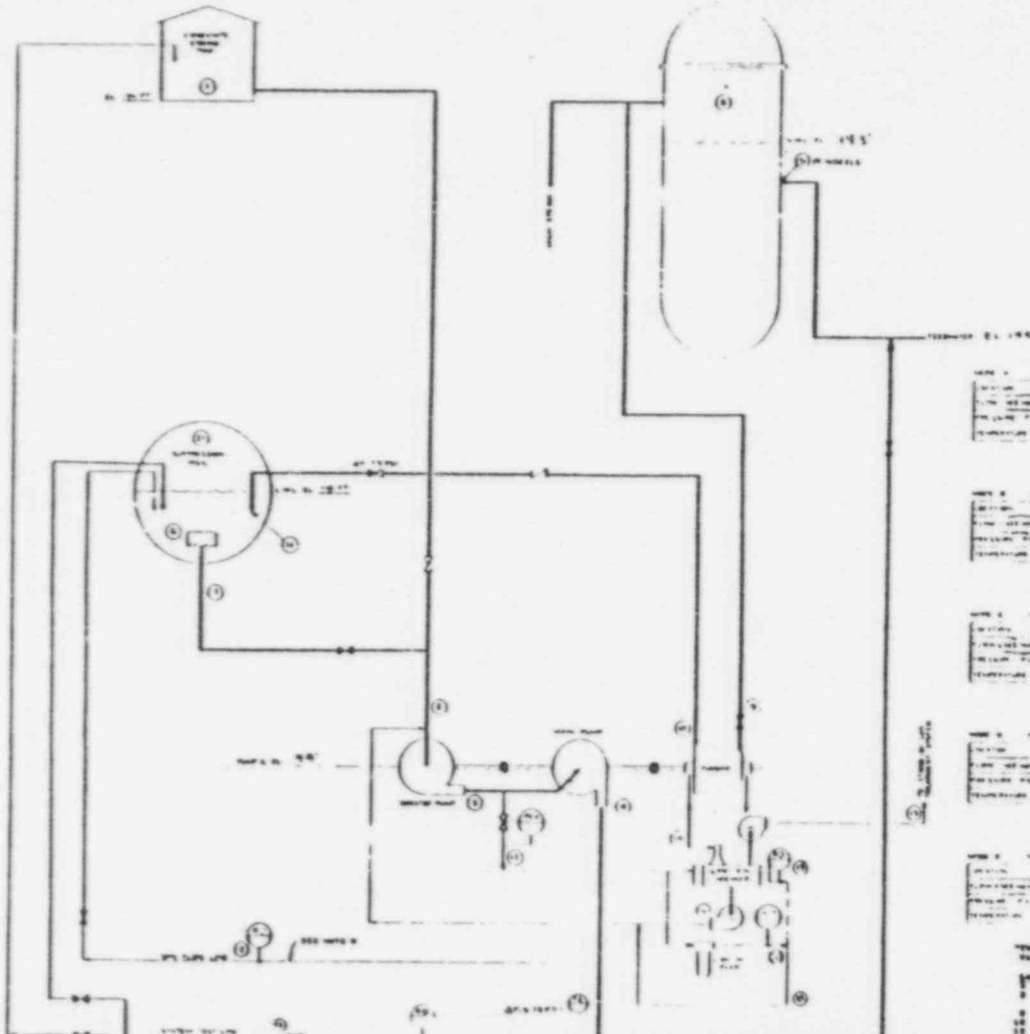
B6.1



Emergency Core Cooling System

HIGH PRESSURE INJECTION SYSTEM

FLOW DIAGRAM



NOTES:

1. ATMOSPHERIC PRESSURE OF 14.7 PSIA WAS USED IN CALCULATIONS
2. WATER FLOWS ARE GIVEN IN GPM, STEAM FLOWS IN 1000 LB/HR.
3. THE MAXIMUM POOL WATER TEMPERATURE FOR CONTINUOUS SYSTEM OPERATION [illegible] EXCEED 140 F. HOWEVER, DUE TO POTENTIAL SHORT TERM OPERATION AT HIG[illegible] TEMPERATURE, PIPING EXPANSION SHALL BE BASED ON 170 F.
4. THE FOLLOWING ADDITIONAL ACCIDENT OPERATING MODES ARE POSSIBLE:

   SUCTION FROM CONDENSATE STORAGE, SUPPRESSION POOL AT LOW PRESSU[illegible] REACTOR AT HIGH OR LOW PRESSURE.

   SUCTION FROM SUPPRESSION POOL, SUPPRESSION POOL AT HIGH PRESSU[illegible] REACTOR AT HIGH OR LOW PRESSURE.

   THESE POTENTIAL OPERATING MODES DO NOT CONTROL PIPE OR VALVE SIZING O[illegible] SPECIFICATION, AND NO DATA IS SHOWN.
5.
6. THE CONTROLLING MODES FOR LINE SIZING AND ARRANGEMENT ARE:

   SUCTION FROM COND. STORAGE ---------------- MODE A
   SUCTION FROM SUPPRESSION POOL---------------- MODE C
   PUMP DISCHARGE ---------------- MODE C
   STEAM SUPPLY ---------------- MODE B
   TURBINE EXHAUST ---------------- MODE A - D
   TEST LINE ---------------- MODE E
   COOLING SYSTEM ---------------- MODE A & E
7.
8. PUMP MINIMUM FLOW REQUIREMENT MAY OCCUR DURING ANY OPERATING MODE. FLOW REQUIREMENT IS 500 GPM, MINIMUM.

PHILADEL[illegible]
PEACH BOTT[illegible]

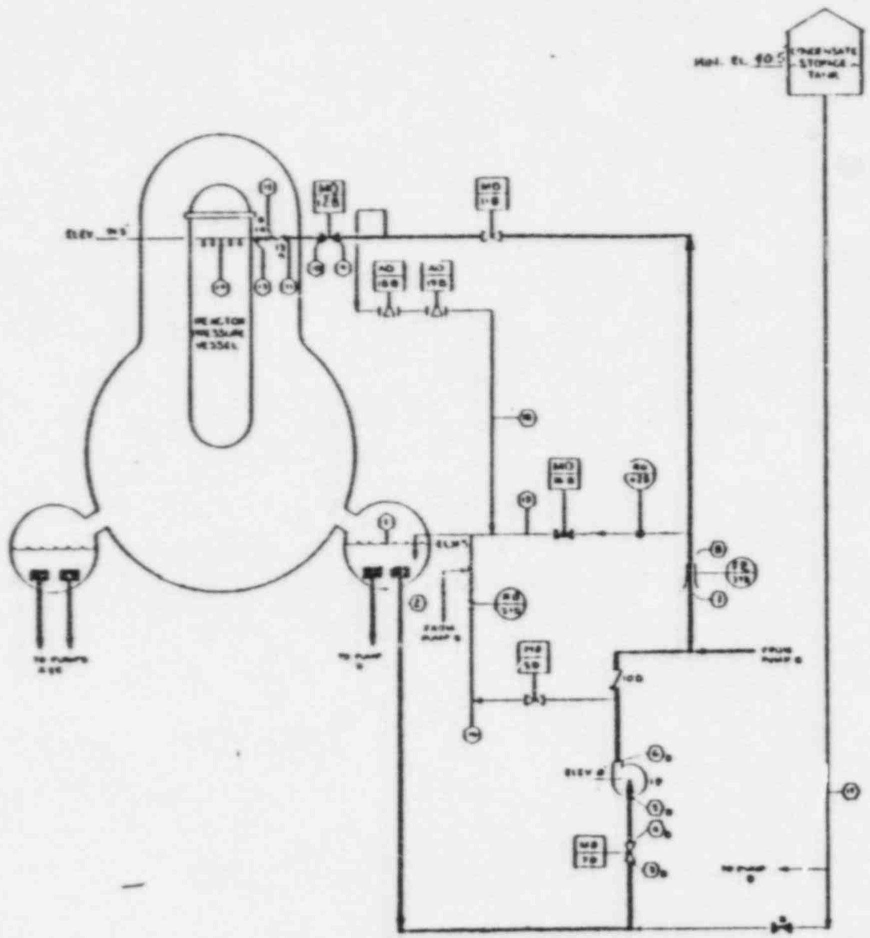
FINAL S[illegible]

HIGH PRESS[illegible]
INJECTION
SYSTEM,

B6.2

CORE SPRAY SYSTEM

FLOW DIAGRAM



CONDITION I NORMAL SYSTEM TEST (2 PUMP OPERATION)

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
FLOW GPM	N/A	[illegible]	←		→	[illegible]	6250	6250	0	0	0	0	0	0	6250	0	0	0
PRESS PSIA	14.7																	
TEMP °F	90	90	90	90	90	90	90	90	AMB	AMB	AMB	AMB	AMB	AMB	90	AMB	AMB	AMB
NOTE																		
ΔP PSI	← [illegible] →		← [illegible] →		← [illegible] →		← [illegible] →											

NOTE: ONLY ONE CORE S

THE SECOND LOOP

CONDITION II TEST USING CONDENSATE STORAGE TANK (2 PUMP OPERATION)

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
FLOW GPM	N/A	0	[illegible]	←	→	[illegible]	[illegible]	←					→	[illegible]	0	0	[illegible]	0
PRESS PSIA	14.7													14.7				
TEMP °F															AMB	AMB		
NOTE					7													
ΔP PSI					← 152 →		← [illegible] →		← [illegible] →		← [illegible] →		← [illegible] →					

← ΔP-27.0 →

CONDITION III PUMP OPERATING ON BYPASS (ISOLATION VALVES CLOSED)

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
FLOW GPM	N/A	[illegible]	←		→	[illegible]	0	0	0	0	0	0	0	0	0	[illegible]	0	0
PRESS PSIA	14.7																	
TEMP °F	140	140	140	140	140	140	AMB	←						→	AMB	140	AMB	AMB
NOTE																		
ΔP					← 342 →													

CONDITION IV CORE SPRAY INJECTING AT REACTOR PRESS (2 PUMP OPERATION)

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
FLOW GPM	N/A	[illegible]	←		→	[illegible]	[illegible]	←					→	[illegible]	0	0	0	0
PRESS PSIA	[illegible]												[illegible]	[illegible]				
TEMP °F	210	←											→	[illegible]	AMB	AMB	AMB	AMB
NOTE																		
ΔP PSI	← 0.5 →		← 0.1 →		← 248 →		← 13 →		← 85 →		← [illegible] →		← [illegible] →					

← ΔP-0.5 → ← ΔP-[illegible] →

VALVE POSITIONS

CONDITION	VALVE NO.								
	5	7	8	11	12	14	18	19	26
I	C	O	C	O	C	O	C	C	P
II	C	C	O	O	O	O	C	C	C
III	O	O	C	O	C	O	C	C	C
IV	C	O	C	O	O	O	C	C	C

O - FULL OPEN
C - FULL CLOSED
P - PARTIALLY OPEN

PHILADELPHIA ELECTRIC C

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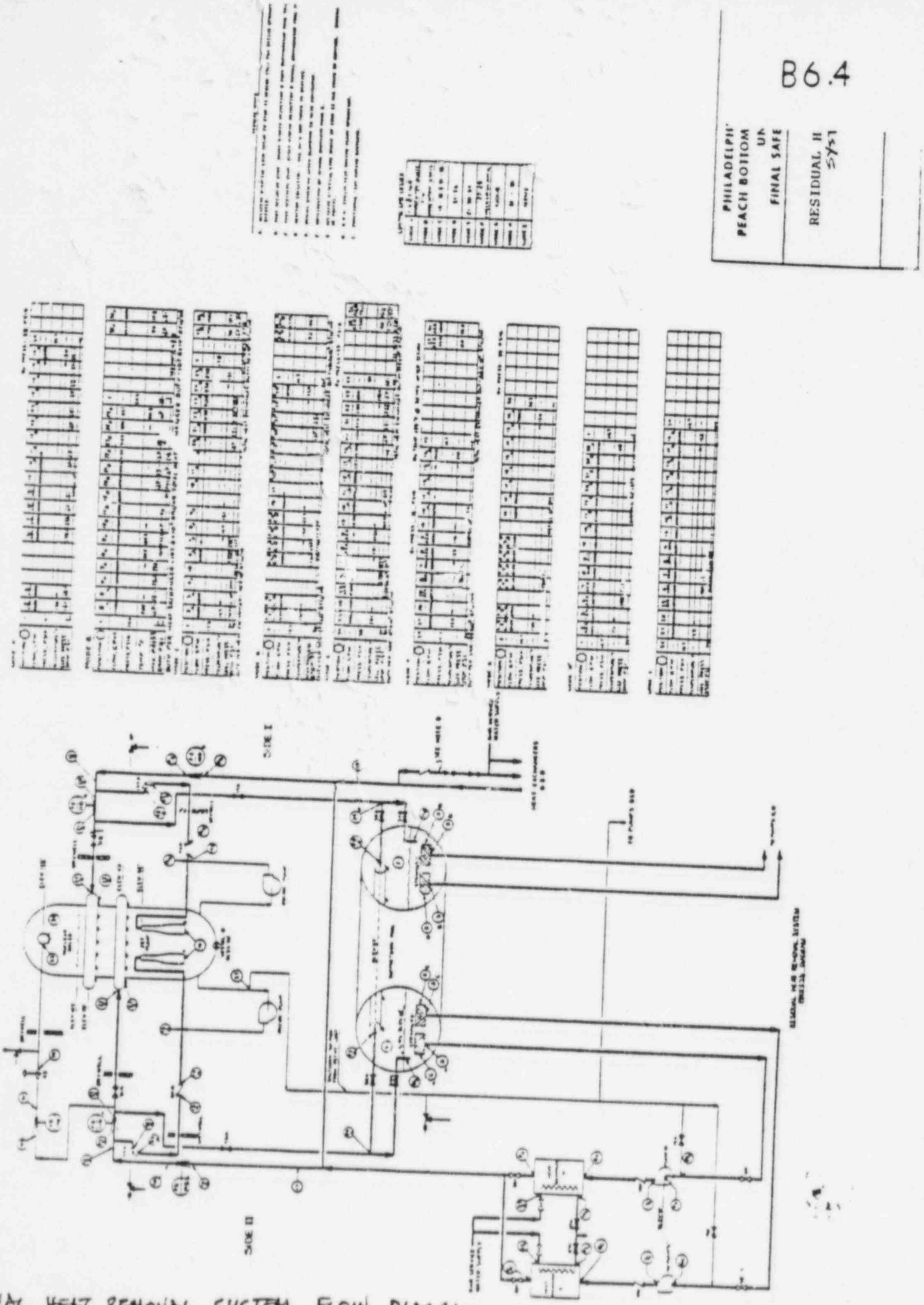
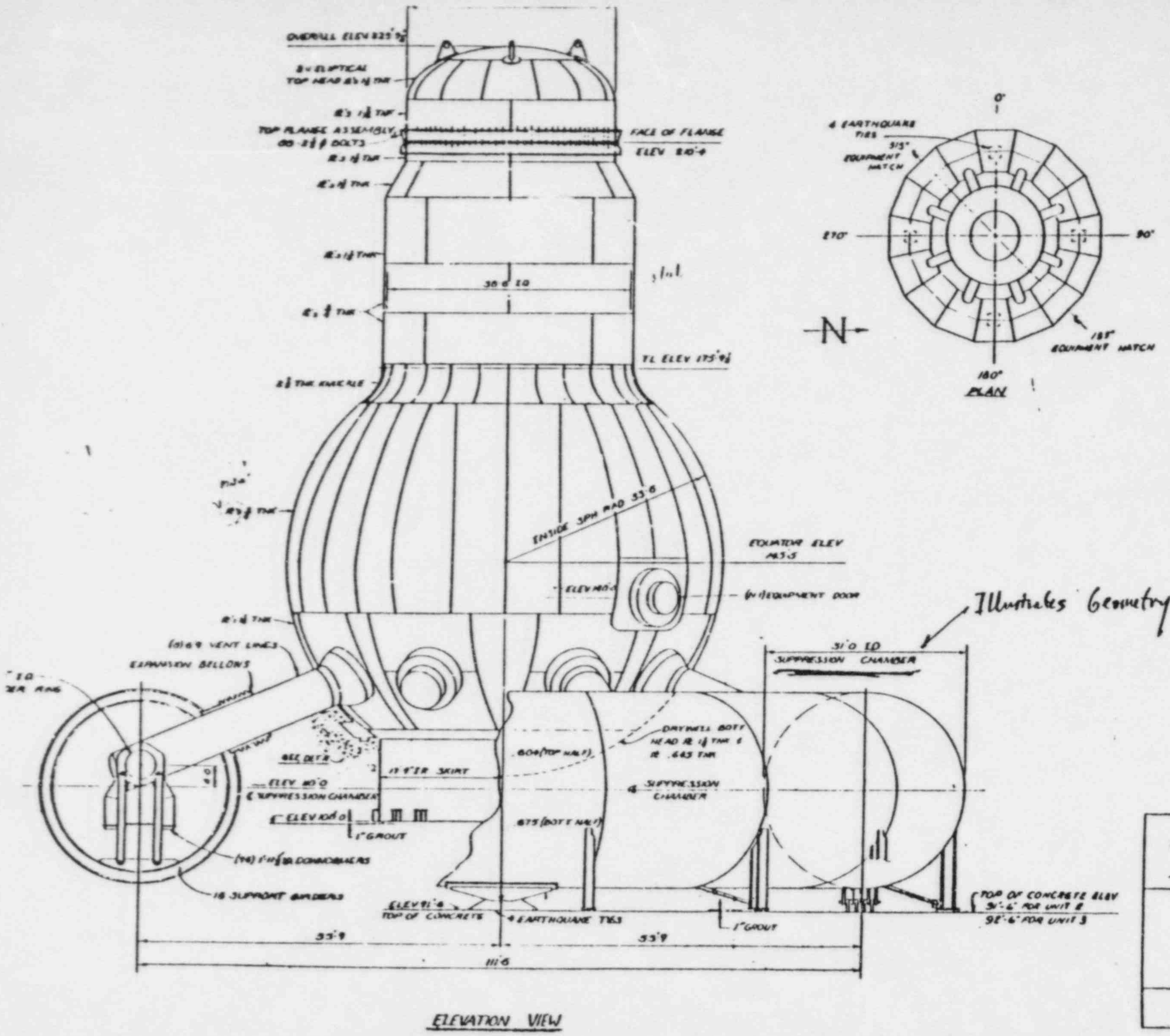
B6.4



RESIDUAL HEAT REMOVAL SYSTEM FLOW DIAGRAM



Illustrates Geometry

ELEVATION VIEW

PHILADELPHIA ELECTRIC
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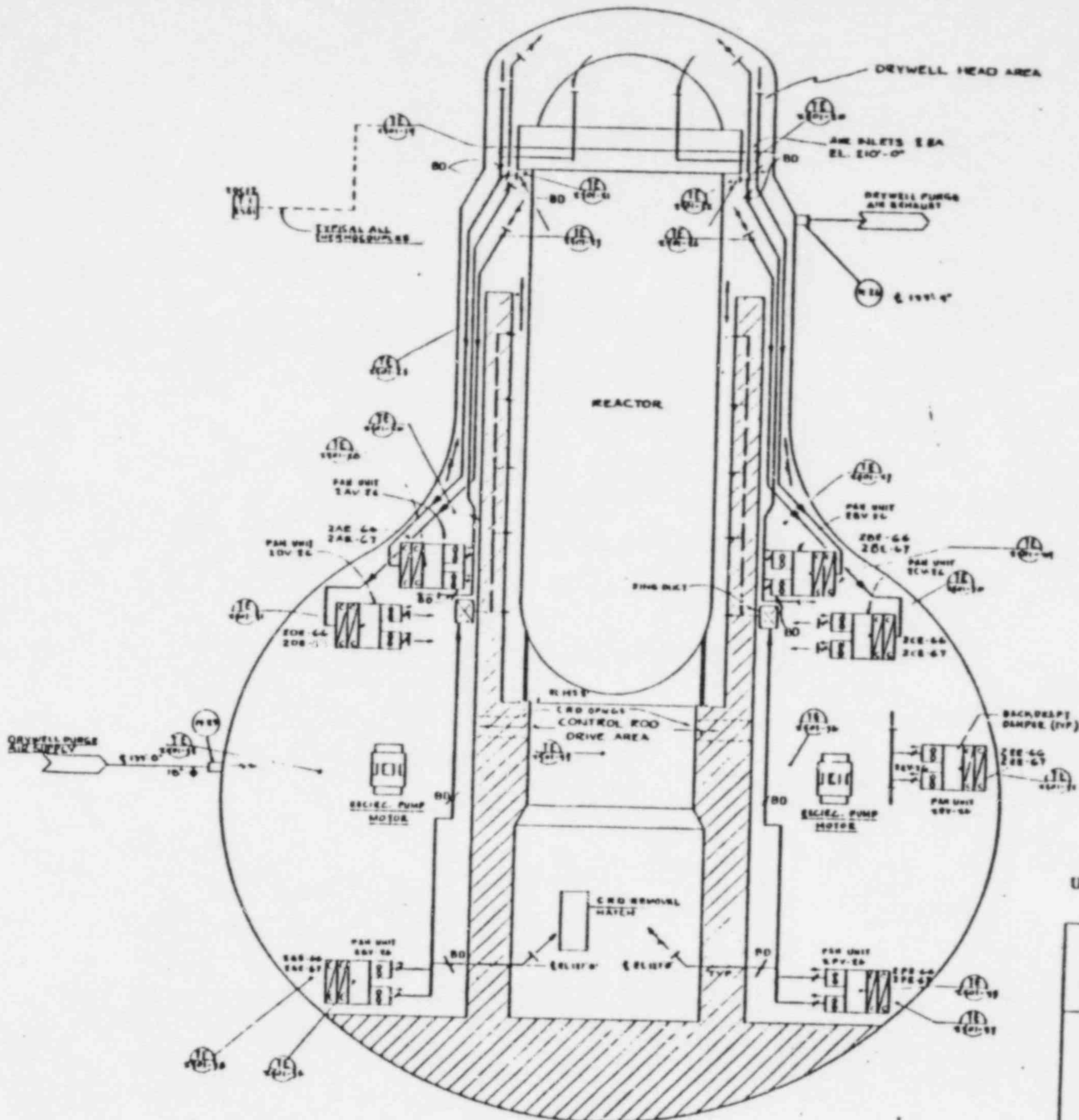
B7.1

PRI

FIGU



UNIT 2 IS SHOWN. UNIT

PHILAD... PEACH BC
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B 7.2

B8.1

COOLANT INVENTORIES AND SELECT COMPONENT VOLUMES FOR THE PEACH BOTTOM 1 BWR

	VOLUME (ft^3)	MASS (lb_m)
REACTOR VESSEL *		
Subcooled Liquid	7 847	131 364
Saturated Liquid	4 005	184 412
Steam	8 814	20 528
PIPING		
Recirculation System	1 228	56 534
Feedwater System	815.9	37 571
Steam	3 126	7 280
PRESSURE SUPPRESSION CHAMBER (TORUS)		
Free Volume	119 000	— NA —
Water	136 000	7 234 642
DRYWELL Free Volume	150 000	— NA —

* FOR FURTHER BREAKDOWN OF WATER MASSES + VOLUMES SEE FIG. B0.1, B0.2

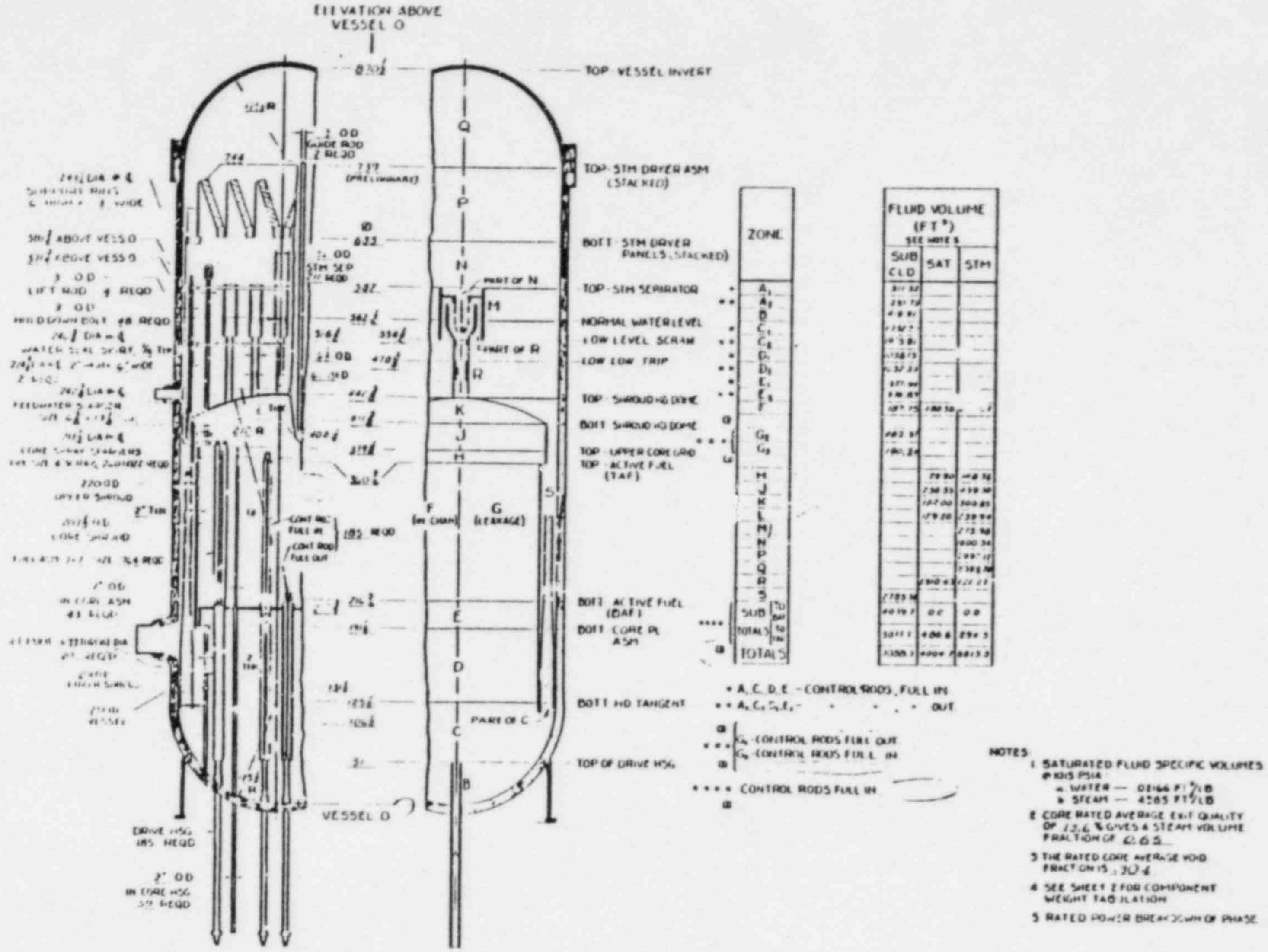
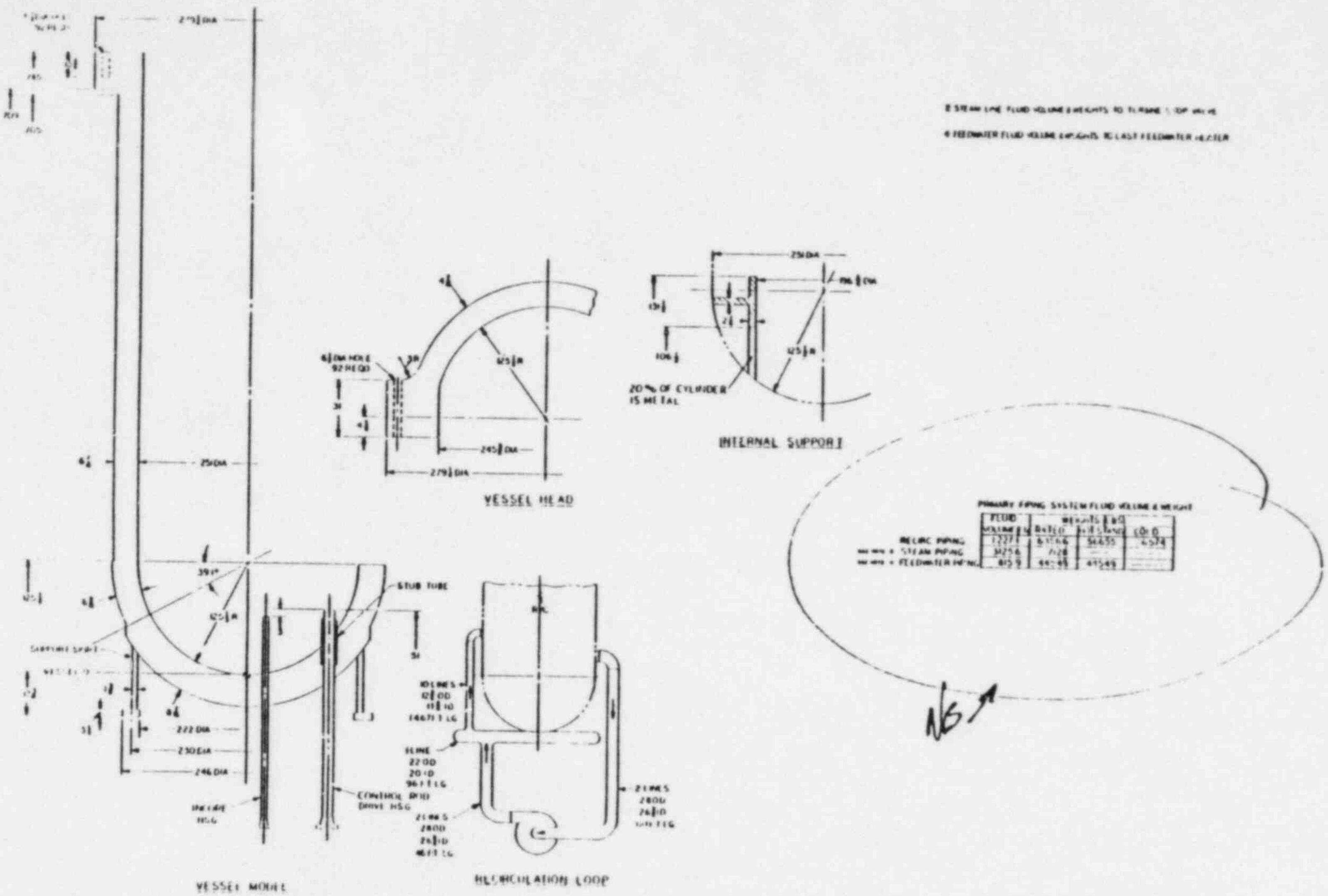


Figure 39. Reactor Primary System Weights and Volumes



PRIMARY PIPING SYSTEM FLUID VOLUME & WEIGHT

	FLUID VOLUME	WEIGHTS		
		RATED	HI STANDBY	COLD
RECIRC PIPING	12271	61166	56655	4574
STEAM PIPING	31256	7128		
FEEDWATER PIPING	8159	44245	41549	

Figure 40. Reactor Primary System Weights and Volumes (Continued)

B 8.3

May 26, 1984

Dear Mel,

Enclosed you will find a couple of illustrations which are meant to serve as a guide for my requests. This is more or less in line with earlier requests, but perhaps I can be a bit more specific this time.

I wish to have in schematic form a diagram of all the relevant control volumes corresponding to the primary system, secondary (steam generator) system and containment volume for both PWRs and BWRs. I should like to have these in a consistent manner. I wish to know the volume of each, in consistent units, and I wish to know how much water is stored in each, appropriate control volume, as the case may be. I prefer not to have water content expressed in lbs, ft^3 and gals, for different units, but rather in the same consistent units. I should like to know the volume of water in cold/hot leg pipes, normally, as well as in all ESS units, auxiliaries, etc. I also wish to know what the normal path (design basis) of water flow is.

[illegible] units

1 [illegible] ?

2 [illegible] ?

I trust that a few schematic diagrams with flow charts would serve my purpose.

Next, I wish to have the pressure vessel, core, etc. system drawn for both PWR and BWR with all appropriate dimensions included, as suggested by my enclosure. I wish to know relative volumes occupied by fuel vs cooling channels, dimensions of downcomer, lower plenum, core, upper plenum, structurals, and so forth. I am, as you will note, interested in having a precise, quantitative picture of those design parameters that are intimately related to the thermal-hydraulics analysis of the problem.

The sooner I can have these, of course, the better. With many thanks for your help,

Sincerely,
Peter
P.L. Auer

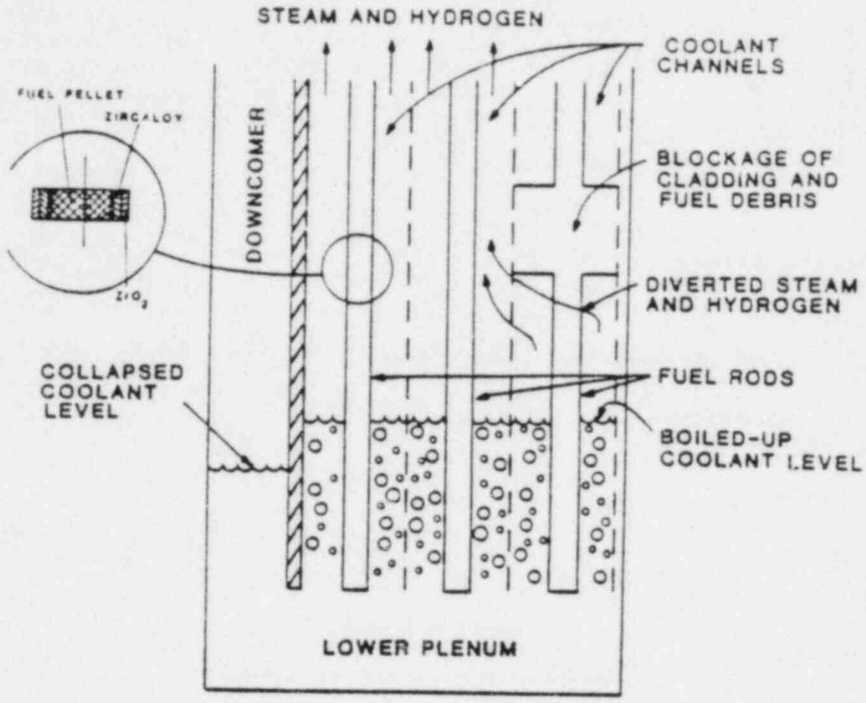
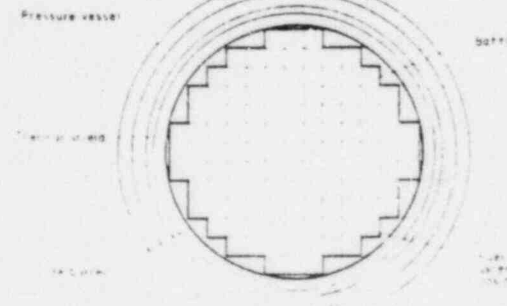


Fig. 1 Schematic description of a PWR core during uncovery.

Please show all appropriate dimensions for constituents/(see 1 and 2) in lower plenum, core and upper plenum.

The baffle is supported by the core barrel. The thermal shield is a thick sheet of steel, designed to reduce the neutron and γ ray dose received by the wall of the pressure vessel. The cooling water flows down on both sides of the thermal shield before entering the core.



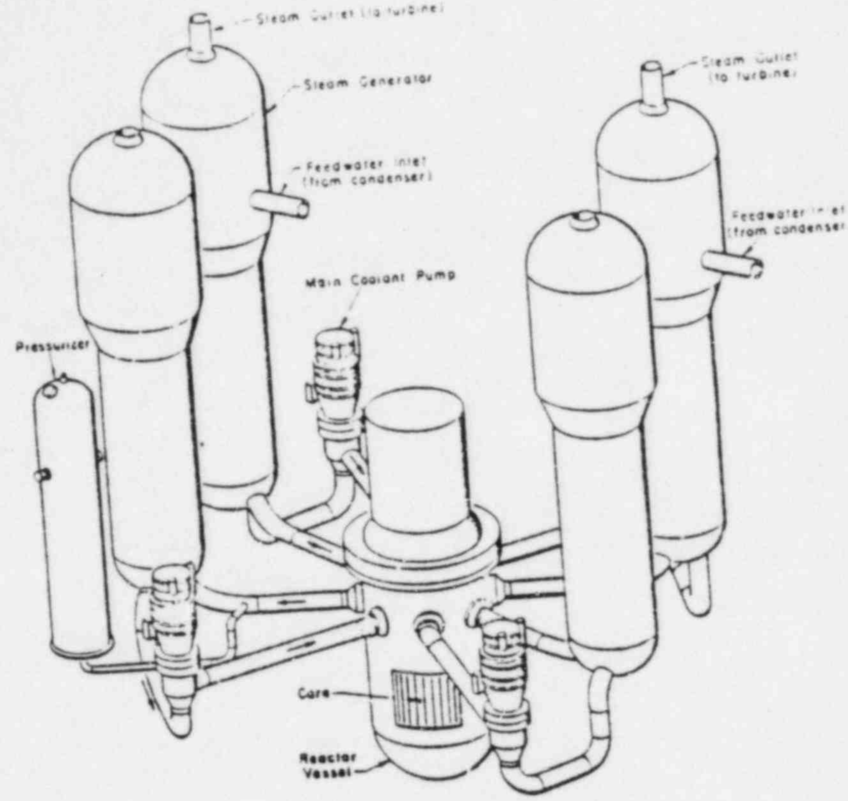
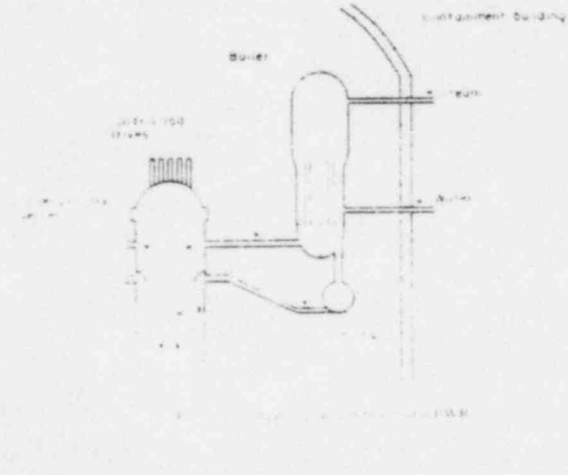


FIGURE 3-3 Schematic of Reactor Coolant System for PWR

[illegible] by schematic [illegible] show [illegible] (RCS, [illegible], etc) containing water, [illegible] of water and [illegible] of water [illegible].
[illegible] show [illegible] F.P. [illegible] to containment.

[illegible] generator enters a centrifugal pump and is returned to the core. The number of coolant loops is variable, but typically for a large reactor there would be four steam generators, each with its own primary circuit pump.



Notes About Auer's Request for Technical Data

I suspect that Dr. Auer would like to estimate the reactor thermal hydraulic conditions to see how they compare to the BMI-2104 results and with his intuition. He seems to want data presented so that he can concentrate on his thoughts rather than on the estimates themselves.

Some, but not all, of the data is in the BMI-2104 reports. However, other data are needed for simple hand-done estimates. Given what is in the BMI-2104 reports and Dr. Auer's letter requesting thermal hydraulic data, I rephrased some of his statements.

Item 1: I believe that the term "control volume" is being used to encompass several notions; one, as a qualitative way of dividing the hardware areas for calculating; two, as a quantitative way of describing the hardware areas. Table 1 illustrates some of the data that is necessary.

Item 2: Both pictorial diagram and simplified schematic diagram would be useful. Describe the complete flow (including return lines) through the hardware during normal operation and emergency operations.

- Reactor coolant system
- Emergency core coolant
- Pressurizer
- PORV
- Surge Tank
- Suppression pool
- Auxiliary storage tanks
- Reactor cavity

Include the PORV setting(s)
the initiation signal for the accumulators

Item 3: Describe the dimensions of the vessel internal including the following information:

- Internal volume
- Downcomer volume
- Lower plenum volume
- Channel volume
- Core volume
- Upper plenum volume
- Pressure
- Temperature of the fuel at the pellet core
- Temperature of the fuel at the pellet periphery
- Temperature of the coolant
- Void fraction

Other data needed to precisely but simply describe the vessel internals.

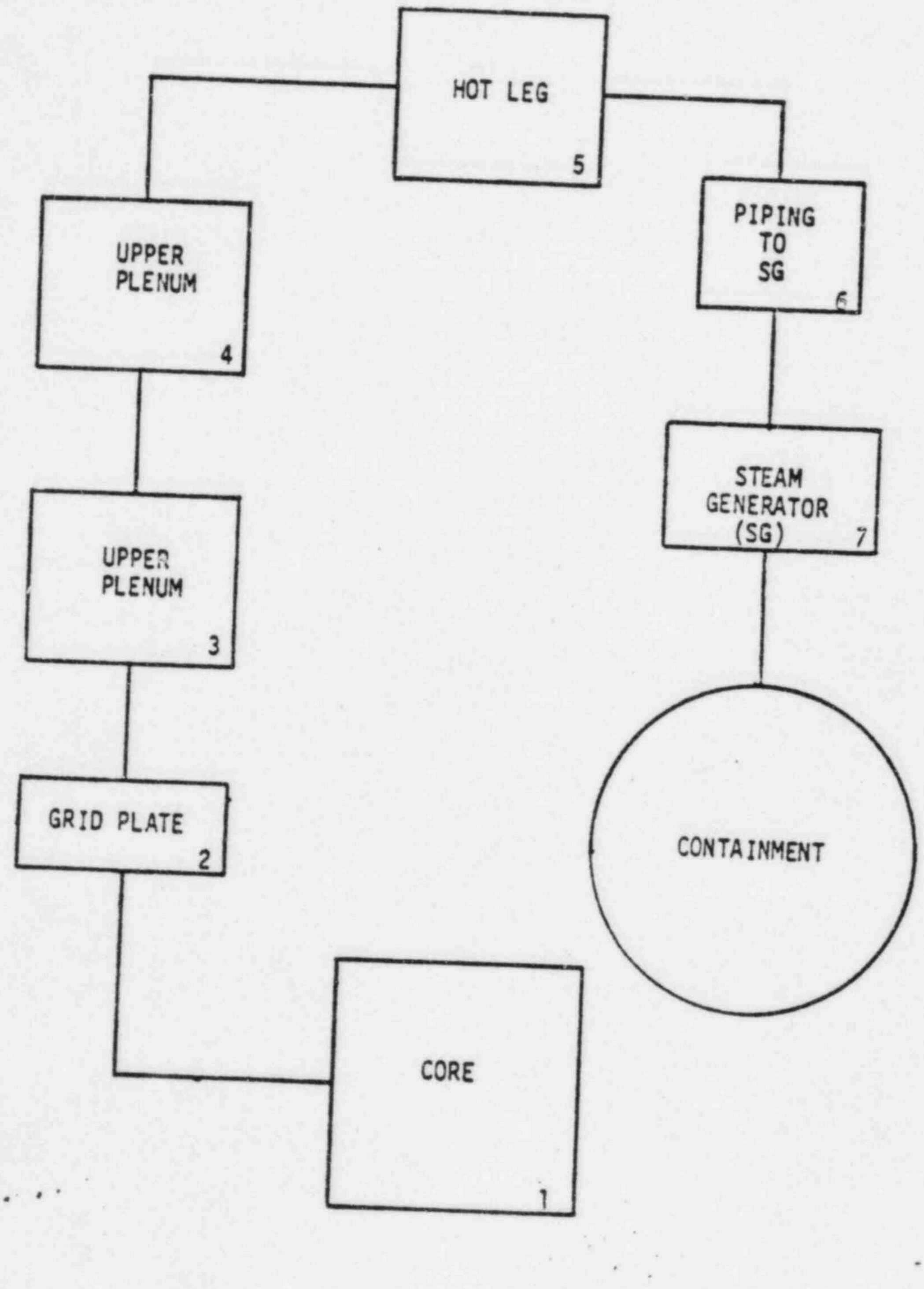
In his letter, Dr. Auer asks for precise data in a consistant manner. By "precise," I assume he means data at an appropriate level of detail for hand-done calculating; too much data or too few data is unacceptable. This will require anticipating his calculations without judging the usefulness of them. By "consistent," I assume that he has two notions; a given datum should appear where ever appropriate in describing the systems; a single set of units should be used (time, ^{o}C, cubic meters). Consistent units should be used not only for a given datum such as temperature but also to make datum compatible with one another.

Attached are the following:

(1) Auer's letter.
(2) Simplified schematics of the type that may be useful to Auer.

Table 1: A suggested format for consistantly presenting numerical data. Pictorial diagrams may also be useful.

	Volume	Water Inventory	Volumetric Velocity	Linear Velocity
Primary System				
Steam Generator				
Secondary System				
Accumulator discharge line			N/A	N/A
RWST discharge line			N/A	N/A



This shows where the control volumes are located, but does not have plant dimensions.

FIGURE 6.16. SCHEMATIC OF CONTROL VOLUMES FOR THE SURRY S_2D SEQUENCE (CASE 1)

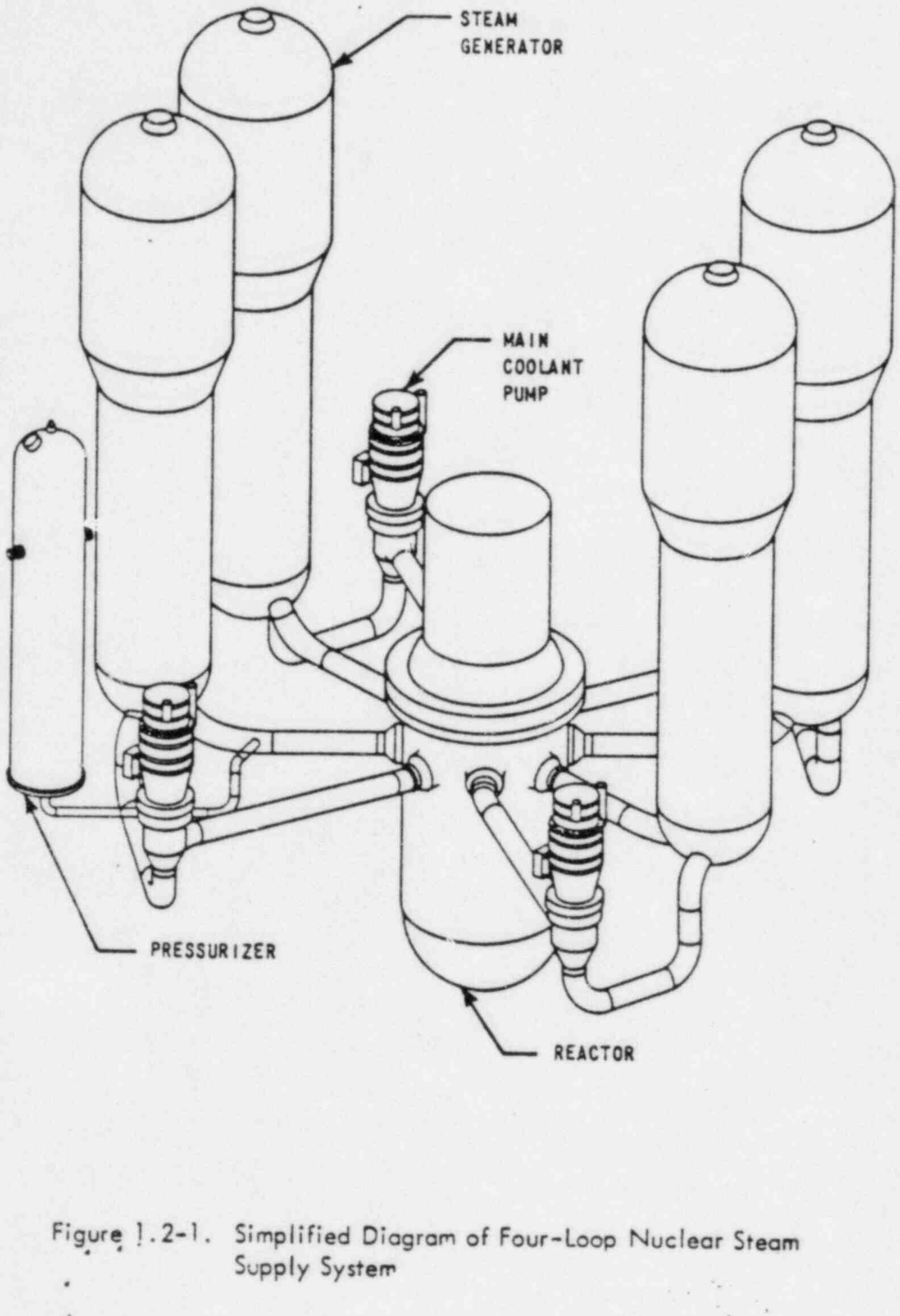
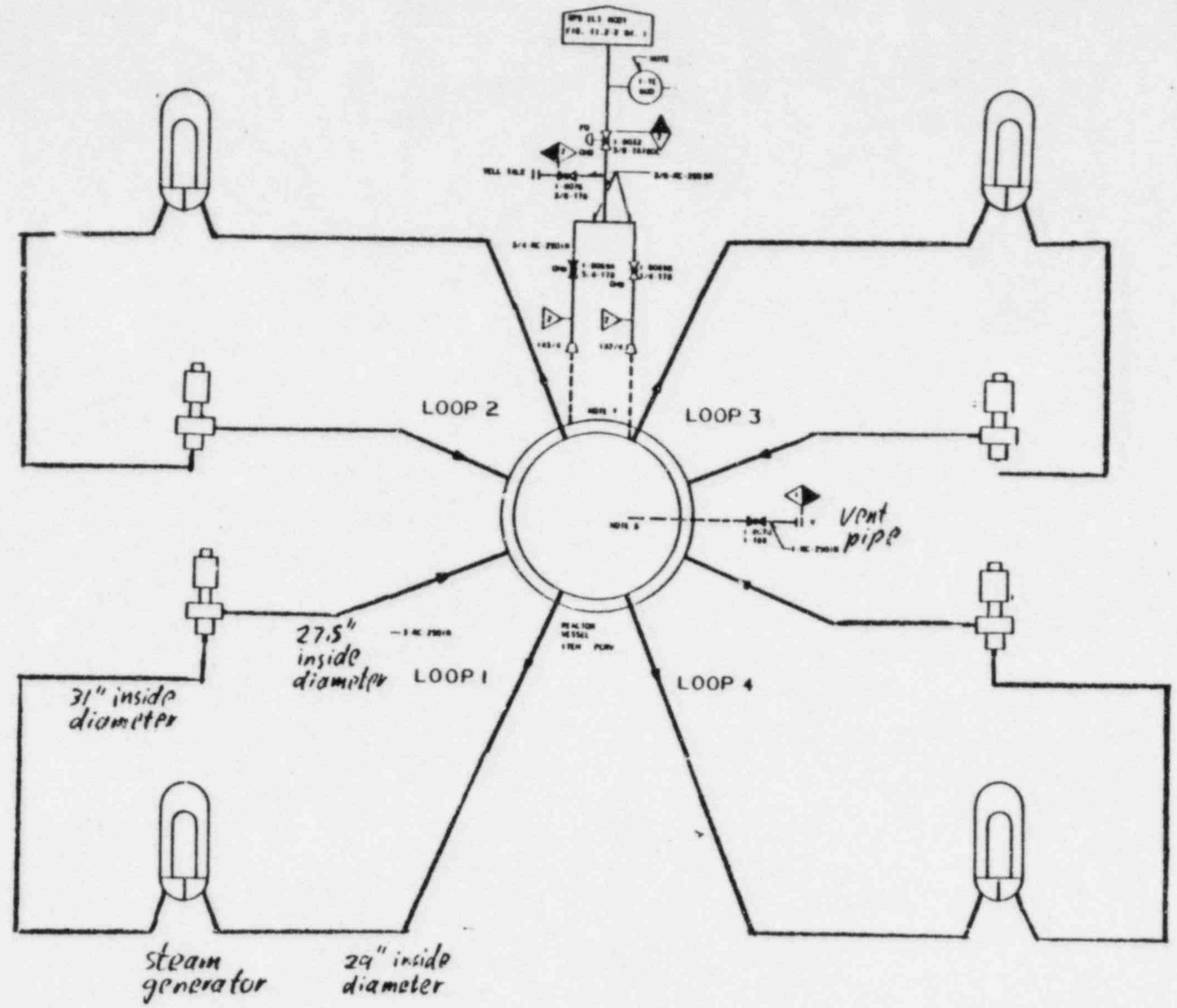
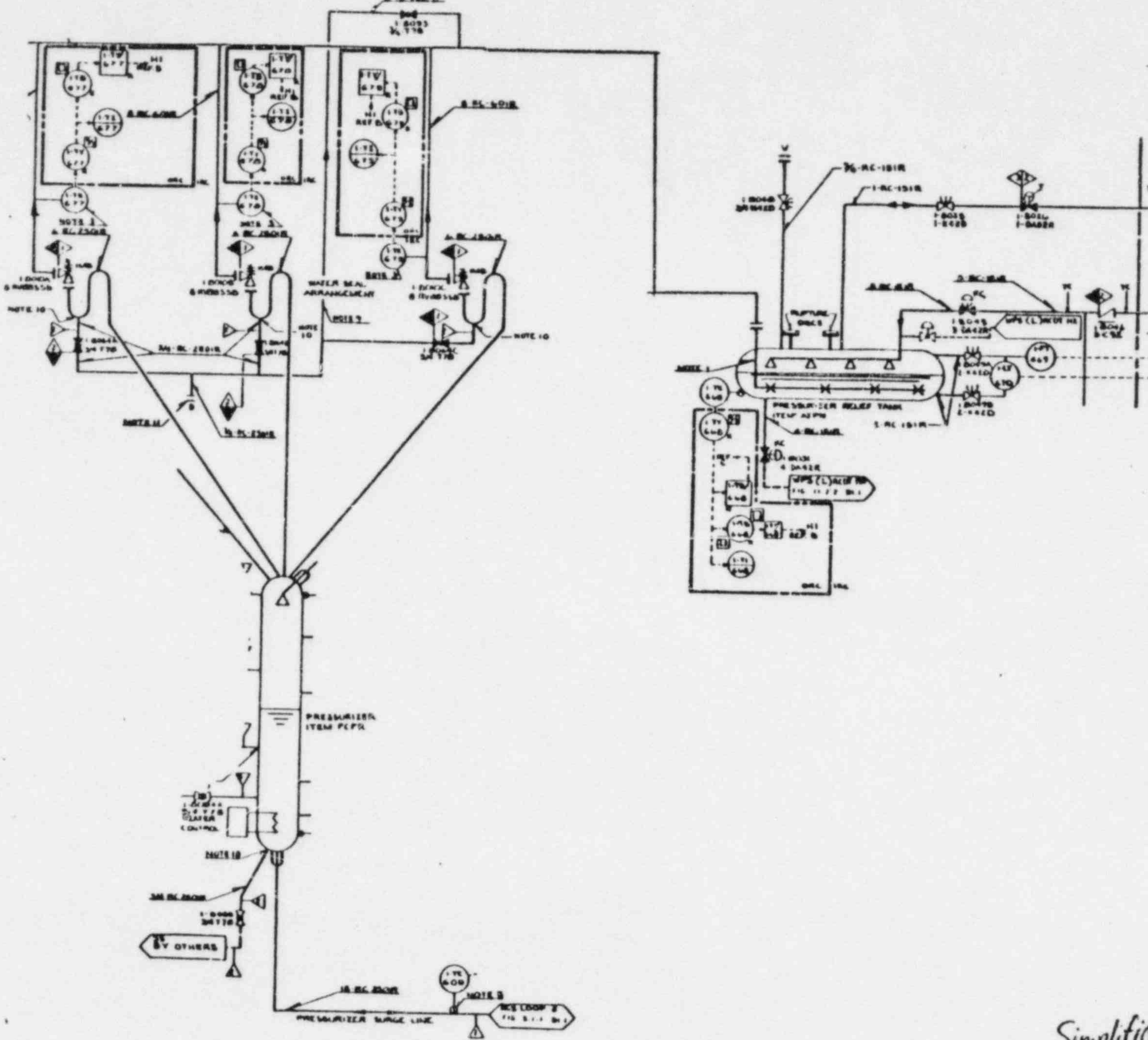


Figure 1.2-1. Simplified Diagram of Four-Loop Nuclear Steam Supply System



Simplified Schematic
-Flow



Simplified Schematic
-Flow

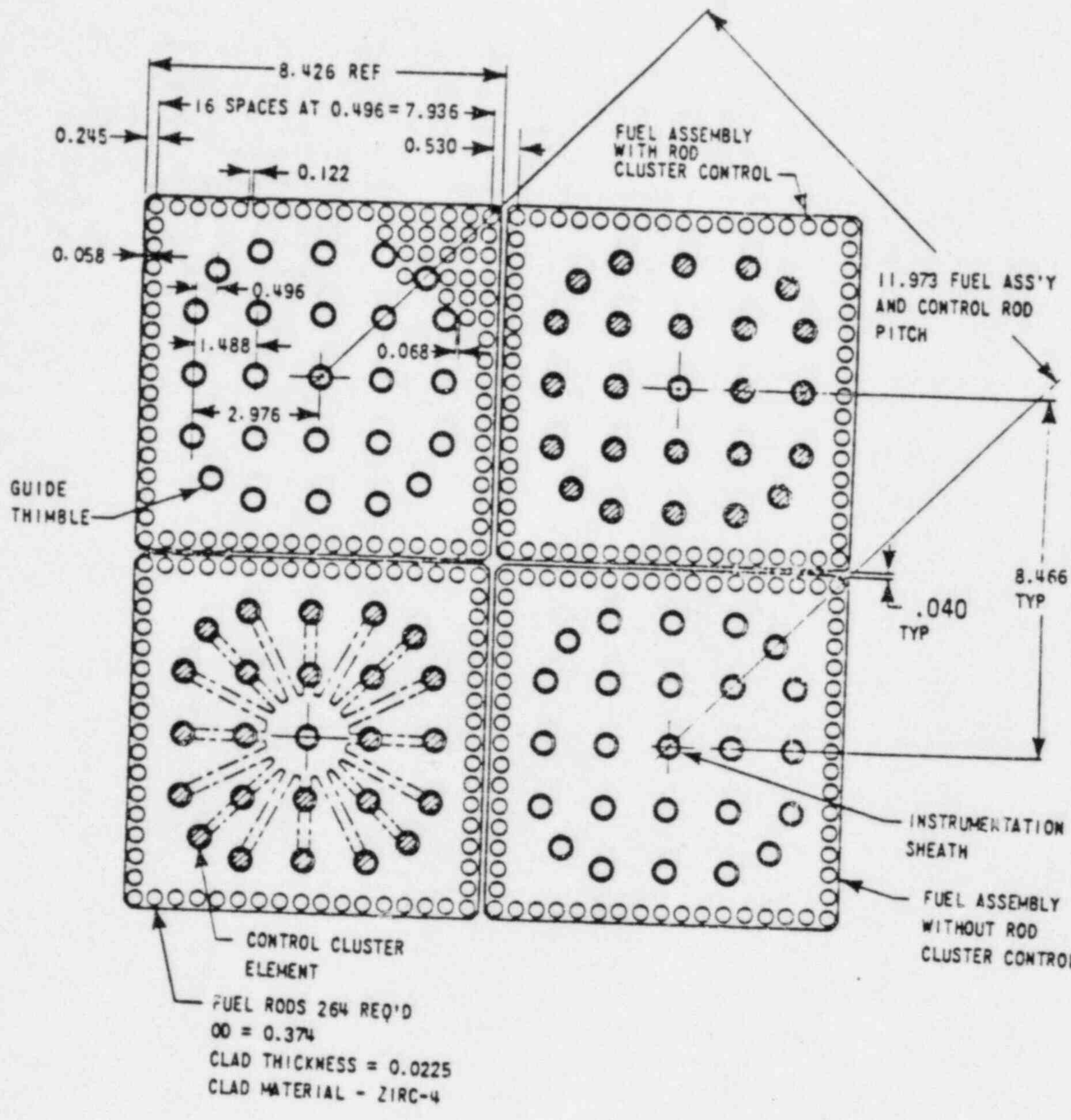


Figure 4.2-1. Fuel Assembly Cross Section 17 x 17

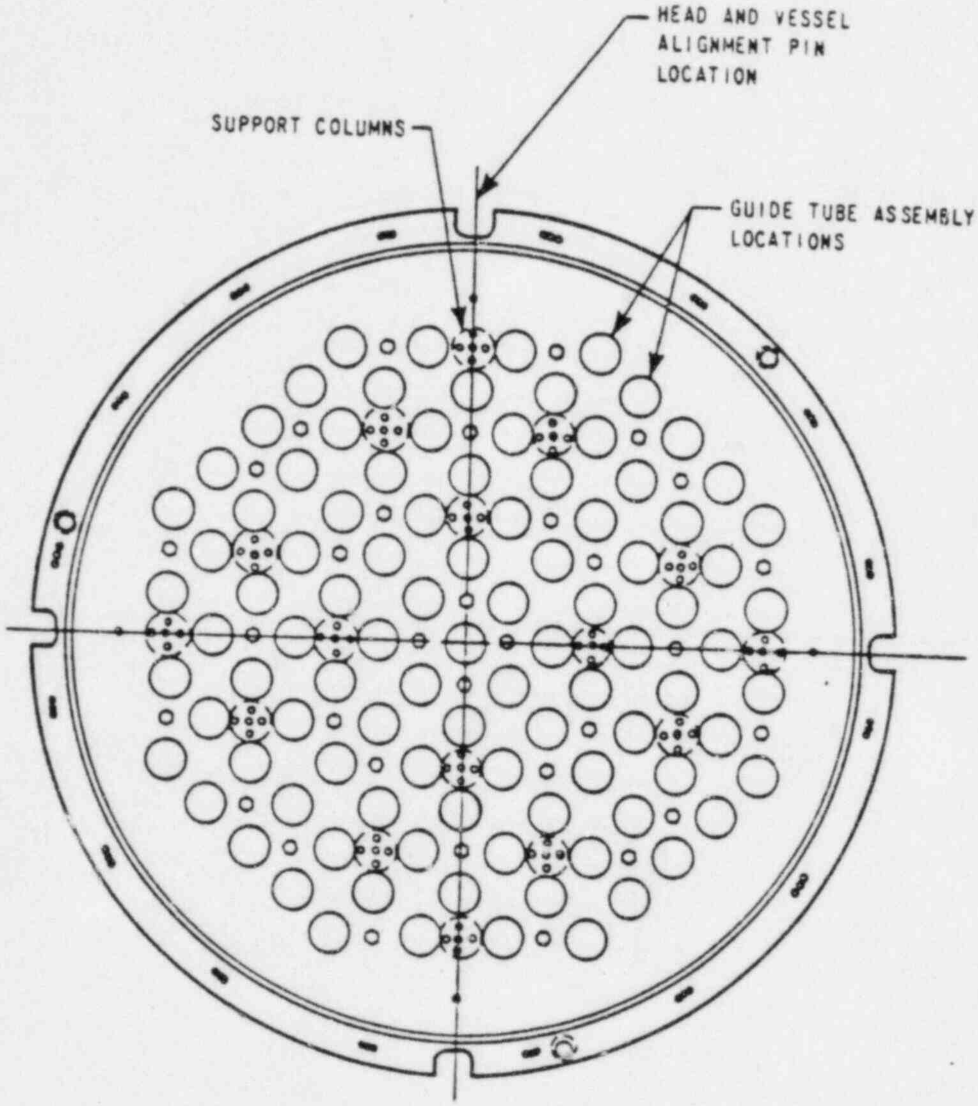


Figure 4.2-8 Plan View of Upper Core Support Structure

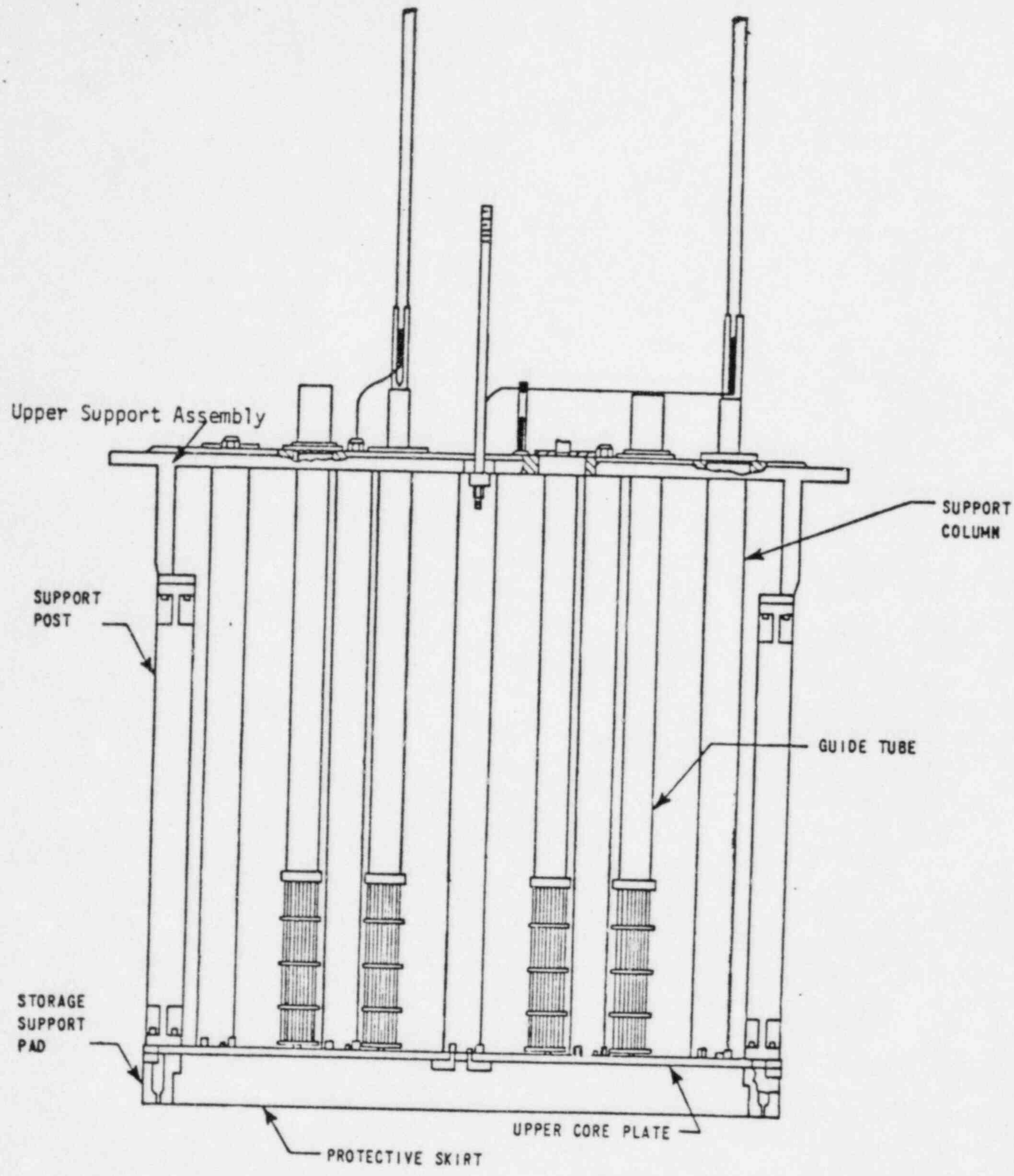


Figure 4.2-7 Upper Core Support Structure

TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Nominal operating pressure, psig	2235
Total system volume including pressurizer and surge line, cu. ft.	13,828
System liquid volume, including pressurizer water at maximum guaranteed power, cu. ft.	12,963
Pressurizer spray rate, max. gpm	1050
Pressurizer heater capacity, kw	2100
Pressurizer relief tank volume, cu. ft.	2100

System Thermal and Hydraulic Data
(Based on Thermal Design Flow)

	4 Pumps Running	3 Pumps Running
NSSS Power MWT	3817	2672
Reactor Power MWT	3800	2660
Thermal Design Flows, gpm		
Active Loop	97,600	103,900
Idle Loop	--	36,000
Reactor	390,400	275,700
Total Reactor Flow, 10^6 lb/hr	144.74	102.80
Temperatures, °F		
Reactor Vessel Outlet	623.8	613.1
Reactor Vessel Inlet	559.8	555.7
Steam Generator Outlet	559.5	555.4
Steam Generator Steam	556.3	554.4
Feedwater	440.0	401.0
Steam Pressure, psia	1100	1083
Total Steam Flow, 10^6 lb/hr	16.96	11.24

TABLE 5.1-1 (Continued)

SYSTEM DESIGN AND OPERATING PARAMETERS

System Flow Summary

Flows, GPM	Thermal Design	Best Estimate	Mechanical Design
4 pumps running, each loop	97,600	99,700	105,200
3 pumps running			
Active loop	103,900	107,900	113,400
Idle Loop	36,000	33,500	35,200
Reactor	275,700	290,200	305,000

System Pressure Drops
(Based on 4-Loop best estimate flow)

Reactor vessel ΔP, psi	54.5
Steam generator ΔP, psi	43.4
Hot leg piping ΔP, psi	1.9
Pump suction piping ΔP, psi	3.5
Cold leg piping ΔP, psi	1.9
Pump head, feet	328

Coolant Flow

Total Thermal Flow Rate, lb/hr	144.7×10^6	142.2×10^6
Effective Flow Rate for Heat Transfer, lb/hr	138.2×10^6	135.8×10^6
Effective Flow Area for Heat Transfer, ft^2	51.1	51.1
Average Velocity Along Fuel Rods, ft/sec	17.2	16.8
Average Mass Velocity, lb/hr-ft^2	2.71×10^6	2.66×10^6

Coolant Temperature

Nominal Inlet, °F	559.8	557.3
Average Rise in Vessel, °F	64.2	59.7
Average Rise in Core, °F	66.8	62.3

Thermal and Hydraulic Design Parameters	RESAR-3 12' (Ammend 5)[b]
Average in Core, °F	590.0
Average in Vessel, °F	587.1
Heat Transfer	
Active Heat Transfer, Surface Area, ft^2	59,900 (59,700)[a]
Average Heat Flux, $Btu/hr-ft^2$	189,400 (189,800)[a]
Maximum Heat Flux, for normal operation, $Btu/hr-ft^2$	454,600 (474,500)[a]
Average Thermal Output, kw/ft	5.43 (5.45)[a]
Maximum Thermal Output, for normal operation, kw/ft	13.0 (13.6)[a,d]
Maximum Thermal Output at Maximum Over-power Trip Point (118% power), kw/ft	(15.2)[a,e]
Fuel Central Temperature (BOL)	
Peak at 100% Power, F	3250 (3500)[a,d]
Peak at Maximum Thermal Output for Maximum Overpower Trip Point, F	4400 (3800)[a,e]
Pressure Drop[c]	
Across Core, psi	28.0±5.6
Across Vessel, including nozzles, psi	47.6±7.1

[a] Using new densification model of Ref. 80

[b] Without densification effects

[c] Based on best estimate reactor flow rate as discussed in Section 5.1.

[d] This limit is associated with the value of $F_Q = 2.50$.

[e] See Section 4.3.2.2.6

TABLE 4.4-2

THERMAL-HYDRAULIC DESIGN PARAMETERS FOR ONE OF FOUR COOLANT LOOPS OUT OF SERVICE

Total Core Heat Output, MWt	2660
Total Core Heat Output, 10^6 Btu/hr	9079
Heat Generated in Fuel, %	97.4
Nominal System Pressure, psia	2250
Coolant Flow	
Effective Thermal Flow Rate for Heat Transfer, 10^6 lbs/hr	102.8
Effective Flow Area for Heat Transfer, ft^2	51.1
Average Velocity along Fuel Rods, ft/sec	12.2
Average Mass Velocity, 10^6 lb/hr-ft^2	1.92
Coolant Temperature, °F	
Design Nominal Inlet	555.7
Average Rise in Core	66.8
Average in Core	590.9
Heat Transfer	
Active Heat Transfer Surface Area, ft^2	68,000
Average Heat Flux, Btu/hr-ft^2	129,900
Minimum DNB Ratio at Nominal Conditions	>1.70
Minimum DNB Ratio for Design and Anticipated Transients	$\geq$1.30

REACTOR CORE DESCRIPTION

Fuel Assemblies

Number	193
Rod Array	17 x 17
Rods per Assembly	264
Rod Pitch, in.	0.496
Overall Transverse Dimensions, in.	8.426 x 8.426
Fuel Weight (as UO_2), lbs.	253,675
Zircaloy Weight, lbs.	53,520
Number of Grids per Assembly	8 - R Type
Weight of Grids (Effective in Core) lbs	2299
Number of Guide Thimbles per Assembly	24
Composition of Guide Thimbles	Zircaloy 4
Diameter of Guide Thimbles (upper part), in.	0.450 I.D. x 0.482 O.D.
Diameter of Guide Thimbles (lower part), in.	0.397 I.D. x 0.4290 O.D.
Diameter of Instrument Guide Thimbles, in.	0.450 I.D. x 0.4820 O.D

Fuel Rods

Number	50,952
Outside Diameter, in.	0.374
Diameter Gap, in.	0.0065
Clad Thickness, In.	0.0225
Clad Material	Zircaloy-4

4.4.3.1 Core Hydraulics

4.4.3.1.1 Flow Paths Considered in Core Pressure Drop and Thermal Design

The following flow paths or core bypass flow are considered:

1. Flow through the spray nozzles into the upper head for head cooling purposes.

2. Flow entering into the RCC guide thimbles to cool the control rods.

3. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel.

4. Flow entering into the core from the baffle-barrel region through the gaps between the baffle plates.

The above contributions are evaluated to confirm that the design basis value of ≤ 4.5% core bypass flow is met. This design bypass value is also used in the evaluation of the core pressure drops quoted in Table 4.4-1, and the determination of reactor flow rates in Section 5.1.

Flow model test results for the flow path through the reactor are discussed in Section 4.4.2.8.2.

4.4.3.1.2 Inlet Flow Distributions

Data has been considered from several 1/7 scale hydraulic reactor model tests[47, 48,55] in arriving at the core inlet flow maldistribution criteria to be used in the THINC analyses (See Section 4.4.3.4.1). THINC I analyses made using this data have indicated that a conservative design basis is to consider a 5 percent reduction in the flow to the hot assembly.[56] The same design basis of 5% reduction to the hot assembly inlet is used in THINC IV analyses.

The experimental error estimated in the inlet velocity distribution has been considered as outlined in Reference [43] where the sensitivity of changes in inlet velocity distributions to hot channel thermal performance is shown to be small. Studies[43] made with the improved THINC model (THINC-IV) show that it is adequate to use the 5% reduction in inlet flow to the hot assembly for a loop out of service based on the experimental data in Reference [47] and [48].

The effect of the total flow rate on the inlet velocity distribution was studied in the experiments of Reference [47]. As was expected, on the basis of the theoretical analysis, no significant variation could be found in inlet velocity distribution with reduced flow rate.

The conservatism of the 5% reduction in hot assembly inlet flow for this plant will be verified by experimental data obtained in a 1/7 scale flow model test described in Section 1.5. Increased flow

TABLE 5.5-3

STEAM GENERATOR DESIGN DATA

Design pressure, reactor coolant side, psig	2485
Design pressure, steam side, psig	1285
Design temperature, reactor coolant side, °F	650
Design temperature, steam side, °F	600
Total heat transfer surface area, ft^2	67,000
Maximum moisture carryover, wt percent	0.25
Overall height, ft-in.	71-6
Number of U-tubes	4864
U-tube outer diameter, in.	0.750
Tube wall thickness, nominal, in.	0.043
Number of manways	4
ID of manways, in.	16
Design Fouling Factor, hr., sq. ft. °F/Btu	
a. Boiler	0.00005
b. Preheater	0.00010

TABLE 5.5-4

STEAM GENERATOR WATER (STEAM-SIDE) CHEMISTRY SPECIFICATION

Total Dissolved Solids, max.	125 ppm*
Total Suspended Solids, max	5 ppm
pH (normal operation) 25°C	9.0 to 10.5
Free Caustic	Zero
Dissolved Oxygen	Essential Zero (less than 0.005 ppm)
Chlorides, max.	75 ppm
Silica, max.	5 ppm

* Up to 600 ppm for 4 weeks.

TABLE 5.5-15

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Spray Control Valves

Number	2
Design pressure, psig	2485
Design temperature, °F	650
Design flow for valves full open, each, gpm	450

Pressurizer Safety Valves

Number	3
Maximum relieving capacity, ASME rated flow, lb/hr	420,000
Set pressure, psig	2485
Design Temperature, °F	650
Fluid	Saturated steam
Transient Condition, °F	(Superheated steam) 680
Backpressure:	
Normal, psig	3 to 5
Expected during discharge, psig	350

Pressurizer Power Relief Valves

Number	2*
Design pressure, psig	2485
Design temperature, °F	650
Relieving capacity at 2350 psig, lb/hr (per valve)	210,000
Fluid	Saturated steam
Transient Condition, °F	(Superheated steam) 680

* Three power relief valves for plant with large load decrease option.

TABLE 5.5-12

PRESSURIZER RELIEF TANK DESIGN DATA

Design Pressure, psig	100
Rupture Disc Release Pressure, psig	85 ± 5%
Design Temperature, °F	340
Total Rupture Disc Relief Capacity, lb/hr at 100 psig	1.6×10^6

TABLE 5.5-8

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION

Residual Heat Removal System start up	~4 hours after Reactor Shutdown
Reactor Coolant System initial pressure, psig	~400
Reactor Coolant System initial temperature, °F	~350
Component cooling water design temperature, °F	105
Cooldown time, hours after initiation of Residual Heat Removal System operation	8
Reactor Coolant System temperature at end of cooldown, °F	150
Decay heat generation at 12 hours after reactor shutdown, BTU/hr	103.9×10^6

TABLE 5.5-9

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump		
Number	3	
Design Pressure, psig	600	
Design Temperature, °F	400	
Design Flow, pgm	3400	
Design Head, ft.	360	
Residual Heat Exchanger		
Number	3	
Design Heat Removal Capacity	39.4×10^6 BTU/hr	
	Tube-Side	Shell-Side
Design Pressure, psig	600	150
Design Temperature, °F	400	200
Design Flow, lb/hr	1.5×10^6	2.45×10^6
Inlet Temperature, °F	150	105
Outlet Temperature, °F	123.7	121.1
Material	Austenitic Stainless Steel	Carbon Steel
Fluid	Reactor Coolant	Component Cooling Water

TABLE 5.5-9

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA

Residual Heat Removal Pump		
Number	3	
Design Pressure, psig	600	
Design Temperature, °F	400	
Design Flow, pgm	3400	
Design Head, ft.	360	
Residual Heat Exchanger		
Number	3	
Design Heat Removal Capacity	39.4×10^6 BTU/hr	
	Tube-Side	Shell-Side
Design Pressure, psig	600	150
Design Temperature, °F	400	200
Design Flow, lb/hr	1.5×10^6	2.45×10^6
Inlet Temperature, °F	150	105
Outlet Temperature, °F	123.7	121.1
Material	Austenitic Stainless Steel	Carbon Steel
Fluid	Reactor Coolant	Component Cooling Water

TABLE 6.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM COMPONENT PARAMETERS

Component	Parameters	
Accumulators	Number	3
	Design Pressure, psig	700
	Design Temperature, °F	300
	Operating Temperature, °F	100-150
	Normal Operating Pressure, psig	650
	Minimum Operating Pressure, psig	600
	Total Volume ft^3	2500 each
	Minimum Water Vol. ft^3	1500-1750
	Vol. N_2 gas, ft^3	750-1000
	Boric Acid Concentration, nominal, ppm	2000
	Boric Acid Concentration, minimum, ppm	1900
	Relief Valve Setpoint, psig	700
	Inleakage Alarm Sounds, ft^3	14
High Head Injection Pumps	Number	3
	Design Pressure, psig	1800

EMERGENCY CORE COOLING SYSTEM COMPONENT PARAMETERS

Component	Parameters	
	Design Temperature, °F	300
	*Design Flow Rate, gpm	800
	Design Head, ft.	2850
	Max. Flow Rate, gpm	1600
	Head At Max. Flow Rate, ft.	1000
	Discharge Heat at Shutoff, ft.	3700
Low Head Safety Injection Pumps	Number	3
	Design Pressure, psig	600
	Design Temperature, °F	300
	Design Flow Rate, gpm	1400
	Design Head, ft.	620
	Max. Flow Rate, gpm	2900
	Head At Max. Flow Rate, ft.	400
	Discharge Head at Shutoff, ft.	700

*Includes miniflow

CONTAINMENT DATA

Passive Heat Sinks

A Material Properties: See table

B Surfaces: See table

C Heat Transfer Coefficients: See figures

Engineered Safety Systems Information

	Full Capacity	Value Used for Containment Analysis
A Passive Safety Injection System		
1. Number of Accumulators	3	2
2. Pressure Setpoint, psig	700	700
B Active Safety Injection Systems		
1. High Pressure Safety Injection'		
a. Number of Lines	3	
b. Number of Pumps	3	
c. Flow Rate, gal/min (each)	800	
2. Low Pressure Safety Injection		
a. Number of Lines	3	
b. Number of Pumps	3	
c. Flow Rate, gal/min (each)	1400	
C Containment Spray System		
1. Recirculation Spray		
a. Number of Lines		
b. Number of Pumps		
c. Number of Headers		
d. Flow Rate, gpm (each)		

CONTAINMENT DATA

	Full Capacity	Value Used for Containment Analysis
Containment Cooling Fan Systems		
1. Number of Units		
2. Air Side Flow Rate, cfm		
3. Heat Removal at Design Temperature (10^6 Btu/hr)		
4. Overall Heat Transfer Rate (Btu/hr-ft^2 F)		
Heat Exchangers		
1. Recirculation Spray System		
a. Type		
b. Number		
c. Heat Transfer (UA), Btu/hr F		
d. Flow Rates		
1) Recirculation Side, gpm (each)		
2) Exterior Side, gpm (each)		
e. Source of Cooling Water		
f. Flow Begins, sec.		
Others		
1. Residual Heat Exchanger		
a. Type		
b. Number		
c. UA Btu/hr F		

CONTAINMENT DATA

	Full Capacity	Value Used for Containm Analysis
d. UA Btu/hr F		
e. Flow Rates		
1) Recirculation Side, gpm (each)		
2) Exterior Side, gpm (each)		
f. Source of Cooling Water		
g. Flow Begins		

Temperature Distributions and Effects

Assumptions Used in Pressure Transient Analysis

A Feedwater Valve Closure Time, Seconds: Not Applicable

B Steam Line Valve Closure Time: Not limiting turbine stop va close at 1.02 seconds

C Trip Time, seconds: 4.2 sec. (includes reactor trip time, de for trip, and time to trip rods. Howeve the core is shut down within one second to voiding reactivity effects.

D Liquid Carryover: The fraction of coolant vaporized, entrai and leaving the core is not constant duri transient, but increases from zero at the ning to 70 to 80 percent of the entering several seconds after initiation of reflo depending on the core flooding rate. Thi supported by FLECHT data. The Westinghou prietary entrainment correlation, present the ECCS rulemaking proceedings, has been to evaluate the amount of mass leaving th of the core as a function of time. The m is over conservative in that fall back in upper plenum is expected. Study of this is continuing. These assumptions are con vative because they result in a high ener release rate.

TABLE 6.2-5

CONTAINMENT SAFETY FEATURES

Spray Systems

Number of spray trains	3
Number of spray trains operating in analysis	2
Spray flow rate per spray train	800 gpm

Fan Coolers

Number of fan coolers	
Number of fan coolers operating in analysis	

Emergency Core Cooling System

Number of ECCS trains	3
Number of ECCS trains operating in analysis	2

Initiation Times

System	Time After Accident (sec)
Emergency core cooling	
Spray	
Fan coolers	
Recirculation	

TABLE 6.2-4

INITIAL CONDITIONS VALUES

Containment Design Parameters

Containment design pressure	60
Containment volume	2.55×10^6
Initial containment pressure	
Initial air partial pressure	
Initial steam partial pressure	
Initial steam partial pressure	
Initial sump water volume	
Initial containment temperature	120°F
Maximum calculated thermal rating	4117 MWt
Refueling water storage tank inventory	400,000
Nuclear service water temperature	

Initial Energies

Water energy	
Air energy	
Steam energy	
Total inital energy	

CONTAINMENT DATA

IV and V   Mass Addition Table and Blowdown Table

PUMP SUCTION DOUBLE-ENDED GUILLOTINE

Time (sec)	Mass Rate (lb/sec)x10[4]	Enthalpy (Btu/lb)x10[2]	Reactor Vessel Pressure (psig)
1×10^{-8}	7.49	5.76	2300
.05	7.49	5.76	1968
.2	8.08	5.77	1960
.3	9.09	5.79	1946
.4	8.46	5.82	1913
.5	8.03	5.87	1887
.6	7.72	5.92	1885
.8	7.48	5.96	1886
.9	7.42	6.00	1871
1.1	7.21	6.05	1855
1.4	6.84	6.11	1809
1.8	6.42	6.16	1746
2.0	5.93	6.23	1708
3.5	4.52	6.45	1360
5.0	3.62	6.62	1261
6.5	3.27	6.65	1211
8.0	2.99	6.62	1137
9.5	2.77	6.67	1105
11.0	2.4	6.91	999
13.0	2.05	7.01	872
15.0	1.64	7.39	804
16.5	1.29	7.74	639
18.0	1.02	7.64	405
19.5	0.81	6.95	301
21.0	0.64	5.89	142
22.5	0.52	4.85	109
24.0	0.26	4.42	57
24.5*	0.12	10.95	52
24.5	0.0	0.0	52
24.8	0.017	13.08	
25.4	0.083	13.08	
34.4	0.118	12.92	
50.0	0.106	12.81	
100.0	0.075	12.60	
159.3**	0.050	12.46	
159.4	0.019	12.48	
200.0	0.018	12.42	
500.0	0.013	12.41	
1000.0	0.009	12.38	
2000.0	0.007	12.30	
10000.0	0.005	12.24	

*End of Blowdown

**End of Entrainment

C. Ryder.

Cornell University

Sibley School of
Mechanical and Aerospace Engineering
Upson and Grumman Halls
Ithaca, New York 14853

June 15, 1984

Mr. Mel Silberberg
Accident Source Term Program
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Mel:

Some time ago Irv Spiewak, partly at my urging, requested information along the following lines. I repeat the request and would appreciate having the information sent to me directly just as soon as possible.

We are interested, in general, in determining how well the various calculations can keep track of the water in the system. In order to do so, we must know <u>where</u> the initial inventories are and <u>how</u> <u>large</u> they are. Accordingly, the following would be most helpful.

A simple block diagram (per reactor type) is needed that will show:

1. Volume of Water
2. Mass of Water
3. Pressure of Water
4. Normal Temperature of Water

in each of the reservoirs, including the reactor vessel, that are connected to the reactor vessel. The paths of connection should be indicated on the diagram (i.e. via pumps or gravity or whatever).

Thus, we are interested in knowing how much water may be available through one or more of the ECCS systems. We wish to know at what pressures they operate. We want to know how much water can be made available by sprays, supression pools, ice baths or whatever other engineered safety system may be in place.

In addition, it would be nice to know how water is distributed between the reactor vessel, the pressurizer (for PWRs) the coolant loops and the steam generators (for PWRs) under normal operating conditions. It might even be helpful to know how large the steam (vapor) bubble is in the pressurizer under normal operating conditions.

In other words, we would much appreciate having a concise, but complete, aqueous description of the plants (2 PWR and 2 BWR) that we have been looking at. Then, should anyone ever ask, "where's the water?", we will be able to answer without a moment's hesitation.

Sincerely,

Peter

P. L. Auer
Professor

cc: R. Wilson - Harvard
I. Spiewak - Oak Ridge

/gf

May 26, 1984

Dear Mel,

Enclosed you will find a couple of illustrations which are meant to serve as a guide for my requests. This is more or less in line with earlier requests, but perhaps I can be a bit more specific this time.

I wish to have in schematic form a diagram of all the relevant control volumes corresponding to the primary system, secondary (steam generator) system and containment volume for both PWRs and BWRs. I should like to have these in a consistent manner. I wish to know the volume of each, in consistent units, and I wish to know how much water is stored in each, appropriate control volume, as the case may be. I prefer not to have water content expressed in lbs, ft^3 and gals, for different units, but rather in the same consistent units. I should like to know the volume of water in cold/hot leg pipes, normally, as well as in all ESS units, auxiliaries, etc. I also wish to know what the normal path (design basis) of water flow is.

other plants

(1) SURV ?

(2) Peach Bottom ?

(2)

I trust that a few schematic diagrams with flow charts would serve my purpose.

Next, I wish to have the pressure vessel, core, etc. system drawn for both PWR and BWR with all appropriate dimensions included, as suggested by my enclosure. I wish to know relative volumes occupied by fuel vs cooling channels, dimensions of downcomer, lower plenum, core, upper plenum, structurals, and so forth. I am, as you will note, interested in having a precise, quantitative picture of those design parameters that are intimately related to the thermal-hydraulics analysis of the problem.

The sooner I can have these, of course, the better. With many thanks for your help,

Sincerely,

Peter

P.L. Auer

[redacted]

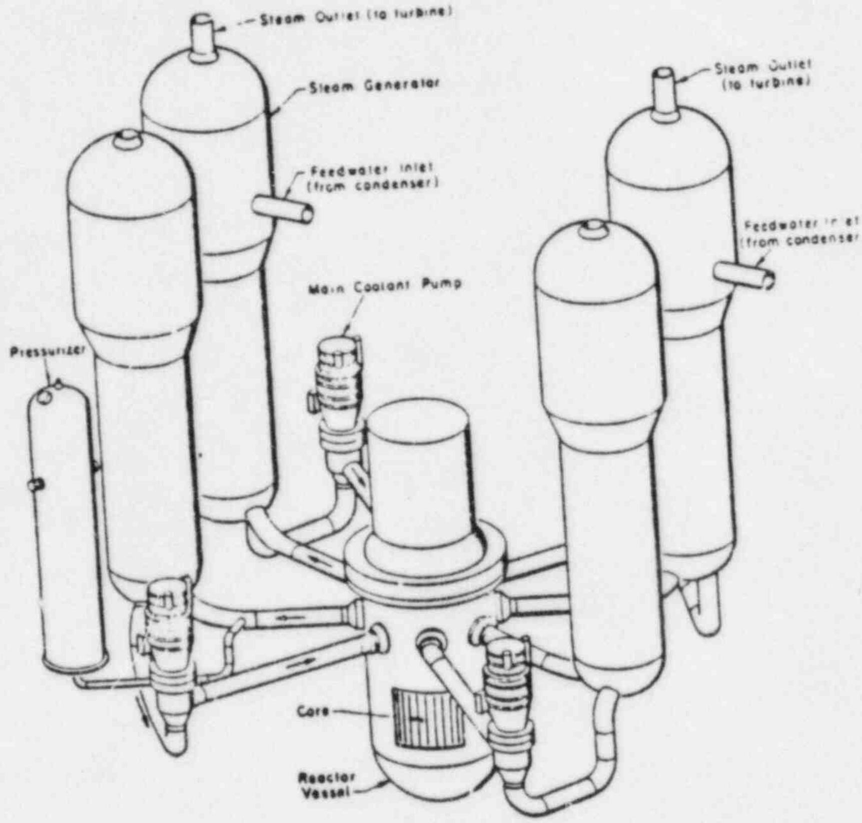
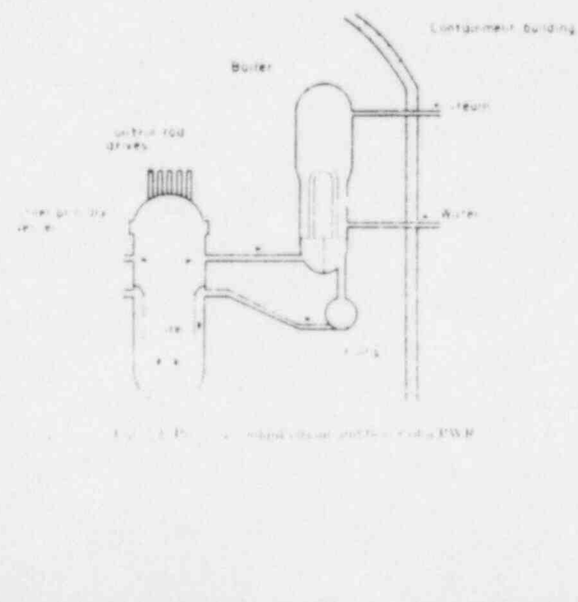


FIGURE 3-3 Schematic of Reactor Coolant System for PWR

Kindly indicate by schematics and flow charts the volumes of all units (RCS, ESS, etc) containing water, amount of water and path of water flow.

Kindly show control volumes and pathways for F.P. transfer, water/gas transfer from RCS to containment.

... steam generator enters a centrifugal pump and is returned to the core. The number of coolant loops is variable, but typically for a large reactor there would be four steam generators, each with its own primary circuit pump.



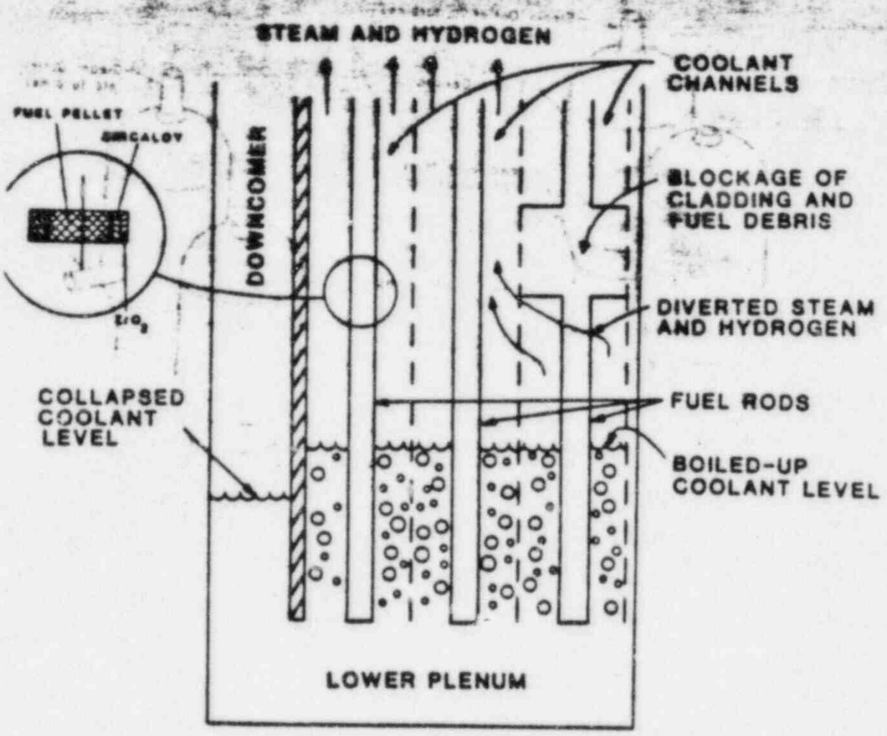


Fig. 1 Schematic description of a PWR core during uncovery.

Please show all appropriate dimensions for constituents/(see 1 and 2) in lower plenum, core and upper plenum.

the hot water from the core ... generator. A separate supply ... boils and produces steam ... ilar to those to be described ... to the turbine. After expan... ... the steam generator ... The number ... four steam

The baffle is supported by the core barrel. The thermal shield is a thick sheet of steel, designed to reduce the neutron and γ-ray dose received by the wall of the pressure vessel. The cooling water flows down on both sides of the thermal shield before entering the core.